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Invited Lectures
ABSTRACT

This paper provides a cursory look at the nuclear reactor safety regulation system and research programs in Japan relevant to nuclear reactor thermal hydraulics in general. After an introduction to Japanese nuclear power program, the nuclear regulation system is briefly described, and nuclear research infrastructure including research institutes and supporting organizations are outlined. In the matured stage of nuclear energy deployment in Japan, safety researches are focused onto those of ensuring the public safety, further improving and reinforcing the safety performances of the current and future nuclear installations, and enhancing nuclear safety regulatory activities. Among them, the prioritized nuclear safety research program is picked up and a brief look at several topics is made in connection more or less to thermal hydraulics. They include those of future risk-informed regulation, material degradation mechanisms principally concerning SCC and ageing management of nuclear power plants, severe accident issues, accident management, code development efforts in the areas of fuel behaviours, core thermal hydraulics, neutronics-thermal hydraulics coupling for high burn-up UO2 and MOX fuelled LWRs with corresponding experimentation, and a series of development programs of safety analysis methods for sodium cooled fast reactors of the next generation.

1 INTRODUCTION

This article is a brief review of the current status of Japanese nuclear reactor safety research and development programs overlooked from a viewpoint of nuclear reactor thermal hydraulics. Because Japan is a country that is not abundant natural resources and that takes an international initiative in reducing the greenhouse gas emission, her reliance on nuclear energy is a logical and inevitable consequence. As one of the most developed countries of nuclear energy deployment, the nuclear safety is of significant importance where both private and government sectors are making unceasing efforts in improving nuclear safety and regulatory systems for current and future generation nuclear reactors.

The safety regulation system is characterized by a “double check system” resulting in a little complicated structure of regulatory authorities. Major organizations that are responsible for the nuclear safety regulations are Nuclear Safety Commission1 (NSC), Nuclear and

1 http://www.nsc.go.jp/NSCenglish/index.htm
Industry Safety Agency (NISA) and its subsidized Japan Nuclear Energy Safety Organization (JNES). See footnotes for the website of these organizations.

Some of the safety researches are under the direct control of NSC being organized in a “prioritized safety research program” and carried out mostly in national laboratories. However a large portion of the safety researches are carried out in connection primarily to the development of APWR, ABWR and their extended versions in the private sectors (licensee including utility companies and reactor vendors) as well as national laboratories, industries and in academia under the sponsorship of the Ministry of Economics, Trade and Industry (METI) and Ministry of Education, Culture, Sports and Science and Technology (MEXT).

In this article, instead of delving into the details of all the safety programs, a cursory look at the major R&Ds on-going is provided and those which are considered to be relevant to nuclear reactor thermal hydraulics and safety issues are listed. Typically they are summarized into the thermal hydraulics safety R&Ds related to:

- Future risk-informed regulation,
- High burn-up UO$_2$ and MOX fuelled LWRs
- Ageing management of nuclear power plants: material degradation mechanisms principally concerning stress corrosion cracking (SCC) and erosion,
- Power up-rating and life extension of LWRs under operation,
- Development of next generation BWRs and PWRs
- Liquid metal-cooled fast reactor development, in particular sodium-cooled fast breeder development

Many efforts are concentrated on code development with corresponding experimentation including those in the areas of fuel behaviours, subchannel analysis, neutronics-thermal hydraulics coupling for high burn-up UO$_2$ and MOX fuelled LWRs, severe accident issues and accident management. Among them, the prioritized safety research program by NSC focuses on the material degradation and ageing management of LWR, and advanced reactor safety evaluation including for high burn-up and MOX LWRs and for sodium-cooled fast reactors as shown in 4.1.

2 NUCLEAR POWER IN JAPAN

As of July 2008, 23 PWRs and 32 BWRs including 4 ABWRs are capable of generating 49.6 GWe (See Fig. 1). In 2006, nuclear power contributed to 31% of total electricity generation in Japan. This year the share would be reduced slightly due to the absence of the seven BWR units in Kashiwazaki-Kariwa Nuclear Power Station of Tokyo Electric Power Company including two ABWRs that are under the thorough check out after the Niigataken-Chuetsu-Oki earthquake July 16, 2007. Currently (July 2008) 3 units are under construction and 3 applications are submitted to obtain construction permit. In addition, 7 more units are being planned to join the grid in 10 years to ensure the most balanced energy supply and security. It would result in 40% nuclear share in the electricity generation by 2017.

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2 http://www.nisa.meti.go.jp/english/index.htm
4 http://www.meti.go.jp/english/index.html
5 http://www.mext.go.jp/english/a02.htm
For general information of Japanese nuclear power industries, visit the website provided by Federation of Electric Power Companies of Japan\(^6\), and for the most updated operating records by Japan Atomic Industry Forum\(^7\).

In 2005, Atomic Energy Commission (AEC) Japan has established a basic policy for the government and industry to promote R&Ds and utilization of nuclear energy (The Framework for Nuclear Energy [1]). The framework aims at 40% of nuclear share from 2030, recycling LWR spent fuels in LWR and high level radio-active waste disposal and commercialization of fast breeder reactors (FBRs) before 2050. Along this policy statement, it is solicited to take actions to promote nuclear R&Ds and utilization activities on the near-term (within a few years of time range), mid-term (~2015) and long-term (~2025) basis. Also the statement emphasizes the importance of international activities as cross-cutting actions to be carried out [2].

Figure 2 displays a nuclear power generation capacity in Japan in the past and future [3]. After 2015, it is expected that the life extension of the nuclear power plants that will exceed 30 years of original plant design life and some of the existing power plants will be operating under the power up-rating condition. These life-extension and power up-rating issues are currently discussed and attracting attentions from both licenser and licensee sides and a number of R&Ds and assessments are being made in both government and private sectors in Japan.

\(^6\) http://www.japannuclear.com/
\(^7\) http://www.jaif.or.jp/english/aij/index2.html

Fig. 1 Nuclear Power Plants in Japan (as of 2006 to 2007)

Figure 2 displays a nuclear power generation capacity in Japan in the past and future [3]. After 2015, it is expected that the life extension of the nuclear power plants that will exceed 30 years of original plant design life and some of the existing power plants will be operating under the power up-rating condition. These life-extension and power up-rating issues are currently discussed and attracting attentions from both licenser and licensee sides and a number of R&Ds and assessments are being made in both government and private sectors in Japan.
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Fig. 2 Nuclear power generation in Japan in the past and future
(Installed capacity is assumed to saturate at 58 GWe just for illustrative purpose) [3]

3 SAFETY REGULATION AND RESEARCH INFRASTRUCTURE

3.1 Nuclear Safety Regulation System

Japanese nuclear safety regulation structure could be illustrated in Fig. 3. It is characterised by the so-called “double check” conducted by the regulatory bodies consisting of NISA and METI and by NSC. Upon receiving license applications, NISA deals with the nuclear power generation related facilities and MEXT the research reactor related facilities. NSC supervises these activities conducted by NISA and MEXT for the items inquired and provides reports to the regulatory bodies.

Fig. 3 Japanese nuclear safety regulation structure
Subsequent regulation after applications for construction permits and during operation phase, NISA and MEXT are in a position to provide approvals of design and safety program and pre-service inspection, and to conduct periodic inspection of facilities, check suitability of safety inspection and emergency preparedness. NSC also supervises the subsequent regulatory activities periodically and carries out audit of the NISA or MEXT regulatory body. Japan Nuclear Energy Safety Organization (JNES) supports NISA in connection to the inspection of the facilities, carries out cross-check analysis of the safety analysis results provided by the licensee as well as R&Ds to be reflected onto the safety regulations by NISA.

3.2 Research and Development Infrastructure

As mentioned before, “The framework for Nuclear Energy,” the statement issued by the Japan AEC in 2005, is the basis of the Japanese nuclear energy policy of the Government [1]. In pursuit of nuclear activities within the scope of this framework, the NSC plays a responsible role in ensuring safety. National institutes and laboratories are a major player in this respect. They include:

- Japan Nuclear Energy Safety Organization (JNES),
- Japan Atomic Energy Agency (JAEA: a coalescence of the Japan Nuclear Cycle Development Institute, formerly Nuclear Power Reactor and Nuclear Fuel Development Corporation, and Japan Atomic Energy Research Institute, since 2005),
- National Maritime Research Institute,
- National Institute of Advanced Industrial Science and Technology,
- National Research Institute for Earth Science and Disaster Prevention,

..., etc.

Current safety issues of the operating reactors are in connection to thermal hydraulics of those of life extension as well as power up-rating. These R&Ds are carried out being aimed at enhancement and reinforcement of the quality of present and next generation power plants safety. They are carried out under the coordinated efforts among government and industries. Namely, R&Ds in connection to these current issues and to the next generation nuclear reactors are promoted in the framework under strong leadership of METI and utility group. Participants to these programs include JAEA and JNES, reactor vendors (MHI, Toshiba-Westinghouse, Hitachi-GE Nuclear, etc), fuel manufacturers (Nuclear Fuel Industries, Ltd., GNF-Japan Co, Ltd., etc), utility companies, Central Research Institute of Electricity Industry (CRIEPI) and universities.

Recently Atomic Energy Society of Japan has organized “Special Committee on Research and Development Plan to Establish Technology Basis for Advanced Thermal-Hydraulic Safety Evaluation” being attended by most of these organizations, where key critical issues are being discussed from the viewpoints of LWR thermal-hydraulics and safety and a road map to the next generation LWRs is going to be established. This kind of activity of Special Committee is unique in the sense that academic society takes a lead in re-directing various and diversified R&Ds otherwise, in salvaging missing important topics, if any, and avoiding duplicated efforts.

Future generation nuclear power reactors include sodium-cooled fast reactor, gas-cooled fast reactor (GCFR), Pb/Bi cooled FR, etc. Among them the sodium-cooled fast reactor has been selected in Japan as a first candidate that is expected to replace gradually the current operating LWRs after 2050. JAEA takes an initiative in this national project, which is participated by the national laboratories, utilities, reactor vendors, and universities. It is noted that a new company, Mitsubishi FBR Systems, Inc. (MFBR) has been established since July 2007 to accommodate the basic design activities of this new FBR development.
4 NUCLEAR SAFETY RESEARCH

4.1 Prioritised Safety Research Program Defined by NSC

The NSC’s overall program covers a wide range of the nuclear safety issues in Japan, including:

- Reinforcement of the current safety regulatory system with:
  - Future introduction of risk-informed regulation, and
  - Use of root cause analysis of troubles and accidents;
- Nuclear facilities such as LWR, advanced nuclear reactor and fuel cycle facilities, where safety evaluation methods, material degradation and ageing management, and seismic evaluation are major R&D subjects;
- Disposal of radioactive waste and decommissioning of nuclear facilities; and
- Radiation effect and nuclear disaster prevention.

Here the safety programs related to nuclear facilities such as LWR and advanced nuclear reactors are of our interest. The background is that the licenses for MOX fuel have been granted for 4 PWRs (Takahama-3&4, Genkai-3 and Ikata-3) and 2 BWRs (Fukushima I -3, Kashiwazaki-Kariwa-3). Also J-Power’s ABWR under construction at Ohma plans a flexible in-core fuel management scheme allowing the operation with the full MOX core.

Selected LWR safety topics include experimentation and model and codes development for fuel behaviour and neutronics and thermal-hydraulics coupling of the advanced high burn-up UO$_2$ and MOX cores, and those associated with power uprating, and severe accidents and accident management issues [4]-[13]. Fuel failure boundaries under RIA and LOCA conditions are examined by in-pile experiments using NSRR and other test facilities. For example, high burnup UO$_2$ and MOX fuels irradiated at LWRs in Japan and Europe including Vandellos-II are loaded in the RIA-simulating experiments at NSRR, a pulse reactor, of JAEA Tokai. Accordingly the database is extended to the higher burnup range and used for safety evaluation methodology. These R&Ds are carried out mostly by JAEA and JNES to provide technical information for the safety examination for licensing by NISA and NSC.

Selected R&Ds for material degradation and technology of ageing management are given higher priority. In particular understanding ageing mechanisms and exploring technology of preventing troubles caused by ageing, development of the early detection and precise measurement of crack and degradation are carried out not only from the water chemistry and structural reliability but also thermal hydraulics point of view [14]-[19]. R&Ds related to the material degradation and ageing management include IASCC investigation using the JMTR (Japan Material Testing Reactor), PFM research at JAEA and development of database and technology for detection and sizing of the defects at JNES.

Finally it is emphasized here that in Japan, national project of Fast Reactor Cycle Technology (FaCT) Development is going on at JAEA being aimed at establishment of conceptual designs of a new sodium-cooled fast reactor and its fuel cycle system by 2015. The project has been joined by MFBR (Mitsubishi FBR, Inc) as a prime contractor from industry. The new design should not only demonstrate economic competitiveness with next generation LWRs but accommodate enhanced safety, reliability and better utilization of fuels with increased proliferation resistance. International coordination frame has been set up among USA, France and Japan trying to reduce the duplication of efforts at the world level. Japan will contribute to this coordination, utilizing her experimental fast reactor Joyo, prototype Monju and fuel cycle development facilities in Tokai Lab of JAEA.
Therefore, fast reactor safety R&Ds are another important issue in the prioritized safety program. Major topics in this area include development of safety evaluation methods for troubles and accidents in connection to the use of sodium as a coolant, for preventing core damages and assuring the accident mitigation and containment including those initiated by ATWS (Anticipated Transient Without Scram) whose occurrence frequencies are considered to be negligibly small. It is expected to demonstrate that the consequence of the accidents dictated by the chemical potential of sodium can be safely accommodated and mitigated by the design and, as a result, the use of sodium as a coolant will never be a direct threat to the fast reactor safety. References are given at [20]-[21].

4.2 Outlook of Some Important Safety Research Topics in Industries

Since the fast reactor development in Japan is carried out as a national project with the leadership of JAEA and as a consequence, there is small room for a private company to have R&Ds initiative at its own. Therefore, it would be worth mentioning Toshiba’s efforts in development of the 4S (Super-Safe, Small and Simple with 10MWe electricity output) fast reactor first in this section. The efforts have been supported by CRIEPI. At present, Toshiba-Westinghouse is applying to USNRC for the pre-licensing of the 4S fast reactor [22], to be followed by Design Approval application in 2009.

A prospective of the key R&Ds related to current and next generation LWR thermal hydraulics and safety in Japan is summarized in Table 1. A group of R&D topics relevant to the current generation LWRs consists of those corresponding to power uprating, adoption of higher burnup fuels and ageing and plant life extension. Topics in connection to the next generation LWRs are classified into subgroups of extended concepts of APWR and ABWR, and small Grid-Appropriate Reactors. R&Ds that are common to the current and next generation NPPs are connected to further development and improvement in the safety evaluation methods, and issues of post accident long-term cooling system, severe accidents and accident management, fire protection and evaluation.

Table 1: Typical R&D topics related to the current and next generation LWRs in Japan [23]

<table>
<thead>
<tr>
<th>NPPs</th>
<th>R&amp;D Group</th>
<th>R&amp;D Topics (B for BWR; P for PWR; P+B in common)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Current Generation LWRs</td>
<td>Power uprating</td>
<td>B: Flow instability, Neutronics-coupling and power oscillation, Post-BT</td>
</tr>
<tr>
<td></td>
<td></td>
<td>P+B: Flow rate measurement</td>
</tr>
<tr>
<td></td>
<td>Advanced fuels</td>
<td>P+B: High burnup UO2, MOX, fuel failure thresholds</td>
</tr>
<tr>
<td></td>
<td></td>
<td>B: RIA transient void, Post-BT</td>
</tr>
<tr>
<td></td>
<td>Aging and life extension</td>
<td>P+B: Material degradation coupled with TH</td>
</tr>
<tr>
<td></td>
<td></td>
<td>P: PTS</td>
</tr>
<tr>
<td>Current &amp; Next Generation LWRs</td>
<td>Safety evaluation method</td>
<td>P+B: BE code Plus Uncertainty method, CSAU, Statistical Methods, more enhanced use of CFD</td>
</tr>
<tr>
<td></td>
<td>Post accident heat removals</td>
<td>P+B: ECCS strainer blockage</td>
</tr>
<tr>
<td></td>
<td>Severe accidents</td>
<td>B: FP transport and release, H2 over pressure</td>
</tr>
<tr>
<td></td>
<td>Fire</td>
<td>P+B: Prevention and consequences with fire PSA</td>
</tr>
<tr>
<td>Next Generation LWRs</td>
<td>APWR and ABWR extension</td>
<td>P: Advanced safety system (SG depressurization)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>B: Severe accident mitigation system</td>
</tr>
<tr>
<td></td>
<td></td>
<td>B: Spectrum shift reactor concept, large bundles</td>
</tr>
<tr>
<td></td>
<td></td>
<td>B: Large natural circulation w/o recirculation pumps</td>
</tr>
<tr>
<td></td>
<td>Grid-Appropriate</td>
<td>P: Integrated Primary System PWRs with boiling two phase flow natural circulation</td>
</tr>
</tbody>
</table>
Some safety research activities are directly connected to troubles/events/accidents that actually occurred in the nuclear power plants. For example, the ECCS pipe rupture accident that took place in November 2001 at Hamaoka BWR Unit-1 urged utility companies, in particular BWR owners, to establish guidelines for preventing the accumulation of hydrogen (and oxygen) in a localized portion of any piping system connected to the primary coolant system. Thermal and Nuclear Power Engineering Society in Tokyo has organized a subcommittee with supervise of METI and first version of the guideline has been published 2005 and revised in 2007 [23]. The guideline has been based on simulated experiments with a help of CFD calculations for the non-condensable gas accumulation inside various part of piping in BWR plants. It is going to be updated with additional information from hydrogen detonation experiments and analysis.

Another example here is the feed-water pipe rupture accident in Mihama-3 PWR in 2004 during the maintenance period. Also similar pipe thinning phenomena were observed in Onagawa BWR Unit-1 (at the elbow of the drain line from MSIV) and Onagawa Unit-2 (in a high-pressure secondary feedwater heater bent pipe: see Fig. 4) in 2006 during annual maintenance period. These events have necessitated understanding the phenomena and the mechanisms of pipe wall thinning. The phenomena have been considered to be a subject of combined research areas of water chemistry and thermal hydraulics, primarily due to corrosion and erosion mechanisms [24].

![Fig. 4 Small hole as a result of pipe thinning, Onahama Unit-2](http://www.tohoku-epco.co.jp/whats/news/2006/05/23b12.pdf)

The troubles of major components and structural materials in LWR power plants have often been caused by flow-induced vibration, corrosion, and their overlapping effects. In order to establish safe and reliable plant operation, it is required to forecast future troubles based on combined analyses of flow dynamics and corrosion and prevent them at very early stages. Flow-accelerated corrosion (FAC) is considered to be one of the major causes of unexpected troubles occasionally occurred in the orifice flow of secondary feed water pipe in BWR. FAC rate increases rapidly as the mass transfer rate increase. In order to analyze this...
FAC phenomenon accurately, a reliable numerical simulation for flow analysis and mass transfer evaluation in orifice flow is carried out [25].

Corrosion analysis models have been combined with three-dimensional flow dynamics and heat transfer analysis models to evaluate corrosion damage, e.g., SCC and FAC of major components and structural materials [26]. The models are divided into the following two parts. First is a prediction model of future trouble on materials. The distribution of oxidant concentrations along the flow path are obtained by solving water radiolysis reactions in the boiling water reactor primary cooling water and hydrazine and oxygen reactions in pressurized water reactor secondary cooling water. Then, the distribution of electrochemical corrosion potential (ECP) along the flow path is obtained by oxidant concentration based on a mixed potential model. Higher ECP enhances the possibility of SCC, while lower ECP accelerates FAC. Second is an evaluation model of wall thinning caused by FAC. At the location with a higher possibility for FAC occurrence, a trend of wall thinning is evaluated, and the lifetime is estimated for preventive maintenance.

5 CONCLUSIONS

A brief review of the nuclear reactor safety regulation system and research programs in Japan has been presented. In the matured stage of nuclear energy deployment, safety researches are focused onto those of ensuring the public safety, further improving and reinforcing the safety performances of the current and future nuclear installations, and enhancing nuclear safety regulatory activities. Among them, the prioritized nuclear safety research program is carried out by Nuclear Safety Commission. Safety R&Ds in Japan include those which will be required by plant life extension, power up-rating, next generation nuclear power plant development, and future risk-informed regulation. In particular, current major issues focus on the material degradation mechanisms in connection to ageing management of nuclear power plants, severe accidents and accident management, experiment and code development efforts in the areas of fuel behaviours, neutronics-thermal hydraulics coupling for high burn-up UO2 and MOX fuelled LWRs, and a series of development programs of safety analysis methods for sodium cooled fast reactors.

ACKNOWLEDGMENTS


REFERENCES


[23] H. Nakamura (JAEA), Handout No. 4-2 at the 4th meeting of the Special Committee on Research and Development Plan to Establish Technology Basis for Advanced Thermal-Hydraulic Safety Evaluation, AESJ, August 5, 2008.


Safety Assessment and Analysis
EVALUATION OF THE IMPACT THAT PARS HAVE ON THE HYDROGEN RISK IN THE REACTOR CONTAINMENT: METHODOLOGY AND APPLICATION TO PSA LEVEL 2

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ABSTRACT

This paper presents a methodology, and its application to a Level 2 Probabilistic Safety Assessment (PSA-2), to evaluate the impact of the Passive Autocatalytic Recombiners (PARs) on the hydrogen risk in the reactor containment in case of a severe accident. Among the whole set of accidental scenarios calculated in the framework of the PSA-2, nine have been selected as representative in terms of the in-vessel hydrogen production rate and in-vessel total produced hydrogen mass. Five complementary scenarios have been added as representative of the core reflooding situations. For this set of selected scenarios the evolution of the conditions in the containment (i.e. pressure, temperature, composition) during the in-vessel phase of the accident has been evaluated by means of a lumped parameter approach. The use of spray systems in the containment has also been considered as well as the presence of recombiners. The impact of recombiners has been accounted for both because use of recombiners is a mean to reduce hydrogen concentration in the containment and atmosphere may be ignited by recombiners due to the high temperatures. Evaluation of the recombiners ignition risk has been done based on experimental results derived from the H2PAR and KALI-H2 experiments. On the basis of these analyses, the regions of the containment (and durations) where the atmosphere ignition was possible were identified for each scenario, and the corresponding pressure obtained in case of an Adiabatic, Isochoric and Complete Combustion (PAICC) was evaluated. The risk of flame acceleration has been then investigated for these regions as well as the risk of transition to detonation (DDT). Respectively $\sigma$ and $L/\lambda$ criteria have been used to assess if the necessary conditions exist for these risks.

1 INTRODUCTION

In the theoretical case of a severe accident in a nuclear reactor with core meltdown, the interaction of the hot core with the cooling water can generate large amounts of hydrogen. Hydrogen may be produced by oxidation of metals present in the corium pool or in the basemat during the molten corium-concrete interaction phase. This hydrogen is transferred into the containment (and transported therein) by convection loops arising mainly from
condensation of steam released via the RCS break or during corium-concrete interaction. Depending on mixing in the containment atmosphere, the distribution of hydrogen is more or less homogeneous. If considerable hydrogen stratification exists, local concentrations of hydrogen may become substantial, exceeding the lower flammability limit for the gas mixture. The distribution and concentration of hydrogen in the containment building may also be modified by its spray systems. Spraying does homogenize the distribution of hydrogen in the containment and lead to “de-inertization” of the mixture through the condensation of steam on water droplets.

To limit the hydrogen concentration in the containment, several methods can be proposed [1]:
- the deliberate ignition of the mixture as soon as the flammability limit is reached,
- the consumption of hydrogen,
- the removal of oxygen,
- the dilution of the atmosphere to prevent the formation of flammable mixtures either by the increase in the volume of the containment, or by the injection of an inert gas.

Hydrogen risk management can be implemented by one or a combination of the previous methods. The choice of a mitigation strategy depends primarily on the design of the containment. For PWRs with large dry containment, the strategy usually consists in combining large free volume to allow dilution, a high value of the design pressure and the use of means, as passive autocatalytic recombiners (PARs), to consume hydrogen. This strategy has been adopted recently in all French PWRs.

In this paper, we propose a methodology to evaluate the impact of PARs on the hydrogen risk in French 900 MW.e reactor containment in case of a severe accident and its application to a Level 2 Probabilistic Safety Assessment (PSA-2).

2 HYDROGEN RISK ASSESSMENT METHODOLOGY

The methodology adopted to assess hydrogen risk in the reactor building must take into account the different loads accounting for the impact of hydrogen production, distribution and mitigation systems. This method uses the following main steps:

**Step 1. Plant design:** the starting point of any analysis is the selection of the plant and geometrical modeling of the containment. This step aims to well describe the containment shape and volume; which influence hydrogen distribution inside the reactor containment.

**Step 2. Selection of relevant scenarios** representative of severe accident sequences and the evaluation of the associated hydrogen production rates and release into the reactor building. These source terms are usually derived from parametric code calculations with best estimates for still uncertain hydrogen production processes.

**Step 3. Evaluation of the containment atmosphere conditions** during the accident transient (i.e. temperature, pressure and gas composition in the different regions and volumes of the containments) accounting for the presence of mitigation systems.

**Step 4. Evaluation of the time evolution of flammable hydrogen-air-steam cloud:** The flammability of the containment gas mixture depends on its temperature, pressure and composition. However, in practice, the point representing the mixture's composition (hydrogen, air, steam) on the Shapiro diagram [2] (see figure 1.) is used to determine
whether the mixture is flammable. In this diagram, the zones of ignition and detonation are respectively delimited by the exterior and interior curves.

![Shapiro diagram for hydrogen-air-steam mixtures](image)

**Figure 1: Shapiro diagram for hydrogen-air-steam mixtures**

**Step 5. Evaluation of the propensity of a premixed flame to propagate inside the containment:** under the effect of hydrodynamic instabilities and turbulence (caused primarily by obstacles in the flame's path), an initially laminar deflagration (with a flame velocity around 1 m/s) may accelerate. Fast combustion regimes may also develop, involving rapid deflagration (a few hundred m/s), deflagration-to-detonation transition (DDT) and detonation (over 1000 m/s). These combustion regimes may generate high pressure loads which could endanger the containment integrity.

To define the transition from slow to fast combustion regime, two types of criteria are considered:

- The “σ” criterion related to flame acceleration. σ stands for the mixture's expansion factor, a ratio of fresh and burnt gas densities at constant pressure. It is an intrinsic property of the mixture.

![Critical value σ* as a function of hydrogen concentration](image)

**Figure 2: Critical value σ* as a function of hydrogen concentration**

The critical value σ*, beyond which flame acceleration is possible, depends on initial gas composition and temperature and flame stability.

- Similarly, prerequisite conditions have been defined for characterizing the transition between deflagration and detonation regimes (DDT). They are based on comparing a characteristic dimension of the geometry with detonation cell size λ.
Flame acceleration and TDD criteria are based on the results of numerous experiments at various scales and in various geometries [3] and are considered as prerequisite criteria, i.e. conditions required for the various combustion modes.

**Step 6. Evaluation of pressure and thermal loads generated by combustion:** two configurations are distinguished:
- If flame acceleration criteria are not met: in this case, dynamic pressure loads are excluded and the pressure load is evaluated by considering adiabatic complete isochoric combustion process.
- If flame acceleration criteria are met: in this case, the induced combustion loads is evaluated using the most appropriate combustion models.

### 3 IMPACT OF PARS ON HYDROGEN RISK

In the following, the previous methodology is applied to investigate the PARS impact on hydrogen risk in French 900 MWe reactor containment by considering severe accidents sequences simulated in frame of PSA level 2. Effect of spray actuation and core reflooding is also considered.

All calculations are performed using ASTEC V1.1 Lumped parameter code jointly developed by IRSN and GRS, i.e. Gesellschaft für Anlagen- und Reaktorsicherheit mbH and dedicated to the simulation of the whole course of severe accidents in light-water reactors.

#### 3.1 Step 1: plant design

The reactor containment is modeled by using multi-compartment approach. This model is based on 65 elementary volumes by considering 6 vertical levels and various radials sections [4]. Moreover, the structure of the casemates surrounding each of the 3 primary loops is described. In addition to elementary volumes, walls and junctions (gas and liquid transport) are considered.
3.2 Step 2: Scenarios definition

In frame of PSA Level 2, 35 core degradation sequences are considered. They correspond to the following situations:

- accidents involving a complete loss of feedwater to steam generators (H2 or TGTA class),
- accidents involving steam generator tube ruptures SGTR,
- accidents involving loss of coolant LOCA,
- accidents involving a complete loss of electrical power (H3 class).

The following figures present hydrogen mass and hydrogen production rate for each scenario:

![Figure 4: hydrogen mass and hydrogen production rate for each core degradation sequences](image)

To take into account core reflooding situations, 3 safety injection rates corresponding to 5.5 kg/s, 41.4 kg and 205.6 kg, are considered. Spray actuation is considered as soon as gas pressure exceeds 2.4 bars.

To reduce the number of scenarios, a classification according to hydrogen production rate and hydrogen mass is carried out. The adopted approach distinguishes three levels of hydrogen mass:

- **Weak** corresponding to a mass of hydrogen produced lower than 300 kg,
- **Medium** corresponds to a produced hydrogen mass between 300 kg and 550 kg,
- **High** corresponds to a mass of hydrogen produced higher than 550 kg.

and three levels of hydrogen production rate:

- **Weak** for H2 production lower than 0.1 kg/s,
- **Medium** for H2 production between 0.1 kg/s and 0.15 kg/s,
- **High** : for H2 production higher than 0.15 kg/s.

The combination of the previous criteria leads to 9 types of scenarios:
Table 1: families of scenarios

<table>
<thead>
<tr>
<th>Families of scenarios</th>
<th>H2 mass</th>
<th>H2 production rate</th>
</tr>
</thead>
<tbody>
<tr>
<td>Type 1</td>
<td>Weak</td>
<td>Weak</td>
</tr>
<tr>
<td>Type 2</td>
<td>Weak</td>
<td>medium</td>
</tr>
<tr>
<td>Type 3</td>
<td>Weak</td>
<td>High</td>
</tr>
<tr>
<td>Type 4</td>
<td>medium</td>
<td>Weak</td>
</tr>
<tr>
<td>Type 5</td>
<td>medium</td>
<td>medium</td>
</tr>
<tr>
<td>Type 6</td>
<td>medium</td>
<td>High</td>
</tr>
<tr>
<td>Type 7</td>
<td>High</td>
<td>Weak</td>
</tr>
<tr>
<td>Type 8</td>
<td>High</td>
<td>medium</td>
</tr>
<tr>
<td>Type 9</td>
<td>High</td>
<td>High</td>
</tr>
</tbody>
</table>

The application of the previous criteria to the various scenarios allowed defining 9 families of accidental scenarios whose members present the same “behavior” regarding to hydrogen risk. Thus the PARs effect investigation will be limited to these 9 sequences.

3.3 Step 3 & 4: time evolution of flammable clouds inside the containment

The flammable gas time evolution is checked by considering gas mixture in each containment zone at each time in Shapiro diagram (see figure 5). This shows that the use of PARs decreases the hydrogen concentration in the containment atmosphere and limit the flammable cloud size inside the reactor.

Figure 5: Effect of PARs on gas mixture flammability

More generally, the analysis of time evolution of reactor containment atmosphere regarding to PARs effect allows defining three classes of scenarios:

- Class 1: constituted of scenarios corresponding to weak productions of hydrogen and to reflooding scenarios with high safety injection rate (205.6 kg/s). For this category, the use of PARs seems to be sufficient to avoid flammable gas formation.
- Class 2: constituted of scenarios corresponding to medium and high productions rate. In these cases, flammable gas mixture is present in certain containment zone during a limited time.
- Class 3: corresponds to reflooding scenarios with safety injection rate of 41.4 kg/s and 5.5 kg/s. In these cases, containment gas atmosphere becomes flammable during reflooding phase.
3.4 Step 5: flame acceleration and TDD

For each scenario, flame acceleration criterion is checked in each compartment of the containment at moments corresponding to high productions rate of hydrogen.

![Figure 6: PARs effect on Flame acceleration](image)

The analysis of PARs effect shows that for:

- Class 1: the use of PARs avoids flame acceleration.
- Class 2: flame acceleration criterion is satisfied in certain containment zone during a limited time.
- Class 3: flame acceleration criterion is satisfied by the reactor containment gas atmosphere during reflooding phase.

3.5 Step 6: pressure loads evaluation

Before performing combustion calculation, ignition sources have to be defined. The ignition must be either predicted mechanistically (self ignition) or must be postulated with respect to time and location.

In the framework of PSA level 2 and due to their hot catalytic sheets, PARs are considered, under specific condition (see figure7), as ignition sources. Indeed, some of the experimental tests performed on KALI H2 and H2PAR [5] show that PARs could ignite the flammable gas mixture. These experimental results show that ignition induced by recombiners occurs for low hydrogen concentrations respecting to the following limits:

![Figure 7: ignition of SIEMENS PARs](image)

These ignition limits correspond to low hydrogen concentration which avoid flame acceleration phenomena. The evaluation of pressure load generated by combustion is then carried out by considering complete, adiabatic and isochoric combustion. Pressure, known as
PAICC, thus calculated depends on gas mixture and on the thermodynamic conditions and does not depend on the geometry of the containment. The calculation results show that PAICC does not exceed in all cases the containment pressure.

4 EVALUATION OF SPRAYING EFFECT

Spray system has been activated for the previous selected scenarios. The effect of this activation has been analyzed and shows that, for all scenarios, steam condensation on spray droplets leads to the creation of consequently flammable cloud. Moreover, flame acceleration criteria are satisfied in the whole of the reactor containment.

5 CONCLUSION

This paper presents a methodology for hydrogen risk assessment and its application to evaluate the effect of Passive Autocatalytic Recombiners in framework of Level 2 PSA.

This analysis has shown that the use of recombines reduces significantly the risk of flame acceleration and transition to detonation. However, the presence of recombines does not eliminate the risk of flame acceleration which lasts for some specific scenarios and short duration.

These analyses have also shown the beneficial effect of recombines as igniters. Indeed, based on the experimental data presently available, it seems that ignition induced by recombines occurs for low hydrogen concentrations, leading to relatively low pressure. These experimental results need however to be corroborated by more detailed experiments and by refined modelling of phenomena occurring in PARs.

6 REFERENCES

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Comparative Analysis of Graphite Stack Behaviour in TKR Facility and Kursk-5 Reactor in the Case of PT Rupture with the Use of U_STACK Code

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A B S T R A C T

It was originally intended to use the results of experimental studies, carried out at ENIC (Elektrogorsk) as part of the programme to deal with the problem of multiple pressure tube ruptures, together with the results of TACIS Project R2.02/99 to prove the safety of RBMK power units belonging to the first and second as well as to the third generation, such as Kursk-5. However, direct application of the results of full-scale experiments with a fragment of the graphite stack built as found in reactors of the first and second generations (TKR facility) proved problematic due to substantial distinctions of the graphite moderator in the Kursk-5 reactor. The problem was solved by numerical simulation of the TKR facility and the Kursk-5 graphite stack with the use of the integral U_STACK Code. The code allows simulating the interrelated mechanical and thermal-hydraulic processes in the graphite stack in the event of channel tube ruptures. It was shown that, under comparable conditions, pressure tubes of fuel channels in the Kursk-5 reactor are found in a better stressed-strained state as compared to those of the TKR facility and NPP reactors of the previous generations.

1 INTRODUCTION

The experimental part of an extensive programme for investigating the problem of multiple pressure tube ruptures (MPTR) in the RBMK graphite stack was completed three years ago [1]. The TKR facility, which comprised a fragment of a graphite stack identical to those of the first- and second-generation RBMKs, was designed for studies on dynamic characteristics of an accident with rupture of a pressure tube (PT) and for demonstrating the impossibility of one PT rupture propagating to other fuel channels in the regular RBMK stack, i.e. in a stack made of square graphite blocks.

Improvements in the neutronic characteristics of the RBMK core were attained at Kursk-5 by optimisation of the uranium-carbon ratio. To this end, about 20% of graphite was removed from the core by cutting away the edges of square blocks and forming four openings of $\phi 16$. The blocks inside the core are octahedral prisms, while those of the top/bottom and side reflectors retain their original shape with a square cross-section [2].

As a result of this modification, the lateral stiffness of the stack columns decreased, the volume of voids in the stack became considerably larger, the cross section of vertical flow increased 200 times within the core height, and the hydraulic resistance across the stack was reduced by a factor of three. The inertial characteristics of the columns changed with reduction of their mass. Under those circumstances, no prospects were seen for extending in
any way the experimental results obtained at the TKR facility [3] to the Kursk-5 reactor stack; i.e., it was by no means obvious whether the stressed-strained state of Kusk-5 pressure tubes may be better or worse under conditions of the accident modelled at the TKR facility. The situation called for development of a computer model which would faithfully reproduce the highly complicated interrelated thermal-hydraulic and structural-mechanical processes following a PT rupture in the reactor stack module (RSM) of the TKR facility. With this done, the same software could be used for simulating the Kursk-5 reactor stack and performing the associated computational studies. The numerical models of the TKR facility and Kursk-5 reactor stack were constructed with the use of the integral U_STACK Code, which was successfully verified both for individual processes and against the results of integral experiments.

2 U_STACK CODE

The U_STACK Code was developed for detailed 3D numerical simulation of two-phase flow movement and heat exchange processes in their interrelation with the motion and deformation of core components in water-graphite reactors during an accident with a pressure tube rupture. The problem of two-phase flow under conditions of heat exchange between the coolant, graphite and metal structures is solved for a system of gaps with variable dimensions, which are evaluated by simultaneous calculation of the movement of graphite blocks subjected to pressure forces when interacting with one another, with pressure tubes and surrounding metal structures. The latter problem, in turn, necessitates calculation of the deformation of the pressure tubes and metal structures under variable loading conditions [4].

The following processes related in time and space can be simulated by U_STACK Code:

− movement and heat exchange of non-equilibrium multicomponent two-phase flow in the circulation circuit, in the graphite stack and outside its limits before and after the pressure tube rupture;
− deformation and rupture of an affected pressure tube with the ensuing destruction of graphite blocks at the condition of overheating;
− displacement and deformation of the graphite stack components under the action of forces resulting from the tube rupture;
− operation of the reactor cavity venting and accident localisation systems.

Calculation of forces and displacements in the graphite stack involves assessment of integrity of the adjacent tubes exposed to internal pressure as well as to additional impacts arising in the course of the accident.

Methodologically, the code combines the mechanical and thermal-hydraulic modules. Thermal-hydraulic and mechanical processes in the stack are studied in their inexorable relation to the processes occurring in the circulation circuit and in the pipelines of the reactor cavity venting, accident localisation and drainage systems. The mechanical analysis module simulates the high-temperature deformation and rupture of the channel tube, the displacement and turning of graphite blocks under the action of rupture-induced forces, the impacts on pressure tubes and their bending. The mechanical analysis based on:

− experimental data and relations describing the behaviour of pressure tubes and graphite blocks;
− momentum equations of solid body with six degrees of freedom to describe the motion of graphite blocks;
− spring beam equation to describe tube deformation by bending.

The module of thermal-hydraulic analysis is based on the best-estimate code RELAP5/mod3.2, in which additions were made to allow for the specifics of processes under conditions of variable geometry of flow paths. The thermal-hydraulic module simulates the processes of coolant movement and heat exchange along the flow path of the affected
channel to the point of rupture and onwards – through the stack to the accident localisation system. This module relies on two approaches using:

- RELAP5/mod3.2, which is based on detailed nodalisation of the relatively small part of the stack adjacent to the rupture, where the geometry and the fluid parameters experience the greatest changes, and which involves a larger parts of the reactor cavity to simulate distant areas;
- additional homogeneous model of two-phase flow, which relies on special interpolation procedures to determine pressure distribution in the large parts of the reactor cavity remote from the ruptured channel.

Besides, the computation model allows for failure of graphite blocks in the affected column and for the stack deformation preceding the pressure tube rupture, for the finite rate of the PT rupture and for movement of graphite fragments in the gaps between graphite columns.

The accident process is analysed according to a scenario defining the initial and boundary conditions of the problem. Both overheating of fuel channels and spontaneous rupture caused by a flaw in the pressure tube under normal operating conditions may be considered as the rupture initiating events [5, 6].

### 3 TKR FACILITY AND KURSK-5 REACTOR STACK

The main purpose of the TKR facility [1] was to reproduce the conditions of an accident with PT rupture under coolant pressure, with rapid increase of the wall temperature. The reactor stack module (RSM) of the TKR facility comprises 45 columns with pressure tubes (Fig.1) and represents a part of the RBMK reactor stack. The central tube contains an internal thermit heater at its mid height. In the initial conditions the tubes, except for the central one, are filled with water whose parameters are equal to the nominal parameters of RBMKs. The central tube is connected via the lower water and upper steam lines to the water preparation unit (WPU). Initially, the central tube is filled with steam of nominal parameters specific to RBMKs.

A system for experimental measurements of parameters in their dynamics provided recording of:

- flow rate, pressure and temperature of the coolant arriving at the damaged channel by the lower (water) and upper (steam) lines,
- axial deformations of the outer tube surface (9 tubes, ~100 sensors). This group of measurements (Fig. 1) provided the most representative and significant data,
- surface temperature of graphite blocks (60 sensors),
- pressure in the stack gaps,
- displacements of graphite blocks in peripheral columns at mid height.

![Fig. 1. Location of deformation sensors at mid height of the RSM, on the right – sensors of block displacement in the outer columns](image)

Graphite block displacements in the peripheral columns were measured by sensors of movement in the direction of the principal axes. Fig. 1 shows the displacement sensors on
one of the RSM sides. The peripheral columns of RSM were in contact with special devices that imitated the elastic reaction of the absent columns of a real stack. Thus, the measurements of parameters made in the process of deformation of the PT-containing graphite columns in the event of the central tube rupture could be regarded as adequate to the conditions of PT rupture in the reactor stack of an RBMK plant belonging to the 1st or 2nd generation (hereinafter referred to as the regular RBMK stack).

The graphite stack of the Kursk-5 reactor, except for the reflectors, is made of octahedral blocks, Fig. 2.

The indicated distinctions between the mechanical and hydraulic characteristics of this stack and a regular one prevent direct application of data from the full-scale experiments at the TKR to the conditions of accidents in the Kursk-5 reactor stack. It proved necessary to assess the effect of competing factors. On the one hand, the lower lateral stiffness and mass of the columns contribute to bending of pressure tubes and, on the other hand, the changes in hydrodynamics caused by sharp decrease of hydraulic resistance of the flow paths in the stack will promote lowering of the maximum pressure in the stack. In other words, the forces arising from pressure difference in gaps between the columns in the Kursk-5 reactor stack will be smaller than in a regular stack with the same leak rates.

![Diagram of graphite blocks in Kursk-5 reactor and in RBMKs of the 1st and 2nd generations](image)

**Fig. 2.** Horizontal sections of graphite blocks in Kursk-5 reactor and in RBMKs of the 1st and 2nd generations

### 4 ANALYSIS OF ACCIDENT PROCESSES

To assess the influence of the reactor stack configuration at Kursk-5 on its safety parameters, a comparative computational analysis of experiments at the TKR facility was carried out, with the conditions of the First and Second full-scale experiments [3] applied to the Kursk-5 reactor stack.

Previous analyses covered three accidents that had actually occurred in operation of RBMK reactors as well as a very broad spectrum of other credible accidents. The results obtained strongly suggest that the maximum horizontal displacement of a graphite block is one of the most symptomatic parameters, which indicates the extent of adverse impacts on the adjacent channel tubes. Axial deformation of the outer tube surface may be taken as a parameter directly pointing to the possibility of the tube wall damage. Comparison of the maximum deformation with a certain limiting value will show the margin to the onset of plastic deformation of the tube.

The data presented in Figs. 3 and 4 suggest a highly complicated nature of the phenomena observed in the experiments at the TKR facility, which accounts for the difficulties in computational simulating.
Measurements of parameters of the steam coming from WPU proved to be out of synchronism and therefore showed insufficient agreement (Fig. 3, left side). Since the impact on the tubes of adjacent channels is greatest at the first second of the accident with tube failure, the dynamics of coolant parameters in the inlet pipeline should be thoroughly

Fig. 3. Second full-scale experiment at the TKR facility. Parameters of the steam coming by the steam line

Fig. 4. Displacements of graphite blocks in the peripheral row
analysed for comparison of the computation results with the experimental data. The results of such comparative analysis made with the use of the U_STACK Code are demonstrated in Fig. 3 (right side).

The consistent and complex patterns of real graphite block motion were revealed by direct measurements, which were only made for several graphite blocks in the peripheral rows of the stack (Figs. 1 and 4). Early in the process, the motions of those blocks were practically coincident with those calculated in terms of both their linear parameters and phase. The periods of motion stabilisation and the displacement extent are also comparable. At the same time, dissimilarities are found in the transition region. Here, the experimental variations in block displacements are perceptibly greater than those in the calculations. This discrepancy is explained by the natural dissimilarity between a real object and an idealised model. Movement of blocks results in a new system of gaps, which change from 1 – 2 mm to several tens of millimetres. The stabilised pattern of block positions will be stochastically dependent on the initial distribution of the gaps and, even more so, on the location, shape and azimuthal orientation of the break in the tube. It is practically impossible to predict either the parameters of this hole or the process of its opening due to the significant influence of random variations in the governing factors. This influence makes the process parameters less determinable locally, but, with the same initiating event, it will have minor implications for the safety-related parameters. This statement will be illustrated here by the data on deformation of the tubes in the TKR facility, obtained during the Second full-scale experiment (Fig. 5).

The initiating impact responsible for the motion and deformation of the reactor stack components is caused by pressure increase in the stack gaps during accidental escape of high energy coolant. The difference in the escaping flow dynamics under the conditions of Kursk-5 and Second experiment at the TKR facility (Fig. 6) in the first 0.5 s of the process is,
most likely, explained by obstruction of the flow path for a short time by fragments of the failing thermit heater. Later on, this obstacle is removed when the fragments are picked up by a steam-gas-water flow rushing into the stack, and carried away.

For Kursk-5, PT rupture was assumed to occur in the peripheral area so that the conditions analysed could be comparable to those of the TKR facility.

Fig. 6. Leak rates

The behaviour of the maximum fluid pressure in the gaps, determined for the cruciform control elements, proved to match the flow rate dynamics (Figs. 7 and 8).

Fig. 7. Maximum pressure in the gaps of the TKR stack

The damaged column is marked by red colour in these figures. The lower hydraulic resistance of the Kursk-5 reactor stack leads to lower maximum and stabilised pressures as compared to those in the TKR stack.
Fig. 8. Maximum pressure in the gaps of the Kursk-5 reactor stack

Figure 9 shows the columns around the broken pressure tube (damaged column) with the “row–channel” identification. Zero coordinates along the principal axes X and Y are assigned to the damaged column.

Fig.9. Columns surrounding the broken channel

In the initial phase of the process under consideration, horizontal motions of graphite blocks in the TRK facility, found closest to the damaged column, proved to be oscillatory, with a period of ~0.8 s and an amplitude commensurable with the value of maximum displacement (Fig. 10). In the Kursk-5 stack, intensive motion of graphite blocks was only revealed within a short initial time interval of 0.1 s (Fig. 11). The motion practically ceased in the next 0.5 s. The maximum displacement of blocks in a stabilised position turned out to be half as large at Kursk-5 as it was in the TKR facility.
The most significant difference between the dynamic characteristics of TKR and Kursk-5 lies in the extent of axial deformation of pressure tubes. The distribution of such deformations was determined with running of U_STACK Code at each time step of the simulated process for each of the reactor channel tubes without exception. The greatest deformations were typically found near the end supports. The examples of estimated deformation dynamics given in Figs. 12 and 13 testify to moderate deformation of the Kursk-5 tubes as compared with the TKR tubes. The graphs show components of deformation only in the direction of one of the two horizontal axes. Clearly, the total tube deformation in this case will exceed the presented values.
Fig. 12. Tube deformations in the X direction at the level of z=-0.7 m (z=0 m correspond to the top support). TKR

Fig. 13. Tube deformations in the X direction at the level of z=-0.7 m (z=0 m corresponds to the top support). Kursk-5

Figures 14 and 15, showing the dynamics of maximum horizontal displacement of graphite blocks (i.e., pressure tube bends) and axial tube deformations, can give an idea of the correlation between the extreme values of safety-related parameters for the columns and pressure tubes surrounding the broken one. The additional scales in these diagrams indicate the sequential numbers of the graphite blocks (starting from the column top), which showed a maximum of the given parameter at each instant of time. The column coordinates are not identified. The upper and lower scales refer to the TKR and Kursk-5, respectively.
Comparisons were also made between the parameters of the process in the regular stack of an RBMK (conventionally, Leningrad-3) and in the Kursk-5 reactor stack under the same boundary conditions of the accident. The maximum displacement of the blocks and axial deformation of the tubes surrounding the broken one, are shown in Figs. 16 and 17.
In all the cases, the bends and axial deformations of the pressure tubes closest to the break were found to be smaller in the Kursk-5 reactor stack than both in the TKR facility and in the regular RBMK stack.

Thus, decrease of hydraulic resistance in the Kursk-5 reactor stack prevailed among all the competing factors mentioned above. Low hydraulic resistance of the stack reduced the maximum pressures in the gaps to such an extent that pressure tubes in the more flexible
and light-weight columns of the modified stack became less susceptible to bending, experiencing lower bending stresses as compared to pressure tubes of the other RBMKs.

5 RESULTS OF COMPARISONS AND SAFETY ANALYSIS RELATED TO RBMK GRAPHITE STACK

The processes taking place in the RBMK reactor stack during an accident with PT rupture are essentially unsteady and of high intensity. These traits manifest themselves to variable degrees throughout the reactor stack, and the safety-related parameters may only be adequately assessed in full-scale modelling of the whole stack. The events resulting from PT rupture are governed by numerous random factors, wherefore computational simulation of the possible variety appears to be the only promising way to make safety analysis sufficiently complete.

As regards the problems resulting from significant changes in the design, as is the case with the Kursk-5 reactor, the highest credibility of safety assessments may be achieved by comparative analysis of the new facility with the well-known prototypes, by analysis using a common methodological and instrumental approach. The prototypes of Kursk-5 are the sufficiently representative RBMK reactors of the first and second generations with the total operating record of about 350 reactor-years.

In the presented sample of the comparative analysis results, the prototype is the TKR facility, which is a full-scale model of graphite stacks found in reactors of the first and second generations. The results of this analysis demonstrated the advantages of the new configuration from the viewpoint of reactor safety during an accident with PT rupture. The analysis relied on routine use of the U_STACK Code, notwithstanding certain dissimilarities between the experimental facility and the reactor, which shows the level of the code universality. It is worth noting the significant similarities in the parameter dynamics observed in the experiments and calculations:

- Fig.3, variations of all three parameters,
- Fig.4, amplitude and modes of oscillatory motions of graphite blocks,
- Fig.5, similarity of the initial deformation pattern, of the scales and frequencies of deformation variations throughout the process.

The traditional requirements for strict agreement between the calculation results and the experimental data for the processes of the type under consideration come into conflict with the random nature of these processes. It may be hardly expected that a computer model will be highly accurate in describing the results of measurements for the processes/phenomena that are exposed to numerous random factors. Most important is computational reproduction of the characteristic attributes of the process and the time scale of the events. This point is confirmed beyond doubt by the practice of verification and application of the integral U_STACK Code.

6 REFERENCES


A Generic Assessment of the RIA Interim Criteria in a Typical 3 Loop 12 Feet PWR with 3D Methods

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ABSTRACT
As a continuation of previous experiences and to provide a provisional answer to the new Standard Review Plan 4.2 rev.3 interim criteria for core coolability and cladding failure during Reactivity Initiated Accidents, a research using codes PARCS and RELAP is performed. The chosen reference core is a typical 12 feet, 17x17 PWR, with very low leakage loading pattern strategy with gadolinium oxide as burnable poison at EOC conditions. The PARCS calculated average nuclear power and nodal power are transferred to a hot spot model for a sequential calculation of fuel temperature and enthalpy responses allowing for independent hypothesis in both calculations. The hot spot analysis is done with a pellet type model with RELAP. The conservative direction of each of the significant parameters was determined by the authors in previous investigations by means of the sensitivity studies and examination of the constitutive equations of the space-time diffusion and heat transfer equations in the fuel regions and conservation of mass, energy and momentum in the flow field as well as the protection system. These parameters include ejected rod worth and ejection time, delayed neutron fraction and yields and nuclear power peaking factor, Doppler, and MTC among other. Bounding representative values for each one of the high and medium ranked parameters were considered including calculational uncertainties. As a result, parameters like fuel prompt and delayed enthalpy and temperature as well as clad temperature were determined and compared to the new Interim criteria in terms of the cladding corrosion layer. This study concludes on the fulfillment of the coolability criteria on a generic basis and also shows that there is significant margin built in the current conservative 1D methods.

1 INTRODUCTION
In order to consider the irradiation effects in the ceramic UO$_2$ pellets and the oxidation and hydriding effects in the clad, regulators and the industry are currently revisiting the criteria to verify coolability and rod integrity during a RIA event. As a first step, the USNRC has issued interim criteria and guidance within the latest revision to NUREG-0800, “Standard Review Plan (SRP), Section 4.2, rev.3, Appendix B, in March 2007, [1]. Although the new criteria apply only to the initial licensing of the new reactor fleet and to advanced reactors and COL applications, it is an anticipation of similar criteria that might eventually apply to the current reactor fleet in the USA in the near term. Whether or not these or similar criteria will be incorporated to the local regulation of other countries is still unknown.
This paper is a generic study of the reactivity insertion accident (RIA) for a reference core, which is representative of the Spanish PWR fleet of Westinghouse design, with 3D modelling (see Table 1). Although these plants are not identical, the fuel design and fuel management strategies are very similar and a common frame of key parameters can be set whose validity could be verified each reload for all units. The methodology, i.e., codes and methods, were described in previous comparative and sensitivity works [2,3] which determined the important phenomena, the set of key parameters as well as the conservative direction. All these parameters, including uncertainties, are considered in a conservative way to define a worst case. Therefore, the method is deterministic and bounding of all hypothetical RIA.

Since the most restrictive condition in the light of SRP 4.2 rev.3 is at higher burnup states, the analysis in this paper is performed at HZP and HFP, both at EOL. The result is compared to the new criteria, and albeit not yet applicable to the current local regulation, is taken as a good comparison exercise.

Table 1: Reference Core

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Reference Value</th>
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<td>Plant</td>
<td>PWR, 3 loop, 12 feet</td>
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<td>Fuel Type</td>
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<tr>
<td>Clad Material</td>
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<td>Rated Thermal Power (MW)</td>
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<tr>
<td>Vessel Flow Rate, min/ max (m$^3$/s)</td>
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</tr>
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<td>Core Flow Bypass (%)</td>
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<td>Fuel Management</td>
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<td>Low leakage Loading Pattern</td>
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<td>Gadolinia as burnable poison</td>
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<td>Rod Burnup state (MWd/TU)</td>
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2 CORE NEUTRONIC -THERMAL-HYDRAULIC AND HOT SPOT SOLUTIONS

2.1 PARCS Model Description

For this demonstration exercise a core corresponding to the reference core in Table 1 was modelled. By using a detailed 3D advanced nodal code, the macroscopic and microscopic cross sections corresponding to EOC were calculated at homogeneous HZP reference conditions and the first-order partial derivatives were obtained from a series of perturbation calculations and based on this design reference model, a PARCS core model was built. The nodalization employed for this core model has been the following: each fuel assembly has been represented in 2x2 radial nodes and twenty-four axial nodes. Additionally, two other axial nodes were set up to represent the upper and lower axial reflectors. The radial reflectors have been modelled as additional non-fuel assemblies with the same 2x2 radial node geometry. To specify the nodewise XS assignment, the fuel has been grouped in five radial regions: “inner” fresh fuel, “outer” fresh fuel, once burned fuel, twice burned fuel, and finally, a special region with a unique assembly, the assembly where the ejected rod is positioned.

Previous calculations and sensitivity analysis performed in the past with this model [2,3], showed enough agreement to the more detailed 3D nodal reference core model,
specially at the EOC limiting condition, not only from the axial and radial power distributions viewpoint but also from the core reactivity parameters (ejected rod worth, rod banks worth, Doppler and power defects, MTC, etc). Adjustments to kinetics and reactivity parameters need to be made through the cross section module of PARCS to adapt to the conservative situation.

2.2 RELAP Pellet Model

The calculation of fuel temperature and enthalpy deposition at the hot spot is performed by means of a pellet-type RELAP model. The core average nuclear power and heat flux peaking factor are determined in a previous calculation with PARCS and then transferred to the RELAP pellet model.

Radial heat conduction is considered but no axial conduction is assumed. Conservative low gap conductance values were determined to maximize the adiabatic behaviour of the hot rod during the transient. Suitable properties have been considered for the UO$_2$ and the burnup effects have been considered. The initial conditions for the HFP case have been adjusted to more detailed fuel rod codes. Heat transfer from clad to coolant is forced convection, local boiling or film boiling depending on the phase of the transient. No DNBR calculation was implemented at this stage.

3 CONSERVATIVE KEY PARAMETERS

Table 2 summarizes the most important parameters considered in the simulation. The current analysis of Rod ejection in the FSAR of each plant has been done with sets of values that do not substantially differ from the chosen reference values in this demonstration exercise, although some values may be different. A discussion about the key parameters and the value assumed in the simulation is given next.

3.1 Reactivity worth

This parameter is the driving force for the transient and it is critical in all known models. There are several factors that have an impact in the control rod worth, mainly: control rod position and fuel burnup distribution within the core (loading pattern), axial power distribution accounting for the Xenon distribution and fuel burnup, load follow policy, and control rod insertion limits.

At HFP, very little insertion is allowed down to the Technical Specifications limit. However, although a typical maximum rod worth is only 23 pcm, a bounding value of 80 pcm is considered in the simulation, which is still very small compared to the effective delayed neutron fraction and one may expect to have proportional (rather than prompt) criticality behaviour. Initial power jumps almost instantaneously to a maximum power before the Doppler feedback turns the power down to a quasi-steady level and finally the scram takes place. The enthalpy deposition in the hot spot is very small.

However, at HZP, more control rods are allowed to a deeper insertion. The maximum rod worth increases with the average core burnup due to the decrease of boron absorption and a typical value is 630 pcm. However, a conservative bounding value of 840 pcm is considered in the simulation. In this situation, delayed neutrons are not important in the neutron kinetics equation and the core response is prompt critical, independent from initial power.

The adjustment in the PARCS model is attained by means of modifying the cross section module (HZP) or by artificially inserting the rods beyond the TS limit (HFP).
Table 2: Key Parameters of the Simulation

<table>
<thead>
<tr>
<th>Parameter</th>
<th>HZP</th>
<th>HFP</th>
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<tbody>
<tr>
<td>RO (pcm)</td>
<td>840</td>
<td>80</td>
</tr>
<tr>
<td>Delayed Neutron Fraction, βeff</td>
<td>0.0044</td>
<td>0.0044</td>
</tr>
<tr>
<td>Doppler Defect from HZP to HFP (pcm)</td>
<td>-1430</td>
<td></td>
</tr>
<tr>
<td>Moderator Temperature Coefficient, MTC (pcm/°C)</td>
<td>-26</td>
<td></td>
</tr>
<tr>
<td>Heat Flux Peaking Factor, Fq</td>
<td>25</td>
<td>4</td>
</tr>
<tr>
<td>Location of the Ejected Rod</td>
<td>H-14 (see Fig. 1)</td>
<td></td>
</tr>
<tr>
<td>Ejection Time (s)</td>
<td>0.10</td>
<td></td>
</tr>
</tbody>
</table>

3.2 Heat flux peak factor, Fq

This parameter is calculated as the ratio between the core maximum local rod linear power density and the core average linear power density determined at the peak power deposited in the fuel during the transient. It has to be noted that PARCS code does not have a pin power reconstruction algorithm, so the reported Fq in PARCS calculations is merely a nodal peaking factor that has to be multiplied by a precalculated intranodal peaking factor. These intranodal peaking factors have been calculated with the 3D advanced nodal reference code previously mentioned.

3.3 Ejection time

Since the rod ejection takes place in a very small period of time, the core response corresponds to a step of reactivity instead of a ramp, and thus, it is nearly independent of the
ejection time. This period of time can be estimated with the kinematics equation of a rod driven by a pressure difference from the core and outside, with no restrictions nor friction. It results in a slightly greater time than 0.10 s, so this value was assumed in the analysis.

3.4 Delayed neutron fraction, $\beta_{\text{eff}}$

Total delayed neutron fraction, $\beta_{\text{eff}}$, depends on the core isotopic content, specially that relative to U235, with minor contributions from fission of U238 and Pu239. The lower U235 content, the lower delayed neutron fraction, so the $\beta_{\text{eff}}$ value at EOC is the smallest.

Delayed neutrons play a very important role in the determination of the core kinetics response, and for this reason, they are treated independently in the core kinetics equations. Prompt criticality transients (RO>$\beta_{\text{eff}}$), like HZP rod ejection accidents, are mainly driven by prompt neutrons. The resulting power excursion is independent on delayed neutrons, but their relative abundance marks the threshold where, depending on the reactivity insertion, a prompt criticality transient becomes a less severe proportional one. The delayed neutron fraction can be specified directly in PARCS.

3.5 Doppler effect

Doppler effect, produced by the broadening of U238 and Pu239 capture cross sections, at resonance energy levels, caused by the fuel temperature increase at power is a key parameter in PWR reactors control, contributing with a reactivity negative feedback. In prompt criticality transients such as HZP rod ejection, the power excursion ends at the point where the reactivity increment beyond the effective delayed neutron fraction is compensated by the correspondent negative Doppler feedback, a few milliseconds after the rod ejection, reaching the maximum power peak.

In these transients, Doppler effect is very sensitive to the power redistribution towards the top of the core and the ejected rod position. For this reason, the simulation of this kind of transients with detailed 3D analysis tools, instead of the simplest traditional 1D methods, is more realistic and a consistent source of margin.

The Doppler effect is evaluated through the use of the “Doppler Defect”, the difference in the core reactivity between two states (HZP and HFP) exclusively due to the corresponding fuel temperature change. The Doppler effect is adjusted by means of modifying the XSEC module of PARCS.

3.6 Scram characteristics

If the reactivity insertion is high enough, the nuclear power will reach the high flux trip setpoint in the very first stages of the transient. After a 0.5 s delay due to signal electronics treatment, breaker opening and release time, all the rods except the one that was ejected, start falling into the core. Conservatively, two other adjacent rods were assumed to remain stuck. The PARCS scram model considers a linear dependence of control rod insertion with time and therefore, it does not consider the acceleration at the beginning nor the deceleration after the dashpot. An additional delay time (nearly 0.5 s) has to be considered to conservatively correct for the non-linearity of the scram kinematics.

Since the scram takes place far beyond the prompt phase of the transient, the sensitivity analysis results in a very small impact for all scram characteristics except for the scram it self.
3.7 Material properties

Density, heat capacity and conductivity are the main properties modelled for RIA analyses. UO$_2$ conductivity has been taken from the FSAR and the remaining properties have been taken from MATPRO [4], which is a compilation of the public databases for a variety of thermo-mechanical properties of UO$_2$ and zirconium alloys, and it is a fundamental reference widely accepted in the nuclear industry. Burnup effects have been considered at the hot spot.

3.8 Moderator temperature coefficient

The presence of the negative moderator temperature coefficient at power conditions in PWR reactors constitutes an additional term to the reactivity negative feedback caused by the Doppler effect. However, this parameter does not play a very significant role in RIA transients, although not negligible, due to their very short time scales. The adjustment is made by means of modifying the XSEC module of PARCS.

3.9 Direct moderator heating

A fraction of the energy generated during RIA is deposited outside of the fuel rod and does not contribute to the incremental of enthalpy in the pellet. Different conservative values can be used for both phases of the transient. However a conservative value has been considered for the nuclear calculation and for the last step of thermal-hydraulic hot spot calculation. A direct moderator heating of 2.6% has been considered.

Sensitivity analyses have been performed for this parameter. Results show a medium rank parameter. Low direct moderator heating leads to more conservative results.

3.10 Heat transfer (gap and clad)

Only forced convection heat transfer from clad to coolant is considered by PARCS. Sensitivity analysis and previous results showed that this simplification is conservative. For the thermal-hydraulic hot spot calculation with RELAP, a set of boundary conditions have been determined to simulate a quick transition to film boiling.

3.11 Other Assumptions

Focusing in the enthalpy deposition criterion, most of initial conditions do not play an important role in RIA behaviour. Variations of pressure around the nominal value have no impact in the kinetic transient and, due to the conservative assumptions, almost no impact in the hot spot calculation either. The same applies for the moderator and initial fuel temperature.

In the nuclear calculation, a maximum flow rate is conservative, since it results in lower moderator heatup and smaller negative moderator feedback. By the contrary, smaller flow rate in the hot spot calculation favours the heat crisis and film boiling heat transfer regime. Since the kinetics and enthalpy calculations are uncoupled, three operative primary pumps are assumed for the first and only two pumps for the later.
4 RESULTS OF THE SIMULATION

4.1 Hot Zero Power

Immediately after the rod ejection, the reactor becomes prompt critical. As a response, the nuclear power increases critical until feedback effects appear up to the maximum value, and then decreases down to the quasi static delayed neutron phase, and finally the scram actuates. $F_q$ increases also almost instantaneously due to the local unbalance of reactivity and reaches quickly the prescribed maximum value. It remains at that level until the fuel temperature rises and drives the Doppler feedback. Most of the energy at the hot pellet is produced at this initial (or prompt) phase. Finally, $F_q$ returns to greater values due to the distortion of the reactivity caused by the rod insertion during the scram, but its contribution to the produced energy is much smaller in this later phase. Figures 2 to 4 show the history of the main parameters.

This conservative case exhibits a maximum nuclear power of 2258% nominal in a 16 ms pulse width. The most relevant results of the calculation are summarized in Table 3.

4.2 Hot full Power

Simulation at full power requires the relocation of the initial position of the control rods, resetting the high neutron flux trip setpoint, the initial fuel temperature coherent to the initial power level and RCS flow rate consistent with the Technical Specifications.

The reactivity worth is lower than the delayed neutron fraction ($R_{\beta eff}$) and the transient does not have a prompt-critical behaviour anymore as it is at HZP. Although the absolute enthalpy is greater than the maximum at HZP, this is because the transient departs from a higher initial enthalpy, but the increase or energy deposition becomes much smaller than in the prompt-critical cases.

Since at part power cases, the core wide radial power distribution does not suffer such a great distortion as at HZP along with less important feedback effects, a 3D simulation does not result in a significant margin gain compared to current 1D methods. The most relevant results of the calculation are summarized in Table 3. Figures 5 to 7 show the history of the main parameters.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>HZP</th>
<th>HFP</th>
</tr>
</thead>
<tbody>
<tr>
<td>Maximum Power (%Nom)</td>
<td>2258</td>
<td>122</td>
</tr>
<tr>
<td>Time of peak (s)</td>
<td>0.168</td>
<td>0.11</td>
</tr>
<tr>
<td>FWHM (s)</td>
<td>0.016</td>
<td>No pulse</td>
</tr>
<tr>
<td>Max. Enthalpy, prompt (cal/g)</td>
<td>90.1</td>
<td>No pulse</td>
</tr>
<tr>
<td>Max. Enthalpy, total (cal/g)</td>
<td>94.2</td>
<td>141</td>
</tr>
<tr>
<td>Enthalpy Deposition, prompt (cal/g)</td>
<td>69.2</td>
<td>20.8</td>
</tr>
<tr>
<td>Peak Fuel Temperature (°C)</td>
<td>1338</td>
<td>2473</td>
</tr>
<tr>
<td>Peak Clad Temperature (°C)</td>
<td>870</td>
<td>649</td>
</tr>
</tbody>
</table>

Table 3 : Results of the Conservative RIA Simulation
Figure 2: Nuclear power at EOC-HZP.

Figure 3: Flux peaking factor at EOC-HZP.
Figure 4: Fuel and clad temperatures at EOC-HZP.

Figure 5: Nuclear power at EOC-HFP.
Figure 6: Flux peaking factor at EOC-HFP.

Figure 7: Fuel and clad temperatures at EOC-HFP.
5 COMPARISON WITH CRITERIA

Appendix B of SRP 4.2 rev.3, provides interim criteria and guidance for the control rod ejection for the USA domestic plants. These criteria addresses the 10CFR50 Appendix A General Criterion #28, which requires core coolability. Besides, although rod failure is allowed, it must be considered as an input to the radiological calculation. A discussion on these criteria is given next.

5.1 Core Coolability

5.1.1 Peak radial average fuel enthalpy must remain below 230 cal/g

This criterion was already considered in RG 1.77 and in previous versions of SRP 4.2 with a former limit of 280 cal/g based in unirradiated fuel and it has been reduced to 230 cal/g to account for effects. The simulation above results in 94.1 and 141 cal/g (HZP and HFP respectively), and the criteria is met with significant margin.

5.1.2 Peak fuel temperature must remain below incipient fuel meeting conditions

This criterion did not exist in previous regulations and a maximum of 10% melting of the innermost pellet was allowed. The new limit challenges the HFP cases for high Fq values. However, the simulation above results in a maximum centerline temperature of 2473 ºC, which is lower than the melting point of UO$_2$ at EOL, 2593 ºC.

5.1.3 Mechanical energy due to (1) non-molten fuel to coolant interaction and (2) fuel rod burst must be addressed with respect to reactor pressure boundary, reactor internals, and fuel assembly structural integrity.

The new regulation does not give any guidance about how to perform this sort of calculation. However, as it will be shown below, the fuel rod integrity during the transient is demonstrated and therefore, there is no fuel to coolant interaction in this case.

5.1.4 No loss of coolable geometry due to (1) fuel pellet and cladding fragmentation and dispersal and (2) fuel rod ballooning

Rather than demonstrating that there is no flow blockage due to ballooning nor fuel dispersal, the shortcoming to this verification is to demonstrate that there is no cladding failure at high burnup stages where rim structure build up and potential for rod dispersion exists. This is verified in Section 5.2 onf this paper.

5.1.5 Obsolete criterion.Maximum clad temperature below 2700 ºF (1482.2 ºC)

This criterion was used in the 60´s and 70´s to conservatively ensure coolability in a Rod Ejection event, but it as never been required by the US regulation in RG 1.77, neither in the original SRP 4.2 nor subsequent revisions. Although still written in some old FSAR, the limit no longer applies and it is superseded by the other criteria above and is being removed from the FSAR in many cases. Furthermore, the recent and ongoing discussions between the regulatory bodies and the industry regarding the effects of burnup on fuel and clad, do not
deal with the clad temperature and do not revisit a PCT limit and therefore, it will remain obsolete in the future. Nevertheless, for the sake of completeness, we can see that the conservative calculation in this paper results in 870 °C which is well below the obsolete limit.

5.2 Clad Integrity

5.2.1 The high cladding temperature criteria for zero power conditions is a peak radial average fuel enthalpy greater than 170 cal/g for fuel rods with an internal rod pressure at or below system pressure and 150 cal/g for fuel rods with an internal pressure exceeding system pressure. For intermediate (greater than 5% rated thermal power) and full power conditions, fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g. DNBR and CPR).

The previous regulation specified only a DNBR limit but there was no mention to the enthalpy limit. In the simulated case at HZP, the maximum total enthalpy (prompt + delayed) is 94.2 cal/g so this criterion is verified. As explained above, no DNBR calculation has been implemented in this PARCS-RELAP exercise, and therefore the full power verification of clad integrity for high temperature can not be conveniently done and is left to a further improvement of this calculation, but it is not expected to be an issue.

5.2.2 The PCMI failure criterion is a change in radial average enthalpy greater than the corrosion-dependent limit depicted in Figure B-1 (PWR) of SRP 4.2 Appendix B.

This new criterion addresses the pellet to clad mechanical interaction (PCMI) and it did not exist in the previous regulation. The corrosion layer of the highly irradiated fuel has been thoroughly measured in the past and 85 micro-meter thickness can be considered as a bounding representative value for the advanced Zirconium alloys, which are included in the original database for the USNRC corrosion-dependent limit. This figure is equivalent to an oxide to wall thickness ratio of 0.15, which results in a maximum limit of 66.2 cal/g for fuel failure. The simulation above provides an enthalpy rise of 69.2 cal/g and is above the limit. However, it can be demonstrated with detailed 3D nuclear calculations, that the actual \( F_q \) value for burned fuel is reduced more than 15% respect to fresh or once burned fuel. If credit is taken to this consideration, then the enthalpy rise at HZP is greatly reduced to nearly 59 cal/g and the limit can be verified.

6 CONCLUSIONS

This paper is just a preliminary and generic study of the RIA for a reference core which is representative of the Spanish PWR Westinghouse designs. These are not license calculations and the final target is to quantify available margins to comply with the future criteria.

Margins to the new preliminary limits proposed by NRC have been identified for fuel enthalpy and temperature. For these cases, crediting a \( F_q \) reduction for high burnup was needed. Further studies should be performed to quantify DNB margins for intermediate and full power conditions.

3D methods versus 1D were a benefit for HZP conditions but HFP did not significantly improve using these sort of codes. However, it is expected that the HZP conditions will be the most limiting, and 3D methods will be an important contributor to satisfy the future criteria for RIA.
REFERENCES


LARGE SCALE EXPERIMENTS ON GAS DISTRIBUTION IN THE CONTAINMENT

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ABSTRACT

Atmospheric stratification and locally enhanced hydrogen concentration in the containment of a light water reactor contribute to the risk of early containment failure. Previous code validation exercises like e.g. the OECD-NEA International Standard Problem ISP-47 have demonstrated the need for further improvement of models to assure reliable predictions of reactor accident transients. In order to establish a more detailed data base for model development, an experimental program on containment gas distribution has been started at the THAI containment test facility which has a cylindrical steel vessel of height 9.2 m, diameter 3.2 m, volume 60 m³, with optional internal structures. The special instrumentation includes multi-sensor probes for gas concentration and temperature, laser-doppler velocimetry and particle image velocimetry. The experimental program follows a strategy of successive enhancement of complexity, beginning with most simple flow conditions (forced flow), then adding specific interactions like buoyancy, heat transfer, or steam condensation in order to allow for stepwise model development and validation. By this way, the contributions of individual physical interactions to the atmospheric mixing processes are investigated separately in relatively simple geometric configurations, thus facilitating the understanding and interpretation of the measurements.

Selected experimental results of relevance for model validation are presented and discussed. In addition, the status of model simulations in relation to these experimental data is summarised, identifying the phenomena and interactions that are modelled better, and those which call for model improvement. The strategic approach of the THAI experimental program, the corresponding design of the experiments, and the experimental characterization of specific mixing phenomena are innovative in the field of nuclear reactor safety research. The approach differs from previous experiments that were primarily designed to simulate reactor accident scenarios partly or fully; in contrast, the THAI tests are more concentrated on the needs for model development and validation. It can be concluded that a substantial amount of new and detailed experimental results is available to support the development of improved simulation models for atmospheric gas mixing in the containment, especially in relation to turbulence effects. Further experimental work will take into account still remaining open questions that may come up from the ongoing model development.
1 INTRODUCTION

Atmospheric stratification and locally enhanced hydrogen concentration in the containment of a light water reactor contribute to the risk of early containment failure. Especially in the large containment volume of a pressurized water reactor, where the atmosphere is not inerted under normal operating conditions, massive release of hydrogen gas from the primary system during the course of a severe accident with core meltdown may lead to high detonable gas concentrations in the upper dome of the containment building [8]. Atmospheric mixing by natural convection currents and turbulent diffusion can lower the gas concentration, thus reducing the potential for energetic combustion and the associated structural loads on the containment shell. Other mitigating measures are passive or active hydrogen recombination, which are also affected by the spatial distribution of the hydrogen concentration. In order to assess the risk of energetic hydrogen combustion, accurate prediction of atmospheric distributions of gas components, steam, temperature, as well as flow and mixing mechanisms is needed.

Computer models for simulating the thermal hydraulics of a containment atmosphere during a severe accident transient are based on complex equations of mass, momentum and energy conservation, complemented by numerous empirical correlations, turbulence closure relations, boundary conditions, and material property routines. The equations are generally solved by means of a spatial discretization and a time-stepping algorithm. Simplifications of the basic equations, mesh sizes in space and time, and numerical solution methods are subject of choice by code programmer or user. They determine the accuracy of the simulation and the computational effort. Validation of models by means of suitable experiments is needed in order to assess the predictive capability of models for the reactor case.

During past validation studies [6], [2], [1], it was found that simulation of atmospheric stratifications is a special challenge to containment codes. In many cases, code predictions resulted in mixed atmospheric conditions in contrast to the stratified experimental situation. In a reactor case, such prediction could lead to a non-conservative underestimation of the risk from hydrogen combustion. Due to the complexity of the models and the experimental conditions, it was generally not possible to identify the causes of the deviations, rather than to get some indication of potential sources of errors. This finding gave rise to the development of an experimental strategy which is intended to establish a data base suitable to identify specific features of thermal-hydraulic containment models which would cause such misprediction of stratified atmosphere, thereby helping to improve the models and to enable more conservative or accurate calculations of reactor cases. This experimental strategy, its current state of implementation, and some related activities in the area of code development and validation are summarized in the present paper.

2 SOME OUTCOMES OF ISP-47

In the OECD-CSNI International Standard Problem ISP-47 [1], experiments on containment thermal hydraulics in the test facilities TOSQAN, MISTRA and THAI were subject to model simulation. Stratified atmospheric conditions were more intensively studied in the ISP-47 THAI experiment. Figure 1 shows the THAI test vessel, a number of measurement transducer positions, and the injection locations of the ISP-47 test TH13. The acronym THAI denotes the research areas of thermal hydraulics, hydrogen, iodine and aerosols and indicates the multi-purpose character of the facility. The present paper addresses THAI tests on thermal hydraulics and hydrogen distribution only.
The cylindrical steel vessel has a height of 9.2 m, diameter of 3.2 m, and volume of 60 m³. Optional internal structures in case of ISP-47 are the inner cylinder of diameter 1.4 m, height 4 m, and four horizontal condensate trays separated by vent openings. The special instrumentation includes multi-sensor probes for gas concentration and temperature, laser-doppler velocimetry (LDA) and particle image velocimetry (PIV).

The THAI experiment TH 13 for ISP-47 was arranged in four successive phases:

<table>
<thead>
<tr>
<th>Phase</th>
<th>Start (s)</th>
<th>End (s)</th>
<th>Injection</th>
<th>Phenomena</th>
</tr>
</thead>
<tbody>
<tr>
<td>I</td>
<td>0</td>
<td>2700</td>
<td>Helium</td>
<td>Helium cloud formation in the upper dome, gas concentration in the cloud 30 % by volume</td>
</tr>
<tr>
<td>II</td>
<td>2700</td>
<td>4700</td>
<td>Upper steam</td>
<td>Helium cloud mixed with steam, gas concentration in the cloud 21 % by volume</td>
</tr>
<tr>
<td>III</td>
<td>4700</td>
<td>5700</td>
<td>Lower steam</td>
<td>Lower half of cloud mixed with lower air layer, gas concentration 23 % in the cloud, 9 % below</td>
</tr>
<tr>
<td>IV</td>
<td>5700</td>
<td>7700</td>
<td>None</td>
<td>Persistence of upper half of cloud</td>
</tr>
</tbody>
</table>
For the numerical simulation of this experiment, the measured data of phases I and II were communicated to the modellers, while the data of phases III and IV were hidden. In this half-blind code benchmark, the helium-steam cloud formation and the atmospheric stratification at the end of phase II were simulated reasonably well by most models. However, the gas cloud erosion by the buoyant plume from the lower steam injection during phase III was overpredicted by most models, resulting in fully mixed atmospheric conditions during most of phase III and during phase IV.

A simplified quantitative estimate of the risk associated with the measured and simulated gas distributions can be derived from the fraction of injected light gas which forms a detonable mixture. In case of hydrogen, energetic combustion is expected at concentrations above 15% if the atmosphere is not steam-inerted, as can be seen e.g. from the Shapiro diagram ([6], p.138). Inertization occurs at volumetric steam concentrations above 50%, while in the THAI test the steam concentration was below 20%. From the measured and simulated gas concentration distributions, the detonable gas fraction was determined by taking the detonability limit as 16%. The results are shown in Figure 2.

![Fraction of detonable gas mass](image)

**Figure 2: Experimental and simulated fractions of detonable gas mass**

The results from the computer codes COCOSYS, FUMO, KUPOL, and TONUS are based on the Lumped-Parameter concept, while the CFX, GASFLOW, and GOTHIC results are based on the field model or CFD concept. As can be seen from the figure, only one of the COCOSYS simulations calculates the correct atmospheric stratification and detonable fractions during the blind simulation phases III and IV, while all other models predict a fully mixed and non-detonable atmosphere during most of this time. The essential weakness of these models is related to the interaction of the rising, buoyant steam plume with the lower edge of the light gas cloud. Because of the near-stagnating flow conditions in the cloud, the turbulence levels in the interaction zone are in the laminar-turbulent transition regime. Similar interactions between buoyant convection streams and atmospheric stratification layers can be of relevance in the reactor case. So the ISP-47 results may indicate a possible deficit in the
turbulence models. On the other side, the good performance of a lumped-parameter model (which uses just a very simplified turbulence correlation with eddy diffusivity parameter) indicates that other sources of model error may exist, and there will be a more general need for model improvement.

3 EXPERIMENTAL STRATEGY

Experiments for containment code validation, especially involving atmospheric distributions, are often of integral type, where many interacting forces determine the atmospheric flow pattern and the associated turbulent mixing. Such complex arrangements are more similar to the conditions of a reactor accident, but do not allow identifying individual model shortcomings. In order to support the necessary model improvement, simpler experiments with less interacting forces are needed. A systematic approach of model validation begins with a case of minimum complexity, and it continues stepwise by successively adding more interactions, until the complete range of reactor-relevant interacting phenomena is approximated in the model. Each step of this validation process is based on the results of the previous step, and the gradual increase of complexity allows building up the model under full control. A corresponding set of experimental data must be available to support such an approach, representing a series of tests with systematic, stepwise increment of physical interactions and complexity. This strategic concept of experimental work has been implemented in the THAI facility and is presented in the following.

3.1 Momentum Exchange

The basis of atmospheric transport processes is the flow field as calculated from the equation of motion. The simplest flow case is associated with a homogeneous atmosphere, where the fluid density is nearly constant. Since under such conditions there is no natural convection, a forced flow experiment is considered. The flow field is determined by the momentum source, turbulent viscosity, and wall friction. The momentum source is realized by a blower mounted at the top of the inner cylinder, see Figure 3.

![Figure 3: THAI arrangement for momentum exchange test TH 18](image)

In test TH 18, two vent openings of different size are provided in the plane of
condensate trays. The blower generates a near-stationary flow rising in the inner cylinder and returning in the annulus, as indicated in the figure. Flow field measurements were taken by PIV and LDA methods.

CFD simulations of this test [3] showed that relatively small obstacles on the vessel wall (typical dimension 5 cm) had a major impact upon the global velocity field. This effect was simulated well, provided the obstacles were represented by the model geometry. Figure 4 shows the overlay of four PIV measurement fields in the upper dome of the THAI vessel. Because of the large size of the measurement fields it is necessary to use special tracer particles in the flow field that are added periodically into the blower channel. The figure shows velocity vectors and colors according to the magnitude of the vertical velocity component. The outlet from the blower is a circular ring where the highest velocities are measured (red color). A zone of backflow is observed in the dead space inside this ring. The effect of the gutter obstacle on the vessel wall is to divert the downflow (blue color) away from the wall.

Figure 4: PIV velocity data in the THAI vessel upper dome

The CFD simulations typically overpredict the flow resistance of the vents between the condensate trays by a factor 3, and largely underpredict the jet dispersion of the flow away from the vents. These deviations are related to the turbulent momentum exchange, which was simulated by variants of the k-ε model. More work on this model is needed in order to achieve a closer comparison with the experimental data. The experimental flow conditions were in the fully developed turbulent regime, the Reynolds number at the vents was $4 \cdot 10^5$. 
3.2 Momentum-induced Gas Mixing

The second step in the experimental strategy adds the effect of buoyancy forces in the momentum balance. The configuration of the experiment is indicated in Figure 5.

![Figure 5: THAI arrangement for gas mixing test TH 20](image)

In test TH 20, the internal structures (inner cylinder, condensate trays) are removed; the blower is mounted in the lower part of the vessel. By helium injection in the upper part of the vessel, a stratified atmosphere is formed. Then the blower is started. Here it is equipped with a nozzle mounted on the outlet in order to form a simple jet instead of the annular jet in test TH 18. The rising momentum jet interacts with the helium cloud; the flow recirculates back along the vessel walls, as indicated in the figure. In the interaction zone at the lower cloud edge, helium is entrained into the flow field and distributed homogeneously in the lower layer. The cloud is eroded at its lower edge; the position of the edge moves upward with a nearly constant erosion velocity until the entire vessel atmosphere is fully mixed. This process is similar to phase III of the ISP-47 THAI experiment, but instead of a buoyant plume, a neutrally-buoyant momentum jet causes the mixing.

The temporal development of the vertical gas concentration profile is shown in Figure 6. The time axis in the figure begins at the end of the helium injection. The vertical concentration profile shows a fairly steep gradient between elevations 5 m and 6.5 m. After some time interval during which the atmosphere is at rest, the blower is started, and the erosion of the gas cloud shows up in a gradual upward shift of the concentration gradient. Simultaneously, the gas concentration in the lower layer rises. This process continues until the entire cloud is eroded, and the concentration in the vessel atmosphere is homogeneous. In order to quantify the elevation of the interface between the lower layer and the gas cloud, for each point in time the concentrations at the upper- and lowermost sensors are averaged, and the location of the resulting concentration value is determined by interpolation in the vertical profile. This time-dependent location is shown as solid blue line in the figure. By similar interpolation the dashed lines are derived from concentrations of ¾ and ¼ between the maximum and the minimum profile values, in order to evaluate the effective width of the transition zone. The small fluctuations of these lines reflect the turbulent character of the gas cloud erosion mechanism.
Similar mixing processes have been investigated elsewhere [9], [7]; the parameters governing the erosion rate are the jet diameter and velocity at the contact with the cloud edge, and the density difference between the cloud and the lower layer. Turbulent waves are generated on the tip of the jet, and the helium entrainment is determined by the breaking and overtopping of these waves. Backflow of the helium-air mixture occurs in an annular region around the rising jet, similar to the flow of a vertical water fountain in air. The rising jet is subject to turbulent momentum and mass exchange with this annular downflow. Flow patterns of the jet and the interaction zone were measured by PIV; the motion of the cloud edge was determined by measured vertical helium concentration profiles. Experiments were undertaken with different initial helium concentrations and different jet velocities. An analysis of the measured data showed that the spreading of the blower jet is affected by the vessel walls, leading to a reduced jet entrainment while rising in the lower layer, and to a lower width of the jet velocity profile, as compared to a jet in free space.

CFD simulations of this test [4] using variants of the k-ε turbulence model resulted in major overestimation of the jet entrainment and width while rising in the lower layer; the calculated results are closer to the values of a free jet, and the influence of the vessel wall is not properly accounted for. The entrainment rate at the lower edge of the helium cloud was underestimated by a factor 5. This is in strong contrast with the large overestimation of the cloud erosion in the ISP-47 THAI experiment. In a parametric calculation using the laminar flow model, the erosion was still underestimated by a factor 3.5 while the jet width was close to the measurement. This indicates that the more important model deficit is related to the entrainment at the cloud edge.

A detailed discussion of the entrainment process at the cloud edge was given by Turner [10]. The magnitude of the entrainment is controlled by relatively large eddies which are formed through the release of mechanical energy from the rising jet. By means of these eddies, parts of the light gas cloud are enclosed by the heavier gas mixture („engulfment“). The interfacial area between light and heavy gas is strongly increased within these engulfments. Small-scale turbulence and molecular diffusion then lead to homogeneous
mixing of the two components. It is evident that the engulfment in the area of jet deflection cannot be adequately described by means of the traditional turbulent diffusion models like k-ε or SST. These models assume that an enhanced diffusion is caused by shear flow configurations which are characterized by velocity gradients. In the present case, however, the entrainment occurs predominantly in a region where vertical density gradients are coincident with strongly curved streamlines. Such curvature terms are taken into account implicitly in the frame of the more involved Reynold stress turbulence models.

### 3.3 Heat Convection

Step three in the experimental strategy adds the effects of heat to the flow field. The configuration of the experiment is indicated in Figure 7. The cylindrical part of the THAI vessel is equipped with three cooling/heating jackets that can be thermally controlled by circulating heat carrier oil with regulated temperature. The jackets and the other parts of the vessel wall carry a thermal insulation layer on the outer side.

In this case, the upper jacket is kept at a low temperature level, indicated by the blue color in the figure, while the two lower jackets are heated to an elevated temperature, indicated by the red color. The temperature distribution along the vessel wall leads to a natural convection flow upward along the heated portion, as indicated in the figure. The atmosphere is cooled in the upper dome and flows back along the vessel axis to the lower plenum, guided by the wall of the inner cylinder. The condensate trays are removed in order to lower the flow resistance. This type of experiment is planned for autumn 2008. The experimental transient will be run over an extended period of time in order to approach quasi-steady conditions. Measurements will be taken of the flow distribution in the upper dome by using the PIV system in addition to numerous thermocouples to determine the atmospheric temperature distribution. From the heat balance of the jackets, atmospheric and vessel wall temperature measurements, it will be possible to determine heat flux and heat transfer coefficients. These will be correlated with flow velocities along the vessel wall.

![Figure 7: THAI arrangement for heat convection test](image-url)
upper dome will form a challenging pattern which is influenced by momentum exchange, buoyancy forces, heat advection and turbulent diffusion. Radiation heat transfer between parts of the vessel wall will be present, but the air atmosphere does not interact significantly with the thermal radiation.

3.4 Buoyancy-induced Gas Mixing

Step four in the experimental strategy combines the effects of heat with the erosion of a light gas cloud. The experimental configuration is shown in Figure 8. A light gas cloud is established in the upper dome of the THAI vessel by helium injection at an elevated position. Then, the vessel walls below the helium cloud are brought to an elevated temperature by means of the heating jackets. The wall heating generates a buoyant upflow of atmosphere along the vessel wall. The flow interacts with the lower edge of the cold helium cloud where it is bent towards the vessel axis. During its travel along the gas cloud edge the hot air loses some thermal energy to the cold cloud above and entrains some helium which is then transported down into the lower layer.

In this case, there is a superposition of vertical gradients in the temperature and the helium concentration fields with partly compensating effects relative to the vertical atmospheric density gradient. Models which have been validated for heat and mass transport separately during the preceding steps should be able to simulate the superposition correctly. The experiment is planned for autumn 2009.

3.5 Buoyancy-induced Steam Mixing

Step five in the experimental strategy introduces the effects of phase transition, especially the condensation and evaporation of steam. The experimental configuration is shown in Figure 9. The arrangement is similar to the gas mixing experiment in step four, but the helium cloud is replaced by a steam cloud.
The presence of steam adds several new interactions to the thermal hydraulic systems considered in the previous steps. Condensation and evaporation lead to changes in the pressure. There may be condensation on walls and in the atmosphere. Steam in the atmosphere interacts with thermal radiation. The condensate (wall films and droplets) is subject to gravitational acceleration; it can accumulate in lower positions and transport thermal energy. Steam effects are most important for all water reactor accident scenarios. The experiment is planned for spring 2011.

### 3.6 Gas Mixing by Buoyant Steam Plume

This experiment has been conducted in the frame of the THAI OECD project [5]. It was not run as a part of the experimental strategy, but it presents an interesting extension, because it follows a similar idea of phenomena simplification.
The experiment begins with the preparation of a stratified atmosphere by helium injection at an elevated position. Then, a buoyant steam plume is generated by steam release in the lower atmospheric layer. Different from ISP-47 phase III, this steam release is directed vertically upward along the vessel axis, in order to avoid any uncertainty about the spatial position of the buoyant plume. Furthermore, no upper steam injection like in ISP-47 phase II is provided. This allows determining the position of the lower edge of the gas cloud not only by means of gas concentration but also temperature sensors. The erosion of the gas cloud by the steam plume is governed by the following parameters:

- plume diameter when it reaches the cloud edge,
- plume density when it reaches the cloud edge,
- density of the gas cloud, and
- density of the lower atmospheric layer.

The inner cylinder and the condensate trays are installed in order to maintain the ISP-47 geometry. Because of the high fog formation from the injected steam, it was not possible to apply the PIV system in this test. In order to measure the plume diameter, equivalent tests with the steam replaced by a specific helium-air mixture gas were conducted, where the same density conditions as in the original experiment using steam were arranged. Furthermore, an equivalent experiment HM-2 was run using hydrogen instead of helium.

In this experiment, the measured gas cloud erosion velocity was slightly lower than in ISP-47. Model simulations were done under half-blind conditions, with open data from the helium injection phase only. They resulted in a broad range of over- and underpredictions of the cloud erosion velocity, and a systematic overprediction of the steam plume width. This is a remarkable difference when compared to ISP-47, where a systematic and severe overprediction of the erosion velocity was observed. The lumped-parameter models appear to be less dependent upon the differences in the experimental conditions of the tests ISP-47 and HM than the CFD or field models. This higher sensitivity can be traced back to the k-ε turbulence model which calculates diffusive fluxes proportional to the flow velocity gradient. Such velocity gradient is induced near the lower gas cloud edge not only by the buoyant steam plume after it has been deflected from vertical to horizontal, but also by a global flow loop in the lower layer driven by the horizontal momentum source from the steam injection in ISP-47, which is absent in the HM test. The horizontal momentum source is mostly neglected in the lumped-parameter models; this may explain their low sensitivity to potential influences of horizontal flow upon gas cloud erosion. The high sensitivity of the field models, on the other side, appears to be an artifact, because the two types of experiment show much less difference in the erosion velocities than the model simulations. The solution of this issue probably lies in a reformulation of the turbulence correlations for the field models.

4 SUMMARY AND DISCUSSION

A series of experiments on light gas distribution and mixing in the containment of a nuclear reactor is being executed at the THAI large-scale experimental test facility. The experimental program follows a strategy of successive enhancement of complexity, beginning with most simple flow conditions (forced flow), then adding specific interactions like buoyancy, heat transfer, or steam condensation in order to allow for stepwise model development and validation. The development of this strategy was motivated by the observation of substantial difficulties in modelling the distribution and mixing of atmospheric stratifications in the containment. The approach differs from previous experiments that were primarily designed to simulate reactor accident scenarios partly or fully; in contrast, the THAI tests are more concentrated on the needs for model development and validation. A substantial amount of new and detailed experimental results is already available to support the
development of improved simulation models for atmospheric gas mixing in the containment, especially in relation to turbulence effects. Further experiments will be run in yearly intervals in order to allow for code and model improvement time. Code validation and model development in connection with the THAI program is mainly undertaken by GRS, University of Stuttgart-IKE, and ANSYS Germany.

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Spatial Discretization in LWR Cell Calculations with HELIOS 1.9:
Influence on $k_{inf}$ and Flux Distribution

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ABSTRACT

Cell- and lattice calculations are the fundament for all deterministic static and transient 3D full core calculations. The spatial discretization used for the cell- and lattice calculations influences the results for these transport solutions significantly. The arising differences in the neutron flux distribution due to different spatial discretization are demonstrated. These differences in the flux distribution cause significant changes in the $k_{inf}$ value. An evaluation of the $k_{inf}$ value for the case of infinitely fine discretization is made. Strategies for improving the discretization are developed and their influence on the calculation time is evaluated.

1 INTRODUCTION

The aim of many reactor calculations is the determination of the neutron flux and the nuclear power distribution. These distributions are in general calculated by solving the space, energy and angle dependent static transport equation. Thus, a reliable computation of the neutron flux distribution within the reactor core would require the solution of the space-, energy- and angle-dependent neutron transport equation for the full nuclear reactor core. It is not feasible to solve exactly the neutron transport equation for realistic reactor core geometries in detail in practical use. Thus deterministic reactor calculations are split into the cell and lattice calculation based on static transport and the core simulation based on nodal methods. The cell and lattice calculations are mostly based on multi group transport calculations within two dimensions considering unstructured meshes. The resulting neutron flux is used for the preparation of few group cross sections like they are used in the nodal 3D full core simulation codes.

One commercial standard product is the Studsvik Scandpower code system HELIOS 1.9. »The transport method of HELIOS is called the CCCP method, because it is based on current coupling and collision probabilities (first-flight probabilities) … the system to be calculated consists of space elements that are coupled with each other and with the boundaries by interface currents, while the properties of each space element – its responses to sources and in-currents – are obtained from collision probabilities« [1].
The applied collision probabilities method is based on the flat flux and flat current approximation. We assume that particles are emitted isotropically and uniformly within each discrete volume. This is known as the flat flux approximation. The flat flux approximation places a restriction on the mesh: if the flux varies rapidly within a region, the mesh should be refined sufficiently to ensure that the flux is well approximated by a piecewise – constant function [2].

This approximation requires a careful spatial discretization to produce reliable results. The problem of the spatial discretization effect on the solution of the diffusion equation by finite difference methods has been studied in the past [3], [4].

The influence of the spatial discretization [5] on the multiplication factor of an infinite system $k_{inf}$ constructed by single cells will be presented here for the collision probability method of the code system HELIOS 1.9.

2 GENERAL DATA

All performed calculations are based on the GRS benchmark for a UO2 fuel assembly [6]. The major parameters are: 18x18-24 UO2 fuel assembly of Konvoi type with 4% initial enrichment and a specific power of 37.5 W/gHM.

The HELIOS 1.9 standard settings are sharpened to reduce uncertainties due to convergence criteria and reproduction of the geometry. The boundaries are coupled (boundary coupling) to reach an infinite lattice. The cross sections are taken from the unadjusted 190 energy group HELIOS 1.9 internal library [7].

3 FLUX DISTRIBUTION

Firstly the influence of different discretization strategies a standard and a detailed fine mesh flux distribution is calculated. For the visualization purposes the original 190 energy group neutron flux is integrated and two group fluxes with energy cut off at 0.625 eV are calculated.

Figure 1 shows the thermal and fast flux distributions for the coarse mesh discretization. In the left picture for the thermal flux, the flux maximum is in the moderator area and the minimum is in the fuel area, while in the right picture for the fast flux, the maximum is in the fuel area and the minimum is in the moderator area. This fact is physically explainable - the fast neutrons are born in the fission reactions in the fuel. These neutrons are thermalized as soon as they reach the moderator. After this process they form the high thermal neutron flux in the moderator region. When thermal neutrons are appearing in the fuel they mostly create fission reactions though they disappear in this region rapidly. Additionally absorption processes influence the neutron economy in the system as well in the fuel as in the moderator and in the cladding. Due to the coarse discretization with three regions only three discrete values in every neutron flux group are available and the neutron flux within every region is flat.

The visualization of the neutron fluxes on the unstructured mesh is done with the visualization tool IRIS terracidata developed at Institute of Safety Research [8].

A very fine mesh discretization is created to calculate the approach to the true neutron flux distribution. The neutron flux distribution for the very fine mesh discretization with 328 regions is shown in Figure 2. The thermal neutron flux is once more lowest in the fuel, but there is now a neutron flux distribution within the fuel where the thermal neutron flux rises from the center of the fuel rod to the surface. This tendency continues in the cladding and the moderator, which leads to a steady rising thermal neutron flux with the maximum in the corner of the single cell. The neutron fast flux of the fine mesh solution increases from the corner of the single cell to the center of the fuel rod.
The difference in the $k_{\text{inf}}$ value for the two presented cases is remarkable. While the $k_{\text{inf}}$ of the coarse mesh solution is 1.315123, the one for the very fine mesh solution is $k_{\text{inf}} = 1.320866$.

Figure 1: Two group thermal (left) and fast (right) neutron flux distribution (cut off at 0.625 eV) in a single cell of an infinite lattice for the coarse mesh discretization consisting of the fuel, the cladding and the moderator area - $k_{\text{inf}} = 1.315123$ (the visualization is created with IRIS [8])

Figure 2: Thermal (left) and fast (right) neutron flux distribution in a single cell of an infinite lattice for a very fine mesh discretization consisting of 328 independent regions - $k_{\text{inf}} = 1.320866$

4 ANALYSIS OF THE DISCRETIZATION EFFECT

For the analysis of the spatial discretization effect further investigation of the $k_{\text{inf}}$ value is shown in Figure 3. The behaviour of the $k_{\text{inf}}$ as a function of the average distance between the regions is illustrated. The convergence behaviour is roughly linear with the averaged distance between regions (averaged cell size) in the beginning of the spatial discretization refinement. This observation coincides with the description in the literature about the accuracy depending on the first order in space [9]. For the further refinement the $k_{\text{inf}}$ value is not completely converged for the solution with 328 regions or an averaged distance between the regions of 0.07 cm ($\sqrt[3]{\frac{328}{1.272 \text{cm}}^2} = 0.07 \text{cm}$). This discretization leads to a $k_{\text{inf}}$ value of 

\[
\left(1.320866 \pm 435 \text{ pcm}ight)
\]

higher than for the coarse mesh calculation. A further refinement by doubling the number of meshes is not possible due to the available computer resources especially memory and the system limits of HELIOS 1.9.
The neutron flux distribution of this calculation, which is used as the reference calculation, is shown in Figure 2. The spatial discretization refinement in the fuel area only (red curve with dots) has already completely converged at an averaged distance of 0.14 cm – visible in the horizontal end of the curve. The spatial discretization refinement in the moderator only (blue curve with triangles) shows the change of the $k_{\text{inf}}$. This result is not yet completely converged. Finally, the spatial discretization refinement in the complete single cell (black curve with squares) represents $k_{\text{inf}}$ as a function of the averaged distance between the regions. This curve is a superposition of the results for the discretization in the fuel and in the moderator. An estimation for the convergence for a next refining step can be made on the basis of the already reached asymptotic convergence in the fuel and the dominating mean free path there (fast group is dominating – mean free path 2.499 cm) and the mean free path in the moderator (thermal group is dominating – mean free path 0.560 cm). The asymptotic convergence in the fuel is reached when the averaged distance is roughly 0.056 mean free paths thick. This result can be used to estimate the averaged distance between the regions to reach a converged result. This estimation leads to an averaged distance of 0.031 cm and means; at least one additional step of refinement would be needed. This leads to the conclusion that the reduced gradient is already a sign for the convergence. A linear extrapolation is used to estimate the worst case. This extrapolation points to an approximate $k_{\text{inf}}$ worth of roughly 1.322 as upper limit for an infinitesimal small distance between the regions (see dotted black line in Figure 3).

The same overall behaviour of the influence of spatial discretization on the $k_{\text{inf}}$ worth and the much slower convergence in the moderator area has been observed by Suslov et al. for a two energy group calculation of a single cell by the method of characteristics [10]. The final converged results for the method of characteristics has been cross checked with Monte-Carlo calculations based on the identical cross section basis. Only very small differences have been observed.

As well the here performed study of the convergence in the collision probability method as the study of Suslov et al. (method of characteristics) show convergence of the $k_{\text{inf}}$ value, but both studies demonstrate the need of a sufficiently fine discretization to reach asymptotic convergence. The over all situation has improved significantly compared to previous benchmark calculations of Buckel et al. [3], [4]. No asymptotic convergence has been reached at all in these calculations for different finite difference schemes and extrapolation was the only possibility for the exact determination of the $k_{\text{inf}}$ value.
5 CHANGES IN EFFICIENCY

The efficiency of the calculation procedure is naturally affected by the spatial refinement of the system. The computation time (used computer: AMD Athlon™ 64 3200+, 2.2 GHz, 2GB RAM) for the used steps for the study of the convergence of the $k_{\text{inf}}$ worth shown in Figure 3 is shown in Figure 4. The indicated time is only the HELIOS 1.9 calculation time without pre-processing with AURORA and post processing with ZENITH. The calculation time becomes especially for the very fine mesh solution really long. The overall time computation time raises nearly exponentially. This is a typical effect caused by the collision probability solution method used in HELIOS 1.9. The characteristic of the collision probability method is a rather full matrix which is complicated to invert especially with increasing size.

A comparison of the thermal and fast neutron flux distributions and the cutting of the regions for the coarse mesh and two different suggested discretization strategies - an optimized mesh and an elaborate mesh discretization [5], [12] is given in Figure 5. The detailed results for the calculation time and the deviations in the $k_{\text{inf}}$ for the 3 optimized discretization steps (see Figure 6) and an elaborate discretization step (20 regions in the fuel, one in the cladding and 72 regions in the moderator) are put together in Table 1. The results are compared to the results for the coarse mesh discretization. The $k_{\text{inf}}$ worth for all three optimized discretization steps is roughly 250 pcm while the $k_{\text{inf}}$ worth for the elaborate discretization step is only 41 pcm away from the converged result whereas the $k_{\text{inf}}$ worth for the coarse mesh discretization is not really reliable because of the offset of more than 400 pcm.

The introduction of additional regions complicates the calculation thus it costs money due to rising computation times. The real computation times for the single cell calculations are indicated in Table 1. The absolute value is not very meaningful because the initial settings for the calculations are fixed for the creation of reference solutions with small calculation regions and not for production purpose. More interesting is the time consumption factor in the last row. To calculate the solution for the optimized discretization takes about 3 to 4 times
more CPU time for the calculation. The elaborate solution consumes a factor of 100 more time than the coarse mesh solution and the reference solution takes more than 1000 times longer. This number does not refer to the time needed for the complete calculation sequence but only to the HELIOS calculation time without pre- and post processing [7].

Figure 4: HELIOS 1.9 computation time dependent on the average length per region for the study of the dependence of the $k_{inf}$ worth on the spatial discretization

Figure 5: Thermal (upper) and fast (lower) neutron flux distribution studied for the coarse mesh discretization (left), the optimized discretization of step 2 (middle) and the elaborate discretization (right)
Table 1: Comparison of criticality and computation time for different discretization schemes for the investigated single cell (AMD Athlon™ 64 3200+, 2.2 GHz, 2GB RAM)

<table>
<thead>
<tr>
<th></th>
<th>coarse mesh discr.</th>
<th>optim. discr. step 1</th>
<th>optim. discr. step 2</th>
<th>optim. discr. step 3</th>
<th>elabor. discr.</th>
</tr>
</thead>
<tbody>
<tr>
<td>(k_{\text{inf}}) worth</td>
<td>1.315123</td>
<td>1.317278</td>
<td>1.317708</td>
<td>1.317906</td>
<td>1.320322</td>
</tr>
<tr>
<td>deviation pcm</td>
<td>-434.8</td>
<td>-271.6</td>
<td>-239.1</td>
<td>-224.1</td>
<td>-41.2</td>
</tr>
<tr>
<td>CPU time (min)</td>
<td>0.14</td>
<td>0.36</td>
<td>0.47</td>
<td>0.61</td>
<td>15.44</td>
</tr>
<tr>
<td>additional time consumpt. factor</td>
<td>1.0</td>
<td>2.6</td>
<td>3.4</td>
<td>4.4</td>
<td>110.3</td>
</tr>
</tbody>
</table>

6 CONCLUSION

A detailed analysis of the influence of the spatial discretization on the neutron flux distribution and the \(k_{\text{inf}}\) worth in a single cell is performed. The analysis of the neutron flux distribution shows a significantly different neutron flux distribution for the coarse mesh discretization and the very fine mesh solution because in every calculation region only one neutron flux value is available. This is due to the basic assumption for the collision probability method as solution of the transport equation called the flat-flux approximation – but this is not only in the collision probabilities the case but also in most other deterministic solution methods for the transport equation. The flat-flux approximation determines that the result for the flux distribution in every calculation is only as good as the used spatial discretization. In former times using structured mesh codes it was the intention of the user to refine the meshing to have the possibility to approximate the geometry as good as possible. In modern codes with unstructured mesh possibilities the geometry can be reproduced very well by a small number of meshes. This improved geometric possibility misguides the user to neglect the detailed reflection about the correct modelling of the system by appropriate discretization, but exactly this appropriate discretization is needed to produce reliable calculation for the neutron flux distribution and the \(k_{\text{inf}}\) worth.

However, not only the flux distribution but also the \(k_{\text{inf}}\) worth is very much dependent on the spatial discretization. It is nearly impossible to get a reliable result for the \(k_{\text{inf}}\) worth without performing a real very fine mesh calculation. The detailed investigation of the effect on \(k_{\text{inf}}\) shows a difference of roughly 430 pcm for the coarse mesh discretization as compared to the almost converged very fine mesh solution for the single cell. The relevance of this value can be seen with some background information. Some years ago the developers of HELIOS created an adjusted library to get better coinciding results for the calculations with existing experiments. The difference in the results for the single cell for using the adjusted
and the unadjusted library is roughly 370 pcm. »This means that the effect of the library adjustment on \( k_{\text{inf}} \) is hidden in an adequate spatial discretization.« [13].

A significant improvement can be gained by using an optimized or elaborate discretization scheme [5], [12]. Unfortunately is neither the optimized scheme nor the elaborate scheme really capable of being automated because elaborated knowledge about the system is needed. At a first glance some ideas for an optimized scheme can be gained. In the area of the fuel where the fast flux is dominant it seems to be helpful to add regions where the major gradient occurs. In contrast in the area of the moderator it seems to be helpful to divide the area by halving the longest straight outward distances. In principle the mesh size should be oriented at the different mean free path lengths of the dominant flux group in the different areas – fuel and moderator. In light water reactors should be the size of the regions in the moderator smaller due to the significantly smaller dominating mean free path there than in the fuel. For a really reliable result by the elaborate scheme important knowledge about the convergence behaviour is needed.

The discretization dependent results will be used for the production of absorption, fission and scattering cross sections sets. This means; The deviations in the results for neutron flux and \( k_{\text{inf}} \) will be carried forward to all operational and safety relevant full core calculations, because these values are finally used for the creation of the few energy group cross section sets needed for the 3D nodal full core calculations for operation and safety evaluation. Exactly these cross section sets are significantly influenced by the used spatial discretization [5], [11], [12].

As a final comment let’s fix - it’s not a bug it is a matter of the used approximation. The user performing the calculation has to think for the appropriate discretization for producing reliable results! The produced results can only be as good as the modelling of the nature in the input of the computer code. It might be worthwhile to have a look at strategies from other areas of computer simulation, e. g. “Best Practice Guidelines for creation of reliable results of CFD calculations” [14].

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The Role of Entrained Droplets in Precursory Cooling During PWR Post-LOCA Reflood

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ABSTRACT

During PWR reflood following a loss of coolant accident, precursory cooling prior to the arrival of the rewetting front is vital to avoid unacceptably high cladding temperatures. This precursory cooling is by a flow of superheated vapour, with entrained saturated drops, which evaporate into the vapour and act as a heat sink. In this paper we investigate a complementary mechanism; the direct cooling of the cladding by the drops themselves. Cladding temperatures are such that wetting does not occur. On the contrary, droplets bounce off a vapour cushion formed during the ~10ms or so they are in close proximity to the cladding. Using a combination of previous experimental correlations and recent CFD calculations, we estimate the ‘Joules per droplet’ and the ‘droplets per square metre per second’ under these circumstances. The heat extracted by those droplets is found to be about 1/10 of the heat extracted by single phase vapour under typical reflood conditions. There are necessarily considerable uncertainties in these figures, but it seems that direct cooling by droplets, not generally incorporated in analyses of reflood, could actually be making a worthwhile contribution to keeping cladding temperatures tolerable.

1 INTRODUCTION

After a design-basis Loss of Coolant Accident in a PWR, cold water is reintroduced to the core by bottom-up reflooding. At the rewetting front, violent boiling generates slugs of liquid, which are entrained in the upward flow of vapour. The slugs break up and form drops. In the region above the rewetting front, the conditions are thus characterized by a flow of superheated vapour between even hotter metal surfaces, with a population of small saturated droplets entrained in the vapour flow. Cooling of the fuel by this droplet-steam mixture above the rewetting front (“precursory cooling”) is vitally important in the reflood process. The main mechanism for this precursory cooling of the fuel is by convective heat transfer to the vapour, with the vapour in turn being cooled by the evaporation of the entrained saturated droplets. The cladding temperatures are such that wetting of the metal by the entrained
droplets does not occur, with droplets instead rebounding from a cushion of vapour generated between the droplet and the surface while they are close together. Whilst heat transfer during this interaction is inhibited by the vapour layer, it may be that in aggregate the multiple droplet-wall interactions are able to extract a worthwhile amount of heat. In this paper we attempt to estimate the rate of heat transfer attributable to these droplet-wall interactions.

The route adopted is, in outline, as follows. The first of the two main stages is to estimate the heat transferred during the rebound of a single droplet. This is achieved using the semi-empirical correlation of Wachters and Westerling [1] for droplet evaporation rate, augmented by the analytical correlations proposed by Biance et al. [2], based on their experimental observations, and by the Level-Set CFD calculations of the droplet spreading behaviour of Chatzikyriakou et al. [3].

Given then a tool to estimate the heat extracted by a single droplet, the second stage is to estimate the rate of droplet-surface interactions. This is done from the correlations of Hewitt and Govan [4] for droplet deposition rates, and can be extended to the case of obstructed flows by using the correlations of Okawa et al. [5].

In section 2, we present a brief review of previous studies of droplet surface interactions, with emphasis on the works on which we rely for the sequence of analyses described above. In section 3 we list the parameters of the problem we have selected. In section 4 we address the first stage of estimation of the heat extracted per droplet. In section 5 we estimate the rates of droplet-surface interactions to be expected in annular dispersed flow. We also combine these results, and estimate the cooling likely to be attributable to non-wetting droplets under reflood conditions. Conclusions are presented in section 6.

2 LITERATURE REVIEW

2.1 Individual droplet behaviour

There is a large body of published work on the interactions of droplets and hot surfaces. We will concentrate here on those previous studies that we used to form the main basis of the present work.

Wachters and Westerling [1] conducted experiments to determine the heat transfer when a liquid drop bounces off from a hot surface. They used 1.15 mm radius saturated water droplets falling vertically onto a gold surface at 400°C inclined at 30 degrees to the horizontal. The surrounding atmosphere was saturated steam. The droplet motion was recorded in a series of high-speed photographs, allowing the hydrodynamic behaviour to be observed. The time dependent surface area of the base of the droplet in (near) contact with the hot plate was determined from the photographs, as was the position of the centre of mass of the droplet.

Conditions, under which the droplets bounced, rather than broke up upon impact, were identified, and it was found that the critical value of Weber number above which droplet breakup occurs was about 80 (this Weber number and all others in this analysis are based on the normal component of the droplet velocity). Below this critical Weber number, the droplet bounced without disintegration. It is perhaps worth noting that other researchers have observed lower critical Weber number values. Yao and Cai [6], for example, reported a large-angle We of about 50, and for small angles of impaction a value of 20 – 30.

Wachters and Westerling [1] found the residence time of the droplet to be approximately equal to the first-order vibration period of a freely oscillating droplet, indicating that the contact time depends on the initial droplet radius and material, rather than the approach velocity.

Using those hydrodynamic observations, in combination with the correlation derived for the evaporation rate in an earlier paper of the same group of researchers [7], Wachters and
Westerling [1] derived a correlation for the vapour film thickness. A second correlation gave the volume of vapour evaporated from the bottom surface of the droplet during the droplet-wall contact period:

\[
\Delta V_{\text{drop}} = \int_0^T \left[ \frac{\pi^3 \rho_c k_s (T_m + T_s)^3}{3 \rho_s \mu_L L_s T_s} \right]^{1/4} R_0 V(t)^{1/4} \left( g + \frac{d^2 h}{dt^2} \right)^{-1/4} dt
\]  

With the droplets being saturated, all of the heat extracted from the surface can reasonably be assumed to be used in generating further vapour, so this correlation in effect gives the heat extracted.

Biance et al. [2] examined the elasticity of a non-wetting droplet both experimentally and analytically. They used a photographic method to observe the bottom radius of a millimetric water droplet incident vertically on a 280°C metal surface. Upon impact, the droplet initially spreads, reaches a maximum extension and then recoils and finally bounces from the surface. At the time when the droplet radius is largest its maximum radius is very close to the radius of the region in ‘contact’ with the hot surface.

The maximum base radius was observed to scale with the perpendicular Weber number in the fashion:

\[
\frac{R_{\text{max}}}{R_s} \propto We^\alpha
\]  

with a value of the exponent of 0.3 .

Additionally, they proposed a simple analytical model to predict the maximum droplet base radius, built upon the earlier work of Chandra and Avedisian [8]. This earlier work was based on the theory that the kinetic energy converts to surface energy (the drop deforms as it hits the solid), and the Chandra and Avedisian [8] analysis concluded that this leads to the maximum radius scaling with Weber number to the power 0.5. The more recent analysis of this by Biance et al. [2] proposed that the value of the exponent was 0.25, in good agreement with their experimental value of 0.3 mentioned above.

Chatzikiriakou et al. [3] conducted numerical simulations for a millimetric water droplet incident on a 300°C solid surface, using the Level-Set method. A correlation for the maximum base radius was identified which was similar in form to that of [2], but with a value of the exponent of 0.23.

These three sets of data (the Biance measurements, the Biance model, and the Chatzikyriakou numerical simulations) are shown in Figure 1.
2.2 Droplet deposition rates

Hewitt and Govan [4] measured the droplet deposition rate in an annular upward air-water flow. Both air-water and steam-water mixtures were studied flowing in 8-40mm diameter tubes at a range of pressures. The liquid and gas mass fluxes were in the range of 100-600 kg/m$^2$s and 100-300 kg/m$^2$s respectively. The liquid content in the flow was in the range 1 - 10 kg/m$^3$, and droplet diameters were typically 0.1 to 0.2mm, corresponding to a droplet number density of $\sim 10^9$ drops/m$^3$.

The liquid deposition rate was measured by removing the annular liquid film, and then observing the rate at which a new film builds up [9].

They obtained a correlation for the droplet deposition:

$$k' = k_d \sqrt{\frac{\rho_L D}{\sigma}} = 0.18 \quad \text{if } C^* < 0.3$$

$$k' = k_d \sqrt{\frac{\rho_L D}{\sigma}} = 0.083 C^{0.65} \quad \text{if } C^* > 0.3$$

where $C^* = \frac{C}{\rho_L}$ is the dimensionless droplet concentration and $k' = k_d \sqrt{\frac{\rho_L D}{\sigma}}$ is the dimensionless deposition mass transfer coefficient. Then, the flux of droplets impinging on the wall (kg/m$^2$s) was given by:

$$N = k \cdot C$$
In a study largely using BWR-conditions, Okawa et al. [5] used similar techniques to examine the effects of an obstacle in the flow on the deposition rate of droplets. They used broadly the same range of pressure and mass flux conditions as Hewitt and Govan [4], and a correlation similar to that of [4] was produced for the obstacle-free case. Then, using obstacles of 20% of the cross sectional area of the tube, they observed an increase in the deposition rate of droplets. For the obstacles studied, the deposition rate was measured to be approximately 50% higher than without obstacles.

3 CONDITIONS CONSIDERED

Our prime purpose here is to investigate the cooling circumstances in a PWR sub-channel during reflood. In Table 1 below are summarised to the conditions which we consider in our study, conditions which are typical of those that might be obtained during reflood.

Table 1: The conditions considered for this study

<table>
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<th>PARAMETERS</th>
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<th>VALUES</th>
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</thead>
<tbody>
<tr>
<td>Sub-channel hydraulic diameter</td>
<td>m</td>
<td>0.0116</td>
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<tr>
<td>Vapour axial velocity</td>
<td>m/s</td>
<td>20</td>
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<tr>
<td>Typical droplet transverse velocity</td>
<td>m/s</td>
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<td>Pressure</td>
<td>MPa</td>
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<tr>
<td>Droplet mass concentration</td>
<td>kg/m³</td>
<td>10</td>
</tr>
<tr>
<td>Droplet radius</td>
<td>mm</td>
<td>1.15</td>
</tr>
<tr>
<td>Cladding surface temperature</td>
<td>K</td>
<td>600-700</td>
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<tr>
<td>Vapour temperature</td>
<td>K</td>
<td>373</td>
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<tr>
<td>Droplet temperature (saturation)</td>
<td>K</td>
<td>373</td>
</tr>
<tr>
<td>Vapour heat transfer coefficient</td>
<td>W/m²K</td>
<td>40</td>
</tr>
<tr>
<td>(via Dittus Boelter)</td>
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<td></td>
</tr>
<tr>
<td>Heat flux to vapour</td>
<td>MW/m²</td>
<td>0.015</td>
</tr>
</tbody>
</table>

Note that (a) the wall temperature is well above the Leidenfrost limit, (b) the perpendicular Weber number, at about 10-20, is below that at which droplet break up is to be expected and (c) radiation effects are ignored.

4 ESTIMATION OF THE AMOUNT OF HEAT EXTRACTED PER DROPLET-WALL INTERACTION

In their experiments, Wachters and Westerling [1] measured, amongst other things, the generation of vapour from the surface of a drop adjacent to the hot solid. They developed Eq.(1) based upon their observations.

This expression gives the total volume of vapour generated as the time integral of a function of (inter alia) the instantaneous droplet base radius during the interaction, and of the instantaneous acceleration of the droplet centre of mass.

It is obviously a trivial matter to convert to the volume of vapour generated into an amount of energy extracted, via the latent heat. However, we cannot use this correlation directly, as neither the time-dependent droplet base radius, nor its acceleration, are known. The a priori information that characterises an interaction is the droplet approach velocity, and the droplet radius.
We will now consider how we can estimate the time dependent contact radius, and acceleration.

4.1 The time integral of the droplet base radius

In the results of [1] there are plotted graphs of droplet base radius versus time, and similar information is available from the measurements of Inada et al. [10]. If these measurements are re-plotted, with the axes scaled respectively by the maximum base radius reached, and the total interaction time, it turns out that the resulting curves lie quite close to each other, for a reasonably wide range of interaction conditions (the velocity varying by a factor of ~2, droplet mass by a factor ~5). This is shown in Figure 2.

![Figure 2: Normalised experimental values for the time dependent droplet base radius, and our empirical fit to these](image)

We have developed an empirical expression (the particular form is a probability density function) that fits this curve well. This curve is given by:

\[
\left( \frac{R}{R_{\text{max}}} \right)_{\text{FITTED}} = 0.19182 \left( \frac{I}{I_{\text{max}}} \right)^{0.52485} \left( 1 - \left( \frac{I}{I_{\text{max}}} \right) \right)^{1.48791}
\]

and is also shown in Figure 2.

We can then use this to obtain a reasonable estimate of the time-dependent droplet base radius, as long as we do know (a) the maximum base radius reached, and (b) the total interaction time.

4.1.1 The maximum droplet base radius

As discussed above, several researchers have addressed the issue of determining the maximum droplet base radius reached as a function of the interaction conditions. These three
sets of data (the Biance measurements, the Biance model, and the Chatzikyriakou numerical simulations) were shown in Figure 1. All propose a similar form of relationship (Eq.(2)) with a modest variation in the value of the exponent to which the Weber number must be raised.

4.1.2 The droplet - surface interaction time

The duration of the interaction between the droplet and the surface was also studied both by [1] and [2]. The duration was found to be rather insensitive to the approach velocity, but much more sensitive to droplet size, and approximately equal to the first-order vibration period of a freely oscillating droplet:

\[ \tau = \pi \sqrt{\frac{\rho R^3}{2 \sigma}} \]  

(6)

4.2 The time integral of the acceleration of the droplet centre of mass

The time-dependent acceleration of the centre of mass is required to evaluate the correlation in Eq. (1) for the vapour generation in the Wachters experiments, and we have attempted the same approach to characterising it as was used above for the base radius.

The time is again normalised by the total interaction time. The height of the centre of mass is normalised by the initial droplet diameter, and we plot in Figure 3 the observed motion for the two approach velocities used by [1]. An approximate fit to these two curves was obtained, using the fourth-order polynomial of Eq.(7), and this is shown in the same figure. The second derivative of this polynomial is then used in the evaluation of Eq.(1).

\[ h(t) = 2 \cdot R_c \cdot (1.8133 \cdot \left(\frac{t}{t_{\text{max}}}\right)^4 - 5.1869 \cdot \left(\frac{t}{t_{\text{max}}}\right)^3 + 6.3550 \cdot \left(\frac{t}{t_{\text{max}}}\right)^2 - 2.2541 \cdot \left(\frac{t}{t_{\text{max}}}\right) + 0.2738) \]  

(7)

4.3 Summary and validation of this heat extraction model

It is perhaps helpful to gather together and summarise the stages involved in applying the model.

Using Eq.(2) and Eq.(6) in Eq.(5) gives us an estimate of the time dependent base radius. Then, using Eq.(6) in Eq.(7) similarly gives us an estimate of the time dependent acceleration of the centre of mass.

Both of these are then inserted into Eq.(1), and the equation is integrated through time to determine the heat transferred during the interaction.

As a test, in Figure 4, we compare the predictions of the base radius given by our model, with those measured in the Wachters experiments from which our model was derived. As is seen, the agreement is reasonable.
Figure 3: The normalised height of the centre of mass for the two droplet approach velocities studied by [1], and the approximate fit to these employed here.

![Figure 3: Normalised Height of Centre of Mass](image)

Figure 4: Droplet maximum base radius as a function of time (We=20, We=7). Experimental results by [1] and values given by our fitting model.

![Figure 4: Maximum Base Radius](image)

As a complementary test, we examine our ability to predict the vapour film thickness underneath the droplet versus time. This too was measured in the Wachters experiments, and the correlation they obtained for this quantity is:

![Graph showing vapour film thickness](image)
The film thickness is seen to depend in a similar fashion upon the time integral of the droplet base radius, and on the acceleration of the droplet centre of mass. We have used the approach outlined above to predict the film thickness by this correlation, and these predictions are seen in Figure 5 to be in reasonable agreement with the measurements of [1].

\[
d_p(t) = \left[ \frac{3\pi \mu k(T_m - T_s)(T_m + T_s)}{16 \rho \rho_L T^3} \right]^{1/4} R_s V(t)^{3/4} \left( g + \frac{d^2h}{dt^2} \right)^{-1/4}
\]

(8)

Figure 5: Vapour layer thickness under the centre of the droplet bottom surface, through time (We=20, We=7). Experimental results from [1] and results computed from the present fitting model.

4.4 Application of the model to predict interaction heat transfer

Applying the above analysis for a 1.15mm radius water droplet near saturation, we show in Figure 6 the computed droplet evaporation rate, volume, and cumulative heat loss versus time. Only the approach velocity, expressed via the Weber number, changes in this parametric study. That is expressed in terms of the perpendicular Weber number. It is seen that an increase in the normal velocity component results in an increase of the evaporation rate. Furthermore, the amount of heat extracted by the droplet by the end of the interaction is larger when the droplet approaches on the wall at a higher velocity.
Figure 6: Computed droplet evaporation rate, volume, and cumulative heat loss versus time. The droplet radius is 1.15 mm. The Weber number here is a measure of approach velocity.

As noted earlier, it is of course the total reduction in droplet volume, or equivalently the volume of vapour generated, which is proportional to the total heat transferred during the interaction. This is found by a time integral. Performing this, we can estimate that a 1.15 mm near-saturation water drop would have extracted about 0.05 J from the hot wall, during its ~10 ms in close proximity. The average heat flux during this period, approximating the average contact area by one half its maximum, is ~2 MW/m². This is large compared to the vapour-only heat fluxes as computed above, of approximately 0.01 – 0.02 MW/m², but of course it obtains for only brief periods, over only small areas.

Below, in Figure 7, we can see how the heat extracted by a millimetric droplet scales with the droplet impaction normal velocity.
although the duration of the interaction does not depend on the approach velocity, the evaporation rate and consequently, the heat loss from the hot surface are affected. The higher this velocity, the higher the Weber number and according to Eq.(2), the larger the maximum spreading of the droplet base. This means that the droplet spreads over and cools a larger area of the hot surface.

5  ESTIMATION OF THE AUGMENTATION OF THE HEAT TRANSFER BY DROPLET-WALL INTERACTIONS

In order to estimate the heat transfer induced by those droplet-wall interactions in a dispersed flow one should initially compute the amount of droplet-wall interactions per unit time. There is inevitably some considerable uncertainty in estimating the rate of droplet deposition on the walls for the conditions we analyse. The approach we have followed is to use the Hewitt and Govan correlation of Eq.(3), but with parameter values for our conditions. Our droplet mass concentration is 10 kg/m³, at the upper end of the Hewitt and Govan range, but our droplet radius, of 1.15mm, is substantially larger. These yield a droplet number density of ~1.5x10⁶ drops/m³, much lower than that observed by [4] due to our droplet radius being ~ten times larger. From these, we compute a rate of arrival of droplets at the wall of ~5x10⁴ drops/m².s.

We have estimated above a heat transfer per droplet – wall interaction of 0.05J, and a rate of interactions of 5x10⁴ /m².s. Combining these, we estimate a wall heat flux associated with wall-droplet interactions of ~2.5 kW/m².

It is helpful to set this in context by comparison with heat fluxes due to single-phase convective heat transfer to the vapour. For the flow conditions we listed in Table 1, we estimated a vapour heat flux of ~15 kW/m².
6 CONCLUSIONS

For the conditions we have studied, the heat flux associated with non-wetting droplet-wall interactions has been estimated to be $O(1/10)$ of that attributable to single phase vapour heat transfer. However, there are necessarily some significant uncertainties in this analysis, associated both with the estimate of ‘Joules per interaction’, but perhaps particularly with the estimate of the number of interactions per second. Direct measurement of ‘Joules per interaction’ is the subject of a current research project (KNOO) in the context of which, direct measurements of the heat removed by very small droplets are being conducted via infrared thermography. The estimate of the interactions per second was derived here from earlier correlations with droplet number densities quite different, and it is perhaps in refinement of this estimate that further effort is required.

Nonetheless, there do seem to be some grounds for believing that a worthwhile increase in cooling, above that achieved by single phase vapour heat transfer, might be obtained during the precursory cooling of reflood.

ACKNOWLEDGMENTS

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# NOMENCLATURE

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# REFERENCES


Role of
RELAP/SCDAPSIM in Nuclear Safety

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ABSTRACT
The RELAP/SCDAPSIM code, designed to predict the behaviour of reactor systems during normal and accident conditions, is being developed as part of the international SCDAP Development and Training Program (SDTP). SDTP consists of nearly 60 organizations in 28 countries supporting the development of technology, software, and training materials for the nuclear industry. The program members and licensed software users include universities, research organizations, regulatory organizations, vendors, and utilities located in Europe, Asia, Latin America, and the United States. Innovative Systems Software (ISS) is the administrator for the program. RELAP/SCDAPSIM is used by program members and licensed users to support a variety of activities. The paper provides a brief review of some of the more important activities including the analysis of research reactors and Nuclear Power Plants, design and analysis of experiments, and training.

1 INTRODUCTION
The RELAP/SCDAPSIM code, designed to predict the behaviour of reactor systems during normal and accident conditions, is being developed as part of the international SCDAP Development and Training Program (SDTP) [1,2]. SDTP consists of nearly 60 organizations in 28 countries supporting the development of technology, software, and training materials for the nuclear industry. The program members and licensed software users include universities, research organizations, regulatory organizations, vendors, and utilities located in Europe, Asia, Latin America, and the United States. Innovative Systems Software (ISS) is the administrator for the program.

Three main versions of RELAP/SCDAPSIM, as described in Section 2, are currently used by program members and licensed users to support a variety of activities. RELAP/SCDAPSIM/MOD3.2 and MOD3.4 are production versions of the code and are used by licensed and program members for critical applications such as research reactor and nuclear power plant applications. The most advanced production version, MOD3.4, is also used for general user training and for the design and analysis of severe accident related experiments such as those performed in the Phebus and Quench facilities. In turn, these experiments are used to improve the detailed fuel behaviour and other severe accident-related models in MOD3.4 and MOD4.0. MOD4.0 is currently available only to program members and is used primarily to develop advanced modelling options and to support graduate research programs and training.
2 RELAP/SCDAPSIM

RELAP/SCDAPSIM uses the publicly available RELAP/MOD3.3[3] and SCDAP/RELAP5/MOD3.2[4] models developed by the US Nuclear Regulatory Commission in combination with proprietary (a) advanced programming and numerical methods, (b) user options, and (c) models developed by ISS and other members of the SDTP. These enhancements allow the code to run faster and more reliably than the original US NRC codes. MOD3.4 and MOD4.0 can also run a much wider variety of transients including low pressure transients with the presence of non-condensable gases such as those occurring during mid-loop operations in LWRs, in pool type reactors, or in spent fuel storage facilities.

The most advanced version of the code, RELAP/SCDAPSIM/MOD4.0[5], is the first version of RELAP or SCDAP/RELAP5 completely rewritten to FORTRAN 90/95/2000 standards. This is a significant benefit for the program members that are using the code for the development of advanced models and user options such as the coupling of the code to other analysis packages. Coupled 3D reactor kinetics and coupled RELAP/SCDAPSIM-SAMPSON [6] calculations are examples where MOD4.0 is used because of a significant reduction in the code development effort and expense to link the packages. MOD4.0 also includes advanced numerical options such as improved time advancement algorithms, improved water property tables, and improved model coding. As a result the code can reliably run complex multi-dimensional problems faster than real time on inexpensive personal computers. Plant simulation and integrated uncertainty analysis are among the most important applications benefiting from the improved speed and reliability of MOD4.0. MOD4.0 includes many enhanced user options to improve the accuracy of the code or to offer new options for the users. For example, the addition of an alternative material property library designed for Zr-Nb cladding materials is important for VVER and CANDU reactor designs, particularly for severe accident related transients. The addition of an advanced water property formulation is important for many transients, in particular those involving super critical water applications.

3 REVIEW OF REPRESENTATIVE APPLICATIONS

RELAP/SCDAPSIM is being used for a variety of applications. As described in Section 3.1, the code is used for the analysis of research reactors and nuclear power plants. The research reactors analysed by the code include TRIGAs, MTR-plate designs, and well as other unique designs. Nuclear plants analysed include Western designed PWRs and BWRs, Russian designed VVERs and RMBKs, Canadian and Indian designed CANDUs and HWRs, and Chinese designed PWRs. The analysis of experiments, as discussed in Section 3.2, is also an important application of the code. This application includes the design of the experiments, the assessment and development of modelling improvements, and finally advanced user training. The application of the code to support the development of improved models and analytic capabilities is discussed in Section 3.3. Section 3.4 presents an overview of the application of the code for training.

3.1 Research Reactor and NPP Applications

A combination of RELAP/SCDAPSIM/MOD3.2 and MOD3.4 is being used to analyze research reactors. A brief summary of the early work by several countries was given in Reference [6]. The research reactors noted in this paper include (a) the LVR-15 reactor located at the Nuclear Research Institute in Rez, Czech Republic, (b) the CARR reactor being built in Beijing, China by the China Institute of Atomic Energy, and (c) TRIGA reactors located at the Atomic Energy Research Establishment in Dhaka, Bangladesh and National
Nuclear Energy Agency in Bandung Indonesia. LVR-15 is a light-water moderated and cooled pool type reactor with a nominal thermal power of 15 MW. The pool operates at atmospheric pressure with an average coolant temperature in the core of 320 K. The reactor also has closed high pressure/temperature loops suitable for testing of materials under PWR and BWR conditions. The reactor core is composed of several fuel assemblies of Al-U alloy arranged in square concentric tubes. CARR is a tank-in-pool design, cooled and moderated by light water and reflected by heavy water. The rated power is 60 MW. The core consists of plate-type fuel assemblies of Al-U alloy. The Indonesian and Bangladesh TRIGA reactors are pool type reactors with 2 MW and 3 MW thermal power respectively. The reactor cores are composed of solid U-ZrH fuel rods arranged in a hexagonal array and are cooled by water in either forced or natural circulation, depending upon the conditions.

More recently, the analysis of two additional reactor types have been reported in References 7-9. The first is for the SAFARI-1 research reactor located in South Africa [7,8]. The second is the University of Missouri Research Reactor located in the United States [9]. The SAFARI-1 research reactor is a tank-in-pool type reactor operated at a nominal core power of 20MW. The core is cooled and moderated by forced circulation of light water. The reactor core can be operated in a variety of configurations from 24 to 32 fuel assemblies. Figure 1 shows an example of one such configuration. The fuel is U-Si-Al plate-type fuel elements. MURR is a 10 MW pool type reactor design with a pressurized primary coolant loop to cool the fuel region. The pressurized primary system is located in a pool allowing direct heat transfer during normal operation and transition to natural convection under accident conditions. The reflector region, control blade region, and center test hole are cooled by pool water (natural convection).

![SAFARI Reactor Core Configuration](image)

Figure 1: SAFARI Reactor Core Configuration.

Because of the unique reactor designs, the RELAP/SCDAPSIM input models were developed separately by each organization and include a range of different nodalizations as presented in the reference papers. However in general terms, the RELAP/SCDAPSIM input models include all of the major components of each reactor system including the reactor tank,
the reactor core and associated structures, and the reactor cooling system including pumps, valves, and heat exchangers. The secondary sides of the heat exchanger(s) are also modelled where appropriate. These input models were qualified through comparison with reactor steady state data, with original vendor safety analysis calculations where available, and with experiments in a limited number of cases. Figures 2 through 4 give examples of the nodalization used for MURR. Figures 2 and 3 show the detailed hydrodynamic nodalization for the pressurized primary cooling system and the bulk pool and pool cooling system, respectively. Figure 4 shows the nodalization of the fuel plates. This input model is also somewhat unique in that all 24 fuel plates were modeled using RELAP5 heat structures.

Figure 2: MURR RELAP/SCDAPSIM Nodalization of the MURR Pressurized Primary Cooling System.

Figure 3: RELAP/SCDAPSIM Nodalization of the MURR Bulk Reactor Pool and Pool Coolant Loop.
Figure 4: RELAP/SCDAPSIM Nodalization of the MURR 24 Fuel Plate Core.

Figure 5 show the nodalization used for the SAFARI research reactor. Figure 5 shows the overall system hydrodynamic nodalization with the upper right corner of the figure showing the core nodalization. Note from the insert of the core nodalization diagram that the bypass or unheated channels were modelled separately from the heated fuel assembly channels. The core nodalization also included two separate hot plate channels located on each side of the hottest plate.

Figure 5: RELAP/SCDAPSIM Nodalization of the SAFARI Cooling System.

A wide variety of transients have been analyzed using the code. Examples are included in the references and include reactivity initiated power excursions and loss of flow or coolant...
transients. Figure 6 shows one such example for MURR. The figure shows the centerline temperatures for the 24 fuel plates during a cold leg LOCA, indicating that the fuel transients remained well below the assumed fuel damage limits of 900°F.

Figure 6: Example of RELAP/SCDAPSIM Calculated Fuel Plate Centerline Temperatures for the 24 Fuel Plates in MURR for a Cold Leg Large Break Transient.

All three versions of RELAP/SCDAPSIM have been used to analyze a variety of nuclear power plant designs. The applications have included RELAP5-only input models for normal operating or transient conditions where core damage is not expected as well as combined RELAP-SCDAP input models that included the possibility of transients with the loss of core geometry. A few representation examples are discussed in more detail in the remainder of this subsection.

Analysts at the Paul Scherrer Institut (PSI) in Switzerland have applied the code to the TMI-2 accident [10], an analysis of a LOCA during cooldown in the Beznau Westinghouse type 2-Loop PWR [11], and an analysis of a station blackout transient in the Gösgen Nuclear Plant [12]. The TMI-2 calculations included comparisons with the limited data available from the accident as well as comparisons with the MELCOR and SCDAP/RELAP5 codes. The Beznau analysis paper summarized the results of the analyses of postulated LOCAs in the Beznau (KKB) PWR, occurring during hot (HS) and intermediate (IS) shutdown with emphasis on large break LOCAs during hot shutdown. The large break LOCA during HS posed the greatest challenge to the plant safety systems. The analysis of the station blackout transient in the Gösgen Nuclear Plant focused on the impact of a potential failure of the depressurization system. In particular, the analysis focused on the timing of the heatup and failure of the RCS piping relative to the relocation of melt into the lower plenum and failure of the lower head. MELCOR, RELAP/SCDAPSIM, and SCDAP/RELAP5 were also used in both Beznau and Gösgen analyses.

The TMI-2 RELAP/SCDAPSIM and SCDAP/RELAP5 nodalization, as shown in Figure 7, used a 2 dimensional representation of the core region with a detailed SCDAP components being used to describe the behavior of the fuel rods and other core structures within each of the five representative flow channels in the core. The transition from the initial
intact core geometry to a damaged state is automatically handled by the SCDAP models including the initial failure of the control rods, liquefaction and relocation of the metallic U-O-Zr fuel rod material, formation and growth of a ceramic [U-Zr]-O₂ molten pool, and relocation of the molten ceramic into the lower plenum. Figure 8 shows one of the set of representative calculations presented in the paper. The figure shows the variation in predicted system pressure for a range of modeling parameters for the relocation of the metallic U-O-Zr fuel rod material for both RELAP-based codes. It was noted in the paper that both RELAP-based codes correctly calculated an in-core molten pool, of which two RELAP/SCDAPSIM cases predicted relocation to the lower head (via the bypass, as observed), while only one MELCOR case did so. It was further noted that the RELAP-based codes correctly calculated that lower head failure did not occur.

Figure 7: TMI-2 Nodalization Used for RELAP/SDAPSIM and SCDAP/RELAP5.

Figure 8: Example of RELAP/SDAPSIM and SCDAP/RELAP5 TMI-2 Calculated Results for System Pressure for a Range of Metallic Fuel Rod Material Relocation Modeling Parameters.
The Beznau RELAP/SCDAPSIM and SCDAP/RELAP5 nodalization, as shown in Figure 9, also used a 2-dimensional representation of the core region with detailed SCDAP components used to describe the behavior of the fuel rods and other core structures within each of the five representative flow channels in the core. Figure 10 shows one of the set of representative calculations presented in the paper. The figure shows the peak core temperatures calculated by MELCOR and the RELAP5-based codes for different assumptions regarding the activation of the Safety Injection pumps including the number of pumps and delays in the actuation of the pumps. The paper concludes that all three codes predict that in the limiting large break case the core is readily quenched without damage, by the nominal operation of the system injection system. However, it was noted that the more mechanistic RELAP-based calculations demonstrated that a larger margin existed (relative to that predicted by MELCOR) with recovery being possible even if only one pump operates after some delay.

The RELAP/SCDAPSIM and SCDAP/RELAP5 nodalization used for the Gösgen analysis also included a detailed core nodalization as described previously. However, the calculations also looked at the effect of hot leg nature circulation using a split hot leg model as shown in Figure 11. The split channel model allows the hotter vapor to move from the vessel to the steam generators along the top of the hot leg and cooler vapor to return along the bottom of the hot leg. The influence of the split hot leg input model relative to a single channel hot leg (which does not allow countercurrent flow of the vapor within the hot leg) is shown in Figure 12. As shown in the figure the split hot leg model predicted a more gradual heatup of the core but both single channel and split channel models still predict rupture of the surge line or hot leg piping before any molten core material relocates into the lower head.

Figure 9: Beznau Nodalization Used for RELAP/SDAPSIM and SCDAP/RELAP5.
Figure 10: Example of RELAP-based and MELCOR Beznau Calculated Results for Peak Core Temperature for a Range of Safety Injection Assumptions.

Figure 11: Gösgen Split Hot Leg Nodalization Used for RELAP/SDAPSIM and SCDAP/RELAP5.
The Politehnica University, Institute for Nuclear Research, and National Commission for Nuclear Activities Control in Romania have used RELAP/SCDAPSIM/MOD3.4 for a variety of analyses of CANDU reactor designs. Reference [13] presents the analysis of reactor inlet header break, looking at the size of the break, the choked flow model employed, the emergency core cooling (ECC) performance and the core nodalization. The results were compared with the original safety analysis results. Reference [14] presents the analysis of the influence of the header manifold modeling for an inlet header break in a CANDU 6. The paper looked at a 35% inlet header break which was expected to produce the highest peak fuel cladding temperatures among all postulated break sizes. Reference [15] presents the analysis of a reactor outlet header break in a CANDU-6. The paper focused on a 100% reactor outlet header break which had the highest potential for fuel failure and release of radioactivity. The paper also compared the results to earlier calculations performed using the CATHENA code [16]. Reference [17] presents an analysis of the late phase of a severe accident in CANDU 6 where bed of dry solid debris or a molten pool of core material had already formed at the bottom of the calandria vessel and was externally cooled by shield-tank water. The study used the SCDAP COUPLE module and included comparisons with earlier results performed using the ISAAC [18] and MAAP4-CANDU [19] codes.

The general system thermal hydraulic nodalization for the CANDU system thermal hydraulic analysis analyses [13-15] is shown in Figure 13. Portions of this nodalization were varied somewhat depending on the analyses being performed. Figure 14 shows an example of the portion of the nodalization that was used in [14] for the study of the influence of the inlet header model. Figure 15 shows an example of some of the results from the inlet header break model study. In the figure, the reference curve is the results from a single average channel circuit model using a single manifold volume and Cases 1, 2, and 3 represent the break location in the multiple inlet manifold volumes in combination with multiple core channels. The curves shown in the figures are the maximum cladding temperatures in fuel bundles contained within the multiple flow channels.

Figure 12: Example of RELAP/SCDAPSIM and RELAP/SCDAPSIM Calculated Results for Peak Core Temperature for Single and Split Hot Leg Model.
Figure 13: RELAP5 Thermal Hydraulic Nodalization for CANDU-6.

Figure 14: RELAP5 Thermal Hydraulic Nodalization of Manifold Headers for CANDU-6 Inlet Manifold Break Analysis.

Figure 16 shows the basic problem analyzed for late phase of a severe accident in a CANDU-6 where a debris bed is present in the bottom of the calandria vessel along with the RELAP5, SCDAP, and COUPLE nodalization used in the analysis. The RELAP5 thermal hydraulic volumes on the right of the figure represent the pool on the outside of the calandria vessel. The RELAP5 and SCDAP volumes above and within the COUPLE mesh provide initial and boundary conditions for the debris bed and calandria vessel wall. (The paper also included a more detailed RELAP5 nodalization of the outer pool at the elevations associated with the debris bed. The more detailed nodalization resulted in significantly lower pool containment pressures upon vessel failure due to the more accurate representation of the external cooling of the calandria vessel.)
Figure 15: Maximum Fuel Bundle Cladding Temperatures for a CANDU-6 Inlet Manifold Break Analysis for Different Break Locations.

Figure 16: RELAP, SCDAP, and COUPLE Module Nodalization and Conceptual Sketch for CANDU-6 Calandria Vessel Analysis.
The papers indicated that RELAP/SCDAPSIM calculations gave comparable results to the CANDU-specific codes, CATHENA for system thermal hydraulics, and ISAAC and MAAP-CANDU for severe accidents. For system thermal hydraulic analysis, RELAP/SCDAPSIM, when using similar input models, provided similar trends as compared to the original safety analysis reports or comparable CATHENA calculations. However, the results were sensitive to the level of detail used in the nodalization, specifically the more detailed nodalization possible using RELAP/SCDAPSIM had a noticeable impact on the results for the inlet header manifold [14], the fuel channels (simulating the effects of horizontally stratified flow in the channels [13], and the outer pool (exterior to the vessel calandria) [17].

RELAP/SCDAPSIM has also been used to analyze VVER reactor designs although the calculations to date have been proprietary and have not been published in the open literature. Figures 17-18 shows a non-proprietary, but representative, input model for a VVER-1000. This representative VVER-1000 input model and associated detailed input model engineering handbook was provided by Risk Engineering in Bulgaria [20] as an in-kind contribution for use in SDTP-sponsored VVER training activities. The nodalization of the basic components of the reactor (vessel and internals) is presented in Figure 17 and includes two channels in the core, the hot and peripheral channels. This basic vessel nodalization would be replaced by a more detailed multi-dimensional model comparable to that used for the PWR calculations described previously for general applications where core damage transients might be considered. Four separate main circulation loops with their corresponding main coolant pumps, cold and hot circulation pipelines are also included in the representative input model as shown in Figure 18.

Figure 17: Representative Vessel Input Nodalization for VVER-1000 for Transients Not Including the Possibility of Loss of Core Geometry.

Analysts at the Lithuanian Energy Institute (LEI) have published a number of papers describing their use of RELAP/SCDAPSIM/MOD3.2 for the analysis of RBMKs. Recent references are cited as [21-23]. LEI also provided non-proprietary, but representative, input models and associated engineering handbooks for use in SDTP-sponsored training activities. Figure 19 shows the nodalization diagram used for the representative RBMK RCS input model.
All three versions of RELAP/SCDAPSIM/MOD4.0 have been used to analyze BWRs although few results have been published in the open literature. Reference [24] describes some of the activities related to the use of the code for plant simulation and training for the Laguna Verde plants in Mexico. Figure 20 shows a representative input model, developed by the Comisión Nacional de Seguridad Nuclear y Salvaguardias (CNSNS), the Mexican regulatory authority, for the analysis of the Laguna Verde plants. This model is also used to support SDTP-sponsored BWR-specific training activities. See Section 3.4 and Reference 24 for more information on these activities in Mexico.
3.2 Experimental Analysis

RELAP/SCDAPSIM/MOD3.4 has been used by a number of organizations to help design experiments, to assess thermal hydraulic and severe accident models, and to support advanced user training. In recent years, the application of the code to experimental analyses have focused on European experimental programs including the German Quench experiments [25-29], French Phébus FPT experiments [30-33], and most recently Russian PARAMETER experiments [34].

The most detailed of the calculations have been involved in the design of new experiments. For example, as described in detail in [26], the design of new experiments requires the development of complex input models to describe the unique features of each experiment and in many cases, the development of specialized new models to treat features of the experiments not previously included in the code. In this example, the analysts from PSI and experimentalists from Forschungszentrum Karlsruhe (FzK) describe their use of RELAP/SCDAPSIM/MOD3.4 in conjunction with MELCOR and a special FzK-developed version of SCDAP RELAP5 [35] to design and analyze three different experiments in the quench facility, Quench-10, Quench 11, and Quench 12. Quench 10 was a unique experiment in that it was the first integral test to look at the influence of air ingestion on bundle heating and reflood. The design and analysis of this experiment required PSI to develop and incorporate special SCDAP models to treat the oxidation of Zircaloy in air/steam mixtures. The experimentalist also ran special small separate effects experiments to help develop the correlations that were then used in these new models. Quench 11 was a unique test for the Quench facility in that the test started with the bundle full of water and then the heat up transient was initiated by the boil-down of the water. (Previous Quench experiments used a mixture of steam and argon during the heat up of the bundle prior to reflood.) Although in this case, it was not necessary to modify any of the RELAP or SCDAP models, the modeling of the auxiliary heaters, added to the lower plenum of the experimental test train to provide...
realistic boildown rates, proved to difficult because of relatively large heat losses in the lower plenum region. QUENCH-12 was unique in that it was designed to determine the influence of a WWER bundle configuration and cladding on heat-up, oxidation, and quench response. Previous Quench experiments used PWR or BWR configurations and cladding materials. The Quench 12 bundle was significantly modified with changes to cladding material (Zr/1%Nb instead of Zry-4), electrical heating, and geometry. Oxidation correlations for Zr/1%Nb in steam were introduced into SCDAP to support the design and analysis of this experiment. Figure 21 shows a schematic of the Quench facility along with the RELAP/SCDAPSIM nodalization diagram.

Figure 21: Example of Experiment Design – Quench Facility Layout and Input Nodalization.

The analysis of the German Quench and French Phebus experiments have also played a pivotal role in the assessment of RELAP5/SCDAPSIM, the development of new improved models as discussed in Section 3.3, and in advanced user training as discussed in Section 3.4. References 28, 29, 31-33 are examples of the analysis of these experiments to assess the accuracy of the code and to identify areas where the models could be improved. References 36 and 37 describe the use of these experiments to support advanced user training.

3.3 Development of improved models and analytic capabilities

The development of improved models and analytic capabilities is also an important part of the overall SDTP cooperative activities. In addition to the modelling improvements driven by large scale experimental programs in the Phebus and Quench facilities as discussed in the previous section, other model and code development activities have been driven by the needs of SDTP members and licensed software users. INSS (Institute of Nuclear Safety System), Japan, one of the original members of SDTP, developed and validated new RELAP/SCDAPSIM models to treat the heat transfer in the gap between a debris bed and the
lower plenum wall [38] and improved correlations for condensation in the presence of non-condensable gases [39]. The application of the improved correlations are described in [40]. IAE/NUPEC (Institute of Applied Energy/Nuclear Power Engineering) Japan, a long time member of SDTP, has been working with the code to develop improved analytic capabilities to support the Japanese nuclear industry. The merger of RELAP/SCDAPSIM/MOD4.0 with the IMPACT/SAMPSON package [41] is one of the most significant projects. However, IAE/NUPEC has also been using the code for a variety of other tasks including the development of enhanced analytical capabilities to analyse corrosive conditions in nuclear power plants using coupled system thermal hydraulics and CFD techniques along with corrosion modelling [42]. CNSNS and ININ (Instituto Nacional de Investigaciones Nucleares) in Mexico have added integrated BWR containment models to RELAP/SCDAPSIM/MOD4.0 and are now working on the possible integration of detailed integrated sub-channel and containment modules developed by IAE/NUPEC [43]. Nuclear plant analyser graphic packages including VISA [44], developed by KAERI (Korean Atomic Energy Research Institute) and RELSIM, developed by RMA (Risk Management Associates) have been linked to RELAP/SCDAPSIM/MOD3.4 and MOD4.0. Other activities by members and licensed users include the coupling of the code with 3D reactor kinetics packages.

The development of improved models and code capabilities for RELAP/SCDAPSIM/MOD4.0 by university members of SDTP has also been an important factor in the improvement of RELAP/SCDAPSIM/MOD4.0 [5]. The rewriting of the code to Fortran 90/95/2000 version of the code has made it significantly easier for university faculty and students to work with. MOD4.0 also provides a well characterized framework for university researchers and students to explore new modeling approaches since the tedious programming details associated with use of complex fluid/material properties libraries, reactor component models such as pumps and valves, input/output, and data base management for tasks such as dynamic data allocation are provided through a standard compile library maintained by ISS. The incorporation of integrated fission product transport models by Honaiser, University of Florida, USA [29] and ongoing work to add an integrated uncertainty analysis package by Perez, University of Catalunya, Spain [45,46], and CANDU-specific models for fuel channel failure by Mladin, Politechnic University, Romania [17] are good examples where university students are key contributors to the development of the code.

### 3.4 Training of analysts and model/code developers

RELAP/SCDAPSIM/MOD3.4 and MOD4.0 are also widely used to support SDTP-sponsored training activities. MOD3.4 is used for basic user and applications training. This includes (a) 1 to 2 week novice and advanced RELAP5 and SCDAP user training workshops and seminars, (b) longer term, 1 to 3 month, user and application training under IAEA and SDTP-sponsored training fellowships, and (c) IAEA-sponsored specialized missions on research reactor applications, severe accident management and others. For example, novice users will use the code to set up basic thermal hydraulic problems such as the flow of water in a pipe or the boildown and quenching of a representative fuel assembly and then move on to the optimisation or expansion of the input model to a representative full research reactor or NPP. More advanced students or participants in longer term training sessions will typically use the code to develop input models for their own facilities or more typically adapt existing input models to run more reliably or run a much wider variety of possible transients. MOD4.0, and to some extent MOD3.4, are also widely used by the SDTP member universities to support their graduate and faculty research programs. Section 3.3 gave some specific examples of university students that started out participating in SDTP-sponsored training activities using MOD3.4 and MOD4.0 and then going on to make significant
contributions to improvement of MOD4.0. Another good example of the use of the codes at universities is provided by Professor Manmohan Pandey and others from the Department of Mechanical Engineering of Indian Institute of Technology Guwahati (IIT-Guwahati), India in a report submitted as an in-kind contribution for their university membership [47].

IIT-Guwahati used RELAP/SCDAPSIM/MOD4.0 and the Nuclear Plant Analyser RELAP/SCDAPSIM-VISA package (ViSA-RS) for numerical simulations of a natural circulation boiling water reactor (NCBWR) and supercritical water cooled reactor (SCWR). Figure 22 shows the example of the NCBWR schematic and nodalization. Their applications included the following areas
(a) Parametric studies of the primary heat transport loop of NCBWR,
(b) Stability analysis of NCBWR,
(c) Stability analysis of SCWR, and
(d) Educational use of RELAP5 and ViSA-RS.

Figure 22: Example of University Applications - Natural Circulation Boiling Water Reactor Applications by IIT-Guwahati.

4 SUMMARY AND CONCLUSIONS

RELAP/SCDAPSIM is widely used by the international community to support nuclear safety studies as well as support the training of a new generation of safety analysts. This paper gives a representative overview of the different applications of the code but, perhaps even more importantly, shows the critical role of the many experimentalists, model developers, analysts, teachers, and students that have contributed to the development of RELAP/SCDAPSIM and other codes like it.
ACKNOWLEDGMENTS

The contributions of the many SDTP members and licensed users are gratefully acknowledged.

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On The Analysis And Evaluation Of Direct Containment Heating With The Multidimensional Multiphase Flow Code Mc3d

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ABSTRACT

Direct Containment Heating (DCH) is a phenomenon that can potentially occur during a severe accident and may threaten the integrity of the containment by pressurization of its atmosphere. Several simplified modelling have been proposed in the past but we find that the models are highly dependent on the exact geometry with high variations of the main parameters. Thus the extrapolations to other reactors, other materials and other scales than those used for the fitting process of the parameters are very doubtful. With the development of multidimensional multiphase flow computer codes, it is now possible to investigate the phenomenon numerically with details. At the IRSN, we are using for this purpose the 3-dimensional code MC3D, mainly developed for Fuel Coolant Interaction (FCI) analysis. In this paper, we present an analysis of the potential of the MC3D code to support the analysis of the phenomenon. We describe and analyse the flow patterns computed with MC3D for the complicated case of geometry with a large pit and deduce a dispersion correlation. The particularly difficult problem of scaling is investigated with quite details. The thermal and chemical aspects are then discussed avoiding however the combustion phenomena itself. Indeed, no combustion model currently exists in MC3D. Finally, we demonstrate the potential of the code to go beyond the “classical DCH” problem with the addition of water either in the pit, either in the vessel itself.

1 INTRODUCTION

Direct Containment Heating (DCH) is a phenomenon that can potentially occur during a severe accident and may threaten the integrity of the containment by pressurization of its atmosphere. It can occur following a melting of the reactor fuel and its relocation at the bottom of the vessel. If the external cooling of the vessel is not efficient, it will fail and the melt will be ejected in the pit. If the (primary circuit) driving pressure is above some threshold, the melt will be ejected out of the pit going for a part into some small rooms (called hereafter sub-compartments) and for the rest directly to the containment dome. If the melt can be oxidized (which is likely), then a very rapid oxidation of a large amount of the melt occurs. The hydrogen produced during this stage will then burn altogether with the pre-existing one. Most of the past studies of this phenomenon where based on geometries of US NPP’s (Surry, Zion…). For a good synthesis of these works, the reader can consult the special issue of Nuclear Engineering and Design [1].
The DCH problem is mainly related to the evaluation of three physical phenomena:
- the melt ejection to the containment;
- the melt oxidation;
- the combustion due to the melt oxidation.

Regarding the melt dispersion, several simplified modelling have been proposed in the past, mainly qualified on US reactor types, and implemented in lumped parameter codes as CONTAIN [2]. However, we find that the models are highly dependent on the geometry with high variations of the main parameters [3]. Thus the extrapolations to other reactors, other materials and other scales than those used for the fitting process of the parameters are doubtful. With the development of multidimensional multiphase flow computer codes, it is now possible to investigate the phenomenon numerically with details. At the IRSN, we are using for this purpose the 3-dimensional code MC3D, mainly developed for Fuel Coolant Interaction (FCI) analysis [4][5].

The melt oxidation is also a complex problem, particularly when considering the specific conditions involved, i.e. high temperature liquid fuel and small time scale (less than 1 second in experiments). Currently, the approach for this problem is mainly heuristic and empirical, based on the experimental evidences of strong and very rapid oxidation. However, some questions regarding the actual characteristics of prototypical corium oxidation are still pending, as well as the question of scaling. This problem is currently under investigation at IRSN, also envisaging the potential oxidation of UO2, but the treatment in MC3D is still parametric.

Last but not least, is the problem of combustion. Experiments have shown that this phenomenon drives the essential of the "heating" problem. Then, it is likely that the heat transfer problem is secondary. In contrast, this means that the treatment of combustion needs a particular attention. For this issue also, the treatment has been mainly heuristic but large uncertainties exist due to important energy losses in experiments, particularly if one considers the problem of extrapolating and scaling. The MC3D software does not currently contain any combustion model and this topic won't then be addressed in the following of the paper.

The present analysis is based on an experimental program investigating the phenomenon with different geometries representative of European reactors such as the EPR geometry (DISCO, performed by FZK, Germany,[6]), the French 1300 MWe P'4 (DISCO-F), VVER and Konvoi German reactors. A general presentation of the program and the main outcomes can be found in reference [7]. We will focus our present study on the 1300 MWe P'4 geometry. In the frame of French-German collaboration between IRSN and FZK, the DISCO facility has been modified to represent the P'4 geometry in a scale 1:16th (Figure 1). The geometry is complex, 3-dimensional, with an important path section towards the bottom access (path 3), and a distribution of fuel ejected upwards towards two different spaces: containment dome (6) and sub-compartments (7). Some of the tests (denoted "2D") were carried out with a simplified geometry supposing a cylindrical cavity and without the access path. The purpose of this simplification is to allow an easier comparison between experimental results and calculations. The program comprised:
- 10 water-tests called DISCO-F(X), including 2 in "2D"
- 4 cold tests with a heavy metal, made from a gallium alloy, called DISCO-FM(X), of which one is in "2D"
- 4 cold tests with a second heavy metal, the Wood's Metal, called DISCO-FW(X), of which one is in "2D" (two other "2D" tests in preparation will close the program)
- 5 hot tests with a thermite alumina-iron, called DISCO-FH(X) with
  - 1 in neutral environment, avoiding chemical effects
- 1 in "2D" geometry.

The pressure differences between the vessel and the pit were varied from 10 to 22 bars approximately. Three diameters of breaches were used: 30, 45 and 60 mm, corresponding to 0.5, 0.75 and 1 m breach diameters at full scale. Note that only tests with a central breach were performed as previous experiments showed that this situation was bounding.

![Figure 1](image)

**Figure 1**: Example of 3-D test calculation geometry with a fine mesh, showing the main volumes and flow paths, for the DISCO-F geometry. 1 reactor pressure vessel, 2 reactor cavity, 3 reactor pit access, 4 pit niche, 5 annular space, 6 exit to containment, 7 exit to sub-compartments

Our purpose in the present paper is 2-fold:
- demonstrate the potential application of CFMD tools for this kind of problem
- give the status of improvement of the study conducted by IRSN with the Multiphase Flow code MC3D.

This will be done by recalling first the main features of the code and its applicability to the current problem. Varying the mesh precision allows us to make several kinds of studies. With an intermediate meshsize, we will discuss the flow patterns calculated with the code. With a somewhat rough mesh, one can make a lot of calculations and thus having a statistical treatment to build correlations for application in simplified models for, e.g., system codes or PSA studies. In contrast, very fine meshes can allow analysis of particularly complex features of the geometry as the upper part of the annular space where occurs a separation of the melt between the sub-compartments and the containment. This is still under investigation and will not be discussed further in this paper.

The qualification of the tool for the specific DCH problem allows us to go far beyond what can be obtained with analytical models with a more comfortable confidence regarding extensions to, e.g. different scales or different initial conditions as including the effect of the presence of water.

2 **MC3D : ANALYSIS OF THE POTENTIAL OF THE CODE TO DESCRIBE THE MAIN FEATURES OF THE PHENOMENON.**

MC3D is a software devoted to study multiphase and multi-constituent flows in the field of nuclear safety. It has one of the reference tools for the evaluation of Fuel Coolant Interaction (FCI) but it is more generally a CMFD tool that can treat various problems [5].
One of the most important particularities of MC3D is that the fuel is described by two fields (optionally three for FCI application). These two fields describe the two possible states of the fuel: either continuous, either discontinuous (Figure 2). These two fields, connected by mass transfers constitutive laws (fragmentation or coalescence). This way, the drop field is supposed to describe only discontinuous fuel state. The continuous fuel field is modelled with a specific VOF-PLIC method, avoiding then numerical dispersion.

![Figure 2: Connections between the continuous and discontinuous fuel fields. Plain arrows: fragmentation; dashed arrows: coalescence towards continuous fuel.](image)

For the DCH evaluations, the fragmentation process is obtained through a model extending the Kelvin-Helmholtz instability model to multiphase fragmentation, taking into account local velocities. Coalescence occurs either from drop collision with the continuous fuel, either from collision between drops. In both cases, the collision models are simple and mostly based on geometrical arguments and fuel temperature.

In the last version, the code can use as many non condensable gases as necessary. Also, an oxidation model is present. Due to the specific conditions that MC3D has generally to face (high temperature melts), the model is however quite simple and parametric. Recent theoretical investigations at IRSN will allow an improvement with more physical constitutive laws in future versions. Currently, we simply set an amount of potential oxidation per surface area of the melt (called capacity), calculated with a transport equation. The oxidation itself occurs at a rate depending on the capacity, a time scale and the vapour content.

The major key limitation is then the evaluation of combustion which not possible currently.

One of the advantages of the MC3D tool is its recognized ability to treat the interaction between the fuel and the water, either for slow mixing, either for steam explosion. It is then possible to combine with the same tool the effect of DCH and FCI. This will be exemplified in the last section of this paper.

3 FLOW PATTERNS ANALYZED WITH MC3D

3.1 Current limitations of scope

We restrict the discussion to 2-D geometries although some 3-D situations have been studied with the code. 3-D aspects come into play regarding:

- The bottom access: this can be analysed with the code but costs in terms of CPU time.
- The separation of melt between the sub-compartments and the containment dome. For heavy melts, a very fine mesh with precise description is required and this problem is still under investigation.
The last point gives a serious limitation in the analysis of the atmosphere heating. However, as the combustion cannot be studied, we restrict then our attention on the phenomena occurring in the pit.

3.2 Pure gas flow

Investigating the pure gas flow has a limited practical interested. However, this is a necessary step particularly if one intends to build analytical simplified models as for System Codes. Indeed, the gas is the carrier fluid and it is important to characterize its flow. It is interesting to notice the existence of an intense convection roll occupying the whole pit (Figure 3) and then definition of a characteristic velocity in the pit, where occurs the entrainment (particularly at full scale) is not an easy task. Along the wall, the velocity shows a very strong gradient. On the mean, we find that the upcoming velocity is in the same order as the one at the annular space.

Figure 3: Example of pure gas flow computed with MC3D at three different times with DISCO P’4 ”2D” geometry. (background colour according to pressure).

3.3 Melt ejection and upward dispersion

The qualification of MC3D regarding the melt and gas ejection out of the vessel have already been discussed in previous analyses [7][8]. We propose now visualization for the purpose of illustration and completeness of analysis in Figure 4. At first, the melt flows as an undisturbed jet. Due to the high ejection velocity, the jet fragmentation is limited: due to the limited distance between the vessel and the floor in the experiment, the travelling time scale is smaller than the fragmentation time scale. This in fact is not true in the full scale: the ejection velocity and then the fragmentation time scale are still the same but the travelling time is increased accordingly to the scale. However, we find that this modification of the flow pattern does not have an important influence on the overall process of dispersion out of the pit.

At the impact with the floor, the code does not (cannot) represent an eventual splashing effect as it was observed in some experiments. However, this splashing does not seem to have a significant effect. More, the effect should again diminish at full scale. Rapidly, a hole is formed inside the melt pool (third picture) and the gas begins to flow out of the vessel with a high velocity. As the gas flows inside the melt, this one is automatically totally fragmented and dispersed in the pit. This stage is seen in the integral experiments through an important
increase of the cavity pressure. However the drops rapidly hit the floor or the wall and undergo a recoalescence.

Before this 2-phase flow stage starts, the melt can be accumulated along the wall simply by the effect of inertia. Depending on the vessel pressure, it can either flow back and return on the floor, either be entrained and flow along the wall.

The gas velocity becomes really important only once most of the melt is ejected. During the stage of two-phase flow, the gas is strongly slowed by the friction with the melt. We find that, in MC3D calculations, the melt along the wall is really entrained and fragmented only once this essentially single-phase gas flow starts. This has been confirmed with inspection of pictures taken during the cold experimental tests. In MC3D, it is found that the fragmentation mostly occurs as the continuous melt approaches the annular space.

It is also seen from Figure 4 that gas flow pattern seen with pure gas flow in Figure 3 are strongly disturbed. We often see the development of two circular spherical rolls on turning a an opposite sense regarding the other (best seen on Figure 5 and Figure 6).

![Figure 4](image)

**Figure 4**: Visualization of melt ejection fragmentation and upward dispersion in a typical simulation of DISCO integral test ($P_v = 19$ bar, $D_b = 30$ mm, Al$_2$O$_3$-Fe thermite at 2500 K, vessel gas : vapour). Legend: top: continuous fuel field volume fraction, middle: fuel drop volume fraction, bottom: drop velocity (m/s).

### 3.4 Gas flow characteristics during melt ejection and dispersion steps

Some characteristics of the gas flow can be visualized on Figure 5 and Figure 6. For a characteristic calculation with hot reacting material, both the gas temperature (Figure 5) and the vapour partial pressure (Figure 6) and strongly affected by the fragmentation stage at the breach. In this calculation, we set the time scale for melt oxidation at the relatively small value of 10 ms. The real characteristics of thermite oxidation kinetic has not been yet investigated. It is however likely that the phenomenon is very rapid due to the high temperature. As soon as the 2-phase flow appears at the breach, the gas is strongly heated. At the experimental scale, the pit is entirely heated at a temperature close to the melt one in a few
milliseconds. Meanwhile (but independently in fact) the vapour pressure has mostly disappeared in the pit (Figure 6). The minimum residual value of the vapour partial pressure is due to the limitation of oxidation set close to 1000 Pa. When the flow becomes essentially a single-phase gas flow, then the pit gas starts cooling down and fresh vapour can re-enter in the pit. As the dispersion process occurs mostly during this single-phase gas flow, it is seen that dispersion occurs with very transient conditions regarding the gas properties.

Figure 5: Visualization of gas temperature during the phase of melt ejection (same calculation as in Figure 4). Legend: gas temperature (K), see also Figure 4 for continuous melt volume fraction and drop velocity.

Figure 6: Visualization of vapour partial pressure during the phase of melt ejection (same calculation as in Figure 4). Legend: vapour partial pressure (Pa), see also Figure 4 for continuous melt volume fraction and drop velocity.
4 DISPERSION CORRELATION

Finding correlations that would work for any case, any geometry has been the subject of work of many researchers in the past (see the important number of correlations in the CONTAIN code). Our purpose is to concentrate on a particular geometry and correlate parameters regarding the initial physical conditions (gas, melt) and the breach section. The limited number of experiments renders this task very difficult. One of the obvious interests of CFD tools is to allow a substitution of the experiments with the calculation. The experiments at our disposal are mainly used to qualify the codes in some particular simulating situations. We use the CMFD code MC3D for this purpose. Then we propose a mixed method where we will seek for a correlation with parameters that need to be adjusted thanks to CMFD calculations and validated with experimental results.

At first, we have to admit that it is necessary to treat the dispersion globally, i.e. without taking into account the particular path to the containment and sub-compartments. The particular problem of separation of melt can be investigated only with a separate method. Also, the bottom exit path cannot be treated easily unless 3-D calculations are performed. Previous estimations tend to show that the 3D complete geometry behaves similarly to the 2D case [8]: the inceptions of entrainment were found very close and mainly the maximum fraction is changed. Then it is likely that we can use the 2D geometry to build a general correlation that will be calibrated for the 3D case with experimental results and some additional particular 3D calculations. It is also rather clear that we cannot have a real precision regarding the maximum possible pressurization. In the code, this depends mostly on the meshing and numerical methods. In the experiment and real reactor case, it depends on local complex geometries particularities that cannot be reproduced.

The first idea that comes in mind to correlate the upward dispersion is to relate the later with the ability of the gas to entrain melt droplets. The relative terminal velocity of a monodispersed droplet population in an ambient fluid can approximately be given by:

$$V \approx V_0 (1-\alpha_d) = (1-\alpha_d) \left[ \frac{8 R_d}{3 C_{d0}} \frac{\rho_d - \rho_a}{\rho} \right] g$$

where the subscript 0 stands for one single droplet.

On the other hand, the relative flow tends to fragment the drops up to equilibrium. If one single drop is stable for a relative Weber number of, say, 12, it is found experimentally [9] that the mean Weber number of drops subjected to fragmentation is lower. More precisely, the drops are not created such that the Weber is equal to the critical one but can be largely smaller. We will call this number the equilibrium Weber number and take it as an uncertain parameter that should be range from 1 to 6. The equilibrium Weber number gives us a second relation between the velocity and the drop mean diameter:

$$We_{eq} = \frac{2 \rho_d \Delta v^2 R_d}{\sigma}$$

Considering only dispersed flow with low drop volume fractions and drop densities far higher than the ambient one, one have an expression of the relative velocity as:

$$\Delta v^2 \approx \left[ \frac{4 \sigma We_{eq} \rho_d}{3 C_{d0} \rho_a^2} \right] g$$

This gives a lower limit for the gas velocity that can lead to melt entrainment. This in fact defines a dimensionless number called the Kutadeladse number:

$$K = \frac{\rho_d v^2}{\sqrt{\sigma \rho_d g}}$$
and a criteria for entrainment can be expressed as:
\[ K \geq K_c \equiv \frac{4 \text{We}_{eq}}{3 C_{d0}} \]

If the drop diameter is in the inertial range, \( C_{d0} \) is about 0.5 and we obtain a critical Kutadeladse of about 3.

The Kutadeladse number gives a criterion for inception of entrainment but our purpose is to find a correlation giving the upward dispersion fraction. It is however likely that we can use this number for this purpose. The dispersion is a function of the mass flow rate and thus we will seek for a relation using the square root of the Kutadeladse number, i.e.:

\[ (2) \quad F_d = f(\sqrt{K}) \]

The problem now is to have a characteristic gas velocity in our system. The fragmentation and entrainment occurs most likely in the reactor pit where no obvious characteristic velocity can be defined. However, it is likely that the velocity along the wall can be related to the velocity in the annular path. Then, for the purpose of a correlation, one can use the annular path velocity as a characteristic velocity to compute the Kutadeladse number.

For the case of pure gas flow, all evaluations show that the pressure inside the pit comes very rapidly to equilibrium and the flow is rapidly in a quasi-steady state. In that case, the mass flow at the entrance and outlet of the pit are balanced. At the vessel breach, the flow is, in all cases of interest, sonic. The classical theory of compressible gas flow gives us the mass flow rate as:

\[ Q_b = 0.56 A_b P_v \sqrt{\frac{\gamma M_m}{RT_v}} \]

Then, the velocity in the annular space is:
\[ v_a = \frac{Q_b}{A_b \rho_a} = 0.56 \frac{A_b}{A_a \rho_a} P_v \sqrt{\frac{\gamma M_m}{RT_v}} \]

The correlation will then take the form:
\[ (3) \quad F_d = f \left( 0.56 \frac{A_b}{A_a} P_v \sqrt{\frac{\gamma M_m}{RT_v}} (\sigma \rho_d g)^{1/4} \frac{1}{\rho_a} \right) \]

In this relation, we identify two difficulties. The first difficulty comes from the influence of the breach diameter. A small breach will increase the duration of the gas flow, having then an influence on dispersion opposite to the one in (3) (i.e. inversely proportional). Also, for a small breach, the two-phase flow stage is longer and there is an important pressure loss during this stage. Then the influence is not so simple. On the overall, the influence should not be linear but rather proportional to \( A_b^n \) with a small exponent n. Then we are seeking for correlation of the form:

\[ (3) \quad F_d = f \left( 0.56 \left( \frac{A_b}{A_a} \right)^n P_v \sqrt{\frac{\gamma M_m}{RT_v}} (\sigma \rho_d g)^{1/4} \frac{1}{\rho_a} \right) \]

The second difficulty comes from the influence of gas density in the pit. This one is not known a priori and we have seen previously that it should change drastically during the transient.

In order to be able to have a statistical treatment, we use the rough mesh shown in, the pit being represented by a 10x10 grid. Such rough mesh has proven to be sufficient to accurately predict the upward dispersion for all tested geometries [8]. All 2-D calculations...
presented hereafter are done with an equilibrium Weber number of 1. The parameters that have been checked to build the correlation are listed in table 1 with the range of variation.

<table>
<thead>
<tr>
<th>parameters</th>
<th>min value</th>
<th>max value</th>
</tr>
</thead>
<tbody>
<tr>
<td>( P_v ) : Vessel pressure (bar)</td>
<td>5</td>
<td>35</td>
</tr>
<tr>
<td>( T_v ) : Temperature of gas in vessel (K)</td>
<td>380</td>
<td>1000</td>
</tr>
<tr>
<td>( T_m ) : Temperature of fuel in vessel (K)</td>
<td>(cold) 350</td>
<td>(hot) 2500</td>
</tr>
<tr>
<td>( D_b ) : Breach diameter (mm) ( \Rightarrow ) defines area ( A_b )</td>
<td>30</td>
<td>90</td>
</tr>
<tr>
<td>( \rho_d ) : Fuel density (kg/m(^3))</td>
<td>1000</td>
<td>9230</td>
</tr>
<tr>
<td>( M_m ) : Vessel gas molar mass (g/mol)</td>
<td>(helium) 4</td>
<td>(air) 29</td>
</tr>
<tr>
<td>( P_0 ) : Containment pressure (bar)</td>
<td>1</td>
<td>3</td>
</tr>
<tr>
<td>( \sigma ) : Surface tension (N.m(^{-1}))</td>
<td>(water) 0.07</td>
<td>(thermite) 1</td>
</tr>
<tr>
<td>( T_0 ) : Containment temperature (K)</td>
<td>315</td>
<td>1070</td>
</tr>
</tbody>
</table>

Figure 7 shows the influence of breach area (scaled with the annular path area). We find that the best fit is obtained with an exponent equal to \( \frac{1}{4} \) instead of 1, probably due to the longer ejection time for small breaches, as expected. The scatter of the data is, as already explained, mostly due to the code and the very rough mesh used. This is a characteristic due to the continuous fuel field numerical treatment (VOF-PLIC), which behaves only approximately with rough meshes. However, in most applications, it is not necessary to have a precise result for a particular situation, but rather a global tendency.

On the overall, it is then found that the breach diameter is not a very critical parameter, thus limiting the impact of the important uncertainty regarding the breach characteristic during a severe accident.

\[
\left( \frac{A_b}{A_a} \right)^{1/4} \cdot \frac{P_{case}}{\sqrt{T_{raw}}}
\]

Figure 7 : Impact of vessel gas temperature and breach diameter on the calculated upward dispersed fraction (all other parameters identical)

The impact of the other parameters not explicitly appearing in (3) (but eventually through the gas density) is found quite mild. We do not see a significant impact of the initial
containment pressure (if limited to 3 bars). In contrast, the melt temperature seems to increase noticeably the dispersion. However, in the hot calculations, the combustion effects are not included and it is likely that the increase in containment pressure would lead to a decrease in dispersion. Due to the various uncertainties, it did not seem relevant to include additional parameters in (3) and the gas density was simply changed to a reference density $\rho_0$ (1 kg/m$^3$).

The overall result of all the 2-D calculations is plotted in Figure 8 with the available experimental data and the best fit correlation of the form:

$$F_d = \frac{F_{d,\text{max}}}{2} \left[ 1 + \tanh \left( A^* \log \left( \frac{\sqrt{K^*}}{\sqrt{K_{50}^*}} \right) \right) \right]$$

(4)\[ F_d = \frac{A^*}{2} \log \left( \frac{\sqrt{K^*}}{\sqrt{K_{50}^*}} \right) \]

$$\sqrt{K^*} = \left( \frac{A_h}{A_a} \right)^{1/4} \frac{P_{\text{cove}}}{\rho_0} \left( \frac{\gamma M_a}{R} \right)^{1/4} \left( \frac{g \sigma}{\sqrt{\gamma T_{\text{cove}}}} \right)^{1/4} \frac{1}{\sqrt{\rho_0}}$$

(5)\[ \sqrt{K^*} = \left( \frac{A_h}{A_a} \right)^{1/4} \frac{P_{\text{cove}}}{\rho_0} \left( \frac{\gamma M_a}{R} \right)^{1/4} \left( \frac{g \sigma}{\sqrt{\gamma T_{\text{cove}}}} \right)^{1/4} \frac{1}{\sqrt{\rho_0}} \]

$$\rho_0 = \text{reference density} = 1 \text{ kg/m}^3.$$

We find, for the DISCO P’4 experimental facility: $\sqrt{K_{50}^*} = 170$, $A = 5., F_{d,\text{max}} = 0.9$.

The correlation can now be compared to the experimental 3-D data, just changing the maximum dispersion in (4). This is done in Figure 9. We also included calculation points obtained with preliminary calculations performed with an equilibrium Weber number of 12, instead of 1 for all other calculations. An equilibrium Weber number of 1 would imply a small but noticeable shift of these points towards the left, so that the agreement would be better. Note that there are in fact 3 thermite points but two of them are overlapping (breach diameter of 42 and 60 mm and initial pressure of 2.39 and 1.98 MPa).

Figure 8: All calculations versus proposed correlation for 2-D case, and available 2-D experimental data.

There is a somewhat important scatter of the data around the correlation. However, considering the very important variations of the physical conditions between a non-reacting water flow and an integral test with chemical reactions, it can be estimated that the scatter is largely in the uncertainty range of actual conditions during a severe accident and probably of some of the experimental conditions. For the use in a particular problem, it is envisaged to fit more particularly on both the calculations and experimental results (if any) with the particular conditions of consideration. For a PSA level 2, the fit should then be calibrated using corium and thermite physical properties ranges, vapour as a driving gas, and high melt temperature.
Figure 9: Comparison of the correlation (4) with all 3-D available experimental data, same parameters as 2-D correlation except maximum dispersion. The calculation points regards preliminary calculations performed with an equilibrium Weber number of 12, instead of 1 for all other calculations.

However, one can already fix ideas of what can be expected for a representative DCH in a severe accident with however the scale of DISCO tests. This is presented in Figure 10 (see legend for conditions). The inception of entrainment is expected to occur approximately at 20 bar. A full dispersion is obtained at approximately 40 bars with a large breach (1 m at full scale) and 60 bars with an intermediate one.

Figure 10: Application of correlation (4) to a case with corium as fuel, and vapor at 550 K in vessel for two breach diameters, at the scale of DISCO tests (1/16th).

5 SCALING

The scaling of the problem is of course an important issue. A large uncertainty can be attached to this problem if one relies only on experimental results and simplified models. The use of CMFD codes as MC3D allows investigating this issue with an increased confidence, although scaling, i.e. the mesh size, can have an important effect on some particular constitutive laws.
Regarding dispersion, the calculations did not show a noticeable effect regarding the global results of melt dispersion. Some of the flow patterns are different particularly regarding melt ejection. As an example, at full scale, the nearly intact jet issuing the vessel is not visible at full scale. In most cases, the aerodynamic processes of fragmentation have sufficient time to break the jet before hitting the floor. For large breach, the 2-phase flow also occurs before the melt-bottom contact. Then on the overall, the initial melt fragmentation is more intense. However, due to the recoalescence along wall and floor, there is no really clear effect. Figure 11 shows the influence of the scale for a calculation based on the integral DISCO experiment FH05, without, however, taking account of chemical aspects. The time has been arbitrarily scaled as the distances. The maximum dispersion is subject, as already seen, to some scattering (the experimental dispersion fraction is 0.54). With the choice done of time scaling, the manly see on this particular example that the dispersion occurs sooner. This is due to the relative sooner occurrence of 2-phase and single-phase flows. Figure 12 gives an more global example of comparison of scaling effect. The dispersion is here plotted as a function of the modified Kutadladse number (see previous section). What has been found from the various trials of calculation at full scale is that the inception of dispersion is not modified. There is however some flattening of the dispersion curves and the full dispersion is predicted to occur for a slightly higher pressure compared to the case at experimental scale.

Figure 11: Evolution of the dispersed fraction with different scale. Calculation parameters based on experiment FH05 (P_v = 19 bar, D_b = 30 mm, thermite at 2500 K) without chemical aspects (oxidation, combustion). time scaled as the distances.
6  FURTHER POTENTIAL CAPABILITIES OF MC3D.

The "academic" problem of DCH does not take into account the presence of water. However, the probability for presence of water either in the vessel, either in the pit is very important for various reasons. MC3D seems a particularly well suited code for taking into account the presence of water. The case of water in the pit is quite obvious, as MC3D is already used for the evaluation of Fuel Coolant Interaction in the so-called ex-vessel situation. This is nothing else, for the code, like a DCH with water in the pit. In the present case, we are interested with a presence of a small amount of water, which was not the case in FCI studies, so no results can be currently given. The case of water in the vessel, above the melt, has been investigated recently for demonstration purpose and is illustrated in Figure 13.

Figure 13 : Example of calculation made with a situation where an important mass of saturated water is present in the vessel at the time of failure. $P_v = 7$ bars.
It is very likely that the water will be close to saturation. There, at the exit from the vessel, a flashing phenomenon will occur, where some part of the water will evaporated quasi-instantaneously. In the example given, the overpressure is only 5 bars, so far below the inception of entrainment without water. Another test with a smaller mass of water and 8 bar as overpressure gives a full dispersion. We find that the flashing increases strongly the pressure in the pit and then the dispersion. In case of large masses of water, there is a large remaining part of water which is expelled together with the melt. This expelled water might inert the combustion. In the case of small mass of water, this one is completely evaporated but produced a huge effect on the dispersion. The water acts in fact as a supplementary vessel pressurisation. In that case, an inverting effect of this vapour is quite unlikely.

7 CONCLUSION AND PERSPECTIVES FOR FURTHER DEVELOPMENTS

This paper highlighted the interest of the use qualified multiphase flow codes as MC3D for the evaluation of the DCH phenomenon during the course of a hypothetical severe accident. Due to the complex geometries of NPP's, this problem has for a long time been faced to the technical difficulties in representing adequately the phenomenon with simplified analytical models. Correlations were largely inspired by experimental results with only restrained applications and weak confidence in extrapolations to real material, real scale.

With the use of MC3D and relatively rough meshes, we could present a method of developing a dispersion correlation. For each geometry (EPR, large pit reactors), the calculations can used to calibrate precisely the correlation, as it has been done for the case of French P'4 reactors.

For a full evaluation of the phenomenon, one needs a model of combustion. However, even codes dedicated to combustion can present some difficulties to evaluate the phenomenon. Then despite the progress made and presented here, we are still faced to a complex problem. In the case of MC3D, it is then envisaged to seek only for a simple model, dedicated to the problem.

Despite this difficulty we have elaborated some patterns of explanation for the dispersion of the melt. We find that 2-D and 3-D situations might not be so different. We are also able to specify with a good confidence the range of physical conditions for which the dispersion can occur and then propose a range of conditions for which the phenomenon cannot threaten the containment. We can also integrate some supplementary conditions that are usually not taken into account as the presence of water.

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REFERENCES


CATHARE Multi-1D Modeling of Coolant Mixing in VVER-1000 for RIA Analysis

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ABSTRACT
The paper presents validation results for multi-channel vessel thermal-hydraulic models in CATHARE used in coupled 3D neutronic/thermal-hydraulic calculations. The mixing is modeled with cross flows governed by local pressure drops. The test cases are from the OECD VVER-1000 coolant transient benchmark (V1000CT) and include asymmetric vessel flow transients and main steam line break (MSLB) transients. Plant data from flow mixing experiments are available for comparison. Sufficient mesh refinement with up to 24 sectors in the vessel is considered for acceptable resolution. The results demonstrate the applicability of such validated thermal-hydraulic models to MSLB scenarios involving thermal mixing, azimuthal flow rotation and primary pump trip. An acceptable trade-off between accuracy and computational efficiency can be obtained.

1 INTRODUCTION
This work is motivated by the need for improved single-phase vessel mixing models in system codes that are able to properly represent local effects in reactivity insertion accidents. The study has been performed in Phase 2 of the OECD VVER-1000 coolant transient benchmarks labelled V1000CT-2 [1], [2]. These benchmarks provide a consistent approach to the testing of coupled neutronic/thermal-hydraulic codes. Separate exercises are devoted to stand-alone testing of thermal-hydraulic and core physics models. Then the validated models are tested in coupled code simulation of asymmetric MSLB transients.

The V1000CT-2 vessel mixing benchmark [1] is based on a steam generator isolation experiment during the plant commissioning phase of Kozloduy-6 in Bulgaria. Local and integral plant data are available for comparison. The purpose of this benchmark is to test the capability of system and CFD codes to represent in-vessel thermal hydraulics. The purpose of the V1000CT-2 MSLB benchmark is to test the core neutronics and coupled N/TH calculations both for separate reactor vessel with boundary conditions and for full plant simulation. The paper presents CATHARE thermal-hydraulic calculations for the VVER-1000 coolant mixing and MSLB benchmarks.

1 Corresponding author
2 THE VVER-1000 REACTOR

The reference plant is Kozloduy-6 with a VVER-1000 V320 in Bulgaria. This is a four loop pressurized water reactor with horizontal steam generators. The steam is supplied to a 1000MWe turbine. The core is of open type and contains 163 hexagonal fuel assemblies, each of which has 312 fuel pins, 18 guide channels for control rods and a central instrumentation tube. The fuel pins are arranged in triangular grid. The geometry of the reactor pressure vessel (RPV) is shown on Figure 1. See ref. [1] for details.

![Figure 1 VVER-1000 reactor pressure vessel and internals](image)

The flow in the lower part of the vessel is throttled through the perforated barrel bottom (1344 holes) and the perforated fuel support columns serving as flow distributors. The support columns are welded at the core support plate so that no flow passes around the support columns. The primary coolant flows through the slots, upward through the support columns and into the fuel assemblies. The flow through the core region consists of flow through the heated part and 2.9% bypass flow of which 2.2% is through the control rod guide channels and water holes.
At the upper core plate most of the fuel assembly heads are connected to guide tubes located in the upper plenum to protect the control rods and instrumentation cables from mechanical impacts. The bulk flow to the upper plenum passes mainly through the holes of the upper plate around the guide tubes. The bypass flow through the guide tubes is about 1% of the total because of the small flow area of the available orifices (see ref. [1]). The outlets of ninety five assemblies are equipped with thermocouples located eccentrically in the upper part of the assembly head. Because of construction peculiarities there is a quasi-stagnation flow at the location of the thermocouples. The cooler jets through the control rod guide channels cause the measured temperature to be somewhat lower than the real coolant temperature at the end of the heated part. This should be taken into account when comparing with computations, or estimated core inlet temperatures can be used for comparison.

The flow in the upper plenum passes upwards, then through the perforated walls of the support ring and the core barrel to the vessel outlet.

3 TEST CASES

3.1 Calculation of the Kozloduy-6 vessel mixing experiment

The Kozloduy-6 SG isolation problem at 9.36% nominal power and the corresponding vessel mixing was chosen as reference problem of the OECD V1000CT-2 Benchmark, Exercise 1 [1] The benchmark provides a validation test of the vessel thermal hydraulics in case of loop temperature and flow disturbances with all MCP in operation. It is relevant to the initial part of VVER-1000 MSLB scenarios from hot full power. The test problem is considered as pure thermal hydraulic problem because the moderator temperature reactivity coefficient was close to zero and the relative power distribution was approximately constant during the transient.

The mixing experiment was initiated by disturbing loop #1. It includes three states: a stabilized initial state, a transient state and a stabilized final state. These states are briefly described below. After the stabilization of the core outlet temperature and the pressure, the experiment was repeated for loop n#2. The transient caused by disturbing loop #1 is selected for the coolant mixing analysis and the data of the second experiment is used indirectly to support the analysis.

**Initial State:** The reactor is at the beginning of Cycle 1 and the power is 9.36% of the nominal. All four MCP and four SG are in operation. The pressure above the core is 15.59MPa, close to the nominal value of 15.7MPa. The coolant temperature at the reactor inlet is 268.6°C and the boron acid concentration is 7.2 g/kg (the coolant temperature reactivity coefficient is zero near 7.5 g/kg). For this initial state, the fuel assembly temperature rise \( \delta T_{k,rel} \) \( k=1,95 \) was calculated from measured cold leg and assembly outlet temperatures. The measured data and the 60-degree rotational symmetry of the fresh fuel core were used to estimate the heat up for assemblies without temperature control, so that the full core temperature rise distribution was obtained.

**Transient:** A transient was initiated by closing the steam isolation valve of SG-1 and isolating SG-1 from feed water. The pressure in SG-1 started to increase and stabilized at 6.47MPa in about 20 min. The MSH pressure was maintained approximately constant during the transient by operating the steam dump to condenser in pressure control mode. The coolant temperature in the cold and hot legs of loop n#1 rose by app.13.5°C and the mass flow rate decreased by about 3.4%. The mass flow rate through the reactor decreased 1%. At 90 s after the disturbance, the temperature of cold leg n#1 exceeded that of the hot leg. The difference stabilized to 0.6-0.8°C in about 20 min. The reactor power changed 0.16% calculated from
primary circuit energy balance. The initially symmetric core power distribution did not change significantly.

Final state: For the analysis presented here, the stabilized state of the experiment 30 minutes after the separation of the SG-1 is considered as “final state”. The core inlet temperatures were calculated from the measured core outlet temperatures and the estimated assembly by assembly temperature rise \( \delta T_k \), \( k=1,163 \) for the initial state. The \( \delta T_{k,rel} \) distribution was assumed constant during the transient due to the approximately constant normalized core power distribution.

For this analysis the benchmark problem was run with vessel boundary conditions from the V1000CT-2 benchmark specifications. The task is to calculate the coolant parameters at fuel assembly inlets and at the reactor outlets. Measured hot leg temperatures and estimated assembly inlet temperatures were used as reference.

3.2 VVER-1000 MSLB Problem

Two scenarios are defined for the purposes of OECD V1000CT-2 benchmark [2]. The first is close to the current licensing practice while the second is a pessimistic scenario derived from Scenario 1 by assuming that all main coolant pumps (MCP) remain in operation and the scram worth is reduced through adjustment of the cross sections.

**MSLB Scenario 1:** Large MSLB between the SG and SIV without loss of off-site power

The analysed transient is initiated by a main steam line break in a VVER-1000 between the steam generator (SG) and the steam isolation valve (SIV), outside the containment.

One of the major concerns for this case is the possible return to power and criticality after reactor scram due to overcooling. Because of this concern, the main objective of the study is to clarify the local 3D feedback effects depending on the vessel mixing.

A burnt fuel loading with three year fuel assemblies is considered. The reactor is at the end of cycle (EOC) and the initial hot full power (HFP) conditions are chosen consistent with the above objective. The SG water inventory is about the possible maximum at HFP. Following the break and the scram signal, one of the most reactive peripheral control assemblies remains stuck out of the core and is assumed to be in the affected sector.

A mechanical failure of the large feed water regulating valve in the broken line is assumed. At the time of the steam line rupture the valve starts to open from about 70% to 100% and then remains stuck in the open position. The main feed water flow to the faulted SG is terminated by closure of the feed water block valve in 52 s. The mass of feed water in the piping between the isolation valve and the affected SG, estimated to about 8000 kg, also contributes to the overcooling.

For benchmark purposes the FW temperature is conservatively fixed to 160°C to the broken SG and 170°C to the intact ones. In case of HPSI operation no credit is taken for the negative reactivity insertion from the boron addition. Other major assumptions are that off-site electric power is available, the MCP of the faulted loop trips and the other MCP run normally during the transient.

**MSLB Scenario 2:** Large MSLB upstream of SIV without loss of off-site power and with all MCP in operation

This is a pessimistic case derived from Scenario 1 by assuming that the MCP in the faulted loop fails to trip on signal and all reactor coolant pumps remain in operation. The scram worth is additionally reduced through adjustment of the cross sections.
For the present analysis, Exercise 2 was considered which consists of separate core-vessel calculation with given pre-calculated vessel MSLB boundary conditions. The reported results were obtained with validated multi-channel vessel models and point kinetics.

4 CATHARE2 VVER-1000 MODEL

The considered vessel thermal-hydraulic model [3] is multi-1D with cross-flow. For the purposes of this analysis the RPV thermal-hydraulics is described by 12, 16 or 24-sector vessel models. The testing includes separate effects (from vessel inlet to the fuel assembly inlets), component scale (vessel with boundary conditions) and full system simulation. The main features of the TH model are summarized below.

4.1 Cathare nodalization scheme

- Multi-1D channel vessel model with N sectors (N=12, 16 or 24) all the way from the inlet to the outlet. Each sector is one main channel, there is no radial subdivision
- Vessel inlet zone modeled with N volumes
- Down-comer modelled with N x 2 volumes and N axial elements
- Lower plenum modeled with 2 layers x N volumes
- Core with N main and N bypass channels
- Ten axial nodes in the core
- Upper plenum (shielding tubes zone) with 2 layers x N volumes
- Upper plenum (annular zone) with N volumes
- Upper plenum (outlet zone) with N volumes
- Upper head modelled with one volume
- Four-loop system model
- SG tube bundle primary side modelled with 6 layers and 10 axial nodes
- One-volume pressurizer model
- Steam generator (SG) secondary side modelled with an axial component both for the down-comer and riser, and two volumes for the upper part of the vessel
- Four main steam lines connected through the main steam header

The elements of the pressure vessel and primary circuit models are schematically shown on Figures 2 and 3.

4.2 Vessel mixing model

The vessel mixing is modelled through cross-flow between the parallel channels and is governed by local pressure drops. Cross-flow is modelled with horizontal junctions and vertical (diagonal) junctions connecting donor cells at given elevation to receptor cells in the neighbour sectors, at higher elevation. Tuning was applied only in the initial steady state, through adjustment of the flow resistance in horizontal cross-flow junctions and flow area in diagonal junctions. Vertical junctions were used to a limited extent, with relatively small flow area and in the lower and upper plenums only. Plant data from ref. [1] and validated CFD calculations by Hoehne [4] were used to validate the multi-1D vessel mixing model.

The assembly by assembly temperature and flow distributions at core inlet were obtained from the corresponding channel parameters through an appropriate mapping scheme. The temperature at the boundary between two sectors was taken as the weighted average of the two sector temperatures. This yields 88 inlet temperatures for the 12-sector model, and 154 ones for the 24-sector model.
Figure 2 Primary circuit nodalization scheme (2 of 16 vessel channels are shown)

Figure 3 Secondary circuit nodalization scheme (SG-2 and steam line 2 are shown)
5 RESULTS

5.1 Simulation of the Kozloduy-6 Flow Mixing Experiment

CATHARE multi-channel calculated results for the final state are given in Tables 1, 2 and Figures 6-8. The comparison illustrates the effects of azimuthal mesh refinement on the prediction of assembly inlet temperatures and the angular turn of the loop flow centres, as well as the vessel outlet temperatures. Although radial refinement is not considered in this study the results are quite reasonable due to the use of appropriate mapping schemes at the core inlet. The results show that for 16 or more azimuth meshes the computed angular shift of the loop flow centre with respect to the loop axis is in good agreement with the plant data. In the present study the loop flow centre is defined as the centre line of the zone of minimal mixing. This zone is formed of assemblies where the temperature difference between the disturbed cold leg and each assembly inlet is ≤ 1.2 K. The maximal deviations between computed assembly inlet temperatures and plant estimated data are 3.41 K for 24 azimuth sectors, 4.88 K for 16 sectors and 5.01 K for the 12-sector model. The average in modulus deviations are 0.94 K for 24 sectors, 0.92 K for 16 sectors and 0.92 K for 12 sectors.

5.2 MSLB Mixing Calculations

Available plant estimated data for the angular turn of loop #4 flow [1] shown on Figure 10 can be used for qualitative comparison of the computed angles, depicted on Figures 11 and 12. The angular turn of loop #4 flow shown on Figure 10 is estimated in terms of loop-to-assembly outlet mixing coefficients Cnk (%), defined as the ratio of flow from loop n into assembly k to the total flow through assembly k. The zone of minimal mixing is formed by assemblies with 90% < Cnk < 100% (app. equivalent to δTn,k < 1.2 K). Reasonable agreement is observed when using validated multi-channel models with cross-flow.

Figure 9 shows a CATHARE-CATHARE comparison of assembly inlet temperatures for Scenario 2 illustrating the effect of angular mesh refinement.

The analysis of the computed results on Figures 11 and 12, and other comparisons for the core inlet parameters with a CFX reference solution [4] (not shown here) permit to conclude that the validated multi-channel vessel mixing models are applicable to MSLB analysis. The maximal errors can be further reduced by radial mesh refinement.

5 CONCLUSIONS

Computationally efficient CATHARE2 multi-channel vessel mixing models were tested on the OECD VVER-1000 coolant transient benchmarks. The results show that such validated models are applicable to the analysis of asymmetric reactivity accidents characterized by sector formation.

ACKNOWLEDGMENTS

This study was carried out by using CATHARE2 V2.5 developed by CEA, EDF, AREVA NP and IRSN.

REFERENCES

Table 1 Initial state

<table>
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<tr>
<th>PARAMETER</th>
<th>Plant data</th>
<th>Cathare 12 sectors</th>
<th>Cathare 16 sectors</th>
<th>Cathare 24 sectors</th>
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<td>281</td>
<td>281</td>
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<td>541.75 (BC)</td>
<td>541.75 (BC)</td>
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<td>0.417</td>
<td>0.418</td>
<td>0.418</td>
<td>±0.043</td>
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Table 2 Final state

<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>Plant data</th>
<th>Cathare 12 sectors</th>
<th>Cathare 16 sectors</th>
<th>Cathare 24 sectors</th>
<th>Uncertainty</th>
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<td>286</td>
<td>286</td>
<td>286</td>
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<td>±1.5</td>
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<td>±1.5</td>
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<td>0.410</td>
<td>0.411</td>
<td>0.411</td>
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Figure 5 Plant data: Estimated temperature differences (K) from cold leg #1 to assembly inlets and angular turn of the loop #1 flow centre
Figure 6 CATHARE 16-sector vessel model: Angular turn of the loop #1 flow centre
Figure 7 CATHARE 24-sector vessel model: Angular turn of the loop #1 flow centre
Figure 8 Kozloduy-6 flow mixing test. Assembly by assembly core inlet temperatures

Figure 9 Snapshot for MSLB Scenario 2 at highest return to power. Assembly by assembly core inlet temperatures for different angular meshes
Figure 10 Kozloduy-6 vessel mixing experiment - assembly outlet data for loop-to-assembly mixing coefficients. Zone of minimal mixing and angular turn of the flow centre for loop #4
Figure 11 MSLB Scenario 2 with stuck rods #117 and #140. CATHARE 12-sector calculated angular turn of loop#4 flow centre.
Figure 12 MSLB Scenario 2 with stuck rods #117 and #140. CATHARE 24-sector calculated angular turn of loop#4 flow centre
ABSTRACT

The main objective for the quantification of the fluid mixing in the downcomer and the lower plenum is the demonstration of the safety of the nuclear plant during non-symmetrical transients. This concerns two main topics: The risk of brittle fracture of the Reactor Pressure Vessel (RPV) due to Pressurized Thermal Shock (PTS) and the risk of core reactivity excursion during non-symmetrical transient such as Main Steam Line Breaks (MSLB) or Boron Dilution Transients (BDT). These scenarios are studied in the 1:5 scaled VVER-1000 reactor model at OKB “Gidropress” in the framework of a TACIS project: “Development of safety analysis capabilities for VVER-1000 transients involving spatial variations of coolant properties (temperature or boron concentration) at core inlet”.

The 3-D computational fluid dynamics (CFD) codes provide an effective tool for mixing calculations. In recent years, the rapid development of both the software and the computers has made it feasible to study the coolant mixing in sufficient detail and to perform the calculations for transient conditions. The CFD-Code used was ANSYS CFX. The geometric details of the construction internals inside the RPV have a strong influence on the flow field and on the mixing. Therefore, a detailed representation of the inlet region, the spacer in the downcomer, the elliptical perforated plate and the complicated structures in the lower plenum was necessary. All parts of the lower plenum structures were modeled in detail. The computational grid contained 4.3 Million nodes. In the VVER-1000 reactor, similar characteristic flow and mixing pattern are observed in the case of nominal flow conditions like for Western type PWR. Sensitivity analyses were performed following recommendations included in the ECORA Best Practice Guidelines.

Regarding the flow field and mixing in the downcomer during four loop operation at nominal flow rates, it has been shown that a sharp sector formation like in western 4-loop
reactors appears. The flow field is inhomogeneous, in fact high velocity values occur beside the loop positions, and not below the inlet nozzles, which indicates the presence of recirculation areas or stagnant zones. Regarding the flow field and mixing at the core inlet, it has been shown that the mass flow rate distribution is more or less homogenous over the core diameter due to the lower plenum internals, the perturbed sector covers more or less one fourth of the core; a sharp sector formation like in western 4-loop reactors appears, weak mixing zones appear (around 99.7% of the unperturbed concentration). In most cases, the sensitivity analyses performed did not show any appreciable dependence of the results with respect to the addressed parameters. A three loop operation was chosen to show the differences of the flow and mixing behaviour compared to the four loop operation.

An extensive experimental program was running, aimed at studying different flow conditions in the reactor mock up, such as the start-up of the 1st coolant pump or natural circulation conditions with density differences of the primary coolant. Pre and post test CFD simulations are being carried out for code validation and for a deeper understanding of the flow and mixing behaviour in the VVER-1000 reactor.

1 INTRODUCTION

For the analysis of boron dilution and Pressurized Thermal Shock (PTS) transients the modeling of the coolant mixing inside the reactor vessel is important, because the reactivity insertion strongly depends on boron acid concentration or the coolant temperature distribution [1], [2]. The 3-D computational fluid dynamics (CFD) codes provide an effective tool for mixing calculations. In recent years, the rapid development of both the software and the computers has made it feasible to study the slug transportation in sufficient detail and as a transient calculation. Model experiments for studying mixing of diluted slugs have been performed at the Rossendorf Coolant Mixing Test Facility ROCOM. Due to the use of this comprehensive experimental database for CFD code validation, the experience in numerical modeling of the coolant mixing in pressurized water reactors ([3], [4], [5], [9] and [10]) could be applied in this study. The Best Practice Guidelines [6], which have been specified for nuclear reactor safety calculations within the ECORA project, have been used to optimize the numerical studies of transients in different reactor types with respect to meshing, selection of time step and physical models (e.g. turbulence models). This knowledge is basically used in this study.

The TACIS Project R2.02/02 (Ref. [1]) involves pre-test Computational Fluid Dynamics (CFD) simulations of the ten experiments that are to be conducted on the Gidropress Mixing Facility (Ref. [8]). The calculations were performed using the ANSYS CFX CFD code. The present paper describes some of the pre-test CFD simulations of the first experiments. The mentioned experiment consists in running the four circulation pumps of the Gidropress Mixing Facility at steady-state conditions corresponding to the nominal mass flow rate, and injecting a tracer in the cold leg of loop #4 for a certain period. The calculations were performed using the ANSYS CFX 10.0 CFD code.

2 GIDROPRESS MIXING FACILITY

The test facility has four circulating loops with a model of the reactor and the pressurizer. Schematic spatial circuit of the test facility is given in Figure 1. The base of experimental facility is one-to-five scale metal model of WWER-1000 reactor where in the geometry of the flow section of Novovoronezh NPP reactor, Unit No. 5 beginning from inlet nozzles to the core inlet is simulated. The reactor model volume is 0,888 m$^3$. The reactor model is also shown in Figure 1. The core is simulated partially. Instead of FA simulators and protective tube unit in the reactor model there is an «assembly» (Figure 3). The «assembly» is a bundle consisting of 91 tubes with 14x2 mm in diameter that is assembled with the help of
three spacing grids by which the pressure loss of the core and the protective tube unit is simulated. Rods with conductivity measurement probes are mounted through the model cover into these tubes.

![Vertical cut through RPV of VVER-1000](image1)
![Horizontal cut through RPV of VVER-1000 at the spacer element positions](image2)

![Vertical cut through RPV of VVER-1000 (lower plenum)](image3)
![Sketch of the reactor model core barrel bottom](image4)

Figure 1: The Gidropress Mixing Facility

In the lower part of the model between the internal surface of the vessel bottom is formed by rotating an ellipse as well as the external surface of the core barrel. Water comes into the core barrel bottom from the lower plenum through 1004 holes of 8.0 mm diameter and 320 holes of 18 mm diameter in elliptical grid. Perforation of supporting tubes located inside the core barrel bottom (Figure 1) was not simulated geometrically; it simulates only a
pressure loss of full-scale supporting tubes. Pressure loss coefficients are reduced to a velocity in pipeline of 170 mm in diameter with four loops being in operation. The test facility has a reactor coolant system consisting of four loops and auxiliary systems for filling and blow down of coolant system, as well as for preparation and injection of salt solution.

Table 1: Pressure loss coefficients (PLC)

<table>
<thead>
<tr>
<th>Component of flow path</th>
<th>PLC</th>
</tr>
</thead>
<tbody>
<tr>
<td>Inlet nozzle</td>
<td>0.73</td>
</tr>
<tr>
<td>Downcomer annular channel</td>
<td>0.05</td>
</tr>
<tr>
<td>Elliptical grid of the core barrel bottom (with a gap of 17 mm)</td>
<td>2.60</td>
</tr>
<tr>
<td>Simulator of supporting tube</td>
<td>0.24</td>
</tr>
<tr>
<td>Outlet nozzle</td>
<td>0.55</td>
</tr>
</tbody>
</table>

In each of circulation loops there is a flash tank intended for simulating the volumetric ratio of coolant in the test facility and in the reactor plant. Coolant volume in each circulation loop is 0.727 m$^3$. Each pump has a frequency-controlled drive for the control of volumetric flow. Flow rates in circulation loops are measured with the help of electromagnetic flow meters. System for salt solution preparation and injection into the facility RCS includes: tank for salt solution preparation, measuring pump, fast response valve and drainage tank. Salt solution is supplied into the system by batching pump with capacity of 60 m$^3$/h and pressure head of 0.5 MPa. Downstream of the valve a collector from which four lines for salt solution supply into each of circulation loops branch is mounted. Salt solution for the experiments simulating a steam line break and an asymmetric boron injection is injected into return pipeline directly upstream of circulating pumps. For measurement of salt concentration during the experiments the conductivity measurement probes are used, these sensors are installed:

- at the outlet of the supporting tubes of the reactor model core barrel bottom
- at the inlet and outlet nozzles of the reactor model
- at different elevation in tank for salt solution preparation
- in drainage tank
- in the line of salt solution supply downstream of fast response valve.

3 THE EXPERIMENT

Ten experiments will be conducted on the Gidropress Mixing Facility, with the purpose of investigating the mixing phenomena occurring inside the RPV of a VVER-1000 reactor during transients involving time and space perturbations of the coolant physical properties at the core inlet. Such experiments belong to the following three categories:

- Experiments simulating the start-up of one Reactor Coolant Pump (RCP), along with the transport of a deborated water slug into the RPV and through the reactor core (two experiments)
- Experiments simulating the onset of a natural circulation regime, along with the transport of a deborated water slug into the RPV and through the reactor core, accounting for different density ratios between the injected solution and the coolant (three experiments)
- Experiments simulating an asymmetric operation of the RCPs, which can occur during Main Steam Line Break (MSLB) scenarios, with the simultaneous injection of deborated water into the coolant circuit (five experiments)
The first experiment E#6 is one of the 3rd type (asymmetric RCPs operation), namely the one involving the steady state operation of all the four pumps at nominal conditions (i.e. 172 m³/hr), and the injection of tracer solution in loop #4 (upstream the circulation pump) for a 60 s period and with a 14 m³/hr volumetric flow rate.

Table 2: Mixing Experiment E#6

<table>
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<tr>
<th>Circulation loop flow rate, m³/h</th>
<th>Flow rate of injection, m³/h</th>
<th>Concentration, g/dm³</th>
<th>t, s</th>
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<tbody>
<tr>
<td>Q1</td>
<td>Q2</td>
<td>Q3</td>
<td>Q4</td>
</tr>
</tbody>
</table>

4 CFD SIMULATIONS

4.1 The computational grids used

The “meshing activity” led to the achievement of several computational grids, and to the selection of a “reference grid” among them, to be used for the pre-test CFD analysis of the first experiment. The selection of the reference grid (Figure 2) was based on computational performance criteria, i.e. the grid achieving a given convergence target within the smallest number of iterations in test calculations was chosen.

The computational domain of the reference grid includes the following parts:

- Four RPV inlet nozzles
- RPV downcomer
- Lower plenum – below the barrel bottom
- Perforated shell (barrel bottom)
- Solid columns region (lower part of the support columns)
- Perforated columns region (upper part of the support columns)
- Periphery (region surrounding the perforated columns)
- Plate&Outlet (core support plate, plus the outlet volume defined in the core region)

In addition, three modified versions of the reference grid [11] were developed in order to perform sensitivity analyses recommended by the ECORA Best Practice Guidelines (BPG, Ref. [6]), and described in the next Sections. The whole set of meshes used for this pre-test activity is summarized in Table 4.

4.2 CFD simulations set-up

This Section describes the set-up of the pre-test CFD simulations that have been performed by the Consultant (in particular, by UNIPI and FZD) for the first mixing experiment to be conducted on the Gidropress Mixing Facility. The two Organizations agreed upon a common approach to follow, based on the following two steps:

UNIPI and FZD set-up and run a reference calculation based on the same grid (i.e. the reference grid), boundary and initial conditions, numerics and any other input parameter. The
aim is to assess a “checked” common set-up, and to evidence possible user-effects of issues related to the different computational resources used by UNIPI and FZD.

A number of sensitivity calculations recommended by the ECORA Best Practice Guidelines are identified and then performed, based on modified versions of the reference calculation set-up and of the reference grid. Some of those sensitivity analyses are carried out by FZD, the other ones by UNIPI.

Table 3: Computational grids used

<table>
<thead>
<tr>
<th>#</th>
<th>Grid Id.</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Assembly 1</td>
<td>reference grid</td>
</tr>
<tr>
<td>2</td>
<td>Assembly 1.B</td>
<td>domain extended to include part of the cold legs</td>
</tr>
<tr>
<td>3</td>
<td>Assembly 1.C</td>
<td>extended outlet volume</td>
</tr>
<tr>
<td>4</td>
<td>Assembly 1.fine</td>
<td>increased number of nodes</td>
</tr>
</tbody>
</table>

Figure 2: Reference Grids
All the simulations were performed with the ANSYS CFX-10.0 package [12]. It has capabilities for simulating a wide variety of fluid flow problems, among which single-phase mixing problems. It embeds the following modules: a pre-processing tool (CFX-Pre) for setting-up simulations, a solver tool (CFX-Solve) that solves for the hydrodynamic equations using a fast coupled solving algorithm, and a post-processing tool (CFX-Post) that allows the visualization and the analysis of simulation results. The solver is based on a finite volumes – finite elements hybrid approach. Here are some common features of all the performed calculations:

- Working fluid: water (incompressible) at 1 atm, 25 °C
- Density: 997 kg/m³
- Dynamic viscosity: \(8.899 \times 10^{-4}\) kg m\(^{-1}\) s\(^{-1}\)
- Turbulence accounted for either with SST or BSL model
- The following field equations have been solved:
  - Mass balance (Continuity)
  - Momentum balance (Navier-Stokes)
  - Transport of turbulent kinetic energy (k)
  - Transport of turbulent eddy frequency
  - Transport of an additional, user-defined, scalar variable simulating the tracer

According to the specifications of the Gidropress Mixing Facility Experiment #6 all the simulations involve a constant and symmetric mass flow rate through all the loops that corresponds to 2.1049 m/s inlet velocity at each inlet nozzle. Concerning the tracer injection, it was decided to simulate it as a continuous injection of the transported additional scalar into the inlet nozzle #1, instead of following the entire time history of the injection. The defined scalar assumes values in the range 0 to 1. The value 0 corresponds to absence of tracer, while the value 1 represents the average normalized tracer concentration of the coolant entering from the “perturbed” loop. The transported scalar will be referred to in the following as the “mixing scalar”. Almost all calculations were run as steady-state problems (i.e. taking advance of the robust and efficient time-independent solver available in CFX), with the only exception of one sensitivity calculation (see below).

4.2.1 Definition of monitor points for target variables

The following target variables are considered for comparison purposes:

- the mixing scalar space distribution in the downcomer
- the mixing scalar space distribution at the core inlet
- the velocity space distribution in the downcomer

Monitor points have been defined in the simulation set-up in order to “extract” the calculated values of the above variables.

A top view of the computational model is represented in Figure 3, in order to show the position of the reference axis with respect to the nozzle; the perturbed nozzle is the inlet number 4 and is indicated with a red arrow in the Figure.
151 monitor points have been defined and located in the centre of each tube of the core inlet. In Figure 4 the location of the monitor points is shown along with the adopted numbering. In addition, 36 further monitor points have been defined in the downcomer, for the extraction both of velocity and mixing scalar. The plane where the downcomer monitor points stay is 0.65 m far from the nozzle plane (see Figure 5).
4.2.2 Hardware used and aspects related to parallelism

All the calculations performed by FZD have been run on a computer-cluster with the following features:

- Operating system: Linux Scientific 64 bit
- 32 AMD Opteron Computer Nodes
- Node configuration: 2 x AMD Opteron 285 (2.6 GHz, dual-core), 8 GB Memory
- pvm protocol for message passing in parallel calculations

All the calculations performed by UNIPI have been run on a computer-cluster (available at the DIMNP – San Piero a Grado Nuclear Research Group) with the following features:

- Operating system: Linux Red Hat Enterprise 4
- 8 dual-processor nodes – AMD Opteron 64 bit
- 32 GB RAM
- pvm protocol for message passing in parallel calculations

Either four or eight processors (out of the sixteen available) were used for the above mentioned simulations.

4.3 CFD SIMULATION RESULTS - USE OF BEST PRACTICE GUIDELINES FOR E#6

Regarding the flow field and mixing in the downcomer, it has been shown that:

- A sharp sector formation like in western 4-loop reactors appears (the same had been previously observed also in ROCOM experiments and simulations, Ref. [3]), the perturbed sector is approximately 1/4 of total area of the downcomer region (Figure 6a and 8).
- The flow field is inhomogeneous; in fact high velocity values occur beside the loop positions, and not below the inlet nozzles (Figure 7a)
- The in-between velocity goes almost to 0 m/s, which indicates the presence of recirculation areas or stagnant zones (Figure 7a)
- Four velocity maxima exist in the downcomer (Figure 7a)
Regarding the flow field and mixing at the core inlet, it has been shown that:

- The mass flow rate distribution is more or less homogenous over the core diameter due to the lower plenum internals (Figure 7b)
- The perturbed sector is approximately 1/4 of total area at the core inlet region (see Figure 8), a sector formation like in western 4-loop reactors appears
- Weak mixing zones appear (around 99.7% of the unperturbed concentration)

4.3.1 Sensitivity analysis on turbulence model

A number of different two-equation turbulence models are available in CFX, i.e. the classical k-ω and k-ε models and more advanced versions which stem from them (see the CFX User Guide, Ref. [12]). Two-equation models are very widely used, as they offer a good compromise between numerical effort and computational accuracy. Both the velocity and length scale are solved using separate transport equations (hence the term ‘two-equation’). The two-equation models use the gradient diffusion hypothesis to relate the Reynolds stresses to the mean velocity gradients and the turbulent viscosity. The turbulent viscosity is modeled as the product of a turbulent velocity and turbulent length scale. In two-equation models, the turbulence velocity scale is computed from the turbulent kinetic energy, which is provided from the solution of its transport equation. An extensive discussion on turbulence models can be found on Ref. [13]. The FZD and reference calculation was run using the Shear Stress Transport (SST) model. For the “sensitivity” calculation (GPMF_FZD_turb_mod) the baseline k-ω (BSL) was used instead. The comparison of results is reported in Figures 6 and 7. No appreciable differences appear for the mixing scalar at the core inlet, while small discrepancies appear in the downcomer mixing scalar and in the velocity profile.

4.3.2 Sensitivity analysis on turbulence intensity at the inlet boundary condition

When turbulence equations are solved for, it is necessary to specify turbulent parameters as inlet boundary conditions. For an easy set-up, CFX-Pre allows choosing among three turbulence intensity levels, i.e. “low” (1%), “medium” (5%, the default value, recommended by the CFX User Guide in case of absence of any information about the inlet turbulence, Ref. [12]) and “high” (10%). The FZD and reference calculation was run using medium inlet turbulence intensity; while for the “sensitivity” calculation (GPMF_FZD_turb_int) a high intensity was used. The values of the mixing scalar at core inlet (Figure 6b) are nearly the same for the two cases analyzed. Little discrepancies are visible in the downcomer mixing scalar (6a) and velocity profile (7a).

4.3.3 Sensitivity analysis on outlet boundary condition

In almost all the simulations an outlet pressure-controlled boundary condition was imposed (referred to in CFX simply as “outlet”). This is the most common choice when it is known a priori that flow is directed out of the domain, and the direction is assumed to be normal to the outlet boundary. Another possible choice consists in using the “opening” boundary condition, which allows the fluid to cross the boundary surface in both directions. For sensitivity analysis purposes, one simulation (GPMF_FZD_oulet_cond) was performed setting an “opening” boundary condition. The compared values of the mixing scalar at the core inlet and at the downcomer are reported in Figure 6. While no appreciable differences are evident in the downcomer, some discrepancies appear for the mixing scalar at few individual channels.
4.3.4 Sensitivity analysis on time-dependent solver

As described before, a “transient” calculation (GPMF_FZD_transient) has also been performed by FZD. It was a 15 seconds transient with a time step of 0.1 s. A comparison of the mixing scalar between the steady-state and the transient case is reported in Figure 6. Streamlines from inlet nozzle #4, and the mixing scalar at the RPV wall are represented in

Figure 6: Comparison of the mixing scalar at the downcomer and core inlet

Figure 7: Comparison of the velocity field at the downcomer and core inlet

Figure 8: Comparison of the velocity field at the downcomer and core inlet
4.3.5 Sensitivity analysis on mesh size

The mixing scalar values at the core inlet and at the downcomer for both the reference mesh and the “fine mesh” are also compared in Figure 6, the velocity fields are compared in Figure 7. The mesh size appears to sensibly influence the flow field (which is a little smoothed in the “fine grid” case”), as well as the mixing scalar distribution over the core inlet (although no difference seems to exist from a qualitative point of view).

4.3.6 Maximum and averaged mixing scalar values

The maximum and the averaged values for the mixing scalar at the core inlet are reported in Table 4, along with the relative error calculated with respect to the UNIPI reference case. In almost all the cases analyzed the averaged value is approximately the same (differences less that 0.5%), except for the finer mesh case (GPMF_FZD_mesh), with differences of -2.0% and 1.2% respectively. The maximum value varies from 0.970 (fine mesh) to 0.997 (FZD reference case), thus it is always very close to the “unperturbed” value (i.e. 1). This means that the presence of zones which are affected by almost no turbulent mixing is predicted by all simulations, and that a grid refinement would probably lead to the prediction of a more effective turbulent mixing.

Table 4: Maximum and averaged values for the mixing scalar at the core inlet.

<table>
<thead>
<tr>
<th>ID</th>
<th>MAXIMUM</th>
<th>%DIFF</th>
<th>AVERAGED</th>
<th>%DIFF</th>
</tr>
</thead>
<tbody>
<tr>
<td>GPMF_FZD_mesh</td>
<td>0.970</td>
<td>-2.41</td>
<td>0.254</td>
<td>1.20</td>
</tr>
<tr>
<td>GPMF_FZD_outlet_cond</td>
<td>0.991</td>
<td>-0.30</td>
<td>0.251</td>
<td>0.00</td>
</tr>
<tr>
<td>GPMF_FZD_turb_mod</td>
<td>0.977</td>
<td>-1.71</td>
<td>0.251</td>
<td>0.00</td>
</tr>
<tr>
<td>GPMF_FZD_turb_int</td>
<td>0.977</td>
<td>-1.71</td>
<td>0.251</td>
<td>0.00</td>
</tr>
<tr>
<td>GPMF_FZD_transient</td>
<td>0.988</td>
<td>-0.60</td>
<td>0.252</td>
<td>0.40</td>
</tr>
</tbody>
</table>

5 SUMMARY

Scenarios are studied in the 1:5 scaled VVER-1000 reactor model at OKB “Gidropress” in the framework of a TACIS project: “Development of safety analysis capabilities for VVER-1000 transients involving spatial variations of coolant properties (temperature or boron concentration) at core inlet”. The CFD-Code used was ANSYS CFX. The geometric details of the construction internals inside the RPV have a strong influence on the flow field and on the mixing. Therefore, an exact representation of the inlet region, the spacer in the downcomer, the elliptical perforated plate and the complicated structures in the lower plenum was necessary. All parts of the lower plenum structures were modeled in detail. The computational grid contained 6.5 Mio. hybrid elements. Sensitivity analyses were performed following recommendations included in the ECORA Best Practice Guidelines. Regarding the flow field and mixing in the downcomer, it has been shown that a sharp sector formation like in western 4-loop reactors appears (the same had been previously observed also in ROCOM experiments and simulations), the flow field is inhomogeneous; in fact high velocity values occur beside the loop positions, and not below the inlet nozzles, the in-between velocity goes almost to zero, which indicates the presence of recirculation areas or stagnant zones. Regarding the flow field and mixing at the core inlet, it has been shown that the mass flow rate distribution is more or less homogenous over the core diameter due to the lower plenum.
internals, the perturbed sector covers more or less one fourth of the core; a sharp sector formation like in western 4-loop reactors appears, weak mixing zones appear (around 99.7% of the unperturbed concentration). In most cases, the sensitivity analyses performed did not show any appreciable dependence of the results with respect to the addressed parameters.

ACKNOWLEDGEMENT

The work reported about in this paper was supported by the EU TACIS Project R2.02/02. “Development of safety analysis capabilities for VVER-1000 transients involving spatial variations of coolant properties (temperature or boron concentration at core inlet)”

LITERATURE

[8] Lisenkov E.A., Description of experiments, SPA Gidropress 320-Pr-656E (Rev. A), TACIS R2.02/02 Task B5.
ABSTRACT

The principles that support the risk-informed regulation should be considered in an integrated decision-making process. Thus, any evaluation of licensing issues supported by a safety analysis should take into account both deterministic and probabilistic aspects of the problem. The deterministic aspects should be addressed using best estimate coupled code calculations and considering the associated uncertainties. In recent years there has been an increasing demand from the nuclear community for best estimate predictions to be provided with their confidence bounds. The ongoing OECD Light Water Reactor (LWR) Uncertainty Analysis in Modelling (UAM) benchmark activities contribute to establishing such unified framework to estimate safety margins, which would provide more realistic, complete and logical measures of reactor safety. This benchmark sequence started in 2007, and integrates the expertise in reactor physics, thermal-hydraulics and reactor system modelling as well as uncertainty and sensitivity analysis. Such an effort has been undertaken within the framework of a program of international co-operation that benefits from the coordination of the NEA Nuclear Science Committee (NSC), and from interfacing with the Committee of safety of Nuclear Installations (CSNI). This paper describes the OECD LWR UAM benchmark by emphasising its safety implications. Reference LWR systems and scenarios for coupled code analysis are defined. Three main LWR types are selected, based on previous benchmark experiences and available data – BWR, PWR and VVER-1000. The objective is to determine the uncertainty in LWR system calculations at all stages of a coupled reactor physics/thermal hydraulics calculation. The full chain of uncertainty propagation from basic data, engineering uncertainties, across different scales (multi-scale), and physics phenomena (multi-physics) is tested on a number of benchmark exercises for which experimental data is available and for which the power plant details have been released. The above-described approach is based on the introduction of nine steps (exercises). These exercises are carried out in three phases, and follow the established routine calculation scheme for LWR design and safety analysis in industry and regulation. This paper describes in detail the Phase I (Neutronics Phase). The paper also discusses the priorities of Phase II (Core Phase) and Phase III (System Phase). Establishing such internationally accepted LWR UAM benchmark framework will help to address current regulation needs and issues related to practical implementation of risk informed regulation,
and offers the possibility to accelerate the licensing process when using best estimate methods.

1 INTRODUCTION

Three of the five principles that support the risk-informed regulation [1] are: 1) when the proposed changes to the licensing basis result in an increase in core damage frequency and/or risk, the increases should be small, 2) the proposed change maintains sufficient safety margins and 3) the proposed change is consistent with the defence-in-depth philosophy. Each of these principles should be considered in an integrated decision-making process. Thus, any evaluation of licensing issues supported by a safety analysis should consider both deterministic and probabilistic aspects of the problem. The deterministic aspects should be addressed using best estimate coupled code calculations and considering the associated uncertainties.

Presently, the tendency is to perform safety studies by best-estimate coupled codes that allow a realistic modelling of nuclear and thermal-hydraulic processes of the reactor core and the entire plant behaviour including control and protection functions. Over the last two decades many organizations from different countries have accumulated experience in developing, validating and applying coupling methodologies for reactor safety analyses [2]. Appropriate benchmarks have been developed in international co-operation led by the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD) that permits testing the neutronics/thermal-hydraulics coupling, and verifying the capability of the coupled codes to analyze complex transients with coupled core/plant interactions [3].

Nuclear regulation in many countries originally specified deterministic criteria to be met by computer codes. A set of results from a single run of the code was compared with a set of criteria. For the last twenty years “realistic methods” are allowed. Such methods are referred to as “best-estimate” calculations, implying that they use a set of data, correlations, and methods designed to represent the phenomena, using the best available techniques. The regulations also require that the uncertainty in these calculations be evaluated. i.e. these best-estimate results should be supplemented by uncertainty and sensitivity analysis. Current Uncertainty Analysis (UA) and Sensitivity Analysis (SA) methods rely on derivative based methods such as stochastic sampling methods (mostly used for evaluation of uncertainties in thermal-hydraulics safety calculations) and on generalized perturbation theory to obtain sensitivity coefficients (mostly used for evaluation of nuclear data related uncertainties in reactor core parameters). Both methods have drawbacks and neither approach addresses all needs.

The best-estimate methodologies supplemented with uncertainty analysis have been applied for thermal-hydraulics Large Break LOCA analysis. In order to incorporate uncertainties into the process usually many runs of the computer code are required and the outcome is a range of results with associated probabilities. The same approach has been applied in some cases to coupled neutronics/thermal-hydraulic calculations. There is a challenge to the technical community to design, implement and validate more comprehensive and robust uncertainty analysis tools for best estimate coupled code calculations.

In the proposed international LWR Uncertainty Analysis in Modelling (UAM) benchmark activity different UA methods for coupled codes will be compared, and their value assessed including the validation of the methodologies for uncertainty propagation [4]. For the first time the uncertainty propagation will be estimated through the whole simulation process on a unified benchmark framework to provide credible coupled code predictions with defensible uncertainty estimations of safety margins at the full core/system level. The
benchmark will allow not only to compare and to assess the current UA methods on representative applications but also will stimulate the further development of efficient and powerful UA methods suitable for complex coupled code simulations and will help to formulate recommendations and guidelines on how to utilize advanced and optimized UA and SA methods in “best estimate” coupled reactor simulations in licensing practices.

2 OVERVIEW OF THE OECD LWR UAM BENCHMARK ACTIVITY

2.1 Background

In recent years there has been an increasing demand from nuclear research, industry, safety and regulation for best estimate predictions to be provided with their confidence bounds. Consequently an "in-depth" discussion on “Uncertainty Analysis in Modelling” was organized at the 2005 OECD/NEA Nuclear Science Committee (NSC) meetings, which led to a proposal for launching an Expert Group on "Uncertainty Analysis in Modelling” and the endorsement to hold a workshop with the aim of defining: future actions and a program of work.

As a result the OECD/NEA Workshop on Uncertainty Analysis in Modelling took place in Pisa, Italy, on April 28-29, 2006 (UAM-2006) [5]. The major outcome of the workshop was to prepare a benchmark work program with steps (exercises) that would be needed to define the uncertainty and modelling task. The other proposals made during the meeting would be incorporated under the different steps (exercises) within the overall benchmark framework for the development of uncertainty analysis methodologies for multi-physics (coupled) and multi-scale simulations.

Following the results from the UAM-2006 Workshop the OECD/NEA NSC at its June 2006 meeting endorsed the creation of an Expert Group on Uncertainty Analysis methods in Modelling [6]. This Expert Group will report to the Working Party on Scientific issues in Reactor Systems (WPRS). Since it addresses multi-scale / multi-physics aspects of uncertainty analysis, it will work in close co-ordination with the benchmark groups on coupled neutronics/thermal-hydraulics simulations and on coupled core-plant problems. The Expert Group will also coordinate its activities with the Group on Analysis and Management of Accidents (GAMA) of the Committee on Safety of Nuclear Installations (CSNI). The Expert Group has the following mandate:

1. To elaborate a state-of-the-art report on current status and needs of sensitivity and uncertainty analysis (SA/UA) in modelling, with emphasis on multi-physics (coupled) and multi-scale simulations.
2. To identify the opportunities for international co-operation in the uncertainty analysis area that would benefit from coordination by the NEA/NSC.
3. To create a roadmap along with schedule and organization for the development and validation of methods and codes required for uncertainty analysis including the benchmarks adequate to meet those goals.

The NEA/NSC has endorsed that this activity be undertaken with the Pennsylvania State University (PSU) as the main coordinator and host with the assistance of the Scientific Board. The 40 participants in the UAM workshop in Pisa (from 26 organizations in 16 countries representing industry, regulatory agencies, national laboratories and research institutions) expressed interest in participating and contributing to this UAM Expert Group and proposed an uncertainty analysis benchmark activity.

To summarize, in addition to LWR best-estimate calculations for design and safety analysis, the different aspects of uncertainty analysis in modelling (UAM) are to be further developed and validated on scientific grounds in support of its performance. There is a need for efficient and powerful analysis methods suitable for such complex coupled multi-physics
and multi-scale simulations. The proposed benchmark sequence will address this need by integrating the expertise in reactor physics, thermal-hydraulics and reactor system modelling as well as uncertainty and sensitivity analysis, and will contribute to the development and assessment of advanced/optimized uncertainty methods for use in best-estimate reactor simulations. Such an effort can be undertaken within the framework of a program of international co-operation that would benefit from the coordination of the NEA/NSC and all participants by interfacing with the CSNI activities.

The first workshop for the OECD LWR UAM benchmark (UAM-1) was held on 10 and 11 May 2007 at the OECD/NEA Headquarters, Issy les Moulineaux, France [7]. The second workshop for the OECD LWR UAM benchmark (UAM-2) was held from 2 to 4 April 2008 in Garching, Germany, and was a follow up to the first workshop [8]. The meeting was organized around an in-depth discussion of the specification and support data for Phase I of the UAM LWR benchmark [4], preliminary results of Phase I, output parameters and format for Phase II, priorities for Phases II and III, and the proposed work plan and time schedule for the UAM LWR benchmark activities.

2.2 Objective

The objective of the work is to define, conduct, and summarize an OECD benchmark for uncertainty analysis in best-estimate coupled code calculations for design, operation, and safety analysis of LWRs. The title of this benchmark is: “OECD UAM LWR Benchmark”. Reference systems and scenarios for coupled code analysis are defined to study the uncertainty effects for all stages of the system calculations. Measured data from plant operation are available for the chosen scenarios.

The utilized technical approach is to establish a benchmark for uncertainty analysis in best-estimate modelling and coupled multi-physics and multi-scale LWR analysis, using as bases a series of well defined problems with complete sets of input specifications and reference experimental data. The objective is to determine the uncertainty in LWR system calculations at all stages of coupled reactor physics/thermal hydraulics calculation. The full chain of uncertainty propagation from basic data, engineering uncertainties, across different scales (multi-scale), and physics phenomena (multi-physics) is tested on a number of benchmark exercises for which experimental data is available and for which the power plant details have been released.

The principal idea is: a) to subdivide the complex system/scenario into several steps or Exercises, each of which can contribute to the total uncertainty of the final coupled system calculation, b) to identify input, output, and assumptions for each step, c) to calculate the resulting uncertainty in each step; d) to propagate the uncertainties in an integral systems simulation for which high quality plant experimental data exists for the total assessment of the overall computer code uncertainty. The main scope covers uncertainty (and sensitivity) analysis (SA/UA) in best estimate modelling for design and operation of LWRs, including methods that are used for safety evaluations. As part of this effort, the development and assessment of different methods or techniques to account for the uncertainties in the calculations will be investigated and reported to the participants.

2.3 Definition of Benchmark Phases and Exercises

The above-described approach is based on the introduction of 9 steps (Exercises), which allows for developing a benchmark framework which mixes information from the available integral facility and NPP experimental data with analytical and numerical benchmarking. Such an approach compares and assesses current and new uncertainty methods on
representative applications and simultaneously benefits from different methodologies to arrive at recommendations and guidelines. These 9 steps (Exercises) are carried out in 3 phases as follows:

**Phase I (Neutronics Phase)**

a) Exercise 1 (I-1): “Cell Physics” focused on the derivation of the multi-group microscopic cross-section libraries

b) Exercise 2 (I-2): “Lattice Physics” focused on the derivation of the few-group macroscopic cross-section libraries

c) Exercise 3 (I-3): “Core Physics” focused on the core steady state stand-alone neutronics calculations

**Phase II (Core Phase)**

a) Exercise II-1: Fuel thermal properties relevant for transient performance

b) Exercise II-2: Neutron kinetics stand-alone performance (kinetics data, space-time dependence treatment, etc.)

c) Exercise II-3: Thermal-hydraulic fuel bundle performance

**Phase III (System Phase)**

a) Exercise III-1: Coupled neutronics/thermal-hydraulics core performance (coupled steady state, coupled depletion, and coupled core transient with boundary conditions)

b) Exercise III-2: Thermal-hydraulics system performance

c) Exercise III-3: Coupled neutronics kinetics thermal-hydraulic core/thermal-hydraulic system performance

Separate Specifications will be prepared for each Phase in order to allow participation in the full Phase or only in a subset of the Exercises. Boundary conditions and necessary input information are provided by the benchmark team. The intention is to follow the calculation scheme for coupled calculations for LWR design and safety analysis established in the nuclear power generation industry and regulation. The specification document that covers Phase I (which includes the first 3 Exercises) was distributed to the participants [4].

### 2.4 Safety Implications

The expected impact and benefits of the OECD LWR UAM benchmark activity for LWR safety and licensing are summarized in [9]. This benchmark project is challenging and responds to needs of estimating confidence bounds for results from simulations and analysis in real applications. Among the expected results of this project are:

a) Systematic identification of uncertainty sources;

b) Systematic consideration of uncertainty and sensitivity methods in all steps. This approach will generate a new level of accuracy and will improve transparency of complex dependencies;

c) All results will be represented by reference results and variances and suitable tolerance limits;

d) The dominant parameters will be identified for all physical processes;

e) Support of the quantification of safety margins;

f) The experiences of validation will be explicitly and quantitatively documented;

g) Recommendations and guidelines for the application of the new methodologies will be established.

The OECD LWR UAM activity will establish an internationally accepted benchmark framework to compare, assess and further develop different uncertainty analysis methods associated with the design, operation and safety of LWRs. As a result the LWR UAM benchmark will help to address current nuclear power generation industry and regulation needs and issues related to practical implementation of risk informed regulation. The realistic
evaluation of consequences must be made with best estimate coupled codes, but to be meaningful, such results should be supplemented by an uncertainty analysis. The use of coupled codes allows to avoid unnecessary penalties due to incoherent approximations in the traditional decoupled calculations, and to obtain more accurate evaluation of margins regarding licensing limit. This becomes important for licensing power upgrades, improved fuel assembly and control rod designs, higher burn-up and others issues related to operating LWRs as well as to the new Generation 3+ designs being licensed now (ESBWR, AP-1000, EPR-1600 and etc.). Establishing such internationally accepted LWR UAM benchmark framework offers the possibility to accelerate the licensing process when using best estimate methods and contributes to establishing a unified framework to estimate safety margins, which would provide more realistic, complete and logical measures of reactor safety.

3 DISCUSSION OF PHASE I

Phase I is focused on understanding uncertainties in prediction of key reactor core parameters associated with LWR stand-alone neutronics core simulation. Such uncertainties occur due to input data uncertainties, modelling errors, and numerical approximations. Input data for core neutronics calculations primarily include the lattice averaged few group cross-sections. Three main LWR types are selected, based on previous benchmark experiences and available data:

a) PWR (TMI-1) [10];

b) BWR (Peach Bottom-2) [11];

c) VVER-1000 (Kozloduy-6, Kalinin-3) [12].

For cross-section generation any type of lattice solver can be used. For core calculations the established two-group energy structure for LWR analyses is proposed as the major part of the benchmark activities. However, provisions are made for utilization of other few-group structures if the participants want to investigate them. The Monte-Carlo method will provide reference solutions for the test problems of each Exercise of Phase I.

3.1 Exercise I-1: Cell Physics

The Exercise I-1 is focused on derivation of the multi-group microscopic cross-section libraries. Its objective is to address the uncertainties due to the basic nuclear data as well as the impact of processing the nuclear and covariance data, selection of multi-group structure, and self-shielding treatment. The intention for Exercise I-1 is to propagate the uncertainties in evaluated Nuclear Data Libraries - NDL - (microscopic point-wise cross sections) into multi-group microscopic cross-sections used as an input by their lattice physics codes. The participants can use any of the major NDLs such as Evaluated Nuclear Data Files (ENDF), Joint European Fission and Fusion files (JEFF), and Japanese Evaluated Nuclear Data Library (JENDL). The evaluation of nuclear data induced uncertainty is possible by the use of nuclear data covariance information. The development of nuclear data covariance files is in progress in major NDLs. For the purposes of the OECD LWR UAM benchmark the availability of covariance data is important for all relevant nuclides (actinides, fission products, absorbers and burnable poisons, structural materials and etc.), present in the reactor core and reflector regions of LWRs, covering the entire energy range of interest (from 0 to 10 MeV), and for all relevant reaction cross-section types.

Table 1 shows the total number of materials and cross-section reactions with neutron cross-section covariance data in the recent versions of the major evaluated nuclear data files.
Table 1: Number of materials and cross-sections with covariances of neutron cross-sections

<table>
<thead>
<tr>
<th>Data files</th>
<th>Number of materials</th>
<th>Number of cross-sections</th>
</tr>
</thead>
<tbody>
<tr>
<td>ENDF/B-VI.8</td>
<td>44</td>
<td>400</td>
</tr>
<tr>
<td>JEFF-3.1</td>
<td>34</td>
<td>350</td>
</tr>
<tr>
<td>JENDL-3.3</td>
<td>20</td>
<td>160</td>
</tr>
</tbody>
</table>

The covariance data in the major data files is scarce in terms of materials (including actinides) and types of covariance matrices available. They contain uncertainty information only for few isotopes and reactions and usually for different number and different isotopes in different files. For isotopes not included, usually their covariances are assumed to be zero, which will result in the underestimation of core parameters uncertainties. In conclusion, the status of available covariance data in the major NDLs is such that it cannot support the objectives of the OECD LWR UAM benchmark. Once the more complete covariance data are ready containing low fidelity covariances that supplement available NDL evaluations available for all required materials, it can be used for the purposes of this benchmark within the framework of Exercise I-1. For example, crude but reasonable covariances for all materials are being developed within the framework of ENDF/B-VII.0. When this development is completed and made available it can be used for the purpose of the OECD LWR UAM benchmark.

The current status of the evaluated cross-section NDLs is such that the most comprehensive covariance library is available with SCALE-5.1. For this reason it was decided to utilize the nuclide dependent multi-group covariance data from SCALE 5.1 for the purposes of Exercise I-1. It is based on a 44-group structure. For other group structures, NEA/OECD has provided the tools for handling and transforming the cross-section covariance in a consistent way. Covariance data are relative values and can be used with different NDLs. In order to analyze the results on a common ground, it is recommended to participants to use only these data and tools. The output uncertainties of Exercise I-1 are input uncertainties in Exercise I-2.

Following a request from the UAM expert group, the authorization was granted by the SCALE management and DOE to use the group cross-section covariance data now distributed with SCALE-5.1 for the purpose of the Phase I (Neutronics Phase) benchmark study and in connection with other codes. As a source of their cross-section data the participants can use the NDLs, which are normally used in conjunction with their lattice physics codes. The three major libraries (ENDF, JEFF and JENDL) are possible candidates. For cross-section covariance data the 44-group covariance libraries from SCALE-5.1 are proposed [13]. 44GROUPV6REC is the recommended covariance library based on several sources, including evaluated data files ENDF/B-VI, ENDF/B-V, JENDL, and JEFF. Data missing from all evaluated data files were represented by the “integral approximation”, for the resonance and thermal energy ranges only. This approximation was used for approximately 300 materials – see Table 2. The SCALE-5.1 recommended covariance library is currently being updated to include recent high fidelity ENDF/B-VII uncertainty evaluations for the nuclides U-235, U-238, Pu-239, Th-232, and Gd isotopes. Some of the integral approximation data also is being revised to more recent measured values. This is being made available for the UAM effort.
### Table 2: The nuclides or materials (in ZA order) for which covariance data are provided

| ZA | Material | ZA | Material | ZA | Material | ZA | Material | ZA | Material | ZA | Material |
|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|----------|----|
| 1  | H-1      | 2  | H-2      | 3  | H-3      | 4  | He-4     | 5  | Li-6     | 6  | Li-7     | 7  | Be-9     |
| 10 | C-0      | 11 | C-0      | 12 | N-14     | 13 | N-15     | 14 | O-16     | 15 | O-17     | 16 | F-19     |
| 23 | Mg-0     | 27 | Al-27    | 31 | Si-0     | 32 | K-0      | 34 | Ca-0     | 35 | Sc-45    |
| 0  | S-0      | 32 | Cl-0     | 54 | Cr-0     | 55 | Cr-54    | 57 | Mn-55    |
| 2  | Cu-0     | 62 | Ni-64    | 70 | Zn-90    | 71 | Zr-90    |
| 3  | B-10     | 11 | B-11     | 14 | C-0      | 15 | N-14     | 16 | O-16     |
| 56 | Fe-56    | 57 | Fe-57    | 58 | Fe-58    | 59 | Co-59    | 60 | Ni-60    |
| 82 | Ge-73    | 74 | Ge-74    | 76 | Ge-76    | 77 | Se-74    | 78 | Se-78    |
| 80 | Se-80    | 82 | Br-79    | 83 | Br-83    | 84 | Kr-84    | 88 | Sr-88    |
| 82 | Sr-88    | 90 | Y-90     | 92 | Zr-92    | 93 | Zr-93    |
| 94 | Nb-94    | 95 | Mo-95    | 96 | Mo-96    | 97 | Tc-97    |
| 96 | Ru-96    | 100| Ru-100   | 101| Ru-101   | 102| Ru-102   | 104| Ru-104   |
| 103| Rh-103   | 105| Pd-102   | 104| Pb-104   | 105| Pb-105   | 106| Pb-106   |
| 110| Pd-110   | 107| Ag-107   | 109| Ag-109   | 111| Cd-0      |
| 112| Cd-112   | 113| Cd-113   | 114| Cd-114   | 115| In-0      |
| 112| Sn-112   | 115| Sn-115   | 116| Sn-116   | 117| Sn-117   |
| 124| Sn-124   | 121| Sn-121   | 123| Sb-123   | 124| Te-124   |
| 124| Te-124   | 125| Te-125   | 126| Te-126   | 127| Te-127   |
| 130| I-130    | 131| Xe-131   | 132| Xe-132   |
| 137| Cs-137   | 138| Ba-138   |
| 140| Ce-140   | 141| Ce-141   | 142| Ce-142   | 143| Ce-143   |
| 143| Pr-143   | 144| Nd-144   | 145| Nd-145   |
| 150| Nd-150   | 151| Sm-151   | 152| Sm-152   |
| 152| Eu-152   | 154| Gd-154   | 155| Gd-155   |
| 155| Dy-156   | 159| Tb-159   |
| 164| Dy-164   | 166| Er-166   | 167| Eu-167   |
| 176| Hf-176   | 177| Hf-177   | 178| Hf-178   |
| 182| W-182    | 184| W-184    | 186| W-186    |
| 206| Pb-206   | 208| Pb-208   | 209| Bi-209   |
| 232| U-232    | 233| U-233    | 234| U-234    |
| 237| Np-237   | 238| Pu-238   |
| 241| Am-241   | 242| Am-242   |
| 249| Cl-249   |
| 253| Cf-253   | 254| Cm-254   |

In addition to the covariance matrices, a utility program for interpolating or collapsing from a given group structure to another one is provided for participants’ use [14]. Participants can choose any energy multi-group structure according to the input requirements of their lattice code to be utilized. The code ANGELO is designed for the interpolation of the multi-group covariance matrices from the original (in this case 44 groups) to a user defined energy structure which is also distributed for the convenience of the users. The algorithm used in the ANGELO code is relatively simple using flat weighting; therefore the interpolations involving the energy group structures which are very different from the original one (especially if the number of groups is reduced considerably) are to be avoided as they may not be accurate. Still, the procedure tends to be conservative. The interpolation procedure was found to give reliable results if the number of groups changed by a factor of up to 4. In this range the procedure can therefore be considered as an adequate and easy-to-use alternative to more rigorous methods, like the ERRORR module of NJOY [15].

LAMBDA is a program to verify some mathematical properties and the physical consistency of the data and the interpolation procedure, in particular the positive definiteness of the multi-group covariance matrices. The trace and the number of positive, negative, and zero eigenvalues are calculated and the matrix is classified on this basis. The correlation matrix is tested to determine if any element exceeds unity. This quality verification is highly recommended before using the covariance information to data consistency analyses with...
integral experiments and to data adjustment. Example of the uncertainties in $k_{\text{eff}}$ (in pcm) due to cross section uncertainties for the benchmark configuration KRITZ-2.13C [17] based on various covariance data is shown in Figure 1. The same sensitivity profiles were used in all three cases and only the covariance data were varied.

Figure 1: Uncertainties in $k_{\text{eff}}$ (in pcm) due to cross section uncertainties

Within the framework of Exercise I-1 the cross-section uncertainty data is processed in a multi-group format. The final multi-group cross-section libraries and associated uncertainties should be consistent with requirements of lattice physics codes, which participants are planning to utilize. In order to perform a comparative analysis of the multi-group cross-section uncertainty data obtained after processing test problems are devised or utilized from the previously defined benchmarks. These are two-dimensional fuel pin-cell test problems representative of BWR PB-2, PWR TMI-1, and VVER-1000 Kozloduy-6. The geometry representation of the BWR unit cell is given in Figure 2. The full geometry and material specifications are provided in [4]. The reflective boundary conditions are utilized at the boundaries of problems. These problems have to be analyzed at Hot Zero Power (HZP) conditions and Hot Full Power (HFP) conditions. The HFP operating conditions cases allow to vary the spectrum in the test problems.

Figure 2: Geometry representation of BWR unit cell
First, one group effective uncertainties will be compared. The effective uncertainties have to be obtained from each participant using his/her multi-group uncertainty data associated with the multi-group library used as input in his/her lattice physics code. The effective uncertainty (relative error) for neutron cross-sections corresponding to neutron flux spectrum of the pin-cell test problem of interest can be obtained as shown in the Equation below:

\[ \Delta^2 = \sum_{i=1}^{n} \text{eff}_i \cdot \text{cov}(i, j) \cdot \alpha_j \]  

(1)

where:

\[ \alpha_j = \frac{\sigma_{\text{eff}}}{\sigma} = \frac{\bar{\phi}}{\bar{\phi}_T} \]

\[ \phi_T = \sum_{i=1}^{n} \phi_i \]

with \( n \) the total number of energy groups;

\[ \text{cov}(i, j) \] is the ENDF multi-group covariance matrix

These effective uncertainties have to be calculated for the neutron cross-sections of the nuclides present in the pin-cell models. In addition, for each test problem participants have to calculate \( k_{\text{inf}} \) and absorption and fission reaction rates for \(^{234}\text{U}\), \(^{235}\text{U}\), and \(^{238}\text{U}\) and associated uncertainties due to multi-group cross-sections based on the processed in Exercise I-1 multi-group covariance matrices.

The NDL effect will be assessed by running Monte Carlo simulations with the major libraries: ENDF/B-VI.8, ENDF/B-VII.0, and JEFF-3.1. Continuous-energy Monte Carlo (MCNP5 [16]) solutions with converged eigenvalue and fission source distribution will be provided for each test problem. Example of the MCNP5 model for the BWR unit cell is shown in Figure 3 while Figure 4 illustrates the MCNP5 \( k_{\text{inf}} \) solution using ENDF/B-VII.0 continuous energy libraries for HZP conditions. The statistical uncertainties in the reference Monte Carlo calculations will be evaluated by the benchmark team. In the calculations of the above-described test problems the participants have to utilize their multi-group cross-section libraries (input to their lattice physics codes) and associated uncertainties. They can utilize their own Sensitivity/Uncertainty (S/U) tools to propagate cross-section uncertainties to calculate quantities of interest in nuclear analysis or the ones available at NEA/OECD.

3.2 Exercise I-2: Lattice Physics

In the current established calculation scheme for LWR design and safety analysis, multi-group microscopic cross-section libraries are an input to lattice physics calculations. The multi-group cross-section uncertainties (multi-group cross-section covariance matrix) should be obtained by participants as output uncertainties within the framework of Exercise I-1. In Exercise I-2 multi-group cross-section uncertainties are input uncertainties and must be propagated through the lattice physics calculations to few-group cross-section uncertainties (few-group covariance matrix). The other input uncertainties in Exercises I-2 are new uncertainties added during the cross-section generation process.
In order to propagate the input uncertainties through lattice physics calculations to determine uncertainties in output lattice-averaged parameters within the framework of Exercise I-2 the utilization of a lattice physics code is necessary. Participants can use/select their own lattice physics codes in conjunction with their own UA and SA tools for the purposes of this exercise.
Different stand-alone neutronics single assembly and mini-core test problems have been designed for the purposes of the Exercise I-2 utilizing information from the previous OECD coupled code benchmarks. Continuous Monte Carlo (MCNP5) solutions with sufficient statistics to assure not only \( k_{\text{inf}} \) (\( k_{\text{eff}} \)) but also fission source convergence will be used as reference solutions. In addition, the KRITZ-2 [17] LEU mini-core test cases and the VVER physics experiments [18] (for which there is available experimental data) are utilized.

The output uncertainty of Exercise I-2 is propagated in Exercises I-3, II-3, III-1 and III-3. The major effort is focused on obtaining uncertainties in two group homogenized parameters based on the standard two-group structure (with 0.625 eV cut-off) utilized in the LWR industrial calculation scheme. Provision for few group (more than two energy group) homogenized parameters and associated uncertainties with selected by participants few-group structures is made.

3.3 Exercise I-3: Core Physics

In the current established calculation scheme for LWR design and safety analysis the lattice averaged (homogenized) few-group cross-sections are an input to core calculations. The few-group cross-section uncertainties (few-group covariance matrix) should be obtained by participants as output uncertainties within the framework of Exercise I-2. In Exercise I-3 the few-group cross-section uncertainties are input uncertainties and must be propagated to uncertainties in evaluated stand-alone neutronics core parameters. The other input uncertainties in Exercises I-3 are new uncertainties added during the core stand-alone calculations.

Understanding the uncertainties in key output reactor core parameters associated with steady state core simulation is important in regard to introducing appropriate design margins and deciding where efforts should be directed to reduce uncertainties. The propagation of the input uncertainties through core calculations to determine uncertainties in output core parameters within the framework of Exercise I-3 requires utilization of a core simulator code. Participants can use/select their own core simulator codes in conjunction with their own UA and SA tools for the purposes of this exercise.

Three-dimensional (3-D) test problems on two different levels are defined to be used within Exercise I-3: HZP core test cases (for which the continuous energy Monte Carlo method is used for reference calculations), and documented experimental benchmark plant cold critical data and critical lattice data.

In summary this exercise is focused on stand-alone neutronics core calculations and associated prediction uncertainties. It does not analyze uncertainties related to cycle and depletion calculations. No feedback modelling is assumed, thus it will address the propagation of uncertainties associated with few-group cross-section generation but not cross-section modelling i.e. methodologies used for cross-section parameterization as a function of history and instantaneous variables. The output uncertainty of Exercise I-3 is propagated in Exercises II-1, II-2, II-3, and III-1 and III-3.

4 PRIORITIES OF PHASES II AND III

The three exercises within Phase I follow the established in the industry and regulation routine calculation scheme for LWR design and safety analysis. The Phase I is focused on understanding uncertainties in the prediction of key reactor core parameters associated with LWR stand-alone neutronics core simulation. Such uncertainties occur due to input data uncertainties, modelling errors, and numerical approximations. Understanding the uncertainties in key output reactor core parameters associated with steady state core
simulation is important with regard to introducing appropriate design margins and deciding where efforts should be directed to reduce uncertainties. The obtained output uncertainties from Phase I of the OECD LWR UAM benchmark will be utilized as input uncertainties in the remaining two phases – Phase II (Core Phase) and Phase III (System Phase).

A new VVER-1000 coupled code benchmark based on the Kalinin-3 plant was proposed [19]. The available data are numerous, accurate and well documented. They include a MCP switch-off transient at nominal power. The Kalinin-3 plant differs from the Kozloduy-6 plant used in the V1000CT benchmark only by the core loading and the control rods. To enhance the value of this benchmark within the UAM activities for the VVER-1000 track, it is suggested to use the Kalinin-3 plant data for Phase I stage (in addition or instead of Kozloduy-6). Hence, uncertainties could be evaluated on the neutronics cross-sections and then propagated in the coupled thermal-hydraulic/neutronics calculations of the plant (steady-state and transient conditions).

Phase II will address core neutron kinetics, thermal-hydraulics and fuel performance, without any coupling between the three physics phenomena. Phase III will include system thermal-hydraulics and coupling between fuel, neutronics and thermal-hydraulics for steady-state, depletion and transient analysis. A strict evaluation of the results obtained in Phases I and II will be made before entering Phase III. Output parameters and targets were proposed and discussed for Exercises of Phases II and III for the three main types of LWR selected in UAM (PWR, BWR and VVER). They will be taken into account for preparing the draft Specifications of Phase II (and later Phase III). For Exercise II-1 (fuel thermal properties relevant for transient performance) the available experimental data can be identified from the CRISSEUE-S data base [20]. The available kinetics experiments (Exercise II-2) for the three LWR types will be searched by the benchmark team with the help of the participants. For fuel bundle thermal-hydraulics (Exercise II-3), data are available for the BWR bundle type (from the OECD/NRC BFBT benchmark [21]), but there is a need for VVER bundle data (the benchmark data form the PSB facility will be examined for this purpose) and for PWR bundle data (the possibility to obtain such data from the NUPEC data base will be explored). A first list of transients to be considered for Phase III is proposed: turbine trip transient, rod ejection accident, main steam line break transient, pump trip or start-up transients, and LOCA. The transient scenarios will be defined for the three main LWR types selected for the UAM benchmark.

The benchmark team will prepare Specifications for each Phase in order to allow participation in the full Phase or only in a subset of the Exercises or in a separate Exercise. Boundary conditions and necessary input information will be provided by the benchmark team. Each organization interested in the UAM benchmark has to identify its own objectives and priorities. In particular for the preparation of Phases 2 and 3, it might be necessary to rank the priorities between the reactor types or the transients to be analyzed.

The Third Workshop of the OECD LWR UAM Benchmark (UAM-3) will be hosted by PSU, USA on April 29 – May 1 2009. The objectives of the UAM-3 workshop will be the following:

a) Discussion of submitted results of Phase I;
b) Discussion of draft Specification for Phase II;
c) Discussion of priorities for Phase III.

5 OUTLOOK AND CONCLUSIONS

It is expected that the application of coupled codes for safety analyses will be continuously growing. In fact, they are the only means to perform best-estimate calculations for accident conditions with a tight coupling of neutronics and thermal-hydraulics effects. The current tendencies in coupled code developments are towards systematic integration of
uncertainty and sensitivity analysis with simulations for safety analysis. Sensitivity and
uncertainty analysis capabilities must be further developed for comprehensive coupled code
simulations with nonlinear feedback mechanisms as well as tested for uncertainty propagation
through multi-physics multi-scale calculations on comprehensive benchmark frameworks as
the described in this paper OECD LWR UAM benchmark.

In this project, for the first time, uncertainties are propagated through the whole process
from microscopic cross-sections to plant transients on a unified benchmark framework to
provide credible coupled code predictions with defensible uncertainty estimations of safety
margins at the full core/system level. The OECD LWR UAM benchmark framework is
expected to help formulating recommendations and guidelines on how to utilize advanced and
optimized sensitivity analysis and uncertainty analysis (SA/UA) methods in “best estimate”
reactor simulations in licensing practices.

This project is challenging and responds to needs of estimating confidence bounds for
results from simulations and analysis in real applications. It will create the favourable
environment for the development of these methods and their use and become a standard. In
order to achieve this, the UAM scientific board members recommended that research
organizations and institutions reserve the necessary funds to support this activity and that an
uncertainty analysis culture is developed in nuclear engineering.

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Influence of Modelling Options in RELAP5/SCDAPSIM and MAAP4 Computer Codes on Core Melt Progression and Reactor Pressure Vessel Integrity

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ABSTRACT

RELAP5/SCDAPSIM and MAAP4 are two widely used severe accident computer codes for the integral analysis of the core and the Reactor Pressure Vessel (RPV) behaviour following the core degradation. Additionally, MAAP4 code has the capability to simulate containment response after release of the primary system inventory. MAAP4 is a fast-running parametric code that produces credible results although it utilizes the simplified form of the conservation equations and the coarser discretization of the reactor systems. RELAP5/SCDAPSIM code is a best-estimate code that contains more mechanistic physics models than the other codes for both severe accident and thermal-hydraulic phenomena.

The objective of the paper was to compare code results by applying different modelling options and to evaluate influence of thermal-hydraulic conditions on core damage progression. The intention was not general validation of codes but examination of code results for the same specific transient by using qualified input models. The analysed transient was postulated Station Blackout (SBO) in NPP Krško with a leakage from Reactor Coolant Pumps (RCP) seals. Two groups of runs were performed where each group had a different break area and thus a different leakage rate from the Reactor Coolant System (RCS). Break flow areas differed by a factor of two. The target results were the time when the molten core material slumps to the lower head and the time when the lower head wall fails due to interaction with the molten material.

Parameters which affect core melt accident progression and debris bed behaviour in the lower head were discussed and their impact on RELAP5/SCDAPSIM and MAAP4 results were presented. Integrity of the RPV lower head was assessed using a) COUPLE code that is a two-dimensional, finite element heat conduction code incorporated in the RELAP5/SCDAPSIM code, and b) the relevant MAAP4 lower head models.
1 INTRODUCTION

RELAP5/SCDAPSIM and MAAP4 codes are Severe Accident (SA) analysis codes capable of modelling all important SA phenomena (reactor coolant system response, core material chemical reactions, oxidation, ballooning and rupture of the fuel rod cladding, core heat-up, degradation and relocation to the lower plenum, etc.). The main difference between them is that RELAP5/SCDAPSIM can only model the in-vessel phase of the SA, while MAAP4 is capable to calculate processes in the containment following the release of water, non-condensable gases and corium from the primary circuit.

RELAP5/SCDAPSIM code, ref. [1], designed to predict behaviour of reactor systems during normal and accident conditions, is being developed at Innovative Systems Software (ISS) as part of the international SCDAP Development and Training Program (SDTP). RELAP5/SCDAPSIM uses the publicly available SCDAP/RELAP5 models developed by the US Nuclear Regulatory Commission in combination with proprietary (a) advanced programming and numerical methods, (b) user options, and (c) models developed by ISS and other members of the SDTP. The code is a combination of RELAP5 code for thermal-hydraulics calculation, SCDAP code for severe accident related phenomena and COUPLE code for a finite element treatment of the vessel lower head.

MAAP4 code is an integral system analysis code for assessing severe accidents in Light Water Reactors (LWR) following large and small break Loss of Coolant Accidents (LOCA) and transients. The code was developed for Electric Power Research Institute (EPRI) by Fauske et al., ref. [2]. MAAP4 employs “generalized models” for BWRs and PWRs in which the type and number of components and the geometry are predetermined. The user inputs various parameters for each component such as volumes or masses.

MAAP4 version used herein was MAAP4.0.5.

RELAP5/SCDAPSIM is characterized by its detailed, mechanistic models of severe accident phenomena; however, the calculations can be rather time-consuming. RELAP5/SCDAPSIM typically uses on the order of hundreds of hydrodynamic components to model the primary system. MAAP4 calculations require minimal computation time with simplified geometry models.

Regarding the thermal-hydraulic model, RELAP5/SCDAPSIM employs detailed RELAP5/Mod3 non-equilibrium, non-homogenous, six-equation representation of single and two-phase flows. The presence of boron and non-condensable gases is also simulated using separate equations for each. The robust RELAP5 modelling is clearly superior to the thermal-hydraulics model of MAAP4 code. MAAP4 has simplified but fast-running models for thermal hydraulics description using a fixed nodalization of the primary circuit. MAAP4 solves a set of lumped parameter, first-order differential equations for conservation of mass and energy. Differential equations for momentum conservation are not employed because MAAP4 considers momentum balances to be quasi-steady, which reduces the momentum equations to algebraic equations, ref. [3].

2 CODE MODELS FOR NPP KRŠKO

2.1 RELAP5/SCDAPSIM Model

The RELAP5/SCDAPSIM model of NPP Krško (NEK) was based on the RELAP5/Mod3.3 model described in ref. [4] and qualified on the steady-state level, ref. [5]. That model has been used for many years now at FER (Faculty of Electrical Engineering and Computing), Zagreb for accident analyses of plant behaviour following large spectrum of initializing events in all modes of operation. NEK RELAP5/Mod3.3 nodalization scheme is
shown in Figure 1. Such detailed nodalization was based not solely on plant geometrical data, but it also took into account the operating systems and the operating conditions.

Figure 1: NEK RELAP5/Mod3.3 nodalization scheme

For the purpose of simulation of core melt progression during a postulated severe accident, core fuel assemblies were divided in five regions by grouping similarly powered fuel assemblies together, Figure 2. Accordingly, to apply correct thermal-hydraulic boundary conditions for the fuel rods, five thermal-hydraulic channels in the core were modelled as well.

Figure 2: Radial cross-section of the NEK core
When a portion of the core has melted it may occupy completely the flow channel and therefore block the coolant flow in the axial direction. The flow will be then diverted in the radial direction. To enable the coolant to flow also in the radial direction, hydraulic channels in the core were interconnected radially by crossflow junctions.

2.2 MAAP4 Model

The NEK MAAP4 model, ref. [6], was used for the development of NEK Severe Accident Management Guidelines (SAMG) and Krško Full Scope Simulator (KFSS) which simulates various severe accident sequences. Krško SAMGs have been developed based on the Krško IPE (Individual Plant Examination) insights, generic WOG SAMGs (Westinghouse Owners Group SAMGs, ref. [7]) and plant specific documents, refs. [8], [9] and [10]. MAAP4 code has been used in the Krško IPE to determine success criteria for the accident sequences.

All deterministic thermal-hydraulic analyses within the NEK IPE project were performed by MAAP3B, Ver. 18, ref. [11]. Since then (end of 1994) that analysis tool has been further improved and new versions have been issued. The version MAAP4.0.5 was used in the presented analyses. MAAP3B was revised to include major model improvements in areas of core heat-up, lower plenum phenomenology, corium-concrete interactions, containment and auxiliary building thermal-hydraulics and hydrogen combustion. Furthermore, new models were added to characterize actions that could stop the accident, i.e. the in-vessel and the ex-vessel cooling. The mathematical solution techniques were implemented to maintain a quick-running code suitable for extensive accident screening and parameter sensitivity applications. As a part of the development, the code underwent a complete design review.

3 ANALYSIS AND RESULTS

3.1 Description of the Scenario

The analyzed accident was Station Blackout (SBO) with a leakage from the reactor coolant system through RCP seals following their degradation. It was assumed that both off-site and on-site (emergency diesel generators) AC power was unavailable. Therefore, the primary system coolant inventory was decreasing due to the unavailability of the High Pressure (HPSI) and the Low Pressure Safety Injection (LPSI) flow. Water was only injected from the accumulators because their operation did not depend on the availability of electrical power. Steam generators (SG) acted as a heat sink since the Turbine Driven Auxiliary Feedwater (TDAFW) pump delivered water to SGs. TDAFW flow was controlled in a manner to maintain the SG narrow range level between 10 % and 50 %. Those values are in compliance with ECA-0.0 "Loss of all AC power" procedure, ref. [12].

Two groups of runs were performed (Table 1) where each group had a different break area and thus a different leakage rate from the RCS. For the first group the break area was taken from the WOG 2000 RCP Seal Leakage Model, ref. [13]. In WOG 2000 Model series of discharge rates with their respective probabilities are defined. For our case the scenario with the highest leakage rate was chosen: the break area was $10^5$ m$^2$ for the first 780 s and it increased afterwards to $2.5 \times 10^{-4}$ m$^2$. For the second group the break area was two times higher ($5 \times 10^{-4}$ m$^2$), but it remained constant throughout the transient. The reason of performing calculations with different break flow areas was to check the influence of thermal-hydraulic conditions on accident progression. The smaller break area means slower depressurization of the RCS and later actuation of the accumulators, thus the earlier core dry-out. The larger break area means an earlier loss of RCS coolant which will force accumulators’ actuation in
the initial phase of the accident. The specific differences between those two cases will be discussed in separate sections.

<table>
<thead>
<tr>
<th>Break area</th>
<th>Group 1</th>
<th>Group 2</th>
</tr>
</thead>
<tbody>
<tr>
<td>0-780 s:</td>
<td>10^{-3} m^2</td>
<td></td>
</tr>
<tr>
<td>&gt;780 s:</td>
<td>5\cdot10^{-4} m^2</td>
<td>2.5\cdot10^{-4} m^2</td>
</tr>
</tbody>
</table>

3.2 Modelling of the Hot Leg Natural Circulation

After the hot legs have been voided and prior to the loop seal clearing, a countercurrent natural circulation between the reactor vessel, hot legs and steam generator U-tubes may develop, ref. [14]. Superheated vapour enters the top of the hot leg displacing saturated vapour, which then flows back to the reactor vessel along the bottom of the hot leg. When hotter vapour enters the steam generator inlet plenum, it will rise toward the steam generator U-tubes. Vapour enters some of the tubes, displacing cooler steam that was in the tubes. Displaced vapour enters the outlet plenum, then reenters other steam generator tubes, forcing vapour into the inlet plenum. A density gradient is thus established between tubes which supports the natural circulation. Once the loop seals are cleared the natural circulation will be terminated.

Hot leg countercurrent flow can affect the structural integrity of the RCS piping. Heating of the pipes and steam generator tubes may lead to melting and creep rupture failure of those components.

Hot leg natural circulation model is an integral part of the MAAP4 code and no specific rearrangement of the input deck is needed to invoke that model. On the other hand, to allow the hot leg countercurrent flow in RELAP5/SCDAPSIM code, the primary system should be renodalized in a manner to split the hot legs and U-tubes as shown in Figure 3. In addition, the loss coefficients need to be adjusted to correctly simulate mixing in the SG inlet plenum. Preliminary runs with the new model did not reveal any increase of the creep failure probability of the RCS piping, contrary to the findings of MAAP4 calculations. The analyzed accident was a Station Blackout (SBO) with a high leakage from the RCS through RCP seals. In such an accident the loop seals will clear early during the transient so the impact of the hot leg natural circulation will not be so significant. Nevertheless, in MAAP4 calculations the hot leg creep failure did occur so this phenomenon can not be ruled out. Taking this into account, two different MAAP4 runs were performed, one with the hot leg creep failure and one without it. That was necessary in order to make correct comparison between the two codes.
3.3 Group 1 Cases

3.3.1 MAAP4 Calculation

Two MAAP4 calculations were performed: one that takes into account and the other that does not take into account the creep rupture of the hot leg piping. Those two calculations were performed to examine the influence of the hot leg piping failure on core damage progression and also to make a comparison with RELAP5/SCDAPSIM results. Namely, contrary to MAAP4 calculation, RELAP5/SCDAPSIM did not calculate the creep failure of the hot leg so to compare correctly the two codes creep failure had to be turned off in the MAAP4 calculation.

Hot leg creep failure at 10300 s caused primary system pressure to decrease rapidly (Figure 4). When pressure dropped to ~5 MPa accumulators started to inject water in the RCS. In the case of no hot leg creep failure accumulators gradually injected water in small intervals because the primary system depressurized at a slow rate. More water was injected in the core when the hot leg failed because all water inventory from the accumulators was discharged into the RCS. Therefore, the core was cooled more successfully and the RPV wall breached later following the relocation of molten corium in the lower plenum.

Figure 5 shows the core maximum temperature for the two cases. The core started to melt in the center and later the process of melting progressed toward the core baffle. Once the core baffle failed, corium slumped through the bypass region between the core baffle and the core barrel in the RPV lower head. In the case with no hot leg creep rupture the core baffle failed at 13800 s, and when the hot leg rupture was taken into account the baffle failed at 17400 s. Accordingly, the RPV wall failed later in the second case.
3.3.2 Influence of Oxide Shell Stability Parameters

One important set of parameters that can be found in both codes and which can have significant influence on the results is the set of oxide shell stability parameters. That set consists of two parameters. The first parameter is the temperature for failure of the oxide shell on the outer surface of the fuel rod cladding. The second parameter is the fraction of oxidation of the clad for the stable oxide shell (FZORUP). If the extent of oxidation of the cladding
is greater than this value, then the oxide shell does not fail even if the cladding surface temperature exceeds the failure temperature. Those parameters were introduced because the mechanistic models in the two codes cannot calculate precisely the temperature at which cladding fails if it is only partially oxidized. If the cladding is 100% oxidized, or if it is not oxidized at all, the cladding will fail when the melting temperature of ZrO$_2$ in former, or metallic Zircaloy in later case, is reached. Otherwise, one has to specify the above mentioned parameters.

It is suggested to use 2500 K, refs. [1] and [2], as the temperature for failure of the oxide shell, and 0.2 as the fraction of oxidation of the cladding for the stable oxide shell. The value of 0.2 was chosen based on TMI-2 benchmark calculations.

Two sensitivity runs were performed in which only the value of FZORUP was varied, while the temperature was set to 2500 K. In one run the value of FZORUP was set to 0.2 and in the other run it was set to 0.675. The value of 0.675 was chosen because it represents 50% of Zircaloy oxidation.

Figures Figure 6 and Figure 7 show accumulated hydrogen mass and total molten mass of core material for MAAP4 runs. Accumulated hydrogen mass and equivalent radius of the in-core molten pool for RELAP5/SCDAPSIM runs are shown in Figures Figure 8 and Figure 9. When FZORUP was set to 0.2 the oxide shell was more stable because smaller fraction of the cladding was needed to be oxidized to keep the cladding intact. It might be concluded that the core would be longer intact in that case and that the RPV wall would fail later. But the results showed different behaviour. A smaller amount of hydrogen was produced in the case of higher value of FZORUP because more cladding was removed from its original location to the lower elevation where temperature was lower and so was the oxidation rate. Also, the amount of molten material was smaller. The reason for that was the reaction between molten Zircaloy and UO$_2$ fuel. Molten Zircaloy dissolves UO$_2$ at temperature much lower than the melting temperature of UO$_2$, thus, the reaction is very harmful regarding the core integrity. The higher value of FZORUP means that more Zircaloy would be removed and so its reaction with UO$_2$ would be prevented. UO$_2$ would not fail until its melting temperature was reached. The smaller value of FZORUP means that even a small amount of oxidized Zircaloy is enough to prevent the cladding to rupture. Metallic Zircaloy inside the oxide shell would melt and dissolve UO$_2$ causing an earlier core meltdown and RPV failure.

![Figure 6: Accumulated hydrogen mass, MAAP4 calculation, Group 1](image-url)
Figure 7: Total molten mass of core material, MAAP4 calculation, Group 1

Figure 8: Accumulated hydrogen mass, RELAP5/SCDAPSIM calculation, Group 1

Figure 9: Equivalent radius of the in-core molten pool, RELAP5/SCDAPSIM calculation, Group 1
3.3.3 Comparison Between MAAP4 and RELAP5/SCDAPSIM Calculation

Figures Figure 10 - Figure 13 show different variables as calculated by the two codes. MAAP4 run did not take into account possibility of the hot leg creep rupture in order to be consistent with RELAP5/SCDAPSIM calculation. Parameter FZORUP was set to 0.2 as recommended by code developers.

A good agreement in primary system pressure (Figure 10) between the two codes is apparent which means that both codes calculated the same rate of coolant discharge from the RCS. That was achieved by setting the discharge coefficient to the same value for both code runs. Core maximum temperature (Figure 11) was also pretty well reproduced. A larger discrepancy was observed in the oxidation power (Figure 12) and consequential hydrogen production (Figure 13). Small variations in primary system pressure around 5 MPa influenced water injection rate from the accumulators and subsequently oxidation kinetics because water injected from the accumulators was evaporating in the core causing the cladding to oxidize. At time when the accumulators began to operate the core was at a very high temperature and as soon as water entered the core it evaporated quickly. Peaks in the diagram of the oxidation power happened shortly after the accumulators injected certain amount of water in the RCS.

![Figure 10: RCS pressure, Group 1](image1)

![Figure 11: Core maximum temperature, Group 1](image2)
A comparison between few important parameters is presented in Table 2.

Table 2: RELAP5/SCDAPSIM and MAAP4 calculation results, Group 1

<table>
<thead>
<tr>
<th>Parameter</th>
<th>RELAP5/SCDAPSIM</th>
<th>MAAP4</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mass of produced hydrogen</td>
<td>240 kg</td>
<td>290 kg</td>
</tr>
<tr>
<td>Mass of core molten material</td>
<td>15700 kg</td>
<td>5900 kg</td>
</tr>
<tr>
<td>Time of core baffle failure</td>
<td>9300 s</td>
<td>11500 s</td>
</tr>
<tr>
<td>Time of RPV failure</td>
<td>9500 s</td>
<td>19900 s</td>
</tr>
</tbody>
</table>

RELAP5/SCDAPSIM calculates earlier core degradation than MAAP4 code. Since both codes use eutectic U-Zr-O model for evaluation of core melt kinetics, the reason for this discrepancy might be the different thermal-hydraulic model adopted in the codes. But, since the core degradation process is stochastic in its nature and depends on large variety of
phenomena, it can be concluded that both codes produce reasonable results for core melt
time.

About 22% of total core material was calculated to liquefy and to slump to the lower
plenum by RELAP5/SCDAPSIM code and 8% by MAAP4 code. As already stated, the
reason for higher core molten mass in RELAP5/SCDAPSIM code was that
RELAP5/SCDAPSIM calculates earlier core degradation than MAAP4 code.

RPV lower plenum temperature distribution was calculated using on one hand the
COUPLE code, a two-dimensional, finite element heat conduction code incorporated in the
RELAP5/SCDAPSIM code and on the other hand a much simpler MAAP4 RPV lower head
model. The COUPLE model of the NEK lower head is shown in Figure 14.

![COUPLE model of the NEK RPV lower head](Figure 14)

Time of the RPV failure was evaluated using Larson-Miller parameter model, ref. [15],
for the creep rupture which is incorporated in both codes. It can be seen from Table 2 that this
time differs in order of magnitude for the two codes. The reason is the difference in the
application of the creep rupture model, ref. [16]. The creep rupture in RELAP5/SCDAPSIM
was calculated using the average wall temperature, while in MAAP4 the creep damage term
was evaluated at each temperature node. Each of these two approaches has its benefits and
weaknesses, but in the scope of the presented analysis, it was not so essential to correctly
calculate time of the RPV failure but more important was to give the answer to question will
the RPV resist molten corium attack or not. Since in the analyzed case no SI was available
and no cavity flooding was provided, RPV finally failed opening a path for fission products to
escape into the containment.

3.4 Group 2 Cases

3.4.1 MAAP4 Calculation

Similar to Group 1 runs, two MAAP4 calculations were performed: one that takes into
account and the other that does not take into account the creep rupture of the hot leg piping.

Hot leg failed due to creep at 5400 s, 5000 s earlier than in the case with a smaller break
area. Accumulators were actuated prior to the hot leg failure and continuously injected water
afterwards. Hot leg break caused primary system pressure to decrease rapidly (Figure 15). The
consequence was that accumulators drained out almost immediately. In the short term that
was positive because injection of large amount of water forced the core temperature to drop to
almost 1600 K (Figure 16). On the other hand, in the longer term, there was no more water
available for core quenching once the temperature increased again. In the case of no hot leg
core failure, accumulators gradually injected water but contrary to the run with the smaller
break area, they injected more water which was enough to keep the core temperature below the UO$_2$ – ZrO$_2$ eutectic temperature. The higher water injection rate was due to the larger pressure drop in the RCS.

For the reasons mentioned above, in the case with no hot leg creep rupture the core baffle failed at 28800 s and when the hot leg rupture was taken into account the baffle failed at 13300 s. Thus, the result was opposite comparing to the previous two runs. Although the analyzed transients were not representative for the real plant situations they show that thermal-hydraulic phenomena dictate the SA progression. Correct thermal-hydraulic modelling of the plant systems is therefore essential for a SA simulation, especially in the longer term.

![RCS pressure, MAAP4 calculation, Group 2](image1)

Figure 15: RCS pressure, MAAP4 calculation, Group 2

![Core maximum temperature, MAAP4 calculation, Group 2](image2)

Figure 16: Core maximum temperature, MAAP4 calculation, Group 2
3.4.2 Comparison Between MAAP4 and RELAP5/SCDAPSIM Calculation

Comparing to the case with the smaller break area, discrepancy between the results is more pronounced. Whereas the calculated RCS pressure (Figure 17) is similar for the two codes, the core maximum temperature (Figure 18), oxidation power (Figure 19) and produced hydrogen (Figure 20) vary significantly.

![Figure 17: RCS pressure, Group 2](image)

![Figure 18: Core maximum temperature, Group 2](image)
Figure 19: Oxidation power, Group 2

Figure 20: Accumulated hydrogen mass, Group 2

MAAP4 calculated that more than two times more hydrogen would be produced comparing to RELAP5/SCDAPSIM calculation. It is interesting to notice that this hydrogen was produced in the early stage of the transient when the accumulators started to operate. Injection of water in the hot core resulted in high oxidation rate. Energy released during oxidation accumulated mostly in the core material and the core temperature rose swiftly (Figure 18). Amount of the oxidation energy transferred to fluid was very small compared to total released energy so it did not affect the RCS pressure (Figure 17). At the same time RELAP5/SCDAPSIM predicted that water flow rate through the core would be enough to
quench the core and prevent oxidation escalation, but once the core began to dry out its temperature increased rapidly due to heat release during the oxidation.

After the oxidation escalation and increase of core temperature to more than 3000 K, MAAP4 calculated successful quenching of the core. Decrease of temperature prevented any significant core damage. On the other hand, when the core temperature reached ~2800 K in the RELAP5/SCDAPSIM calculation no quenching onward was possible. The core started to melt and the process of core degradation proceeded rapidly with no possibility for stopping it. The consequence was that mass of molten core material as calculated by RELAP5/SCDAPSIM code was seven times higher than mass calculated by MAAP4 code.

The heat released during the oxidation in MAAP4 calculation in the short period when the accumulators were turned on was later successfully removed. RELAP5/SCDAPSIM calculated that the energy accumulated in the fuel due to oxidation was too high to be removed by water injected from the accumulators only. The ECCS system is therefore necessary to be operable in order to prevent core damage when looking the RELAP5/SCDAPSIM results. On contrary, MAAP4 results indicate that water from the accumulators is enough to cool the core and that SI pumps are not needed in the early phase of a transient.

A comparison between some results is presented in Table 3.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>RELAP5/SCDAPSIM</th>
<th>MAAP4</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mass of produced hydrogen</td>
<td>110 kg</td>
<td>270 kg</td>
</tr>
<tr>
<td>Mass of core molten material</td>
<td>15100 kg</td>
<td>2000 kg</td>
</tr>
<tr>
<td>Time of core baffle failure</td>
<td>9600 s</td>
<td>29000 s</td>
</tr>
<tr>
<td>Time of RPV failure</td>
<td>9800 s</td>
<td>34600 s</td>
</tr>
</tbody>
</table>

There are many reasons for large differences in the results. First of all, models included in codes are different; not only thermal-hydraulic models, but also models that simulate core damage progression. RELAP5/SCDAPSIM is mechanistic code, while MAAP4 is parametric code that uses more phenomenological models. The analyzed transient was quite demanding and necessitated careful preparation of the input data and initial and boundary conditions. Due to those reasons the uncertainties of the input data were not taken into account because the differences between code predictions could not be covered by the uncertainties only.

### 3.5 Influence of Some User-Selected Parameters

Severe accident codes generally use a large set of user-defined parameters to cover the uncertainty in analytical description of SA processes. MAAP4 code is regarded as a parametric code because its results rely on proper selection of many of parameters describing phenomena starting from an early stage of core degradation to the late stage of containment fission products behaviour. RELAP5/SCDAPSIM code utilizes more mechanistic models and user-defined parameters are primarily used in sensitivity calculations.

One of MAAP4 parameters which significantly influences the results is the porosity of core material below which the flow area and the hydraulic diameter of core nodes become zero. Setting that parameter to a higher value resulted in earlier core degradation because less material relocation was needed for a core node to become a crust node resulting in shorter times for oxidation and heat transfer. The user has also the possibility to set the size of the in-core crust opening after the failure of the in-core molten pool side crust such that sideward relocation of corium to the lower head is possible. Following the relocation the user has to specify the minimum and the maximum crust thickness surrounding corium in the lower head.
that forms during the solidification of molten material. Sensitivity calculations revealed that influence of those two parameters was not as significant as the influence of the first parameter (porosity of core material).

Influence of few RELAP5/SCDAPSIM parameters was also tested. 1) The rate of melting of the core baffle in contact with the molten material, 2) selection of a model that results in the earliest or the latest possible slumping of corium to the lower head and 3) the porosity of particles in the lower head were some of the tested parameters. It was found that their influence on core degradation kinetics and the timing of the lower head failure was negligible. Nevertheless, the user influence on RELAP5/SCDAPSIM results cannot be ruled out because correct modelling requires a qualified and experienced user with knowledge of plant systems and their interconnections.

4 CONCLUSION

Capabilities of RELAP5/SCDAPSIM and MAAP4 codes in simulating in-core severe accident progression were compared focusing on influence of thermal-hydraulics and selection of user-defined parameters. The input decks for both codes were prepared taking into account the actual geometric and operational data of NPP Krško making them qualified for a comprehensive and systematic analysis.

The correct prediction of RCS thermal-hydraulic behaviour is important for the later SA progression. Oxidation rate and heat-removal from the core are dictated by availability of coolant, thus production of hydrogen and core heat-up and degradation depend primarily on RCS thermal-hydraulics. Calculations have shown that MAAP4 results are more sensitive to RCS pressure variations and coolant discharge rate from the break. Increase of the break area substantially affected the timing of both core melt process and failure of the RPV lower head. RELAP5/SCDAPSIM, on contrary, regardless the size of the break area predicted RPV damage almost at the same time, meaning that in the case of a large break, whatever the size of the break actually was, core would lose its geometry early during the transient unless mitigation measures are undertaken before.

A number of user-defined parameters, especially in MAAP4 code, have to be entered during preparation of an input deck. While in RELAP5/SCDAPSIM code these parameters are used only in sensitivity calculations, in MAAP4 code their selection could significantly alter the results. The solution is to use parameter values recommended by MAAP4 developers or ones which are experimentally measured and confirmed.

The comparison between the codes showed some similar trends, but also large disagreements in the obtained results. Code validation against plant and experimental data is therefore a necessary tool in testing the code’s accuracy. Nevertheless, both codes showed capability of modelling complex interactions between core materials and overall core behaviour during harsh severe accident conditions.

REFERENCES


Main results of Phase IV BEMUSE project. Simulation of LBLOCA in a NPP

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ABSTRACT

Phase IV of BEMUSE Program is connected with previous phase II, being a necessary step for an uncertainty analysis: the simulation of the reference scenario and sensitivity analysis. Present phase analyses a LBLOCA scenario in Zion 1 NPP, a 4 loop PWR unit. Thirteen participants coming from ten different countries have taken part in the Phase IV of the program.

The BEMUSE (Best Estimate Methods plus Uncertainty and Sensitivity Evaluation) Program has been promoted by the Working Group on Accident Management and Analysis (GAMA) and endorsed by the Committee on the Safety of Nuclear Installations (CSNI).

The paper presents the results of the calculations performed by participants and emphasizes its usefulness for future uncertainty evaluation, to be performed in next phase. The objectives of the activity are basically to simulate the LBLOCA reproducing the phenomena associated to the scenario and also to build a common, well-known, basis for the future comparison of uncertainty evaluation results among different methodologies and codes.

Following activities performed in phase II, a list of sensitivity calculations was proposed and performed by participants to study the influence of different parameters such as material properties or initial and boundary conditions, upon the behaviour of the most relevant parameters related to the scenario.

1. INTRODUCTION

Models and codes are an approximation of the real physical behaviour occurring during a hypothetical transient and the data used to build these models are also known with certain accuracy. Therefore code predictions are uncertain. The BEMUSE programme is focussed on the application of uncertainty methodologies to large break LOCAs. This introduction deals with some background considerations and establishes the objectives of the programme along with its steps and phases.
1.1. Background

One of the goals of computer code models of Nuclear Power Plants (NPP) is to demonstrate that these are designed to respond safely at postulated accidents. To deal with uncertainties the analyses can either use conservative or best-estimate (BE) codes:

- The conservative codes contain assumptions to try to cover not known uncertainties. These assumptions are often unphysical and lead to predictions that could be worse than reality.
- BE codes are designed to model all the relevant processes in a physically realistic. A calculation with a BE code is then considered the best approach of what is more likely to occur. In any case it is necessary to evaluate the uncertainty of the estimation.

The reasons and motivation for using BE codes has been explained in many occasions [1], [2], [3]. The OECD BEMUSE started with the aim of achieving a deeper understanding of such methods [4].

1.2. Objectives

The BEMUSE programme is focussed on the application of uncertainty methodologies to large break LOCAs. The objectives of this programme are:

- To evaluate the practicability, quality and reliability of best-estimate methods including uncertainty evaluations in applications relevant to nuclear reactor safety.
- To develop common understanding.
- To promote/facilitate their use by the regulator bodies and the industry.

Using the same codes and similar methods should allow comparing the potential important uncertain parameters and the effects of different modelling for uncertainties can be evaluated. Therefore, the assessment of each methodology by comparison with experimental data is also one of the purposes of the programme.

1.3. Steps and phases

The BEMUSE program is divided in two steps. The first step is to perform an uncertainty and sensitivity analysis of LOFT L2-5 test calculations and the second is to perform this analysis for a NPP-LB.

Each of these two steps is made up of three phases:

- First step (Phases 1, 2 and 3):
  - Phase 1: presentation a priori of the uncertainty evaluation methodology to be used (lead organisation: IRSN)
  - Phase 2: re-analysis of the ISP-13 exercise, post-test of LOFT L2-5 test (lead organisation: University of Pisa)
- Phase 3: uncertainty evaluation of the L2-5 test calculations (lead organisation: CEA)

• Second step (Phases 4, 5 and 6):
  - Phase 4: best-estimate analysis of an NPP-LBLOCA (lead organisation: UPC)
  - Phase 5: uncertainty evaluation of the NPP-LBLOCA (lead organisation: UPC)
  - Phase 6: status report, conclusions and recommendations (lead organisation: GRS)

2. LESSONS LEARNED FROM PREVIOUS PHASES

Participants to Phase II achieved significant results. Almost all performed calculations appear qualified against the fixed criteria and few mismatches between results and acceptability thresholds have been characterized. Dispersion bands of results appear substantially less than years ago in ISP-13. Modelling techniques used by participants are the most fruitful outcome of phase II to be used in phase IV analysis.

The Input/Output Specification of Phase IV has been prepared by the coordinator team taking into account achievements and recommendations basically of Phase II but also of phases I and III.

3. PHASE IV SCOPE AND OBJECTIVES

The scope of Phase IV of BEMUSE programme is the simulation of a LB–LOCA in a Nuclear Power Plant using experience gained in previous Phase II [5]. Calculation results will be the basis for uncertainty evaluation, to be performed in next phase.

The objectives of the activity are:

• To simulate a LB–LOCA reproducing the phenomena associated to the scenario.

• To have a common, well-documented basis for the execution of the uncertainty evaluation step in Phase V.

4. PLANT AND SCENARIO

The selected plant was Zion Station a dual-reactor nuclear power plant operated and owned by the Commonwealth Edison network. No other options were available. This power generating station is located in the extreme eastern portion of the city of Zion, Lake County, Illinois. It is approximately 40 direct-line miles north of Chicago, Illinois and 42 miles south of Milwaukee, Wisconsin.

The main features of the plant are:

• 4 loops
• Pressurized water reactor
• Westinghouse design
• Net Output: 1040 MWe
• Thermal power 3250 MWth
• Permanently shut down
• Date started: June 1973
• Date closed: January 1998

The Steady-State conditions are summarized in Table 1

Table 1: Steady-State main parameters

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Steady-State value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Power (MW)</td>
<td>3250.0</td>
</tr>
<tr>
<td>Pressure in cold leg (MPa)</td>
<td>15.8</td>
</tr>
<tr>
<td>Pressure in hot leg (MPa)</td>
<td>15.5</td>
</tr>
<tr>
<td>Pressurizer level (m)</td>
<td>8.8</td>
</tr>
<tr>
<td>Core outlet temperature (K)</td>
<td>603.0</td>
</tr>
<tr>
<td>Primary coolant flow (kg/s)</td>
<td>17357.0</td>
</tr>
<tr>
<td>Secondary pressure (MPa)</td>
<td>6.7</td>
</tr>
<tr>
<td>Steam generator’s downcomer level (m)</td>
<td>12.2</td>
</tr>
<tr>
<td>Feed water flow per loop (kg/s)</td>
<td>439.2</td>
</tr>
<tr>
<td>Accumulator pressure (MPa)</td>
<td>4.14</td>
</tr>
<tr>
<td>Accumulator gas volume – only tank (m3)</td>
<td>15.1</td>
</tr>
<tr>
<td>Accumulator liquid volume – only tank (m3)</td>
<td>23.8</td>
</tr>
<tr>
<td>Reactor coolant pump’s velocity (rad/s)</td>
<td>120.06</td>
</tr>
</tbody>
</table>

The scenario is a cold leg Large Break LOCA in double guillotine without HPIS. The following statements specify the scenario description:

• LPIS injection with a pressure set point of 1.42 MPa (driven by a flow-pressure table)
• Accumulators injection with a pressure set point of 4.14 MPa
• Containment pressure imposed as a function of time after the break
• Reactor coolant pumps velocity imposed as a function of time after the break

Table 1: Time sequence of imposed events

<table>
<thead>
<tr>
<th>Event</th>
<th>Time(s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Break</td>
<td>0.0</td>
</tr>
<tr>
<td>SCRAM</td>
<td>0.0</td>
</tr>
<tr>
<td>Reactor coolant pumps trip</td>
<td>0.0</td>
</tr>
<tr>
<td>Steam line isolation</td>
<td>10.0</td>
</tr>
<tr>
<td>Feed water isolation</td>
<td>20.0</td>
</tr>
<tr>
<td>HPIS</td>
<td>NO</td>
</tr>
</tbody>
</table>
All the information needed to carry out Phase IV calculations was organized by the coordinator as the “BEMUSE Phase IV Input Specification” [5] and distributed among participants. The specification includes information on:

- Decay power multiplier
- LPIS pressure-flow curve
- Containment pressure
- Pump velocity for primary coolant pumps in intact loops
- Pump velocity for primary coolant pumps in broken loop

All the available details related to the plant lay-out were also included in the specification.

It is important to point out that, as the plant was in permanently shutdown condition from 1998, no detailed information could be made available if needed during the development of the project. In order to work out this problem along with plant parameters, the main features of the LBLOCA scenario were specified in order to ensure common initial and boundary conditions.

5. CODES AND NODALIZATIONS

Table 2 shows the features of codes and nodalizations used by each participant. The table includes:

- Number of hydraulic nodes;
- Number of mesh points for the heat structures;
- Number of core channels (not including the bypass channel);
- Number of axial core nodes per channel.

<table>
<thead>
<tr>
<th>Participant</th>
<th>Code's name</th>
<th>Hydraulic nodes</th>
<th>Mesh points (heat structures)</th>
<th>Core channels (core channels)</th>
<th>Axial active core nodes per channel</th>
</tr>
</thead>
<tbody>
<tr>
<td>AEKI</td>
<td>ATHLET 2.0A</td>
<td>580</td>
<td>1839</td>
<td>2</td>
<td>18</td>
</tr>
<tr>
<td>CEA</td>
<td>CATHARE V2.5 1 mod.3.1</td>
<td>NS</td>
<td>NS</td>
<td>NS</td>
<td>NS</td>
</tr>
<tr>
<td>EDO</td>
<td>Tech-M-97</td>
<td>87</td>
<td>811</td>
<td>5</td>
<td>12(*)</td>
</tr>
<tr>
<td>GRS</td>
<td>ATHLET 2.1A</td>
<td>395</td>
<td>526</td>
<td>2</td>
<td>18</td>
</tr>
<tr>
<td>IRSN</td>
<td>CATHARE2 V2.5_1 mod5.1</td>
<td>NS</td>
<td>NS</td>
<td>NS</td>
<td>NS</td>
</tr>
<tr>
<td>JNES</td>
<td>TRACE ver4.05</td>
<td>743</td>
<td>10660</td>
<td>16</td>
<td>42</td>
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<tr>
<td>KAERI</td>
<td>MARS 3.1</td>
<td>1116</td>
<td>NS</td>
<td>3</td>
<td>18</td>
</tr>
<tr>
<td>KINS</td>
<td>RELAP5/MOD3.3</td>
<td>252</td>
<td>2145</td>
<td>1</td>
<td>18</td>
</tr>
<tr>
<td>NRI-1</td>
<td>RELAP5/MOD3.3</td>
<td>306</td>
<td>2055</td>
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<td>18</td>
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<td>PSI</td>
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<td>NS</td>
<td>NS</td>
<td>NS</td>
</tr>
<tr>
<td>UNIPI-1</td>
<td>RELAP/MOD3.2</td>
<td>NS</td>
<td>NS</td>
<td>NS</td>
<td>NS</td>
</tr>
<tr>
<td>UNIPI-2</td>
<td>CATHARE2 V2.5_1</td>
<td>79</td>
<td>12017</td>
<td>5</td>
<td>21</td>
</tr>
<tr>
<td>UPC</td>
<td>RELAP5/MOD3.3</td>
<td>305</td>
<td>2193</td>
<td>2</td>
<td>18</td>
</tr>
</tbody>
</table>

(*) The fuel path is simulated by 10 axial nodes.
Five active heat structures were nodalized simulating the fuel elements. Figure 1 shows a sketch of core heat structures zones, listed below:

Zone 1: average fuel rods in peripheral channels
Zone 2: average fuel rods in average channels
Zone 3: average fuel rods in hot channels
Zone 4: average fuel rods in hot fuel assembly
Zone 5: hot rod in hot fuel assembly

Figure 1: Core heat structures

Figures 2 and 3 show the sketch of two different nodalization schemes used by two different participants.
The most relevant differences among the nodalizations used are the core vessel detail and the fuel rods. Core vessels have been modelled using one dimensional and three dimensional codes. In each particular case the resulting flow distribution, ECC bypass and the behaviour of liquid in the upper head, among others, significantly explain the diversion of results. Among one dimensional codes an influential feature in nodalization is the use or the availability of cross flow junctions between the core channels and between the downcomer pipes. Related to fuel rods, one participant simulated the oxidation of the cladding while the others did not compute it.

The specifications document for BEMUSE phase 4 devoted a whole section (section 3 in [5]) to list a number of requirements and recommendations for nodalization performance with the aim to have a common basis for comparison. Among them:

- Some initial conditions
- Some nodalization characteristics (core, downcomer, lower plenum and the break itself)
- The use of code options (reflood and CCFL)

The level at which each participant followed the recommended procedures strongly affects the dispersion of the results.

**Figure 2: Example of 1-D Relap5 nodalization scheme used by UPC.**
6. MAIN RESULTS

The nodalization development and the steady-state results were compared among participants in a systematic way. Figure 4 shows a piece of information related to the complete comparison. The example is the normalized pressure drop curve which is quite acceptable. Most of the participants manage to reproduce the reference curve. Some of the differences are due to the small changes performed by participants after the reference curve was supplied. These small changes (like those related to re-splitting the downcomer from 2 to 4 pipes in the coordinators case) produced only small deviations in the comparative plot but came up with some improvements in the reference case.
The comparison of participant results for the reference case has also been performed in a systematic way which includes:

- Calculated sequence of events
- Time trends
- Relevant Thermalhydraulic Aspects (RTA)
- Comments on similarities and discrepancies found among the different groups

All this information can be found in detail in Phase IV report. Among the 25 compared time trends the most significant have been selected and are shown below in Figures 5 to 10.
Figure 5: Time trends of intact loop 1 pressure in hot leg.

Figure 6: Time trends of accumulator 1 pressure.
Figure 7: Time trends of integral break mass flow.

Figure 8: Time trends of ECCS integral mass flow.
Most of the events related to the scenario are strongly dependent on primary pressure time trend. Despite of the dispersion shown in some of the Figures, some events are predicted in a consistent way by participants among these:
- Subcooled blowdown ended
- Cladding temperature initially deviated from saturation (DNB in core)
- Pressurizer emptied
- Accumulator injection initiated
- LPIS injection initiated

Events related to the partial top-down rewet need some explanation. After analyzing the corresponding Figures, despite of a non-negligible dispersion, the shape of the curves shows some consistency. All participants predict a first PCT, a temperature decrease (at the initiation of the partial rewet) and a further temperature increase (at the end of the partial rewet). These events are not so clearly shown when participants are asked to define a time quantity related to each event but there is a general agreement on the shape of the curves. Clearly the time trend analysis (instead of the simple comparison of the time of occurrence of the events) is the best way to show the discrepancies and similarities among results.

A similar comment can be made regarding accumulator behaviour. Despite injection initiation is consistently predicted by participants and properly shown in Figure 8, the prediction of accumulators emptying shows some dispersion. As it is a phenomenon depending on intact leg pressure, pressure error and cumulative time error have a strong effect on the occurrence of the event and dispersion increases.

Finally, the core thermal behaviour, and mainly the full quench, is another event needed of clarification. Figure 10 is maybe the best information for discussion that has some comments involving code effect. The spread of results for the first PCT and for the second is not so high (roughly 200 K for each peak). The lowest of PCT have been obtained by KAERI (1159.1 K) and highest of PCT by EDO "GIDROPRESS" (1326.15 K). Difference between lowest and highest of PCT for RELAP users is about 100 K, for CATHARE and ATHLET users is about 40 K, and for TRACE users is 20 K. Eight participants predicted the time of PCT between 40 s and 60 s except for NRI-1, CEA, GRS, JNES and IRSN. These participants predicted more early the time of PCT (about 10 s). Ten participants predicted the time of PCT between 40 s and 60 s except for NRI-1, CEA and IRSN. NRI-1, CEA and IRSN predicted more early the time of PCT (about 10 s). The major differences between results come with the reflooding behaviour and mainly its duration. Concerning this aspect, among the 13 participants, 8 of them show a medium reflood duration (total core quench obtained between 160 and 250 s), 3 other computations show a long reflood duration (total core quench between 320 to 420 s) and the other 2 show a kind of slow cladding temperature decrease in which it is difficult to establish the time of full quench.

It is clear that dispersion bands exist but it is also clear that the effort of explaining the reasons of such dispersion is a valuable outcome from this phase. The outcome of BEMUSE Phase IV is also helpful to understand the nuances existing inside the user effect. The discussion on the point related to the full quench has been useful to clarify the "border" between user effect and code effect. Figure 11 enlightens these considerations putting together CATHARE and RELAP5 calculations results for this particular aspect. Despite the consistency of both groups of calculations some code effect appears. This point is a minor result of Phase IV detected within the programme although it cannot be solved in its framework.
7. SENSITIVITY CALCULATIONS

Different sensitivity calculations were performed in Phase IV with the aim of helping to prepare the following Phase V of BEMUSE project. The results can be used by participants individually either when deciding which parameters are to be included in their respective uncertainty analysis or after running the uncertainty calculations (for those participants using methods based in Wilks' formula) when deciding whether to accept or to put in question the results of the sensitivity analysis post-calculation.

In order to provide the reader with a better sight of the sensitivity analysis results, the values for $\Delta PCT$ and for $\Delta REFLOOD$ given by all participants have been averaged. As reasonable ranges of variation have been assumed for the input parameters, $\Delta PCT$ and $\Delta REFLOOD$ values provide a good measure of the influence that these input parameters can have on the calculation results.

For the $\Delta PCT$, participants in average have found that the most influential parameters are those related to the energy stored in the fuel elements (i.e. fuel and gap conductivity, power - before and after the scram, and fuel dimensions) and, among them, fuel conductivity, radial power factor (hot rod power) and fuel dimensions.

Regarding the $\Delta REFLOOD$, the average participant has encountered that the parameters having more influence in the time of reflood are containment pressure, power after scram (decay power), radial power factor (hot rod power), power before scram (steady state power) and volume of liquid in accumulators.
The sensitivity study performed in Phase IV has proved to be useful in order to set up the Specification for Phase V.

8. **CONCLUSIONS**

Conclusions can be summarized as it follows:

- All participants managed to simulate the scenario and predict the main parameters with credible consistency
- Maximum values of PCT predicted by participants are quite close one each other
- PCT time trends and timing of complete core rewet still show some disagreements
- A database, including comparative tables and plots has been produced. This database is suitable for providing the explanations needed for the following phases

About the announced difficulty of dealing with a plant that was in permanent shutdown condition from 1998, one can conclude that the participants in the exercise managed to work it out. Although no detailed information could be made available during the development of the project, the specification itself and further contacts among participants were sufficient to reach a suitable definition of common initial and boundary conditions.

The final calculation results had a credible consistency and are considered a good basis for the comparison work of next phase of the project, in which uncertainty bands will be calculated.

Phase IV results are a step forward that contributes to the general goals of BEMUSE project.

**ACKNOWLEDGEMENTS**

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AN INNOVATIVE TRAINING PROGRAM TO INCREASE SAFETY CULTURE ON ELECTRABEL-SUEZ NUCLEAR POWER PLANTS

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ABSTRACT

Training is a key element to maintain and improve safety culture. Progress made in management of safety culture requires new training approaches. Technical competencies are vital but more and more human factors and safety behaviours are also considered as essential. Electrabel has recently introduced a major training program for both its own staff and the contractors working on the sites of its Nuclear Power Plants. This training program stresses the importance of safety culture on both theoretical and practical level and is mostly focused on the behavioural aspects during activities performed at the site of a Nuclear Power Plant. The training contains a theoretical part introducing the basic concepts of nuclear safety and safety culture and a practical exercise in a simulated environment. This program has been built in coherence with the WENRA harmonized guidelines and the IAEA safety standards. An innovative element in the training cycle is the use of a simulated environment, where the actual working conditions in the nuclear part of the installation are simulated. This mock-up installation enables the workers to train the nuclear safety constraints linked to the actual installation and to enhance safety culture by responding on simulated problems and changing conditions possibly being encountered during an intervention at the real working site. A complete circuit with pumps, pipes, valves, sensors, and electrical panels...has been built where technicians can prepare and realize an intervention. The training facility is designed for scenarios like interventions in radiological area, in confined space, at height... Training exercises are recorded on video and trainees are evaluated on their safety behaviours (rigorous and prudent approach, questioning attitude, procedure adherence, communication...) during the intervention but also during preparation of work, pre-job briefing, post-job briefing. During the presentation, details will be provided on the training programme, the training facility, the training programme and the results obtained so far.
1 BRIEF INTRODUCTION OF ELECTRABEL-SUEZ

SUEZ, an international industrial and services Group, designs sustainable and innovative solutions in the management of public utilities as a partner of public authorities, businesses and individuals. The Group aims to answer essential needs in electricity, natural gas, energy services, water and waste management. SUEZ businesses are structured around four operational business lines: SUEZ Energy Europe (SEE), SUEZ Energy International (SEI), SUEZ Energy Services (SES), SUEZ Environment (SE).

SUEZ companies meet the expanding needs of cities and businesses as they are challenged with new constraints tied to demographic growth, urbanization, higher standards of living and environmental protection. Each day, the 140,000 men and women at SUEZ work at local level to resolve these issues through partnerships based on performance, innovation and dialog. Their technical and managerial know-how serves to restrain energy consumption, reduce greenhouse gas emissions, preserve natural resources and provide access to sanitation while continuously monitoring risks that could have an impact on the health and safety of populations.

Thanks to their expertise and commitment, today more than 200 million individuals, 3,000 municipalities and 500,000 industrial clients have daily access to clean energy, treated water and environment-friendly waste services.

Key figures

- €44.3 billion in 2006 revenues
- 200 million individual customers
- 500,000 industrial and commercial clients
- 140,000 employees throughout the world
- 60,000 MW in power production capacity
- 3,000 municipalities served daily
- 600 researchers and experts in 8 R&D centres

Belonging to the operational business line SUEZ Energy Europe, Electrabel is a leading European energy company and number one on the Benelux market. Electrabel provides comprehensive and tailor-made energy solutions for industrial enterprises. Electrabel is strengthening its local geographical presence with generating activities in a number of regions of Europe. It manages diversified generating equipment totaling 30 021 MW. The company’s main objectives are high-energy efficiency and the lowest possible impact on the environment. The European facilities are primarily made up of high-energy yield gas turbines, of extremely reliable nuclear facilities and of renewable energy of generation is CO2-emission free. The 7 Electrabel’s nuclear units are located in Belgium on two sites: Doel with 4 units and Tihange with 3 units. Table 1 provides the dates of first connection to the grid of each unit and the nominal power.

<table>
<thead>
<tr>
<th>Table 1: Electrabel’s nuclear units</th>
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<tbody>
<tr>
<td>Unit</td>
</tr>
<tr>
<td>--------</td>
</tr>
<tr>
<td>Doel 1</td>
</tr>
<tr>
<td>Doel 2</td>
</tr>
</tbody>
</table>
NUCLEAR SAFETY: CONTINUOUS EVOLUTION

The term 'Safety Culture' was first introduced in INSAG's Summary Report on the Post-Accident Review Meeting on the Chernobyl Accident, published by the IAEA in 1986 Ref [1]. It is fairly recent in the history of the nuclear industry and since that time, a large number of publications refer to this concept of “safety culture”.

Obviously, there is an increasing awareness of the contribution that human behavioural sciences can make to developing good safety practices. (IAEA documentation Ref [2])

With the operating experience, our understanding of the factors influencing nuclear safety has improved:

- At the early stage of the nuclear industry, the nuclear plant organizations saw safety as an external requirement and not as an aspect of conduct that will help the organizations to succeed. There was little awareness of behavioural and attitudinal aspects of safety performance. Safety was seen very much as a technical issue.

- With time and progress made in quality management, good safety performances became an organisational goal. An organization at this stage had a management, which perceived safety performance as important even in the absence of regulatory pressure. Although there was growing awareness of behavioural issues, this aspect was largely missing from safety management methods, which comprised technical and procedural solutions. Safety performance was dealt with, along with other aspects of the business, in terms of targets or goals. The organization concentrated primarily on day-to-day matters. There was little in the way of strategy. Management’s response to mistakes was to put more controls in place via procedures and retraining. There was a little less blaming. Safety was thought to imply higher cost and reduced production. There was growing awareness of the impact of cultural issues in the workplace. It was not understood why added controls do not yield the expected results in safety performance.

- And now on, nuclear plants organisations have adopted the idea of continuous improvement and applied this concept to nuclear safety. There is a strong emphasis on communications, training, leadership, and human performances. Everyone in the organization can contribute. Some behaviour is seen within the organization, which enables improvements to be made. The level of awareness of behavioural and attitudinal issues is high, and measures are being taken to improve behaviour. Progress is made one step at a time and never stops. The organization begins to act strategically with a focus on the longer term as well as awareness of the present. There is no goal conflict between safety and production performance. Safety and production are seen as interdependent. Management’s role is seen as coaching people to improve business performance. People are aware of the impact of cultural issues, and these are factors considered in key decisions.

More than ever, human behaviour is considered as a central element for good safety performances. So, progress made in the management of nuclear safety has a direct impact on training of workers and contractors. The training programmes and the training approach must...
also continuously be adapted to follow this evolution. At the beginning, regulators and safety authorities focused on training programmes for control room operators. Operators have to be licensed, follow refresher training on full-scope simulators… The evolution has conducted the nuclear plants to adopt a systematic approach to training on a general basis for all workforces: (e.g. also maintenance, engineering support, radiological protection workers and managers and also contractors). The number of training hours/year-man is higher than in the past. The content of training programmes has shifting from one-off technical training towards initial and continuous training. The methodology needs to be updated in order to be more oriented on behavioural training. Teamwork, leadership, communication, procedure adherence, pre-job and post job briefing, questioning attitude, prudent and conservative approach are much more emphasised than in the past. It requires new training techniques.

3 ELECTRABEL TRAINING PROGRAMME ON SAFETY CULTURE

In the above-described context, Electrabel has put in place in 2006 a new comprehensive training program for both its own staff as well for the personnel of contractors. This program includes a theoretical part that covers the generic principles of successful safety cultures and also a practical part with different work scenario on a full-scale simulator replicating several technical rooms. Both training courses and exercises are concluded with an individual evaluation. A positive result of the evaluation is hereby required to gain access to the site.

3.1 Initial training programme

For contractors: this initial training programme last four days. During the three first days, the trainees will be informed on the potential risks they could encounter during their stay in the installations. They will be trained to assess the danger. They will be informed on the behaviors to adopt in order to prevent risks in normal and accidental situation.

The last day is devoted to practical exercises on the full-scale simulator. A positive result of the evaluation is required to gain access to the site.

For Electrabel staff, the initial training programme is divided in three modules adapted to the level of experience of the trainees:

- Module 1: Initial training for young technicians+managers
  5 days “basis on Nuclear safety”: classroom training + 2.5h/day e-learning + evaluation at the end of each day
- Module 2: advanced training for experienced EBL staff (min 6 months):
  » 2 days safety culture (safety behaviours)
  » 3 days on training workshop (practical exercises pre-post job briefing analyse
- Module 3: advanced training for experienced EBL staff
  » 2 days “Safety Culture – cases study”

In function of the management level and the type of job, additional training is also provided:
- Training on Human error reduction tools (1 day for technicians)
- Training on task observation + coaching (3 days for managers, foremen and team leaders)
3.2 Continuous training programme

The initial training programme has been completely reviewed. It has been done recently. The next step will be to establish a continuous training programme for safety culture. It is now in under development phase. The frequency will most probably be a refresher training every three years.

3.3 Arbitration committee

For people who have already gained a nuclear experience elsewhere before entering the Electrabel’s nuclear sites, it is possible to be exempted of some training courses if they can provide documentary evidence that they have got the appropriate training, qualification during their recent professional experience. This information should be obtained prior to the access on site. The arbitration committee will decide on basis of this documentation if derogation will be admitted and what will be the scope of this derogation (in compliance with IAEA guidelines ref [3]).

3.4 Assistance of partners and registration via internet

In order to facilitate the registration, contractors can easily do it on internet. All the information is also available through this media.
Electrabel has worked with external partners. We have taken advantage of the know how in the development of training programmes and evaluation of competences from the SIFOP [5]. We also work with competence centers like Technifutur [6], Technofutur [7] and Syntra [8] who are officially recognized and so give access to subsidies for professional training.
Centre Secure [9] provide us their expertise in safety and radioprotection. Having the support of their instructors, we are able to conduct much more training sessions per year.
Contractors may also choose different locations to follow the theoretical training (Charleroi, Antwerp, Liège) and Doel or Tihange for the practical exercise on simulator.

3.5 Financial charge

Contractors could simply consider this training of 4 days as a loss of working hours for the business. To avoid such a barrier, Electrabel has carefully examined the potential financial incentives and fixed the registration costs to a minimum of 595 €/person.
Depending from the sector of activity where the contractor works, it is possible to gain access to different financial support. Such aids are substantial enough to transform this training in a profitable financial operation.
In addition, the competences acquired by contractor’s personnel will reduce their accident and rework rate. It is certainly financially interesting but it is unfortunately not accurately quantifiable.
3.6 Training programme for contractors

The programme is detailed in table 2. The main objective of this training is to provide trainees with an awareness of human performance and the factors that may affect nuclear safety. Practical exercises will focus on correct behaviours to adopt and error prevention tools to be used. After this training, contractors will be able to fulfil their job safely, and to contribute to the improvement of safety culture. The four day programme contains legal background, information on specific dangers, occupational health, environment, and measures in place at site to prevent accident and minimise the risks.

Table 2: Initial training programme for contractors

<table>
<thead>
<tr>
<th>Day 1: Safety Culture and Radiation Protection</th>
<th>Morning 4 hours</th>
<th>Afternoon 4 hours</th>
</tr>
</thead>
<tbody>
<tr>
<td>Welcoming objectives</td>
<td>Nuclear Safety</td>
<td>Safety Culture</td>
</tr>
<tr>
<td>(1 h)</td>
<td>(3 h)</td>
<td>INSAQ 4</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Day 2: Safety Culture and Radiation Protection</th>
<th>Morning 4 hours</th>
<th>Afternoon 4 hours</th>
</tr>
</thead>
<tbody>
<tr>
<td>Safety Culture</td>
<td>Nuclear Safety</td>
<td>Radiation Protection</td>
</tr>
<tr>
<td>Risk analysis, safety material, FMEA...</td>
<td>Radiation Protection</td>
<td>Radiation Protection</td>
</tr>
<tr>
<td>(2 h)</td>
<td>(1 h)</td>
<td>(1 h)</td>
</tr>
<tr>
<td>Belgian Plant</td>
<td>(2 h)</td>
<td>(3 h)</td>
</tr>
<tr>
<td>Phase out, WANO KPI Benchmark...</td>
<td>(1 h)</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Day 3: Health &amp; Safety Environment</th>
<th>Morning 4 hours</th>
<th>Afternoon 4 hours</th>
</tr>
</thead>
<tbody>
<tr>
<td>Health &amp; Safety</td>
<td>Safety Culture</td>
<td>Radiation Protection</td>
</tr>
<tr>
<td>Risk management Legislation</td>
<td>Risk management</td>
<td>Environment management</td>
</tr>
<tr>
<td>(1 h)</td>
<td>Explosive atmosphere</td>
<td>ISO 14001</td>
</tr>
<tr>
<td>Specific risks, Electricity, confined space,</td>
<td>(2 h)</td>
<td>Waste treatment, Energy use...</td>
</tr>
<tr>
<td>work at height, fire, toxic products...</td>
<td>(1 h)</td>
<td>(2 h)</td>
</tr>
<tr>
<td>Personal Protective Equipment</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Firefighting equipment</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fire signs</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Rehearsal and theoretical evaluation</td>
<td>(1 h 30)</td>
<td>(1 h 30)</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Day 4: Practical Exercise Evaluation</th>
<th>Morning 4 hours</th>
<th>Afternoon 4 hours</th>
</tr>
</thead>
<tbody>
<tr>
<td>Practical exercise on training simulator</td>
<td>Safety Culture</td>
<td>Rehearsal and theoretical evaluation</td>
</tr>
<tr>
<td>Work scenario on technical equipments</td>
<td>Radiation Protection</td>
<td>Synthese</td>
</tr>
<tr>
<td>(1 h)</td>
<td>(3 h)</td>
<td></td>
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</table>

3.7 Description of the training facilities used for practical exercises

One floor of the training building has been adapted in different rooms:
- a classroom equipped with multi-media projector and computers (use of informatics application to prepare the work order and receive the work permit as in real situation)
- Cold and hot change rooms similar to those at the plant
- A warehouse where trainees will bring the tools and measure instruments
- 8 different technical rooms to practice different exercise scenario like:
  - work at height
  - work in radioactive environment
  - work in air contaminated area
  - work in confined space
  - work on mechanical components
  - work on electrical components
• work on I&C
In these rooms, several circuits are installed. So, exercises can be conducted on job very similar to those realize in reality at the plant. Equipments like ventilators, valves, pumps are fully operational, some circuits are filled with water, dosimeters and whole body counters at the exit can be operated by the trainers to simulate alarms. The technical rooms are equipped with camera. Videos are used during debriefing.

Two similar training facilities have been built: one in Tihange and the other one in Doel.

3.8 Evaluation during practical exercise

In addition to the camera, one trainer per group of 3 to 4 trainees observe the exercise and evaluate the trainees on basis of a check-list with multiple criteria such as;

• Access in the “controlled area”and exit:
  o Undress and dress practices
  o Use of dosimeters, contamination control devices (body+material)
  o Respect of safety rules defined in the work permit
• Job in controlled area
  o Respect of RP signs on doors and equipments
  o Use of measurement devices (O2meter, explosion meter, radiation meter...)
  o Use of personal protective equipment
  o Application of principle time/distance/shield
  o Waste segregation
  o Response in case of alarm
• Safety culture
  o Questioning attitude (good question at the right time)
  o Alertness (detection of deviation, abnormalities)
  o Communication
  o Rigor (procedure adherence)
  o Housekeeping, FME

3.9 Further developments

Tihange NPP was the first plant equipped with this training facility. It was in September 2006. Since that time and till end 2007, about 2300 contractor’s persons have attended the training. At Doel, the training facility is available since October 2007 and at the beginning of 2008, about 1200 contractor’s persons have attended the training.

It is a fairly new training tool and the training departments will develop more exercise scenarios (up to 25 different scenario by the end of 2008).

In the near future, the training facilities will also be used for refresher training.

There is also a high interest in promoting this training at technical schools. Investigations are under way to develop a partnership with schools. As it becomes more and more difficult to find technicians on the market, it will certainly be beneficial to attract technician students in the nuclear sector.

Other nuclear operators are also interested in developing the behavioural training programmes. Electricité de France has shown a big interest in visiting our installations and has decided to build a similar training facility at each of their nuclear stations. Through the
Tacys program, Electrabel has also exchanged its experience in this area with Khalinin NPP in Russia. We expect to see this “good practice” replicated by other nuclear operators rapidly and as it is the goal of this Topsafe, we are open to share our experience with them.

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SARNET: SEVERE ACCIDENT RESEARCH NETWORK OF EXCELLENCE

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ABSTRACT

Fifty-one organizations network in SARNET (Severe Accident Research NETwork of Excellence) their capacities of research in order to resolve the most important pending issues for enhancing, in regard of Severe Accidents (SA), the safety of existing and future Nuclear Power Plants (NPPs). This project, co-funded by the European Commission (EC) under the 6th Framework Programme, has been defined in order to optimise the use of the available means and to constitute sustainable research groups in the European Union. SARNET tackles the fragmentation that may exist between the different national R&D programmes, in defining common research programmes and developing common computer tools and methodologies for safety assessment. SARNET comprises most of the actors involved in SA research in Europe, plus Canada.

To reach these objectives, all the organizations networked in SARNET contribute to a Joint Programme of Activities, which consists in:

- Implementing an advanced communication tool for accessing all project information, fostering exchange of information, and managing documents;
- Harmonizing and re-orienting the research programmes, and defining new ones;
- Analyzing the experimental results provided by research programmes in order to elaborate a common understanding of relevant phenomena;
- Developing the ASTEC code (integral computer code used to predict the NPP behaviour during a postulated SA), which capitalizes in terms of physical models the knowledge produced within SARNET;
- Developing Scientific Databases, in which all the results of research programmes are stored in a common format (DATANET);
- Developing a common methodology for Probabilistic Safety Assessment of NPPs;
- Developing short courses and writing a text book on Severe Accidents for students and researchers;
- Promoting personnel mobility amongst various European organizations.

This paper presents the major achievements after four and a half years of operation of the network, in terms of knowledge gained, of improvement of the ASTEC reference code, of dissemination of results and of integration of the research programmes conducted by the various partners.

After this first period (2004-2008), co-funded by the EC, a further contract for the next four years is under negotiation with the EC as part of the 7th Framework Programme. During this period, the networking activities will focus mainly on the remaining pending issues as determined during the first period, experimental activities will be directly included in the common work and the network will evolve toward a complete self-sustainability. The bases for such an evolution are presented in the last part of the paper.

1 BACKGROUND AND OBJECTIVES OF SARNET

The Nuclear Power Plants (NPPs) existing in Europe are designed with the principles of defence in depth. In particular, they incorporate a strong containment and engineering systems to protect the public against radioactivity release for a series of postulated accidents. Nevertheless, in some very low probability circumstances, severe accident (SA) sequences may result in core melting and plant damage leading to dispersal of radioactive material into the environment and thus constituting a health hazard to the public.

In 2004, many achievements had been already obtained in the field of research on Water Reactor SA, thanks in particular to the numerous European actions undertaken within the 4th and 5th Framework Programmes (FP) of the European Commission (EC). In spite of the accomplishments reached, several issues were remaining where research activities were needed in order to reduce further uncertainties considered of importance and to consolidate severe accident management plans.
Research programmes on SA were - and still are - decided at national levels. Cooperation agreements are often concluded around these national programmes, but on a case-by-case basis. Facing the inevitable reduction of the national budgets in this field, it was necessary to coordinate better the national efforts to optimise the use of the available expertise and experimental facilities in order to resolve the remaining issues for enhancing the safety of existing and future NPPs.

Therefore, since April 2004, fifty-one organizations involved in R&D in SA, including technical supports of safety authorities, industries, utilities and universities, decided to seize the opportunity offered by the EC in the framework of the 6th FP to network in SARNET (Severe Accident Research NETwork of Excellence) their capacities of research in the severe accident area in a durable way. These organizations are coming from 18 Member States of the European Union (Austria, Belgium, Bulgaria, Czech Republic, Finland, France, Germany, Greece, Hungary, Italy, Lithuania, the Netherlands, Romania, Slovakia, Slovenia, Spain, Sweden, United Kingdom), Switzerland, Canada and the Joint Research Centres of the EC (Fig. 1). The network benefits from, and strengthens, the existing complementarities between the different partners (corium / fission product chemistry experts, small scale / large scale testing, simulants / real materials, experimentalists / model developers / code developers) [1].

![Figure 1: SARNET organization countries](image)

The general objectives of SARNET consist in:
- Tackling the fragmentation that exists between the different R&D organizations, notably in defining common research programmes and developing/qualifying computer tools;
- Harmonizing the methodologies applied for assessing risk and improve Level 2 Probabilistic Safety Assessment (PSA) tools;
- Disseminating the knowledge to Newcomers to the European Union more efficiently and associating them with the definition and the conduct of research programmes more closely;
- Bringing together top scientists in SA research to constitute a world leadership in advanced computer tools for SA risk assessment.

This paper presents the major achievements after four and a half years of operation of the network, in terms of knowledge gained, of improvement of the ASTEC reference code for SA, of dissemination of results and of integration of the research programmes conducted by the various partners.

After this first period (2004-2008), co-funded by the EC, a further contract for the next four years is under negotiation with the EC as part of the 7th FP. During this period, the networking activities will focus mainly on the remaining pending issues as determined during
the first period, experimental activities will be directly included in the common work and the network will evolve toward a complete self-sustainability. The bases for such an evolution are presented in the last part of the paper.

2 ORGANIZATION

The SARNET Network is organised on the basis of a two-level structure. On the first level, a Governing Board involving all members is in charge of strategic decisions and is advised by an Advisory Committee. On the second level, a Management Team is entrusted with the task of the day-to-day management of the Network.

2.1 The Governing Board

SARNET is steered by a Governing Board. It reviews the progress made by the Network, in particular in terms of progressive integration, and makes recommendations on future orientations. It decides upon the allocation of the financial contribution of the EC and approves the activity programme. The Governing Board is composed of:

- 1 member designated by each Contractor;
- 1 representative of the EC as observer.

Members are in general of high management level and may commit the resources of their organization for performing activities decided by the Governing Board.

2.2 The Advisory Committee

The role of the Advisory Committee is to provide the Governing Board with advice on strategic orientations of the research activities of SARNET. It involves managers of end-user organizations, including Vendors, Utilities and Regulatory Bodies (which are not necessary SARNET members), appointed by the Governing Board.

2.3 The Management Team

The Management Team is in charge, on behalf of the Governing Board, of the day-to-day administration of SARNET.

Based on the Joint Programme of Activities (JPA) organization (see part 3), the Management Team is composed of the SARNET Coordinator heading the team, of the Advanced Communication Tool Leader, of the ASTEC Coordinator, of the PSA2 Leader, of the Experimental Database Leader, of the Severe Accident Research Priorities Leader, of the three Scientific Coordinators (corium, containment and source term domains), of the Excellence Spreading Coordinator (Education and Training, Book and Mobility work-packages) and of a Secretary.

The Management Team monitors the progress made in the different activities, examines any difficulty which may arise and examines with the corresponding project leaders the possible actions to overcome them, examines the new projects, promotes collaborations both within the Network and with external organizations (OECD, ISTC, IAEA, etc), makes proposals to the Governing Board for updating the programme of activities, manages the communication system and the databases of the Network, organises the training and education activities, and disseminates information inside and outside the network, in particular by organizing periodic conferences and topical seminars, and through a public Web site.

The Coordinator acts under the control of the Governing Board, and reports to it on his duty. He provides the Governing Board with technical and financial reports and implements
its decisions, notably by updating the programme of activities. He is furthermore responsible for the relations with the EC, which, yearly, organises a review of the progress of the project by a panel of independent experts.

3 THE JOINT PROGRAMME OF ACTIVITIES

To achieve the objective of SARNET a JPA has been defined and yearly updated. All the organizations networked in SARNET contribute to the JPA, which is broken down into twenty work-packages (Fig. 2) which can be presented in three elements:

- Integrating activities to strengthen links between organizations;
- Joint research activities to resolve remaining outstanding issues;
- Spreading of excellence activities to diffuse the knowledge.

![Figure 2: SARNET Joint Programme of Activities](image)

4 MAIN ACHIEVEMENTS

The main achievements, after four and a half years of operation of the network, are described below.

4.1 Integrating Activities

The integrating elements of the programme are considered of highest importance and constitute key elements of the JPA.

One integrating element is the implementation of an Advanced Communication Tool (ACT) for promoting information exchange. ACT is a key concept to achieve SARNET goals; it provides unified support for efficient communication within the network, concerning:

- Access, search, publication of documents and codes (concept of knowledge storage),
- Contact and communicating with partners (interactive and collaborative services),
- Joint co-ordination of actions/programmes (co-operative management of the network),
- List of links to satellite community projects (R&D projects, related sites).

A platform has been developed and deployed. Access is given by Web Browsers, enabling access from any Internet connection. Around 250 users of SARNET have been
granted access to this tool and the ACT is used really intensively and efficiently (500 to 1000 accesses per month, 8000 items stored, etc).

The SA integral code ASTEC [2], jointly developed by IRSN and GRS, is the main integrating component and contributes efficiently to the dissemination of the knowledge. Furthermore, most of the “Joint Research Activities” are linked to ASTEC, as one of their ultimate goals is to provide physical models to be integrated in ASTEC. The exchange of information on the detailed models developed by the various experts through interpretation of experiments leads to generic common models in the different detailed codes (example of ICARE/CATHARE and ATHLET-CD for core degradation). Model improvements for the ASTEC code are then derived from these detailed models.

Twenty-seven organizations collaborate with IRSN and GRS on the code development and assessment. ASTEC describes the behaviour of a whole NPP under SA conditions, including SA Management by engineering systems and procedures. It is intensively used by IRSN for Level 2 PSAs for 900 and 1300 MWe PWRs, and more and more partners use it for complete accident scenario calculations.

A close and efficient collaboration between ASTEC users and developers has been set up using ACT and the MARCUS web tool for code maintenance. Training courses on code use were organized. Three main code versions were released to the partners: V1.1 mid-04, V1.2 mid-05 and V1.3 end of 2006 (the latest V1.3 rev2 update being released in Dec.07). A large improvement of the code documentation was reached since SARNET beginning. Three ASTEC Users’ Club meetings, the latest being held in April 2008, allowed fruitful direct discussion between users and developers.

Model developments were performed by CEA on corium behaviour in lower head and on vessel external cooling, including validation on LIVE, SULTAN and ULPU experiments. Other models were proposed or improved after discussion between partners, for instance: release of Silver-Indium-Cadmium and of ruthenium from the core, B$_4$C oxidation, iodine mass transfer, radiolytic oxidation in containment (see details or other examples in part 4.2). Other new models are planned in the near future like aerosol mechanical resuspension in the circuits. ASTEC was also used for preparing and interpreting several experiments such as QUENCH and LIVE in FZK. The model adaptations to BWR and CANDU were specified and ranked, along with promising scoping calculations with the present code versions.

An extensive code validation was carried out by the partners on 70 experiments (analytical and integral ones), often OECD International Standard Problems. Generally, the results can be ranked as good, even very good for circuit thermal-hydraulics, core degradation and fission products and aerosol behaviour. Many plant applications were performed on various NPPs (PWR, Konvoi 1300, Westinghouse 1000, VVER-440, VVER-1000 and RBMK), including benchmarks with other codes. The agreement was good with the integral codes MELCOR and MAAP4 on the trends and orders of magnitude of the main sequence results, and very good with detailed results of mechanistic codes such as ATHLET-CD and ICARE/CATHARE for core degradation. ASTEC numerical robustness has been significantly improved since SARNET beginning.

IRSN and GRS are now taking into account all users' requirements on ASTEC evolution for the new series of ASTEC V2 versions. The 1$^{st}$ version V2.0, to be released end of 2008, will be applicable to the EPR, in particular its core-catcher, and will include most of the ICARE2 (IRSN code for core degradation) mechanistic models.

Harmonization of Level 2 PSA methodology and development of advanced tools is also an integrating activity. Level 2 PSA is a powerful tool to assess plant-specific vulnerability regarding NPP SA. It evaluates possible SA scenarios in terms of frequency, loss of containment integrity and radioactive release into the environment and quantifies the contribution of prevention and mitigation measures in terms of risk reduction. Different
approaches are used in Europe, derived more or less from what has been implemented in the USA. A description and comparison of the main elements of methods used by the different partners to develop their PSA has been written. For many issues regarding Level 2 PSA, questionnaires have been set up and the answers have been analyzed in order to define next steps of harmonization. Case studies for harmonization have then been performed on specific issues as hydrogen combustion, iodine chemistry, melt corium concrete interaction (MCCI), large early releases, reactor end states definitions and methods for level 1 and 2 PSA interface. A certain level of harmonization has been reached and recommendations have been written. A State on the Art Report on Dynamic Reliability methods has been produced and the limitations of classical methods, which could be exceeded using these reliability methods, were identified. Examination of the benefit of one of the possible methods (Monte Carlo Dynamic Event Tree) has been achieved on station black out situation. Large efforts were dedicated to a benchmark exercise (quantification of the risk of containment failure induced by the activation of safety system during the vessel core degradation phase). This benchmark allowed the comparison of dynamic reliability methods with classical ones. Finally, a precise definition of ASTEC requirements for level 2 PSA needs and work on ASTEC coupling with probabilistic codes are ongoing.

The objective of DATANET, the SARNET experimental database network, is to develop and maintain an instrument that ensures preservation, exchange and processing of SA experimental data, including all related documentation. The data are both existing experimental data that SARNET partners are willing to share within the network and all new data produced within SARNET. DATANET is based on the STRESA tool developed by JRC Ispra and consists of a network with several local databases (or nodes). From the central database, it is possible to connect with other local databases; direct connections to the local databases are also possible, which increases the potential and the power of this type of system. Currently, nine nodes exist: the central one at JRC Ispra, and local ones at FZK, IRSN, CEA, CIEMAT, FORTUM-VTT, AEKI, KTH and GRS. Training weeks have been periodically organized at JRC Ispra. The results of more than 100 experiments have been implemented so far.

Research priority assessment is also an integrating activity. It identifies research priorities and intends to re-orientate progressively the existing national programmes, to contribute to launch new ones in a coordinated way, eliminating duplications and developing complementarities. This activity was performed in close collaboration with participants, representing Technical Support Organizations, industry and utilities. As in the Phenomena Identification and Ranking Table (PIRT) carried out in EURSAFE [3], the whole spectrum of SA situations, extending from core uncovering to long term corium stabilization, long term containment integrity, and fission product release to the environment was considered. Special attention was brought to a risk oriented approach in order to really focus on the most relevant pending issues. As a result, a consensus was reached and 18 main issues were ranked into four categories.

Six issues are regarded to be investigated further with high priority:

- core coolability during reflood and debris cooling,
- ex-vessel melt pool configuration during MCCI and ex-vessel corium coolability by top flooding,
- melt relocation into water, ex-vessel Fuel Coolant Interaction (FCI),
- hydrogen mixing and combustion in containment,
- oxidising impact (Ru oxidising conditions/air ingress for high burn-up and MOX fuel elements) on source term,
- iodine chemistry in Reactor Cooling System (RCS) and in containment.
Four issues are re-assessed with medium priority; these items should be investigated further as already planned in the different research programmes. The risk significance is reduced due to considerable progress of knowledge, but some questions are still open:

- hydrogen generation during reflood and melt relocation in vessel,
- corium coolability in lower head,
- integrity of Reactor Pressure Vessel due to external vessel cooling,
- direct containment heating.

For five issues the current knowledge is considered as sufficient assessing the state of knowledge and the risk and safety relevance and taking into account ongoing activities; these issues are assessed with low priority, they could be closed after the related ongoing activities are finished:

- corium coolability in core catcher with external cooling,
- corium release following vessel rupture,
- crack formation and leakages in concrete containment,
- aerosol behaviour impact on source term (in steam generator tubes and containment cracks,
- core reflooding impact on source term.

Three issues are marked as “issue could be closed”. Due to the risk significance and the current state of knowledge, which is regarded as sufficient in comparison to other issues with greater risk relevance and larger uncertainties, no further experimental programme is needed:

- integrity of RCS and heat distribution,
- ex-vessel core catcher and corium-ceramics interaction, cooling with water bottom injection,
- FCI including steam explosion in weakened vessel.

This ranking of issues will allow redistribution of competence and manpower on high priority issues, both in the EC 7th FP and in other international projects (e.g. OECD projects).

### 4.2 Jointly Executed Research Activities

These activities constitute the R&D basis of the network. Linked to the above research priorities, they aim at resolving the priority pending issues. They are split into three areas: corium behaviour, containment integrity and source term.

In all three areas, the same method has been adopted: review and selection of available relevant experiments, synthesis of analyses and interpretation of data from these experiments, and model review, synthesis and proposals of new or improved models for ASTEC.

Corium behaviour is a large topic dealing with more than half of the issues selected in the EURSAFE PIRT [3]. The corium area ranges from early phase of core degradation to late phase core degradation and ex-vessel corium stabilization. A major effort is also underway on the development and improvement of the corium thermodynamic and material physical property databases. Joint activities have been deployed in 22 SARNET organizations [4], such as the contribution to the definition and the interpretation of tests, with benchmark exercises and associated model improvements: OECD-CCI tests around the MCCI programme; QUENCH-10 test on air ingress in bundle geometry; QUENCH-11 test on boil-down and quench; QUENCH-12 test with VVER bundle; COMET-L1 and L2 tests to study MCCI in 2D geometry; LIVE or VULCANO-COMET tests. Similar activities have been carried out for ongoing and new projects from the International Scientific and Technical Centre (ISTC): PARAMETER project on core top flooding models, METCOR on the impact of thermo-mechanical interaction on the vessel behaviour, CORPHAD on the corium thermodynamic (improvement of the NUCLEA database). In the International Source Term Programme (see Source Term part below), FZK and IRSN have harmonized their test matrices on Zircaloy oxidation by air/steam mixtures and on B4C oxidation and degradation.
Among the main achievements on the corium activities [4], we can quote as examples:

- The understanding of the oxidation phenomena in steam and in air has progressed and oxidation correlation have been validated. The importance of material composition has been demonstrated. Further research is required on new cladding materials, especially in relation to the hydrogen and fission product source term issues.
- Data on B4C oxidation has been collected, thanks to experiments at FZK and IRSN, which allows a common interpretation of the integral test PHEBUS FPT3.
- The launching of the late-in-vessel LIVE experiments has started a new series of modelling and analytical work on in-vessel pool behaviour.
- The different models of vessel failure by creep rupture have been compared and a common understanding of the OECD OLHF-1 test has been achieved. Modelling of the crack evolution is in progress.
- 2D Debris bed coolability analyses for inhomogeneous bed structures showed an increased coolability compared to earlier 1D particle beds. This launches a new interest in this issue both experimentally and numerically including a pursuit of the debris bed formation and its characteristics of coolability importance.
- Likewise, the recent 2D MCCI test resulted in unexpected results with a marked ablation anisotropy for silica-rich materials. Interpretation and modelling of this behaviour must be pursued.
- Core catcher concepts based on spreading (EPR) and bottom flooding (COMET, downcomers) are being studied and progress has been achieved in their modelling.
- The validation of the chemical thermodynamics database NUCLEA has been extended thanks to the analysis of EC-funded experiments and a round-robin exercise has qualified the uncertainties of energy dispersive X-ray spectrometry (EDX) analyses.

The research efforts on energetic phenomena that could potentially threaten containment integrity concern hydrogen behaviour and fast interactions in the containment. For the former, the hydrogen combustion and associated risk mitigation is studied, concentrating on the formation of combustible gas mixtures, local gas composition and potential combustion modes, including reaction kinetics inside catalytic recombiners. Hydrogen distribution within the containment is studied to assess the risk of high concentrations. Experimental programmes on combustion with gradients (ENACCEF, IRSN) and recombiner kinetics (REKO-3, FZJ) have been performed and/or are ongoing. ENACCEF experimental results have been used in a benchmark using 3 different 3D Computational Fluid Dynamic (CFD) codes (FLUENT, TONUS-3D and REACFLOW). The results revealed some weaknesses of the combustion models used when simulating combustion with negative hydrogen gradient. Further experiments in the ENACCEF facility will focus on the interaction of hydrogen flames with sprays. New tests in REKO-3 are focused on the gas ignition on hot catalyst elements. The results will be used for the validation of numerical codes.

The major achievement in the topic of containment atmosphere mixing is the simulation of containment spray experiments that were performed in small scale (TOSQAN, IRSN) and large scale (MISTRA, CEA) facilities. Both atmosphere depressurization and stratification break-up phenomena were addressed. Experiments were simulated using CFD and lumped parameter codes. The calculated results demonstrated the feasibility of such simulations, which could also be applied to actual containments, provided that sufficient computing capacities are available. Apart from that, CFD simulations of the operation of Passive Autocatalytic Recombinators (PAR) in simplified 2D containment models represent a first significant step towards comprehensive simulations of PAR-atmosphere interaction in real plants. Finally, simulations of condensation experiments, performed in the CONAN (Univ. Pisa) facility allowed the comparison and assessment of different steam condensation models.
that are used in CFD codes and significantly influence the behaviour of the simulated atmosphere.

Concerning fast interactions, FCI is studied to increase the knowledge of parameters affecting steam explosion energetics during corium relocation into water, and determine the risk of vessel or containment failure by investigation of specific processes like premixing, melt fragmentation and particle heat transfer mode. The work on FCI in SARNET was closely linked to Phase 1 of the OECD/SERENA programme, which had the objective of evaluating the capabilities of the current generation of FCI computer codes to predict steam explosion loading of the reactor structures and reaching consensus on the understanding of important FCI phenomena relevant to the reactor situations. Mainly, MC3D and IKEMIX/IDEMO have been used for this purpose. In addition a number of experiments were performed in MISTEE and DEFOR (KTH), and KROTOS re-started at CEA-Cadarache after move from JRC-Ispra, with the aim to complement understanding of fragmentation and explosion behaviour of corium melts. Main accomplishment was a consensus that in-vessel steam explosion would not induce failure of the vessel, thus closing the in-vessel steam explosion issue from the risk perspective, and that ex-vessel steam explosion could induce some damage to the cavity. However, the level of loads in the latter could not be predicted due to a large scatter of the results. Major reasons of this scattering were found to be the uncertainties on void distribution in the pre-mixing region, inducing large discrepancies on the initial conditions of the explosion, and on explosion behavior of corium melts, inducing more or less arbitrary tuning of the explosion parameters. These uncertainties will be addressed experimentally and analytically in Phase 2 of OECD/SERENA programme and in SARNET2.

The second issue concerning fast interactions is Direct Containment Heating (DCH). This includes melt dispersion into various reactor compartments, heat transfer and chemical processes such as production and combustion of hydrogen. The consequences of DCH are essentially related to cavity geometry; therefore a database has been established for the plant types EPR, French PWR-1300, VVER-1000 and the German Konvoi by an experimental programme performed in the DISCO-facilities (FZK). For EPR and VVER1000 plants the DCH issue can be considered as closed, due to their cavity design. The efforts to improve the predictive capabilities of the CFD code MC3D on the one hand and the system codes COCOSYS and ASTEC on the other hand will be continued. Benchmark exercises have revealed severe deficiencies in the current modelling: debris dispersion correlations need to be better adapted to the respective cavity geometries and oxidation and hydrogen combustion models are needed as they are the really challenging phenomena as regards containment integrity (experimental results showed that direct thermal effects alone should not challenge the containment integrity). Based on available experimental data from separate effect tests in DISCO, the scaling of combustion of hydrogen jets in an air-steam-hydrogen atmosphere must be established by applying dedicated combustion codes (COM3D, REACFLOW), then the modelling parameters must be transferred to codes with DCH models (CFD codes or COCOSYS) and finally adapted to ASTEC.

In the Source Term area, fission product release, transport and deposition were studied, including air ingress (the influence of an oxidising environment on release and circuit phenomena). For transport and deposition, one study focused on iodine, both its volatility in the primary circuit (particularly, for interaction with concomitant release of Ag/In/Cd) and its gas-liquid partitioning in the containment (also extended to ruthenium), while the other looked at aerosol behaviour in scenarios of special significance for risk: by-pass sequences (particularly, Steam Generator Tube Rupture (SGTR)), through-containment cracks and thermal and mechanical remobilization. The main achievements were detailed by Haste [5].

Extensive effort was devoted to the continuing experimental International Source Term Programme (ISTP) launched by IRSN, CEA and EDF with the support of the EC [6].
Interpretation of available AECL and of RUSET (AEKI) data showed that Ru release occurs in oxide form after an incubation period during which full oxidation of fuel and cladding occurs. RUSET and VTT tests showed that oxide forms can stay volatile enough in lower temperature regions to be transported to the reactor containment in a stable volatile form, a very significant result. These tests continue. Test conditions in the future VERDON programme (CEA) [6] are being defined through air ingress reactor scenario simulations. These new data will be used to validate independently the improved ASTEC modelling. The ISTC VERONIKA experiment proposal on fission product release from highly irradiated VVER fuel was reviewed and SARNET proposals on the test matrix were adopted; this followed a similar exercise with the EVAN iodine chemistry project, whose results are now under evaluation. FIPRED experiments (INR) provided data on UO$_2$ pellet self-disintegration.

Concerning fission product transport, IRSN interpreted iodine chemistry in the circuit, based on data from PHEBUS FP, and VERCORS HT (CEA). Under reducing conditions, and without absorber material, it seems relatively straightforward, the iodine being transported mainly as caesium (and rubidium) iodide. In oxidizing conditions it is more complicated since Cs take-up in forms other than CsI affects the chemistry. Hence, iodine can either still be principally CsI or tends to form other metal iodides such as with control rod materials or, if these are not present, conditions become conducive to HI formation. These statements still need to be confirmed. The experimental CHIP programme (IRSN) [6] is providing kinetic and thermodynamic data on iodine transport through the primary circuit, particularly concerning key systems such as \{-I-Cs-O-H\}, which will be used to improve ASTEC models, and help in the scaling of experimental results to reactor conditions. VTT are providing experimental support. In the QUENCH-13 bundle test (FZK), performed with strong experimental (PSI, AEKI) and modelling (PSI, GRS, EDF) support, with associated small-scale tests, production of Ag/In/Cd aerosols following PWR control rod failure was measured continuously for the first time, along with speciation at intervals. These data are being correlated with the core degradation and with earlier analytical tests EMAIC (CEA). Post-test calculations are being performed on this and on PHEBUS FP data; model improvements will be carried out as required.

Several facilities investigated aerosol retention within the steam generator under SGTR conditions: PSAERO/HORIZON (VTT), PECA/SGTR (CIEMAT) and ARTIST (PSI). Overall, these tests showed that wet scenarios (those with the breach under the secondary side water level) would provide effective scrubbing of particles, and even dry scenarios could capture a fraction, albeit smaller, of the particles entering the secondary side. The VTT tests showed that resuspension is important in aerosol retention within horizontal tubes and is enhanced by sudden velocity changes. All the available resuspension models are being assessed by comparison to data (new VTT results, EC/JRC Ispra STORM) and with each other, while the new VTT data will be used to develop a new empirical model. Revolatilisation tests in the small-scale REVAP facility (JRC/ITU) on PHEBUS FP samples show that Cs revapourisation can be very high (~95%) on flat metallic substrates. CsOH deposits on stainless steel have the same behaviour as that of the PHEBUS FP deposits. VTT are starting new tests on speciation from revapourisation aerosols complementary to the CHIP programme mentioned above.

Retention of aerosols in containment cracks can be effective, particularly in the presence of steam (SIMIBE tests, IRSN). Stand-alone and in-code models have been developed. They will be tested against data from experiments in COLIMA (CEA) to be performed under the PLINIUS platform; the successful proposal for which was coordinated within the Source Term area. A SARNET team is providing pre-test support.

The facilities involved in Containment Chemistry are PHEBUS FP, CAIMAN, SISYPHE, the Chalmers facility, PARIS and EPICUR [6], with data recently released from
RTF P9T3 (AECL). The Iodine Data Book compiled by Waste Management Technology provides a critical review of chemistry data and models. CAIMAN results showed that in the presence of paints, irradiation and high temperature, organic iodide can be the dominant form of volatile iodine; in alkaline conditions, gaseous iodine concentrations decrease by several orders of magnitude. Mass transfer between sump and gas phase was addressed in SISYPHE; evaporating conditions increase the transfer rate from the liquid to the gas phase and change the steady-state iodine concentrations, the sump iodine concentration being reduced. The well-known two-film model was extended to these conditions, with improvements introduced into ASTEC. A more mechanistic model is now under validation [7].

The effect of radiation on the containment atmosphere and the effect of metallic impurities in the sump were investigated in the PARIS and Chalmers experimental programmes, respectively, while Chalmers/VTT are investigating speciation of iodine oxides/nitroxides formed in the atmosphere under radiolysis. The organic iodine formation models consider thermal and radiolytic mechanisms in the gas and liquid phases. There are however discrepancies in the aqueous modelling, essentially concerning the organic sources. Data from EPICUR are being interpreted using codes like ASTEC/IODE, INSPECT and LIRIC. Ruthenium behaviour in-containment has been studied experimentally and theoretically by IRSN (further EPICUR tests) and by Chalmers, as well as the effect of fission product heating on passive autocatalytic recombiners, where ASTEC/SOPHAEROS has successfully modelled the bench-scale RECI tests (IRSN); scaling effects are now being considered. Finally, cooperative evaluation with several codes of the ThAI-Iod9 integral test on containment iodine chemistry was completed under the coordination of GRS.

From the above paragraphs, we can note that a real integration of these research activities has been achieved thanks to:

- Collaborations on pre and post calculations of experiments: for instance PSI performed pre and post-test calculations of the FZK QUENCH tests,
- Joint realization of experiments: for instance experimentalists from VTT have installed and operates specific instrumentation on the IRSN CHIP experiments,
- Joint definition and interpretation of experiments (many “interpretation circles” created and really active),
- Benchmarking of codes,
- Distribution of codes to partners in order to achieve a common understanding of experimental phenomena (ASTEC modules),
- Exchanges on the application of R&D results to the reactor scale,
- Round-robin exercise on EDX analyses of a prototypic corium sample by three European laboratories (Cadarache, France; Karlsruhe, Germany; Rez, Czech Republic),
- Yearly technical meetings in each of the three areas (corium behaviour, containment integrity and source term), complemented by a large number of specialists’ meetings.

### 4.3 Spreading of Excellence Activities

The third major type of activity concerns spreading of excellence. The more experienced organizations started to contribute to disseminating the excellence by preparing an educational course on SA phenomenology addressing PhD students and young researchers, given in January 2006 over 5 days. A training course on “Accident progression (data, analysis and uncertainties)”, addressing instead more experienced nuclear safety specialists, was given in March 2007 over 5 days. Finally, a third course was organized in Budapest in April 2008 covering both phenomenology and severe accident scenario codes. From 40 to 100 persons...
attended to each of these courses. Besides this, a textbook on SA phenomenology is being written. This covers historical aspects of Light Water Reactor safety and principles, phenomena concerning in-vessel accident progression, both early and late containment failure, fission product release and transport; it contains a description of analysis tools or codes, of management and termination of SA, as well as environmental management. It also gives elements on Generation 3 Reactors. The partners who agreed to work together in preparing these courses and writing this book are universities, TSOs\(^1\), national laboratories and industrial organizations that share their great talent and experience within SARNET.

A mobility programme, under which students and researchers can go into different laboratories of SARNET for training, complements these spreading of excellence activities. 33 mobilities, with an average duration of 3 months, have been funded by SARNET.

Three conferences (European Review Meetings on Severe Accident Research – ERMSAR) have been organized in France, Germany and Bulgaria as a forum to the Severe Accident community. They are becoming one of the major events in the world on this topic.

5 SARNET FOLLOW-UP

The SARNET contract with the EC covered a four and a half year period from April 2004 to September 2008. From 2006, a specific working group composed of 9 representatives of the Governing Board was created to prepare a SARNET Follow-up. It facilitated a positive response to the second call for proposals of the EC 7\(^{\text{th}}\) FP, and the contract is currently under negotiation. So, the Network should still be co-funded by the EC for four more years.

During this period, the networking activities will focus mainly on the high priority issues (see part 4.1), experimental activities will be directly included - and partly funded - in the common programme and the network will evolve toward full self-sustainability. The bases for such an evolution are presented hereunder.

Forty-one partners from Europe, plus Canada, Korea and the United States will network their research capacities in SARNET2. In the continuity of SARNET, the project has been defined in order to optimize the use of the available means and to constitute a sustainable consortium in which common research programmes and a common computer tool on SA (ASTE integral code) are developed.

The SARNET2 Network will be organised on a similar way as SARNET. To increase efficiency at the decision-making level, the SARNET Governing Board will be replaced by a Core Group of 10 members in charge of strategy, advised by an Advisory Committee of managers of end-user organisations. A General Assembly, constituted by one representative of each Consortium Contractor, plus the EC representative, will be called yearly in order to inform and consult all the Consortium Contractors on the progress of the network activities, the detailed implementation plan and the decisions taken by the Core Group. On the second level, as for SARNET, a Management Team (Coordinator and seven work-package leaders) will be entrusted with the task of the day-to-day management of the Network.

The key integrating and spreading of excellence activities will be pursued: gathering of available experimental data in a common format in a scientific database; integration of acquired knowledge in ASTEC (including necessary code developments and, in particular, its applicability to all types of European NPPs); spreading of the knowledge through the public SARNET WEB site and the ACT; organization of conferences and seminars; organization of education and training courses; encouragement of exchange of students and researchers. Besides, research priorities will be periodically updated. PSA2 activities will not be pursued within SARNET2, as a specific EC contract on harmonization of PSA2 studies has been set up, starting early in 2008 (Cf. \url{www.asampsa2.eu}).

\(^1\) Technical Support Organizations to Safety Authorities
The joint research activities will include experimental programmes on high priority issues defined during SARNET (see part 4.1), with a special focus on corium and debris coolability and on MCCI. Common analyses of experimental results and benchmarking activities, to elaborate a common understanding of relevant phenomena, will be pursued through various technical circles. Links with end-users and with other international programmes and organizations as OECD/NEA, ISTC and other programmes co-funded by the EC (mainly SNE-TP, PHEBUS FP, ISTP, ASAM-PSA2) will be maintained and reinforced. Progressive opening of the activities to new generations of reactors will be also considered.

Finally, the network will evolve toward real self-sustainability, through the creation of a legal entity.

6 CONCLUSION

The SARNET Network of Excellence activities started in April 2004 with the ambitious but highly important objective to provide an appropriate frame for achieving a sustainable integration of the European severe accident research capacities.

By capitalizing the acquired knowledge in the ASTEC code and in the DATANET database, SARNET is really producing conditions necessary for preserving the knowledge produced by thousands of person-years of research, and disseminating it to a large number of end-users. The ASTEC code is being actively used as well as DATANET.

By fostering collaborative work in the PSA2 domain, SARNET has started to create the necessary conditions for harmonizing the approaches and making Europe a leader in SA computer code and risk assessment methodology.

Through an education and training programme (organization of courses, writing a text book addressing young scientists), SARNET is developing synergies with educational institutions to keep attractive this domain of activity for young people. This is reinforced by an efficient mobility programme, which allows fruitful exchanges between European laboratories for young students and researchers.

By fostering collaborative R&D work in the domains of corium behaviour, containment integrity and source term, SARNET is progressing to solve remaining outstanding issues and to provide ASTEC with modelling recommendations. Proposals have been elaborated for various ASTEC models and their implementation is either done or under way.

Finally, SARNET is clearly becoming a reference, in terms of research priorities in the field of SA, having impact on national programmes and associated budgets. Progressively all the research activities in this field will become coordinated by the Network, which contributes to an optimised use of European resources.

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Two New Models in SCDAP for CANDU Fuel Channel under Severe Accidents

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ABSTRACT

It is generally known that detailed models for severe accident evolution developed for PWR are not applicable in a straightforward manner to CANDU reactors. The main reason is the existence, with this reactor design, of the horizontal fuel channels separated by moderator inside Calandria vessel. The SCDAP/RELAP5 code can calculate for CANDU the heat-up phase of an accident, including oxidation, and can be used to some extent in simulating degradation phenomena such as channel sagging or bundle slumping, but cannot describe properly the evolution of a severe accident after the loss of the initial channel geometry for this reactor type. The paper first evaluates the application of the “as-received” code to a CANDU6 fuel channel and then introduces two new models, created inside the routines structure of SCDAP, and dedicated to the early phase of a Loss of Coolant Accident coincident with Loss of Emergency Core Cooling (LOCA/LOECC). The two models are: a) metallic melt relocation model in the horizontal channel geometry, taking advantage of the original Liquefaction-Flow-Solidification (LIQSOL) model existent in SCDAP, and b) bundle slumping and heat transfer model between fuel elements inside the bundle and between the elements and the cylindrical structure simulating the pressure tube. Finally, the effect of the application of the new models on the overall performance of the SCDAP/RELAPSIM/Mod3.4(bi7) code is highlighted and the conclusions are drawn regarding the future steps in developing and validating a new version of the code for CANDU.

1 INTRODUCTION

The SCDAP/RELAP5 code has been developed for best-estimate transient simulation of light water reactor coolant systems during a severe accident. The code models the behaviour of the reactor coolant system, core and fission products released during a severe accident transient. The code is the result of merging the RELAP5 code, calculating thermal-hydraulics, control system interactions, reactor kinetics, and the transport of noncondensible gases and SCDAP code, which models the core behaviour during a severe accident. SCDAP/RELAP5 is a flexible tool because of its generic modelling approach that permits as much of a particular system to be modelled as necessary and consequently it is used to a wide range of plant transients, research reactor transients and experiments in small scale facilities.

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Although developed mainly for PWR and BWR, the code has already been used for a range of applications to PHWR [1],[2],[3]. The majority of the applications to this reactor design use the thermal-hydraulic code RELAP5. SCDAP code includes models for vertical fuel and core components. Consequently, the horizontal geometry of the fuel channel in CANDU makes any application of SCDAP which would attempt to go further than heat-up and oxidation to be questionable.

The paper evaluates first the possibility to model the early phase of a LOCA accident coincident with a loss of emergency core cooling (LOECC) with the ‘as-received’ version of the code, investigating its capability to treat oxidation and deformation phenomena inside a CANDU channel under postulated inlet steam flow rate. This constitutes the starting point for developing specific physical models for the degradation of a horizontal channel. Then, the main purpose of the paper is touched, namely new developments in SCDAP code. Two new models for treating the early phase phenomena are introduced, including the metallic melt relocation, and the very first results are discussed. Specific developments for CANDU should extend in the next years, if a code version capable to perform a complete severe accident sequence up to the failure of Calandria vessel is wanted.

2 USER MODELLING – ‘AS-RECEIVED’ CODE VERSION

2.1 Deformation phenomena

Three modes of deformation may appear with a CANDU channel rather soon after LOCA/LOEC:

- a) pressure tube ballooning;
- b) pressure tube sagging;
- c) fuel bundle slumping.

If the pressure is still high enough in an early heated channel, the pressure tube can balloon uniformly and contact the calandria tube, establishing an effective heat transfer path to the moderator [4].

At low pressure, the dominant mechanism for pressure tube deformation is sagging, appearing at a temperature lower limit of 850 °C [5], in which the contact between the pressure tube and the calandria tube is over a contact angle $\psi_0$, as in Figure 1.

![Figure 1. Sagged pressure tube in contact with the calandria tube](image1)

![Figure 2. Close-packed configuration as a result of bundle slumping](image2)

The sagged condition of the pressure tube is the one adopted in the calculations. It is assumed to produce higher temperatures than pressure tube ballooning because the heat transfer to the moderator is less efficient compared to the latter case.
Fuel bundle slumping is a phenomenon likely to appear in both balooned and sagged pressure tube state and is due to the distortion of the end plates resulting in ineffective end support for the fuel. The elements can slip from their central spacers sideways, allowing the bundle to assume a settled geometry which is sometimes called the „close-packed configuration”. Experiments show that CANDU fuel elements will distort quite early during the transient heat-up, at cladding temperatures ranging from 1200 °C to 1400 °C [6].

Fuel bundle slumping will produce flow redistribution inside the channel and bypassing effect. A settled bundle configuration will result, with elements in direct contact with each other and with the pressure tube in the lower part of the bundle. As a result, steam starvation can appear for the majority of the elements in the process of oxidation, and better cooling conditions for some of the upper elements (see Figure 2).

2.2 SCDAP/RELAP5 Model for the CANDU6 Channel

A standard 5.94-m-long CANDU fuel channel, containing 12 bundles of 37 fuel elements each was modelled without feeders and end fittings, with the RELAP/SCDAPSIM /Mod 3.4(b17). In the hydrodynamic schematic drawing presented in Figure 3, the two RELAP5 “time-dependent” volumes simulate the pressure and temperature conditions during the steady-state regime and the pressure and static quality conditions in course of the transient, in the inlet and outlet headers of the reactor. The inlet junction, modelled as a “time-dependent” junction is used to control the mass flow rate boundary condition, introducing the nominal water flow rate during steady-state and a mixture of water and steam during the transient.

![Figure 3. Hydrodynamic representation of the CANDU6 channel with RELAP5](image)

SCDAP components representing fuel rods rings are associated to the hydrodynamic channels. The pressure tube-annular gas-calandria tube system was modelled using a 'shroud' component. The moderator outside the calandria tube is simulated by a large stagnant heavy water volume at low pressure and temperature (1.25 bar and 74 °C). Actually, unless the failure of the moderator systems, the moderator can be assumed as being in forced cooling, with rather intricate flow pattern inside the Calandria vessel [7].

Sagging modelling was realized by altering the thermal conductivity of the input declared material representing the carbon dioxide inside the annular gap in order to take into account the radiation heat transfer between the pressure tube and the calandria tube:

\[
k_{\text{eff}} = k_{CO_2} + \frac{\delta_{\text{gap}} \sigma}{\varepsilon_{pt} + \frac{1}{\varepsilon_{ct}} - 1}. \frac{T_{po}^4 - T_{ci}^4}{T_{po} - T_{ci}}
\]

where: \(k_{\text{eff}}\) - effective conductivity of the gap (W/m·K); \(\delta_{\text{gap}}\) - thickness of the gap (m); \(T_{po}\) - outer pressure tube temperature (K); \(T_{ci}\) - inner calandria tube temperature (K); \(\sigma\) -
Stefan-Boltzman constant $(W/m^2\cdot K^4)$; $\varepsilon_{pt,ct}$ - emissivities of the surface of the pressure tube and calandria tube.

Enhanced heat transfer due to sagging is calculated by multiplying by a constant the value the gas thermal conductivity in eq.1. The constant is determined from a two-dimensional steady-state heat transfer model as a function of the pressure tube temperature and the contact angle between the tubes [8].

$$k_{CO2} = C \cdot k_{CO2} \quad (2)$$

where: $C$ is the conductivity multiplier due to sagging.

In principle, ballooning can also be modelled by means of the same procedure, defining a more extended contact angle.

Concerning bundle slumping, a close-packed configuration was calculated by producing a restart file at the approximate moment when the escalating fuel temperature reaches a predefined value. Beyond this moment, re-nodalization was made by introducing a bypass hydrodynamic channel and altering the fuel channels area and hydraulic diameter on the entire length of the fuel channel. After slumping, the bypass channel created is estimated - on the basis of HTBS test #3 configuration reproduced by [9] - to occupy 48% of the initial cooling area of the channel, whereas the total fuel channels area is now only 52% of the cooling area.

2.3 Results with ‘as-received’ version

‘As-received’ version means here that the original physical models of the Mod3.4(bi7) version are used to generate results. Some elements of the original metallic melt relocation model (LIQSOL) will be provided further, when the new models will be introduced. Calculations are performed with a series of constant steam mass flow rates (1 g/s, 5 g/s, 11.7 g/s 20 g/s, 50 g/s) for a maximum rated 7.3 MW channel and for an average 5.7 MW channel. The former is used to obtain maximum temperatures inside the channel and the later for hydrogen production and for the evolution of the components of channel power: oxidation and power transferred to the moderator. Results of SCDAP/RELAP5 application are represented against results obtained with CANDU channel dedicated code CHAN-II [10] for the same cases. Only excerpts of the entire series of results are illustrated, since the purpose of presenting them is just to provide an image of the ‘as-received’ version capabilities with respect to CANDU channel phenomena.

Figures 4, 5 and 6 display the evolution of temperature inside the 7.3 MW channel for three steam flow rates: 1 g/s (starvation), 11.7 g/s (maximum hydrogen production), and 20 g/s (maximum temperature), whereas Figure 7 is a synthetic representation of all cases integral hydrogen production. Corresponding CHAN-II results (at maximum temperature bundle) are also plotted. ‘S/R5’ (SCDAP/RELAP5 curves) calculations include both the central and outer ring elements temperature while ‘CHAN-II’ are fuel average temperature values. Calculations are performed with two sets for the emissivities of the pressure tube and of the calandria tube:

(i) $\varepsilon_{pt} = 0.85$ and $\varepsilon_{ct} = 0.9$ as used in [11] for the pressure tube and for the calandria tube with a black oxide during manufacture;

(ii) $\varepsilon_{pt} = 0.7$ and $\varepsilon_{ct} = 0.2$ as considered in the CHAN II calculations. This case is basically similar to the non-oxidized zircaloy in the MATPRO library [12] for which $\varepsilon = 0.325$

All results in the discussed figures are obtained with the already exposed method for thermally quantifying sagging, using $C=6$ as corresponding to $\psi_0 = 5^\circ$. The agreement with CHAN-II might be considered good, but there exist differences between the SCDAP/RELAP5 results with emissivities set (ii) and CHAN-II. These can be due either to user effects (CHAN-II input deck details were not available) or/and to differences in the thermal treatment by the
two codes. Figures 7 and 8 present the components of power for 1 g/s and 11.7 g/s case. The underestimation of temperatures in CHAN-II persists both with oxidation powers higher and lower than in SCDAP/RELAP5.

Figure 4. 7.3 MW channel SCDAP/RELAP5 cladding temperature for central and outer elements. Axial level 4. Steam flow: 1 g/s.

Figure 5. 7.3 MW channel SCDAP/RELAP5 cladding temperature for central and outer elements. Axial level 6. Steam flow: 11.7 g/s.

Figure 6. 7.3 MW channel SCDAP/RELAP5 cladding temperature for central and outer elements. Axial level 10. Steam flow: 20 g/s.

Figure 7. Integral hydrogen production. Calculations for an average 5.7 MW channel with emissivities set (ii).

One particularly interesting result of SCDAP/RELAP5 is the maximum clad temperature in the 20 g/s steam flow rate case. As indicated by the green curve maximum temperature in Figure 6, ceramic melt is reached and molten pool is formed. That is a result that could not be reproduced by CHAN-II since models for severe damage do not exist in this code. But for such extent of the degradation, the application of SCDAP/RELAP5 is not justifiable, because of the axial gravitational relocation of the degraded fuel which is typical for vertical fuel. Also, two questions appear: if the existing physical models do treat the fuel channel correctly after the onset of zircaloy melting and, the second, if the formation of molten pool is possible in this phase of the accident after all. The second part of the paper will attempt to discuss these issues and to introduce a new metallic melt relocation model for horizontal geometry.
The maximum clad temperatures for the bundle slumping calculations are presented in Figure 10. It appears that the effect of steam flow diversion through the by-pass channel is severe, since high steam flow rates are necessary for reaching 2400 K. But again, is the restart and resize of the sub-channels an adequate treatment for the bundle slumping phenomenon, as the individual bundles are not reaching the slumping temperature threshold at the same time? A modelling improvement that is presented further on is created to offer a more realistic approach to this issue.

3 NEW SCDAP MODELS DEDICATED TO CANDU CHANNEL

3.1 Metallic melt relocation in horizontal geometry

The original Liquefaction-flow-Solidification (LIQSOL) model in SCDAP [13] calculates the change in fuel rod configuration due to meltdown and also calculates the oxidation and heat transfer at the locations in a fuel rod at which liquefied cladding is slumping. Figure 11 summarizes the processes occurring during meltdown of the fuel rods.
The LIQSOL model performs calculations in three steps. The first calculates where the cladding and fuel have been liquefied. The second calculates where and when the cladding oxide shell is breached. The third calculates the configuration and the relocation of the liquefied cladding and fuel that flows through a breach, down the outside surface of the fuel rod, and solidifies.

The relocated material is in the configuration of drops with 3.5 mm radius and is slumping down with a constant velocity of 0.5 m/s. The flow is exclusively on the outside surface of the cladding. There is an oxide layer temperature (default 2500 K) beyond which the oxide does not retain inside the U-Zr-O mixture and slumping, drop by drop, takes place. However, if cladding is oxidized more than a prescribed value (default 60%) the oxide layer does not fail until its melting temperature is reached. The drops flowing down interacts both with the steam and with the intact fuel rod surface. The processes of relocation, oxidation and heat transfer that drops are undergoing are represented in Figure 12. The solidification of drops is produced, depending on the user choice, either when they encounter a sufficiently low surface temperature or when the thermal balance leads to the phase change of the relocating mixture. The collection of frozen drops may cause a significant blockage to coolant flow. Previously slumped drops that exceed the failure temperature of oxide shell will slump again. If this slumping occurs, all the drops are calculated to slump at the same time.

This is very briefly the description of the original LIQSOL model, designed for vertical rod geometry. The phenomenology of metallic relocation for horizontal bundles is qualitatively described in [9] based on high temperature experimental tests. In the bundle configuration tests, the typical local appearance of the metallic relocation is like in the photo reproduced as Figure 13. At the end of the test, the frozen drops of metallic melt are forming a mass common to more than one fuel element.

The new model for horizontal geometry consists actually in a series of modifications to LIQSOL model. The first assumption is the existence of inter-rods relocation. A new possibility is introduced: the third step of the original model (relocation) is allowed on components that do not have oxide scale breach or re-melting of previously frozen drops but are identified as horizontal relocation receivers from components which exceed the criteria. Fraction of a drop emerged from a fuel rod can come in contact other one fuel rod and even with the ‘shroud’ component surface modelling the pressure tube. Then, the drops will not change axial position; they can only travel along rods circumference.
The original version of the code allows to set the melt velocity to zero. This was done in
the calculations with the ‘as-received’ version of the code, to have the LIQSOL model work-
ing in what concerns the thermal processes without the change of axial position of the drops
which meet the slumping condition (default 2500 K was used in the calculations).

The new model uses without change the first step in LIQSOL modelling, that is the
cladding liquefaction and the dissolution of UO$_2$ by molten zircaloy. The philosophy of relo-
cation in drops and the radius of drops are preserved. Drops are formed from the intact clad-
ding one by one, separated in time by a re-slumping correlation originally dependent on the
temperature gradient between axial nodes. This was replaced with the time needed for the
drop at 0.5 m/s to run over half the outer circumference of the rod. For the actual dimensions
of CANDU fuel rods, this re-slumping time interval will be around 4E-2 s.

The above concerns the inter-element relocation mode, dominated by the liquid/solid in-
terfacial forces. The threshold between running and wetting behaviour of the emerging melt is
considered 1 weight % of oxygen in the cladding. Above this weight percent value the drop
will integrally belong to the component generating it.

The new model relocates a drop of U-Zr-O along the circumference of a fuel rod, for
running behaviour, until the drop encounters a narrow space (inter-elements contact) and be-
comes a mass belonging to two fuel rods. The drop splits according to an input matrix whose
coefficients are the component-to-component (including shroud) relocation fractions. This
square matrix is composed for the arrangement of components and number of rods in each
component, as in Table 1.

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<th>1</th>
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<th>ncomp</th>
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<td>0</td>
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</table>

The drop is presumed to appear at contact points between components because there is
experimental evidence that the oxide shell will be ruptured at such locations (see Figure 14).
This is due to the disappearance of any previous ZrO$_2$ layer in the contact zone caused by
the contact pressure. Anyhow, the existing oxide layer in the moment of bundle slumping should
be very thin because this slumping phenomenon is appearing early in the transient. It is true
that for the upper rods in the bundle, the extent of the inter-elements contact is smaller since
there is no weight above, but the mechanism for the appearance of melt at contact points
when the melting point of the mixture U-Zr-O is reached should be valid for the bundle as a
whole on the axial length. The small possible separation intervals in different axial locations
are considered irrelevant.

Since the LIQSOL model runs through the components one by one, the matrix defines
the mass fraction of the mixture materials (U,Zr,O) in the drop that will belong to compo-
ponents, from the lower numbered to the upper numbered components. So, for instance, component no. 1 is possibly producing relocation terms instantly for components 2 to ncomp (last component, which is a shroud), but there will be no relocation terms from component no. 2 (or higher) to component no.1. This means that the arrangement of the components inside the channel should be from no. 1 in the upper part of the geometrical cross section to ncomp, so that the drop running under the action of gravity may belong partially to the originating component and to other lower positioned components in the arrangement, including the shroud. The particular intact geometry arrangement used for testing the model on a bundle with 37 rods in CANDU6 channel is given in Figure 15. Physically, the drop is generated at one contact point (see Figure 16) and, if running, will stop at another contact point, in a lower position of the same rod. The coefficients of the matrix is composed by averaging over the rods of the same component, considering that half of the mass will stay with the originating rod and the other half with the rod participating in the contact point at which the drop stops.

Figure 15. Components identification inside the intact channel. Shroud is component no. 8.

Figure 16. Close-packed configuration with components and rods identification.

The use of the model assumes that metallic melt drops are released when an input temperature for failure of oxide shell is reached (40000300 card, W1), with a fraction for stable oxide shell = 1. (40000300 card, W2). This means that melt will be available for relocation at the input temperature regardless the degree of cladding oxidation. This input temperature can be as low as 2098-2125 K (or beta-Zr melt interval); both 2200 K and 2125 K were used in the tests. The re-slumping temperature of the previously slumped drops was set 50 K above the intact cladding slumping temperature, to avoid intermixing of intact cladding slumping and re-melting of drops, although the new model allows the coexistence of the two relocation modes. The physical reason is the increase of melting temperature of drops compared with intact cladding due to addition of oxygen.

The modifications in SCDAP do not make any changes to the model of drops oxidation. The total oxidation heat generation is treated with the LIQSOL original surface eq. for a hemisphere of 3.5 mm radius on the surface of the rod, and with the fractional number of drops $N_k$ in the formula:

$$q_{totk} = N_k \cdot 2\pi \cdot r_d^2 \cdot Q_{dk}$$

where: $Q_{dk}$ is the heat generation per unit surface area of drop located at node k (W/m$^2$);

$q_{totk}$ is the total heat generation due to oxidation of drops at axial node k (W); $r_d$ is the radius of drop (m); $N_k$ is the number of drops at axial node k.
The drops will tend to coalesce and reduce the area exposed to steam for the relocated material that is composed of repeated drops slumping; therefore preserving the original formula (3) may lead to an overestimation of drops oxidation which is not quantified yet.

There is also a treatment for the relocation inside a horizontal channel from fuel rods to shroud. The coding of this part of the model involved modifications in a greater number of SCDAP routines than the relocation between fuel rods only. The components are sending fraction of drops towards the pressure tube and the model calculates the heat addition due to the internal energy of the drops and also a crust that grows on the shroud due to mass addition. The model supposes that drops are freezing during time step on the inner face of the shroud. In SCDAP shroud is homogeneously treated, when actually, in case of a horizontal fuel channel, there is asymmetry, both thermally and in material relocation. So, the code identifies the fuel component neighbouring the lower shroud (component no. 7 in our example) and the material is added in the crust of this component, attempting an adequate channel obstruction.

For relocations to the shroud coming from fuel, nuclear heat transported by the \( \text{UO}_2 \) in the melt had also to be considered. Note that the relocation of shroud itself is not considered, so the \((\text{ncomp} \rightarrow \text{ncomp})\) coefficient will not be taken into account. Moreover, the old SCDAP model for zirconium shrouds relocation in vertical geometry was disabled.

In the original LIQSOL model, the existence of liquid drops at an axial position prevents the formation of a new drop. The new model renounces to this restriction considering that the existence of a not solidified drop in contact with the rod does not preclude the formation of a new drop, at contact points positioned higher than that at which the previous drop is stopped. This approach implied additional treatment in the LIQSOL routines in order to take into account the liquid drops mass and number from previous time steps.

The characteristics of the horizontal relocation model are summarized in Table 2.

All the above modelling assumptions describe the relocation on the outer surface, or the inter-elements relocation. The other relocation mode described in [10] has to meet two conditions:

- sufficiently oxygen dissolved (in the clad) to cause the wet of ceramic by the melt;
- availability of capillary volumes.

No treatment of this mode is provided because the second from the above conditions seems difficult to quantify. In the hypothesis of its existence, the intra-element relocation will incorporate the melt inside the cracks of the fuel pellet and will reduce the observed number of drops (or slugs) on the outer face of the rod. If modelling assumptions would become clear regarding the availability of capillary volumes, the modification of fuel pellet density, thermal properties and radial power distribution will be necessary, which might prove to be a rather difficult task.

The new model acts toward reducing the maximum temperature inside the channel as will be proved by results further on.

Table 2 Summary of main assumptions for horizontal relocation

<table>
<thead>
<tr>
<th>No.</th>
<th>Use of original LIQSOL options</th>
<th>New/Modified LIQSOL features</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Melt available for relocation regardless the oxidation degree and at lower temperature due to contact pressure</td>
<td>Matrix of inter-components relocation; fraction of drop transferred to lower located components in case of running behaviour of the melt; results in existence of melt on cladding without breach of its oxide</td>
</tr>
<tr>
<td>2</td>
<td>Axial melt velocity 0.0</td>
<td>New liquid drops without axial relocation or freezing of previous drops</td>
</tr>
<tr>
<td>3</td>
<td>Fixed time interval for drop by drop relocation</td>
<td></td>
</tr>
<tr>
<td>4</td>
<td>Relocation towards shroud</td>
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3.2 Bundle slumping and contact heat transfer at contact between components

3.2.1 Bundle slumping

The possibility of slumping for each individual bundle (axial node) inside the channel was introduced. An average temperature of the fuel is calculated for the node, based on the rods surface temperature and their respective number in the bundle. At exceeding input criterion (1673 K was used in the tests), slumping takes place and it will remain like that for the rest of the transient, because the phenomenon is not reversible.

This temperature criterion triggers both the modification of sub-channels area and hydraulic diameters inside the fuel channel and the heat transfer at direct contact between components that will be described in the next subchapter. One way to quantify the modification of channels at a temperature threshold is to produce restart records at moments chosen through a previous run of the case and to re-nodalize the model. But this method can prove unproductive even for one channel because different axial nodes will reach the threshold at different moments, and the job might be fragmented in different runs with re-nodalization input decks.

In order to automatically quantify the reduction of channel areas and the formation of by-pass steam channel due to bundle slumping, at the axial nodes at which the slumping condition is fulfilled, special treatment was necessary.

The area of the channels and the hydraulic diameter are modified, both for the fuel channels and for the by-pass channel according to input values. That means that the input deck should incorporate from the beginning the by-pass channel, with small volume areas (but with internal junction areas equal to the new area of the by-pass volume after bundle slumping). Identification of by-pass volumes is provided by input, too.

3.2.2 Contact heat transfer

As already mentioned, it is produced after exceeding the input criterion for average temperature of the bundle. A metal-metal conduction term is introduced:

\[
q_{mm,j} = n_{c,j} \cdot h_{cd} \cdot (t_{s,j} - t_{s,k}) / L_k
\]

(4)

where: \( q_{mm,j} \) is the heat transfer from the axial node \( k \) of component \( j \) (W/m); \( h_{cd} \) is the power transferred by metal-metal contact per axial node and temperature degree (W/K); \( n_{c,j} \) is the number of contact points between component \( j \) and component \( i \); \( t_{s,j} \) is the surface temperature of the axial node \( k \) of component \( j \) (K); \( t_{s,k} \) is the surface temperature of the axial node \( k \) of component \( i \) (K); \( L_k \) is the length of the axial node \( k \) (m).

The matrix of contact points \( n_{c,j,i} \), \( i=1\text{,ncomp} \) and \( j=1\text{,ncomp} \) should be composed by the user, averaging for rods in a fuel component to find how many points they have with their neighbours that are fuel rods or shroud, using a close-packed configuration as in Figure 14.

\( h_{cd} \) should be an input parameter and its value is roughly 1 W/K for 2000 W/m²K contact conductance, contact angle \( \psi = 5^\circ \), and 0.5 m length of the axial node and a fuel radius of 0.0065 m.

With the metal-metal conduction term the power density in the clad of the fuel is modified:

\[
q_{cl,dt} = o_{x,dc} + q_{cdr,ps} + q_{cdr,gs} - q_{mm}
\]

(5)

where: \( q_{cl,dt} \) is the cladding power density (W/m); \( o_{x,dc} \) is the oxidation heat generation of clad (W/m); \( q_{cdr,ps} \) is the oxidation heat generation of the solidified drops (W/m); \( q_{cdr,gs} \) is
the heat transferred into fuel rod by flowing drops of mixture (W/m); $q_{mm}$ is the metal-metal heat transferred by conduction (W/m).

Metal-metal term concerning the shroud is taken into account in the corresponding power density relationship for this component type.

### 3.3 Testing the new models

The horizontal melt relocation and the metal-metal models (without bundle slumping) could be tested on the same input model presented already (Figure 3). The maximum cladding temperature in the transient for metal-metal model alone with 1 g/s steam is shown in Figure 17, and with horizontal relocation, for the 20 g/s steam, is presented in Figure 18. Both time evolutions are plotted with different values of $h_{cd}$ (different values for contact conductance) for the maximum rated channel (7.3 MW). It is expected $h_{cd} = 1$ W/K to be a reasonable value whereas $h_{cd} = 2.5$ W/K represents only a higher value to highlight the effect of the model.

![Figure 17. Effect of metal-metal model on maximum clad temperature. Flow: 1g/s. First input deck.](image)

![Figure 18. Effect of metal-metal model and horizontal relocation model on maximum clad temperature. Flow: 20 g/s. First deck.](image)

Concerning the effect of the horizontal relocation model in Figure 18, there is an important limitation of the temperature excursion inside the channel compared to the same case with original code version (see Figure 6). On one hand, there is a reduction of the surface of metal exposed to steam by drop formation and on the other hand a quick transfer of drop fraction to components that are colder. No molten pool formation is seen with the new model.

Testing the bundle slumping model and by-pass channel formation is not possible with the simple input deck that allowed obtaining all the SCDAP/RELAP5 results up to this point. A new input deck was produced, because it was necessary to define mixing volumes between the bundles in order to quantify separately individual bundle slumping. This was done using a large number of single volumes and single junctions. The schematic drawing of the second input deck is given in Figure 19. The power distribution inside channel (maximum bundle/average bundle = 1.415) is applied here on the active length of the fuel, making room for the mixing volumes of 1.8 cm length, while in the first input deck, power is applied over the entire length of the channel. Both with the first and the second input deck the heat transfer model option for drops freezing is active. With the second input deck calculations, the temperature for ZrO$_2$ layer failure was set to both 2200 K and 2125 K. The effect of modifying this temperature is seen in the relocated melt composition; lowering it reduces the UO$_2$ molten fraction in the drops.

Figure 20 presents synthetically the effects of the new models on the code performance for the transient with 20 g/s steam. In this graph, ‘all-models’ curves describe maximum clad
temperature with horizontal relocation, bundle slumping and metal-metal models simultaneously. The input deck with mixing volumes and by-pass channel was used.

Figure 19. Diagram of the second input deck with mixing volumes and by-pass channel.

Figure 20. Effect of new models. Flow: 20 g/s. (a) \( T_{\text{ZrO}_2} = 2200 \) K (b) \( T_{\text{ZrO}_2} = 2125 \) K Second input deck.

Using the two input models (with and without mixing volumes) we obtained: a) no melting of the cladding for low steam flow rates (1 g/s); b) significant metallic melt relocation for high steam flow rates (20 g/s) when bundle slumping model is not active, with metal-metal model active on first deck (Figure 18), and without metal-metal model on second input deck (Figure 20, ‘horizontal relocation’ curve); c) using all new models relocation melt is seen up to \( T_{\text{ZrO}_2} \approx 2190 \) K. Beyond that, the maximum temperature does not exceed the relocation threshold. In all calculations, the procedure to simulate thermal transfer for sagged pressure tubes described in 2.2 was applied.

4 CONCLUSIONS

The above presented work has two interlinked parts. First is the modelling of the early phase of a LOCA/LOEC in a CANDU6 fuel channel with ‘as-received’ version of RELAP5/SCDAPSIM. It appears from the calculation results that the code can model this accident up to severe degradation. Then, some new models dedicated to horizontal geometry and to deformation phenomena specific to this reactor design and the results of testing them are presented. These are the first results with the new models which have no validation yet. With the coding modifications related to the addition of the new models the first steps have been made towards a SCDAP/RELAPSIM version for CANDU.

It has to be mentioned that the developing stage of the models allows the treatment of one radiation enclosure only. Using them for reactor applications would require more efforts. Nevertheless, applying the models for one channel should be enough for attempting their validation which will hopefully constitute the next step in continuing the work. For that purpose, bundle heating experiments with oxidation and formation of melt relocation would be suitable but data are not available yet. Comparison against experimental data can underline the degree of adequacy of the physical modelling assumptions, especially for the horizontal metallic relocation model: relocation in drops, oxidation of drops in the form of hemispheres, neglecting the intra-element relocation mode, etc.

All the calculations in the paper assume that moderator is surrounding the outer wall of the fuel channel (calandria tube). Also, the horizontal relocation model makes this assumption when calculating the freezing of relocation melt coming from fuel once it contacts the shroud. Uncovered fuel channels may not fulfil the solidification condition of the molten slugs which
drop to the lower region of the pressure tube. Accordingly, if there is a loss of moderator inventory during the contingent metallic melting period, the horizontal relocation model should be improved with respect to the shroud treatment. However, after the channel uncovery, its disassembly will be rather fast as the outer face of calandria tube is exposed to steam and is rapidly oxidizing.

A more important improvement for describing the CANDU channel severe accidents phenomena would concern the existence of a non-symmetrical ‘shroud’ component. As pointed out, actually there is asymmetry, both thermally and in material relocation, for the shroud. The treatment in the horizontal relocation model takes into account an axially symmetric shroud by calculating a homogenous crust and by reducing sub-channel area of a neighbouring component due to relocation to shroud. Creating a non-symmetrical ‘shroud’ component will lead to the modification of these modelling assumptions. Probably, a non-symmetrical ‘shroud’ component would bring the calculated temperature distribution inside the channel closer to real case distribution.

Results with the new models, where the maximum case from the ‘as-received’ version calculations (20 g/s) was analyzed, suggest that transition towards ceramic melting may not be reached in the first phase of the accident. Nevertheless, the results so far do not rule out a hypothetic configuration of inter-components relocated frozen drops of molten material that might plug individual sub-channels and block the local cooling of the fuel rods. If such a case arises from further studies or from experiments, the need to modify the relocation of debris and molten pool in horizontal geometry will arise too.

An even further step of SCDAP development for the CANDU severe accident studies concerns the failure of the ‘shroud’ type component after the channel uncovery and the subsequent treatment of the fuel channel.

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ABSTRACT

The objective of the current study is to investigate the capability of the US NRC code TRACE to predict the CHF position and temperature profiles for different axial heat flux distributions in the reflooding of a hot single tube. Measurements of each of Bennett, Keeys and Becker were used for this. Hydrodynamic and post-dryout heat transfer calculations were performed using TRACE Code. CHF and critical quality correlations (based on the ‘look-up’ tables of Groeneveld, the ‘local conditions’ hypothesis, and the boiling length/quality relationship) are usually implemented in system codes. Each of these has been used to analyse the experiments. These simulations showed that generally the CHF position was well predicted whereas the estimation of the wall temperature was not correct for particular ranges of mass and heat fluxes. This is investigated and possible causes, associated with ‘local conditions’ issues, are proposed.

1 INTRODUCTION

The integrity of the cladding of the nuclear fuel of Light Water Reactors (LWR) as first containment barrier is of high importance in nuclear safety. The cladding is intended to operate with a temperature less than 2200°F (~1477°K) to avoid both oxidation and radial expansion. The situations for which such a temperature can be reached are numerous. A typical description is the presence of a blanket of vapour surrounding the pin inducing a very sharp decrease in heat transfer and consequent temperature increase.

While the design intent is generally to avoid this condition, predicting the Peak Clad Temperature (PCT) is important. Geometric factors and operating parameters (pressure, power profiles, mass flow rates, inlet sub-cooling to name a few) affecting the Critical Heat Flux (CHF), the maximum heat flux applied to a solid surface for which unstable vapour patches appear, have been investigated in the past [1-3] and resulted in many correlations.

Hence, it has become increasingly difficult to select the correct CHF prediction methods that can be used for a variety of conditions and geometries. Additionally, a lack of
knowledge of the undergoing heat transfer regimes encountered in the post-CHF region has limited the development of theoretical models.

This present work intends to first provide a study of the main correlations implemented in system codes, (such as the 1995 Groeneveld look-up table [1], the Biasi and CISE-GE correlations). A focus will be on the quench front (CHF location with a wall superheat of about 200°K [2]) and the temperature profiles in the Dispersed Flow Film Boiling (DFFB) region. The computer code used as a basis for these simulations is the thermal-hydraulics system code TRACE, developed by the US NRC. A feature of the TRACE code, the fine mesh technique, was used in order to gain in accuracy; we will investigate this issue.

Along these same lines, the above-mentioned CHF prediction methods will be assessed in the case of uniform and cosine axial-heat flux distributions (AFD) for a range of pressures and mass fluxes in tubes.

In a second part, we will discuss the modelling of the post-CHF heat transfer and the sensitivity of the code to the different correlations accounting for the interfacial droplet drag, the interfacial heat transfer and the wall heat transfer.

Finally, we have performed some calculations with a selected set of post-dryout heat transfer correlations where a ‘Blowing’ factor accounting for the effect of the mass transfer upon the heat transfer was added to the interfacial droplet drag and the interfacial heat transfer. A two-phase enhancement factor, which represents the enhancement of the convective heat transfer in the expression of the wall-to-gas heat flux due to the presence of the dispersed droplets in the superheated vapour flow, is commonly implemented and tuned in system codes to fit data. In here, we will conserve its original form and study its influence on the de-superheating of the vapour phase.

### Nomenclature

- \( B \) Blowing factor
- \( c_p \) Specific heat (J/kg-K)
- \( C_D \) Drag coefficient
- \( D \) Diameter
- \( D_H \) Hydraulic diameter (m)
- \( F \) Friction factor
- \( g \) Gravitational acceleration (m/s\(^2\))
- \( G \) Mass flux (kg/m\(^2\)-s)
- \( Gr \) Grashof no.
- \( k \) Conductivity
- \( K \) Dummy factor
- \( L_B \) Boiling Length (m)
- \( Nu \) Nusselt no.
- \( P \) Pressure (Pa)
- \( Pr \) Prandt no.
- \( q^* \) Heat flux (W/m\(^3\))
- \( Re \) Reynolds no.
- \( T \) Temperature
- \( V \) Velocity (m/s)
- \( x \) Quality

### Greek symbols

- \( \alpha \) Void fraction
- \( \beta \) Exponent
- \( \mu \) Viscosity
- \( \rho \) density
- \( \psi_{2\phi} \) Two-Phase enhancement factor

### Subscripts

- \( CHF \) Related to CHF
- \( conv \) Related to convection
- \( d \) Related to droplet
- \( FC \) Related to forced convection
- \( g \) Related to gas
- \( l \) Related to liquid
- \( lam \) Related to laminar flow
- \( rad \) Related to radiation
- \( sat \) Related to saturation
- \( turb \) Related to turbulent flow
- \( v \) Related to vapour
- \( w \) Related to wall
2 CRITICAL HEAT FLUX AND POST DRYOUT HEAT TRANSFER

2.1 Critical Heat Flux (CHF) prediction methods

As noted, the critical heat flux (CHF) is the heat flux for which a heated surface becomes vapour-blanked with increasing heat flux inputs. The vapour patches/blanket resulting from this excess in heat flux leads to a deterioration of the convective heat transfer exchanges between the wall and the coolant. The superheat beyond the CHF point could significantly be of the order of 600°K.

In general, three approaches are adopted in the prediction of the CHF:
- The 1995 AECL\textsuperscript{1}-IPPE\textsuperscript{2} “look-up” tables [1],
- Flux-quality relationships, and
- Boiling length/quality relationships.

The differences between the two latter relationships are reported in [3].

It is usually accepted that the Groeneveld look-up tables comply with the margin requirements imposed by both nuclear regulatory bodies and energy producers. However, the main assumption is that CHF can be determined on local conditions basis, i.e.:
\[
\dot{q}_{\text{CHF}}' = K \cdot fn\{P, G, x, D_H\}
\]

where \(\dot{q}_{\text{CHF}}'\) is the critical heat flux, \(K\) is a product of factors accounting for the geometry, low flow conditions etc. \(P\) is the pressure, \(G\) the mass flux, \(x\) the quality and \(D_H\) is the hydraulic diameter. From Eq. (1), it could be understood that the Groeneveld et al. tables [1] imply a specific relationship between CHF and local quality for a given mass flux and pressure. Nevertheless, the relationship between heat flux (\(\dot{q}_{\text{CHF}}'\)) and quality (\(x\)) is not a universal one and this is amply proved by data for non-uniform heat flux, exemplified by some Refrigerant-114 data from Shiralkar [4].

The two other approaches, namely the \(F\)-factor and the boiling length approaches, allow better results to be obtained.

The \(F\)-factor method developed by Tong et al. [5] takes into account the influence of the non-uniformity of the heat flux upon the prediction of the CHF. The so-called \(F\)-factor is defined at a given enthalpy as the ratio of a uniform heat flux (CHF) to the non-uniform CHF. As this approach assumes the importance of the effect of the upstream heat flux profile on the local CHF, it has to be noted that at low qualities, the ‘memory effect’ is small, and thus the CHF is conditioned by the local heat flux. At high qualities, this same ‘memory effect’ is sizeable and the average heat flux fixes the CHF position.

On the other hand, the boiling length/ critical quality (\(x_{\text{CHF}}\)) relationship introduced by Bertoletti et al. [6] is expressed as:
\[
x_{\text{CHF}} = K_i \frac{A \cdot L_H}{B + L_B}
\]

where \(L_H\) is the boiling length, i.e. the distance from which the fluid reaches saturation, \(x=0\); \(K_i\) is a factor accounting for the radial peaking factor and, in the case of the Biasi correlation, accounting for the ratio between the heated perimeter to the wetted perimeter. ‘\(A\)’ and ‘\(B\)’ are two system-dependent functions, i.e.: \(A = fn\{P, G\}\) and \(B = fn\{P, G, D_H\}\).

\textsuperscript{1} AECL: Atomic Energy of Canada Limited.
\textsuperscript{2} IPPE: Institute of Power and Physics, Obninsk, Russia.
In the TRACE code, the CHF position is essentially determined by the 1995 look-up tables (notwithstanding the non-physical nature of the local condition hypothesis). The reason is that the overall average and the RMS errors associated with the adoption of this approach are 0.38% and 8.17% respectively, which in practice, is difficult to better. In addition to this prediction method, the user can combine the prediction of the CHF with the CISE-GE [7] or Biasi [8] critical quality correlation. The critical quality value is only used as a transition criterion from pre to post-CHF while the CHF point \( (\dot{q}_{CHF}^*, T_{CHF}) \) remains determined by the Groeneveld et al. tables [9]. No attempt was made to modify the determination of the CHF point by other means than the look-up tables.

In part 2, we present results of the effect of the different correlations described above on the temperature distribution.

### 2.2 Dispersed Flow Film Boiling (DFFB)

The regimes present in the post-CHF region are the inverted annular flow and/or the dispersed flow film boiling (dispersed droplet mixture in a superheated continuous vapour phase). Here we discuss only the latter given the high void fractions \( \alpha \geq 0.80 \) encountered above the dryout location. The modelling of DFFB heat transfer is, in computer codes, a three step process: wall-vapour heat transfer, vapour-drop convection, and wall to drop heat transfer.

#### 2.2.1 Wall heat transfer

The wall heat transfer is a combination of forced convection to the superheated vapour and thermal radiation to the liquid droplets. Therefore, the heat flux is expressed as follow:

\[
\dot{q}_{DFFB}^* = \dot{q}_{wg,FC}^* + \dot{q}_{wg,rad}^* + \dot{q}_{wl,rad}^* + \dot{q}_{wd}^*
\]  

(3)

The wall temperature is often greater than the Leidenfrost temperature, the droplet-to-wall contact could be considered by an enhanced Forslund-Rohsenow correlation at high vapour Reynolds no. \( (Re_v \geq 4000) \):

\[
h_{dw} = 0.00638(Re_v - 4000)^{0.6}(1 - \alpha)^{2/3} \left[ \frac{k_v h_{fg} g \rho_d \rho_v}{(T_w - T_g) \mu_s D_g} \right]^{0.25}
\]  

(4)

The convective wall-to-gas heat transfer \( \dot{q}_{wg,FC}^* \) is usually calculated by making the key assumption that the heat transfer is similar to single-phase forced convection heat transfer. A correction factor \( (\Psi_{2\Phi}) \), which accounts for the two-phase heat transfer exchanges enhancement due to the presence of the dispersed droplets in the continuous vapour phase, is added to the Gnielinski correlation [10]:

\[
\dot{q}_{wg,FC}^* = \frac{k_v}{D_H} \cdot Nu_{wg,FC} \cdot \Psi_{2\Phi} \cdot (T_w - T_l)
\]  

(5)

where:

---

3 In highly turbulent flows (and at high pressure), the droplet radial velocity increases, therefore, droplets-wall interactions should not be neglected since they participate to the rewetting hot walls.

4 At the moderate Reynolds nos. entrance length effects can enhance the wall heat transfer but are neglected in this present study.
In Eq. (6) \( f \) is a friction factor, and:

\[
\Psi_{2\Phi} = \left[ 1 + \frac{(1-\alpha) \cdot g \cdot \Delta \rho \cdot D_L}{2 \cdot f_w \cdot \rho_g \cdot V_g^2} \right]^{1/2}
\]

where \( f_w \) is the wall friction factor and \( V_g \) is the gas velocity.

In general, the axial gradient of vapour superheat first increases and then, due to the precursory cooling effects, decreases with increasing axial distance downstream from the quench front. The effect of heat transfer on the vaporization of the liquid phase could be caused by either heat transfer from the superheated vapour to the entrained liquid or from heat transfer from the superheated wall to the liquid. Hence, the interfacial heat transfer between the vapour phase and the droplet mixture plays a key role in determining the maximum superheat of the vapour, which in turn limits the Peak Clad Temperature (PCT). We will discuss this point in the next section.

### 2.2.2 Drop drag and interfacial heat transfer

The modelling of the dispersed phase requires a particular attention in regard to the interfacial droplet drag and the interfacial heat transfer. A critique of the key assumptions in the modelling of DFFB regimes in multiphase flow system codes can be found in [11]. We will only mention the assumptions relevant to our analysis.

First, for the calculation of both interfacial drag and heat transfer, it is assumed that the cloud of droplets of a range of sizes can be represented by a single representative spherical droplet with the same area to volume ratio as that of the entire population. The physical mechanisms that affect the droplet size distribution (impingement on the wall, evaporation, break-up, coalescence) are ignored. The droplet size is taken to depend only on local conditions, namely relative velocity and fluid properties and is calculated from the local critical Weber no. Thus, the number density of droplets generated at the locus of CHF varies in the computational domain whereas this transport of the droplet population should be history-dependent and consistent.

Droplet-to-droplet interactions are incorporated through empirical approximations. In TRACE, for example, this is accomplished through the evaluation of an empirical correlation based on the local flow conditions and is calculated using the correlation of Kataoka, Ishii and Mishima [12]. In some other computer codes, the recent trend of multi-field modelling intends to solve this problem [13].

Moreover, since the droplets are mainly generated at the quench front where the fluid properties are at saturation, the droplet temperature is expected to be at saturation (no conduction/convection inside the drop). Only heat transfer exchanges on the vapour side are considered. Then, the interfacial heat transfer coefficient could be based either on the Lee and Ryley correlation developed at atmospheric pressure:

\[
Nu_d = 2 + 0.74 \Re^{1/2} \Pr_v^{1/3}
\]

Or on the correlation suggested by Ranz and Marshall [14]:

---

5 This representation is limited, in fact, a same volume may produce different interfacial.
The drag coefficient is expressed as:

\[ C_D = \frac{24}{Re_d} \cdot (1 + 0.1 \cdot Re_d^{0.75}) \]  

Studies on the sensitivity of the temperature distribution to different drag coefficients and interfacial heat transfer correlations are presented in the results section. The main modification is the addition of the so-called ‘Blowing factor’ accounting for the effect of mass transfer upon heat transfer in the expressions of drag coefficient and Nusselt no.

2.3 Experiments

From the experimental studies available in the literature, we have selected the Bennett et al. [15] data set for uniformly heated tube. To investigate the effect of the axial heat flux distribution, we use the Keeyes et al. [16] tests for which a cosine heat flux distribution was applied to a tube. Thirdly, the Becker et al. [17] study on the post dryout heat transfer which, for instance, reveals the importance of pressure on the dryout location, has been modelled.

The objective of the Bennett experiment was essentially to obtain a measure of the wall surface temperature in the region beyond the dryout point. The dryout ‘interface’ between the heated wall and the dry regions was investigated and has led to the demonstration of the non-equilibrium effects associated with developing flows in heated tubes. The variables reported were the mass flux, system pressure, wall heat flux, inlet sub-cooling and quality. Systematic experiments in a 12.6 mm internal diameter tube were conducted at the constant pressure of 6.89MPa. At a given mass flow rate, the heat flux was varied by an increment of \(\sim 1\%\), and the axial temperature profiles measured.

In another UKAEA study, Keeyes et al. in 1971 examined the post-CHF heat transfer in a 3.66m long tube with a cosine heat flux distribution (with a form factor of 1.4). The data set are summarized in table 1. These two experiments demonstrated the dependency of the local critical heat flux on the axial heat flux distribution. In fact, at the same dryout quality, the CHFs of the non-uniformly heated tube were noticeably lower than those measured on the uniformly heated tubes. Hence the necessity of modelling the post dryout heat transfer in a non-uniformly heated tube by taking into account the flow history, in other terms: the boiling length.

Becker et al. [17] conducted at the Royal Institute of Technology (RIT) in Stockholm post dryout experiments where the wall temperatures were measured on a set of 7 m long electrically heated tubes with inner diameters of 10, 14.9 and 24.7 mm. The tube was cooled by upwards flow of water with mass fluxes from 500 to 3000 kg/m2-s. However, in this present work, the cases selected covered pressures ranging from 3 to 14MPa, heat fluxes from 400 to 1060kW/m² and inlet sub-cooling from 8.5 to 12K.

Table 1: Experimental data

<table>
<thead>
<tr>
<th>Year</th>
<th>Experiment</th>
<th>P (MPa)</th>
<th>G (Kg/m²s)</th>
<th>D (mm)</th>
<th>L (m)</th>
<th>X</th>
<th>Q (MW/m²)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1967</td>
<td>Bennett</td>
<td>6.9</td>
<td>393-5235</td>
<td>12.6</td>
<td>5.56</td>
<td>-</td>
<td>0.35-1.84</td>
</tr>
<tr>
<td>1972</td>
<td>Keeyes</td>
<td>6.9</td>
<td>700-4100</td>
<td>12.7</td>
<td>3.66</td>
<td>0.15-0.95</td>
<td>0.8-1.5</td>
</tr>
<tr>
<td>1983</td>
<td>Becker</td>
<td>3-20</td>
<td>500-3000</td>
<td>14.9</td>
<td>7.0</td>
<td>0.03-1.60</td>
<td>0.1-1.25</td>
</tr>
</tbody>
</table>

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3 SIMULATIONS

3.1 TRACE (TRAC-RELAP Advance Computational Engine)

TRACE is a thermal-hydraulics computational modelling code for nuclear power systems developed by the US Nuclear Regulatory Commission (US-NRC). TRACE is a code designed to analyze reactor transients and accidents up to the point of fuel failure. The partial differential equations that describe the two-phase flow and heat transfer are solved using the finite volume numerical methods. The heat-transfer equations are evaluated using a semi-implicit time-differencing technique while the Navier-Stokes equations in the spatial one-dimensional (1D) and three-dimensional (3D) components use, by default, a multi-step time-differencing procedure that allows the material Courant-limit condition to be exceeded. In this present paper, all results presented arise from one dimensional modelling.

3.2 Nodalization and Fine Mesh Technique

Nodalization studies with shorter test section cells show that decreasing the length from 0.30 m to 0.10 m did not noticeably increase the accuracy in predictions as illustrated in figure 1. CHF and temperature profile predictions require correct numerical and physical models. The approach adopted by developers is to improve the numerical techniques implemented in two-fluid codes such as TRACE in order to track both the variations of the void fraction (level tracking methods) and the thermal gradient at the quench front (fine mesh rezoning techniques) [9].

![Figure 1. Nodalization study for the Bennett run 5353, on the left, the case was run with the fine mesh technique activated. On the right, the nodalization shows that a number of 20 cells is sufficient.](image)

With a fine mesh technique, the axial resolution of the axial gradient involves the insertion of transitory nodes whenever the difference between adjacent nodes exceeds a heat transfer regime dependent value $\Delta T_{\text{max}}$, usually of the order of 25°C. The figure 2-c, modelling the Bennett run 5358, shows indeed a good agreement, the predictions fit perfectly the data at low mass fluxes and low heat fluxes while using the Biasi correlation. The analysis of the output reveals that even though the last part of the graph tends to be similar, the heat transfer regimes are different: we encounter the DFFB in the run performed with the fine mesh technique and single phase vapour heat transfer in the other.

A maximum number of 1000 transitory nodes and a minimum distance of 1 mm between two adjacent nodes were imposed. These criteria suggest that the technique is consistent with the space-averaging theory on which is based the two-fluid formulation [18]. In the selected data, the regimes are liquid and vapour single phase, nucleate boiling, transition boiling and...
DFFB. With a drop size lower than 1 mm, we are assuming that the fine mesh technique is suitable for DFFB. According to this preliminary study, most of runs were performed using this fine mesh technique, generally used in transient and quasi-steady reflood-like situations.

However, those calculations failed to predict well the position of the CHF and the temperature profile in the dryout region; at this stage, one could question the validity of the physical models implemented in system codes. The precursory cooling effects (due to slugs and droplets generated at the quench front) could be important beyond the CHF; this phenomenon controls the wall temperature by essentially cooling down the superheated vapour. Recent work in this area has been carried out by Andreani and Yadigaroglu [19-21] and contributed to a better understanding of the physics involved in the post-dryout region.

Also, the wall-droplet interactions could be significant, but they are usually not modelled. In fact, the wall temperature in the DFFB regime is often above the temperature at which a droplet will wet the wall (the Leidenfrost point). The recent strategy to tackle this physical problem is to implement a multi-field model [22, 23] and to have a better treatment of the interfacial area by solving the interfacial area transport equation [24-26] coupled to a macroscopic $k-\varepsilon$ model [27].

3.3 Simulations, results and comparisons

The code assessment process is based on the comparison between the computed predictions and experimental data in order to validate individual physical models. In this paper, we are interested in the prediction of the critical heat flux position but also in the interfacial droplet drag, interfacial heat transfer and wall heat transfer in the post dryout region. In the following results, we have added when necessary a Blowing factor $B$ to account for the evaporation process. In fact, the drag dependency on the mass efflux (evaporation or condensation) is real and proved.

In TRACE, the model of Renksizbulut and Yuen [28] is attended to re-transcribe the effect of mass transfer upon heat transfer.

3.3.1 CHF and critical quality correlations

The determination of the CHF location is crucial and determines the extent of the temperature rise in the post-CHF region. As mentioned earlier, we used here the different options available in the TRACE code:

- AECL-IPPE (Look-Up Tables),
- AECL-IPPE with the Biasi critical quality correlation,
- AECL-IPPE with the CISE-GE critical quality correlation.

The latter was developed for rod-bundle geometries and is not expected to provide the best fit to data. Moreover, the correlation is adequate only for mass fluxes in the range: $300 \leq G \leq 1400$ (kg/m$^2$·s). On the contrary, the Biasi correlation developed for tubes is stated to be valid for mass fluxes in the range: $100 \leq G \leq 6000$ (kg/m$^2$·s); we recall here that the Biasi correlation was corrected to account for the conversion from ATA to Bar units [29]. Again, using the Bennett run 5359, the comparison (Figure 2-d) establishes that both the look-up table method and the Biasi critical quality predict satisfactorily the data. However, the simulation of a Keeys test at high power and high mass flux shows that none of the correlation predicts the dryout point. The CISE-GE correlation is the best prediction for this particular run despite the fact that it was primarily developed for rod bundle geometries.

As the Biasi critical correlation is based on the boiling length approach essentially accounting for the history of the flow upstream from the dryout point, it was used for our simulations.
An important step in the code assessment process is to check whether TRACE is capable of estimating the dryout point for uniform and non uniform heat flux distributions. Figure 2 demonstrates that TRACE is able to predict well the CHF location and the temperature profile for a uniform heat flux at low mass flux \((G = 380 kg/m^2s)\) for different heat inputs.

![Figure 2. CHF position and temperature predictions for a uniform heat flux (Bennett Data).](image)

In the case of non-uniform heat flux, the prediction is less good. Indeed, we notice on figure 3-b that the position of dryout is over-predicted by about 0.6m.

![Figure 3. CHF position and temperature predictions for a non-uniform heat flux (Keeys data).](image)

The pressure also influences the prediction of the dryout point and the consequent post dryout heat transfer exchanges. At constant mass flux and power input, an increasing pressure favours the occurrence of CHF; the predictions of the Becker tests confirm this behaviour (figure 4). Downstream from the dryout point, the temperatures appear to tend to an asymptotic value. This could be explained by the fact that in this region, only vapour exists and some sort of equilibrium homogenises the thermal exchanges.
Figure 4. Effect of pressure on the dryout position at a constant mass flux of 1500kg/m²-s and at a constant heat flux of 760kW/m² (Becker data).

Figure 5. Dryout position and temperature profiles for the calculations of the Becker tests at constant pressure (14MPa) and varying mass fluxes (from the top left corner to the right bottom corner: 1977, 1970, 1494, 1006, 503 kg/m²-s).

In short, these previous results demonstrate the ability of the code to predict the CHF location and post-dryout heat transfer reasonably well at low mass fluxes and low uniform heat fluxes using the Groeneveld et al. [1] ‘look-up tables’ with the Biasi critical quality correlation. Indeed, in Figure 5 and 6, one can notice that at high heat fluxes and high mass fluxes, the CHF is still well predicted but the distribution of temperatures is now less well predicted. Also, at a given heat flux and pressure, and for conditions of low mass velocities, the code fails to estimate either the CHF position and/or the temperature profile. It is worth noting that when the CHF position is forced by the authors to occur at a given elevation...
(Figure 6-b), the temperature distribution beyond the dryout point does not coincide with experimental data and the PCT is under-predicted. Indeed, two reasons could be suggested, (1) non-equilibrium effects become significant and hence the reduction in accuracy and (2) upstream history effects are predominant in the development of film boiling.

Since the CHF point was, in most cases, well estimated; we decided to focus on the DFFB regime by implementing modified and/or new models into the thermal-hydraulics TRACE code.

![Figure 6](image1.png)

Figure 6. Comparison between calculated temperature profiles and: (left) the Bennett experimental run 5379 (P=6.9MPa, G=3800kg/m²-s and Power=377kW); (right) the Keeyes data (P=6.9MPa, G=2000kg/m²-s and Power=191kW) while the computed CHF position is fixed

### 3.3.2 Two-phase enhancement factor

In this section, the two-phase enhancement factor ($\Psi_{2\Phi}$) is used in the following form:

$$\Psi_{2\Phi} = 1 + K_{\psi} \left[ \frac{(1-\alpha) \cdot Gr_{2\Phi}}{Re_{g}^{2}} \right]^{n}$$

(11)

where $Gr_{2\Phi}$ is the Grashof no. ($Gr_{2\Phi} = (\rho_{g} \cdot g \cdot \Delta \rho \cdot D_{h}^{2} / \mu_{g}^{2}$)), $K_{\psi}$ is a wall friction factor dependent coefficient ($2 \leq K_{\psi} \leq 100$) and the exponent $n$ takes either the value of 0.5 or 0.85 depending on the droplet diameter considered. While $K_{\psi}$ is too high, the two-phase enhancement factor becomes important in Eq. (5) causing a higher heat flux and then under-predicting the PCT. However, as it could be noticed in figure 7, the TRACE is not sensitive to a change of values for $K_{\psi}$.

![Figure 7](image2.png)

Figure 7. Effect of different $K_{\psi}$ values on the temperature distribution for the Becker run 232 (P=10MPa and G=1500kg/m²-s)
3.3.3 Interfacial Droplet Drag

As mentioned in part 2, the evaluation of droplet drag is crucial. This encouraged us to investigate the impact of different correlations on post-CHF regimes. For conditions of constant relative velocity, the droplet drag $C_D$ is related to the droplet Reynolds no. as follow:

$$
\begin{align*}
0 < Re_d < 2 & \quad C_D = 24 / Re_d \\
2 \leq Re_d < 500 & \quad C_D = 0.4 + 40 / Re_d \\
500 \leq Re_d \leq 10^5 & \quad C_D = 0.44
\end{align*}
$$

(12)

As an alternative to the expression of the droplet drag $C_D$ in Eq. (10), we used a drag law which has been obtained in the context of an accelerating cloud of droplets; Ingebo [30] has studied particle acceleration and has correlated a wide range of data by the expression:

$$
C_D = \frac{27}{Re_d^{0.84}}
$$

(13)

Indeed, a model for accelerating cloud of droplets is likely better to represent the physics involved. Thermal gradients can cause the continuous vapour phase to be accelerated increasing the droplet mean velocity.

In order to account for the effect of evaporation on the drag coefficient, a factor has been added to the previous expression:

$$
C_{Dnew} = \frac{C_D}{(1 + B)^{0.2}}
$$

(14)

where the blowing factor $B$ suggested by Renksizbulut and Yuen [28] is:

$$
B = \left( \frac{c_{P,g} \cdot (T_g - T_{sat})}{h_{fg}} \right) \left( 1 - \frac{q_{ad}^*}{q_w} + \frac{q_{rad}^*}{q_{conv}} \right) \approx \left( \frac{c_{P,g} \cdot (T_g - T_{sat})}{h_{fg}} \right)
$$

(15)

From the calculations performed (figure 8), one can state that the droplet drag coefficient does not have a significant effect on the PCT. The addition of a blowing factor increases the PCT by no more than 5°K.

Figure 8. Influence of the droplet drag on the temperature profile prediction for the Becker Run 232 (P=10MPa and G=1500kg/m²-s).
3.3.4 Interfacial Heat Transfer

Despite the minor effect of the blowing factor on the droplet drag coefficient (figure 8) and therefore on the temperature distribution; figure 9 reveals the central role of the single particle Nusselt no. in the temperature and CHF predictions. The Renksizbulut and Yuen correlation is the correlation implemented in the TRACE code. We have implemented few more correlations (Ranz-Marshall, Lee and Ryley and Beard and Pruppacher) to quantify their respective influences.

![Figure 9. Comparison of different interfacial heat transfer correlations in the DFFB region for the Becker Run 232 (P=10MPa and G=1500kg/m²-s).](image)

The results obtained with the different correlations show that the best agreement was obtained with a modified version of the Beard and Pruppacher (BP) correlation[31] to account for the mass evaporation:

\[
\frac{1.56 + 0.616Re_d^{1/2}Pr_d^{1/3}}{(1 + B)^\beta} = \text{Nu}_d
\]  

where 0.07 < \beta < 2.79.

Eq. (16) determines the single-particle Nusselt no. In our calculations, a multi-particle Nusselt no. is used; this latter contains an additional factor dependant on the void fraction.

The use of the Beard and Pruppacher correlation with a value of 1.1 for the exponent \beta does not predict the first temperature peak well; but, upstream from the quench front, the precursory cooling effects seem to be well estimated. It is worth noting that the validity of the above correlations can be discussed in the sense that most of them were obtained at atmospheric pressure and at relatively low Reynolds nos.

3.3.5 A selected set of constitutive laws for heated single tubes

In this section, we have chosen a set of correlations which (1) better suited the conditions of the Bennett, Keeys and Becker experiments and (2) accounted for more physical phenomena (3) enables us to over-predict the PCT (measure of safety). The graphs (figure 10) were obtained while expressing the interfacial droplet drag from a modified version of the Schiller-Naumann correlation accounting for the mass efflux:

\[
C_D = \frac{24}{Re_d} \left( 1 + 0.15 \cdot \frac{Re_d^{0.687}}{(1 + B)^{1/2}} \right) \quad 0 \leq Re_d \leq 1000
\]  

However, we select the maximum value of the drag in between the latter, 0.44 and the Ingebo drag coefficient value.

The interfacial heat transfer was based on the Beard and Pruppacher correlation:
\[
N_{ud} = \frac{1.56 + 0.616 R_e^{1/2} P_r^{1/3}}{(1 + B)^{1.1}} \tag{18}
\]
and the two-phase enhancement factor considers a wall friction factor value for a smooth pipe of 0.005:

\[
\Psi_{2\phi} = 1 + 100 \left[ \frac{(1 - \alpha) \cdot G r_{2\phi}^{2}}{R e_{g}^2} \right]^{1/2} \tag{19}
\]

In our modelling, we do not account for the droplet-to-wall heat transfer or for radiation. This set of correlations (Eq. (17), (18), (19)) is supposed to better represent the physics involved such as the droplet mass evaporation, droplet drag coefficient derived from accelerating cloud of droplets etc. Nevertheless, some results (figure 10) exhibited large discrepancies right after the CHF point. The calculation of both the droplet diameter (at a critical Weber number value of 12) and the relative velocity might cause these differences. A too high relative velocity value induces an under-prediction of the temperature at low mass flux.

Regarding the Blowing factor, the mass efflux from the droplet reduces the convective heat transfer from the superheated steam: the evaporation process increases the boundary layer thickness leading to a sharp decrease in heat transfer to the droplet surface, which in return reduces the rate at which additional mass can be added. This shielding effect is limited in our calculations and droplets completely evaporate.

4 CONCLUSION

The analysis and the assessment of the TRACE codes demonstrate its ability to well predict the point of critical heat flux reasonably well. However, the temperature profile, and
thus, the peak clad temperature, is not well predicted for the case of a non-uniform heat flux. For the case of uniform heat flux distributions, the combination of the so-called ‘Look-Up Tables’ and the Biasi critical quality provides results that match well with the Bennett experimental data. This is not true at high mass and heat fluxes.

The study of the DFFB models usually implemented in thermal-hydraulics system code such as TRACE established the crucial character of the interfacial heat transfer. A key point would be to study the variations of the heat transfer coefficient and its sensitiveness to both the mentioned correlations and the interfacial area concentration (which has not been treated in this paper for simplicity). The droplet drag did not play an important role in the calculations performed here. The two-phase enhancement factor applied in the wall heat transfer correlation, namely, the Gnielinski correlation, had a little influence on the temperature predicted for the experiments simulated. The droplet diameter and the number density are important in the prediction of both the position of dryout and the temperature profile upstream from the quench front (precursory cooling effects).

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6 REFERENCES


Design and Safety Issues
APPLICATION OF CFD CODES IN NUCLEAR REACTOR SAFETY ANALYSIS

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ABSTRACT

Computational Fluid Dynamics (CFD) is increasingly being used in nuclear reactor safety (NRS) analyses as a tool that enables safety relevant phenomena occurring in the reactor coolant system to be described in more detail.

Numerical investigations on single phase coolant mixing in Pressurised Water Reactors (PWR) have been performed at the FZD for almost a decade. The work is aimed at describing the mixing phenomena relevant for both safety analysis, particularly in steam line break and boron dilution scenarios, and mixing phenomena of interest for economical operation and the structural integrity.

On the other hand slug flow as a multiphase flow regime can occur in the cold legs of pressurized water reactors, for instance after a small break Loss of Coolant Accident (SB-LOCA). Slug flow is potentially hazardous to the structure of the system due to the strong oscillating pressure levels formed behind the liquid slugs. For the experimental investigation of horizontal two phase flows, different non pressurized channels and the TOPFLOW Hot Leg model in a pressure chamber were built and simulated with ANSYS CFX.

In a common project between the University of Applied Sciences Zittau/Görlitz and FZD the behaviour of insulation material released by a LOCA released into the containment and might compromise the long term emergency cooling systems is investigated. Whereas in FZD CFD models are developed in Zittau the corresponding experiments are performed.

Moreover, the actual capability of CFD is shown to contribute to fuel rod bundle design with a good CHF performance.

1 INTRODUCTION

The last decade has seen an increasing use of three-dimensional CFD codes to predict steady state and transient flows in nuclear reactors because a number of important phenomena such as pressurized thermal shocks, coolant mixing, and thermal striping cannot be predicted by traditional one-dimensional system codes with the required accuracy and spatial resolution. CFD codes contain models for simulating turbulence, heat transfer, multi-phase flows, and

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chemical reactions. Such models must be validated before they can be used with sufficient confidence in NRS applications. The necessary validation is performed by comparing model results against measured data. However, in order to obtain a reliable model assessment, CFD simulations for validation purposes must satisfy strict quality criteria given in the Best Practice Guidelines (BPG).

Our partner for CFD code qualification is ANSYS CFX [1], which is one of the leading CFD codes worldwide. Based on this partnership the models developed are implemented into the code and thus contribute to the code qualification. The following topical issues, where CFD calculations have been performed, will be briefly discussed in the paper:

1. Coolant Mixing
2. Horizontal Stratified Flow Phenomena in the Hot Leg of PWR
3. Debris Transport Phenomena in multidimensional water flow
4. Sub-cooled boiling - Application to fuel rod bundle safety assessment

The material presented has been prepared by FZD partly under the sponsorship by the European Commission and the German Government (BMWi).

2 COOLANT MIXING

Numerical investigations on coolant mixing in Pressurized Water Reactors (PWR) have been performed by other institutes and at the FZD for more than a decade [2-9]. The work was aimed at describing the mixing phenomena relevant for both safety analysis, particularly in steam line break and boron dilution scenarios, and mixing phenomena of interest for economical operation and the structural integrity.

With the set-up of the ROCOM [8,9] test facility (Fig. 1), a unique data base has been created to be used for the validation of Computational Fluid Dynamics (CFD) codes for the application to turbulent mixing in nuclear reactors. Benchmark problems based on selected experiments were used to study the effect of different turbulent mixing models under various flow conditions, to investigate the influence of the geometry, the boundary conditions, the grid and the time step in the CFD analyses. In doing the calculations the Best Practice Guidelines for nuclear reactor safety calculations have been followed [5].
stationary and transient flow and mixing studies of the coolant in the PWR Konvoi and the ROCOM test facility with CFX-4 and ANSYS CFX-5-11 during oron dilution transients (start-up of the first coolant pump), Fig. 3
• main steam line break scenarios, Fig. 4
• density driven flows after an inherent dilution with ECC injection (generic experiments at the ROCOM test facility), [7]

The CFD calculations were carried out with the CFD-codes CFX-4 and CFX-5. Calculations were performed on the FZD LINUX cluster (operating system: Linux Scientific 64 bit, 32 AMD Opteron Computer Nodes, node configuration: 2 x AMD Opteron 285 (2.6 GHz, dual-core), 16 GB Memory). Using the block-structured code CFX-4 internals were modeled using the porous media approach and additional body forces. Sensitivity studies showed, that the k-ε turbulence model together with the higher order HYBRID discretization scheme give the best results. Within ANSYS CFX-5-11 it was possible, to model all internals of the RPV of ROCOM in detail. A production mesh with 7 Million elements was generated (Fig. 2). Detailed and extensive grid studies were made. It was shown, that a detailed model of the perforated drum in CFX-5 gives the best agreement with the experiments. Sensitivity studies.
showed that the SST turbulence model and the automatic wall functions together with higher order discretization schemes should be used if possible (further details see [5]).

In the case of stationary mixing, the maximum value of the averaged mixing scalar at the core inlet was found in the sector below the inlet nozzle, where the tracer was injected (Fig. 4). The mixing scalar is a dimensionless representation of the tracer concentration in the experiment or boron concentration/liquid temperature in reality. There is a good agreement between the measurement and the CFD calculations, especially in the averaged global mixing scalar at the core inlet. At the local position of the maximum mixing scalar the time course of the measurement and the calculations is also in good agreement (see Fig. 4).

At the start-up case of one pump due to a strong impulse driven flow at the inlet nozzle the horizontal part of the flow dominates in the downcomer (Fig. 3). The injection is distributed into two main jets; the maximum of the tracer concentration at the core inlet appears at the opposite part of the loop where the tracer was injected [6].

3 CFD-SIMULATIONS FOR STRATIFIED FLOWS

Slug flow as a multiphase flow regime can occur in the cold legs of pressurized water reactors, for instance after a small break Loss of Coolant Accident (SB-LOCA). Slug flow is potentially hazardous to the structure of the system due to the strong oscillating pressure levels formed behind the liquid slugs. It is usually characterized by an acceleration of the gaseous phase and by the transition of fast liquid slugs, which carry out a significant amount of liquid with high kinetic energy. For the experimental investigation of air/water flows, a horizontal channel with rectangular cross-section was built at Forschungszentrum Dresden-Rossendorf (FZD) [10,11]. Experimental data were used to check the feasibility to predict the slugging phenomenon with the existing multiphase flow models build in ANSYS CFX. Further it is of interest to prove the understanding of the general fluid dynamic mechanism leading to slug flow and to identify the critical parameters affecting the main slug flow parameters (like e.g. slug length, frequency and propagation velocity; pressure drop). For free surface simulations, the inhomogeneous multiphase model was used, where the gaseous and liquid phases can be partially mixed in certain areas of the flow domain. In this case the local phase de-mixing after a gas entrainment is controlled by buoyancy and inter-phase drag and is not hindered by the phase interface separating the two fluids. The fluid-dependent shear stress transport (SST) turbulence models were selected for each phase. Damping of turbulent diffusion at the interface has been considered.

The picture sequence (see Fig. 5) shows comparatively the channel flow in the experiment and in the corresponding CFD calculation. In both cases, a slug is developing. The tail of the calculated slug and the flow behind it is in good agreement with the experiment. The entrainment of small bubbles in front of the slug could not be observed in the calculation. However, the front wave rolls over and breaks. This characteristic of the slug front is clearly to be seen in Fig. 5. It is created due to the high air velocity.
Furthermore, pre-test calculations CFD were carried out to simulate a slug current in a real geometry and under parameters relevant for the reactor safety. These calculations were performed for a flat model of the hot leg which represents the geometry of a 1:3 scaled Konvoi reactor (Fig. 6). Steam and water were taken as a model fluid with a pressure of 50 bar and the accompanying saturation temperature of 264°C. To be able to perform the experiments at high pressure, the whole hot leg model is put into a pressure chamber.

The pretest calculations began with a partial water-full channel and quiescent gas phase. At the beginning of the steam supply the surface of the still standing water phase rises in the direction of the steam generator simulator. This effect is caused by the momentum exchange between flowing out steam and quiescent water. The calculation shows spontaneous waves...
which grow in the elbow to slugs originate in the horizontal part of the hot leg model. The
Fig. 7 shows this state as a snapshot of the results of the calculations.

![Fig. 7 Snapshot of the results of the calculations](image)

### 4 INVESTIGATIONS OF INSULATION FIBER TRANSPORT PHENOMENA IN WATER FLOW

The investigation of insulation debris generation, transport and sedimentation becomes more
important with regard to reactor safety research for PWR and BWR, when considering the
long-term behaviour of emergency core coolant systems during all types of loss of coolant
accidents (LOCA). The insulation debris released near the break during a LOCA incident
consists of a mixture of disparate particle population that varies with size, shape, consistency
and other properties. Some fractions of the released insulation debris can be transported into
the reactor sump, where it may perturb/impinge on the emergency core cooling systems [12-15].

Open questions of generic interest are the fibre transport in an aqueous flow, the
sedimentation of the insulation debris in a water pool, its possible re-suspension and transport
in the sump water flow and the fibre load on strainers and the corresponding pressure drop.

A joint research project on such questions is being performed in cooperation of the University
of Applied Sciences in Zittau/Görlitz and the Forschungszentrum Dresden-Rossendorf. The
project deals with the experimental investigation and the development of CFD models for the
description of particle transport phenomena in coolant flow. While the experiments are
performed at the University Zittau/Görlitz, the theoretical work is concentrated at
Forschungszentrum Dresden-Rossendorf. Details were published by Krepper et al. 2008 [16].

The main topics of the project are
- Primary particle constitution: Experiments are performed to blast blocks of
  insulation material by steam under the thermal hydraulic conditions to be
  expected during a LOCA incident (i.e. at pressures up to 11 MPa). The material
  obtained by this method is then used as raw material for further experiments.
- Sedimentation of the fibres: The transport behaviour of the steam-blasted material is investigated in a water column by optical high-speed video techniques. The sinking velocities of the fibres are then used to derive the drag coefficients and other physical properties of the modelled fibre phase, which is necessary for the implementation of an adequate CFD simulation. Fig’s. 7 and 8 show the measured distribution of sinking velocities and particle size for the insulation material MD2.

![Fig. 8 Distributions of the sinking velocities](image1)

![Fig. 9 Distributions of the particle size](image2)

- Transport of fibres in a turbulent water flow: For these investigations, a narrow channel with a racetrack type configuration was used with defined boundary conditions. Laser PIV measurements and high-speed video were used for the investigation of the water flow-field and the fibre concentration. Besides the drag acting on the particles, the turbulent dispersion force plays an important role in determining the momentum exchanged between the water and the fibrous phase.

- Deposition and re-suspension of fibres: The deposition and re-suspension behaviour at low velocities was investigated by the same techniques and the narrow racetrack channel. Except that, in this case obstacles were inserted into the channel to change locally the flow regime. The experiments are designed to work with laser PIV measurement and high-speed video to investigate the fibre agglomeration in the obstacle region. CFD approaches consider the influence of the fibre material on the mixture viscosity and the dispersion coefficient on the transport of the solids.

- Effect of strainers: A test rig was used to study the influence of the insulation material loading on the pressure difference observed in the region of the strainers. A CFD model was developed that uses the approach of a porous body. The calculated differential pressure considers compactness of the porous fibre layer. Correlations from the filter theory known in chemical engineering are adapted to the certain fibre material properties by experiments. This concept enables the simulation of a partially blocked strainer and its influence on the flow field.

- Behaviour of a plunging jet in a large pool and impact on fibre transport: By using high-speed video and laser (LDA and PIV) measurements, the progression of the momentum by the jet in the pool is investigated. Of special importance is the role that entrained gaseous bubbles play on disturbing the fluid and potentially influencing the fibre sedimentation and re-suspension. Fig. 10 shows...
that under certain flow conditions the entrained air causes a swirl, which transports the injected fibres below the jet. In Fig. 11 the fibre mass accumulated in the tank dependent on the inlet jet velocity is shown. In the case of only 1.5 m/s a left turning swirl was found and the fibre material was transported directly through the tank. For the other case of 5 m/s jet velocity a right turning swirl occurred (s. Fig. 10), which deposited the fibres below the jet and accumulates fibres for longer time in the tank (s. Fig. 11).

![Water flow field induced by the entrained air](image)

![Accumulated fibre mass dependent on the inlet velocity](image)

**Fig. 10** Water flow field induced by the entrained air  
**Fig. 11** Accumulated fibre mass dependent on the inlet velocity

## 5 CFD-CALCULATION OF A HOT CHANNEL OF A FUEL ROD BUNDLE

Boiling is a very effective heat transfer mechanism. Liquid cooling including phase transfer very large heat fluxes can be established. Exceeding the critical heat flux, however, the heat transfer coefficient suddenly decreases, the temperatures increases leading to possible damaging of construction material. The critical heat flux depends not only on fluid properties but also on flow conditions and on geometric circumstances [17].

For the case of a fuel rod, the permissible heat flux can be influenced by the geometrical design. Especially the spacer grids equipped with mixing vanes play an important role to increase the permissible heat flux. The verification of design improvements and their influence on the critical heat flux require very expensive experiments. Therefore, the supplementation or even the replacements of expensive experiments by numerical analyses are of relevant interest in fuel assembly design.

Although the CFD modelling of critical heat flux is not yet state of the art, the simulations shall demonstrate the capability of CFD supporting the fuel assembly design. In the calculations only sub-cooled boiling is simulated, which is here considered as a preliminary phenomenon towards departure of nucleate boiling (DB). DNB might occur at the thermal hydraulic conditions of a PWR. A situation was investigated, when at full power and full pressure the inlet temperature rise caused undesired boiling in the channel.

A section of coolant channel between two spacer grids having a length of z=0.5 m was simulated. The grid represents a sub-channel between 4 rods having a diameter of 9 mm and a rod distance of 12.6 mm. The thermal hydraulic and transport water properties were set for a pressure of 15.7 MPa, typical for PWR conditions. The heat flux at the rod surface was
assumed to be $1.0 \times 10^6 \text{ W/m}^2$ and the sub-cooling at the inlet was set to 12 K expecting the generation of vapour in the simulated section. The axial water velocity was set to $V_Z = 5 \text{ m/s}$. The faces at the low and high x respective at low and high y were simulated as periodic boundary conditions, assuming that the channel is infinitely extended in these four directions.

![Fig. 12 Hot channel vapour flow streamlines and rod surface temperatures](image)

The figure shows the flow condition in the considered channel section (axially shortened presentation). The mixing vanes generate a strong swirl in the actual calculation given as inlet condition. They are not modelled in this calculation, but a swirl was introduced into the flow as boundary condition at the inlet of the channel section. The overall vapour generation can be decreased by the swirl effect. Due to the centrifugal force, the heavier fluid component - the water - is pushed outwards, whereas a large amount of the lighter component - the vapour - is accumulated in the centre of the channel. The streamlines show vapour bubbles moving in the centre of the channel caused by the centrifugal forces. The colours represent the temperatures of the metal surface. Their distribution can be used as qualitative criterion of the effect of a mixing vane. Further details were published by Krepper et al. (2007), [18].

**CONCLUSION**

Computational Fluid Dynamics (CFD) is increasingly being used in nuclear community to model safety relevant phenomena occurring in the reactor coolant system. For this reason the long-term objective of the activities of the FZD R&D program lies in the development of theoretical models for basic phenomena of transient, three-dimensional single and multiphase systems. Local geometry independent models for mass, momentum, heat transfer and scalar transport are developed and validated. Such models are an essential pre-condition for the application of complex fluid dynamic codes to the modelling of flow related phenomena in nuclear facilities. Our partner for CFD code qualification is ANSYS. Their code CFX is one of the leading CFD codes worldwide. Based on this partnership the models developed are implemented into the code and thus contribute to the code qualification.
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SAFETY IMPROVEMENTS IN MOCHOVCE 3&4 DESIGN

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ABSTRACT

In 2006 ENEL became the majority shareholder of Slověnské Elektrárne (SE) and after the completion of a feasibility study it was decided to complete Mochovce Nuclear Power Units 3 and 4. The Basic Design was revised to include additional safety improvements; the Preliminary Safety Analysis Report was updated accordingly. Before the start of the Basic Design revision process, ENEL/SE decided to establish an independent Safety Board composed of six outstanding nuclear safety experts to supervise the development of the Basic Design revision with specific reference to issues of primary interest from the point of view of nuclear safety.

In the following the major safety improvements and the main conclusions drawn by the Safety Board are presented.

1 INTRODUCTION

The construction of Mochovce Units 1 and 2 started in November 1982. Construction of Mochovce Units 3 and 4 started three years later, in 1985 (a view of the site is provided in Fig. 1).

In 1992 the construction works were suspended due to lack of financial resources. In 1996, construction of Units 1 and 2 was resumed. Unit 1 was connected to the grid in 1998, and Unit 2 in 2000.

Since suspension of works, the civil structures and mechanical components of Mochovce 3&4 were put under a preservation program which was concurred with the Slovak Nuclear Regulatory Authority (NRA).

In 2006, ENEL became the majority shareholder of Slovenské Elektrárne (SE) and after the completion of a feasibility study it was decided to proceed with the completion of Mochovce Units 3&4.
In the mean time a large amount of safety upgrades had been implemented in other Nuclear Power Plants of this type (VVER 440/213) and, considering the evolution of the design criteria for the current-generation reactors, it was agreed with the Nuclear Regulatory Authority to implement additional major safety improvements in the design of Mochovce 3&4. The Basic Design and the Preliminary Safety Analysis Report of Mochovce 3&4 were therefore revised.

The safety-improvement program took into account the current international safety requirements (IAEA, WENRA [1], [2]), the recommendations of all international missions carried out in Mochovce 1&2 and, in general, the experience feedbacks from similar nuclear power plants currently in operation in several countries (Slovakia, Czech Republic, Hungary, Ukraine, Finland and Russia). In addition, further consideration was given to the results of research and experimental activities carried out in the last fifteen years on severe accidents and on the VVER 440/213 pressure-suppression containment.

Many important safety improvements were implemented, including those to cope with severe accidents and related effects on containment integrity. Several other important modifications were introduced, such as the improvement of the reliability of the power supply during accident conditions. Furthermore, a state-of-the-art I&C has been specified.

An expert Board ("Safety Board") was established in 2006 by ENEL as an independent body to assess the safety aspects during the execution of the revision of the Basic Design of Mochovce 3&4. The Safety Board was composed of 6 peer members internationally recognized as leading experts in nuclear power plant safety (Prof. Boeck from Austria, Mr Lipar from Slovakia, Ms Carnino from France, Prof. Birkhofer from Germany, Prof. Bolshov from Russia and Prof. Cumo from Italy), with the technical support of SOGIN.

2 MAJOR SAFETY IMPROVEMENTS FOR SEVERE ACCIDENTS

The design work performed on Mochovce 3&4 took advantage of the results of extensive studies and experiences of several reactors of the same design.
In this paper, the features already identified by IAEA to be implemented in this type of plants and already largely included in most or all plants are not described. All these safety features along with all the safety features already implemented in Bohunice V2 Nuclear Power Plant and in Mochovce Units 1&2 have been implemented also in the design of Mochovce 3&4.

This paragraph mainly focuses on the additional improvements which were included in the design.

2.1 Severe Accident Prevention

The improvements implemented for the prevention of core melt sequences proved to be very effective, further reducing the Core Damage Frequency (CDF) to less than 1E-5 per year, as recommended by INSAG [3] for new plants.

The most important preventive measures are:

1. interconnection of all the Units at the 6 kV level (non–safety switchboards) that allows, when needed, to feed the auxiliaries from another Units. This helped reduce significantly the contribution to the CDF.
2. implementation of several measures, such as the possibility of manual interconnection between the safety divisions of different Units at the 6 kV level, to avoid the evolution of station black-out to severe accident.
3. improvements in the design of safety and safety-related systems (e.g., HP and LP ECCS, Service Water System) aimed at increasing the reliability of these systems by incorporating the operating experience feedbacks of Mochovce 1&2 and Bohunice NPPs.

It has to be remarked that, within the revision of the Basic Design, the impact of hardware modifications on the Probabilistic Safety Assessment (PSA) figures has been fully evaluated while the impact of human error on the CDF has not been completely assessed. Ad-hoc operating and emergency procedures are meant to be used to evaluate the impact of human error, which will be prepared during the development of the Detailed Design. For this reason, in the first design phase (Basic Design revision) the results of the human reliability analysis already carried out for Mochovce 1&2 has been preliminarily used for Mochovce 3&4.

A significant reduction of the full-power contribution to the total CDF has been already evaluated for Mochovce 3&4 with respect to Mochovce 1&2, while a meaningful estimation of the impact of the proposed design modifications on the low-power/shut-down contribution to the CDF will be done later. Nonetheless, it is expected that one of the important results of the safety-improvement program of Mochovce 3&4 will be to obtain a PSA being well balanced between full-power and shut-down and with no initiating events having an excessive impact on the CDF.

2.2 Severe Accidents Mitigation

Particular emphasis has been given to severe accidents mitigation. Severe accidents have to be taken into account in the design and in the licensing documents as required by the Slovak Law in compliance with recommendations by IAEA Safety Standards.

The main elements of the severe accident mitigation strategy are:

1. Primary circuit depressurization. To avoid scenarios of high-pressure vessel failure, which may pose a serious threat to containment integrity, an additional 80 mm diameter pipe has been designed to be connected to the pressurizer relief
and safety valves header. This pipe shall be provided with two valves in series which shall be operated according to emergency operating procedures. The steam shall be discharged into the containment; pressure will be suppressed by the bubbler condenser. In this way the pressure inside the Reactor Pressure Vessel will reach values below 2 MPa in a few minutes, thus excluding scenarios of direct containment heating.

2. Retention of corium inside the Reactor Pressure Vessel by flooding the reactor cavity and removing the heat through the reactor vessel wall (eliminating any ex-vessel corium interaction with water, containment atmosphere and reinforced concrete structures).

3. Limitation of hydrogen concentration by self-actuated igniters and recombiners preventing detonation risks.

4. An additional power supply to cope with station-blackout scenarios.

5. Additional independent water storage tanks available for containment spray also during station-blackout scenarios.

6. Heat sinks composed of structures and other masses inside the containment have been taken into account to control the initial pressure peak while containment spray by external circulation of sump water has been accounted for in the long term.

7. Also the adequacy of the containment leak-tightness has been ascertained.

The modifications that have been implemented in the design are grouped in the following categories:

1. Management of containment atmosphere;
2. Corium in vessel retention;
3. Additional water sources;
4. Monitoring systems dedicated to severe accidents;
5. Control room habitability;
6. Instrumentation and Control.

2.2.1 Management of the Containment Atmosphere

The containment for the Mochovce Units is of the pressure-suppression type and relies on a large amount of water for the condensation of steam in case of large loss-of-coolant accidents. This important feature of the Mochovce containment has allowed a significant reduction of the containment design pressure and allows for the early termination of radioactive releases in case of design basis accidents (DBA) by providing for a quick reduction of containment overpressure.

In case of a large Loss of Coolant Accident (LOCA), a considerable amount of air is displaced from the containment zone housing the primary circuits (steam generator boxes) to separate containment areas (air traps). In this way, the effectiveness of the containment spray is greatly enhanced providing a quick reduction of pressure in the steam generator boxes. A set of pipes connecting the air-traps with the steam generator boxes has been designed (complying with redundancy and diversification criteria required for systems designed to cope with DBA scenarios) to avoid excessive containment under-pressurization. Such pipes are normally closed but are opened automatically in case of low pressure in containment zones housing the steam generator boxes to equalize the pressure. Appropriate verifications will be performed during the execution of the Detailed Design also to make sure that no drawbacks exist from the point of view of hydrogen management (due to the ingress of air in potentially hydrogen-rich areas).
The issue of hydrogen management has been investigated for a long time in the field of nuclear safety, as hydrogen uncontrolled burning is recognized to pose a severe threat to containment integrity. In addition, the fact that the containment of VVER 440 reactors is divided in many compartments makes the hydrogen issue even more important for this type of reactors, as high concentrations of hydrogen can be reached locally more easily than in “western-type” containments.

In the early phase of such investigations, strategies like pre- and post-inertization of the containment atmosphere were also evaluated but then have been rejected as being unpractical. The solution identified as the most appropriate is based on the use of a combination of Igniters and Catalytic Recombiners: Igniters to quickly burn the hydrogen produced at the beginning of the accident at a high production rate (up to 150 g/s during core relocation in the worst cases) and Catalytic Recombiners to maintain a low concentration of hydrogen in the long term. Both will be passive and autocatalytic. Such solution is widely adopted also in many other operating NPPs where retro-fittings to cope with severe accident scenarios have been introduced.

It is expected that the capacity of the recombiners will be sufficient for adequate hydrogen management for the large majority of severe accident scenarios while the igniters are deemed to be necessary to cover only few limiting cases of lower probability of occurrence but potentially responsible for the most severe consequences.

It is foreseen to install 32 passive autocatalytic recombiners and 30 autocatalytic igniters plus sensors qualified for severe accident conditions. It is expected that the igniters will be located in the steam generator boxes (sprayed zones) although the precise position of such devices inside the containment will be determined during the detailed design.

2.2.2 Corium In-Vessel Retention

The concept of keeping the corium inside the reactor vessel by external cooling with water is not new. It has been already designed for Westinghouse AP600/AP1000 reactors and implemented in Loviisa NPP as a plant modification. The solutions have been approved in all cases by the respective safety authorities and have been used successfully to eliminate the need to deal, in the safety analyses, with ex-vessel phenomena such as ex-vessel steam explosion or molten core-concrete interaction.

Taking into account that the feasibility of IVR has been demonstrated for a very similar plant of the same power (i.e., Loviisa NPP), a similar concept has been developed for Mochovce 3&4, although with some differences in the way it has been implemented.

The key point is to take advantage of the large amount of coolant already available in the containment as energy vector for removing residual heat from the degraded core and transferring it to the ultimate heat sink. Depending on the preceding scenario, such water may be completely available at the floor of the steam generator boxes or partially stored in the 12 trays of the bubbler condenser tower. Taking into account also the volume of water of the primary circuit (including the hydro-accumulators) and the water from the ECCS tanks, it can be estimated that more than 2000 m$^3$ of water are available for ex-vessel core cooling.

By modification of the existing ventilation lines in and around the reactor cavity, a connection for reactor cavity flooding has been designed. Water shall flow from the connection corridors between the steam-generator box and the bubbler-condenser tower, leading to the reactor cavity and then through the gap between the thermal shield and the reactor pressure vessel up to the reactor vessel nozzles, allowing the steam to flow to the steam generator boxes. The steam released in the steam-generator boxes will be condensed by the containment spray water supplied by the additional water storage tanks dedicated for severe accident scenarios.
For the IVR, a considerable amount of thermo-hydraulic analyses had to be performed together with the implementation of a certain amount of design modifications (as depicted in Fig. 1), basically aimed at:

- effectively cooling the bottom part of the vessel (by modifying the thermal shield to allow the ingress of water at the upper part of the reactor cavity);
- preserving the water inventory, by excluding the possibility of water losses to ex-containment rooms of the reactor building (e.g., through the ventilation lines or through the drainage line of the reactor cavity);
- flooding the reactor cavity and removing heat also in station-blackout conditions (by connecting all the relevant equipment to the station-blackout power supply),
- ensuring that, with the additional source of water, the overall thermal capacity of the containment is sufficient to manage the severe accident for the time the ultimate heat sink is postulated to be unavailable (12 hours).

Fig. 2: Sketch of the design modifications adopted for the implementation of the In-Vessel Retention strategy for severe accident management

2.2.3 Additional Water Sources

Three 500 m$^3$ additional tanks external to the containment have been included in the design to ensure the containment spray during the early phase of a severe accident also in station-blackout conditions and to increase the inventory of water available for core cooling, simultaneously increasing the overall thermal capacity of the containment. By means of a dedicated pump, the system is capable of injecting water: a) into the primary circuit via the low-pressure ECCS lines in case of open vessel scenarios; b) to the containment spray system; c) inside the spent fuel pool. The system features its own diesel generator for station-blackout scenarios.
The requirement to add an additional external source of water derived from the results of analyses of hydrogen management. Additionally, a need arose, which was identified during preparation of both emergency operating procedures and severe accident management guidelines for Mochovce 1&2, to increase the availability of the coolant (borated water) for emergency supply into the primary circuit. Finally, as identified by the Probabilistic Safety Assessment (PSA) for similar units (Bohunice V2), the open reactor sequences needed to be considered. For this reason, by evaluating various alternatives, the direct connection to an external source of coolant was selected to be most beneficiary. The proposal of the system is based on the following requirements:

- to provide early spray of the containment in order to reduce the steam content of the containment atmosphere to allow a quick start of the igniters. Igniters should be able to burn hydrogen from the very beginning of the severe accident (i.e., within minutes from the beginning of core relocation when an intense oxidation of the fuel cladding is expected). This is fundamental to prevent the formation of high hydrogen concentrations inside the containment (i.e., 10-12%) which would lead to unbearable consequences for the containment integrity.
- to provide extra thermal capacity for the adequate management of containment temperature and pressure during the first 12 hours since accident initiation, (postulated time of ultimate heat sink unavailability);
- to decrease the radioactive content of the containment atmosphere, thus reducing the off-site consequences of a severe accident;
- to decrease the safety risk, identified by PSA analyses, deriving from the loss of coolant from the spent fuel pool.
- to decrease the safety risk, identified by PSA analyses, associated with open reactor sequences.

2.2.4 Monitoring Systems Dedicated to Severe Accidents

A significant set of newly-monitored parameters has been included into the Plant and Unit monitoring systems in order to include severe accident control and mitigation. The system is designed as an integral part of the unit monitoring system (it is incorporated into the post accident monitoring system). The list of parameters monitored for severe accident purposes has been derived also taking into account the design inputs resulting from the development of the Severe Accident Management Guidelines (SAMGs) for Mochovce 1&2. The list can be divided in two parts. The first part contains containment and system parameters needed directly for SAMG purposes (decision taking). Systems for monitoring the following parameters are included in this group:

- Core outlet temperature
- Pressure in containment
- Temperature of containment in selected rooms
- Hydrogen concentration in selected rooms
- Coolant level in steam generator boxes
- Coolant level in reactor cavity
- Pressure in air locks
- Temperature in the air locks
- Pressure in reactor pressure vessel
- Pressure difference between primary and secondary circuit
- Radiation level in representative points in the containment
- Selected additional informative parameters, as e.g.:
- Coolant level in steam generators
- Feedwater flow into steam generators
- Pressure in hydro-accumulators

The second group of parameters to be monitored for severe accident mitigation purposes contains the parameters necessary to control and monitor the equipment dedicated to severe accident management such as: indicators of position of relevant valves, interlocks, flows, etc.

All monitoring systems and related components will withstand severe accident conditions and will provide reliable and full-scale information during a severe accident.

2.2.5 Control Room Habitability

The issue of assuring the habitability of the Main Control Room (MCR) in all conceivable situations – including severe accident scenarios - has been analyzed extensively.

This issue has found increasing interest in the recent years with the objective of creating a safe working environment for MCR personnel also in case of a severe accident.

Based on relevant information available for modern reactor designs, the general main basic features of the MCR emergency habitability issue can be summarized as follows:

- normally, severe accidents are managed from the MCR, which is a part of the radiologically non-controlled area, and obviously contains no radioactivity.
- the two potential radiological impacts on the MCR staff are: a) direct radiation; b) inhalation of the radioactive substances from the air supplied by the Heat, Ventilation and Air Conditioning (HVAC) system into the MCR. For all modern designs, direct radiation is negligible as compared with inhalation.
- the MCR ventilation system is equipped with sensors to detect hazardous substances such as smoke, airborne radioactivity or, in special cases, chemical and/or biological agents.
- in case of detection of high radioactivity at the ventilation intake, the MCR emergency habitability system is completely isolated from the outside atmosphere. Afterwards, a slight overpressure and acceptable working conditions are ensured in the MCR by a supply of breathable air from compressed air tanks. The capacity of the tanks is sufficient to maintain MCR habitability for several hours.
- in addition and as a back-up to the MCR emergency ventilation and compressed air tanks, there are protective clothing, respirators, and portable breathing equipment with air bottles stored inside the MCR pressure boundary for individual use by the MCR staff members.

In Mochovce 3&4, a set of pressurized air bottles provide fresh air for at least 12 hours after the beginning of the accident. In the assumed conditions, the internal contamination of the MCR has been eliminated for the first most significant part of the accident and therefore the doses to the MCR staff are limited to irradiation from the source term external to the MCR.

2.2.6 Instrumentation and control

Instrumentation and control is a key element to ensure the achievement of a high safety level in the plant. For Mochovce 3&4, the system has been re-designed and shall be supplied entirely new with state-of-the-art design criteria and equipment.
The new design of the I&C will be strongly oriented to the improvement of the human-machine interface (e.g., through the adoption of a safety parameter display system) in order to allow an optimized management of the plant by the MCR staff in all plant conditions.

3 THE CONCLUSIONS OF THE ADVISORY SAFETY BOARD

The mandate of the Safety Board, which has lasted 18 months, has been to supervise the development of the design revision with specific reference to the issues of primary interest from the point of view of nuclear safety, to assess the design adequacy with respect to the best current international practices and also to provide recommendations to ENEL/SE for the next stages of the Project.

The Safety Board recognized that the safety of VVER 440/213 Nuclear Power Plants had been reviewed by international and western organizations and had been continuously upgraded during the last fifteen years. Eighteen reactors of the same type in six different countries have been and are operating with very good safety and operational performances. However the safety of new nuclear power plants is always reviewed and their design is improved taking into account the development of technical knowledge, evolution of the safety criteria and lessons learned from operating experience. In this respect, the Safety Board highlighted that the Mochovce 3&4 revised design is the most advanced among those so far implemented in VVER 440/213 plants, and that the safety and operational targets of Mochovce 3&4 are comparable to those of other reactors currently under construction in Europe.

In particular, it is worthwhile to mention at least the following points highlighted by the Safety Board:

- the severe accident prevention and mitigation are addressed explicitly during the design process.
- PSA results in terms of core damage frequency are in the range of most modern plants.
- the safety review made by IAEA and operating experience from similar plants have been fully considered to improve the design.
- several systems are being designed and will be built with state-of-the-art components (e.g. Instrumentation and Control System, Electrical Systems).
- all IAEA Safety Requirements for the design have been considered and largely implemented.
- several aspects have been reviewed in depth by IAEA ad-hoc teams in relation with the Mochovce site and with Mochovce 1&2 Units, always with very satisfactory results (including an OSART mission in September 2006).

The Safety Board has been able to identify all safety issues and to propose solutions to be implemented during the design revision phase or in the future Project phases, or even at the operational phase. ENEL/SE assured that all recommendations have been or will be taken into account and that the implementation is being tracked.

In conclusion, the Safety Board believes that no design aspect that has been reviewed and discussed prevents Mochovce 3 and 4 units from achieving a very high safety standard and protecting the workers, the public, and the environment according with current applicable international standards.

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Analysis, by RELAP5 code, of Boron Dilution Phenomena in Small Break LOCA and in Mid Loop Operation Transients, Performed in PKL III Test Facility

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ABSTRACT

The present paper deals with the post test analysis of PKL III E2.2 and E3.1 experiments, by RELAP5/Mod3.3 code. These experiments have been executed in the framework of the OECD/SETH Project in the integral test facility PKL III which is operated by AREVA NP GmbH in Erlangen, Germany. The main purpose of the project is to investigate pressurized water reactor safety issues related to boron dilution phenomena. In particular, the E2.2 experiment investigates the boron dilution issue during a small break LOCA transient and the E3.1 experiment investigates the boron dilution issue during shut down conditions (for refueling) with the reactor coolant system closed and the reactor placed in mid loop operation conditions. This analysis is focused on assessing the capability of the RELAP5 code to correctly predict boron dilution phenomena and the thermal-hydraulic parameters in transients with a) asymmetric loop behavior, b) natural circulation in one and two-phase flow, c) steam generator in reflux condenser mode, and (only for E3.1 experiment) d) the primary system under low pressure conditions in the presence of nitrogen. The E2.2 calculated results show a good agreement with regard to the main thermal-hydraulic experimental parameters but show that is challenging for the code predict the different natural circulation onset in the loops. A sensitivity analysis, carried out by simulating the SG primary side with seven U-tubes of seven different lengths, show that the code predicts the delay of the onset of the natural circulation in the loop 2. The E3.1 experiment demonstrates that it is challenging for the code because it is at low pressure and in the presence of non-condensable (nitrogen). Sensitivity analyses have focused on these transients. The results of these sensitivity analyses show the importance of the mass flow through the upper head by-pass to predict the main thermal-hydraulic parameters during the transients.

1 INTRODUCTION

Assessing the safety of a nuclear installation requires the use of a number of highly specialized tools: computer codes, experimental facilities and their instrumentation, special measurement techniques and so on. Of these tools, thermal-hydraulic system codes are widely
used to perform design, safety and licensing analyses of Nuclear Power Plants (NPP). The performance assessment and validation of large thermal hydraulic codes and the accuracy evaluation, when calculating the safety margins of Light Water Reactor (LWR), are among the objectives of international research programs.

In this framework, the execution of experiments in Integral Test Facility (ITF), simulating the behaviour of a NPP, plays an important role for the system code assessment and for the possibility to identify and characterize the relevant phenomena during off-normal conditions. Organization for Economic Cooperation and Development (OECD) sets up a test campaign named SETH (SESAR (Senior Group of Experts on Nuclear Safety Research) Thermal-Hydraulics) project to be carried out in the Primärkreisläufe (PKL) facility, addressing the investigation of inadvertent boron dilution events in a Pressurized Water Reactor (PWR) [1]. The present paper is related to the application and the assessment of RELAP5/mod3.3 code against boron transport experiments [2].

The first part of the paper is focused on the E2.2 test which simulates a break in a cold leg (CL) with a contextual high pressure injection in two CLs and a secondary system cooling rate of 100 K/h [3,4]. The transient has been analyzed by comparing experimental and RELAP5 calculated data. This activity has been performed in the framework of a collaboration between the Department of Nuclear Engineering (DIN) of the University of Palermo and APAT (Agenzia per la Protezione dell’Ambiente e per i Servizi Tecnici) started in the 2002.

The second part of the paper is connected with the participation in the OECD SETH/PKL Benchmark on test E3.1 [5,6,7], and in particular aimed at understanding the limits of the current generation of thermal hydraulic system codes. This second activity is performed in the framework of a collaboration between the DIN and the San Piero a Grado Nuclear Research Group of the University of Pisa started in the 2006.

1.1 Boron (Mixing and) Transport (Including Dilution)

Boron is a highly neutron absorbing material and will consequently affect the core power when inserted or removed from the core. Thus boron mixing and transport will not directly influence the thermal hydraulics but will do so indirectly through the core power which is a thermal-hydraulic boundary condition.

In PWRs boron is added to the coolant-moderator and its concentration will control the long term variation of core reactivity. Because of the high liquid velocity in the reactor coolant system (RCS) the added boron will be mixed quite homogeneously with the coolant without steep gradients and its transport through the RCS will closely follow the liquid transport. However there are low flow conditions, for example at idle reactor coolant pumps, when the pump power has temporarily been lost or when inhomogeneous boron concentration can be attained in the loops with rather steep concentration gradients, in which the diffusion of boron within the liquid can be quite substantial and thus the boron transport will be different from the liquid flow.

Boron mixing and transport within the RCS is of crucial importance for core power control in LWRs. The mixing and transport of boron influences through reactivity feedback the local core power during normal and off normal operation of PWRs. During normal operation of a PWR the initial excess reactivity of a fresh core is compensated by adding boron to the coolant. This added boron is gradually decreased as the fuel burnup increases in order to keep the core reactivity essentially constant throughout the core lifetime.

During a Small Break Loss of Coolant Accident (SBLOCA) in a PWR with U-tube Steam Generators (SG) it has been found that in the reflux condensation phase there exists a
mechanism which can accumulate boron-free condensate in the loop seal (LS). These transients could become a reactivity induced accident (RIA) initiator. It also a well known fact that a PWR cannot be kept in a sub-critical cold shutdown condition immediately after refueling without the boron concentration in the coolant being above some value, despite all the control rods being inserted. Thus the boron concentration and its mixing and transportation with the coolant are also of major concern at shutdown conditions. It also has been found that in some specific cases, during shutdown conditions, in PWRs there could be situations where pronounced heterogeneities in the boron concentrations could develop and thus could potentially cause a RIA [8].

1.2 Small Break LOCA Transients

In general, SBLOCA’s are characterized by an extended period (this can be tens of minutes to several hours at the lower end of the break spectrum) after the occurrence of the break, during which the primary system remains at a relatively high pressure and the core remains covered. As soon as the pumps are tripped, either automatically or manually, gravity controlled phase separation occurs and gravitational forces dominate the flow and distribution of coolant inside the primary system. The subsequent sequence of events, whether or not the core uncovers and is recovered or reflooded, depends not only on the location, shape and size of the break, but also on the over-all behaviour of the primary and secondary systems. This behaviour is strongly influenced by both automatic and operator initiated mitigation measures. In a SBLOCA however due to the continuous loss of primary coolant inventory through the break, degradation of core cooling is expected to occur or not, depending on break size and location, availability of mitigating equipment and nature of operator actions. The critical flow through breaks is a very important element in analyzing plant behaviour during the course of the complete spectrum of LOCAs. The break flow determines the depressurization rate of the system and the time to core uncovering which in turn are of, major concern for when and how different mitigation auxiliary systems will be initiated and function [9].

1.3 Shutdown and Mid Loop Operation Transients

The design of many PWRs requires that, during certain phase of the shutdown period, maintenance operations are performed while the water level in the primary system is lowered. The water level may be reduced to the mid-point of the outlet leg (Hot Leg) of the primary coolant system, and thus the term mid loop applies. During this operation condition it is necessary to provide a continuous heat removal system for the reactor decay heat. Inasmuch as the SGs can no longer perform this function (there is no forced or Natural Circulation (NC) through the SGs) the sole heat removal system is the Residual Heat Removal System (RHRS). During the heat removal process it is important to maintain the reduced level within a somewhat narrow range. The level must be low enough to enable opening of the inspection hatches, but not so low as to uncover the pipe leading from the HL to the RHRS (this pipe is sometimes referred to the drop line). If the water level is too low, the water supply to the RHRS is disrupted. At first a vortex may form and air may be ingested. A complete interruption of water supply leads to a pump damage mechanism known as cavitation; this can lead to irreversible pump damage. At such times a small error in accomplishing the desired water level can lead to drainage below the HL and interruption of the heat transport to the RHR heat exchanger system [10].
2 BRIEF DESCRIPTION OF THE PKL III FACILITY AND THE E2.2 AND E3.1 EXPERIMENTS

2.1 PKL III Facility

The PKL facility [11] is a full-height ITF that models the entire primary system and most of the secondary system (except for turbine and condenser) of a PWR of KWU design of the 1300-MW class. All the elevations are scaled 1:1 while the volumes, power and mass flows are modeled by the scaling factor 1:145. Similarly to other test facilities of this size, the scaling concept aims to simulate the thermal hydraulic system behaviour of the full-scale power plant.

This facility is subdivided in RCS, SG secondary side and the interfacing systems on the primary and secondary side.

The RCS comprise the rod bundle vessel containing the heater bundle (simulation of reactor core), the four loops, with the reactor coolant pumps and the SGs, the down-comer (DC) model and the pressurizer (PRZ). The four loops on the primary side are symmetrically arranged around the pressure vessel (RPV). The PRZ is connected to the RCS by the surge line. The vessel models the Upper Head (UH) plenum, the Upper Plenum (UP), the reactor core, the reflector gap and the lower plenum (LP). The reactor core model consists of 314 electrically heated fuel rods and 26 control rod guide thimbles. The maximum electrical power of the test bundle is 2512 kW. The fuel rods, arranged in three concentric zones, can be heated independently of one another. The DC is modelled as an annulus in the upper region and continues as two stand pipes connected to the LP. This configuration permits symmetrical connection of the 4 CLs to the RPV, preserves the frictional pressure losses and does not unacceptably distort the volume/surface ratio. The UH bypass is modelled by four parallel by pass lines associated with the respective loops to enable detection of asymmetric flow phenomena in RCS.

The four steam generators of the facility are vertical U-tubes bundle heat exchangers. In keeping with the scaling factor of 1:145 the number of tubes for each SG was therefore obtained as 28. The tubes have seven different lengths.

The operating pressure of the PKL facility is limited to 45 bar on the primary side and to 56 bar on the secondary side. This allows simulation over a wide temperature range (250 °C to 50 °C) that is particularly applicable to the cooldown procedures investigated.

2.2 E2.2 Experiment

The test E2.2 investigates the inherent boron dilution during SBLOCA. The test was executed January 15, 2002 [3].

The test E2.2, simulates a SBLOCA with these main boundary conditions:

- break (32 cm$^2$/145) in the CL of loop 1, between RCP and RPV;
- all 4 SGs running down at a cooling rate of 100 K/h;
- active High Pressure Injection Systems (HPIS) in loop 1 and 2;
- active Low Pressure Injection Systems (LPIS) in loop 1 and 2 when the pressure in RCS drops below 10 bar for the first time.

The aim of the test was to answer the following questions:

- what are the mass flow rates as NC start up and how long is the time lag between the commencement of circulation in the individual loops and the condensate slugs from the individual loops reaching RPV;
what is the lowest boron concentration that can ever be attained at the inlet of the RPV in the event of an accident involving a small cold-side break, cold-side injection of emergency cooling water, inherent boron dilution.

The test initial conditions are established in two phases: a pre-conditioning phase and a conditioning phase. The first one (subcooled NC) is performed to reach a steady state condition before the conditioning phase.

At the end of the pre-conditioning phase:

- the primary system is completely filled with 2300 kg of water with an homogenous boron concentration of 1000 p.p.m.;
- the heater rods supply a constant power of 530 kW;
- the pressure in RCS is about 42 bar and the core outlet temperature is 250 °C (3°C of subcooling);
- heat is removed by NC in all 4 loops;
- main steam pressure is about 28 bar.

The conditioning phase initiates 6450 s before test starts (t=0) (Start of the Transient (SOT)) by insulating the main steam line with the purpose of increasing the pressure in secondary side to reduce the primary to secondary temperature difference. This induces saturated conditions at core outlet and consequently an increasing pressure in RCS. When the RCS pressure reaches 44 bar (5270 s before the SOT), the break in CL 1 is opened and the pressure difference between primary and secondary falls down to 1-2 bar. The level in RPV drops rapidly reaching the lower CL edge, while U-tubes of the SGs and PRZ completely empty. At 4430 s before SOT the main steam line is opened again so the primary pressure reaches about 40 bar. The pressure of secondary system is controlled at about 39.4 bar in order to decrease the NC in RCS. When the coolant inventory is about 1170 kg (3950 s before SOT), the break is closed and the primary system remains in reflux-condenser conditions for 3240 s. At the end of this phase, about 200 kg of condensate are formed per loop and low-borated water is accumulated in LSs. At 710 s before SOT an high pressure injection in CLs 1 and 2 begins at a reduced flow rate in order to avoid rapid condensation which could cause high mass flows in RCS, and, therefore, mixing processes between low-borated and high-borated water. At the end of this phase, the mass inventory is about 1440 kg.

At t=0 the test starts with the opening of the break in CL 1, the injection by HPIS in CLs 1 and 2 of a total flow rate of about 0.8-0.9 kg/s and the cooldown of the SGs at 100 K/h. At the beginning, the mass flow which goes out through the break is greater than the mass flow injected by HPIS. The coolant inventory has a minimum at about 1200 s, than return to increase owing to a decreasing RCS pressure and an increasing flow rate injected. When RCS pressure drops below 10 bar, LPIS is activated in CL 1 and 2, and it is switched off again when the same pressure rise again above 10 bar. At t=5980 s the HPIS is switched off and the test is considered ended at t=8430 s, when the RCS and the secondary side pressures are about equal to external pressure.

This experiment shows that the restart of the NC is a “very sensitive mechanism” which is strongly influenced by the prevailing boundary conditions in the individual loops e.g. location of the break, connection of the PRZ and Emergency Core Cooling Systems (ECCS) injection. In this connection some of the relevant experimental results are here reported. NC starts at different times in the different loops. NC first sets in the two unfed loops 4 and 3. In the loop 4 NC starts, with an initial very small mass flow, at 3020 s (final phase of refill), about 160 s earlier than in loop 3. The onset of the NC in the loops supplied with ECCS is experimentally detected after LPIS injection ends. In particular, a forward flow starts in the loop 1 at 3830 s and the NC starts in the loop 2 at about 6450 s, only after the HPIS is switched off.

As regard the boron concentration it is noteworthy that the condensate slugs with the lowest boron concentration are formed only in the loops 4 and 3 which are not feed with ECC water.
In particular the minimum boron concentration measured at the RPV inlet is about 350 ± 100 p.p.m. at 3050 s in the loop 4 and at 3400 s in the loop 3. After the onset of the NC in the loops supplied with ECC water there is no significant drop in the boron concentration at the RPV inlet. As regard the loop 1 it is noteworthy that, though the NC is started at 3830 s, the entire water flowing through SG 1 is escaping via the break with any condensate present. Therefore no drop in the boron concentration is observed at the CL 1 RPV inlet. In the loop 2, only after t=6450 s a distinctly forward moving two phase NC is starting (due to the steam that enters HL 2 via the surge line from the PRZ). However the boron concentration at the CL 2 RPV inlet don’t drop but rise.

2.3 E3.1 Experiment

The second experiment that is analyzed in this paper is the test PKL III E3.1 “Loss of Residual Heat Removal System (RHRS) in 3/4 loop operation with the RCS closed” which was conducted at the facility on July 25, 2002 [5].

The scenario starts simulating the failure of the RHRS which, during the stationary condition of the experiment, is removing the decay heat of the core simulator. The boundary conditions of the experimental facility simulate the condition of the prototype NPP in preparation for the refueling. The reactor coolant inventory is reduced at the 3/4 loop operation level, the space above the water inventory is filled with nitrogen that is injected into the RCS, the primary side pressure is 1 bar and the temperature at core outlet is 61 °C. For the purpose of this test, it is postulated that the PRZ has already been largely cooled down when the transient starts, and its temperature varies between 56 °C (bottom) and 49 °C (top).

The test was designed to investigate:

- the capability of the water-filled SG to remove the decay heat following the failure of the RHRS via only one operational SG;
- the heat transfer mechanism when nitrogen is present in the primary side and in the SG;
- what is the primary side pressure when the heat removed by the secondary side allows stable equilibrium conditions;
- the deboration process connected with the reflux condenser mode occurring following the failure of the RHRS.

To better address this issue the test is performed with borated coolant and special instrumentation suitable for boron concentration measurements.

The experiment can be subdivided in the following phases:

0. phase prior to starting the test, before the RHRS failure (until 0 s);
1. test phase until the set point for SG 1 operation (from 0 s to 8225 s);
2. test phase with SG 1 pressure controlled (from 8225 s to 33095 s);
3. test phase with the accumulator injection (from 33095 s to 37650 s);
4. test phase following the restoration of RHRS (from 37650 s to the end of the experiment).

The phase prior to starting the test is performed in order to reach the condition of the test begins. At the end of this phase the primary side is at about 1 bar, in stable conditions with core power at 217 kW. The energy is removed by normal operation of RHRS. The RCS mass inventory including the PRZ has been reduced to 1300 kg which correspond to 3/4 loop level on the primary side. Secondary–side pressure in all SGs is approximately 1 bar. SG 1 and 2 are filled with water on the secondary side (level approximately 12.2 m) and the secondary sides of SG 3 and 4 contain no water and are completely filled with air. Only the SG 1 is ready for operation while the SG 2, 3 and 4 are not operated during the experiment.
The experiment starts simulating the loss of the RHRS (phase 1). Following this event the primary side energy is not removed and coolant temperature in the core increases. Nearly 700 s after the SOT, saturated condition are reached in the core outlet. At 1270 s after the SOT, following the increase of voiding in the core, the PRZ level rises because the two phase mixture in HL 2 is entrained into the PRZ and here it condenses due to the low temperature of the PRZ wall. During this phase the core power is removed mainly from the secondary side of SG 1 and 2, filled with water. Due to the increased flow of steam into SG 1 and 2, the swell level in the inlet chambers of these SGs continues to rise and reaches the U-tubes with resultant heat removal in reflux condenser mode 1955 s after the SOT. The swell level in SG 3 and 4 simultaneously drops back into the HLs. An accurate investigation of the balance of energy of the system shows that notwithstanding the SG 3 and 4 are filled with air their contribution is not negligible. The energy removed from the primary side is accumulated in their structural materials. The energy removed by the SG 1 and 2 causes the temperature increase in the mass inventory of these systems, and after 5500 s the saturated conditions are reached in SG 1. The SG 2 follows the SG 1 behaviour with a delay of 1600 s. When the saturated conditions are reached, the secondary side pressure rises more quickly. Secondary side pressure in SG 1 rises to 2 bar (SG 1 pressure set point) 8225 s after SOT resulting in activation of the SG 1 secondary side pressure control system.

The SG 1 starts to be operated (phase 2) and the pressure is maintained at 2 bar for the remaining of test period. On the contrary the SG 2 pressure follows the trend of the primary side. Following the start of SG 1 operation, as a result of the steam flowing from the core to this SG, a displacement of mass from SG 2 U-tubes into SG 1 U-tubes takes place. At 12180 s, the primary mass inventory of the SG 2 is completely displaced. As a result of the SG 1 U-tubes level increase, at 13300 s a first notable over-spilling happens. That causes a rapid dilution of the boron concentration in the LS 1 under the SG 1 outlet. Between 13500 s and 14780 s after SOT the boron concentration drops from an initial value of approximately 2150 p.p.m. to approximately 830 p.p.m. This overspill creates a positive flow conditions in these U-tubes. That allows coolant to be transported from the inlet to the outlet of the SG and onward through the LS. Primary coolant with a low boron concentration below the reactor coolant pump and in the upper DC region is consequently transported to the core, producing a short term decrease in the boron concentration there. Then the boron concentration in the LS rises because the diluted water accumulated at the inlet of the SG U-tubes is replaced by the higher borated water in the HL, and then it starts to decrease slowly (reflux condenser mode conditions occurring in the primary system). The primary pressure is stabilized at 4.8 bar, with the SG 1, now, capable to remove the energy of the primary side. During this phase, the pressure in the secondary side of SG 2 remains above the pressure of the primary side. This is related to the presence of the non-condensable in the closed system.

Until the first accumulator injection a quasi steady state condition prevails in SG 1 with intermittent flow through individual U-tubes and the reflux condenser conditions in the remaining U-tubes.

The third phase of the experiment is characterized by the hydro-accumulators activation. Five accumulator injections are performed: four of these into CLs of loops 1-4 and one into HL of loop 4. The purpose of this phase is to investigate the effect of the injection location (Cold and HLs) and the effect of different injection masses on the RCS pressure profile. The last phase (phase 4) is connected with the restoration of the RHRS system that removing the decay heat causes a continuous decrease in the pressure and temperature. From 39380 s till the end of the test the core remains in subcooled condition. The experiment is stopped at 42725 s with core outlet temperature of 98 °C and the pressure of 4.3 bar.
3 E2.2 CODE APPLICATION

3.1 RELAP5 Model for E2.2 Test

The PKL III RELAP5 model used to study the E2.2 experiment is reported in figure 1 and is the DIN modified version of the model delivered by FRAMATOME-ANP [4].

The RCS nodalization comprises the rod bundle vessel containing the heater bundle, the four loops, with the reactor coolant pumps and the SGs, the PRZ and the DC model. The four loops are modelled separately. The vessel nodalization comprises the UH, the UP, the reactor core, the reflector gap and the LP. The core region is modelled with a pipe and an annulus which simulates the reflector gap of the facility. The 314 core channels are lumped in only one thermal hydraulic region, thermal coupled with three different active heat structures in order to simulate the three concentric zones. The annular part of the vessel DC is modelled with a “fictitious” 3D model taking into account the geometric location of the CLs. The four SGs are modelled separately and the two SG DC pipes are lumped in only one pipe in each SG. The U-tubes are modelled with three different tubes each one of different height that respects the elevation vs volume ratio and the real heat-exchange surface. The nodalization models the main steam piping system present in the facility. In particular the main steam line, the warm up line, the steam header and the silencer are modelled separately for each SG. The PRZ heaters and SG heaters are modelled as well.

Figure 1: PKL III reference RELAP5 nodalization of A) the vessel and B) the loop 2.

3.2 RELAP5 Model Qualification Process

A nodalization representing an actual system (ITF or NPP) can be considered qualified when a) it has a geometrical fidelity with the involved system, b) it reproduces the measured nominal steady-state conditions of the system and c) it shows a satisfactory behaviour in time dependent conditions. Taking into account these statements, the standard procedure reported in [12] has been considered. The steps of the qualification process performed in this phase of the activity are: a) the steady state results analysis, b) the reference calculation results analysis, c) the results from sensitivity studies (still in progress).
3.3 RELAP5 Model Steady State and Conditioning Phase Level Qualification Process

The steady state level qualification process has been performed and the steady state acceptability criteria described in Ref [12] have been verified. The conclusions of this step of the code assessment procedure are: a) the criteria for nodalization qualification are fulfilled, b) the steady state and the conditioning phases are reproduced with a very good accuracy. In particular, during the reflux condenser phase, the mass of condensate predicted by this RELAP5 simulation is in a good agreement with the experimental data (about 850 kg). The initial conditions obtained in the RELAP5 simulation, at the end of the conditioning phase, are in a very good general agreement with E2.2 experimental values as shown in figure 2 A [4].

3.4 E2.2 Reference Calculation and Sensitivity Analysis Results

The E2.2 reference calculated results show a good agreement with the experimental data for a number of main important variables as the primary system pressure, the level in RPV, the level in PRZ, reported in figure 2 A, and SGs and the value of mass flow in core. The primary mass inventory has the same behaviour as the experimental case, reaching the minimum value (about 1250 kg) at about 1200 s after the start of the test and the maximum value (about 2880 kg) when the HPIS is switched off. In the reference calculation, only three equivalent U-tubes for each loop are used to model the SG primary side, the results show a simultaneous restart of NC in the four loops and a minimum boron concentration of 700 p.p.m. (experimental data below 500 p.p.m.) at CL 3 and 4 RPV inlet after the start of NC. A preliminary sensitivity analysis (indicated as SEN7) aiming at the improvement of the NC restart in the four loops, is carried out by simulating the SG primary side with seven tubes of seven different lengths. The simulation shows that the NC restart almost simultaneously in the loop 1, 3, and 4 and later in the loop 2, figure 2 B and 3A.

![Figure 2: A) E2.2 experiment and reference analysis results for the PRZ level; B) sensitivity analysis (SEN7) results for the mass flow rate in the different loops.](image)

The boron concentration in the CL 3 and 4 RPV inlet drops at about 500 p.p.m., for a short time, as shown in figure 3 B; boron concentration in the loops 1 and 2 remains constant. The results of other sensitivity analysis show the importance of the mass flow through the reflector gap and the UH by-pass to predict the main thermal-hydraulic parameters during the transient. These quantities shown itself to be key parameters to correctly predict the UP, UH and the annulus DC thermal-hydraulic behaviour.
4 E3.1 CODE APPLICATION

4.1 RELAP5 Model for E3.1 Test

The nodalization of PKL III facility used to simulate the E3.1 experiment and to participate in the “OECD SETH/PKL benchmark on test E3.1” in 2006, is a scaled nodalization of ANGRA-2 NPP, that is a PWR Siemens-Framatome reactor. After the analysis of the previous reference calculation [7], presented at the “OECD SETH/PKL benchmark on test E3.1” [6], and the analysis of some sensitivity calculations [13], a preliminary review of the SGs secondary side model was carried out in order to better reproduce its thermal hydraulic behaviour.

The PKL III nodalization vessel consists of a core channel heated by 3 active heat structures representing the 3 radial concentric heated zones. The two DC pipes have been modelled with 2 pipes connected to the LP region and to a common volume representing the annular region of the DC in the upper part of the vessel where the CLs are connected. In the above mentioned previous sensitivity study, the annular part of the vessel DC was modelled with a “fictitious” 3D model taking into account the geometric location of the CLs, but it is not used in this first results updating. Two pipes model the UH in order to allow recirculation in this zone. The four loops are modelled with pipes, primary pump component in each loop and only one pipe representing the U-tubes preserving the heat exchange surface area. The secondary side of the facility is also modelled. Each loop has a SG composed by a main pipe, the riser; the 2 DC pipes are represented by one single pipe. The time dependent volumes representing the steam lines are connected in the top side of the SGs nodalization. The steam pressure and the level control in the SGs are represented by time dependent volumes and junctions. The length of the volumes in the riser region follows the same length progression increase of the U-tubes to which they are associated. The compensation for heat losses circuit has also been modelled but not activated during the E3.1 simulation.

4.2 RELAP5 Model Steady State Level Qualification Process

The steady state level qualification process has been performed and the steady state acceptability criteria described in reference [12], have been verified. The conclusions of this
step of the code assessment procedure at steady state level are: a) the criteria for nodalization qualification are generically fulfilled, b) the initial conditions obtained in the RELAP5 simulation, at the end of the preliminary phase, are stable and in general agreement with the E3.1 experimental values. Few discrepancies in the initial conditions at the SOT are due to the assumed hypotheses to simulate the preliminary phase.

4.3 E3.1 Reference Calculation and Sensitivity Analysis Results

The post test analysis has been limited to the first two phases of the experiment, just before the accumulator’s injection, as in the specification of the previous OECD SETH/PKL benchmark on test E3.1. A comprehensive comparison between measured and calculated trends or values has been performed.

In the new reference calculation analysis (indicated as REF) at the beginning of the test when the RHRS has been shutdown, the core is completely covered by subcooled water and there is no mass flow to remove the decay heat. The calculated core outlet temperature reaches the saturated value after about 578 s of delay compared to the experimental data, due to a prompt liquid circulation between the core and the reflector gap. The core inlet temperature increase starts around 200 s after the SOT (about 1300 s before the experimental data) due to the beginning of the liquid circulation inside the core. The core outlet temperature shows the same increase as the experimental value in the first part of its rise; an overestimation is present in the second part and during the “quasi” steady state condition. When saturation temperature is reached at core outlet the simulation shows a decrease of the vessel collapsed level and a two phase mixture enters in the HL of loops 1 and 2. After that, the mixture levels arrive at the inlet of SGs and the reflux condenser mode starts followed by the heat exchange from primary to secondary. The calculated results show an insurge of two phase mixture in the PRZ via the surge line and an increase of the PRZ collapsed level. The collapsed level in the PRZ has the same trend as the experimental data in the first part of its rise, and shows an overestimation in the second part reaching a “quasi” steady state condition, as shown in figure 4 A. The PRZ pressure begins to rise about 100 s before the experimental data and shows an overestimation during its rise and during the “quasi” steady state condition. The pressure overestimation during the “quasi” steady state condition is due to the overspilling absence in the calculated results. In the primary side the steam flows to UH via UP then arrives in CL and DC pipes after passing through the UH by-pass and DC annulus. The effect of this flux is one of the reasons, in the calculated results, of the rise of the DC, CL and LS temperature although the absence of the overspilling phenomenon. The condensation in CL and DC produces boron dilution. Differential pressure between HL and CL across LS produces boron dilution and transport. The simulation doesn’t predict the overspilling in the SG 1 but predicts the mass displacement from SG 2 to SG 1 (primary side) figure 4 B, although the SG primary side is modelled with only one tube. The calculated results predict the temperature build-up in the core but don’t show the transit of a cold-water slug at the core inlet. The PRZ temperature behaviour is qualitatively predicted and shows an overestimation compared to the experimental data. The boron concentration dip and the dilution rate in the LS of loop 1 is not predicted because of the overspilling absence in the calculated results, as shown in figure 5 A. The short term decrease in boron concentration at the core inlet is not predicted.

The primary side thermal hydraulic parameters, about 21000 s after the SOT, show a temporary perturbation. This perturbation seems due to a temporary increase of the UH by pass flow. This flow increase creates a temporary reverse flow into the core from the HLs. This reverse flow creates a decrease of the water level in the SG 1 U-tubes and a temporary
decrease of SG 1 heat exchange. The PRZ level shows a temporary decrease due to a reverse flow through the surge line.

Secondary side pressure increase is due to the heat exchange from primary to secondary side by reflux condenser mode. The SG 1 trend is correctly predicted but is needed a more detailed model of the steam line in order to correctly reproduce the SG 1 experimental pressure oscillation, as shown in figure 5 B. The SG 1 pressure control starts about 350 s before the experimental data. The SG 2 pressure shows an overestimation with regard to the experimental parameters. The temperatures behaviour on the SG 1 secondary side is well predicted by RELAP5 calculations in the “quasi” steady state condition. The feedwater and steam flow in SG 1 show continuous oscillations; the degree of the SG 1 feed water and steam flow oscillations, present in the calculated results, depend on the method adopted for the feedwater and steam discharge regulation.

Figure 4: E3.1 experimental and calculated level results for A) PRZ; B) SG 2 tube 1 inlet.

Figure 5: E3.1 experimental and calculated results for A) boron concentration in the LS 10; B) SG 1 secondary side pressure.

The sensitivity analysis (indicated as SEN) is carried out in order to investigate the code sensibility to the effect of the UH by-pass [14] in the prediction of the level and pressure of the PRZ and in the prediction of the main thermal hydraulic primary parameters, during the transient. This sensitivity analysis, carried out by closing the UH by pass, shows a better prediction of some primary side thermal hydraulic parameters; in particular the PRZ level
shows a significant decrease, as shown in figure 4 A. The simulation again doesn’t predict the overspilling in the SG 1. The DC pipes temperature shows a reduction with regard to the reference calculation results and continues to show a missed behaviour with regard to the experimental data. The SG 1 and 2 pressure start their rise before the reference and experimental data, figure 5 B. The previous temporary perturbation of some primary side thermal hydraulic parameters is now not present.

5 CONCLUSIONS

The analyses here presented are focused on assessing the capability of the RELAP5 code to correctly predict boron dilution phenomena and the thermal-hydraulic parameters in transients with a) asymmetric loop behaviour, b) NC one and two-phase flow, c) SG in reflux condenser mode, and (only for E3.1 experiment) d) the primary system under low pressure conditions in the presence of nitrogen.

The E2.2 reference calculated results show a good agreement with the experimental data for a number of main important parameters. By using only three equivalent U-tubes for each loop to model the SG primary side, the calculated results show a simultaneous restart of NC in the four loops and an overestimation of the boron concentration at CL 3 and 4 RPV inlet after the start of NC. The sensitivity analysis, carried out by simulating the SG primary side with seven U-tubes of seven different lengths, shows that the NC restarts almost simultaneously in the loops 4, 3, and 1 and later, in a good agreement with the experimental data, in the loop 2. The boron concentration at the CL 3 and 4 at the RPV inlet drops at about 500 p.p.m., for a short time after the start of the NC in these loops, in agreement with the experimental data and their related uncertainty.

The results of other sensitivity analysis show the importance of the mass flow rates through the reflector gap and the UH by-pass to better predict the UP, UH and the annulus DC thermal-hydraulic behaviour. More accurate studies of condensation processes in the PRZ and UH region in this transient could improve the simulation results during the phase in which the pressure rises rapidly, after the LPIS intervention.

The E3.1 experiment is challenging for the code because it is at low pressure and in the presence of non-condensable. The difficulty to simulate the experiment is due to the simultaneous presence of low pressure condition and of nitrogen in various parts of the facility, the variation in the performance of individual U-tubes and the large influence of pressure drop and heat losses including their spatial distribution. The correct prediction of the transient is strongly influenced also by the nitrogen and the void fraction distribution at the beginning of the transient.

The model with only one equivalent U-tube resulted not suitable for the prediction of the overspilling phenomenon in this kind of transient. Therefore the evolution of the boron concentration in the LS and core inlet is not reproduced by the code. The results of the previous E3.1 benchmark show that it might be necessary to simulate the steam generator U-tubes in a “tube-by-tube” fashion in order to correctly predict the unplugging/overspilling phenomenon observed in the experiment, but the long computing time for the current modelling also shows that this approach is unfeasible with current monoprocessor machines.

The fill up of the PRZ shows an overestimation compared with the experimental trend.

The E3.1 sensitivity analysis, shows a better prediction of some primary side thermal hydraulic parameters although the overspilling is not predict by the code. In particular the PRZ level shows a significant decrease.

Again, more investigations need the effect of the reflector gap mass flow rate and the UH by pass, in order to investigate the code sensibility to them, in this kind of transient, and
better reproduce the general thermal-hydraulic behaviour. The prediction of the water levels and the non-condensable redistribution during the transient should be more analyzed.

REFERENCES


ABSTRACT

Process plant safety and nuclear plant safety have grown up as two substantially separated branches of the safety technology. There are good reasons for that because nuclear plants employ essentially one process only (heating water by nuclear power, producing steam and finally electricity), process plants use many different processes; the size (investment) of the plant also is very variable, as opposed to the situation of a nuclear plant. However, both technologies might benefit of enhanced knowledge exchanges. Fields where such exchanges could be increased are: more intrinsic and passive safety, probabilistic analysis of large plants, fire prevention and mitigation, defence against external natural and man-made events. On the contrary, fields where beneficial interactions seem less promising are: index-based analysis methods, probabilistic analysis of small plants, containment technology. All the previously listed issues are discussed in the paper. Finally, possible ways to enhance knowledge transfer between the two fields are addressed.

1 INTRODUCTION

Process plant safety and nuclear plant safety have initially grown up as two substantially separated branches of the safety technology. There are good reasons for that because nuclear plants employ essentially one process only (heating water by nuclear power, producing steam and finally electricity), process plants, on the other side, use many different processes. The size (investment) of the plant also is very variable, as opposed to the situation of a nuclear plant.

However, both technologies might benefit of enhanced knowledge exchanges. Fields where such exchanges could be increased are:
- more intrinsic and passive safety
- probabilistic analysis of large plants
- fire prevention and mitigation
- defence against external natural and man-made events

On the contrary, fields where beneficial interactions seem less promising are:
- index-based analysis methods
- probabilistic analysis of small plants
- containment technology

2 FIELDS WHERE EXCHANGES SHOULD BE INCREASED

2.1 More Intrinsic and Passive Safety

2.1.1 General Remarks
The nuclear reactors now operating incorporate both passive and active safety features. As an example, reactors have a passive limitation of power excursions through a negative power coefficient of reactivity, which is, for most of them, the outcome of an early recognition that a power excursion might be difficult to limit in presence of self-enhancing dynamic reactor features. On the other side, most reactor emergency cooling systems are active. The variety of solutions does not reflect a precise choice in the early days of nuclear power towards active or passive systems: it reflects the best choice according the sensibility of the designers of that time. Passive and intrinsic safety solutions were adopted when recognized as effective and economically convenient. Moreover, the fundamental safety functions to be accomplished in a nuclear reactor are limited to reactor shutdown, reactor and containment cooling and containment of radiotoxic products: the most natural engineering solutions for these functions were in general adopted, with obvious variations, in all of the reactor designs developed.

With passing time, in depth safety studies and data of operating experience both tended to expand the safety requirements beyond those originally devised. Plants became more complex and part of the passive safety originally present tended to disappear. This is evident, as an example, in containment cooling, which was originally more entrusted to passive, natural mechanisms.

At a certain time in its development, the nuclear industry had, unfortunately, to suffer the two big accidents of Three Mile Island and of Chernobyl, different in many respects from each other, but equally rich of lessons in their applicable technical environment.

Besides these two events, and in a certain sense, anticipating, some of their indications, the integral safety studies of typical plants, starting with the Rasmussen study, called the attention of the technical experts on the need for a complete rethinking on the safety approach until then followed.

Since then, everybody in the technical field was convinced, or, I should say, even more convinced, that accident prevention and mitigation in nuclear plants deserved a very special attention: serious accidents could be avoided, but a continued attention to safety in design and operation was warranted, also including the consideration of important plant design alternatives.

Some facts, in particular, became even more evident than before: in the first place the potential importance of multiple failures in complex safety systems and, secondly, the possible serious consequence of human errors.

Hence, attention was called on passive and, as an extreme, on inherent or intrinsic safety systems which needed less auxiliary systems, were simpler, with a lower number of parts which could fail and did not require as much operator intervention as active systems.

“Passive safety” is the expression currently used to qualify the operating safety features of structures and devices designed to counteract specific events without reliance on mechanical and/or electrical power, forces or “intelligence” signals external to the same structures and devices. These features should rely only on natural laws and properties of materials, as well as on lack of human action. Different degrees of passivity exist; as an example, a safety system may operate without external power but may require some sort of active actuating signal: in this case, too, the system is qualified as passive even if not to the highest degree.

“Inherent safety” means the elimination of hazard by choice of material or design concept; as an example, elimination in a plant of any combustible material (if possible) would implement inherent safety in front of the danger of fires.
In the last few years, much discussion took place on the merits of passive and intrinsic safety, mostly because, to my opinion, it was too easily postulated by the public that such a thing as an intrinsically safe plant, or, at least, as a passive safety plant, should necessarily exist: the attention was erroneously misplaced more on "intrinsic-passive" than on "safety".

Although it is evident that a substantial research and development effort on simpler and less vulnerable nuclear plants is still warranted, it appears now more generally recognized that the best possible and safest plant at this point in time and one in which serious accidents can be avoided throughout all of its life, probably includes both active and passive features in an optimization perspective. Passive systems, although at first sight attractive for their simplicity, may have drawbacks, as the one of being less powerful and slower in their action; moreover, their reliability is more difficult to evaluate.

The conceptual development in the process (mainly chemical) industry is somewhat similar. Even there, a number of Three Mile Island-Chernobyl type of events exists, which are named Flixborough, Seveso, Bhopal and others.

The first one was an open-air explosion of a flammable gas (Unconfined Vapour Cloud Explosion, UVCE) released into the air by a Nylon plant, which killed the 28 plant employees present and caused extensive property damage in the surrounding territory. Too large inventory of flammable substances and faulty maintenance operations were mainly blamed as the cause of the event. The second is well known for the dangerous release of dioxin due to poor plant safety features and to human underestimate of the possibility of a runaway reaction. The third one, which killed a still unknown number of people in the order of 4000, was referred again to too large inventory of toxic substances and to very poor staff attention to operability of safety features.

Also in the process industry, plants tended to grow bigger and bigger with passing time and to became, therefore, more complicated and dangerous for the high amount of stored chemicals.

A rethinking period then started also in the chemical industry, pointing to the study of "more inherently safe" plants. The wording chosen is indicative of the need to eliminate the wrong idea of a completely safe plant.

In the following, the main directions followed by these studies in the two fields explored and the main results will be discussed.

2.1.2 Some passive systems and components for nuclear plants

Systems and components discussed in the last few years range from complete reactor concepts to single components.

References [1] and [2] list and comment most of the existing proposals.

A rather arbitrary selection of a few among such proposals is presented in the following. They are all well known concepts in the nuclear industry and they are here recalled because they are considered among the most interesting ones.

Passive plant reactors (e.g., AP600, AP1000 W) are proposed future reactors that use the technology of current reactors, but include also significant changes in plant design and layout.

Safety, in the event of an accident, depends on passive safety systems and on safety systems which are passive in operation although started up by a simple action such as valves opening.

In AP 600 and AP 1000, PCCS (Passive cooling containment system) is provided to remove heat from the steel reactor containment. The operation of PSIS (Passive safety injection system) following a LOCA results in steam released from...
the reactor core being passively condensed inside containment. Steam condensation reduces containment pressure. The PCCS firstly consists of a large tank above the containment structure that allows gravity drain of water on the outside of steel containment vessel. Secondly, opening of air dampers supplies natural circulation air cooling of the external surface of the steel containment. The air and evaporated water exhaust through an opening "in the roof of the shield building. The passive containment cooling system is capable of removing the thermal energy following a design basis event so that the containment pressure remains below the design value with no operator action required for (three) days. The passive containment cooling system is designed to reduce containment pressure to less than one-half its design pressure within 24 hours following a LOCA. After three days, if there's no supply of water, the heat removal is assured by air alone with an increased pressure (till about design pressure).

In nuclear power plants, the containment is the final barrier to prevent the radioactive release to the environment during accident events. Because of containment importance in mitigating the postulated consequences of an accident, it is necessary not only to assess its integrity during, but also to ensure that it is and stays leak-tight after accident occurrence.

Typical allowable primary containment leakage rates lay in the range of 0.1-1% volume/day, but the operating experience sometimes has indicated "real world" values above allowable limits. That is usually due to excessive valves or penetrations leakage, valves or penetrations left open after testing, airlocks failure etc.

Studies have been made on the following aspects:
- containment leak tightness enhancement (better choice of valves type, reduction of the number of penetrations, valves stems leakage reduction etc.)
- research of root causes for leak tightness degradation (e.g. debris reduction and deposition on valve seal surfaces and valves behavior under severe accidents)
- conception of a secondary containment to reduce the primary containment releases by hold-up, deposition, filtration, elevated release (for example a secondary containment that envelopes possibly affected buildings equipped with filtration systems)
- monitoring capabilities to detect pre-existing openings in the containment boundary (e.g. monitoring nitrogen leaks in inerted containments)

The ALWR passive plants employ safety grade passive decay heat removal (PDHR) systems in order to enhance the capability (relative to current plants) to maintain the plant in a safe shut-down condition following non-LOCA events.

The approach developed for these systems is founded on meeting the following requirements:
- the PDHR system is employed for both the hot standby and long-term core cooling modes. This system can operate at full reactor coolant system pressure and places the reactor in the long-term cooling mode immediately after shut-down.
- operation in the long-term cooling mode is automatic.
- operation of the system does not require any ac power, either on site or off site.
- operation of the system does not require any pumps or valve operation once initial alignment is established.
- no make-up water is required for a period of at least three days following reactor shut-down.
- the systems are located entirely within containment.
The passive RHR systems haven't, however, the ability to bring the plant to cold shut-down conditions of 100° C. This is inherent in the passive heat removal process itself because heat removal is accomplished by heat exchangers located within a pool of water, and the temperature on the reactor coolant side of the heat exchanger tubing will, of necessity, exceed the boiling point of water at normal pressure. Cold shut-down can be achieved by the reactor shut-down cooling system, proposed as a non-safety-grade system.

More in particular for non LOCA events the AP600/1000 PRHR system, for example, is designed to perform the following functions:

- automatically actuate to provide reactor coolant system cooling and to prevent water relief through the pressurizer safety valves;
- removing core decay heat assuming the steam generated in the in-containment refuelling water storage tank (IRWST) is condensed on the containment vessel and returned by gravity into the IRWST. The PRHR should provide decay heat removal for at least 72 hours if no condensate is recovered;
- the PRHR heat exchangers are designed to cool the reactor coolant system to 400 F (200 °C) in about 72 hours;
- during a steam generator tube rupture event, the PRHR system remove core decay heat and reduces reactor coolant system temperature and pressure, equalizing primary pressure with steam generator pressure and terminating break flow, without overfilling the steam generator.

During the TMI accident, one of the strategies unsuccessfully tried by the operators to regain control of core cooling was to depressurize the reactor system: the reactor was not designed for that operation and the manoeuvre did not succeed. A reactor depressurization system would probably have helped there.

Moreover even the initial PRAs did evidence the possibility of high pressure severe accident sequences for current LWRs. The idea then started to be studied of designing a depressurization system into LWRs. This was a new thing especially for PWRs [3] since BWRs had relief system in order to cope with the possibility of loss of condenser accidents. In principle, a primary depressurization system has many advantages: its operation tends to create an immediate, yet temporary, reactor shutdown effect; it decreases the primary water temperature and favors core cooling; finally, it allows water to be supplied to the core either by high pressure injection systems and by low pressure "jury-rigged" emergency systems (fire truck water and so on). New passive LWRs incorporate a powerful depressurization system which allows emergency water injection to be made by gravity driven (passive) arrangements. Moreover the operation of the primary depressurization system also ensures that the reactor coolant system would be depressurized during a severe accident. Therefore, violent ejection of molten core debris from a pressurized reactor coolant system is highly unlikely for the passive plant with a corresponding reduction in the potential for direct heating of the containment atmosphere. That is also applicable to the evolutionary light water reactors, in fact the NRC staff has concluded (SECY 90.016) that ALWR designs (evolutionary and passive) should include a depressurization system to preclude the ejection of molten core debris under high pressure from the reactor vessel.

Nevertheless the reactor coolant release to containment has the potential for adverse effects on in-containment equipment.

Accordingly, the ALWR plants should be designed to minimize such adverse effects by ensuring that the frequency of inadvertent actuation is extremely low (2x10⁻³/y for passive plants, EPRI, Electric Power Reasearch Institute, U.S.A., requirements) ensuring that recovery from such inadvertent actuation is feasible.
without compromising plant availability for a long period (EPRI requirements for passive plants: recovery within 30 days or less). As an example a short description of the AP 600/1000 depressurization system is mentioned in the following.

The AP600/1000 automatic depressurization system consists of sixteen valves divided into four depressurization stages. These valves are installed in the reactor coolant system at three different locations. The first three stages valves are connected to nozzles on top of pressurizer. The fourth stage valves are connected to the hot leg of reactor coolant loop. The main actuating signals for each depressurization stage come from different level set points in the core makeup tanks (CMTs that provide high pressure make-up by gravity). When the CMT is going to deplete, the depressurization takes place to allow low pressure injection from IRWST (in-containment refuelling storage tank) by gravity.

Moreover the depressurization system together with passive injection of borated water from IRWST could ensure safe shutdowns in the long term in case of ATWS if other active systems are not available for this purpose.

The design of hydraulic engineered safety features of LWR’s has been traditionally performed according to high reliability and leak tightness standards. These systems are usually called into operation to protect the fuel barrier in the case of a loss of the primary system barrier. In addition, being strictly connected to the primary circuit pressure boundary, they have to be equipped with leak tight isolation devices, normally closed during plant operation. Squib valves, initially used for applications in the space industry, have been considered very attractive for the application in an advanced passive reactor. These valves are characterized by a no-leak capability and, once actuated, they are designed to maintain the open position.

The inlet chamber of the valves in normally closed by a sealing cap. When the valve is actuated, an explosive initiator pushes a plunger that shears the cap off.

This kind of actuation has resulted very reliable from operating experience and qualification tests. These valves require very limited maintenance. If fact no periodic intervention, other that the substitution of the initiator, is necessary.

Additional benefits associated to their use in Automatic Depressurization Systems are related to the possibility of providing a flow area larger than that traditionally obtained with standard Safety Relief Valves (SRV). Such a large area is very important in passive reactors to depressurize the primary system at very low pressures, consistent with the operation of injection systems based on gravity.

The installation in the core cooling injection system, in addition to the benefits associated to the leak tightness characteristics, allows to ensure, during normal operation, a pressure shielding function on the upstream check valves. Therefore, these valves do not remain forced in the closed position for long times, and that improves their reliability when called to open under a low differential pressure.

The "density locks" (or "hot-cold interfaces") are passive devices which perform the same function of normally closed valves during normal operating conditions. However in case of transient or accident conditions they allow cooling flow without need of power supply or motion of mechanical parts.

The density locks have been applied in the PIUS (Process inherent Ultimate Safety) reactor concept [1] [2]. In it the reactor core is immersed in a large pool of pressurized, cold, borated water. The hot primary water and the cold pool water are in contact by two "hot-cold interfaces" (high and low elevation in the cooling
circuit) where, during normal operation, substantial mixing is prevented by design details and by pump speed (head) adjustment, governed by the lower interface temperature. In case of uncontrolled accidents of any origin, the core will tend to overheat causing water boiling and the decrease of the hydrostatic head in the riser pipe above it, beyond the correction capability of the pump speed control system. In these conditions, natural circulation between the cold pool, the core and the riser pipe will be established through the two "hot-cold interfaces" along an always open natural circulation path. The pool cold borated water will then enter into the core and will shut the reactor down and remove the decay heat. In a certain sense, PIUS safety is based on the use of an essentially unstable cooling circuit, which needs active pump action to ensure stability during normal operation; in off-normal conditions, the system automatically switches to its stable condition which also is a safe shutdown condition. The "density locks" carry on a fundamental role in PIUS to ensure core cooling during emergency conditions and thus the potential for their blockages caused by gas collection, material distortion or plugging by detached insulating materials should be analyzed in depth. The density lock concept has been used in other new reactor schemes. Fluidic diodes and vortex valves are passive devices whose application to future NPP's is currently under evaluation with reference to their potential of use as check valves or actuation valves in safety related systems. Fluidic diodes, used in reprocessing plants and chemical industries, are one-way valves with no moving parts. They are characterized by a very high flow resistance in one direction with respect to the other. This characteristic allows their application to NPP's as flow limiters to maintain core coolant boundary integrity in the case of a LOCA event. In a potential application to a typical PWR system, a fluidic diode is installed on the reactor pressure vessel nozzle of cold legs to avoid reverse flow conditions following a pipe break. Due to the diodes characteristics, instead of a massive release of coolant, only limited leaks would occur. Vortex valves are "normally active - passive during emergency" devices designed to maintain separation between environments normally operating at different pressures. This function is performed by the fluid movement provided by a normally operating pump. A potential application to NPP's safety features is as actuation valves in case of transients or accidents. In fact during normal operation the two environments remain isolated as if they were provided by a standard isolation valve. Following a transient the pump operation is expected to be interrupted or its head capacity to be overcome and water can flow from the environment at high pressure to that at low pressure.

In the field of Process Industry Plants the concept of more inherently safe design is a recurring theme in the three reports of the Advisory Committee on Major Hazards (ACMH), which was set up in U.K. after the Flixborough accident. These reports set the general principles of the "new" process industry safety in U.K. and they represent in their field what, as an example, the IAEA "Safety Fundamentals" documents does in the nuclear industry. A full account of the developments of this concept is given in References [4], [5],[6]. The magazine "Loss Prevention Bulletin, Institution of Chemical Engineers, England" is also a "must" for interested people; it is available in most technical libraries and a list of the main articles appeared over the years is included in Ref. [4], Vol.3. The basic principles of inherently safer designs in the process industry are:

- intensification, namely carrying the chemical reaction in a smaller volume in order to have a lower inventory of dangerous substances and smaller consequences of an accident,
- substitution (of a dangerous process or substance, e.g. an heat transfer medium, with a less dangerous one),
- attenuation (adoptation of a less hazardous process condition, e.g. of a lower pressure in combination with the improvement of a catalyst)  
- simplicity ( e.g. designing a vessel or pipe for full overpressure instead of adopting a pressure-relief system); as Henry Ford used to say "What you don't fit costs nothing and needs no maintenance",
- operability (adaption of a process which can be easily controlled and adjusted to off-normal conditions) 
- fail-safe design (where the failure of the system leads directly to a safe condition) 
- second chance design (second line of defense)

Interesting examples of proposals in the process industry follow.

The first typical example concerns the fabrication of nitroglycerine. It has to be classified as an "intensification" of the process, namely as a drastic reduction of the inventory of the dangerous substance. Nitroglycerine is manufactured by the reaction between glycerin and a mixture of concentrated nitric and sulphuric acid. The reaction is highly exothermic and the mixture has to be continuously cooled and stirred, otherwise a violent explosion may occur due to the uncontrolled decomposition of nitroglycerine. Originally the reaction was performed in batches using large (one ton) pots. The operator had to continuously monitor the temperature and check that stirring was effective: since the reaction lasted a rather long time (hours) there was the danger for the operators to fall asleep and, therefore, they used to work sitting on one-legged stools, as it can be seen in historical pictures, one of which is sketched in Figure 1.

![Manufacture of nitroglycerine in old times](image)

Figure 1: Manufacture of nitroglycerine in old times

This kind of process continued to be used until fifty years ago with a number of casualties and complete plant losses. The same reaction is now obtained in a, small injector where the acid jet entrains the correct amount of glycerin and, due to the turbulent mixing, the reaction time has been reduced down to minutes and the reaction is complete at the exit from the injector. The amount of nitroglycerine in the reactor is reduced to a
few kilograms and the operators can be protected by a blast wall. In another reaction, the adipic acid reaction, the process was previously performed in a huge reactor with external circuits for cooling; now it is performed in a smaller integral vessel with internal cooling and agitation and with a very smaller possibility for leaks. A similar evolution has taken place in nuclear reactors which were transformed from external to internal recirculation units or in integral proposals for future reactors.

It is also worth mentioning the so called Higee ICI process, where the process of gravitational separation is enhanced by centrifugal forces in a rotating unit, with consequent decrease in amount of substance in the separator.

Many examples are available concerning the substitution of one process with a less dangerous one.

In a number of cases in the chemical industry the choice has to be made between the availability of a large storage of substances and the reduction of stored substances concurrent with the continuous production of them on site. In the first-case, continuity of production is better assured but the risk of the storage is present. In the second case, the opposite advantages- drawbacks are present. The concept of inherent safety would incline to make the second choice. It has to be remembered that in the case of Bhopal, a total pessimization of the situation was created because at a certain point in time it was decided to produce methyl isocyanate (MIC, the poison which was released in the accident) on site instead of importing it from another factory, but the already existing huge MIC tanks continued to be used with the consequent risk. Major reductions of inventories have, however taken place in the last years on safety grounds, following the Bhopal event and some new regulations concerning, in particular, hazardous substances as ethylene oxide, propylene oxide and sulphur trioxide.

A large progress in the safety of the chemical industry is being made in a field of strong interest for the nuclear plants too: the reduction of the possibility of leaks from containments through the reduction in the number and the dimension of penetrations.

Simplification of detailed design is also pursued by such measures as design for overpressure and design modification to avoid instrumentation, simple cases of the latter operation is the use of suitable piping arrangements to avoid reverse flow and to provide for automatic sump voiding (high turns of pipe with anti-syphon openings, selfpriming syphoons and so on).

It seems worth noting, concerning the operability-,concept of the above included list, its similarity with those provisions, in the nuclear plants which tend to provide a longer "grace period" in case of mistakes or accidents (increase of the water inventory in water reactors, and so on).

Speculative proposals for the long range also exist. One of them considers the advantages to have distributed manufacture of chemicals using miniaturized plants at the users site; such plants would be more environmentally friendly and would deliver their products on a "just in time" basis; they should also be completely automated, highly reliable, self cleaning and sealed for life.

As it is apparent from what discussed above, in a number of instances the process industry has gone beyond the study phase in the way of the adoption of more inherently safe provisions.

Safety experts in the process industry, however, complain that not enough has been done [5] in this direction. Some of the constraints towards a higher level of inherent safety are: - time ("the technical options available for the next plant are usually limited by time, so if major advances are to be made there has to be thought about the "Plant after next", namely during the design stage of a plant there is not enough time to discuss and to develop alternative designs); - desire for certainty of production (if a new process or a
new equipment is used, then unforeseen difficulties may cause trouble during start-up, perhaps delay or prevent the achievement of design output or efficiency); - the influence of the process licencers is often on the side of tradition (for the possibility of unforeseen snags and surprises); - technical misconceptions (like the belief that, e.g., reduction in the inventory of dangerous substances may render the control of the process more difficult); - organization scheme (the organization of a company in business areas instead of in functional departments is not favorable to innovations because of the strong influence of the control of expenditures); ill defined responsibility for design innovation (research departments or design departments).

It has been remarked that it is difficult to convince interested people that there is a problem of improvement of safety level: many are accustomed to think that hazard is inherent in industry (which may be true to a certain extent) and it didn't occur to them that in many cases it may be possible to avoid hazards.

2.2 Probabilistic Analysis of Large Plants

A full complete probabilistic analysis of a large plant may cost millions of euro. Although this amount is reduced for smaller, simpler plants, it often occurs for process plants that a full probabilistic safety analysis is not justified on economical grounds. This is the reason why this kind of analysis will be confined to nuclear plants or to large process plants or industrial areas, as it was the case with the famous Canvey Island complex analysis in years 1975-78 or with the Rijnmond area in The Netherlands.

2.3 Fire Prevention and Mitigation

Fire prevention provisions and techniques were, till several years ago, much more advanced in the process plant field than in the nuclear plants. Presently, nuclear plants regulations have made important progress in the fire protection area. A fire risk analysis is now common place in presently operating plants. Various accidental events contributed to increase the attention towards this phenomenon in nuclear plants. The most famous event was the Brownsferry Fire in 1975, where a penetration leak-tightness check made by a candle flame (very sensitive instrument, indeed) caused a local fire and the incapacitation of a number of essential electrical circuits [7]. Remedial actions by the plant personnel were put into effect, including “heroic” and creative actions not envisaged by the plant emergency procedures and the function of damaged emergency systems was recovered at last. The event was very dangerous indeed and thorough inquiries followed by general recommendations for the future.

2.4 Defence against External Natural and Man-made Events

Natural events like earthquakes and tornadoes are of interest both to nuclear and process plants. The most significant advancements in the study of these phenomena and in the response of industrial structures and components were made firstly in the nuclear field and then adapted to the more general case of industrial plants. The most vivacious developments started in the nuclear field at the end of the years ’50s. Before that time, essentially civil building rules were applied also to industrial plants. The problem is that the objectives of the seismic study of civil buildings are different from those of nuclear and process plants. In fact, as it is known, the legislator intends to reach two objectives for the protection of civil buildings:

1. Avoiding any form of damage to structures in case of an earthquake with a return time roughly equal to the normal life of a building (e.g. 100 years),
2. *Avoiding the collapse of the structure*, even accepting damages, in case of the most violent earthquake expected on the site.

On the contrary, for an industrial plant either nuclear or in any case with a risk of a serious accident, the protection objectives could be expressed in the following way:

1. *To ensure the continued operation of the plant* in the occasion of an earthquake with a return time equal to its normal life, possibly after an inspection and simple and occasional repairs of damaged components,

2. *Avoiding a serious accident* in the case of the most violent earthquake expected on the site.

As it is evident, the two points of view are different and, while the current norms considers damages and collapses, the needs of protection of a plant concern its functionality and the absence of accidents; these concepts imply, in particular, the absence of significant leaks of noxious gases and liquids, the absence of reactions and of uncontrolled and destructive phenomena and the functionality of the safety equipment (shutdown, cooling, containment and control). Moreover, the components and the phenomena of interest in industrial plants are not covered by the methods used in the civil structures. In particular, in the plants, phenomena not taken into consideration by the norms can happen and therefore the need arises to indicate acceptable verification methods, which are in any case logically compatible with the spirit of the norms themselves. A typical case concerns the phenomenon of liquid oscillations in industrial tanks caused by earthquakes and of the possible consequent effects (in particular for large atmospheric tanks, Figure 2: impact of the liquid against the roof and consequent damage, A, increase of the overturning moment on the tank and possible damage of anchors, C, and elastic-plastic instability of the vertical wall, B).

![Figure 2: Weak points of a tank in case of earthquake](image)

Concerning man-made events, it is generally thought that this issue is mainly of interest to nuclear plants, for which any accident has a wider resonance in the media and in the public opinion. They should, however, be also considered possible for other types of plants and of installations. The September 11 2001 attack to the World Trade Center in New York City was, indeed, aimed at buildings, although of extremely high significance for the public opinion. In any case, the methods of analysis and the provisions against an aircraft attack have firstly been developed for nuclear plants. This is one of the cases where a bulk of knowledge and of experience can turn to be useful also for process plants. I can quote from my experience of work the study of an aircraft impact on an offshore Liquid Natural Gas (LNG) Terminal, which was performed with satisfactory results using mainly assumptions and methods taken from the nuclear plant experience [8].
3 FIELDS WHERE EXCHANGES SEEM LESS PROMISING

3.1 Index-based Analysis Methods

These methods have originated in the chemical industry for the simplified yet complete evaluation of the safety level of a plant. Examples are the “Dow Index method” and the “Mond Index method”. In substance, a safety index is calculated as the sum of various contributions which are calculated, by formulas and tables, starting from physical quantities pertaining to the various parts of a plant, as amount of noxious substances present, operating pressure of vessels and piping systems, etc. According to the value assumed by these index, the plant is considered acceptable or not. No general consensus exists on the merits of these methodologies. The principal criticism is based on the doubts about the capability of such methods to give the correct importance to specific features of a plant, which could not be considered in the list of items included in the calculation of the index. However, these methods are rather widely used in a specific context, like sets of similar plants originated by a single design organisation; they are also used as first, quick check of the safety level in view of a subsequent complete safety analysis or for very simple plants. The index-based methods are not considered generally useful for nuclear plants and none of them has been proposed. However it is true that for a class of plants, as for example PWRs of a certain vintage, simplified analysis can be made which are based on experience and knowledge of the most likely weak points of a plant of the category under consideration. These kind of analyses are currently made during the safety reviews performed by working groups assembled upon request by international organisations (e.g. IAEA) in one or two weeks of work. The issues considered in these reviews are chosen on the basis of the past experience of the expert reviewers, who, after many safety reviews, well know where to look for possible weak points in a plant design [7].

3.2 Probabilistic Analysis of Small Plants

The cost of a probabilistic analysis is here the difficulty. The investment involved in a small chemical plant does not warrant the performance of a full probabilistic analysis. It is more convenient to exceed in the implementation of additional and superabundant safety margins in design and operation of the plant. A full probabilistic analysis is likely to stay confined to large nuclear plants or chemical complexes. Some form of limited probabilistic analysis has, however, been studied for smaller plants or parts of plants, like the Hazan method [4].

3.3 Containment Technology

Even for nuclear plants the original concept of a containment system for the confinement of possible noxious releases has for a long time been considered by many as an undue waste of money. After the Three Mile Island accident, however, where the containment played a mayor role in mitigating the external consequences of the accident, a general consensus consolidated on the advisability of a pressure resistant, leak-tight containment around most types of nuclear plants (PWRs, BWRs, FBR and HTRs in particular). No such provision is likely to be applied to other kinds of process plants, with the possible exception of small parts of them. The experience gathered with nuclear plant containments can be useful in such cases. Indeed, the concept of containment is not so simple as it appears at first glance. The containment system is, primarily, a system and not a simple passive container; automatic
isolation devices, leakage reduction and collection-filtration systems, structural protections against thermal damage in accidents and against impact effects of internal origin, protections against external threats etc., are all components of this complex concept [7].

4 Ways to further enhance knowledge exchanges between nuclear safety and process plant safety

The main point is very simple and is that if responsibles within organisations in both fields are convinced that an increased exchange of information can be beneficial to all, all the possible actions will quite naturally follow. Examples are:
- publish in technical magazines of one field some paper coming from specialists in the other field if the subject can be applicable to both,
- give room to papers in both fields in seminars and conventions, as it is done here with this invited paper,
- organize courses, masters, specialisation schools which deal with both fields,
- favor within interested organisations exchanges between the two fields, avoiding a sharp separation between differently oriented organisation units and creating occasions for knowledge exchanges (seminars, mixed working groups, etc).

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Use of PSA in the Design and Construction Phase of NPPs in Finland

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ABSTRACT

Teollisuuden Voima Oy (TVO) owns and operates two BWR units (OL1 and OL2) in Southern Finland, and it has one PWR unit (OL3) under construction. This paper presents the main requirements on the use of PSA for new nuclear plants to be built in Finland, and highlights the experiences from the application of PSA in the design and construction of Olkiluoto 3. The Finnish Regulatory Body STUK requires that the design of new nuclear power plants be supported by PSA. The paper gives an overview of the detailed Authority requirements of PSA in numerous applications. The Authority requirements are somewhat tighter for the new plants than for the operating plants, because early use of PSA makes it possible to perform design comparisons and also correct the design weaknesses and errors before commissioning. STUK reviews thoroughly the analysis two times before the commercial operation, namely before granting the construction license and before granting the operation license. Construction license assumes an approved construction phase PSA. The PSA is complemented and updated during the construction phase, and used in several applications in the detailed design. The result is the living PSA for operation phase, approval of which is a condition for the operation license.

1 INTRODUCTION

Teollisuuden Voima Oy (TVO) owns and operates two BWR units (OL1 and OL2) in Southern Finland, and it has one PWR unit (OL3) under construction. In connection with the licensing of the operating plants, probabilistic methods have been used late since the 1970’s. The Finnish Regulatory Body STUK published in the year 1985 the first version of the guide YVL 2.8, in which the requirement on PSA including levels 1 and 2 was presented. TVO started the preparation and analysis of level 1 PSA for the identical boiling water plants in the same year, and sent the first report, consisting of the analysis of internal initiating events, to STUK four years later. The PSA for OL1 and OL2 has grown during the last decade as a comprehensive PSA program, and it has been actively used in the safety management of the utility and as a communication tool in discussions with STUK.

The Finnish Regulatory Body STUK requires that the design of the new nuclear power plants is supported by PSA. The requirements are somewhat tighter for the new plants than for the operating ones, because early use of PSA makes it possible to perform design comparisons and also correct design weaknesses before commissioning. The regulatory guide YVL 2.8 also requires a PSA to be performed and applied during the design and construction of the plant. It stipulates that the contents, the documentation and the applications shall be
completed in different phases of the plant life cycle and sets requirements on the quality management. STUK reviews thoroughly the analysis twice before the commercial operation, namely before accepting the construction license application and before accepting the operation license application.

AREVA NP as the vendor is responsible for the conduction of the PSA in the OL3 project, while TVO as the owner reviews and accepts the used methods in advance and the results. However, both parties benefit on having the up-to-date model to reflect most recent design status of the plant. The owner can review in a detailed manner the applications by the vendor, assumed that the models and calculations are transparent. TVO can also make its own analyses and study design alternatives. The common PSA model improves the transparency of safety management of the OL3 project.

2 PSA OBJECTIVES

2.1 TVO practices and objectives

TVO has developed its own living PSA concept. It is widely used in safety management of the plant and fulfils the regulatory requirements. The living PSA is based on four elements:

1. Plant specific and up-to-date PSA model.
2. Plant specific and up-to-date probabilistic data.
3. Fast enough and transparent computer code to run living PSA and to support the operation department and the technical department of TVO.
4. Plant specific and up-to-date procedures for use and updating of the PSA in order to define the responsibilities for the separate organizations.

The speed of the code and the transparency of the model and the results it produces is vital. Flexible tracing of the origin of the min cut sets and studying the results of individual accident sequences are crucial in the serious use of PSA as a support for design and for daily plant operations. Efficient use of PSA requires extensive possibilities for post processing of the minimal cut sets. Similar application of living PSA concept is the objective of TVO also in the construction project of the OL3 unit. TVO has selected fifteen year ago the Finnish code SPSA. The codes for OL3 level 1 PSA are FinPSA, which is a new Windows code based on SPSA, and SPSA for level 2.

2.2 Regulatory safety objectives

The Finnish regulatory guide YVL 2.8 “Probabilistic safety analysis in safety management of nuclear power plants” shows how probabilistic safety analyses are to be performed and used in the design, construction and operation of light water reactor plants.

According to the Government Resolution (395/1991) “accidents leading to large releases of radioactive materials shall be very unlikely”. STUK has defined probabilistic target values for the core damage frequency and for the release of radioactive materials. The mean value of the probability of core damage shall be less than $1 \times 10^{-5}$/year. The mean value of the probability of a release exceeding the target value defined in section 12 of the Government Resolution (395/1991) must be smaller than $5 \times 10^{-7}$/year. The target value for the release is defined so that the release shall not have “acute harmful health effects to the population in the vicinity of the nuclear power plant nor any long-term restrictions on the use of extensive areas of land and water. For satisfying the requirement applied to long-term effects, the limit for an atmospheric release of cesium-137 is 100 TBq.” Such a release corresponds less than 0.01% of the Cs 137 inventory in OL3 core.
STUK requires that PSA is used in the design phase to support the balance of the design. The following requirement in YVL 2.8 also sets certain requirements for the tool used: “The risks associated with various initiators and accident sequences, taking into account their uncertainties, are to be compared with the numerical safety objectives and with each other in order to ensure that no single or few prevailing risk factors will stay at the plant.”

3 REQUIREMENTS ON THE SCOPE OF PSA

3.1 Authority requirements

There are requirements on the content and documentation of PSA in YVL 2.8. The guide gives a list of topics, which one shall be able to trace from assumptions of the PSA to the final results. In addition to power operation, low power and shut down states and the transfers between them shall be considered in the PSA. Events such as internal failures, disturbances and faults, loss of off-site power, fires, floods, harsh weather conditions, seismic events and other external and human caused initiators shall be included as initiating events. However, YVL 2.8 does not cover intentional damaging of a plant.

The level 1 PSA shall identify the accident sequences leading to core damage and to determine their probabilities. The level 2 PSA shall determine the amount, probability and timing of radioactive substances released out from the containment in consequence of core damage. Besides leaks and ruptures of the containment, also the bypass sequences and controlled release of radioactive materials shall be assessed. YVL 2.8 gives a list of issues that level 2 shall include, e.g. interface with level 1, containment event trees, systems reliability analysis. Source terms from the reactor, transportation, retention and respective probabilities have to be analysed. Appropriateness and efficiency of the strategy of accident management and the balance between systems have to be assessed. YVL 2.8 also gives a list of severe accident issues to be analysed, e.g. reaction forces, hydrogen issues and recriticality. However, the list gives only examples, and all the plant specific issues have to be mapped out and analysed as realistically as possible.

3.2 Scope at the beginning of the life cycle

Voluntary use of PSA in safety management is an integrated part of the developed safety culture, and the fulfilment of the authority requirements is only one part of it. A simplified PSA in the feasibility study of a plant concept can be based on the analysis of an existing plant corrected with the main modifications in the design. Then, it is important to foresee that there are no fundamental problems in the design.

In the bidding phase it is important to show that the plant can meet the regulatory targets. A quite comprehensive level 1 and level 2 PSA is necessary, but it can be based on the analysis of existing plants of similar design or delivered earlier by the same vendor.

STUK reviews the preliminary level 1 and level 2 PSA for the first time in connection with the application for the construction license i.e. design phase PSA. STUK may require supplementing the analysis before approving it for the construction license. In any case the PSA needs to be complemented during the construction phase, and during this phase it will be used for several applications.

The result of the work during the construction phase is a complete level 1 and level 2 PSA for the operation phase. An approval of this PSA is a condition for the operating license. STUK reviews the PSA, i.e. construction phase PSA, the second time in connection with the application for an operating license.
4 REQUIREMENTS ON THE APPLICATION OF PSA

4.1 Authority requirements in the PSA specific regulatory guide YVL 2.8

In the guide YVL 2.8 there are several requirements of applications and references to corresponding guides. In YVL 2.8 following applications are required:

In design phase:

• Safety classification shall be assessed by PSA. The assessment shall be used to demonstrate that the requirements for quality management system concerning the safety classification of each component are adequate compared with the risk importance of the component. The probabilistic review of the safety classification shall be submitted to STUK in conjunction with the safety classification document.

In construction phase:

• The purpose of the level 1 and 2 construction phase PSAs is to ensure the conclusions made in the design phase PSA on the plant safety and to set a basis for risk informed safety management during the operation phase of the plant. The level 1 and 2 PSAs shall be based on the plant specifications submitted in conjunction with the application for an operating license.

• The application for an operating license shall demonstrate that the plant meets the numerical design objectives set forth in section 2.1 of this Guide. Should substantial risk factors not recognised earlier appear before the commissioning of the plant, the applicant for a licence shall upgrade the safety of the plant. In conjunction with the design of safety upgrades the applicant for a licence shall demonstrate that the safety of the plant assessed after the upgrades is substantially at the same level or better than the objectives presupposed for the design phase.

• The technical specifications shall be reviewed by PSA in such a way that the coverage and balance of technical specifications are ensured. The review must cover all operating states of the plant.

• The results of PSA shall be applied in the review of safety classification as in the design phase if extensive changes are performed in the plant design in the construction phase.

• The results of PSA shall be applied in the working up of programs of safety significant systems testing and preventive maintenance during operation, and in the working up of disturbance and emergency operating procedures.

• The results of PSA shall be used in the drawing up and development of the inspection programs of piping as per Guide YVL 3.8. While drawing up the risk informed inspection program, the systems of classes 1, 2, 3, 4 and EYT (not safety related) must be regarded as a whole.

• The results of PSA shall be taken into account in the planning of personnel training.

4.2 Authority requirements on PSA applications in other YVL guides

STUK assumes in several other guides application of risk informed methods and PSA. E.g. the top-level guide YVL 1.0 “Safety criteria for design of nuclear power plants” states in connection of human errors that “According to section 19 of the Council of State Decision (395/91), special attention shall be paid to the avoidance, detection and repair of human
The possibility of human errors shall be taken into account both in the design of the nuclear power plant and in the planning of its operation so that the plant withstands well errors and deviations from planned operational actions. In that guide, a requirement for PSA application is stated as follows: “In failure analyses required in Guide YVL 2.7, human error shall be considered and it shall be demonstrated that individual errors do not prevent safety functions. The possibility of multiple human error shall be assessed in the plant probability safety assessment (PSA) and the necessary measures to avoid or reliably detect errors shall be planned”.

The guide YVL 2.0 “Systems design for nuclear power plants” states that: “Systems design shall employ both deterministic and PSA-based methods.” It continues: “With probabilistic safety assessment (PSA) the reliability of various safety functions and the balance of design between them is evaluated. A plant shall be so designed that calculated risks are distributed such that no individual component, system, phenomenon or other factor is risk-dominant and that the share of hard-to-manage risks is as low as possible. A plant designed in such a way has a well balanced design. High reliability of operation is required of all systems and of safety systems in particular. This is why system operation in various failure situations shall be assured. This is accomplished by applying the redundancy, diversity and separation principles (Section 18 of Government Resolution No. 395/1991).”

The requirement in YVL 2.0 assumes a transparent computer code and a PSA model in order to be able to check the design balance: “It shall be demonstrated by PSA methods that a plant’s design is well-balanced in terms of reliability, as per subsection 2.1. It shall specifically be demonstrated that a well-balanced design has been reached between
- various safety functions
- different systems carrying out the same function
- main systems and support systems
- subsystems of the same system

In addition, it shall be ensured that risks (in terms of both core melt and/or environmental release frequency and severity) are distributed between various initiating events in such a way that no individual event sequence, system, subsystem, structure or component causes a major contribution to overall risk.”

Making comparisons between design alternatives requires using a fast and flexible computer code with enhanced post processing capabilities. The FinPSA code used by TVO fulfills this requirement.

PSA support for system evaluation is required during the PSAR phase: “a description of a system’s importance in the accomplishment of a safety function proper if the system supports a system performing a safety function and the reliability target of the safety function in whose implementation the system contributes” Accordingly, in the FSAR phase (for safety class 1 - 4 systems): “a probabilistic assessment of a system’s significance for plant safety using importance measures (see Guide YVL 2.8)”. YVL 2.0 continues: “The system’s analysis demonstrates the fulfillment of its design bases and requirements. Essential analyses to be included in a safety analysis report or topical reports include among others an analysis of the system’s physical operation, a single-failure analysis, a Failure Mode and Effect Analysis (FMEA), and importance measures. The mutual order of importance of the various analysis types varies according to the field of technology”.

The guide YVL 2.6 “Seismic events and nuclear power plants” requires seismic PSA in the design phase in the demonstration of earthquake resistance: “The results of a design-phase PSA shall also demonstrate that the implementation of seismic design is acceptable from the viewpoint of the nuclear power plant’s overall safety. It continues with more detailed requirements “The most important initiating events, due to earthquake-induced failures and
malfunctions, shall be incorporated in the design phase PSA. When choosing the initiating events, the following factors shall be considered: S2\(^1\) category structures and components plus their supports as well as experiences of the susceptibility to failure of different types of structures and components in actual earthquakes of varying magnitudes. The possibility of failure chains attributable to the simultaneous dynamic loading of large component entities and of common cause failures shall be analysed."

The guide YVL 2.7 “Ensuring a nuclear power plant’s safety functions in provision for failures” also requires, in connection of general design principles, that: "Both deterministic and probabilistic design principles shall be employed in the design of safety systems. When setting reliability requirements for the safety functions the likelihood of occurrence of the initiating event and the severity of its consequences shall be considered."

The guide YVL 3.0 “Pressure equipment of nuclear facilities” states: “Risk-informed methods may be used in choosing the components to be inspected. The procedure is described in Guide YVL 3.8.”

The guide YVL 3.8 “Nuclear power plant pressure equipment - In-service inspection with non-destructive testing methods” requires using risk informed methods: “In the drawing up of inspection programmes for Safety Class 1, 2, 3 and 4 piping and Class EYT (non-nuclear) piping as well as in the development of inspection programmes for operating plants, risk-informed methods shall be utilised to ascertain the inclusion in the inspection scope of those components posing the highest risk.”

In guides YVL 3.2 “Nuclear facility pressure vessels” and 3.5 “Ensuring the firmness of pressure vessels of a NPP” there is a requirement of estimating the brittle fracture probability.

Fire PSA is required by the guide YVL 4.3 “Fire protection at nuclear facilities”: “Together with the initiating events analysed in the design phase PSA the fires shall be assessed in order to evaluate the fire protection arrangements and to identify the risks caused by fires.” Especially, the following analyses are addressed: “Fire hazards analyses shall always be performed for the containment and the control room. By means of the containment fire hazards analysis it shall be demonstrated that the safety functions of the plant can be reliably accomplished during and after any potential fire accident in the containment: the reactor can be shut down and maintained subcritical, the plant can be cooled down to cold shutdown condition and the residual heat can be removed. By means of the fire hazard analysis of the control room it shall be demonstrated that the control of the necessary safety functions can be accomplished in the event of a fire in the control room or in any other fire compartment”. Also, the YVL 4.3 addresses the need of keeping PSA up-to date as follows: “Maintaining and continuously advancing the fire safety of nuclear power plants is a part of the safety culture conducted in the operation of the nuclear power plants. Maintaining and advancing the fire safety includes updating the PSA, described in Guide YVL 2.8. Also fire hazards analyses and other documents shall be updated if the conditions of the power plant change or modifications of the plant fire protection arrangements are performed. New research results in the fire field, the general progress in the field, accumulated knowledge of the fire events as well as the ageing effects of the components and materials shall be taken into account in the fire hazards analyses and in the operation and inspection activities of the power plant”.

\(^1\) Seismic classification: the structures and components of Nuclear Power Plants are to be classified into the seismic categories S1 and S2 according to their required resistance to earthquakes. Seismic category S1 comprises structures and components whose failure could cause an accident situation directly or indirectly leading to a reactor core melt. Seismic category S2 comprises all other structures and components and no earthquake resistance requirements relating to their own operation and integrity are set but their failure must not compromise structures and components in seismic category S1.
The guide YVL 5.2 “Electrical power systems and components at nuclear facilities” requires use of PSA for several purposes, e.g.: “in the drawing up of testing programmes and preventive maintenance programmes” and “to assess alternative solutions”. Also, the guide contains requirements for safety analyses: “Demonstration of the fulfilment of functional and performance requirements by analyses is part of the qualification of electrical power systems and components. Safety Class 2 and 3 electrical power systems shall be subjected to a failure mode and effects analysis, a common cause failure analysis, an operating experience analysis, a selectivity analysis to demonstrate the fulfillment of the selectivity requirements for electrical protection, a safety analysis to demonstrate the fulfillment of safety requirements. In addition, Safety Class 2 electrical power systems shall be subjected to a quantitative reliability analysis and Safety Class 3 electrical power systems to a quantitative reliability analysis according to their safety significance”. In addition, specific requirements are given for computer-based systems and components in the YVL 5.2.

Similarly, the guide YVL 5.5 “Instrumentation systems and components at nuclear facilities” sets several requirements for use of PSA e.g. Safety Class 2 systems shall be subjected to a failure mode and effects analysis, a common cause failure analysis, an operating experience analysis, and a quantitative reliability analysis. Safety Class 3 I&C systems shall be subjected to a failure mode and effects analysis, a common cause failure analysis and an operating experience analyses, and, depending on the safety significance, a quantitative reliability analysis. Plant specific PSA shall be updated to correspond to the modified system”. The YVL 5.5 contains specific requirements in case of using programmable I&C systems.

5 CONCLUSION

The Finnish Regulatory Body STUK has demanding and detailed requirements on PSA and risk informed methods in the design, construction and operation of nuclear power plants. Besides the dedicated regulatory guide YVL 2.8, there are many requirements for risk informed applications in various regulatory guides. Successful fulfilment of the makes it important to begin the PSA programme very early in the design phase. STUK reviews the PSA and first applications in connection with the construction license. An Operating license requires a complete level 1 and level 2 PSA ready for living use. STUK has granted the construction license, and the complementation of the PSA for OL3 is ongoing.
ABSTRACT

This paper outlines those key aspects of the processes of design and assessment of generic designs of Nuclear Power Plant (NPP), which aim to provide assurance of safety. Indications are given of the approach and criteria likely to be required for such process. It is emphasized that design aims for defence in depth, while assessment provides assurance that the defence in depth is adequate for the safety goals of the design. In practice design and assessment would overlap, with the findings of assessment being fed back into design re-iteration. Both deterministic and probabilistic approaches are necessary inputs into assessment. The paper notes the complementary aspects of deterministic and probabilistic assessment approaches. It outlines the key requirements for safe design, and the deterministic evaluation of a design. It then focuses on the requirements for probabilistic risk estimates and criteria. Recommendations are made for the approach to be adopted for generic designs, using deterministic and probabilistic criteria within some form of integrated risk-informed decision making process as aids to judgment.

1 INTRODUCTION

1.1 Background

The International Atomic Energy Agency (IAEA) has recognized a need to provide guidance for NPP vendors, operators and regulators on the information to be supplied for the evaluation of the safety of generic designs. The over-arching design safety goals are to protect people and the environment from the harmful effects of ionizing radiation. This can only be achieved through a systematic process of compliance with a number of fundamental safety principles. This process includes the deterministic safety design aspects and the system features (safety margin, defence in depth, redundancy, diversity, automation, etc.) which aim for safety by design, and the associated specified operational practices. It also includes the deterministic and probabilistic analyses to provide assurance that a high degree of safety will be achieved in practice.

The process of generic design and evaluation is different in context and detail from a specific installation proposal. For example, generic design would be based on an assumption of a ‘typical’ site, or a specification of site characteristics, and then it would need to be reviewed against the characteristics of a particular site. Considerable experience has been
accumulated in some Member States (MSs) where generic designs are certified (e.g. USA). However, much of this is with evolutionary designs of Light Water Reactors (LWRs) based on experience of earlier versions. Consideration is now needed of the possibility of new concepts, passive safety features and other technical developments, together with the developing emphasis on integrated Risk-Informed Decision Making (RIDM) and the use of Probabilistic Safety Assessment (PSA) to complement the deterministic approach.

In past generic design certifications in individual States, there had been considerable interaction between the vendors, operator organizations and the State’s regulatory body in the evolution of particular approaches to design and its evaluation. Reference [1] notes that some MSs have detailed deterministic regulatory approaches; while others prefer goal-setting with flexibility on how to achieve the goals.

1.2. Structure of the Paper and IAEA’s Safety Standards

This paper adopts a top-down structure, reflecting the IAEA Safety Standards (SSs), which begin with Safety Fundamentals (SFs) [2] providing basic concepts and principles of nuclear and radiation safety. These are supported by a number of Safety Requirements (SRs) publications providing high-level requirements addressing different aspects of safety, e.g. nuclear power plant safety. These in turn are supported by more detailed guidance in the category of Safety Guides (SGs) providing recommendations on the actions needed to comply with SRs. The paper also follows the assurance process presented in SRs for Safety Assessment [3], which begins with the setting of safety and other design objectives. Design proceeds according to good engineering standards and practice, with emphasis on defence in depth. The design is evaluated by deterministic analysis and further reviewed by probabilistic analysis. The results of analyses and comparisons with objectives may lead to re-iteration of aspects of the design; and analyses may be carried out in parallel with the detailed design development.

In this paper, the term ‘safety assurance’ is used to cover the information required to underpin the safety-related aspects of design plus the safety assessment process. ‘Safety analysis’ covers the deterministic and probabilistic analyses of a design to indicate the degree to which safety objectives have been achieved. Safety analysis is a major input to ‘safety assessment.’

2 SAFETY OBJECTIVES

In 2006, the IAEA published its revised Safety Fundamentals, SF-1 [2]. This states that: “The fundamental safety objective is to protect people and the environment from harmful effects of ionizing radiation”. This is to be achieved by adherence to ten fundamental principles, of which numbers 5, 6 and 8 pay particular attention to safety assessment.

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1 The IAEA’s Safety Standards are not legally binding on Member States, but may be adopted by them, at their own discretion, for use in national regulations in respect of their own activities. The IAEA’s Safety Standards are binding on the IAEA for application in relation to its own operations and on States in relation to operations assisted by the IAEA.

2 “Safety assessment involves the systematic analysis of normal operation and its effects, of the ways in which failure might occur and of the consequences of those failures. Safety assessments cover the safety measures necessary to control the hazard, and the design and engineered safety features are assessed to demonstrate that they fulfil the safety functions required of them…” [2].
Principle 5 of SF-1 states that “Protection must be optimized to provide the highest level of safety that can reasonably be achieved”. To show adherence to this principle, risks must be assessed. A graded approach to assessment should be used, with the effort put into such assessment being commensurate with the scale of the hazard. For NPP, with the main hazards being the very large radioactive isotope inventory of the core and the spent fuel in store, a very high effort on assessment is expected.

Principle 6 of SF-1 [2] states that “Measures for controlling radiation risks must ensure that no individual bears an unacceptable risk of harm”. Thus “…doses and radiation risks must be controlled within specified limits…” [2]. Quantitative limits for doses are indicated in the Basic Safety Standards [4]. Criteria for risks used in different States are discussed in Ref. [5]. Often, Member State regulations and/or guidance would specify the national implementation of IAEA standards for dose and risk limits. An NPP operator (licensee), or the vendor of a generic NPP design, would be expected to demonstrate that an installation would give doses and risk below such ‘specified limits’.

SF-1 also states that the specified limits “…represent a legal upper bound of acceptability….they therefore have to be supplemented by the optimization of protection…” [2]. This is in line with a trend towards a three-zone approach for risk consideration, with criteria defining the boundaries between zones (for details see INSAG-6 [6]). The following is the framework suggested in INSAG-6:

“… decisions concerning the tolerability of risk should be based on three principles:

– There exist levels of risk from technology to individuals or society that should not be tolerated irrespective of the technology's benefits. Such levels are often referred to as tolerability limits.

– At risks lower than that level, safety cannot be absolute, and the knowledge of how to improve it is never complete. Responsible action includes continued striving for risk reduction, provided that the effort to achieve these improvements is not unreasonably high.

– Well below the tolerability limit, risks are so low that they should be regarded as negligible in order to avoid unnecessary deployment of resources which diverts attention from substantial safety issues which could lead to larger risks of other types. That corresponding low level is sometimes called a ‘de minimis’ limit.”

The three-zone approach is illustrated in Fig. 1. The boundary between the upper zone and middle zone corresponds to a reference level for action (called ‘tolerability limit’ by INSAG-6). If a risk metric lies above this level, every effort must be made to improve safety. Some MSs indicate criteria for this level. In contrast to dose limits, the reference level for action for risk based on PSA results is not usually applied by MSs as a formal legal limit.

![Figure 1: Three-Zone Approach to Safety Consideration](image-url)
In the lowest zone the risk is broadly acceptable and the emphasis is on ensuring that the risk has been properly assessed and on maintaining the safety situation. In the middle zone the risk should be optimized (i.e. correlate with SF-1 [2] Principle 5) i.e. as low as reasonably achievable (ALARA). There may also be criteria for the boundary between the lowest zone and the middle zone; such lower criteria may be called ‘objectives’.

SF-1 Principle 8 [2], dealing specifically with accident risks, states that “All practical efforts must be made to prevent and mitigate nuclear or radiation accidents”. The text indicates the basic means to achieve this objective, with particular emphasis on defence in depth, fitness for purpose, and the general principles of good engineering design. It is also usually expected that a designer will begin by attempting to eliminate or minimize risks by using non-hazardous materials and conditions wherever possible. Safety assessment would use deterministic and probabilistic analyses to demonstrate compliance with this Principle. The general interconnection between Deterministic Safety Assessment (DSA), PSA, and Safety Objectives is illustrated in Fig. 2.

![Figure 2: Interconnection between DSA, PSA and Safety Objectives](image-url)

**DETERMINISTIC Safety Assessment (DSA)**
- Defence-in-depth
- Safety margins
- Diversity and redundancy within and between safety systems
- Single failure criterion

*Success-oriented approach – if deterministic requirements are met, the associated risk is believed to be negligible*

**PROBABILISTIC Safety Assessment (PSA)**
- Comprehensive analysis of potential accident scenarios without restrictions on the number of potential component failures and human errors

*Failure-oriented approach – answers to the questions:

- What can go wrong?
- How likely is it?
- What are the consequences?*

**REQUIREMENTS TO PSA QUALITY:**
- FULL SCOPE PSA +
- STATE-OF-THE ART IN METHODOLOGY +
- INDEPENDENT REVIEW

**Comparison of the assessed risk with SAFETY OBJECTIVES**

**MAINTAIN SAFETY LEVEL**

**IMPLEMENT MEASURES TO FURTHER ENHANCE THE ACHIEVED SAFETY LEVEL**

**NON-compliance identified**
It is noted in Figure 2 above that DSA is ‘success-oriented’ while PSA is ‘failure-oriented’. The use of both approaches therefore provides a diversity of perspective within the assessment, and enriches the inputs into decisions on the adequacy of safety. This is recognized in the development of integrated risk-informed decision-making that is discussed further.

3 DESIGN AND DETERMINISTIC ASSESSMENT

The basic safety aims of the design and construction are: to limit routine exposures and releases, to prevent accidents, and to limit and mitigate accident consequences. The key aspects to achieve these aims are:

- prevent failures or abnormal conditions (including breaches of security) that could lead to loss of control;
- the primary means to do these is Defence in Depth (D.i.D.), i.e. “the combination of a number of consecutive and independent levels of protection that would have to fail before harmful effects could be caused…” [2];
- D.i.D. should ensure that no single failure (technical, human or organisational) could lead to harmful effects; and combinations of failures that could do so are very low probability;
- independent effectiveness of D.i.D. levels is necessary;
- D.i.D. combines: management of safety, site selection, design and engineering features providing safety margins, diversity and redundancy of safety systems;
- design and engineering should feature: high quality and reliability (of design, technology and materials); passive and active control, limiting and protection systems, and surveillance; inherent and engineered safety features; operational and accident management;
- structures, systems and components (SSCs), as well as human operators, should have adequate reliability. This is achieved by classifying SSCs according to their safety significance, and implementing them with commensurate quality;
- technology should be already proven in operation or qualified for the proposed application, applying conservative acceptance criteria with margins of safety;
- design should be adequate to withstand initiating events with widespread effects on multiple components.

The proposed IAEA Draft Requirements for the Safety Assessment of Nuclear Facilities and Activities [3] includes the following points which are particularly relevant for generic design evaluation:

- use of proven SSCs where practicable, plus research and development (R&D) to show adequacy of any novel aspects, and equipment qualification;
- clear identification of design principles (as stated above);
- show relevance of international codes and standards, taking account of SSC safety classification;
- cover all relevant internal and external hazards;
- indication of design to take account of the eventual need for safe decommissioning.

General principles of good design should be applied, including:

- elimination or reduction of hazard where possible by choosing inherently safe materials and conditions;
- preference for passive over active systems or their effective combination;
- diversity and redundancy within and between safety systems;
– no single component failure can disable a safety function;
– systems which are fail-safe and whose deterioration is visible;
– explicit consideration of human factors, with automation where beneficial, good ergonomics, and robust procedures.

An NPP design would need to make transparent the design philosophy and its detailed application, to ensure that regulatory bodies and operating organizations understand the reasons for design decisions. This aids the evaluation of the design, and it also reduces the risk of inappropriate later modifications which inadvertently weaken a system. The design would also need to specify various aspects to be addressed in the deterministic and probabilistic assessments such as the schedules, site layout, reliability of off-site services, plant condition monitoring, operator staffing levels and competences, etc.

Starting with an initial design concept and performance (including safety) goals, design proceeds from the definition of the design bases for SSCs, derived by consideration of the possible initiating events which might place demands on them. In principle this requires the analysis of all routine and potential abnormal states or events affecting the plant. Usually some selectivity is applied to exclude plant states or events which are highly improbable, and the remainders are grouped within bounding envelopes to facilitate conservative analysis while not requiring excessive detail. The rationale of selectivity and grouping is an important part of a safety case. Assessment continues with engineering analysis of the SSCs subject to the design basis parameters to demonstrate withstand capability with safety margins as required by appropriate standards.

The deterministic design analysis and assessment aspects form the larger part of a safety case, and there is much detailed guidance in IAEA and international technical standards, and MS’ own regulatory standards. Very clear documentation of the standards basis would be required in a vendor’s safety case, to provide a route-map for reviewers from different backgrounds. A vendor would itself need to arrange an independent review of the design and the safety case, and to provide evidence of this. In due course, the vendor would also need to ensure that any operating organization has clear information and understanding of the required Technical Specifications, Operating Limits and Conditions, staffing levels and competences, and component specifications as assumed in the generic safety case.

4 PROBABILISTIC SAFETY ASSESSMENT

Historically, the design and operation of NPPs were based on deterministic concepts. The main elements were defence-in-depth provisions, safety margins, diversity and redundancy, and single failure criterion. The implications were that if deterministic criteria are met, the plant would be safe enough, and the risk of unacceptable radiological releases would be sufficiently low. However, the major accidents at NPPs showed that this was not always the case. The PSA technology that started in 1975 with the famous study, WASH-1400 [7], provided the possibility to get additional and new safety-related insights and to unambiguously assess the risk dealing with a particular NPP. To complement the deterministic approach, PSA presented a failure-oriented approach that was aimed at finding answers to the questions:
– What can go wrong?
– How likely is it?
– What are the consequences?

As with deterministic assessment, there is a large body of standards and guidance on the appropriate technical features of PSA. IAEA has recently completed the final draft Safety
Guides on Level 1 and Level 2 PSA (see Refs [8] and [9]). Details of the contents of the safety guides are discussed in Ref. [10]. In addition, TECDOC-1511 [11] details the attributes required in PSAs for particular applications. TECDOC-1511 focuses on Level 1 PSA for NPP at power and internal initiating events only; some MSs would expect more comprehensive coverage. TECDOC-1436 [1] notes that the emerging standard is for at least Level 2 PSA for new NPP. It is not proposed to discuss here how PSA should be done, but rather ‘what should be done’ at an elevated level of detail.

While PSA is valuable to provide a quantitative indication of risk for overall safety evaluation, it has other uses as well. Such uses derive from the holistic nature of PSA, including the assessment of risk from events beyond the design basis, and the understanding of the interactions within complex systems. For example, PSA may be used to inform accident management planning. PSA is also a valuable tool for assessing the effects of options and the sensitivity of the risk to assumptions on SSC performance. For overall risk assessments using PSA, it is necessary to ensure that all potential initiating events and fault sequences are covered.

It is generally accepted that PSA should be best-estimate, rather than the conservative approach often applied in deterministic assessment. This should be borne in mind when developing or using criteria. Some regulators expect an indication of uncertainties, as well as basing judgements on the 50% confidence results. Others may indicate criteria based on a higher confidence level, such as 95%. INSAG-6 [6] recommended the use of 95 percentiles when testing compliance with broad technical safety objectives such as Level 1 PSA Core Damage Frequency (CDF).

An important need with PSA, as with DSA, is to ensure full and transparent recording of the assumptions and judgments. These would be particularly necessary where a generic NPP design is to be sold to several different operating organizations regulated by different MS licensing bodies that might have different safety objectives stated in their regulations. It would also be necessary to ensure that the regulatory body and operating organization have access to the PSA code and model or arrangements to use them, for reviewing the PSA, for the assessment of modifications during operation, and for periodic safety reviews. It is very important that the PSA used in the licensing process should necessarily undergo a thorough independent review to assure its compliance with the state-of-the-art in methodology and the absence of inadvertent mistakes, omissions, and inconsistencies.

There is international consensus that PSA can provide an in-depth understanding of the level of safety achieved in design, and it should be used as a complement to deterministic safety assessment and as a tool for analysis of compliance with safety objectives. The general iterative process of assurance of the compliance with safety objectives outlined in Fig. 2 could be applicable in the licensing process. It is now recommended that a full-scope Level 1 and Level 2 PSA should be performed. Level-3 PSA is also desirable. Provisions for a ‘Living PSA’ are desirable as well.

5 RISK INFORMED DECISION MAKING

5.1 RIDM Considerations in IAEA Publications

TECDOC-1436 [1] discusses the achievements in RIDM and risk informed regulation (RIR), mostly in relation to licensing decisions by a regulatory body. RIDM could also be used by a licensee in making a safety case, or an NPP vendor in assessing a proposed design.

The basic aim of RIDM is a ‘balanced’ decision. This is a decision which takes into account all the available information to an appropriate extent. RIDM combines in a formal process the results from deterministic and probabilistic analyses and other assessment
requirements, plus legal, regulatory, cost-benefit, and any other requirements. The degree to which a requirement is met may be combined with a weighting for that requirement, to give an overall evaluation.

In RIDM, explicit consideration is given to the likelihoods and consequences of events, as well as good engineering practice, deterministic safety margins, and the arrangements for the management of safety. It is recognized that the maturity of PSA allows the explicit use of quantitative risk information, but quantitative PSA results compared with risk criteria are not the sole determinants of design adequacy. For example, PSA alone should not be used to justify a new design which apparently reduces the safety protection compared to generally-accepted deterministic good practice.

RIDM recognizes the strengths of deterministic assessment, for example that it is tried and tested, and very well-developed. However, deterministic analysis alone may have weaknesses, such as focusing on issues which are not the major or only contributors to risk, and being unable to prioritize potential improvements in terms of risk reduction. It may also be difficult to demonstrate deterministic safety margins with a high degree of confidence for radically new designs. Deterministic consideration of possible ‘reasonably achievable’ safety improvements can only be done qualitatively.

The inclusion of PSA in RIDM recognizes the benefits which PSA should provide in addition to quantitative risk information, such as: comprehensive coverage of initiating events; insights into the importance of events and safety measures; explicit indication of uncertainties and sensitivity analyses of options; degree to which defence in depth, single-failure criterion etc. are met; prioritization of improvements; insights into the safety aspects of novel designs. There are potential difficulties in PSA performance and use, such as: completeness of identification of initiating events; large uncertainties of numerical risk results; data relevance; modelling issues (for example, human errors of commission); limitations (for example, PSA of a restricted scope, e.g. Level 1 PSA, does not address severe accident analysis). It may also be difficult to cross-compare the results of PSAs done with different codes; but the use of the same PSA code to compare options is much less problematic.

TECDOC-1436 [1] states or implies several points about RIDM, for example:

- Avoid compensating for a weakness in one aspect by claiming a strength in another; the aim is to consider all aspects separately and strengthen them as far as reasonably achievable, and then make a balanced judgement about adequacy overall.
- Process requires staff resources with the expertise for the various components, plus the ability to take an overview. (It is recommended in the TECDOC that the final decision is taken by an expert interdisciplinary panel).
- RIDM aids transparency in the decision-making process; and it should aid the retention of corporate knowledge about the reasons for design features.
- Vendors of generic designs may find that the weighting of various aspects (such as deterministic, probabilistic), and key criteria, may differ between MS. Transparency in RIDM data and assumed weightings should facilitate the tailoring of safety cases to fit MS regulatory requirements.

Having recognized the growing incentive for RIDM, in 2007 the IAEA launched a project on developing a guidance document on RIDM [12]. It will provide principles and suggest approaches to integrate the results of deterministic and probabilistic safety analyses, as well as other important aspects to make sound, optimum, and safe decisions. The primary focus of the publication is to provide guidance to Member States on how to adequately and responsibly perform, document, and report the results of RIDM. An advanced draft document is currently available [12].
5.2. Discussion: Goals for PSA

The three-zone approach outlined above may be taken as a model. Where the PSA risk estimates at 50% confidence level are above a reference level for action, very high priority should be given to reducing the risk by physical and operational measures. Where the risk estimate is below this level, effort should still be made to reduce the risk to as low as reasonably achievable. Where risks are estimated to be below a ‘de minimis limit’ with a high degree of confidence (such as 95%), the emphasis should be on maintaining that situation. In all cases, there should continue to be periodic and interim reviews to seek reasonably achievable improvements, taking account of operating experience and new knowledge.

With Level 1 PSA, a reference level for action for CDF of 1E-5/ry may be used for new NPP designs, with a lower risk to be aimed for where reasonably achievable (for existing NPPs, the tolerability level of 1E-4/ry recommended in INSAG-12 [13] may be used). If it is desired to have a ‘de minimis level’, fractions (for instance, 10% or 1%) of the limit figures could be used. Care is needed when defining ‘core damage’ for designs other than LWR. This would be for MSs, licensees and vendors to propose, by analogy with the significance of a level of core damage in LWRs.

For Level 2 PSA, for Large Early Release Frequency (LERF), a reference level for action set at a factor of 10 lower than the CDF figure is often used, giving 1E-6/ry for new NPP designs. A lower risk should be aimed for where reasonably achievable. If it is desired to have ‘de minimis limits’, fractions (for instance, 10% or 1%) of the reference level for action figure could be used. For new designs, INSAG-12 [13] now expects the “practical elimination” of accident sequences which could lead to large early release. However, Level 2 PSA would still seem to be necessary, to show completeness in the defences and to show that the overall risk is acceptably low.

A definition is needed for ‘Large Early Release’. INSAG-12 [13] says “large off-site releases requiring short-term off-site response”. Some MSs use a definition in terms of quantity of a dominant isotope (e.g. Cs 137); others use a fraction of core inventory; others may use a criterion based on breach of containment. It is suggested that further consideration be given to this issue.

Care is needed when comparing PSA information with quantitative criteria. The following points summarise some aspects of this. (These points are derived from a recent IAEA Questionnaire on the use of PSA criteria, see Ref. [5]).

(a) Words used for criteria include ‘goals’, ‘objectives’, ‘targets’, ‘limits’, ‘acceptable’, ‘standards’, etc. It may be unclear whether these terms mean: ‘a level of risk which should not be exceeded’ (i.e risk tolerability limit discussed above); or ‘a level below which the risk is broadly acceptable’ (i.e. ‘de minimis limit’ discussed above); or some other definition. In line with the IAEA Glossary, the words ‘reference level for action’ should be used for the first meaning, and ‘risk objective’ for the second, following the principles provided above. This acknowledges that the criteria may not be intended to be used as stand-alone formal or legal criteria, and so should not be called ‘limits’.

(b) It should be clear whether an explicit assessment of options to test for reasonably achievable improvements is required where the risk falls between a ‘reference level for action’ and a ‘risk objective’. For new generic designs, it is suggested that such an assessment should be presented as part of a safety case package.

(c) The PSA, and the criteria, should cover all plant states during the operational lifetime. It is suggested that the main metric (CDF or LERF, units ry-1) should be the integral over time of the overall risk from all plant states, assuming the intended operating cycles. PSA should include all initiating events and hazards. The number of units on
the site could be taken into account while defining the ‘risk objective’ and ‘reference level for action’; this issue may become important when several units are to be located at the same site. Subsidiary metrics may be proposed for commissioning and for particular plant states, including the derivation of Technical Specifications. Where PSA is used as an aid to operational decision-making (for example a Risk Monitor), subsidiary criteria may be needed for the action to be taken if the point-in-time risk level rises.

(d) PSA involves many assumptions or specifications relating to the Technical Specifications, quality of construction, operation, maintenance, inspection, testing etc. A generic design would need to make these assumptions explicit so that operating organisations and regulatory bodies can ensure adherence to them. A comprehensive independent peer and/or regulatory review is a standard practice that should be followed. The technical quality of PSA should be in accordance with the state-of-the-art that is reflected in safety standards developed by IAEA and other standards internationally, e.g. ASME PRA Standard [14].

(e) Arrangements are needed for the operating organisation to use, or contract for the use, of the PSA for the assessment of modifications and periodic reviews throughout the NPP lifetime. Also, a ‘Living PSA’ programme is encouraged.

(f) There may be a requirement for Level 3 PSA in a particular MS; arrangements would then be needed to associate Level 1 and 2 PSA with a Level 3 PSA that is site-specific.

6 CONCLUSIONS

The main design safety goals are to protect people and the environment from the harmful effects of ionizing radiation. A systematic iterative process of compliance with a number of fundamental safety principles at the design stage based on deterministic analysis and complemented by PSA (as outlined in Fig. 2) is likely to result in a design that meets the type of numerical criteria suggested above. Such criteria should not be used as stand-alone decision determinants. They should be used within some form of integrated risk-informed decision making process, as aids to judgment. The PSA used in decision making should be of appropriate technical quality. The new IAEA safety guides on PSA [8, 9] and document on RIDM [12] are specifically designed to provide advanced guidance in these matters.

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Licensing and Harmonisation
Regulatory Challenges Posed by the “Best-estimate Plus Uncertainty” Methodologies

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ABSTRACT

Realistic methodologies are gradually replacing the simpler and overconservative traditional ones in the realm of the deterministic safety analysis (DSA) of nuclear power plants. Such new approaches make use of realistic rather than pessimistic models and hypotheses, and must include uncertainty assessments of the important results, the term “best-estimate plus uncertainty” (BEPU) methodologies stemming from that fact. The safety acceptance criteria are correspondingly moving from a deterministic to a probabilistic shape. As a result, probabilistic definitions are currently being proposed for the safety margins, and statistical methodologies are now an essential part in the DSA toolkit. In this paper the main ideas about the DSA methodologies, and the differences between the conservative and realistic approaches, are described. The most widely used BEPU methodologies are based on Monte Carlo calculations performed with the predictive models, and feature order statistics method of uncertainty assessment (also known as Wilks method). In the present paper, such methodologies are described and discussed from a regulatory point of view, and insights obtained from the evaluation work conducted by the authors for Spain’s regulatory authority are presented.

1 INTRODUCTION

Deterministic Safety Analysis (DSA) is one of the frameworks used in the design and safety evaluation of nuclear power plants (NPP). The risk posed by a NPP is described by the accidents which can occur therein, each one typified by its frequency and consequences. The DSA basically consists in assessing the damage level which is exceeded with a given frequency, and comparing it with a limit or acceptable value, whereas the Probabilistic Safety Analysis (PSA) has the complementary task of calculating the frequency of exceedance of a given damage level and comparing it with a limit or acceptable value.

The basis of the DSA consists in assessing the consequences of a group of selected transients or accidents, named Design Basis Transients (DBT) and defined as enveloping or bounding transients in the sense that their consequences (damage) are worse than those of the great majority of transients deriving from the same initiating event.

The consequences of the DBTs must be calculated using predictive models, and the results are compared to the safety limits. Nevertheless, as it is well known, the calculation processes introduce uncertainty in the results, for two main reasons:
The inputs to the calculation (for instance, the conditions existing in the plant at the beginning and during the transient) are imperfectly known.

The predictive models used in the calculation are not perfect, because they are simplified views of the physical reality.

The conclusion is that the calculated consequences of a DBT have uncertainty, and this fact must be taken into account when checking the fulfilment of the safety criteria. In the safety sciences, a principle exists to deal with uncertainty: the risk must be assessed in a conservative (or pessimistic, or bounding…) fashion. In the DSA this means that the calculated consequences must be worse than the real ones with high confidence.

In this paper we will describe the formalism of DSA and distinguish between the different types of deterministic methodologies according to the type of predictive models and calculation hypotheses being used. The realistic (also known as BEPU) methodologies will be introduced, as an evolution of the traditional conservative ones. The most widely used BEPU methodologies are based in Monte Carlo analysis, and some challenges posed by them in the regulatory activities are described. Most insights have been obtained in the evaluation work performed by the authors for the Consejo de Seguridad Nuclear (Spain’s nuclear authority).

2 THE FORMALISM OF DSA. CONSERVATIVE METHODOLOGIES

The DSA is based on calculations of accident consequences performed with predictive models. The models are deterministic, in two senses: first, they do not explicitly involve probabilities; second, they do not contain random number generators, so that two calculations with exactly the same input give the same results.

To fix some ideas let us suppose an accident or transient A deriving from an initiating event IE in a NPP and being simulated with a realistic and deterministic predictive model M, which can be viewed as a transformation from an input vector X into an output vector Y:

\[ Y = M(X) \]  

It is assumed that the model M is composed by several submodels and correlations and the input vector X contains all the data which are needed by the model to perform the calculation of Y for the transient A, including parameters which describe the plant and the accident, initial and boundary conditions and model parameters. Most of these inputs correspond to physical magnitudes which have a real or true value, and are estimated by measure, control, calculation, expert elicitation…or combinations of them. Therefore, these inputs have uncertainty. It is the case of initial and boundary conditions, geometrical parameters,…, and also of the empirical parameters contained in the submodels and correlations of M, which are adjusted through comparison with real data.

On the contrary, there are other inputs which have no “true” value, and their existence shows the imperfection of the model. That is the case of the parameters related to the discretization process conducted for solving differential equations, such as time step and spatial node sizes. They are not properly uncertain parameters, but their presence introduces errors in the value of the outputs (i.e. deviation from their true value). And such errors are uncertain, because they are not perfectly known.
Therefore, the components of $X$ can be roughly grouped in two categories:

1. Parameters representing real magnitudes
2. Parameters stemming from the imperfection of the model

The output vector $Y$ includes the results of the calculation which are important from the safety point of view, especially those intervening in the acceptance criteria for the transient. According to our previous discussion, the outputs have uncertainty coming from two main sources: the uncertainty of the inputs, propagated through $M$, and the imperfection of $M$. In what follows we will suppose that fixed values are assigned to the parameters in the category 2, so as to give a conservative bias to $Y$. Then, all the uncertainty of $Y$ is propagated from the inputs of category 1.

The uncertain magnitudes are classically described as random variables. In the sequel $X$ will represent a multidimensional random variable.

We will focus on the simplest case, when $Y$ is a scalar safety output, and further assume that the higher $Y$, the worse the consequences, so that $Y$ has an upper safety limit $L$. The limit $L$ is considered as a scalar, non-random magnitude, determined as a conservative (i.e. low) estimation of a damage threshold. This simple setting permits to lay down the main ideas about the deterministic analysis and the statistical methodologies.

Following [1,2], we define the probabilistic safety margin (PSM) of $Y$ in $A$ as:

$$PSM(Y; A) = PR\{Y < L | A\}$$

(3)

, namely the probability of $Y$ being under the limit $L$ conditioned to the occurrence of the transient $A$. It is clearly a safety margin, because it measures the distance from $Y$ to its “forbidden region” (from the safety point of view). The definition (3) is still valid in the case that $L$ has uncertainty. Also, the definition is easily generalized when $Y$ is a multidimensional vector (e.g when $Y$ describes multiple failure modes of a safety barrier) [1,2]

The objective of DSA is to demonstrate that the PSM takes a high value, or, in other words, that, with a high enough probability, the safety limits are not violated. The procedure to achieve this goal has two steps:

1) A highly conservative value of $Y$, which we will name $Y_{lic}$ (the subscript deriving from “licensing”) is obtained
2) The safety acceptance criterion is stated as $Y_{lic} < L$

For obtaining $Y_{lic}$, the concept of Design Basis Transient (DBT) is used. In our simple case, given two realizations $X_1$ and $X_2$ of the input $X$, we will say that $X_2$ bounds or envelops $X_1$ (and write $X_2 \succ X_1$) whenever $Y_2 = M(X_2)$ is higher than $Y_1 = M(X_1)$:

$$X_2 \succ X_1 \Leftrightarrow M(X_2) > M(X_1)$$

(4)

In mathematical terms, (4) defines an order relation, and the inputs to $M$ can thus be ordered according to their bounding or enveloping character.
Given an initiating event IE in a NPP, a DBT for IE is an accident sequence deriving from IE and built in such a way that it bounds a majority of the sequences deriving from IE. The DBT is constructed by using the criteria of “defence-in depth”, for instance postulating some “additional failures” in safety systems that must cope with the accident (this means that some inputs are set to overconservative values) and furthermore choosing the time and space discretization schemes so that additional conservatism is introduced. Let us call XD the input corresponding to the DBT and YD=M(XD). Then the following probability

$$PR\{XD > X\}$$

which, according to (4), coincides with

$$PR\{YD > M(X)\}$$

takes, by definition, a very high value (i.e. very close to 1). The probability in (5) represents the fraction of accidents deriving from IE and bounded by the DBT. The probability in (6) is an analogue to the PSM defined in (3), but with YD instead of L, and so it measures the smallness of the random variable Y=M(X) with respect to the random variable YD. We call (6) the probabilistic analytical margin associated to the DBT, and obviously measures the DBT conservative character.

In summary, DBTs are defined as transients which bound a majority of transients with the same origin (initiating event), so that they keep a high analytical margin with respect to the safety output.

XD still has uncertain components. Thus YD has uncertainty, propagated from these inputs. A probabilistic safety margin can be associated to YD:

$$PSM(YD) = PR\{YD < L\}$$

The bounding condition of the DBT implies that PSM(YD) is lower that PSM(Y).

The methodologies for performing the DSA can be classified according to their more or less realistic character. Here we will refer to two basic categories, the so-called conservative and realistic methodologies. The terms are a little misleading, because both of them use the DBT concept and so they are clearly conservative. The DBT is *a priori* built (i.e. before performing safety calculations), and the difference appears in the procedure for calculating Y\textsubscript{lic}. In conservative methodologies, Y\textsubscript{lic} is obtained by assigning fixed values to the uncertain inputs of XD. A worst case scenario is defined, in such a way that a number of important inputs (i.e. those influential on the results) are given conservative (even overconservative) values and the remaining, less important, inputs are kept in nominal or mean values. In this way, XD transforms into a non-random, very bounding input X\textsubscript{lim} (limiting input) and Y\textsubscript{lic} = M(X\textsubscript{lim}). Therefore, pessimistic predictive models and hypotheses are used, and the very conservative condition of Y\textsubscript{lic} justifies its calculation without uncertainty. Y\textsubscript{lic} will be the figure-of-merit compared to the safety limit L.
3 REALISTIC METHODOLOGIES

In realistic or BEPU methodologies, the DBT is built similarly to the conservative ones. Some inputs are given conservative values (e.g. postulating additional failures). The time and space discretization schemes are chosen with a criterion of global conservatism. The main difference with the conservative approach lies on the use of realistic predictive models and the way in which $Y_{lic}$ is obtained, through a statistical assessment of the uncertain variable $Y_D$, based on calculations of $M$ with different realizations of $X_D$. $Y_{lic}$ is finally obtained as an estimate of a high quantile of $Y_D$. Therefore, in this case $Y_{lic}$ is not calculated with a programmed input, but it is statistically estimated.

The conservative methodologies were absolute rulers over the deterministic realm for many years. With the increasing knowledge of the accident phenomenology in NPPs (mainly in the thermohydraulic and fuel behaviour fields) boosted by experimental and theoretical research programmes, the realistic analysis of accidents started to consolidate. The regulation had therefore to tackle the requirements to be imposed on the realistic methodologies.

In 1989 the USNRC released a new version of the rule 10 CFR 50.46 wherein the use of realistic methodologies was accepted to perform LOCA-ECCS analyses. In the Regulatory Guide 1.157 [3] the acceptable models for performing such analyses were described, and the acceptance criteria were given a probabilistic shape, so that the requirement was no more the strict compliance of the criteria, but the compliance with a high probability. In our simple framework, the traditional acceptance criterion for conservative methodologies:

$$Y_{lic} < L$$ (8)

transforms into a probabilistic criterion for realistic methodologies

$$PR\{Y_D < L\} \geq Q$$ (9)

where $Q$ is a high (close to 1) value, imposed by the regulatory authorities.

Development and assessment of BEPU methods have been subject of extensive work within the nuclear safety community. The first realistic methodology, developed for LOCA analysis, was CSAU [4]. International efforts have promoted the development of uncertainty analysis methods [5,6]. Two basic approaches have been considered: i) based on uncertainty propagation through the predictive model (as described in section 2) and ii) based on the so-called “internal assessment of uncertainty”, which benefits from a direct calculation of the model error uncertainty using experimental data. In the present paper regulatory experience on the licensing of type i) methodologies is described.

The basic stages in a realistic or BEPU methodology of safety analysis are:

1) The most important inputs to the predictive model are selected, and its uncertainty is estimated. The result is a set of basic uncertainties.
2) The basic uncertainties are propagated through the model, to obtain the uncertainty of the safety outputs.
3) The safety outputs are compared to their limits, and an estimation of the safety margin is obtained.
4 DESCRIBING UNCERTAINTY

As we have already pointed out, the classical description of uncertainty is probabilistic, in such a way that an uncertain magnitude is represented as a random variable. From this point of view, the probability distribution of the variable completely describes its uncertainty. An uncertain magnitude \( V \) can be described by the cumulative distribution function, cdf, defined as:

\[
F_v(v) \equiv PR\{V \leq v\}
\]  
(10)

or by the probability density function, pdf, which is the derivative (when it exists) of the cdf:

\[
f_v(v) \equiv F_v'(v)
\]  
(11)

But uncertainty, in a more restricted way, can be described by:

- A scalar parameter, e.g. standard deviation or difference between quantiles
- An interval containing the value of the variable with a high probability.

If uncertainty is probabilistically described, it should be statistically estimated. On estimating uncertainty through random samples, the sample counterparts of the aforementioned descriptors are used, for instance the sample standard deviation or the so-called statistical intervals.

A statistical interval is, informally speaking, an interval which contains the true value of the magnitude with a high likelihood. The term statistical stems from the fact that the endpoints of the interval are statistics (i.e. functions of the variable’s sampled values). We will next focus on a specific type of statistical intervals, named tolerance intervals. First we will introduce the concept of coverage. Given a random variable \( V \) and an interval \((L,U)\), we define:

\[
W_v(L,U) \equiv F_v(U) - F_v(L) \equiv PR_v\{L < V \leq U\}
\]  
(12)

as the coverage of \( V \) by \((L,U)\). We also say that \((L,U)\) covers \( V \) with a probability \( W_v(L,U) \).

A two-sided tolerance interval of level \( A/Q \) (where \( A \) and \( Q \) are both real numbers higher than 0 and less than 1) for the uncertain variable \( V \) is a statistical interval \((L_T, U_T)\) covering a fraction higher than \( Q \) of the variable \( Y \) with a confidence level \( A \):

\[
PR_{L_T, U_T}\{PR_v\{L_T \leq V \leq U_T\} > Q\} = A
\]  
(13)

It should be noticed that both \( L_T \) and \( U_T \) are random variables, dependent on the sample of \( Y \) values.

One-sided tolerance intervals are described by tolerance limits. Upper and lower tolerance limits with level \( A/Q \) for \( V \) are, respectively, statistics \( U_T \) and \( L_T \) such that:
\[ PR_{U_r} \{ PR_r \{ V \leq U_r \} > Q \} = A \]  \hspace{1cm} (14)

and

\[ PR_{L_r} \{ PR_r \{ L_r \leq V \} > Q \} = A \]  \hspace{1cm} (15)

It seems obvious that, when both Q and A are high values (i.e. close to 1), a two-sided tolerance interval is an adequate descriptor of a generic uncertain magnitude, because it covers a high fraction of the V values with a high statistical confidence. Nonetheless, the classification as “safety” of a magnitude affects the description of its uncertainty. For a safety magnitude having an upper (resp. lower) safety limit, the natural descriptor of uncertainty is an upper (resp. lower) tolerance limit. This seems quite logical, because the important values of the output, from the safety point of view, are those on the conservative side. For a safety magnitude like YD an A/Q upper tolerance limit will be used, A and Q being high values in the interval (0,1).

Tolerance interval endpoints and tolerance limits are statistically estimated, and thus they have statistical uncertainty, due to the finiteness of the random sample used in the estimation. So we have descriptors of the uncertainty which in turn are uncertain. This uncertainty of the uncertainty can be properly called metauncertainty, and has the property of tending to zero when the random sample size tends to infinity.

5 METHODOLOGIES BASED ON MONTE CARLO

The uncertainty propagation is a fundamental step in the BEPU methodologies, and the purpose of this paper is not to give a thorough survey of the propagation methods, but rather to refer to the methodologies based on Monte Carlo calculations and specially to those based on the propagation method that has become the most popular and widely used in the last years, namely the order statistics (OS) method, also known as Wilks method. The Monte Carlo method will be considered as directly applied to the predictive model M. Surrogate models, like the response surfaces proposed in the CSAU methodology [4], will not be treated in this paper.

5.1 Methods based on order statistics

The OS method is based on a pure Monte Carlo analysis of the magnitude YD=M(XD), wherein the uncertain inputs making up XD are randomly sampled so that a simple random sample of XD, (XD_1, …, XD_N) is obtained. The model M is then run for the N sampled inputs and thus a simple random sample (YD_1, …, YD_N) of the safety output YD is obtained. The hypothesis that YD is a continuous random variable with a continuous pdf will be adopted from now on.

The peculiarity of the OS method refers to the statistical analysis of the YD sample. The sampled values can be sorted (YD_{1:N}, …, YD_{N:N}) low to high, YD_{1:N} and YD_{N:N} being
respectively the sample minimum and maximum. Then, the value \(Y_{D_{r:N}}\) is called the \(r\)-th order statistic (OS) of the sample. Conventionally, statistics of order 0 and \(N+1\) are respectively defined as the minimum and maximum of the complete \(Y_D\) range.

The usefulness of order statistics in uncertainty assessment stems from the fact that they can be used as endpoints of tolerance intervals for the uncertain variable. Let \(Y_{D_{r:N}}\) and \(Y_{D_{s:N}}\) two OS such that \(r<s\). The coverage of \(Y_D\) by the interval \((Y_{D_{r:N}}, Y_{D_{s:N}})\) is:

\[
W_{Y_D}(r,s) = F_{Y_D}(Y_{D_{s:N}}) - F_{Y_D}(Y_{D_{r:N}}) = PR\{Y_{D_{r:N}} \leq Y_D \leq Y_{D_{s:N}}\}
\]

(16)

The interesting property is that \(W_{Y_D}(r,s)\) has a well known probability distribution, namely the beta distribution with parameters \(s-r\) and \(N-s+r+1\), and this is true whatever \(F_{Y_D}\) be [7]. This is the reason why the OS method is described as nonparametric or distribution-free.

Then it is clear that:

\[
PR\{PR\{Y_{D_{r:N}} \leq Y_D \leq Y_{D_{s:N}}\} > Q\} = PR\{\text{beta}(s-r, N-s+r+1) > Q\}
\]

(17)

By comparison with (13) it is concluded that \((Y_{D_{r:N}}, Y_{D_{s:N}})\) is an \(A/Q\) tolerance interval for \(Y\) whenever

\[
PR\{\text{beta}(s-r, N-s+r+1) > Q\} = A
\]

(18)

The probability distribution of the beta variable is codified in statistical packages and spreadsheets, so that the equation (18) can be implicitly resolved to find out pairs \((r,s)\) giving the desired tolerance level. Actually, (18) has, in general, no solution for integer values of \(N\), \(s\) and \(r\), and thus it is replaced by the inequation

\[
PR\{\text{beta}(s-r, N-s+r+1) > Q\} \geq A
\]

(19)

so that integer values are obtained fulfilling (19) while minimizing the probability on the left hand side.

But, as already pointed out, the uncertainty of our safety output \(Y_D\) is more properly described by an upper tolerance level, rather than a two-sided interval. The expression (19) with \(r=0\) reads:

\[
PR\{\text{beta}(s, N-s+1) > Q\} \geq A
\]

(20)

that can also be expressed as

\[
_{1-A}\text{beta}(s, N-s+1) > Q
\]

(20-bis)

the left hand side being the \((1-A)\) quantile of the beta variable. Integer values of \(N\) and \(s\) fulfilling (20) or (20-bis) while minimizing the respective left hand sides are such that \(Y_{D_{s:N}}\) is an \(A/Q\) upper tolerance limit for \(Y_D\). There is a minimum sample size \(N\) needed to obtain a tolerance interval or limit for a prescribed level \(A/Q\). For instance, at least 59 samples are necessary to obtain a 95/95 tolerance limit.
In Wilks methodologies the licensing value $Y_{lic}$ is finally obtained as an order statistic $Y_{D_{CM}}$ from the $Y_D$ random sample.

In figure 1 a schematic view of the safety outputs to the BEPU analysis is shown. The safety output $Y$ transforms into $Y_D$ when the DBT is considered. $Y_{lic}$ is an upper tolerance limit for $Y_D$, covering a high quantile of $Y_D$ with high confidence. Each safety output is depicted by a pdf. It is interesting to point out that $Y_{lic}$ has metauncertainty; if the Monte Carlo sample size increases, the pdf of $Y_{lic}$ will become narrower, whilst those of $Y$ and $Y_D$ will remain unchanged.

We have described the Wilks method for a single scalar magnitude. When considering multidimensional magnitudes the method becomes more complex to apply. For safety analysis the univariate version is enough, because there is actually a single scalar safety output, namely a Boolean index taking the value 1 when all the safety criteria are fulfilled and 0 otherwise. The multivariate version should be used in the process of validation of the predictive model [10].

### 5.2 Parametric Methods

The OS method uses a nonparametric inference about $Y_D$, and hence it is especially appealing when there is no *a priori* assumption about the probability distribution of $Y_D$. When there is information supporting the ascription of $Y_D$ to a known family of probability distributions, for instance when the random sample of $Y_D$ values fulfil a goodness-of-fit test, statistical parametric methods can be used instead of OS. The most typical instance is when the $Y_D$ data fit a normal distribution, or when they can be transformed to data that fit a normal distribution. In [8,9] expressions are given for normal tolerance intervals, usable as uncertainty descriptors. A normal upper tolerance limit for $Y_D$ is obtained as

$$m_{Y_D} + k_{AQ} s_{Y_D}$$  \hspace{1cm} (21)

where $m_{Y_D}$ and $s_{Y_D}$ are respectively the mean and standard deviation of the $Y_D$ sampled values. $k_{AQ}$ is a coefficient depending on $A$, $Q$ and $N$.

Contrary to the OS, the normal intervals do not require a minimum sample size to reach the tolerance level $A/Q$. As a counterweight to it, a small sample can be considered as insufficient to test the hypothesis of normality.

### 6 INTERPRETING AND EVALUATING MONTE CARLO METHODOLOGIES

When the regulator faces the task of evaluating a safety analysis methodology based on the Monte Carlo method, the three stages quoted in section 3 must be properly checked. The OS-based methodologies propagate the uncertainties by means of pure Monte Carlo calculations and the use of OS. This method has been worldwide accepted. Nevertheless there are points that the evaluation should focus on, roughly corresponding to the three aforementioned stages:

- Construction of the basic uncertainties
- Performance of the Monte Carlo calculations
- Interpretation of the results and calculation of safety margins
6.1 Checking basic uncertainties

The uncertainty of a safety output must not be underestimated, especially in the conservative range. Whenever all the uncertainty is propagated from the inputs, and the propagation method is sound, the interest is centred in the proper estimation of the input uncertainty, in such a way that it is not underestimated in the conservative side.

Let us denote as $Z$ an uncertain component of the vector $XD$ (i.e. a scalar input to $M$). Different procedures exist in order to estimate the distribution function of $Z$, in the form either of pdf $f_Z(z)$ or of cdf $F_Z(z)$. When a random sample exists of values of the parameter, the data can be used to build an empirical pdf (a histogram) or an empirical cdf (ecdf), which is a stepwise function defined as:

$$F_Z^{emp}(z) \equiv fr(Z \leq z) \quad (22)$$

i.e. the fraction of sampled data being lower or equal to $z$. The disadvantage of the histogram is that it depends on the data binning criterion; instead, the ecdf defined in (22) does not require any binning criterion.

But it is important to point out that a distribution function estimated with a finite sample has statistical uncertainty, that we have called metauncertainty, and when it is taken into account, the ecdf defined in (22) must be supplemented with confidence bands. These can be of two types:
- **Point-wise**: for each value \( z \) they contain the real \( F_Z(z) \) with a prescribed confidence level
- **Joint**: they contain, with a prescribed confidence level, the whole real cdf curve

The cdf should then be estimated by means of a one-sided confidence band. For instance, if the conservative values of \( Z \) are the higher ones, a lower confidence band should be used, because it favours such high values.

Sometimes, the data are used to build a probability distribution belonging to a certain family (e.g., normal, lognormal, gamma, etc). In such cases, a statistical test should previously be conducted, keeping in mind that the fitting of distribution should be conservatively performed.

In fact, when data are really scarce, more drastic measures should be taken. For instance, sometimes only the range of the parameter is known, and then a possible choice is fixing the value of the parameter to a conservative value. In this case, the uncertainty of the parameter is really taken into account, but it finally contributes to the conservative bias of the prediction model rather than to the output uncertainty.

Another possibility is to introduce some expert opinion. For instance, the expert can recommend the use of a uniform distribution spanning the range of the parameter, hence assigning the same probability to all the values in a certain interval. In such case, the interval should be wide enough in the conservative direction. Other possibility is that the expert assigns to the parameter a known distribution function, for instance a normal distribution having as mean a nominal value and as standard deviation a certain percentage of the mean. The expert’s assumptions must be properly justified.

It is not unusual the truncation, for practical reasons, of the parametric distributions being used. For this procedure to be acceptable, the eliminated tail in the conservative side should be small enough.

### 6.2 Monte Carlo calculations

Once a random sample of inputs \( XD \) has been obtained, the predictive model \( M \) must be run for the different input realizations. This pure Monte Carlo analysis can be viewed as a binomial experiment (just like tossing \( N \) times a coin), the two outcomes being success (the safety criteria are fulfilled by the run output) and fail (some safety criteria are not fulfilled).

Some provision should be made about the runs failing to completion. In general, most of the failed cases can be successfully completed when rerun with a change in time step, or in some convergence criteria. If there are runs failing to completion even with such measures, the conclusion is that the predictive models have shortcomings, and such conclusion should be communicated to the experts in the model (developers, V&V team). If the experts believe that the shortcomings of the model in some isolated cases do not imperil the capacity to simulate the transients, then the strategy of obtaining a new random input to replace the one causing the failure could be acceptable. In fact a binomial model is being tested (from the probabilistic point of view) wherein only two results are accepted (namely success or fail in compliance of the safety criteria). The success probability, which is the probabilistic safety
margin, is estimated through the number of successful runs and the total number of completed runs.

6.3 Interpreting the results

When the Wilks method is applied to a N-sized random sample of the uncertain variable YD, the minimum integer s satisfying (20) is obtained, so that YD,s:N is an A/Q upper tolerance limit for YD. The safety criterion takes the form

\[ YD_{s:N} \leq L \]  

(23)

The OS methodologies have a prompt interpretation in the realm of DSA, stemming from the definition of tolerance intervals. For a safety output as YD, nonparametric upper tolerance limits are obtained, covering a high quantile of the safety output with a high statistical confidence, and this fact (together with the bounding nature of the DBT) evidences the conservatism of the obtained Y_{lim} to be compared to the safety limit L.

But a more powerful interpretation can be done, in terms of probabilistic safety margins. The precise meaning of (23) is that at least s out of N random values of YD have fallen under the safety limit. A binomial experiment allows us to estimate the probability of the outcomes, for instance the probability of ‘head’ when tossing a coin. The Monte Carlo results allow us to estimate the probability of Y being under L, which is the PSM. Therefore, the use of the Monte Carlo method in the field of DSA not only permits to state that “enough safety margins exist”, but to quantify such safety margin in terms of probability.

A clear relation exists between giving a tolerance limit for YD and giving a confidence limit for PSM(YD). In [2] it is mentioned that, when N Monte Carlo calculations are conducted and s of them give YD<L, a lower confidence limit with level A for PSM(YD) is the (1-A) quantile of the beta distribution with parameters s and N-s+1 (Clopper-Pearson limit). Comparing this statement with (20-bis) it is concluded that running N Monte Carlo cases, producing an A/Q upper tolerance limit YD,s:N and checking that it is less than L, is equivalent to say that PSM(YD) is higher than Q with at least a confidence level A:

\[ \Pr\{\Pr\{YD < L\} \geq Q\} \geq A \]  

(24)

The innermost probability in (24) refers to the uncertainty of YD; the outermost one refers to the metauncertainty stemming from the finiteness of the sample. (24) is the form assumed by the probabilistic acceptance criterion (8) when the metauncertainty is taken into account.

In [2] several methods for calculating probabilistic safety margins from random samples of safety outputs are gathered.

For instance, if N=105, then s=104 is the minimum solution of (19) with A=0.97 and Q=0.95, so that YD_{104:105} is a 97/95 upper tolerance limit for YD. And if we confirm that YD_{104:105} < L, then it is concluded that PSM(YD) is higher than 0.95 with a confidence 0.97.

We introduced the concept of limiting output in connection with conservative methodologies. For BEPU methodology the definition is the same:
\[ Y_{\text{lic}} = M(X_{\lim}) \]  

(25)

In OS-based methodologies \( X_{\lim} \) is obtained as the input giving rise to \( Y_{D:s:N} \). According to the order relation between inputs defined in (4), the XD sample can be sorted low to high, \( (X_{D_{1:N}}, \ldots, X_{D_{N:N}}) \) in such a way that \( Y_{D_{k:N}} = M(X_{D_{k:N}}) \), \( k=1, \ldots, N \). Then it is clear that \( X_{\lim} = X_{D_{s:N}} \), \( s \) being the minimum integer fulfilling (16). The conclusion is that the OS-based methods directly produce not only the licensing value, but the limiting output as well.

On the contrary, when the normal distribution is used for uncertainty propagation, the limiting output is not obtained as a direct by-product of the uncertainty assessment. The reason is that \( Y_{\text{lic}} \) is not obtained as an OS, but as a function, like (21), of the mean and standard deviation of the YT values.

In (6) we have defined the probabilistic analytical margin for YT, as a measure of the DBT conservative character. An analogous measure of the conservativeness of \( Y_{\text{lic}} \) with regard to YT can be defined. Let us suppose that a new random value of YT is sampled, additionally to the N previously sampled values. The probability that the new value is not higher than the s-th order statistic of the previous N-sized sample

\[
PR\{Y_{D_{N+1}} \leq Y_{D:s:N}\} \]

(26)

is a measure of the conservativeness of YT:s:N with regard to YT. An advantage of the OS method is that the probability (26) can be easily calculated [7] and found to be

\[
PR\{Y_{D_{N+1}} \leq Y_{D:s:N}\} = \frac{s}{N+1}
\]

(27)

In our previous example, with \( N=105 \) and \( s=104 \), the probability after (27) is 0.981.

There is an infinite number of pairs \((s,N)\) of integer values such that \( Y_{D:s:N} \) is an A/Q tolerance limit. But they have different degree of conservativeness, described by (27). \( Y_{\text{lic}} \) is expected to be less conservative as \( N \) increases. In the limit, the metauncertainty disappears, and the OS tends to the Q quantile of YT:

\[
\lim_{N \to \infty} PR\{YD \leq Y_{D:s:N}\} = Q
\]

(28)

Then, when \( N \) increases the knowledge of the Q quantile of YT enhances. On the other hand, when \( N \) decreases the expected conservativeness of \( Y_{\text{lic}} \) increases.

As described in 5.2, if the YT data pass some goodness-of-fit test, parametric methods can be used in order to obtain \( Y_{\text{lic}} \). The normal distribution is the standard example. Normality tests are a special class of statistical tests wherein the null hypothesis is the normality of the analyzed variable. The convention exists of accepting the null hypothesis unless the sampled data show a clear discrepancy with it. The so-called p-value of the test is a measure of the agreement or compatibility of data and the null hypothesis. Sometimes a very low value of the p-value (e.g. \( >0.05 \)) is admitted as sufficient to accept the normality hypothesis. This can lead to an underestimation of the uncertainty, for instance when the real distribution is heavy-
tailed. When conducting this kind of tests for a safety magnitude, the default hypothesis should be non-normality rather than normality. In other words, the replacement of the nonparametric OS tolerance limits by normal ones should only be accepted if there is strong evidence of normality, including a high test p-value.

7 OTHER REGULATORY IMPLICATIONS

Whenever a new methodology is accepted for application in a safety case the implications in the regulatory frame must be assessed. Typically a revision of the Safety Analysis Report (SAR) will be issued where the analyses affected by the new methodology will be updated. As a result the limiting transient for a given safety limit may change. Also there may coexist within the SAR both methodologies, conservative and BEPU, applied to different set of transients. A careful analysis of the consistency of the analytical frame in the SAR must be conducted.

A special analysis should be made on the impact of the new method and the aforementioned analytical frame to the Technical Specifications (TS) of the plant. TS set limits on several components of X, which are operational parameters, defining a region in the X space where safe operation of the NPP is analytically supported and thus allowed by the regulator. In a simplified way, this means that any transient within the design basis has acceptable consequences provided that the input X is inside the TS acceptance region. The TS concept has been directly linked to the conservative methodologies of DSA, because they permit an easy checking of the safety acceptance criteria (using a single calculation, equation (6)) and therefore a quick exploration of the acceptance region boundaries. On the contrary, BEPU methodologies need N calculations in order to check the safety criteria (equation (24)), and perform a random sampling of the X space instead of looking for the boundaries of the acceptance region. A simplified procedure should imply analyzing a worst case scenario where the TS parameters are set to their analytical value in the input XD, and this would require a set of N Monte Carlo calculations additional to those performed for the input XD. In summary, the computational effort to check TS could soar up for BEPU methodologies.

A looser interpretation, in which the TS simply set limits on the ranges of some uncertain operational parameters, would not allow to state that operation in any point of the TS acceptance region is analytically supported. The interpretation of TS in relation with BEPU methods should be further studied.

Another relevant issue regarding the regulatory perspective is the treatment of plant modifications and how the validity of the SAR is maintained. In the case of Monte Carlo methods it is important to determine when a complete reassessment must be made, and how to perform it. For performing the input’s random sampling a set of pseudorandom numbers is use, using a random seed as starting point. If the safety analysis is repeated, the random seed of the base case can be maintained or it can be changed. The statistical inferences that can be performed in each case are different, and hence the regulator should pay attention to this “reseeding policy”. The same situation would be apparent when errors are discovered in the methodology or the predictive models.

Last but not least, a decision has to be taken on the parameters included in the probabilistic acceptance criteria shown in expression (24), namely the level of coverage Q and of confidence A.
8 CONCLUSIONS

The increasing knowledge on accident phenomenology has promoted the use of realistic or BEPU methodologies. Those based in the use of Monte Carlo with order statistics for uncertainty propagation have become the most widespread. Such methodologies allow not only to state that “enough safety margins exist” but to quantify such safety margins in probabilistic terms (which are the really useful ones). The evaluation of a BEPU methodology must focus on the proper estimation of the basic uncertainties, which must not be underestimated, and on a right interpretation of the results.

Parametric methods can be used as an alternative to the OS-based; the most typical are those based in the normal distribution. When testing normality, the default hypothesis should be the non-normality, so as to minimize the risk of underestimating the uncertainty.

An interesting point from the regulatory side is to study the relation between Technical Specifications and the BEPU methodologies.

REFERENCES

Abstract

The IAEA is already long time involved in activities related to accident management. It has developed various documents on this subject, and also initiated a review service, where the accident management measures of an individual plant are reviewed.

Recently, the IAEA compiled a draft Safety Guide on Accident Management that comprises the main elements of such management in a complete and consistent way. It describes which steps should be taken in setting up an accident management program, from the conceptual stage down to a complete set of instructions - procedures and guidelines - to the plant operators.

Today’s nuclear power plants have multiple barriers against the release of radioactive substances, plus a number of supportive systems that protect these barriers. There is a set of instructions, usually called Emergency Operating Procedures (EOPs), to deal with a large variety of credible events, both inside and outside the design basis. Nevertheless, there is a remote chance that these instructions are not successful, and that core damage will occur. In that case, plants still have capabilities to mitigate potential releases. The IAEA draft Safety Guide gives guidance on how such measures should be defined and how they should be executed.

The process is in a number of well-defined steps. First, the plant vulnerabilities for severe accidents (i.e. accidents with substantial core damage) are defined, as well as their timing. This gives a certain sequence of challenges to fission product boundaries. Then the plant capabilities to mitigate those challenges are researched. This leads to a number of possible strategies, which then are converted to mitigative measures. From there, procedures and guidelines are developed. Priorities are defined on the basis of the timing and magnitude of potential releases.

Major differences with the EOP-domain are that the plant status may be known only partially, the outcome of actions cannot always be predicted, and planned actions can even have severe negative consequences (e.g., ignition of hydrogen may lead to loss of containment). This complicates the decision making process. The IAEA draft Safety Guide gives guidance on how the evaluation and decision-making processes should be set up and where the authority should be placed.

A relatively new type of IAEA service in this field is the Review of the Accident Management Program (RAMP) of an individual plant. The objective of the RAMP is to assist at the utility and plant designer in preparation, development and implementation of effective plant specific accident management programme. However, assistance can also be provided to the regulatory body in reviewing of accident management programme. Such a review includes a one-week visit of a group of experts (about 5) to a nuclear site, a review of the relevant documentation, interviews with plant staff and, finally, discussion of findings and formulation of
recommendations. A guide has been prepared to structure this process. Three plants have been reviewed so far. These reviews have been received very well by the parties involved.

1. INTRODUCTION.

The IAEA has set up a series of documents that specify requirements and guidance for the design and operation of nuclear power plants (NPPs). Member States can use these as regulatory requirements and guidance in their national programs. Requirements are obligatory (i.e. ‘shall’ statements), guidance specifies recommendations how the requirements can be met (‘should’ statements).

The documents NS-R1 [1] and NS-R2 [2] specify the design and operations requirements, respectively; they require NPPs to also consider both design provisions and accident management procedures for severe accidents. The draft Safety Requirements [3] on Safety Assessment for Facilities and Activities states that the assessment of defence in depth is required to determine whether adequate provisions have been made at each of the levels of defence to identify accident measures to control severe accident conditions and to mitigate the radiological consequences of potential release. Before, the IAEA had already developed a number of documents for Member States to support them in establishing accident management at their plants. The Agency had compiled documents on the development of Emergency Operating Procedures (EOPs) [4] and on the implementation of accident management programs in NPPs [5], which latter document deals mostly with accidents beyond the design basis and severe accidents. The phenomenology of severe accidents and various analysis methods have been described in [6, 7]. Recently, a draft Safety Guide [8] has been compiled to give further guidance to the implementation of the requirements of [NS-R1], which is the subject of this article.

Today’s nuclear power plants have multiple barriers against the release of radioactive substances, plus a number of supportive systems that protect these barriers. The EOPs deal with a large variety of credible events, both inside and outside the design basis. Nevertheless, there is a remote chance that the EOPs are not successful, and that core damage will occur. In that case, plants still have capabilities to mitigate potential releases. The IAEA draft Safety Guide gives guidance on how such measures should be defined and in which way they should be executed.

The ultimate goal of accident management is to reduce risk in the unlikely case of a severe accident. Even if these are low probability events: ‘operators should never be placed for an accident that has not been previously analysed by an engineer’ - a paramount lesson from the TMI-2 event, as formulated by one of its operators.

2. STRUCTURE OF THE SAFETY GUIDE

The Guide has been developed in two parts:
Part 1 contains the basic principles, which are the ‘foundations’ of the program, i.e. fundamental characteristics that should be observed as core elements of the program.
Part 2 contains the technical and organisational elements of developing the AM-program, i.e. all work that must be done to finally arrive at a set of guidelines for use by plant personnel.
3. SETTING UP ACCIDENT MANAGEMENT GUIDANCE.

Accident management is a series of actions with the objectives:
1. to prevent a beyond design basis event to escalate into a severe accident,
2. to terminate the progress of core damage once it has started;
3. to maintain the capability of the containment as long as possible;
4. to minimize releases of radioactive material; and
5. to achieve a long term stable state.

The last four points constitute ‘severe accident management’ (SAM). The guidance in this field is usually called ‘severe accident management guidance / guidelines’ (SAMG).

Setting up a AMG-program for an NPP requires a number of steps in a top-down approach:
1. define the objectives, as described above.
2. define strategies to achieve the objectives
3. define measures to execute the strategies
4. develop procedures and guidelines for the actions to be taken by plant personnel

The process is illustrated in Fig 1.
4. CONCEPTUAL ELEMENTS

This section specifies some main principles, aspects of equipment upgrades, forms of the accident management guidance and roles and responsibilities. They are the basis for further development of the guidance. The text of this section is - mainly - a quote from the draft Guide and marked as such, where explanations / examples are here added in *italics* - they do not form part of the Guide. Paragraphs are not exact quotes, for reasons of brevity; numbering follows the numbering of this article, not that of the Guide.

**MAIN PRINCIPLES**

1. In view of the uncertainties involved in severe accidents, accident management guidance should be developed for *all physically identifiable challenge mechanisms* for which the development of accident management guidance is feasible; this should be performed largely independently of predicted frequencies of occurrence.
   
   *Arguments that certain plant damage states need no mitigation because of perceived/calculated low probability should, in principle, not be credited, as long as guidance for such states can be developed.*

2. Accident management should be *symptom based*, i.e. the strategies and the associated measures should be based on directly measurable plant parameters or parameters derived from these by simple calculations.
   
   *Strategies should not be developed on complex parameters such as peak clad temperature*

3. Accident management guidance should be set up in such a way that it is not necessary for the responsible staff to identify the accident sequence or to follow some pre-analysed accident

4. Accident management should cover at all modes of plant operation and also appropriately selected external events, such as fires, floods, seismic events and extreme weather conditions (e.g. high winds, extremely high or low temperatures, droughts) that could damage large parts of the plant. The accident management guidance should consider specific challenges posed by these events such as loss of the power supply, loss of the control room or switchgear room, and reduced accessibility to systems and components.

5. When developing guidance on accident management, consideration should be given to the plant’s full design capabilities, using both safety and non-safety systems, and including the possible use of some systems beyond their originally intended function and anticipated operating conditions and possibly outside their design basis.
EQUIPMENT UPGRADES

6. Design features important for the prevention or mitigation of severe accidents should be evaluated. Accordingly, existing equipment, and/or instrumentation should be upgraded or new equipment, and/or instrumentation should be added, if needed or useful for the development of a meaningful accident management program.

7. If a decision is taken to add or upgrade equipment and/or instrumentation, the design specification of such equipment and/or instrumentation should be such as to ensure appropriate independence from existing systems and preferably appropriate margins with regard to their use under accident and/or severe accident conditions.

8. The installation of new equipment or the upgrading of existing equipment should not remove the need for the development of guidance for the situation that such equipment fails, even if such failure has a low probability.

FORMS OF ACCIDENT MANAGEMENT GUIDANCE

Preventive domain

9. In the preventive domain, the guidance should consist of descriptive steps, as the plant status is known from the available instrumentation and the consequences of actions can be predetermined by appropriate analysis. The guidance, therefore, takes the form of procedures, usually called emergency operating procedures (EOPs), and is prescriptive in nature. EOPs cover both design basis accidents and beyond design basis accidents, but are generally limited to actions taken prior to core damage.

Mitigative domain

10. In the mitigative domain, uncertainties may exist both in the plant status and in the outcome of actions. Consequently, the guidance should not be prescriptive in nature but rather should propose a spectrum of potential mitigating actions and thus leave space for additional evaluation and alternate actions. Such guidance is usually termed severe accident management guidelines (SAMGs).

11. The guidance should contain a description of both the positive and negative potential consequences of proposed actions, including quantitative data where available and relevant, and should contain sufficient information for the plant staff to reach an adequate decision during the evolution of the accident.

12. The guidance should be sufficiently detailed to support the responsible staff in its deliberations and decisions in a high-stress environment, with a minimum chance to delete or overlook relevant information. The guidance should not be shaped in such a form and with such
detail that responsible personnel will tend to follow it verbatim, unless that is the intended type
of action.

13. The overall form of the guidance and the selected amount of detail should be tested in drills
and exercises. Based on the outcome of such drills, it should be judged whether the form is
appropriate and whether additional detail or less detail should be included in the guidance.

**ROLES AND RESPONSIBILITIES**

14. Accident management guidance should be an integral part of the overall emergency
arrangements at a nuclear power plant. The execution of the accident management guidance is
the responsibility of the emergency response organization at the plant or the utility. Roles and
responsibilities for the different members of the emergency response organization involved in
accident management should be clearly defined and coordination among them should be ensured.

15. A specialized team or group of teams (referred to in the following as the technical
support centre) should be available to provide technical support by performing evaluations and
recommending recovery actions to a decision making authority, both in the preventive and
mitigative domains. It should also provide appropriate input to the people responsible for the
estimation of potential radiological consequences. For multiple teams, the role of each team
should be specified.

16. The decision-making authority should be placed at an appropriate level commensurate
with the complexity of the task and the potential for on-site and off-site releases. In the
preventive domain, the control room shift supervisor or a dedicated safety engineer should be
largely able to carry this responsibility, whereas a higher level of decision making is
recommended in the mitigative domain.

**5. PROCESS OF DEVELOPMENT OF AN ACCIDENT MANAGEMENT PROGRAM**

The AM guide contains a full chapter on the process of setting up the accident management
programme, outlined in 128 numbered paragraphs. Further technical detail which can be helpful
is contained in references, such as [5]. The following subjects are treated:

1  General remarks
2  Identification of plant vulnerabilities
3  Identification of plant capabilities
4  Development of accident management strategies
5  Development of procedures and guidelines
6  Hardware provisions for accident management
7  Role of instrumentation and control
8  Responsibilities and lines of authorization
9  Verification and validation
10  Education and training
11  Processing new information
Ad 1. General Remarks

This section addresses a.o. the selection of events to be considered. It describes the process of obtaining the events, not to exclude events because of assumed/calculated low probabilities, and also to check, at the end, whether indeed the important risk contributors are covered with means that indeed reduce risk. A PSA is helpful, in obtaining the events to be considered, but is not the only tool - other insights should be used as well, e.g. similar studies from other plants, operating experience and research on severe accidents.

After completion of the accident management guidance, it should be checked whether indeed all important accident sequences, in particular those obtained from the PSA, are covered, and whether plant risk is reduced accordingly.

Ad 2. Identification of plant vulnerabilities

A comprehensive set of insights should be obtained on the behaviour of the plant during a beyond design basis accident and severe accident; these should identify the phenomena that will occur and their timing and severity. In the severe accident domain, these insights are collected in the technical basis for severe accident management. This technical basis is often documented separately, as a collection of insights from where the AM program is developed.

The insights should be obtained using appropriate analysis tools. Also other inputs should be used, such as the results of research on severe accidents, insights from other plants and engineering judgment. Uncertainties in severe accident models and the assumptions made should be considered in developing the insights. Often, the plant PSA, if available, is used, but also other insights, as described, are important.

Ad 3. Identification of plant capabilities

All plant capabilities available to fulfil the safety functions should be investigated, including the use of non-dedicated systems, unconventional line-ups and temporary connections (hoses, mobile equipment) and use of systems beyond their design basis, up to and including the possibility of equipment damage. It should also be considered whether failed systems can be restored to service and, hence, can again contribute to the mitigation of the event. Where unconventional line-ups and temporary connections are identified, consideration should be given to adaptation of equipment necessary to use these capabilities.

This is a complex matter. Although the principle elements of such strategies are relatively straightforward (e.g. depressurising the RCS), the strategies are interlinked and influence each other. Unlike in the preventive domain, strategies in the mitigative domain can also have negative consequences, which complicates the matter of prioritising and decision-making. As severe accidents evolve over a larger time frame, there should be time for evaluating positive and negative consequences. But for fast-developing scenarios, such time may not be available and the decision making process should account for it. Strategies are not derived from a perceived underlying scenario, but from measurable parameters indicative of plant damage.

There should be a systematic evaluation of the possible strategies that can be applied, taking into consideration the evolution of the accident. In selecting and prioritising strategies, it should be noted that there is an increased importance of evaluation due to the presence of multiple potential negative impacts, and of increased levels of uncertainty in plant status and potential response to actions.

Special attention should be devoted to strategies that have both positive and negative impacts in order to provide the basis for a decision as to which strategies constitute a proper response under a given plant damage condition. An example is flooding the cavity, with the negative impact of the possible occurrence of a steam explosion.

Insights into the plant damage states in the evolution of the accident should be obtained wherever possible. They are helpful as they can help to select strategies, because some strategies can be effective in one plant damage state, but may be ineffective or even detrimental in another. In addition, such insights are relevant for the estimation of the source term and, if available, should be used for this purpose. Examples are: adding water to a core is generally beneficial; however, if it is done at the moment the fuel stack is still intact but the control rods have melted away, a large power spike may follow. If only a limited amount of water is available, injection may not cool the core, but just generate hydrogen. If the containment pressure is high, spraying it may be beneficial. However, it can also de-inert the containment atmosphere and cause hydrogen burns.

Priorities should be set between strategies, because possible strategies can have a different weight and/or effect on safety, and because not all strategies can be carried out at the same time. The basis of the selection of the priorities should be documented. The setting of priorities should include the consideration of support functions (vital auxiliaries such as power and cooling water).

Ad 5. Development of procedures and guidelines.

The strategies should be converted to procedures and guidelines, so as to make them suitable tools for the control room operators and their supportive organisation (Technical Support Centre). Elements of the procedures /guidelines are:

1. objectives and strategies;
2. initiation criteria;
3. the time window within which the actions are to be applied (if relevant);
4. the possible duration of actions;
5. the equipment and resources (e.g. AC, DC, water) required;
6. actions to be carried out;
7. cautions;
8. throttling and termination criteria;
9. monitoring of plant response.

The set of procedures and guidelines should include a logic diagram, which describes a sequence of relevant plant parameters which should be monitored and which are linked to the initiation / throttling / termination criteria of the various procedures and guidelines. The sequence should be in line with the priority of associated procedures and guidelines, as is described before.

Possible positive and negative consequences of proposed strategies should be specified in the guidelines, in cases where the selection of the strategies has to be done in the evolution of the accident. The technical support centre should check whether additional negative consequences are possible, and consider their impact.

Priorities should also be defined among the various procedures and among the various guidelines, in accordance with the priority of the underlying strategies. Conflicts in priorities, if any, should be resolved. The priorities may change in the course of the accident and, hence, the guidelines should call for selection of priorities to be reviewed at regular time intervals. The selection of actions should be changed accordingly.

The EOPs should be interfaced with the SAMGs, and proper transition from EOPs into SAMGs should be provided for, where appropriate. Functions and actions from strategies in the EOPs that have been identified as relevant in the mitigative domain should be identified and retained in the SAMGs.

Guidance should be developed to diagnose equipment failure and to identify methods to restore such failed equipment to service. The guidance should include recommendations on the priorities for restoration actions. In this context the following should be considered:

- the importance of the failed equipment for accident management;
- possibilities to restore the equipment;
- the likelihood of successful recovery if several pieces of equipment are out of service;
- dependence on the number of failed support systems;
- doses to personnel involved in restoration of the equipment.

Ad 6. Hardware provisions for accident management

In principle, each plant should develop SAMG irrespective of its hardware configuration. However, for certain functions, specific hardware may be useful or even needed. A list of such possible hardware is provided. Hardware modifications are always to be considered if it otherwise is not possible to develop a meaningful SAMG-program.
Ad 7. Role of instrumentation and control

The plant parameters needed for preventive accident management measures and mitigative accident management measures should be identified. It should be checked that all these parameters are available from the instrumentation in the plant.

The effect of the environmental conditions on the instrument reading should be estimated and included in the guidance. It should be considered that a local environmental condition can deviate from global conditions and, hence, instrumentation that is qualified under global conditions may not function properly under local conditions. The expected failure mode and resultant instrument indication (e.g. off-scale high, off-scale low, floating) for instrumentation failures in severe accident conditions beyond the design basis should be identified.

Dedicated instrumentation that is qualified for the expected environmental conditions is the preferred method to obtain the necessary information. Where instruments can give information on the accident progression in a non-dedicated way, such possibilities should be investigated and included in the guidance.

Every key instrumentation reading from a non-qualified dedicated instrument that is used for diagnosis or verification should have an alternate method to verify that the primary reading (i.e. the reading from the dedicated instrument) is reasonable. Alternate instrumentation should be identified where the primary instrumentation is not available or not reliable. When an alternate means of obtaining a key parameter value cannot be identified, consideration should be given to upgrading or replacing the instruments in order to provide that alternate indication. Alternatively, other strategies that do not use this instrumentation should be developed.

The ability to infer important plant parameters from local instrumentation or from unconventional means should also be considered. For example, the steam generator level can be inferred from local pressure measurements on the steam line and steam generator blowdown lines.

The need for development of computational aids to get information where parameters are missing or their measurements are unreliable should be identified and appropriate computational aids should be developed accordingly.

Ad 8. Responsibilities and lines of authorization

Transfer of responsibilities and decision making authority from the control room staff to a higher level of authority should be made at some time point in an event that degrades into a severe accident, as decision making is highly complex in view of the uncertainties involved, and because it may involve actions with consequences beyond the information available in the control room or even at the plant. In the mitigative domain, the Technical Support Centre (TSC) should be charged with performing evaluation and recommending recovery actions to the decision-making authority.
This decision-making authority should be with a high level manager, here further denoted as the Emergency Director (ED). The ED holds the authority to decide on the implementation of accident management measures proposed by the technical support centre or, if needed, based on his own deliberation. The ED should have a broad understanding of the actual status of the plant and of other relevant aspects of the emergency response, including off-site effects. If there is any involvement of the regulatory authority in the decision-making, it should be defined how this is to be done.

An example of a typical layout of the main elements of the Emergency Response Organization (ERO) of a plant is shown in Fig. 2.

![Diagram](image)

**FIG. 2. Typical layout of the main elements of the plant Emergency Response Organization**

Criteria for activation of the TSC should be specified, and accident management measures should be carried out by the control room staff until the technical support centre is functional.

The rules for information exchange between the teams of the ERO should be defined. The flow of information between the TSC and the control room as well as from the TSC to other parts of
the ERO, including those responsible for the execution of on-site and off-site emergency plans, should be specified.

The accessibility and habitability of the physical locations of the evaluator and implementer teams as well as of the emergency director under severe accident conditions should be checked and maintained. A widely applied arrangement is that the team of evaluators is located in the technical support centre room, and the team of implementers is in the control room of the plant.

**Ad 9. Verification and validation**

All procedures and guidelines should be validated. Validation is the evaluation that confirms that the actions specified in the procedures and guidelines can be followed by trained staff to manage emergency events.

Possible methods to validate SAMG are the use of a full scope simulator (if available) or an engineering simulator or other plant analyser tool, or a tabletop method. The most appropriate method should be selected. On-site tests should be performed to validate the use of equipment. Scenarios should be developed, and they should describe a number of fairly realistic (complex) situations, which should require the application of major portions of the SAMGs.

Staff involved in the validation of the procedures and guidelines should be different from the staff that developed the procedures and guidelines.

**Ad 10 - 13. Training; new information; analysis; management of AM program development.**

The remaining items discuss the education and training of plant staff, and the processing of new information such as revision of the generic guidelines and new results of research in severe accidents. Comprehensive guidance is formulated regarding the analysis needed and an example presented. Finally, the management of development of the AM program is linked to applicable other IAEA documents.

**6. PRESENT STATUS**

The AM-Guide has been completed and approved by the NUSSC for collecting Member State comments. These have been processed and the Guide will now be considered by final approval by NUSSC and publication.

**7. IAEA SERVICE: REVIEW OF THE ACCIDENT MANAGEMENT PROGRAM (‘RAMP’).**
Apart from existing services such as OSART, IPSART, the IAEA has created a service called ‘Review of the Accident Management Program’ (RAMP) [9] at an NPP.

Objectives of the services can be summarized as follows:
1. to explain to licensee personnel principles and possible approaches in effective implementation of AMP based on experience world-wide;
2. to give opportunities to experts from the host plant to broaden their experience and knowledge in the field;
3. to perform an objective assessment of the status in various phases of AMP implementation, compared with international experience and practices; and
4. to provide the licensee with suggestions and assistance for improvements in various stages of AMP implementation.

The service consists of two parts, one for the analysis and one focussed on the implementation of the accident management program:

Review of Accident Analysis for Accident Management (RAAAM): this review is intended to check completeness and quality of accident analysis covering BDBA and severe accidents. The review should be typically performed prior to use of accident analysis for development of AMP. It is considered that 2 experts and 1 IAEA team leader in one-week mission can perform the review.

Review of Accident Management Programme (RAMP): this review of AMP, in particular appropriate prior to its implementation, is intended to check its quality, consistency and completeness. The review of accident analysis as described in the previous paragraph is a part of the overall review. It is considered that a group of 4 invited experts and one team leader (IAEA staff) during one-week mission will be capable to perform the task. Such composition of the team is sufficient, if detailed review of accident analysis as described in the previous bullet was done separately as a different task, or e.g. within the framework of review of level 1 or level 2 PSA study. If this is not the case, than two more experts should be included in the team to take care of the accident analysis.

At present, two full RAMP and one preparatory-RAMP missions have been completed. The IAEA is in contact with various Member States for further RAMP-missions.

8. CONCLUSIONS.

The IAEA has compiled extensive guidance on the development and implementation of accident management at an NPP. As such, it gives guidance on how to implement the requirements of [NS-R1]. Accident management is a set of actions to prevent a BDBA to escalate into a severe accident and, if not successful, to mitigate the consequences of such accidents.

The development of accident management guidance is a process with a number of dedicated steps in a top-down approach. First the objectives are defined, followed by the development of strategies to achieve these objectives. From the strategies, means are developed to execute the
strategies. Finally, procedures and guidelines are set up, to give detailed guidelines on the actions to be taken on plant equipment.

The process of developing these guidelines is described in some detail. It has 13 elements, as described in sec. 5. It starts with a search for plant vulnerabilities, then it defines the plant capabilities. In defining these latter ones, use is made of all plant systems, also in a non-conventional way and, if needed, outside the design basis of the system. The role of instrumentation is discussed and possible hardware modifications are treated. In developing the guidelines, it is considered that many actions are not unique safety-oriented, but can have negative consequences as well.

As severe accident are complex events, appropriate evaluation and decision-making is required. Mechanisms to achieve this are described, e.g. the support by a Technical Support Centre and decision-making by a high-level authority, mostly the Emergency Director. The AM Guide discusses this item at length, in view of the importance of the issue.

The AM Guide further describes the process of verification and validation of the guidelines. Should new information arise, a process should exist to incorporate it into the guidelines.

Finally, it is described that IAEA has developed the service of Review of Accident Management Program (RAMP) at an NPP. This is a mission of about one week with 4 - 6 experts to a plant, and investigates the existing AM program in detail. Its objective is to further strengthen and improve the AM program. A number of RAMP missions have been performed in last years.

9. REFERENCES.


THE SGTR (STEAM GENERATOR TUBE RUPTURE) LICENSING EVOLUTION AND THE ASSOCIATED MODIFICATIONS FOR NUCLEAR UNITS IN BELGIUM

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ABSTRACT
All over the world, the Steam Generator Tube Rupture (SGTR) event has been a difficult and controversial subject of nuclear plant safety, at least for generation II plants. In some penalizing conditions, this event may lead to a direct radioactive release to the environment, by-passing both the Reactor Coolant System (RCS) and the containment barriers. From the time of the conception of generation II plants, several aspects have evolved. First of all, the event frequency has appeared to be higher than expected, because some SGTR events have really occurred, mainly due to corrosion of Steam Generators' (SG) tubes. Secondly, it became clear that besides the SG overfilling leading to liquid release, other phenomena could be possible, like the atomisation (small droplets of primary water escaping from SG’s) of the break flow in case of low liquid level in the affected SG. Therefore, since the 90’ies, the Belgian Safety Authorities (BSA) requested to improve the situation of the plants in operation, and to provide a safety demonstration taking into account scenarios associated to these newly identified phenomena. After several years of efforts, 2007 has to be recognized as a milestone for the licensing, as an agreement has finally reached between the Utility and the BSA. It consists in adopting different modifications enhancing both the prevention and the mitigation capacity of the plants. Complex and various supporting studies have demonstrated that the radiological impact of a SGTR occurring on any Belgian unit can be significantly reduced once these modifications are carried out. These are concerning Nitrogen-16 chains qualification, installation of control room operated motorized isolation valves upstream of the SG relief valves, reduction of the maximal iodine concentration during operation, reduction of the allowed primary to secondary system leak rate and improvement of the accident procedures. The fact that practically all the original steam generators have been replaced from 1993 until today, and the adoption of new chemical treatment of the secondary circuit (all volatile treatment), are major modifications reducing significantly the SGTR occurrence frequency. Finally, a specific training program helps the operators to react adequately and efficiently in case of any possible SGTR scenario. It must be underlined that this final agreement is the result of a fruitful collaboration between the Utility (Electrabel) and the Architect Engineer (Tractebel Engineering). Many experts have worked on the subject during a given period or even followed the entire project since its beginning.
1 INTRODUCTION

The Steam Generator Tube Rupture (SGTR) event is a design basis accident that must be taken into account for the licensing of PWR’s. This accident is complex, with many particularities distinguishing it from other design basis events. Considering the three major safety functions defined by the IAEA [1] (capability to safely shut down, to remove residual heat and to limit radiological consequences), the SGTR has the potential to challenge each of them.

Firstly, the residual heat removal is a concern because there is a primary inventory loss that could lead to core melting.

Secondly, the control of the radiological release is maybe the most important concern characterizing the SGTR, because even without any core damage, there is a possibility to bypass all the three barriers (fuel, primary circuit, containment). The fuel cladding barrier is never perfectly leakproof and an iodine spiking, due to reactor trip, can lead to a significant activity increase in the primary coolant. The SGTR itself is a breach in the second barrier, while the opening of secondary relief valves allows release to the environment. In 1997 [2], the BSA (Belgian Safety Authorities) were considering the containment by-pass, and particularly the SGTR event, as a key point of future generation III plants.

Thirdly, the capacity to safely shutdown is also a concern recently highlighted again in the licensing of some generation III plants, because when primary and secondary pressures are similar, there is a possibility of back-flow that can cause a boron dilution and a reactivity rise in the core [3].

Even with design improvements that are included in generation III plants, the debates are still in progress.

Another particularity of this event is its relatively high occurrence frequency, since it has happened several times in the past operating experience of PWR’s. Luckily these events have not lead to significant consequences, but this experience led to the observation of new phenomena, and to re-evaluations, in general showing that the SGTR risk was probably underestimated [4]. That explains the international pressure to re-evaluate the SGTR licensing of existing PWR’s.

The present paper will treat the evolution of the international context, and in particular what has been required and achieved in Belgium to improve the safety of generation II PWR’s. It is here supposed that the reader has a general knowledge on the SGTR scenario.

2 DESCRIPTION OF THE SGTR EVENT

The major causes of SG tube break can be the corrosion (different types), the wear, or the presence of a foreign object.

There are many different possibilities of SGTR scenarios, and a distinction should be made between realistic scenarios and scenarios resulting from a deterministic licensing approach using conservative initial and boundary conditions. Hereafter, an attempt is made to make a general description of what can happen after the initiating event, the figure 1 represents the primary and secondary pressure (one of the most important parameters) evolution.

The reactor can be initially at power or even in hot standby state. Regarding the radiological aspect, there is no risk of release to the environment if the primary coolant is perfectly clean and if the fuel cladding remains hermetical during the event. However, the primary coolant activity can be initially at the limit allowed by the technical specifications. In this situation, there is a continuous activity release through small cracks in the fuel rod cladding. This is the first barrier failure.

By definition, the initiating event is a SG tube break, somewhere between the tube plate and the apex.

From this time on, the Chemical and volumetric control system (CVCS) is not able to compensate the break flow, and the primary mass inventory begins to decrease, as well as the pressuriser level and the primary pressure. Regarding the radiological aspects, the initiating event is the second barrier failure.

If the reactor is at nominal power, the feedwater mass flow to the affected SG is controlled to maintain the narrow range level at the value corresponding to the SG’s initial mass. Due to the break flow, the feedwater mass flow to the affected SG is lower than for the other two. With a reactor being initially in hot standby, the feedwater flow is small and the affected SG inventory will increase, this can be an important indication for the operator.
Even without any operator intervention, the reactor is tripped (if initially at power) on the low advanced pressuriser pressure signal. After that, the low low pressuriser pressure signal will start the safety injection.

From that time on, there is a stabilisation of the primary inventory and pressure, high pressure safety injection being able to compensate the break flow. This is a good factor for the core cooling, but not for the contamination of the secondary circuit that continues.

Many different hypotheses can be taken from the time of reactor trip, enhancing either the overfilling of the affected SG (risk of liquid releases), or it’s emptying (risk of by-pass releases). A Loss Of Offsite Power (LOOP) can result from the grid perturbation after reactor trip, stopping primary pumps and normal feedwater pumps.

The turbine is also tripped and in case of unavailable condenser, the secondary pressure will increase until the actuation of a relief device (relief valve or safety valve). Regarding radiological aspect, this is the third barrier failure, and releases to environment begin.

For most plants, operator actions are necessary to reach a safe shutdown state. From the observation of activity in the secondary circuit using Nitrogen-16 (N-16) monitors), or abnormal SG level, the operator will identify the affected SG. The objective is firstly to isolate the affected steam generator (which is done by adjusting relief valve setpoint and closing the main steam isolation valve), and to control the level (which is done by auxiliary feedwater isolation if necessary). Thereafter, the operator cools down the primary circuit, via opening of the intact steam generator relief valves. This cool down is necessary to recover a sufficient subcooling margin to allow to depressurize the primary system without risk of massive boiling. This process requires to stop the safety injection pumps. A controlled state is reached when primary and secondary pressures match, terminating the break flow.

On the long term, the primary and secondary cooldown and depressurization are pursued until connection to the Residual Heat Removal system (RHR), in order to reach a safe shutdown state.

![Figure 1: Typical pressures evolution during SGTR event (starting at Hot Full Power)](image)
3 LICENSING EVOLUTION DUE TO THE US EXPERIENCE

Originally in Belgium, only the aspects of core melting and SG overfilling were investigated for the SGTR accident. Radiological consequences were evaluated considering the partitioning phenomenon, by which a fraction of the iodine in the liquid phase of the affected SG, is carried by the steam. A dilution of the break flow into the affected SG coolant was also assumed. By these two effects, the calculated radiological consequences were not important.

In the 80ies, two SGTR events occurred in the US. They both drew the attention of the NRC (Nuclear Regulatory Commission) and showed that these events were more than hypothetical and could have some unexpected characteristics:

- in 1982, Ginna event, during which an overfilling of the affected SG was observed. During this event, the SG safety valves opened 5 times, and failed to reseat twice. The radiological consequences were not important
- in 1987, North Anna event, with a break located at the top of the tube bundle.

Following this, NRC warned the industry \[5\] for a potential of non-conservatism in SGTR safety analysis for plants with inverted U-tube SG, using the following arguments:

- the North Anna tube rupture demonstrates that SG tube failure near the top of tube bundle cannot be excluded
- the tube uncovery can produce a direct path for fission product release
- for those plants where the steam generator tubes are thought to remain covered following tube rupture, the previously calculated safety analysis offsite dose might be exceeded

In order to better illustrate the situation, the following figure 2 shows the different possible release mechanisms:

![Figure 2: Release Mechanisms](image-url)
Besides the possibility of SG overfilling and liquid releases, figure 2 shows the situation with a low liquid level in the affected SG (uncovered break). The break flow can be divided in three parts:

- the part of the break flow which is flashed (directly transformed into steam due to sudden depressurization)
- a first liquid part mixing with the SG water (liquid)
- a second liquid part composed of very small droplets (atomization) that can be carried out of the affected SG (steam dryer being not efficient for very small droplets)

At the relief device of the affected SG (relief or safety valve), three contributors of releases are possible:

- flashing, carrying the iodine primary activity
- partitioning, which is the affected SG liquid vaporization carrying a fraction (usually one hundreds) of the iodine in the liquid phase
- the by-pass (atomization), carrying the primary iodine activity

The response of the industry to the NRC requirement was an evaluation [6], using a model of primary water by-pass validated on a test facility. Moreover, a stuck-open relief valve was taken into account in the evaluation (not the case in the design basis). The report concluded that less than 1% of the break flow escapes through the open relief valve even when the break is uncovered. The report concluded that the contribution of this scenario to the global risk was negligible, and the design basis evaluations were still valid.

4 BELGIAN PUSGR PROGRAMS

After the SGTR events in US, the BSA reacted in 1989 by reclassifying the SGTR from class IV to class III according to ANSI 18.2, since an occurrence frequency lower than $10^{-2}$ event/year could not be justified anymore. As a result, a lower dose limit had to be respected.

The first Power Uprating and Steam Generator Replacement (PUSGR) in Belgium took place in 1992 for the Doel 3 unit, justifying the complete review of the Final Safety Analysis Report (FSAR). In particular, the SGTR event attracted the attention of the BSA in the safety review due to the following reasons:

- the degradation of the steam generators (addressed further in the document) was partially explaining the decision to launch this project
- the US experience as described before
- the Belgian experience, with the occurrence of an SGTR in Doel 2 in 1979

Therefore, despite the fact that the SG replacement was improving the situation, the requirements of the BSA regarding the new SGTR study were more stringent as compared to the design study:

- SGTR must be evaluated as a class III event, as required since 1989
- a stuck-open SG relief valve must be taken into account as possible single failure
- scenario with by-pass release must be considered

These requirements led to long and difficult discussions during the licensing process. The next PUSGR project in Tihange 1 (1995) had similar problems. Following these two difficult licensing situations, it appeared that the conclusions of the Belgian evaluations were not in accordance with the evaluation for US plants [6] since the scenario’s that enhanced an uncovered break could lead to unacceptable consequences. The poor knowledge of the phenomena and the non adapted tools, were supplementary difficulties for a convincing safety demonstration.

In front of this situation, in agreement with the BSA, it was decided to create a SGTR generic project (i.e. concerning all 7 Belgian units) having as main objectives to deeply investigate and evaluate the consequences of an SGTR, mainly for scenario’s leading to an uncovered break.
5 THE SGTR GENERIC PROJECT

Several tasks were defined in the beginning of the generic project in 1995, and they evolved in parallel. The work performed since then is quite important and described hereafter.

The Iodine-131 (I-131) is the most significant isotope regarding radiological impact. Therefore, it was fundamental to study its behavior during the SGTR calculations, going from the source term in the primary circuit, to the release to the atmosphere. First of all, a model of fission product transport was developed [7] and included into the RELAP5/MOD2 [8] input deck of a typical 3-loop Belgian nuclear unit. It consists in calculating the I-131 concentration in the primary circuit and in each SG, including all the release mechanisms described in figure 2.

The integration of the I-131 activity balance is performed for the 6 following systems starting from the break opening time:

- Primary without pressurizer
- Pressurizer
- Liquid phase of the affected SG
- Liquid phase of the leaking SG
- Liquid phase of the intact SG
- Condenser

For each of these systems, a I-131 balance equation is integrated, taking into account the incoming and outgoing flows with their respective activity, the source terms in the system and the radioactive decay. More details can be found in reference [7].

5.1 Spiking model

The Quantification of the source term of iodine in the primary circuit was a main task of the generic project.

The iodine spiking is a phenomenon during which the primary coolant activity in iodine is rising, following operational transients like scram, important load variation, or primary depressurization. Due to a poor understanding of this phenomenon, the first spiking model coming from the NRC in the 80ies was simple, expressing the fuel release rate (Bq/s) during the spiking as a multiple of the fuel release rate in normal operation: $R_0$.

The drawbacks of this model were the unspecified period of release and the absence of available iodine inventory. Its use was leading to very high and unrealistic activities (Bq/m3) in the primary circuit.

Therefore, for the SGTR generic project, a more efficient model was developed, starting from a sufficient comprehension of the key phenomena, as described hereafter.

Even with a good quality of fuel, fuel cladding cracks are always present. During an experience in which helium was injected in the space pellet-cladding, no iodine release was observed. It is thus supposed that iodine can be expelled out of the fuel if water has first penetrated into the space pellet-cladding, through the existing cracks. In this space, iodine is supposed to be present in the form of a salt which is water soluble, and representing about 1% of the total fuel iodine inventory.

Moreover, a supplementary iodine inventory that originates directly from the pellets, can be released during the spiking:

- Due to the thermal shock between entering water and the pellets, that generates cracks in the pellets, forming a path for the iodine to escape
- Due to the decrease of pellets conductivity after oxidation with water, that generates higher temperatures in the pellets, enhancing iodine diffusion

This second contributor of iodine inventory is more important in case of fuel damage. After a spiking, about 3 days of irradiation at full power are needed to recover half of the iodine inventory which is present in the space pellet-cladding.
Based on these observations, the new model can be build.

In a first step, an iodine Inventory (Bq) that can be released in the primary during the few hours after spiking initiation, is calculated, as function of $R_0$. It has been verified that this model of inventory covers all the spiking phenomena that have been observed in the Belgian units during a long operation period.

$R_0$ is easily obtained because it is in equilibrium with the filtration rate of the CVCS system, imposing a given primary coolant activity. For a conservative evaluation in the SGTR analysis, this activity can be just at the limit authorized by the technical specifications at stable conditions (see later explanations on these limits), due to a pre-spiking occurring before the accident.

In a second step, the spiking itself is initiated at scram, releasing the remaining inventory (because not enough time to recover the total inventory), by the following relation:

$$R(t) = \frac{\text{Inventory}}{T} \cdot e^{-\frac{t}{T}} \quad (1)$$

The acceptance of this model by the BSA was an important milestone of the project.

5.2 Atomization model

By lack of experimental data, the following simple and conservative model has been developed in accordance with the BSA. From figure 2, the fraction of the liquid break flow which is atomized and bypasses the dryers in the form of small droplets (carrying coolant activity out of the affected SG) depends on 2 thresholds:

- If the affected SG is not stratified (sufficient recirculation), which is considered to be the case before the scram, the by-pass fraction is 0.0001
- If the affected SG is stratified (which is the case after scram), the NLH (Net Liquid Height) value has to be calculated and represents the collapsed liquid level above the break location. If NLH is greater than 2 feet with uncertainties, the break is covered and the by-pass fraction is 0.01. Otherwise, the by-pass fraction is 0.3.

That explains the reason why the worst break location for scenarios enhancing the by-pass of the primary coolant, is at the top of the tube bundle.

5.3 Operator intervention timing model

As explained in figure 1, operator actions are necessary to reach a safe shutdown state following a SGTR event. For scenarios with uncovered break, the release is terminated when primary cool-down is initiated (except for a single failure with relief valve stuck open), because the affected SG pressure decreases and its relief valve closes. For the overfilling scenario, the primary-secondary pressure balance must be reached to terminate the break flow.

In order to defend a reasonably conservative timing, a new method has been developed, discussed and accepted by the BSA, specifically in the frame of SGTR generic project.

The method consists in classifying all operator actions in five different types, each with its specific duration: opening of procedure, simple verification, verification with a judgment, simple action, complex action. This method requires characterizing the type of each operation action in one of the five categories that are proposed. For some of them, this interpretation can be discussed (especially when making the difference between simple and complex actions).

In a transient simulation, not all the operator’s actions are necessarily simulated. Nevertheless, the delay and duration associated to these operators' actions have to be characterized and taken into account in order to obtain a realistic “time table”.

A first remark is that the procedures are specific for each unit (number and nature of steps to follow), and the “time table” is thus also specific. Unless a bounding approach is adopted, the plant specific accident procedures are thus directly linked with the safety assessment.

A second remark is the technical differences between plants: a simple action in one plant can correspond to a complex action in another plant.
An assessment of the proposed method has been made, performing different comparisons with simulator tests: WCAP simulator test results [9] using the standard Emergency Response Guidelines (ERG) procedures, simulator test results for Tihange 3 using the specific Tihange 3 E-0 and E-3 procedures, simulator test results for Doel 4 using the specific Doel 4 E-0 and E-3 procedures.

It appeared that the method covers the majority of the operator teams without being too conservative. The following example concerns some important actions that are simulated in the calculations for the overfilling scenario initiated at nominal power. Conservatively, the first evident sign for the operator that something is going wrong is the scram. The following events then take place:

- after 12 minutes, identification of the affected steam generator and going from E-0 to E-3 of the ERGs. Adjustment of the affected SG atmospheric relief valve setpoint.
- after 14 minutes, closing MSIV and by-pass valves
- after 16 minutes, checking the intact SG’s level and keeping the level in a given range by a “switch on – switch off” feedwater system
- after 21 minutes, start of the Reactor Cooling System (RCS) cool-down by opening the intact SG relief valves.

The duration of the cool-down (about 15 minutes) depends on the RELAP model behavior, and is close to what is observed on simulator. The same occurs for the depressurization phase (about 10 minutes). From the scram to break flow inversion, the total duration is about 45 minutes.

5.4 Main results of generic evaluations

With all the detailed models described before, and with a RELAP input deck representing a typical 3-loops plant, calculations have been performed for three main scenarios that are summarized below.

**Low water inventory and by-pass release**

The reactor is initially at power. A break is simulated at the top of the tube bundle. Different possibilities of single failure have been examined and the most significant is the relief valve of the affected SG which remains stuck open after scram. The condenser is assumed unavailable. As aggravating failure, a LOOP is postulated at scram, leading to the coast-down of the primary pumps and of the SG feedwater pumps. It is the only scenario for which the radiological consequences are calculated.

Next figure 3 shows the importance of each contributor in the total release:

![Figure 3: contributors (% of total release)](image)

**Figure 3:** contributors (% of total release)
The absolute value of release was slightly higher than the limits, and moreover, could only be terminated if the operator had the opportunity to isolate the stuck open relief valve. This result demonstrated the non-applicability of the US evaluations [6] for Belgian licensing:

- due to different parameters influencing the results, as the smaller SG tube size, and the more important SG relief valve capacity. These differences completely change the affected SG mass balance and enhance the capability of SG emptying, increasing the risk of releases by atomization

- due to the stringent atomization model which is described in §5.2

It appeared that the difficulties encountered during PUSGR projects were confirmed, even using the very detailed models that were developed during SGTR generic project.

Anyway, this result was consistent with the European benchmark exercise realized in 1999 [4], concluding that bypass release mechanism could be an important contributor to risk, and that the prevailing licensing methodologies may in some cases be non-conservative.

**Overfilling scenarios**

The event was analyzed starting with reactor initially at full power, or at hot shutdown. A break is simulated at the base of the tube plate in order to maximize the break flow. SG level measurement error (for case at full power) and SG designer liquid inventory uncertainty, are added to enhance overfilling. For the case at power, the regulation of normal feedwater is considered as blocked at the initial position.

In both cases, an overflow of several tens of tons of liquid out of the affected SG was predicted. The major contributor of the affected SG overfilling is the break flow, demonstrating that the rapidity of the operator action is a key point.

Again for this scenario, difficulties were encountered, even using the operator intervention timing model described in §5.3.
6 FINAL AGREEMENT WITH BSA, SUPPORTED BY IMPROVEMENTS OF BELGIAN PWR UNITS

In 2007, a demonstration was performed in front of the BSA, regarding the reduction of the SGTR risk from the beginning of the SGTR generic project. Improvements concern both the prevention and the mitigation of the consequences. Some of the associated modifications are already performed, and others are planned on the short term. These are described hereafter.

6.1 Prevention improvements due to new steam generators, new chemical treatments and FME procedures

As mentioned in §4, despite the SG replacements in Belgium that improved gradually the situation, the requirements of the BSA regarding the new SGTR study were more stringent compared to the design study. This can be explained by the fact that there was no sufficient operation feedback and no sufficient reason to be more confident in new SG’s.

The NRC has identified 10 SGTR events, all of them happened between 1975 to 1993, among them one in Belgium: Doel 2 in 1979. A remarkable point is that all these events happened in plants using SG’s in Inconel 600 MA (Mill annealed).

In 2006, 10 years after the start of SGTR generic project, 13 years after the first SG replacement in Belgium, and 18 years after reclassification of SGTR event by the BSA, a new evaluation was made in the frame of the SGTR generic project. This evaluation was based on Belgian and international feedback and showed that the situation was significantly improved:

- the behavior of Inconel 690 or Incoloy 800 for new SG’s appeared to be much better than the Inconel 600 regarding different types of corrosion phenomena
- the secondary circuit chemical treatment (all volatile treatment) was better than old treatment methods that led to corrosion
- procedures of Foreign Material Exclusion (FMA) applied for primary and secondary circuit, significantly reduced the probability of SGTR due to foreign object

For plants using SG’s in Inconel 690 or Incoloy 800, the evaluation made counts about 1000 years of operation without any problem. Using a binomial distribution, it was demonstrated that:

- the frequency of 7 SGTR events per 10000 reactor-years is the most likely one
- there is 95% probability to have a frequency smaller than 3 SGTR events by 1000 reactor-years

Even considering that tomorrow, an SGTR event happens somewhere, the same evaluation shows slightly different results. In this case, there is more than 95% probability that the real SGTR frequency would be smaller than 5 events per 1000 reactor-years.

The situation is summarized in figure 4 below. Using the ANSI 18.2 classification as it is the case in Belgium, with the current conditions of operation (new types of SG, chemical treatment, FME procedures…) the SGTR event can be considered as a condition IV event. Even if the reclassification from condition IV to condition III that occurred in 1989 was justified, this is not the case anymore.
Without any SGTR event observed during 1000 years of operation

With one single SGTR observed during 1000 years of operation

Condition I and II

Condition III

0.003 (95 % confidence)

0.0007 (best-estimate)

Condition IV

0.005 (95 % confidence)

0.002 (best-estimate)

Figure 4 : SGTR classification following ANSI 18.2

As already mentioned; the first SG replacement in Belgium occurred 1992 (Doel 3). Since then, most SG's have been replaced on the Belgian units. Only Doel 1 is still equipped with its genuine SG’s (their replacement is planned for 2009). So far, no SG tube has been plugged in any new SG.

Since the beginning of the SGTR generic project, this is a significant improvement regarding prevention.

6.2 Prevention improvements regarding treatment of SG leakage

The current technical specifications allow a maximum primary to secondary leakage of 80 kg/h for each SG during power operation. The allowed leakage will be reduced to 24 kg/h in order to limit the likelihood of propagation of flaws to SG tube rupture. The alarm set-points on N-16 radiation monitors will also be reduced.

The training program for the licensed operators will be adapted in order to bring the proper response to an SG leakage.

6.3 Mitigation of consequences : qualification of the SG tube leakage monitoring

In §5.4 dealing with evaluations performed, it is supposed that the operator has an efficient indication to detect the accident and to identify the affected SG.

The continuous on-line N-16 radiation monitors provide for rapid detection and response to leakage. They allow the operator:
- to identify a SG tube leakage at an early stage and to take appropriate actions,
- to identify which SG is affected by a tube rupture and has to be isolated, according to the incident or accident response guidelines.

The N-16 chains are not effective at low power. Nevertheless, if a SGTR occurs at high power, the N-16 detection significantly reduces the overall operator response times when applying the accident procedures E-0 and E-3. The generic studies have shown that a shorter operator response time leads to the significant reduction of the radiological impact of the SGTR.

On the Doel units, the N-16 radiation monitors are located in safety-grade cabinets and receive a safety-grade power supply. Their output signals are sent to a safety-grade recorder in the control room.

On the Tihange units, the N-16 radiations monitors are not powered by a safety-grade power supply. Their output trigger an alarm in the control room but are not recorded.
The qualification level of the N-16 monitors will be improved on all units so as to ensure that their outputs are memorized and can be relied upon to ease the identification of a SGTR when applying E-0 and E-3 procedures.

Firstly, the Utility has charged the Architect Engineer to define for all units the required modifications to be carried out to reach the desired qualification level of N-16 radiation monitors. The modifications will be performed in a second phase currently in preparation.

6.4 Mitigation of consequences: remote controlled isolation of SG relief valve

As explained in §5.4 dealing with evaluations performed, the single failure applied on a stuck open relief valve led to unacceptable consequences, even considering an efficient isolation by the operator. However, this isolation could not be ensured on all Belgian plants.

As a consequence, on all Belgian plants, the SG relief valves will be provided with a qualified motor-operated isolation valve, capable of being remotely operated.

The electrical power supply to the isolation valve will be safety grade. For plants with a bunker, second level electrical power is also acceptable.

The remote controlled SG isolation valves will be subject to new technical specifications that can be added to the already existing specifications about the SG relief valves.

Some plants (e.g. Doel 4) already have the required isolation valve, but without the remote commands located in the main control room. Sending an operator to the bunker in order to perform isolation of a stuck-open relief valve takes about 1 minute, which is acceptable.

This modification has been carried out in 2006 on Tihange 2 and 3.

All Belgian units are thus today consistent with the SGTR generic evaluations.

6.5 Mitigation of consequences: improvement of emergency response guidelines and licensed operator training

The following modifications have been introduced in ERG procedures in Doel and Tihange. At an early stage in the procedures, the operator is explicitly asked to pay attention to the narrow range level and feed-water flow of the affected SG. Early contingency actions are asked should, for some reason, the SG level be out of control:

- isolation of Main Feedwater (MFW), Auxiliary Feedwater (AFW) and Emergency Feedwater (EFW) in case of uncontrolled SG overfilling,
- isolation of the SG relief valve in case of uncontrolled SG draining by a stuck-open SG Relief Valve.

Specifically for the Doel units:

- the operator is asked to use the (recorded) N-16 indications to identify a SGTR,
- the subcooling target at the end of the cooldown phase has been lowered from 30°C to 20°C so as to reduce the delay needed to resume the subcooling target and bring the RCS pressure to the affected SG pressure.

On the Tihange units, the use of the N-16 indications to identify a SGTR will be added to the procedures once these indications are adequately qualified.

Training program for licensed operators will be adapted in order to (continuously) clarify their knowledge and understanding of:

- the physical phenomena that influence the radiological impact of a SGTR,
- the purpose and impact of above mentioned system and technical specifications modifications,
- the proper response according to each leakage or SGTR possible situation,
- the usage of incident procedures as well as emergency response guidelines.

As the Hot Zero Power (HZP) scenario of the overfilling (in §5.4) was also sensitive, the future simulator training will also focus on the HZP case and not only on the Hot Full Power (HFP) case.
6.6 Mitigation of consequences: reduction of the technical specifications limits on primary specific activity

The technical specifications limit the allowed primary activity, for both steady state (equilibrium) and transient situations. These limits are expressed as the specific activity of the isotope I-131.

The Utility intends to reduce the radiological impact of a SGTR by adopting for all Belgian units reduced and uniform activity limits:

Table 1

<table>
<thead>
<tr>
<th>Unit</th>
<th>Current TS activity limit</th>
<th>New TS activity limit</th>
</tr>
</thead>
<tbody>
<tr>
<td>Doel 1/2</td>
<td>0.65 / 3.75</td>
<td>unchanged</td>
</tr>
<tr>
<td>Doel 3</td>
<td>9.11 / 253.85</td>
<td>1.80 / 18.00</td>
</tr>
<tr>
<td>Doel 4</td>
<td>2.80 / 28.46</td>
<td>1.80 / 18.00</td>
</tr>
<tr>
<td>Tihange 1</td>
<td>14.23 / 40.38</td>
<td>1.80 / 18.00</td>
</tr>
<tr>
<td>Tihange 2</td>
<td>7.69 / 40.00</td>
<td>1.80 / 18.00</td>
</tr>
<tr>
<td>Tihange 3</td>
<td>4.46 / 38.46</td>
<td>1.80 / 18.00</td>
</tr>
</tbody>
</table>

The activity limits of Doel 1/2 have already been significantly reduced after SG replacement carried out on Doel 2. The main reason was the non-applicability of the SGTR generic evaluations for a 2 loop plant (see §5.4).

For all the units, it was demonstrated to the BSA that the radiological consequences of the scenarios examined in §5.4, were reduced quasi proportionally.

The new technical specifications will progressively be adopted by the Utility.

7 CONCLUSIONS

The licensing of the SGTR event has evolved from the conception of generation II plants until today, both due to the occurrence of the event, and to the consideration of new phenomena. In Belgium, difficulties have led to the creation of the SGTR generic project in 1995, in which different tasks were defined in order to deeply investigate the subject. Using the RELAP code, a model of fission product transport was specifically developed. Moreover, precise models were also developed for the spiking, the atomization, and the timing of operator intervention. The last step of this work was a final evaluation using all these models.

For the scenarios enhancing SG overfilling, with the entire conservative initial and boundary conditions, it appeared that the affected SG overfilling is very difficult to avoid, particularly in HZP. However, this situation is less frequent than HFP.

For the scenarios enhancing low water inventory and bypass release, there remains an important uncertainty on the atomization model that was very conservative for the evaluations.

The radiological consequences of all scenarios are limited with the adoption of new technical specifications limits on primary specific activity. This modification is an important challenge for the Utility, which is consistent with the objectives of the Institute of Nuclear Power Operations (INPO) regarding the fuel integrity: “0 failures by 2010”.

During the last 15 years, important efforts were made in the frame of SGTR generic project, in various areas regarding studies and modifications on site, concerning both the Utility and the Architect Engineer, working together to find a solution.

The final agreement with the BSA in 2007 is the demonstration of the success of this work.
8 REFERENCES


[2] “TSO study project on development of a common safety approach in EU countries for large evolutionary PWR’s” : ICONE5-2160 (Nice, 1997)


NPP Safety In Ukraine: Recent Developments and Trend

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ABSTRACT

Significance of nuclear power industry in the Ukrainian fuel and energy complex continually increases. Nuclear energy development program has envisages construction of more than 10 new power facilities during next 20 years. It should be noted that for new nuclear power plants (NPP) more strict requirements for safety (comparing to existent NPPs) are established. The safety has to be ensured by means of NPPs proper siting, design, construction and commissioning, followed by the proper management and operation of the plant.

International and national researches show that the design and operation of existent NPPs meet internationally approved safety principles. However, understanding significance of safety, existent plants promote activities to meet safety requirements for new NPPs. These activities include (but are not limited) such directions, as periodic safety review and life extension/management for older plants, safety upgrade program (for all plants), continuous enhancement of operational safety and implementation of quality assurance system. This paper presents an overview of current trend in the development of these directions.

In particular, in order to meet national requirements, both the Utility and Regulatory Authority undertake activities on NPP modernization using both deterministic and probabilistic justifications. Adopted approach allows increasing safety by more effective use of efforts and means for elimination of safety deficiencies; increase regulatory efficiency; and reduce licensee undue burden while maintaining required safety level. This paper summarizes the main insights and recent results of realization of Safety Upgrade Program on NPPs in Ukraine.

1 INTRODUCTION

Currently four nuclear power plants (NPP) with 15 nuclear reactors of WWER type produce more than 50% electricity in Ukraine. The initial design of these plants and their safety has been continuously improved over the years by taking into account the feedback from operational experience. It can be considered that the safety of these NPPs designed to earlier standards is sufficient. Furthermore, the Utility promotes activities to meet more strict safety requirements for future plants. However, the plants (both future and existing NPPs) must meet a difficult challenge: to be both safe and economically competitive. The way to achieve competitiveness includes (a) enhancement of safety by modification of design / operation and (b) application of risk-informed techniques in decision making process of both the Utility and Regulatory Authority. This paper deals with trends and issues in the development of these directions in Ukraine.
2 SAFETY OF NUCLEAR POWER PLANTS

International and national researches (specifically, safety analysis reports for each type of WWER in Ukraine) show that the design and operation of existent NPPs meet internationally approved safety principles. However, understanding significance of safety, the Utility intends to ensure that operating NPPs maintain safety level comparable to the plants designed and constructed today. To attain this goal, Utility needs, among other aspects, a stable regulatory system that should not be excessively conservative. In order to achieve a stable regulatory system, new regulatory rules must be carefully considered. The rules should be such that Utilities can use their resources for those improvements which are essential to safety rather than on items which are less important, and only marginal to safety. Therefore, the rules should quantitatively define the safety goals. Top level regulatory document, [1], renewed in 2008, numerically defines the safety goals for operating and future NPPs. These goals (as well as the whole document) are consistent with the IAEA safety standards. As a most effective tool in evaluating and comparing NPP safety against safety goal, probabilistic approach is used. Activities in this field are described in Section 3.

To meet safety requirements for future NPPs, the Utility undertakes the efforts, which include (but are not limited) realization of measures developed under periodic safety reviews, implementation of safety upgrade program, and continuous enhancement of operational safety.

2.1 Periodic Safety Review

Regulatory documents, Refs. [1] and [2] require that each power unit of each NPP must perform periodic safety review (PSR) with periodicity of 10 years. The main purposes are to evaluate current level of safety, and to assess whether NPP designed to earlier standards can maintain an adequate safety level, to satisfy operating safety standards and rules. Moreover, it is necessary to conduct PSR to clearly identify any deficiencies in defence in depth, followed by a modernization program so that all barriers can be timely reinforced.

The Utility has developed guide (see Ref. [3]) on contents and structure of report on periodic safety review. At the first stage, Ref. [3] provides requirements for older units, commissioned at 1980-1982 years (Rivne NPP Units 1 and 2; South Ukraine NPP Unit 1). Taking into account experience from conducting PSR for these three units, PSR requirements for other NPPs will be refined.

PSR report should provide real status and prognosis for each safety factor for a period to the next PSR or to the expiration of NPP life time. List of evaluated safety factors should include the following: (1) design; (2) current technical condition of systems and components; (3) equipment qualification; (4) aging; (5) analysis of internal and external hazards; (6) deterministic safety assessment; (7) probabilistic safety assessment; (8) operational safety; (9) management and control; (10) operational documentation; (11) human factors; (12) experience feedback from other plants of similar design and scientific researches; (13) emergency preparedness; and (14) environment impacts. One of the important results from PSR will be NPP modification program that includes measures which are necessary for compliance with more modern regulatory requirements and international good practices; and aging management measures. The measures should be developed taking into account for operating experience, maintenance history, aging, and severe accident analysis. Both deterministic and probabilistic approaches should be properly combined to achieve a more balanced modification program.

Probabilistic safety analyses (PSA) have become an important part of PSR. From one hand, during PSR should be determined to what extent the existing PSA remains valid as a representative model of the plant when the following aspects have been taken into account:
changes in the design and operation of the plant; new technical information; current methods; and new operational data. From other hand, according to the extent of PSA (i.e., completeness), PSAs can be used to evaluate proposed (under PSR) modifications. It is understood that to keep the PSA as an effective tool, it has to be maintained in a living status in the sense that it should be updated to reflect plant conditions at any given time.

According to regulatory document [2] and guide [3], the first PSRs in Ukraine will be used to support justifications of lifetime extensions for RNPP Units 1 and 2 and SUNPP Unit 1. Currently, there is no international experience (or experience is very limited) in conducting such type PSR (up-to-now PSRs were applied mostly for recovery of design basis, or for licence renewal). So, it is believed that performing PSRs for older units in Ukraine will provide a good experience and valuable results for justification of lifetime extension and for further development and enhancement of safety not only for Ukraine, but for all countries where lifetime extension issue becomes actual.

2.2 Safety Upgrade Program

In 2005 the Ukrainian Government has approved the Complex Program of the NPP modification and safety improvement, Ref. [4]. The Program includes safety enhancement areas and associated safety measures developed using information on risk from safety analyses, considering qualitative and quantitative outcomes from PSA. The main goal is to further enhance safety of the Ukrainian NPPs to account for enforcements in regulations and best current practices in rational and the most efficient way.

Each safety enhancement area contains a number of safety measures for each type of WWER reactors. PSA informed areas are presented below. In addition to PSA-informed areas, Safety Upgrade Program includes areas identified based on international recommendations/advanced international experience, deterministic safety analysis etc. Table 1 shows total number of areas/measures for each type of reactors operating in Ukraine.

Table 1 Safety upgrade areas

<table>
<thead>
<tr>
<th>SAFETY UPGRADE AREAS</th>
<th>PSA insights</th>
<th>PSA insights</th>
<th>IAEA recommendations</th>
<th>PSA insights</th>
<th>IAEA recommendations</th>
<th>PSA insights</th>
<th>IAEA recommendations</th>
<th>Regulatory requirements</th>
</tr>
</thead>
<tbody>
<tr>
<td>Area 1: LOCA from primary to secondary side</td>
<td>11</td>
<td>7</td>
<td>5</td>
<td>6</td>
<td>10</td>
<td>5</td>
<td>4</td>
<td>13</td>
</tr>
<tr>
<td>Area 2: Dependent and common cause failures</td>
<td>12</td>
<td>9</td>
<td>3</td>
<td>6</td>
<td>12</td>
<td>4</td>
<td>4</td>
<td>13</td>
</tr>
<tr>
<td>Area 3: Secondary heat removal</td>
<td>10</td>
<td>4</td>
<td>6</td>
<td>7</td>
<td>7</td>
<td>5</td>
<td>6</td>
<td>13</td>
</tr>
</tbody>
</table>

Table 1 depicts 9 safety upgrade areas covered in Ref. [4]. The brief description of PSA informed areas is provided below.
2.2.1 PSA informed areas

Area 1 “Medium LOCA from primary to secondary side”. This initiating event is the most complex for control and has a most impact into the CDF (and large release frequency, LRF) values. This is because of IE diagnostics is difficult, emergency control is complicated for personnel, time available for decision making and actions performing is limited with coolant inventory for make-up, probability of steam dump valves (SDV) on affected steam generator (SG) non-closing after opening is high (taking into account for SDVs presence, which are not qualified for operation on a steam-water mixture), and as a consequence containment bypass, irretrievable loss of coolant and radioactive materials release into the environment are possible. Safety issue realization implies the complex of tasks including: (a) preventive measures which decrease the probability of primary to secondary circuit leakage occurrence (use of 100% non-destructive monitoring of SG manifold metal and welded splices, implementation of adequate water-chemical mode, etc) and constrain the leakage rate; (b) organization measures directed on the emergency preparedness by the way of: revision of the emergency procedures aimed on leakage localization within the affected SG, prevention of containment bypass and excluding or minimization of the radioactive release into environment under leakage; and personnel training using the full-scale simulator; (c) plant modifications directed on expansion of safety systems capabilities to overcome given accident and to facilitate the personnel tasks on emergency control.

Area 2 “Dependent and common cause failures”. Measures of this area are directed on reducing multiple malfunctions of equipment due to dependent failures and common cause, such as: (a) spatial interactions (steaming, spraying, piping whip, steam and water jet impingement) resulted from high energy line breaks; (b) internal flooding and fires; (c) blasted insulation due to LOCA inside containment.

Area 3 “Secondary heat removal”. For this safety function the following safety deficiencies were identified: (a) SDVs availability is not ensured under the steam-water mixture outflow; (b) some of existing components may be used only for limited number of accident modes, that significantly decrease redundancy and consequently the safety function reliability. Safety analyses showed, for example, that for ensuring SG feeding from the auxiliary feedwater pumps the intersystem dependencies are not balanced and have not enough redundancy.

Area 5 “Primary heat removal and pressure control”. This area is directed on implementation of primary heat removal in feed and bleed mode. This method is well known and effective mode which applies to prevent the reactor core damage under emergency situations. List of measures includes: changes in systems design and construction, modifications of operational and emergency procedures. List of modified systems includes pressurizer pilot operated relief valves, high pressure and low pressure injection systems, instrumentation and control.

Area 7 “Emergency power supply”. Measures under this area are associated with total station blackout. Under total blackout core damage will occur in few hours. If the external grid is restored during this time interval, the potential exists to prevent the core damage, if the NPP will be timely switched on to the restored grid. Availability of batteries is the important condition for providing that after external power supply recovery the front-line systems may be powered. In a case of batteries complete exhausting, switching on to the external grid becomes more complicated, while this possibility still remains under condition if batteries of open switchyard are available.
2.2.2 Safety areas ranking

It was decided that implementation of safety measures will be performed by step-by-step procedure, taking into account their safety efficiency. Currently, the Utility activities in this field are directed on prioritisation of safety measures and areas using probabilistic safety assessments. Brief description of risk-informed techniques in decision making process is provided in Section 3.

Regulatory guide (Ref. [5]) establishes methods for prioritization, qualitative and quantitative ranking criteria, as well as, requirements for PSA technical quality for such application. Quantitative ranking criteria according to regulatory guideline are shown on Table 2.

Table 2 Ranking criteria for safety measures/issues

<table>
<thead>
<tr>
<th>Category name</th>
<th>Criterion, $\Delta CDF_i$ 1/year</th>
<th>Criterion, $\Delta LRF_i$ 1/year</th>
</tr>
</thead>
<tbody>
<tr>
<td>Category 0: Insignificant influence on the NPP safety level.</td>
<td>&lt;1E-07</td>
<td>&lt;1E-08</td>
</tr>
<tr>
<td>Category I: Low influence (importance) on the plant safety</td>
<td>&lt;1E-06</td>
<td>&lt;1E-07</td>
</tr>
<tr>
<td>Category II: Medium influence (importance) on the plant safety</td>
<td>&lt;1E-5</td>
<td>&lt;1E-6</td>
</tr>
<tr>
<td>Category III: High influence (importance) on the plant safety</td>
<td>&lt;1E-4</td>
<td>&lt;1E-5</td>
</tr>
<tr>
<td>Category IV: Very high influence (importance) on the plant safety</td>
<td>&gt;1E-04</td>
<td>&gt;1E-05</td>
</tr>
</tbody>
</table>

Prioritisation of safety areas/measures for new units (KhNPP Unit 2 and RNPP Unit 4) is near completion, while for ZNPP Unit 5, SUNPP Unit 1 and RNPP Unit 1 prioritization process is underway. Preliminary ranking of the safety areas for different reactors, and change in CDF, based on probabilistic analyses, are shown in Table 3.

Table 3 Preliminary ranking of the safety upgrade areas

<table>
<thead>
<tr>
<th>Safety area</th>
<th>WWER-1000/302</th>
<th>WWER-1000/320</th>
</tr>
</thead>
<tbody>
<tr>
<td>Rank</td>
<td>Potential to CDF decrease, $\Delta CDF_i$</td>
<td>Rank</td>
</tr>
<tr>
<td>------</td>
<td>------------------------------------------</td>
<td>------</td>
</tr>
<tr>
<td>Area 1 &quot;LOCA from primary to secondary side&quot;</td>
<td>III</td>
<td>2.4E-05</td>
</tr>
<tr>
<td>Area 2a &quot;Dependent and common cause failures - ECCS sump issue&quot;</td>
<td>II</td>
<td>7.5E-06</td>
</tr>
<tr>
<td>Area 2b &quot;Dependent and common cause failures - Control systems issue&quot;</td>
<td>I</td>
<td>2.5E-07</td>
</tr>
<tr>
<td>Safety area</td>
<td>WWER-1000/302</td>
<td>WWER-1000/320</td>
</tr>
<tr>
<td>------------------------------------------------</td>
<td>---------------</td>
<td>---------------</td>
</tr>
<tr>
<td>Area 2c &quot;Dependent and common cause failures - Spatial interactions issue&quot;</td>
<td>III</td>
<td>3.2E-05</td>
</tr>
<tr>
<td>Area 3 “Secondary heat removal”</td>
<td>II</td>
<td>3.6E-06</td>
</tr>
<tr>
<td>Area 4 &quot;Cold overpressure&quot;</td>
<td>I</td>
<td>1.0E-07</td>
</tr>
<tr>
<td>Area 5 “Primary heat removal and pressure control”</td>
<td>II</td>
<td>3.6E-06</td>
</tr>
<tr>
<td>Area 6 &quot;Containment reliability&quot;</td>
<td>II</td>
<td>N/A</td>
</tr>
<tr>
<td>Area 7 “Emergency power supply”</td>
<td>I</td>
<td>4.2E-07</td>
</tr>
<tr>
<td>Area 8 &quot;EOI and emergency preparedness&quot;</td>
<td>III</td>
<td>1.0E-05</td>
</tr>
<tr>
<td>Area 9 &quot;Safety Analyses&quot;</td>
<td>N/A</td>
<td>N/A</td>
</tr>
</tbody>
</table>

Figure 1 illustrates tentative decrease in total CDF as result of fulfilment of measures under safety upgrade program.

Figure 1 CDF due to realization of safety upgrade program

Industry activities on realization of safety upgrade program and continuous enhancement of operational safety will further decrease CDF estimates.

## 3 RISK-INFORMED TECHNIQUES IN DECISION MAKING PROCESS (RIDM)

To ensure that NPPs are both safe and economically competitive, the Utility and Regulatory Authority take efforts on implementation of risk-informed techniques in decision making process. Activities on RIDM are covered by Policy decisions, Ref. [6], and Implementation Plan, that was approved by Utility and Regulatory Authority in 2003. Adopted approach allows increasing safety by more effective use of efforts and means for
elimination of safety deficiencies; increase regulatory efficiency; and reduce licensee undue burden while maintaining required safety level.

It was found that balanced use of PSA technique and methods to improve NPP safety jointly with the deterministic approaches supports making of well-founded regulatory decisions on NPP safety. However, whether risk informed regulation is of benefit to Utilities depends to a large extent on the common understanding developed with the Regulatory Authorities. Since PSA development imposes a considerable burden, in terms of the human and financial resources that need to be expended, it is very important to clearly define what is expected from the Utility and how the results will be used. The counter parties try to develop common understanding through wide discussion of RIDM and development of a suitable regulatory framework. To achieve common understanding, a discussion framework was established through interagency coordination board. The board is advisory organ on RIDM for both Regulatory Authority and Utility. The coordination board consists of senior engineers and managers representing Regulatory authority and their technical support organizations, the Utility, NPPs and design organizations. Place of the board in organizational structure is illustrated on Figure 2.

![Figure 2: RIDM Organizational structure](image)

The following responsibilities are imposed to the coordination board: General organizational oversight and coordination of works envisaged by the Implementation Plan; wide discussion and development of recommendations in order to: (a) resolve actual technical issues; (b) form a common industry policy in the area of RIDM; (c) analysis and evaluation of experience, updating of plans and schedules, upgrading the Implementation Plan. The first results of coordination board meetings have involved wide discussion and approval of general regulatory document on risk-informed regulation, see [7]; introduction of new projects on PSA application; rescheduling of pilot projects, etc.

Top level regulatory document, [1] numerically defines the safety goals (core damage frequency and large release frequency) for operating and future NPPs. Based on these targets, general regulatory document, Ref. [7] proposes risk acceptance criteria for RIDM. The criteria are stated in terms of risk metric (CDF; LRF) and change in risk metric due to plant upgrade. Depending on relation between base risk metric and change in risk metric, regulatory body makes decision on allowance of plant upgrading and necessity for compensatory measures. For example, if base CDF is less than 1.0E-04, the plant upgrade can...
be permitted, on conditions that: change in CDF is small; no degradation of defence in depth; and compensatory measures are considered. In the RIDM context, the criteria of [4] should be interpreted as targets. They are intended to provide an indication, in numerical terms, of proposed changes acceptability. Criteria are intended for comparison with a full scope (including internal and external events, full power, low power and shutdown) assessment of the risk metric. Illustration of criteria is shown on Figure 3.

Figure 3: Risk Acceptance criteria

The plant upgrade can be permitted if base CDF and LRF are consistent with the safety goals, on conditions that: changes in CDF and LRF are satisfactory in numerical terms; and compliance with such engineering safety principles is ensured as (a) consistency with the defence in depth philosophy should be maintained; (b) sufficient safety margins should be ensured; (c) impact of plant modification on safety should be monitored; and other.

To ensure PSAs of high quality to support RIDM, from one hand, the Utility involves high-professional engineering support organizations to develop internal guidelines on PSA and prepare full-scope PSA. From other hand, the Regulatory Authority develops regulatory requirements to PSA and PSA applications and provides regulatory review services for associated reports developed by the Utility. Recent regulatory requirements include guideline on Living PSA, guideline on PSA Level 2, guideline on evaluation of safety measure/issues, and other. The development of procedures and technical guidelines assumes using advanced international experience. In this respect, international workshops, forums, etc. are especially valuable as opportunity for transferring to Ukraine the advanced technologies, best regulatory practices and experiences in PRA field.

According to regulatory requirements, PSA used for NPP licensing must be full scope PSA. The Utility undertakes efforts on gradual development of full scope PSAs for each plant through a logical step by step procedure. Three pilot units, which represent three types of WWER reactors (Unit 1 of South Ukraine NPP - WWER-1000 of “small series”, Unit 5 of Zaporizhzhya NPP – ‘serial’ WWER-1000 and Unit 1 of Rivne NPP - WWER-440) operating in Ukraine, have developed safety analysis reports with full scope PSA chapters. New units (Unit 2 of KhNPP and Unit 4 of Rivne NPP) also have completed full scope PSAs. PSA for other 10 units in Ukraine will be developed by adaptation of PSA models and documentation from pilot units to non-pilot units of similar design. Scope and degree of adaptation depend on differences in design, construction, procedures, operational practice and experience between NPPs. To a certain extent, adaptation can be considered as harmonization of PSA. Expected
results of this process will be estimated level of overall safety for each power unit in such way, that the differences in PSA results can by explained only by differences between units, not by differences in PSA teams, methods and approaches.

To realize adaptation procedure in justifiable and consistent way, a standardized approach was developed, which was approved by SNRCU and Utility in 2007.

Based on established infrastructure for PSA application, the Utility and Regulatory Authority proceed now with practical use of PSA and PSA applications. Ongoing activities in this field are associated with:

(a) evaluation and prioritization of safety areas/ safety upgrade measures (all NPPs);
(b) optimization of test, maintenance and repair procedures (e.g. SUNPP Unit 1, ZNPP Unit 5);
(c) using PSA together with full scope simulator trainings to improve operator training programs and enhance human reliability, and to evaluate and improve emergency operating procedures (e.g., SUNPP Unit 1); Main insights show that integral evaluation of accident progressions provides important information on the benefits and drawbacks of various operations in abnormal plant states; and there is significant potential for safety increase due to improvement of EOP and more efficient use of full scope simulator at NPP.
(d) introduce compensatory measures to decrease safety deficiencies (e.g. for turbine hall issue at RNPP Units 1 and 2).

Further industry activities on RIDM are connected with risk-informed in-service inspection; prioritization of regulatory inspections; precursor analysis, etc.

4 CONCLUSIONS

Permanent safety improvement activities at Ukrainian NPPs provide considerable enhancement of safety level. However, potentials for safety increase are not exhausted. Ukrainian Nuclear Regulatory Authorities and Utility promote use of PSA and PSA application as advanced tools that complement traditional methods and facilitate decision making on the safety. Enhancing safety and efficiency of NPP operation is especially important for Ukraine, with 50% share of nuclear power in total energy production. Pilot PSA applications have demonstrated their efficiency for regulatory tasks and for Utility. There is a need in further strengthening the regulatory framework and technical basis for practical use of RIDM.

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Exploitation of BEPU Approach for Licensing Purposes

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ABSTRACT

Best estimate codes supplemented by uncertainty evaluation (i.e. BEPU = Best Estimate Plus Uncertainty) achieved a suitable maturity and can be applied to the licensing process of water cooled nuclear power plants. The Final Safety Analysis Report (FSAR) constitutes the key element for demonstrating the safety of a nuclear power plant. Thus BEPU approach is relevant to the FSAR.

The present paper deals with a proposal to apply a BEPU approach to the FSAR of Atucha-2 PHWR (Pressurized Heavy Water Reactor) now in construction in Argentina with operation start expected in 2010. Atucha-2 is a vessel type two loop KWU-Siemens-AREVA plant with vertical core channels and hydraulically separated moderator and coolant circuits.

The following key aspects are at the basis of the possible application of the BEPU approach in the Atucha-2 licensing process:

- The accident classification: the standard distinction between Anticipated Operational Occurrences, Design Basis Accident and Beyond Design Basis Accident is considered. However, different assumptions are proposed for the analysis of selected transients belonging to individual classes.
- Conservatism is embedded into the reactor design including protection system and related set-points; furthermore, conservatism can be added in the analyses through the proper choice of boundary and initial conditions or by (typically) preventing redundant trains of emergency core cooling systems (ECCS) from contributing in recovering the plant safety functions.
- A best estimate analysis of phenomena expected in the concerned class of transients implies the coupled application of system thermal-hydraulics, computational fluid dynamics and three-dimensional neutron kinetics computational tools other than structural mechanics codes. The demonstration of suitable qualification for the computational tools constitutes a challenge for the present approach.
1 INTRODUCTION

The Atucha-2 nuclear power plant is designed to produce 745 MW of electrical power, with a pressurized heavy water cooled and moderated reactor (PHWR). The Atucha-2 Construction License was issued in July 1981, upon a previously submitted Preliminary Safety Analysis Report (PSAR) [1], basically fulfilling the requirements on Safety Analysis Reports, established by IAEA standard [2], although its format has been prepared in accordance with a largely adopted US standard [3].

Consistent with the fundamental radiation protection objective, the Argentinean regulatory standard AR 3.1.3 [4] establishes a criterion which provides an upper limit for the radiological impact for nuclear power plant operation (restriction of radioactive releases), including the consideration of the complete spectrum of accidents and the correspondent probabilities of occurrences. Additionally, this standard requires a probabilistic safety analysis to support the acceptability of the design of a nuclear power plant.

During the period between 1977 and 1994, significant progresses have been observed in the activities of design finalization, components fabrication and of the erection of structures and buildings. The established licensing compromises have been strictly observed and are embedded in the Atucha-2 safety design concept.

Despite the fact that the licensing procedure in Argentina follows basically a probabilistic approach, a deterministic safety analysis must also be submitted. With the exclusion of the maximum credible accident from the range of the design basis spectrum for Atucha-2, a break size of ten percent on reactor coolant pipe (0.1 A) was recognized [5] as the basis for fulfilling traditional regulatory requirements.

After a long period of delay, a decision was taken to resume the construction, and to bring the plant into operation until the year 2010. Consequently, strong demands have been derived for design finalization, including the issuance of a Final Safety Analysis Report (FSAR). To this aim, a twofold strategy is foreseen: the original safety design philosophy must be preserved, and recent advances in nuclear safety technology should be incorporated, as long as possible.

Significant progress has been observed in areas of the nuclear safety field, in the last twenty years. Many of them are related to improvements in the ability to predict plant behavior during normal and accident conditions. Evolution of analytical models, supported by comprehensive experimental efforts made available powerful computational tools which provide support for detailed calculation on relevant phenomena for nuclear reactor dynamics.

Within the framework of Atucha-2 project finalization, NA-SA (Nucleoeléctrica Argentina Sociedad Anónima) and Unipi (University of Pisa) have signed an agreement for supporting activities in the area of deterministic safety analysis methods. Separate efforts are also undertaken in the area of probabilistic safety analysis, which are, however, out of the scope of this paper.

Derived from the connected probabilistic approach, the double ended guillotine break is considered as a beyond design basis scenario. Nevertheless, the demonstration of the design capability to overcome this event has still a relevant role in the safety performance evaluation. For this aim, however, currently used conservative approach for safety analysis may not be sufficient to guarantee that safety margins still exist. Quantification of available safety margins seems to be an adequate strategy.
2  OBJECTIVE AND SCOPE

This paper aims at to describe a comprehensive and modern methodology proposed for performing the deterministic safety analysis to be included in the chapter 15 of the Atucha-2 Final Safety Analysis Report. Such analyses provide the basis to determine the plant limiting conditions for operation, limiting safety system settings, and design specifications for safety-related components and systems, as well as the deterministic technical basis for the demonstration of the existence of adequate protection of public health and safety.

The proposal has been developed starting from the original SIEMENS methodology planned to be used for accident analysis [6], but further enhanced to comprise the use of modern best estimate computer codes and methods, including evaluation of uncertainty in the calculated results (Best Estimate plus Uncertainties or BEPU approach). It follows well established safety practices, but also includes some advanced solutions compatible with the state of the art in the field of nuclear reactor safety.

The proposed approach has been built to cover the design basis spectrum of events, but it was intentionally extended to address additional safety relevant events beyond design basis, including the double-ended guillotine break (DEGB) loss of coolant accident (LOCA).

The acceptance criteria adopted for these analyses are of deterministic nature, even for radiological consequences associated to the events.

The application of the proposed methodology does not include, however, the analysis of severe accidents, by recognizing the complementary role of probabilistic safety analysis and by understanding that related issues are to be addressed by an independently and separately performed study under NA-SA responsibility.

3.  PROPOSED BEPU APPROACH

The event sequences postulated in the design of the plant are analyzed to demonstrate that in operational states, on the occurrence of a design basis accident and, to the extent practicable, on the occurrence of some selected accident conditions that are beyond the design basis accidents, the following three fundamental safety functions are performed:

- Safe shutdown and long term subcriticality
- Residual heat removal
- Limitation of radioactive releases.

The proposal follows well accepted design philosophy for nuclear power plants, which recognizes the principle that plant states which could result in high radiation doses or radioactive releases are of very low probability of occurrence, and plant states with significant probability of occurrence have only minor or no radiological consequences. Postulated initiating events (PIE) are grouped according to their anticipated probability of occurrence in anticipated operational occurrences (AOO), design basis accidents or selected beyond design basis accidents (SBDBA).

The third event category is proposed to address specific scenarios beyond design basis, including DEGB LOCA and ATWS. Accident conditions which stand out of these ranges of probabilities should be treated separately, through the probabilistic safety analysis.

The approach takes credit of the concept of evaluation model (EM), and comprising three separate possible modules depending on the application purposes:

- For the performance of safety system countermeasures (EM/SA)
- For the evaluation of radiological consequences (EM/RA)
- For the review of components structural design loadings (EM/CA).
All selected scenarios are grouped in a classical nine families of events, already established within the scope of the PSAR, where each family covers events with similar phenomena.

For the FSAR Chapter 15 analyses, and for each category of events, the results of the analyses will be assessed in terms of the fulfilment of safety functions which are graded according to the expected frequencies of occurrences for the correspondent PIE.

To keep a consistently flat risk profile over the entire spectrum of AOO and DBA, the more frequent the event is, the less tolerable its consequences are. In this sense, acceptance criteria are selected for different event categories, for safety parameters as fuel and cladding temperatures, departure from nucleate boiling ratio (DNBR), primary circuit pressure, containment pressures, and total effective dose equivalent (TEDE).

To start analyzing typical events scenarios for the chapter 15 of an FSAR, evaluation models rely mostly on system thermal hydraulic codes (as for EM/SA) to solve the transport of fluid mass, momentum and energy throughout the reactor coolant systems. The extent and complexity of the physical modes needed to simulated plant behavior are strongly dependent of the reactor design and of the transient itself.

For some scenarios, or regarding some analysis purposes, the system thermal hydraulic code may, for example, be complemented by (or coupled with) a three-dimensional neutron kinetics code or the reference model may need an expansion to include a detailed simulation of controls and limitation systems which play a relevant role for determining the plant response.

For the scope of the proposed approach for accident analyses, the complexity of the evaluation model may range from a simplified qualitative evaluation (EM/QA) to a complete combination of the three possible modules (EM/SA + EM/RA + EM/CA).

Additionally to the computers codes and the selection of modelling options, the established procedures for treating the input and output information are also recognized as comprising key parts of the evaluation model. The adopted procedures to select initial and boundary conditions, which follows the original design safety philosophy, are of particular importance for supporting the regulatory acceptability of the results provided by the EM.

As the foreseen use of this EM is for licensing purposes, it is necessary to evaluate the suitability of conservative assumptions or to adopt best estimate approaches with the quantification of uncertainties.

Suitability of conservatism should be understood as addressing the issue of “how conservative is conservative enough”. Alternatively, when a best estimate approach is adopted, then realistic assumptions will be input to best estimate models, conducting to realistic estimates for plant behavior. In these cases, licensing applications demand the quantification of uncertainties in the calculated results to ensure that safety margins are still available. For the scenarios were the conservative assumptions may provide enough safety margins, the proposal includes a criterion to determine the need for uncertainty calculations. Typically, SBDBA will involve quantification of uncertainties. It is a relevant part of the proposed approach for accident analyses, the application of a comprehensive method for estimate uncertainties.

The figure 1 shows a simplified flowchart with the steps followed by the proposed approach.

Starting from the selection of the event to be analyzed, after establishing the purposes for the analysis, EM requirements are derived from the identification of event-related phenomena.
Figure 1: Simplified flowchart for the proposed BEPU approach to be applied for Atucha 2 accident analysis
The two main aspects which have been considered for developing the evaluation model with the ability of adequately predict plant response to postulated initiating events are intrinsic plant features and event-related phenomena characteristics.

For the two modules EM/SA and EM/CA, the first set of requirements for the evaluation model is imposed by the design characteristics of the nuclear power plant, its systems and components. Requirements on the capability of simulating automatic systems are of particularly importance for anticipated operational occurrences, in which control and limitation systems play a key role on the dynamic response of the plant. For evaluation of radiological consequences, the EM/RA module has demanded additional appropriate site-related features to be built in.

The third set of requirements is derived from the expected evolution of the main plant process variables and the associated physical phenomena. For the proposed approach, this is performed through the process of identifying the Phenomenological Windows (PhW) and the Relevant Thermal-hydraulic Aspects (RTA). The relevant timeframe for the event is divided into well defined intervals when the behaviour of relevant safety parameters is representative of the physical phenomena.

For the adequate simulation of the identified phenomena, computational tools were selected from those which have previous qualification using an appropriate experimental data base. Satisfactory qualification targets provide basis for acceptability of the postulated application.

For the proposed approach, with the full scope of application of best estimate plus uncertainty quantification (BEPU), a pre-requisite is the availability or the support of the most advanced-qualified computational tools. This comprises a key feature of the proposed approach: all computer codes which have been selected are the best codes available in the market.

For most event scenarios, the single purpose evaluation model EM/SA may be necessary and sufficient to be developed. In this sense, the availability and the application of qualified system thermal-hydraulic code (SYS TH) and reliable uncertainty methodology (UM) should be the minimum requirement.

Additionally, depending on the specific event scenario and on the purpose of the analysis, it is necessary the availability of calculational methods that are not embedded in the SYS TH code, as for burst temperature, burst strain and flow blockage calculations. This may imply an evaluation model EM/CA composed by a fuel rod thermal-mechanical computer code.

For the present proposal, the first selected computational tool would usually comprise a qualified SYS TH code with proven availability of models capable of simulating each individual phenomenon expected to occur during the safety relevant scenarios.

The second selected computational tool, for a minimum requirement scope, is an uncertainty methodology (UM) tool that has been qualified by a comprehensive regulatory body peer review process and by international scrutiny evaluation through participation on projects like OECD/NEA/CSNI Uncertainty Methodology Study [7] and BEMUSE [8-10].

For some specific event scenarios, model requirements derived from identified phenomena may demand for the use of computational tools that have capabilities not available to system code and that allow the ‘best’ simulation of the phenomena expected to be relevant for the safety demonstration of the concerned system.

Two typologies of codes are used as support for application in some accident scenarios. Geometric complexities of the Atucha-2 primary system and three-dimensional mixing phenomena within the reactor vessel play relevant role in the safety performance. In this sense, selected set of codes may involve a coupled SYS TH code with 3 dimensional neutronic capabilities or a coupled use of computational fluid dynamics tool. In the present
case, the local (three-dimensional in nature) effects principally of channel voiding and of boron upon the reactivity justify the use of coupled techniques. Currently, qualification of CFD codes is a very active field (see e.g. [17,18,19]). When it comes to licensing, CFD codes are used mainly as supporting tools (e.g. [16]). Recent works draw attention to quality assurance in CFD applications for reactor safety analysis (“Best Practice Guidelines”, [20]).

Qualified code and qualified uncertainty method imply qualified interfaces between the code and the simulated systems (this is called nodalization in the case of thermal-hydraulic system code) and qualified group of users, where:

- Qualified code-input data set or nodalization implies fulfilment of qualitative and quantitative acceptance criteria, separating nodalization development, achievement of steady state and transient analysis.
- Qualified group of code users should be available for the selected SYS-TH code. This implies the documented consideration of recommendations included in internationally agreed documents, e.g. references [11] to [13].
- Qualified UM input data set and UM users. Although internationally recognized as an open issue, (e.g. as discussed in ref. [10]), and with main concern being regarding the identification the input uncertain parameters and their range of variation, for the present approach this problem is not applicable, as an internal assessment of uncertainty capability is available.

4. ASSUMED BOUNDARY AND INITIAL CONDITIONS

To build a complete and detailed EM for a particular plant and event, properly selected code options, boundary conditions, and temporal and spatial relationships among the component devices, code input specific data set are derived.

The strategy for the analyses follows the original design safety approach [6]. Plant behavior is investigated under design basis specific predetermined operational states and accident conditions, as well as for some selected beyond design basis conditions, applying a specific set of rules to provide enough insurance (as per regulatory requirements) on design adequacy.

Basiclly, for each event to be analyzed two complementary scenarios are investigated:

A. Realistic Case – in a first step, the realistic sequence of events is calculated under normal (best estimate) conditions for which all systems which did not fail as a consequence of the postulated initiating event are assumed to be available. Additional failure assumptions in case of normal conditions are not postulated.

B. Conservative Case - Regarding the fact that additional failures are conceivable or have to be postulated according regulations, in addition to the normal case, and as a second step of calculations, a conservative case of each respective event is also analyzed.

As a general assumption for conservative cases of accidents, it is assumed that, beyond the postulated situation of a subsystem being repaired, when a safety system is needed, there will be a single failure (random failure) in one of the safety devices.

Failure assumptions for control, limitation and reactor trip are also postulated to achieve a sufficient level of conservatism in the analysis of AOO and DBA.

The evaluation of these conservative cases shall demonstrate that measures of reactor protection system and safety grade systems are available (n+2 prove) for event control and for the successful performance of the correspondent subsidiary safety functions.

The realistic case calculations are performed to demonstrate that anticipated operational occurrences will not escalate into accident conditions and that, as a rule, there is no need for safety-grade system to operate. Reactor trip is possible, and even necessary, in some cases.
For the proposed approach, the analyses of the events are performed, both the realistic and the conservative cases, starting from the same set of nominal parameter values (e.g. temperature, pressures) corresponding to that power level for which the analyses are performed. For example, some reactivity transients are analyzed for 0% power level, although most of the cases are analyzed for 100% power level. Deviations of the plant parameters from their rated setpoints are, in general, not considered.

For AOO with postulated additional failures, an escalation into an accident condition is also conceivable under certain circumstances (such as the coincident failure of the reactor power limitation system and a control system). Analyses of AOO for conservative conditions are conducted to demonstrate that, despite additional system failures, the next level of acceptance criteria (that means for DBA) is met due to the action of safety systems (reactor protection system, engineered safety features).

By contrast, for design basis accidents, the realistic case is calculated only to show the expected behavior of the plant, with all safety systems countermeasures providing enough margins to the applicable acceptance criteria. In the conservative case, it is necessary to demonstrate the effectiveness of the safety system, with credit taken only for those systems or system redundancies which may deterministically be taken as available for mitigation of the consequences of the event.

For beyond design basis accidents, measures are foreseen to mitigate their consequences. These measures are introduced under consideration of achieving a reasonable balance between the engineering effort and its achievable risk reduction. They are event specific actions using design margins of the plant.

Anticipated transients without scram (ATWS) are analyzed only for the normal case, which means best estimate conditions, to demonstrate that due to common acting of all available systems using thereby design reserves of the plant, the subsidiary safety functions of heat removal and long term subcriticality of the reactor are ensured.

For extended spectrum of LOCA recognized as SBDBA, or extremely low probability scenarios, only best estimate cases are evaluated. As additional system failures are not required to be postulated, to quantify the available safety margins, regarding the same acceptance criteria as for DBA, involved evaluation model uncertainties are quantified.

For extended spectrum of loss of coolant accidents (from 0.1 to 2A break sizes), the use of a systematic approach which quantifies the uncertainty (BEPU) supports the demonstration of the existence of margins to the safety limits of the activity barriers.

5. UNCERTAINTY QUANTIFICATION – CRITERIA AND METHOD

In principle, whenever a best estimate method is applied for licensing purposes, uncertainty quantification is needed. For the present proposal, as a realistic conservative approach, it should include uncertainty quantification. Nevertheless, due to the conservatism embedded in some assumptions, and due to efficient safety performance of limitation and protection systems, in many cases there is no real need for such calculations. The proposed BEPU approach derived a non-safety related criterion to decide upon the need for performing uncertainty calculation. Whenever the safety parameter, as calculated by the evaluation model, comes within an established range or distance from the limit value, the uncertainty in the calculated results is quantified (BEPU). For any case, however, such calculation may be performed, and specific additional demands from the regulator may be fully addressed.

The application of acceptance criteria is also addressed for SBDBA, but with the need to determine the available safety margins, the uncertainties are always quantified.

For the proposed methodology, all events in the category SBDBA and some events in the DBA category will be analyzed with the full scope of the BEPU approach.
This is particularly valid for the evaluation model module developed for performing safety systems analyses (EM/SA). Usually, radiological consequences and component stress analyses follow well established conservative and deterministic procedures. In this sense, the correspondent evaluation modules (EM/RA and EM/CA), in principle, will not include uncertainty methods.

For the EM/SA, however, uncertainty analyses are performed to confirm that the combined code and application uncertainty is less than the design margin for the safety parameter of interest when the code is used in a licensing calculation. Examples of safety parameters are peak cladding temperature (PCT), cladding oxidation thickness and departure from nucleate boiling ratio (DNBR).

The internal assessment of uncertainty is a relevant capability for thermal-hydraulic system codes. This consists of the possibility of obtaining proper uncertainty bands each time a nuclear power plant scenario is calculated. At the basis of the derivation of the code with (the capability of) internal assessment of uncertainty (CIAU), there is the uncertainty methodology based on the accuracy extrapolation (UMAE), although other uncertainty methodologies can be used for the same purpose.

The UMAE method [14] focuses not on the evaluation of individual parameter uncertainties but on the propagation of errors from a suitable database calculating the final uncertainty by extrapolating the accuracy from relevant integral experiments to full scale NPP.

The basic idea of the CIAU [15] can be summarized in two parts, as per Figure 2:
- Consideration of plant status: each status is characterized by the value of six relevant quantities (i.e. a hypercube) and by the value of the time since the transient start.
- Association of an uncertainty to each plant status.

![Figure 2](image-url)

Figure 2. – Outline of the basic idea of the CIAU method.

In the case of a PWR the six quantities are: 1) the upper plenum pressure, 2) the primary loop mass inventory, 3) the steam generator pressure, 4) the cladding surface temperature at
2/3 of core active length, 5) the core power, and 6) the steam generator down-comer collapsed liquid level. These quantities are also considered in the large break LOCA analysis of Atucha-2 [21], a PHWR.

A hypercube and a time interval characterize a unique plant status to the aim of uncertainty evaluation. All plant statuses are characterized by a matrix of hypercubes and by a vector of time intervals. Let us define Y as a generic thermal-hydraulic code output plotted versus time. Each point of the curve is affected by a quantity uncertainty (Uq) and by a time uncertainty (Ut). Owing to the uncertainty, each point may take any value within the rectangle identified by the quantity and the time uncertainty. The value of uncertainty, corresponding to each edge of the rectangle, can be defined in probabilistic terms.

The idea at the basis of CIAU can be made more specific as follows: the uncertainty in code prediction is the same for each plant status. A Quantity Uncertainty Matrix (QUM) and a Time Uncertainty Vector (TUV) can be set up including values of Uq and Ut derived by an uncertainty methodology.

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Cultural And Organizational Factors Leading To Major Events

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ABSTRACT

More than ten events from a range of industries (e.g. nuclear, petrochemical, transport) on a worldwide basis, including several recent nuclear events, have now been analysed in detail. The work has allowed common organisational and cultural factors to be identified, from leadership issues to operational cultural issues and the impact of commercial pressures. It is argued that if these are recognised and addressed, the risks of further events might be reduced. These common factors will be presented and discussed in the paper. In addition, organisational objectives and specific supporting question sets have been generated against the above identified key factors with the intention that operating organisations and regulators can carry out assessments of the vulnerability of organisations to these complex ‘organisational accidents’. The aim of continuing research is to develop a software tool capable of addressing organisational vulnerability. Approaches to structuring such a tool are discussed.

Keywords: organisational safety culture, systems, processes, evidence, risk, vulnerability.

1 INTRODUCTION

Major accidents in the nuclear industry are rare, not least, because significant effort has been spent on designing and operating plant to minimise the risk of such events. Nonetheless, in the last decade there have been several lesser incidents and ‘near-hits’ which provide further learning to continue the process of risk reduction. Some of this learning relates to technical and procedural issues. However, maintaining high levels of safety requires also the understanding of causes involving more deep-seated and complex issues relating to organisational and cultural shortcomings.

The nuclear industry is not alone in having to deal with such issues in order to ensure continuing high levels of safety. Parts of the chemical/petrochemical, transport and civil engineering industries also have to operate to high standards and pay great attention to such issues. There have been several significant events in these sectors over the last decade or so, from which we can collectively learn.

In a previous study carried out as part of an internal BNFL project, five major accidents were studied including one nuclear related event (the JCO criticality accident in Japan in 1999) [1]. The findings on the organisational and cultural causes of these events and some preliminary conclusions attracted interest in the nuclear industry and beyond. Understanding
such ‘organisational’ accidents has also become of growing interest in industry and among
regulators following more recent events such as the BP Texas City oil refinery disaster.

The present paper involves a deeper study of ten events (including three of those from
the BNFL study). A description of the identified issues is the main subject of this paper. It
should be emphasised that in discussing these and other events we have drawn on the findings
of reviews and Inquiries. We have attempted to do this in a spirit of learning and it is not our
intention to establish blame or to criticise organisations or individuals. In each case, the
organisations involved were subject to pressures and difficulties (many of which we attempt
to identify).

One important finding of the BNFL study was that when considering organisational and
cultural causes, similar issues appear to surface whatever the nature of the event or the
industrial sector involved. Such a recurrent and repetitive pattern was again revealed across
all the ten events studied in this research. This is important because it provides an
opportunity to address such issues generically and to develop tools for diagnosis and action
which might provide further opportunities to reduce risks in a wide range of circumstances.
This will be relevant to organisations themselves seeking to improve safety, particularly in a
nuclear or process safety context but also to regulatory bodies.

A wider discussion of these issues and the development of diagnostic question sets is an
important first step. A longer term objective of the current research is to attempt to develop a
‘vulnerability tool’ which attempts to structure the issues and gain clearer systems insight.
The tool to be developed will be rooted in previous work at the Safety System Research
Centre (SSRC) in the modelling of complex systems [2]. Also important objective will be to
make such a tool. Some preliminary approaches to modelling this are outlined in the final
section of this paper.

2 THE FINDINGS FROM TEN EVENTS

To illustrate the relevance of organisational safety culture failings and its implications
from a safety point of view, ten events have been studied in some depth. The current work,
has used a similar methodology to that described in Taylor and Rycraft [1]. This in turn was
broadly based on the approach to such accidents developed by Turner [3], Blockley and
Pidgeon [4] and Reason [5]. The ten events studied were:

1) Port of Ramsgate walkway collapse (UK, September 1994) [6];
2) Heathrow Express NATM tunnel collapse during construction (UK, October
   1994) [7];
3) Longford gas plant explosion (Australia, September 1998) [8, 9, 10];
4) Tokai-mura criticality accident (Japan, September 1999) [11];
5) Hatfield railway accident (UK, October 2000) [12];
6) Davis Besse pressure vessel corrosion event (USA, February 2002) [13];
7) loss of the Columbia Shuttle (USA, February 2003) [14];
8) Paks Nuclear Plant fuel cleaning event (Hungary, April 2003) [15, 16];
9) Texas City oil refinery explosion (USA, March 2005) [17, 18, 19];
10) loss of containment at the THORP Sellafield reprocessing incident (UK, April
    2005) [20].

Organisational and cultural findings contributing to each event were assembled from the
published reports for each of the ten cases studied. The analysis revealed strong similarities
between the findings. As a step in the development of a ‘vulnerability tool’, these have been
grouped under eight generic/key headings. The main areas identified and discussed in more
detail below are:

1. leadership issues;
2. operational attitudes and behaviours (operational ‘culture’);
3. the impact of the business environment (often commercial and budgetary pressures);
4. oversight and scrutiny;
5. competence and training (at all levels);
6. risk assessment and risk management;
7. organisational learning;
8. communication issues.

These issues were relevant to nearly all of the events studied. In addition, several (but
not all) of the events had specific learning relevant to external regulation and to the interface
with contractors. More detailed questions sets based on the identified issues have been
developed and these will be used as an input to the vulnerability tool. These issues will be
addressed in the final project report.

2.1 Leadership

Weak/ineffective leadership is considered by the authors to be the most fundamental
issue leading to most of the events analysed. Specific issues include the following:

• The need for commitment to nuclear/process safety from ‘the top’ and the
communication of this as a core value to the workforce in a compelling and
intelligible way, such that the priority attached to this in the organisation is beyond
question.
• A requirement for a strong understanding of operational ‘reality’ obtained from high
leadership visibility and a questioning attitude about matters as they really are, rather
than encouraging the transmission upwards primarily of ‘good news’.
• A sufficient understanding of nuclear/process issues so that information received and
decisions taken can be considered in an informed way and properly integrated in the
business decision making process.
• The development of clear organisational structures, which minimise complexity.
This also ensures clarity about roles and responsibilities. It also facilitates good
communication and minimises the existence of ‘silos’ which can reduce team
working and learning.
• The need to ensure that the organisation maintains its capability as the ‘controlling
mind’ and is an intelligent customer for services that it buys in, with an
understanding of the role of licensee or equivalent.
• Ensuring that there is an effective safety management system (SMS), that this is
supported by a strong safety culture and that ‘policy’ is translated into operational
requirements and procedures in such a way that the users of the SMS understand the
basis of requirements and receive help and advice, where necessary, in
implementation. In particular there is a need for a clear and well-understood
‘balance’ between requirements from the ‘centre’ and discretion given to operational
units.
• The need for sufficient information effectively to monitor and review performance –
for example, reviewing on a regular basis a suitably detailed range of performance
indicators for nuclear/process safety which contain leading as well as lagging parameters. This is further discussed in section 2.6.

- The existence of processes which recognise the importance of nuclear/process safety issues and integrate these with decision making about other aspects of business performance. Issues relating to nuclear/process safety must always be given sufficient prominence (e.g. when compared to the review of financial and commercial performance).

- The existence of an approach to communication which transmits key expectations and issues to the workforce and which encourages and facilitates feedback which is then used to drive improvement. An effective system allows key messages to be cascaded into the organisation in a suitable form and thus ensures that the ‘right messages are received by the right people at the right time’.

- The enabling of processes and systems which ensure that risks are properly assessed and reviewed and that this is done in such a way that independent challenge is welcomed, that learning is encouraged and shared and that there is clarity about priorities backed by adequate resources. When actions are taken to address risks it is essential that leaders confirm that these have been satisfactorily implemented.

- An awareness by leaders of the nuclear/process safety risks which they are managing and a recognition that when commercial and other pressures require organisational changes to be made, this is done after carefully considering the effect on these risks and adequate resources are available to manage them.

2.2 Operational attitudes and behaviours

Analysis of the events studied has provided many examples of issues which are brought together under this broad heading.

- Poor quality procedures (sometimes not reflecting risk assessment or safety case findings) and/or failure to comply with them – in particular, a failure to distinguish between ‘what is written and what is done’. This led to ‘workarounds’, violations and/or the development of informal procedures.

- Failure to ensure that operators have a sufficient understanding of the risks that procedures and instructions are designed to control. In some cases, the workforce had not received sufficient training on nuclear/process risks and there was a false belief that the control of industrial safety risks (e.g. slips, trips and falls) would necessarily lead to good performance across the spectrum of safety risks.

- Significant attention needs to be given to the vital role of first line supervisors in both setting standards and challenging unacceptable practices.

- Failure to encourage a questioning attitude and constructive challenge allows the development of mindsets. In some cases this resulted in important risks being ‘normalised’ and risks being taken on a habitual basis by default. In such cases, risks which were once identified as significant and worthy of particular attention became neglected because they did not lead to major problems.

- The need to ensure that there is ‘conservative decision making’ such that nuclear/process issues are always given sufficient attention and priority. This is particularly relevant in cases where novel processes are being used or in the case of ‘new plant culture’ – the view that a new plant or process at the cutting edge of technology is unlikely to fail. It can also be important, however, where a process has become familiar and operators are no longer sufficiently cautious.
• Failure to address issues of complacency/overconfidence often arises from a view that the organisation has ‘always done it this way’. In some cases the organisation had previously been successful but unrecognised organisational drift then led to degraded performance.
• Poor communication, particularly at shift handovers or between engineering/specialists and operational staff, were factors in several events.
• A willingness to operate with equipment which is in an unacceptable condition or in a working environment which is conducive to poor quality in operations.
• A failure to encourage and involve individuals and teams in identifying improvement opportunities and ‘challenging’ poor standards.
• Weaknesses in providing sufficient capability in recognising and dealing with abnormal events and/or recurring issues. This was exemplified in several events by a failure to understand the significance of alarms, to deal with information overload and to seek assistance when issues had escalated beyond normal operations.
• The development of inappropriate patterns of work with casual transfer of roles and in some cases the working of long hours leading to fatigue and possible deterioration in the ability to make important decisions.

2.3 Business environment

Nearly all of the events studied arose against a background of significant commercial and/or operational pressure. In any organisation there is always a balance to be struck between the pressures of production/delivery and the achievement of acceptable levels of safety performance. It is when the balance leans towards an emphasis on achieving commercial results at the expense of safety that danger arises. The following are among the specific issues which have arisen:

• A failure to consider the nuclear/process safety implications of changes to the organisation in terms of either people or other resources, sometimes because required changes are perceived as urgent and sometimes because there is insufficient analysis to make leaders and managers aware of the implications of the change.
• In some cases, business decisions from ‘above’ have overburdened plants so that they have been overloaded with initiatives and requirements. This has led to a loss of direction and sense of priority. More specifically, personnel have regarded changes as ‘flavour of the month’ and commitment and trust has been lost. Loss of direction was sometimes exacerbated in cases where there were very rapid changes in the composition of the leadership team.
• In organisations where resource reductions become the norm (e.g. cost cutting in continuing attempts to restore profitability in the face of changing market conditions), ‘salami slicing’ of resources has taken place without the review of the cumulative impact of such changes on nuclear/process safety.
• Where new facilities are acquired, this can lead to positive steps to improve the material condition and people-related issues at the facility. Sometimes, however, the fact that infrastructure is in a relatively poor state is not fully recognised and acted upon and it is allowed to deteriorate further with the new owners unwilling or unaware of the need to seek substantial improvement.
• Commercial and ‘political’ pressures have led to organisations outsourcing or passing substantial safety related responsibilities and competences to contractors often in order to minimise costs. This can result in a loss of clarity about accountabilities, a failure of the contracting organisation to maintain its competence
as an informed and intelligent customer and in some contexts, to abrogate its responsibilities as a licensee/duty holder.

- Incentives have sometimes been introduced which fail to take account of nuclear/process safety issues and which concentrate on financial or quality-related issues – sometimes with a negative impact on safety. Where such incentives are introduced, it is important to examine the potential impact of these and introduce balancing requirements or incentives to give safety sufficiently high priority.
- Changes in the business environment which have led to processes or plant becoming neglected. The ‘orphan plant’ issue, as exemplified by the Tokai-mura accident, illustrates the potential of this as a factor in events. In this case there was an apparent lack of ‘ownership’ of a peripheral plant which was not in the mainstream of the organisation’s business. A similar issue relates to ‘organisational drift’. In this case, a once ‘high performing’ plant deteriorates and standards drop whilst leaders and regulators fail to notice and continue to act as though the plant has retained its previous high standards.

2.4 Competence

Most of the events studied have shortcomings in competence as an issue. The following issues have been identified from the events studied:

- In some events, there was a gradual erosion of competence and a lack of process-related knowledge. This was not identified because of a failure to review competencies for nuclear/process safety on a regular and systematic basis – particularly during or following major organisational change. This relates to positions at all levels in the organisation and often includes contractors.
- There is a need to ensure that senior managers and organisational leaders have sufficient understanding of the risks which they are seeking to manage and to ensure that the consequences of failing to do so has been highlighted.
- Some events studied highlighted the need for front line staff and their supervisors to have a greater understanding of nuclear/process safety risk and an ability to recognise when abnormal and potentially dangerous situations are developing. In these situations they need to be able and willing to draw on competent specialist support.
- In some events, training was superficial and based on a ‘tick box’ approach without adequate planning, assessment and direct personal support to ensure a deeper understanding of principles and the underlying issues.
- Technical competence is vital but issues relating to non-technical capabilities such as communication, team working and issues relating to safety culture (such as the need for a questioning attitude and the importance of reporting and learning from events and precursors) in some cases did not receive the necessary attention.

2.5 Risk assessment and management

This area has again been highlighted by almost all of the events studied and includes a wide range of issues from the strategic to the specialist, through to the assessment and management of risks in day-to-day operations. The specific issues from the events studied include the following:

- Failure to have in place an overarching process by which nuclear/process safety risks can be assessed and minimised. For some of the events studied, the organisations
had no systematic process to identify and prioritise the risks and their response to them. In some cases they were overwhelmed by competing priorities such that the key risks did not receive the attention they deserved.

- In other cases, the organisations had drifted into a mindset or state of complacency as a result of previous good performance and/or excellence. This meant that emerging and (in some cases) long-standing nuclear/process safety risks were not identified or were not seen to be significant. Where they were recognised, they were ‘normalised’ and actions were sometimes inappropriate or ineffective, with no check on their effectiveness.
- In several cases, there was a lack of rigour in assessing risks, developing a suitable safety case, ensuring that issues were reflected in operational procedures and that these were then adequately controlled.
- Mechanical integrity programmes and related inspection programmes were not maintained and remedial actions prioritised.
- Important technical findings, such as good practices identified in Hazops and safety reviews, were deferred (sometimes for budgetary reasons).
- Indicators associated with abnormal conditions (e.g. alarms and data trends) were not systematically addressed in several cases, particularly during start-ups and shut-downs.
- There was no recognised and useable process for assessing the effects of organisational and, in some cases, technical changes on nuclear/process safety.

2.6 Oversight and scrutiny

When failures occur in systems and/or as a result of a weak organisational culture, this can be put right before a major failure occurs by oversight systems designed to alert different layers of the organisation to the deficiencies. Failures in oversight were (perhaps unsurprisingly) a common feature of all of the events studied. The following specific issues were identified:

- A failure to have in place a hierarchical, layered system of checks and balances. In some cases there was only a conventional audit process - often solely within the line and consequently lacking clear independence. In some cases this did not look beyond paper systems and did not identify failures to comply and deficiencies in the underlying safety culture.
- Oversight processes were sometimes ineffective because they were either poorly resourced, reports and feedback were not given sufficient weight and/or were not the subject of sufficient questioning by the recipients of the reports. This was sometimes reinforced by a ‘good news culture’ in which unpalatable aspects were not highlighted or acted upon.
- In some cases, information being fed up to senior leaders was aggregated such that weaknesses relating to particular plants or functions could not easily be identified and addressed. On occasions, also, there was a failure to prioritise remedial actions and then to check that actions had been carried out and had achieved the desired outcome.
- Early warning of emerging issues can most effectively be identified in the oversight process if key measures and issues are integrated. Thus it is not sufficient to rely just on performance indicators. An effective system uses these together with audit findings, event reports and through the commitment of senior leaders to question safety performance systematically to the same depth and intensity to which financial
and project related programmes would usually be scrutinised - for example through regular face-to-face scrutiny meetings between leaders and their direct reports. In few of the events studied, did leaders appear to exercise a formal, integrated process.

- Safety Departments (which might be expected to provide independent authoritative advice) were not sufficiently resourced or competent and/or did not have sufficient authority to stop potentially unsafe operations.
- In several of the events studied, organisations had once been strong performers with a good reputation, but this had gradually eroded without the organisation being aware of this. This ‘organisational drift’ is often an important precursor to organisational accidents.
- Failure to detect weaknesses in nuclear/process safety performance also arose from the lack of suitable nuclear/process safety metrics. In some cases over-reliance was placed on metrics relating to personnel/industrial safety and it was wrongly assumed that successful performance in these areas of safety would ‘guarantee’ excellence in nuclear/process safety. In nearly all cases suitable metrics relating to nuclear/process safety were not available or contained only lagging indicators.
- There was evidence in many cases that leadership teams at the top of the organisation were unaware of the reality of safety shortcomings at plant level. Findings were not always questioned, in some cases probably because of a lack of expertise at this level about the nuclear/process safety issues involved but, in some cases, because it appeared that the needs of the broader business agenda did not ‘align’ with the information being made available through the oversight processes.

2.7 Organisational learning

For most of the events studied there had been previous events from which there was suitable learning available. If this had been acted upon, the event would not have occurred. The following issues were identified:

- There sometimes did not appear to be an effective system for event reporting particularly in relation to nuclear/process safety. Reporting was poor for a variety of reasons, including apparent concerns from staff that their reports would not be part of a ‘just’ or ‘blame free’ response, that bad news would not be welcome at more senior levels, that there was insufficient knowledge to recognise precursors and/or that there was simply a culture of mistrust and/or complacency which did not encourage open reporting.
- Previous events had not been investigated on a systematic basis. This was reflected in a failure to investigate some events at all and in other cases there was a failure to consider root causes. Learning from events was often not shared within the organisation or beyond as part of an effective OEF programme.
- In many cases there were historical events which provided significant learning opportunities. Some of these had happened in the organisation and others were from other companies within the same industrial sector. Where these had been recognised as learning opportunities, they had often faded in significance within the corporate memory or improvement actions taken had not been tracked, completed or carried out effectively.
- Members of the workforce were sometimes not aware of the risks being run through poor practices or failed equipment. For many of the events studied there appeared to be little evidence that organisations were actively encouraging the workforce to
become involved in improvement activities in the area of nuclear/process safety as individuals or as teams.

- The existence of ‘organisational silos’ also meant that important knowledge which might have minimised the risk of the resulting event was not transferred. There was, for example, a failure to transfer learning between engineering or technical staff and operations staff or to share learning with contractors.

2.8 **Use of contractors**

The impact of contractors and their control by the contracting organisation was an important factor in about half of the events studied. Identified issues included:

- A gradual loss of control with more and more responsibility being ceded to contractors and without the contracting organisation always being aware of its failure to retain the necessary control.
- In doing this, the contracting organisation sometimes lost competence (or failed to develop the necessary competence) and was thus unable to determine whether the contractor was carrying out its operations safely and/or with an acceptable safety culture.
- In some cases the contractor was more aware of an emerging issue than the contracting organisation. However, failures of communication, a lack of competence or commercial and other pressures, meant that advice from the contractor was not acted upon.
- Contractors were in some cases the subject of contracts which did not properly reflect the importance of safety as part of their role. Incentives to complete on time and to cost sometimes reinforced this.

2.9 **Communication**

Communication issues enter into all events in one form or another. Relevant issues were:

- Failure of communication between leaders and the workforce about the high priority to be attached to nuclear/process safety. Where communication had occurred it was often based on written systems and the element of personal commitment shown in good face to face communication had been lost. This was also often not a two way process with leaders failing to ask the workforce about the real operational issues affecting nuclear/process safety.
- Breakdowns of communication occurred between contracting organisations and their contractors.
- Breakdowns of communication occurred between operational staff and those providing engineering and/or technical support.
- The existence of organisational ‘silos’ where communication was only channelled through certain routes and where communication between individuals and teams was outside the norms of organisational behaviour.
- In several events, there was evidence of a breakdown of communication at critical points in the progression of an event. A particular example was breakdown in communication at shift handover. In some cases this involved the failure to use an existing procedure, whilst in others, communication was generally carried out within an informal setting and was not subject to a formal process requirement.
• Failure of communication with other organisations, particularly in terms of improvement ideas and the sharing of learning, was an important element in some of the events studied. Many of the organisations involved had developed a ‘closed’ culture in which communication in a spirit of questioning and learning from others was no longer a practice which was supported and encouraged.

2.10 External regulation

Regulatory bodies provide a vital safeguard in acting as a last line of defence in providing oversight. When effective, they can also provide a stimulus to the achievement of good practice and enable organisations to realise that complacency, organisational drift or overconfidence is becoming a danger.

Many of the issues discussed above are applicable to the regulatory body (e.g. leadership, competence, organisational learning and review and oversight of the regulator’s own role). Commercial pressures may also be significant both in terms of the level of resource available to regulators and the need for them to challenge the development of an unhealthy business environment in which safety is not being given an acceptable degree of priority. This is difficult because it may sometimes mean challenging the safety implications of broader (e.g. government) policy when this is likely to have a negative impact on safety.

Issues relating to external regulation which arise from the events studied include the following:

• For most of the events studied, the striking finding in relation to external regulation was its absence or weakness. In some cases there was an inadequate inspection regime (e.g. Tokai-mura), in other cases the regulator appeared to trust the judgement of the operator/licensee rather more than was justified (e.g. Paks and Davis Besse). In several instances, the relationship between the parties may have become too ‘comfortable’. This was particularly highlighted in the case of ‘good performers’ such as Davis Besse, where scrutiny was reduced but where subsequently performance appeared to have declined.

• In addition to general inspection of plant and awareness about overall plant condition and culture, the events highlight the significance of regulators being aware of major emerging engineering developments on which focussed scrutiny may be required.

• The need for regulators to take a view on the capability of the organisation as a whole in the context of such issues as leadership commitment, safety culture, the management of change, the role of contractors, the impact of the business or ‘political’ environment (as discussed above) and evidence of ‘organisational drift’. These are difficult issues which require significant judgement but are the ‘seeds’ out of which many of the events studied have grown. They are not always part of standard regulatory procedures and practice, but regulators need to ensure that they have the remit and ability to ‘test’ such issues.

• At a more practical level, for some events there had been regulatory involvement in the emerging issue but insufficient follow up of the actions taken by the operating organisation.

• Several of the events have pointed to the need for regulatory bodies to improve their own internal communication and thus to integrate better the knowledge which was available in the organisation. This appeared to be a particular issue between technical specialists and inspectors who had to agree priority actions with the operating organisation.
Finally, some of the events pointed to shortcomings in the regulatory framework. An obvious case was the need for a safety case/permissioning regime in the State of Victoria at the time of the Longford accident. A further issue may also attach to the need to identify ‘orphan’ plant or areas of risk which are currently not regulated to a level that their safety significance might suggest is appropriate.

3 PROCESS MODELLING

The key areas identified in section 2 emphasise how safety performance of a whole system (e.g. an organisation) is affected by both technical and human factors. It can be a very challenging task to assess safety performance in a way that allows a view to be developed of the ‘big safety picture’ in a clear and transparent way. Also, it is particularly difficult to capture the rationale behind the reasoning. This section investigates tools to support and help to analyse these relations.

It is suggested that a systems methodology such as that set out by Blockley and Godfrey in their book ‘Doing it Differently’ [21], can provide structure and form the basis of a new software tool to help to achieve this. In particular, a modelling technique known as hierarchical process modelling (HPM) offers a new approach to safety assessment [22]. Such an approach describes a complex process at different levels of definition and contains a rich analysis of the connectivities between different parts of the system under study. HPM is currently the subject of further research with the aim of building a tool to assess organisational safety and safety culture within an organisation, based on issues such as those discussed in Section 2 above and associated evidence gathered from inspections/questionnaires etc.

The following sections provide insight into the different concepts involved in the systems methodology and how they can be used in practice. These concepts are holons, interval probability (Italian Flag) and the hierarchical process model.

3.1 The need for a holistic approach

Assessing the safety performance of the whole system requires a holistic approach, addressing all factors such as technical systems, safety management systems, organisational safety and leadership. The approach must assess individual components of the system but also the interactions between those components.

As a tool for analysing this process, traditional reductionism has proven useful when addressing well defined physical systems but does not work well for human driven systems. In contrast, HPM is a holistic approach, capable of modelling key emergent system properties and has advantages over the traditional reductionist approach. Using hierarchical abstraction, a complex process is described at different levels of definition, as are the connectivities between different parts of that process. Also, it provides an opportunity to integrate all of these parts, both ‘hard’ and ‘soft’, within one unified framework [23].

3.2 A need for a hierarchical process model

HPM provides a detailed understanding of the top-process (i.e. the highest level process that defines the purpose of the activity) in terms of the factors that lead to the success of that process. The hierarchy elaborates these factors in increasing levels of detail. This improves transparency by enabling stakeholders to walk through the model and understand how lower level processes affect the performance of higher level processes, allowing them to discuss, elicit, evaluate and update parts of the process model. It also facilitates a ‘whole view’ of the safety system. This will enable those responsible, whether within an operating organisation or regulatory body, to demonstrate or identify deficiencies in safety performance with greater
confidence and clarity and to justify action and eventually address process safety issues which if not addressed could be precursors to events as those studied as part of this research project.

3.3 Defining a process holon

In HPM, processes are described in terms of holons. A process holon is a well-described characterisation of a process. A high level process is described as an interacting collection of lower level process holons. A holon is usually viewed as an activity which is ‘a way of getting from where you are to where you want to be’. A holon can identify both ‘hard’ as well as ‘soft’ processes and is therefore useful for complex problems involving human components.

A process holon is defined by its name and a set of attributes that define the success of the activity. The name is in a phrase form which describes what the objective of the owner of the process is. The owner is responsible and accountable for attaining the purpose successfully. The attributes of the process formed by asking six fundamental questions: ‘what’, ‘where’, ‘who’, ‘when’, ‘how’, and ‘why’. These key questions are used to provide the evidence to support judgements regarding the success of the process.

3.4 Modelling uncertainty

It is seldom possible to present the conclusions of a safety analysis with absolute certainty. It can be difficult to obtain objective quantitative data in support of conclusions, as is often the case in assessment of safety culture or human factors. One of the greatest difficulties of safety performance assessment resides in the treatment of evidence or data which might be uncertain, incomplete, inaccurate or inconsistent (or even simply missing). Because uncertainty is often unavoidable, it is important to help safety analysts to manage it within an appropriate, well founded reasoning framework. Several mathematical approaches exist to quantify uncertainty and have been discussed in various sources [24, 25]. One such approach is Interval Probability Theory [26, 27]. This can be used to handle in a simple way, fuzziness, incompleteness and randomness and has been found to be particularly appropriate for managing uncertainty in an HPM.

3.5 Performance measurement in HPM

The ‘state’ of an HPM process holon and its progress towards eventual success are defined in terms of performance measures, referred to in this context as Performance Indicators (PIs). PIs are derived from a variety of sources ranging from measurements, inspection documents and model calculations to expert elicitation. The degree of success of a process is determined by comparing the values of its PIs to target values and the results are used to assess process performance. The values of all the PIs for a single process are combined into an ‘Italian Flag’ (green-white-red) diagrammatic representation to visualise process performance. This allows non-experts to get a ‘feeling’ of how well a process is performing without knowing all of the technical and non technical details.

3.6 A simple hierarchical process model example

The theoretical ideas discussed above are under development as part of this project to design and build a practical vulnerability assessment tool. This work is not complete and is not possible yet to provide a developed organisational safety assessment, but a simple example is provided to demonstrate the key elements of the methodology. One top process and four sub processes are shown in Figure 1.

The sub processes H1 ‘Having good safety leadership’, H2 ‘Containing both risks and hazards’, H3 ‘Having commercial growth without reducing organisational safety’ and H4
‘Having good communication’ are linked to the top process H₀ ‘Maintaining a high level organisational safety performance’. The arcs are directed from the sub processes to the top process indicating that success in all the sub processes is necessary and sufficient for the success of top process H₀. More specifically, the arcs reflect the necessity (N) and sufficiency (S) relationship existing between each sub process and the associated top process. These N/S relationships are given values on a [0, 1] interval to model the relative importance the sub processes.

Performance evidence is acquired at the level of processes H₁, H₂, H₃ and H₄. The effect of this evidence on the top level process is modelled by developing an Italian Flag associated with each of H₁, H₂, H₃ and H₄. These flags are derived from the PIs associated with H₁, H₂, H₃ and H₄. Each Italian Flag bar represents an interval probability statement summarising the degree of success of a process. The performance of H₀, represented by its Italian Flag, is calculated from the Italian Flags assigned to the sub processes, and the S/N links. It is also possible to define a local PI for the top process. Comparing the Italian Flag from this with the propagated ones enables the process owner to highlight possible situations of conflict.

The amount of white in a flag indicates the level of uncertainty in assessing the success of the process. A wide white band would be an indication that, in the case of H₀ for example, an operating organisation or regulatory body needs to undertake further evidence collection in order to reduce the amount of uncertainty associated with the success of the process. Similarly, green indicates positive evidence in support of success and red indicates evidence in support of failure. The process model provides the opportunity to examine how the component parts of the flag for a process are computed thus offers key insight into the causal factors that determine the (success) of a process.

Figure 1: A hierarchical process model representing a top process with its four sub processes.

4 CONCLUSIONS

Detailed study of the reports into the recent major events across a range of industries (including several events in the nuclear industry) has allowed a range of organisational and cultural factors to be identified which seem to be common precursors to the events. It is believed that if operating organisations can develop a better understanding of these issues and a ‘vulnerability tool’ can be developed to assess the ‘health’ of organisations with respect to these factors, it may be possible to reduce the risks of so-called ‘organisational accidents’.

This paper has reviewed the issues identified for the events studied and categorised them under broad headings. A question set is being drawn up, based on the identified issues, which should be helpful in enabling regulators and operators to assess performance.
However, it is believed to be important that process modelling based on a ‘systems thinking’ approach should be developed in order to structure, clarify and potentially ‘measure’ organisational vulnerability with an appreciation of the uncertainties involved. The paper has presented some approaches which have been developed by the SSRC at the University of Bristol and describes some of the concepts and how in outline these might be combined with the identified issues to provide a hierarchical process model as part of a structured holistic approach to assess the impact of the complex, wide-ranging factors involved.

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A NEW METHOD TO RESPECT THE REAL STATE OF KNOWLEDGE ON UNCERTAINTIES IN THE EVALUATION OF SAFETY MARGINS

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ABSTRACT

This paper is devoted to some recent developments in uncertainty analysis methods of computer codes used for accident management procedures in nuclear industry. A quick overview on current practices as for uncertainty methodology is first given with a special attention devoted to the probabilistic modelling which is the most classical approach currently used by analysts. It turns out that despite its attractiveness relying on a simple implementation and convenient available tools to study the statistics of the code response, probability theory does not provide satisfactory results for uncertainty quantification in presence of incomplete knowledge or when the uncertainty is not only of aleatory nature. Therefore, a new approach, called the RaFU method, is introduced to avoid the subjectivity which may exist in the choice of a single probability distribution when it is not justified. Finally, an application of the RaFU method to uncertainty analysis of a LBLOCA transient (LOFT-L2-5) is given and a comparison with probability-based methods is provided as well. This application has been performed in the framework of the BEMUSE OECD project.

1 INTRODUCTION

Best estimate computer codes are increasingly used in nuclear industry for the accident management procedures and have been planned to be used for the licensing procedures. Unlike conservative codes, they attempt to calculate accidental transients in a more realistic way. Therefore, it becomes of prime importance, in particular for the French Institut de Radioprotection et de Sûreté Nucléaire (IRSN) in charge of safety assessment, to know the uncertainty on the results of such best estimate codes.

A large majority of uncertainty analysts uses uncertainty methodologies based on a probabilistic modelling and Monte-Carlo simulations to propagate the uncertainties through their computer codes. However, the two following limitations can reduce the efficiency of such an approach and deteriorate the relevance of the decision-making process:

- These methods require a lot of knowledge to determine the probability law associated to each uncertain parameter and all the possible dependencies between the uncertain parameters. In practice, such information is rarely fully available.
- Working within the probability theory framework implicitly assumes that all uncertainties are aleatory (i.e. due to the natural variability of an observed
phenomenon). In practice, uncertainties can arise from imprecision (a variable has a fixed value which is badly known due to the lack of data, knowledge or experiment).

Therefore, recent works have focused on new methods able to avoid the subjectivity which may exist in the choice of a single probability distribution in presence of incomplete knowledge or when uncertainties are due to imprecision. Existing methods are often computationally costly and are thus applicable to relatively simple models, which limits the efficiency of such approaches in fields (such as nuclear safety) where models can be very complex and where computational costs have to be taken into account.

We propose in this work a new numerical treatment of such methods based on Monte-Carlo sampling techniques which reduces the computational cost and can be applied to complex models. Moreover, by using notions of order statistics, our method proposes a way to estimate the numerical accuracy of the results. The key point of our work mainly consists in setting some decision step before the uncertainty propagation, whereas usual methods postpone this step after the propagation.

Section 2 gives a quick overview on current practices as for uncertainty methodology with a special attention devoted to the classical probabilistic modelling. Since it will become clear that this approach does not provide satisfactory results in many cases, we introduce, in Section 3, our new method, called the RaFU method. It allows to work within an unified framework to take into account the nature of uncertainty sources and to properly represent the real state of knowledge on uncertainties. It leads also to a numerical implementation ensuring a minimal computational cost. Finally, in the framework of the BEMUSE OECD project, an application of the RaFU method to uncertainty analysis of a LBLOCA transient (LOFT-L2-5) is given in Section 4 and a comparison with probability-based methods is provided as well.

2 UNCERTAINTY ANALYSIS AND CURRENT PRACTICES

2.1 Main steps of an uncertainty analysis

Uncertainty analysis methods are performed in four steps:

**Step 1: Identification of uncertain parameters**
All important factors affecting the model results must be identified. These factors are generally referred to as the “uncertainty sources” or as the “uncertain parameters”.

**Step 2: Quantification of the knowledge about uncertain parameters**
The available information about uncertain parameters is formalized. The uncertainty of each uncertain parameter is quantified. If dependencies are known between uncertain parameters (or classes of uncertain parameters) and judged to be potentially important, they also need to be specified.

**Step 3: Propagation of uncertainties through the computer code**
The propagation requires, except for very simple computer codes, a coupling between the code and a statistical software.

**Step 4: Treatment and interpretation of the code responses**
The code responses are used to get quantitative insights regarding the output variable.
For example, in risk studies, the main concern is to estimate the likelihood of the code response to be above a critical value.

When performing practical studies, the two following requirements have to be respected in order to guarantee a relevant uncertainty evaluation:

- The method has to respect the state of knowledge in the quantification of the information about uncertain parameters (Step 2 in the sketch of the uncertainty propagation methods).
- The method has to lead to a tractable algorithm for uncertainty evaluation (Step 3 and 4 in the sketch of the uncertainty propagation methods) i.e with a reasonable computational cost.

A classical method for uncertainty analysis is the one based on the probabilistic approach. This modelling, as well as its advantages and limitations are recalled in the following section.

2.2 Probabilistic Modelling

2.2.1 Construction of the probabilistic modelling

Here, we specify Steps 2, 3 and 4 within the probabilistic modelling:

Quantification of the knowledge about uncertain parameters

The uncertainty of each uncertain parameter is quantified by a probability density function (pdf). If dependencies between uncertain parameters are known and judged to be potentially important, they are quantified by correlation coefficients.

Propagation of uncertainties through the computer code

The numerical estimation is obtained thanks to Monte-Carlo simulations ([1]). In Monte-Carlo simulation, the computer code is run repeatedly, each time using different values for each of the uncertain parameters. These values are drawn from the probability distributions and dependencies chosen in the previous step. In this way, one value for each uncertain parameter is sampled simultaneously in each repetition of the simulation. The results of a Monte-Carlo simulation lead to a sample of the same size for each output quantity.

The advantage of Monte-Carlo methods with respect to deterministic uncertainty analysis methods (i.e. that do not involve stochastic or statistical approaches) is that the combination of uncertain parameters values are performed in such a manner that they allow to quantify the likelihood of any combinations of parameter values.

Treatment and interpretation of the code responses

Using the central-limit theorem ([2]), the output sample is used to get any typical statistics of the code response such as mean or variance and to determine the cumulative distribution function (CDF). The CDF allows to derive the percentiles of the distribution (if X is a random variable and F\_X its CDF, the \( \alpha \)-percentile, \( \alpha \in [0;1] \), is the deterministic value \( X_{\alpha} \) such that \( F_X(X_{\alpha}) = \text{Proba}(X \leq X_{\alpha}) = \alpha \). Its estimation is crucial for safety assessment since the CDF allows to estimate whether the code response can exceed a critical value.

A simple and robust way to get information on the CDF is to use order statistics ([2]). The principle of order statistics is to derive statistical results from the ranked values of a sample. If \( X=(X^{(1)},...,X^{(L)}) \) denotes the output sample, the key idea is that the cumulative distribution of
\(X^{(k)}, F_X(X^{(k)})\), follows the Beta law \(\beta(k,L-k+1)\) which does not depend on the distribution of \(X\). Therefore, it is possible to derive confidence intervals for any percentiles directly from the sample values without having to determine the probability distribution of the random variable. This relevant result is very popular in the safety assessment community. It is often used in two ways: when the sample size is fixed, it provides the numerical accuracy (due to the finite sample size) associated to the estimation. It also gives for a fixed numerical accuracy the minimal sample size (and therefore the minimal number of computer runs) to perform in order to reach this accuracy. The connection between accuracy and minimal sample size is often quoted as the Wilk’s formula.

### 2.2.2 Advantages and limitations of the probabilistic approach

The probabilistic model is simple to implement thanks to the uncertainty propagation by Monte-Carlo simulations. Moreover, the use of order statistics provides both simple and robust estimators of percentiles for any output quantities without using response surfaces or fit tests. However, it assumes that each uncertain parameter can be modeled by a random variable (i.e, the uncertainty is due to the natural variability and cannot be reduced by the arrival of new information, it is referred to “aleatory uncertainty” in the sequel). This is not true in many applications:

- When the uncertain parameter is measured through an experimental design leading to systematic errors, a pdf is not adapted to the modelization of such uncertainties which are due to imprecision instead of variability.
- Even in the case of random uncertain parameters, choosing an unique pdf and specifying all the possible dependencies between the uncertain parameters is not always affordable since the knowledge related to uncertainties is often incomplete. In practice, following a principle of minimal information, the engineers select an uniform distribution as pdf when only the uncertainty range of the parameter is known and take an independence assumption between two uncertain parameters for granted when no information is known about their dependencies. Uniformity means equiprobability of any values within the uncertainty range which is not justified when knowing only the uncertainty range. Independence implies that it is unlikely to have simultaneously extreme values between random variables and often leads to uncertainty compensation. Therefore, these assumptions do not lead to conservative results and do not meet the precautionary principle when poor information is known.

One speaks about epistemic uncertainty when the choice of a single pdf cannot be assumed or known. This type of uncertainty can be reduced by increasing the state of knowledge. Therefore, it comes out that the probabilistic modelling is not tailored to handle all the practical issues coming from safety assessment applications. It may lead to an unjustified reduction of the final uncertainty of the model response and affect the decision-making process in risk studies. Indeed, in the worst case, because of such an artificial reduction, the decision maker could underestimate the risk and accept a too high level of risk but a more relevant quantification of uncertainties (i.e, another choice of pdf and dependency assumption or another modelization to take into account imprecision) would have shown that the code response is likely to exceed the critical value. For safety reason, it becomes of prime importance to provide a new methodology that gives the engineer a tool to measure the impact of a misleading modelization of uncertainty due to poor knowledge.
Therefore, we propose in the rest of this paper a new method for uncertainty evaluation (called the RaFU method). It allows us to mix different kinds of knowledge representation in order to respect the available information about uncertain parameters and about the nature of their uncertainty. It also integrates an efficient numerical strategy to reduce the computational cost to its minimum.

3 THE RAFU METHOD

We describe in the sequel Steps 2, 3 and 4 within the RaFU modelling.

3.1 Quantification of the knowledge about uncertain parameters

The RaFU ([3]) method allows to handle two kinds of uncertainties: aleatory and epistemic uncertainties. As mentioned previously, aleatory uncertainty is due to the natural variability or randomness of an observed phenomenon. This kind of uncertainty cannot be reduced by new information and its modelling by a pdf is well appropriate. Epistemic uncertainty is due to imprecision or lack of knowledge and thus can be reduced by the arrival of new information. It can come from systematical error such as measurement error but also from the scarcity of information about random phenomena that prevents from choosing an unique pdf for the uncertainty modelization. Possibility theory ([4]) provides an attractive framework to quantify this second type of uncertainty. A possibility distribution is well fitted to the situation where a given variable has a fixed value but badly known. Moreover, if $\pi$ denotes a possibility distribution, it induces the set of probabilities $P_\pi = \{P/\forall A, P(A) \leq \sup_{x \in A} \pi(x)\}$, which is particularly well fitted to represent a badly-known random phenomenon due to an incomplete state of knowledge. In this sense, a possibility distribution can be seen as a model of partial probabilistic information. For example, it can be proved that the probability set induced by a trapezoidal possibility distribution (Figure 1, right) contains all the probabilities with the same core (i.e the most likely values are located in the same interval, [2;4] in our example) and the same support ([1;7]). In other words, if the uncertainty attached to a parameter P is summarized by its range of variation ([1;7]) and an interval of values within this range that P is more likely to take ([2;4]), then the trapezoidal possibility distribution of Figure 1, right, can be chosen for uncertainty quantification. Similarly, the set of pdfs induced by triangular possibility distributions (Figure 1, left) contains all the pdfs with the same mode (i.e. the same most likely value, 3 in our example) and the same support ([2;4]). It is therefore well fitted when the information related to P is its range of variation ([2;4]) and its nominal value (3).

![Figure 1: Example of possibility distributions. Left, triangular possibility, right, trapezoidal possibility.](image-url)
It turns out that the possibility theory is a convenient way to quantify epistemic uncertainty but it can lead also to unrealistic uncertainty margins when, in the case of aleatory uncertainty, enough information is available to select an unique pdf. Therefore, our method allows the analyst to select a probability distribution or a possibility one with respect to the amount of information about uncertain parameters and to the nature of uncertainty. This is achieved by working in an unified framework for probability and possibility called the theory of evidence ([5]). In the same way that the probability theory assigns weights to the different values taken by each uncertain parameter (for example all the points within the core of the trapezoidal distribution), the idea of the theory of evidence is to put weights on subset of values (and not necessarily on single values) such as intervals.

A remaining crucial question concerns the choice (and the simulation) of dependencies between uncertain parameters. Indeed, the classical independence assumption that is taken when few information about dependencies is available is not always justified. It can lead to compensate uncertainties and can affect the uncertainty margins. Based on precautionary approach, it becomes important to allow uncertainty accumulation if there is no information about compensating effect. This can be achieved within the RaFU method thanks to a special propagation strategy.

Therefore, our new method is derived to allow the engineer to answer independently to these two following questions:

- **Probability or possibility?**
  In the case of epistemic uncertainty, possibility distributions are used in order to relax assumptions currently made on the choice of the pdfs associated to some uncertain parameters. For aleatory uncertainty, specific pdfs can also be used if a substantial amount of information is available, limiting the over-conservatism encountered in standard interval calculations.

- **Compensation (independence) or accumulation of uncertainties (ignorance of dependencies)?**
  When the amount of information to ensure compensating effects is sufficient, one might assume independence whereas simulate uncertainty accumulation in the case of poor information.

### 3.2 Propagation of uncertainties through the computer code and treatment of the responses

The propagation is based on an extension of Monte-Carlo simulations and therefore first requires a sampling of random (aleatory uncertainty)/badly-known (epistemic uncertainty) variables. Note that sampling a random variable gives a value (Figure 2, left) (as in classical Monte-Carlo simulations). As for badly-known variables, the sampling (Figure 2, right) is performed on the possibility distributions associated to each variable. We focus in this work on convex possibility distributions which are the most encountered in practical studies. Therefore, sampling badly-known variables leads to a set of nested intervals called α-cuts (a set of nested intervals \( \{I_{\alpha}, 0 \leq \alpha \leq 1\} \) satisfies \( \forall \alpha \in ]0;1[, I_{\alpha} \subset I_{\alpha -} \subset I_0 \)).
It is therefore similar to performing Monte-Carlo simulations on intervals (i.e. calculations are performed with values at the extreme of each sampled interval). Monte-Carlo techniques offers also an attractive framework for the simulation of dependency. More precisely, for uncertainty compensation, the sample is constructed by randomly combining values/interval drawn from probability/possibility distributions. When assuming uncertainty accumulation, this combination is performed at the same confidence level. Since both probability and possibility are used to model parameters, the output of the computer code is no more a random variable but a random fuzzy variable. This also implies that the uncertainty derived by this methodology cannot be summarized by a pdf (or a CDF) but by a pair of lower and upper CDFs, \([\bar{F}, F]\), called probability boxes ([6]). The difference between these CDFs comes from the lack of knowledge modeled by possibility distributions.

There exist many recent works that handle, like the RaFU method, both aleatory and epistemic uncertainties and derive a pair of CDFs. Among them, one can mention the work of Ferson and Ginzburg ([7]) and Baudrit et al ([8]) that propose a post-processing technique to extract the relevant information from the resulting random fuzzy variable. Up to now, they concern very simple models or are computationally costly, which limits their efficiency in fields such as nuclear safety where computational cost has to be taken into account.

On the contrary, the RaFU methodology integrates a computational cost reduction strategy. The underlying idea is that in many studies, analysts are interested in some particular statistical summary (such as \(\alpha\)-percentiles) which can be evaluated without building the whole random fuzzy variable. Since the RaFU propagation can be seen as an extension of Monte-Carlo simulation to the theory of evidence framework, each statistical quantity of interest is directly estimated using standard results coming from the probabilistic modelling. Moreover, one can exploit convergence theorems to derive the numerical accuracy associated to the limited sample size. The computational cost reduction strategy of the RaFU is then to set a decision step before propagating uncertainties and leading to an optimal (in term of number of code runs) sampling. More precisely, the RaFU method is pre-defined by a triplet of parameters \((\gamma_S, \gamma_E, \gamma_A)\) specified by the analyst:

- Parameter \(\gamma_S\) is related to the aleatory uncertainty. It provides the statistical quantity the analyst is interested in (usually \(\alpha\)-percentiles in safety studies).
- Parameter \(\gamma_E\) is related to the epistemic uncertainty. It determines how \(\alpha\)-cuts are drawn from possibility distributions.
- Finally, parameter \(\gamma_A\) measures the desired numerical accuracy on the final result. In the case of \(\alpha\)-percentile estimation, \(\gamma_A\) comes from the use of order statistics.
According to the analyst, the RaFU method then determines the minimal sample size and the nature of the required sampling to build the wished response. Number of calculations is thus reduced to its minimal number, in accordance with the analyst’s choice. Moreover, computational cost can be easily evaluated, allowing the analyst to eventually revise her/his choices before uncertainty propagation. It is also possible for her/him to provide the maximal number of code runs that can be made and the RaFU method will derive the numerical accuracy that can be reached.

Let us come back in detail on the sampling procedure that plays a key role in the implementation of the RaFU method once the triplet \((\gamma_S, \gamma_E, \gamma_A)\) has been chosen by the analyst. This procedure is fully specified in the following example:

Let \(P_1, \ldots, P_N\) be the \(N\) uncertain parameters (identified by a preliminary sensitivity analysis for example) of a computer code. We denote by \(T\) the output variable and for sake of simplicity, \(T\) is assumed to be monotonous increasing with respect to \(P_1, \ldots, P_N\). Moreover, we consider that, according to expert’s judgement for example, the uncertainty associated to \(P_1, \ldots, P_k\), \(k < N\), is aleatory (i.e. quantified by a pdf) whereas the uncertainty related to \(P_{k+1}, \ldots, P_N\) is of epistemic nature (i.e. quantified by a possibility distribution). Once the analyst has chosen the triplet \((\gamma_S, \gamma_E, \gamma_A)\), the sampling procedure is performed in three steps:

1. Generate \(L\) samples \(X^{(i)} = (P_1^{(i)}, \ldots, P_k^{(i)})\), \(i = 1, \ldots, L\), for the \(k\) first random variables according to the identified pdfs and the dependencies between uncertain parameters (\(L\) denotes here the number of samples chosen by the analyst or associated to the selected numerical accuracy, \(\gamma_A\), to reach).

2. Associate to each \(X^{(i)}\) one sample of intervals corresponding to the \(N-k\) badly known variables \(P_{k+1}, \ldots, P_N\) and denoted \(I^{(i)} = ([P_{k+1}^{(i)}, P_{k+1}^{(i)}], \ldots, [P_N^{(i)}, P_N^{(i)}])\). Each sample is constructed according to the identified possibility distributions and to the sampling strategy of epistemic uncertainty represented by Parameter \(\gamma_E\).

3. Propagate the two sets of samples \(\underline{S}^{(i)} = (P_1^{(i)}, \ldots, P_k^{(i)}, P_{k+1}^{(i)}, \ldots, P_N^{(i)})\) and \(\overline{S}^{(i)} = (P_1^{(i)}, \ldots, P_k^{(i)}, P_{k+1}^{(i)}, \ldots, P_N^{(i)})\) \((i = 1, \ldots, L)\) through the computer code and get the two corresponding output samples \((T_1^{(i)}, \ldots, T_L^{(i)})\) and \((\overline{T}_1^{(i)}, \ldots, \overline{T}_L^{(i)})\).

Figure 4 displays a flowchart of the RaFU method.
Figure 4: Flowchart of the RaFU method.

4 APPLICATION OF THE RAFU METHOD TO UNCERTAINTY ANALYSIS OF A LBLOCA TRANSIENT (LOFT-L2-5)

4.1 Description of LOFT L2-5

The Loss-of-Fluid Test (LOFT) facility (Figure 5) simulated the major components and the system responses of a commercial PWR during a loss-of-coolant accident (LOCA). The core was a semi-scale one with an active height of 1.70m. The experimental assembly included five major subsystems which were instrumented such that system variables can be measured and recorded. The L2-5 experiment has been successfully completed on June 16, 1982 in the LOFT facility at INEL (Idaho National Engineering Laboratory). This experiment simulated a guillotine rupture of an inlet pipe in a pressurized water reactor with a true nuclear core. The experiment L2-5 was initiated, after operating the reactor at 36.0 MW for 40 effective full power hours to build up a fission decay product inventory, by opening two quick-opening...
blow-down valves upstream a blowdown suppression tank simulating the reactor containment behavior.

Since the L2-5 experiment, several computations have been conducted to simulate the behavior of the LOFT system and to compare with experimental results. Among them, one can mention the OECD/CSNI program on Best Estimate Methods for Uncertainty and Sensitivity analysis (BEMUSE) whose goal is to apply uncertainty methodologies to a Large Break Loss Of Coolant Accident (LB-LOCA transient) performed on an integral test facility [9].

4.2 Uncertain parameters

After sensitivity analysis, a list of 27 uncertain parameters (Table 1) has been proposed by IRSN. Table 1 provides also the available information related to the range of variation and the nominal value associated to each parameter.
If we choose a probabilistic modelization, the state of knowledge does not allow to identify an unique probability law to represent the uncertainty attached to each uncertain parameter. According to Table 1, only the support (i.e. the range of variation) and the mode (i.e. the nominal value) can be derived. Therefore, the partial probabilistic modelling of the RaFU framework turns out to well fitted to this situation. In the next section, we propose several numerical tests to illustrate the capabilities of the RaFU method. Our goal is here to show how it can provide robust margins to the assumptions related to the choice of an unique pdf in classical Monte-Carlo simulations. The question of computational cost is also fully detailed in the sequel.

### 4.3 Numerical tests

We focus on the uncertainty analysis of the first peak cladding temperature (PCT) of a hot rod in a hot channel. A preliminary study performed within the probabilistic framework has shown that selecting the three most influential uncertain parameters (i.e. Parameters N°12, 18 and 20 in Table 1) leads to very similar uncertainty margin estimations as in the case of 27 parameters. Therefore, we only consider in these numerical tests the 3 most influential ones. However, the RaFU method is not limited to small numbers of uncertain inputs and can be applied in higher dimension. In order to show the capability of our approach, the uncertainty attached to PCT is estimated using both probabilistic and RaFU modellings. In the first series

<table>
<thead>
<tr>
<th>N°</th>
<th>Phenomenon</th>
<th>Nom. value</th>
<th>Range</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Critical heat flux</td>
<td></td>
<td>[0.8 ; 1.2]</td>
</tr>
<tr>
<td>2</td>
<td>Interface to liquid heat flux - flashing</td>
<td></td>
<td>[0.05 ; 1]</td>
</tr>
<tr>
<td>3</td>
<td>Interface to liquid heat flux - &quot;Shah&quot; correlation</td>
<td>1</td>
<td>[0.1 ; 1]</td>
</tr>
<tr>
<td>4</td>
<td>Interface to liquid heat flux - stratified flows</td>
<td>1</td>
<td>[0.3 ; 3]</td>
</tr>
<tr>
<td>5</td>
<td>Minimum stable film temperature</td>
<td>1</td>
<td>[-42 ; 60]</td>
</tr>
<tr>
<td>6</td>
<td>Interface to liquid heat flux - turbulences induced by injection</td>
<td>1</td>
<td>[0.3 ; 3]</td>
</tr>
<tr>
<td>7</td>
<td>Interface to liquid heat flux - droplet flows standard model</td>
<td>1</td>
<td>[0.3 ; 3]</td>
</tr>
<tr>
<td>8</td>
<td>Condensation by injection of under-saturated water</td>
<td>1</td>
<td>[0.5 ; 2]</td>
</tr>
<tr>
<td>9</td>
<td>Vapor wall heat flux - vaporization, Vapor wall heat flux - condensation</td>
<td>1</td>
<td>[0.5 ; 2]</td>
</tr>
<tr>
<td>10</td>
<td>Interfacial friction - annular flows</td>
<td>1</td>
<td>[0.5 ; 2]</td>
</tr>
<tr>
<td>11</td>
<td>Interfacial friction - stratified flows</td>
<td>1</td>
<td>[0.5 ; 2]</td>
</tr>
<tr>
<td>12</td>
<td>Liquid-wall friction</td>
<td>1</td>
<td>[0.8 ; 1.9]</td>
</tr>
<tr>
<td>13</td>
<td>Interfacial friction downstream quench front</td>
<td>1</td>
<td>[0.29 ; 3.4]</td>
</tr>
<tr>
<td>14</td>
<td>Vapour-wall friction</td>
<td>1</td>
<td>[0.8 ; 1.9]</td>
</tr>
<tr>
<td>15</td>
<td>Interfacial friction (churn-bubble flows) in pipe geometry</td>
<td>1</td>
<td>[0.2 ; 10]</td>
</tr>
<tr>
<td>16</td>
<td>Interfacial friction (churn-bubble flows) in assembly geometry</td>
<td>1</td>
<td>[0.6 ; 1.8]</td>
</tr>
<tr>
<td>17</td>
<td>Interfacial friction (churn-bubble flows) in annular geometry</td>
<td>1</td>
<td>[0.5 ; 2]</td>
</tr>
<tr>
<td>18</td>
<td>Vapour-wall heat transfer (forced convection regime)</td>
<td>1</td>
<td>[0.5 ; 2]</td>
</tr>
<tr>
<td>19</td>
<td>Vapour-wall heat transfer (natural convection regime)</td>
<td>1</td>
<td>[0.5 ; 2]</td>
</tr>
<tr>
<td>20</td>
<td>Film-boiling (Berenson/Bryce) Standard model</td>
<td>1</td>
<td>[0.15 ; 6.5]</td>
</tr>
<tr>
<td>21</td>
<td>Interface-wall heat transfer downstream quench front</td>
<td>1</td>
<td>[0.5 ; 2]</td>
</tr>
<tr>
<td>22</td>
<td>Liquid-wall heat transfer (nucleate boiling) Standard model</td>
<td>1</td>
<td>[0.5 ; 2]</td>
</tr>
<tr>
<td>23</td>
<td>Liquid-wall heat transfer Laminar/turbulent forced convection</td>
<td>1</td>
<td>[0.8 ; 1.2]</td>
</tr>
<tr>
<td>24</td>
<td>Fluid-wall heat transfer (2D conduction near quench front)</td>
<td>1</td>
<td>[0.5 ; 2]</td>
</tr>
<tr>
<td>25</td>
<td>Droplets fall velocity</td>
<td>1</td>
<td>[0.5 ; 2]</td>
</tr>
<tr>
<td>26</td>
<td>Bubbles rise velocity</td>
<td>1</td>
<td>[0.4 ; 5]</td>
</tr>
<tr>
<td>27</td>
<td>Phases distribution coefficient in volumes</td>
<td>1</td>
<td>[0.85 ; 1.15]</td>
</tr>
</tbody>
</table>
of tests, we assume that for computational cost reason, the analyst intends to build a sample of fixed size and we compare both probabilistic and RaFU methods in term of capability to represent uncertainty according to the real state of knowledge. In the second one, we focus on the numerical treatment of the RaFU approach and show how it can drastically reduce the computational cost associated to existing methods handling both aleatory and epistemic uncertainties such as [7] and [8]. In every test, the computer code is CATHARE V2.5 mod 6.1 ([10]). The statistical treatment is performed with the software SUNSET ([11]) developed by IRSN.

4.3.1 First series of tests

Uncertainty quantification

As mentioned in Section 4.2, due to the lack of information, there exists several suitable families of pdfs to modelize the uncertainty attached to each parameter. When only the support (i.e. the range of variation) is known, an uniform probability distribution (Figure 6, left) is classically chosen: it corresponds to a minimal state of knowledge. It assumes that each value in the support is likely to be taken by the uncertain parameter. When the support and the mode (i.e. the most likely value or the nominal value in our study) are known (which is the case for many parameters of Table 1), the previous modelization does not emphasize that one particular value within the support (i.e the mode) is more likely to be taken. Therefore, analysts often switches to histogram distributions with the 50%-percentile for the nominal value (Figure 6, right). This is this last type of law that is considered in the sequel.

Within the RaFU framework, each uncertain parameter follows a triangular possibility distribution (Figure 1, left). Note that the family of pdfs encoded by each triangular possibility encompasses the histogram distribution chosen in the previous modelization. The probabilistic and RaFU modelings are compared assuming first uncertainty compensation then uncertainty accumulation. Tables 2 summarizes the uncertainty quantification.
Table 2: Uncertainty quantification.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Probabilistic modeling (Aleatory uncertainty)</th>
<th>RaFU modeling (Epistemic uncertainty)</th>
</tr>
</thead>
<tbody>
<tr>
<td>PHBO (n°20) Nom. Val.: 1, Range: [0.15;6.5]</td>
<td>Histogram probability distribution</td>
<td>Triangular possibility distribution</td>
</tr>
<tr>
<td>PHCFV (n°18) Nom. Val.: 1, Range: [0.5;2]</td>
<td>Histogram probability distribution</td>
<td>Triangular possibility distribution</td>
</tr>
<tr>
<td>PI1CLx (n°12) Nom. Val.: 1, Range: [0.8;1.9]</td>
<td>Histogram probability distribution</td>
<td>Triangular possibility distribution</td>
</tr>
</tbody>
</table>

Uncertainty propagation and statistical treatment

The propagation is performed by Monte-Carlo simulations from probability distributions (probabilistic modeling) and also from possibility ones (RaFU). The statistical quantity of interest is the 95%-percentile which is the most relevant quantity to estimate in safety studies. Table 3 summarizes the 3 parameters required for the RaFU propagation.

Table 3: The three parameters of the decision step within the RaFU modeling.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Probabilistic modeling</th>
<th>RaFU modeling</th>
</tr>
</thead>
<tbody>
<tr>
<td>γS</td>
<td>0.95</td>
<td>0.95</td>
</tr>
<tr>
<td>γE</td>
<td>random α-cut for each badly-known parameter within each sample (Uncertainty compensation)</td>
<td>α-cut identical for all badly-known parameters within each sample (Uncertainty accumulation)</td>
</tr>
<tr>
<td>γA</td>
<td>95%-accuracy #samples=200</td>
<td>95%-accuracy #samples=200</td>
</tr>
</tbody>
</table>

The numerical tests allow to derive an estimation of the 95% percentile within the probabilistic modelling and a couple of type [95%min,95%max] when choosing the RaFU approach (Figure 7). The lower (resp.upper) 95%-percentile corresponds to the “most optimistic” (resp. the “most pessimistic”) choice of pdf according to the real state of knowledge.

Figure 7: Estimation of the 95%-percentile with the probabilistic and the RaFU modellings.
As expected, the 95% percentile obtained with the probabilistic method is lying between the lower and upper percentiles derived from our approach as for the same assumptions about uncertainty compensation or accumulation. This figure illustrates two main effects that need to be taken into account if a decision-making process follows the uncertainty analysis:

- Effect of the choice of pdfs: a difference of 100K on the 95%-percentile is noticeable between all pdfs with same mode and support.
- Effect of the assumptions related to compensation or accumulation of uncertainties: assuming uncertainty accumulation (which is similar to ignoring dependencies) instead of uncertainty compensation (i.e independence between parameters) leads to a difference of 30K in the estimation of the 95%-percentile.

4.3.2 Second series of tests

Contrarily to the previous section, the number of sample within the RaFU modeling is not fixed here by the user but automatically derived to reach a given accuracy (95%-accuracy on the estimation of the 95%-percentile in our case).

Table 4 provides a comparison between our methodology and the classical approaches of [7] and [8] in term of sample size. We recall that [7] and [8] assume that the whole random fuzzy variable has been built before deriving the statistical quantity of interest. This construction requires Monte-Carlo simulations for each $\alpha$-cut of the possibility distributions (the number of $\alpha$-cuts is set to 21 in this test, i.e $\alpha=(0,0.05,\ldots,1)$).

<table>
<thead>
<tr>
<th>Methods</th>
<th>Classical methods</th>
<th>RaFU</th>
</tr>
</thead>
<tbody>
<tr>
<td>Baudrit et al</td>
<td>#sample=1239</td>
<td>#samples=59</td>
</tr>
<tr>
<td>Ferson and Ginzburg</td>
<td>#sample=1239</td>
<td>#samples=118</td>
</tr>
</tbody>
</table>

Clearly, the RaFU method, knowing the desired final quantity before propagation and exploiting relevant results from order statistics, reduces drastically the number of calculations to perform: a factor $\sim$20 in the case of Baudrit et al and $\sim$10 in the case of Ferson and Ginzburg.

5 CONCLUSION

A new approach, called the RaFU method, has been constructed and applied in this paper to uncertainty analysis in presence of incomplete knowledge. Its construction is based on the theory of evidence framework that allows to handle both aleatory and epistemic uncertainties in order to respect the real state of knowledge. It is coupled with an optimal numerical treatment (based on an extension of Monte-Carlo simulations to the theory of evidence framework and on the introduction of a decision step before the propagation) that minimizes the required computation and allows the analyst to possibly revise her/his desires. Moreover, this method offers a way to control the numerical accuracy of the result. The RaFU method has successfully been applied to the uncertainty analysis of a LBLOCA transient (LOFT-L2-5). It came out that this new approach provides robust uncertainty margins related to the assumptions about pdfs that are required within the classically probabilistic modeling. These results are less precise (i.e an interval instead of a value) but are more reliable for safety
studies. The effect of pdf choices can be evaluated. Moreover, thanks to its numerical strategy, it is well fitted to uncertainty analysis of complex computer codes such as CATHARE where computer cost has to be taken into account. In the frame of the BEMUSE OECD program, it is currently tested to derive uncertainty margins for a real nuclear power plant in case of LBLOCA.

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NUCLEAR SAFETY RISK-INFORMED DECISION MAKING

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ABSTRACT

The Discipline:

Nuclear Safety Risk-Informed Decision Making (RIDM) is a discipline. It involves considering, weighing, and integrating often complex inputs and insights from traditional nuclear safety engineering (deterministic) analyses, nuclear safety probabilistic analyses, operational experience, compensating or mitigating measures, or other pertinent considerations. It considers each aspect in context with each other aspect and in context with the whole. It assesses conformance to guidance or criteria. It involves assessing safety or risk. It involves a way of thinking to integrate such inputs, insights, and assessments to result in safe, sound, and optimum management or operational actions or decisions.

The international nuclear community increasingly recognizes and emphasizes that probabilistic safety assessment (PSA) and other nuclear safety risk evaluations provide extremely valuable complementary insight, perspective, comprehension, and balance to deterministic safety assessment (DSA) of nuclear installations.

The Guidance:

Accordingly, there is a need to establish international standards for RIDM. The International Atomic Energy Agency is working to fulfill such need through its international Safety Standards. One of such standards, the developing high level RIDM Safety Guide and its implementation program will provide guidance to Member States on how to adequately and responsibly establish an infrastructure to perform, document, report, track the results of, and to follow-up on RIDM.

There is good consensus for the RIDM Safety Guide among the 35 delegates from the 18 Member States, the OECD/NEA, and the IAEA who have contributed to it. The delegates represent nuclear regulators, nuclear power plants (NPP), and nuclear safety support organizations. RIDM continues to be a critical aspect for nuclear safety in mature and maturing Member States. In parallel, there is a need to transfer that discipline to Member States which are emerging into the nuclear energy technology arena.

The Implementation:

A number of Member States, their regulators and NPPs, have expressed interest in being pilots for the RIDM Safety Guide and its implementation program. Similarly, it will be important to assist other Member States. Accordingly, the IAEA, in the implementation program, is pursuing a service to assess organizational culture, to implement RIDM, and to implement specific RIDM applications. This service will include progressive training as well as independent audits and coaching of Member States’ progress and success.
1 INTRODUCTION

While many individuals, organizations, and nations appreciate, support, and use RIDM, there appears to be a genuine need for continued safety culture enrichment with respect to RIDM and its benefits within the international nuclear community. This enrichment would especially benefit that part of the community not routinely involved in RIDM. As with any cultural development, safety culture enrichment involves change. Culture change requires first a recognition of a need and then time and effort to satisfy the need. A critical aspect for culture enrichment is vision. A nation, national organization, or private organization needs to see and appreciate the end point of such enrichment in order to build consensus in the correct direction.

During an organizational culture survey of a nuclear engineering staff at an internationally respected organization, one engineer offered a very significant comment. He said that while he appreciated that RIDM key elements were to be integrated, he did not know how his deterministic work could really contribute, and he did not know how to accomplish integrating it with risk assessment. Another individual expressed that he felt that when risk considerations entered the picture, they took precedence over deterministic considerations. It appears as if such perspectives are not isolated, and some national regulators, NPPs, or support organizations remain cautious about the value of RIDM. However, such perspectives appear to be in the minority, and they become more favourable toward RIDM following a genuine and mutual interchange of information.

Because of this awareness, the IAEA is working to provide informative guidance and appropriate support through programs, assessment missions, and information access.

2 DISCIPLINE

RIDM is a discipline. It starts with a need, an issue or situation for which a decision is required. It involves considering, weighing, and integrating often complex inputs and insights from deterministic analyses, probabilistic analyses, operational experience, compensating or mitigating measures, or other pertinent considerations. It considers each aspect in context with each other aspect and in context with the whole. It assesses conformance to guidance or criteria. It involves assessing safety or risk. It involves a way of thinking to integrate such inputs, insights, and assessments to result in safe, sound, and optimum management or operational actions or decisions. The discipline, while having static aspects, is fundamentally dynamic or fluid and is sensitive to responsible long term and near term feedback; it is ongoing. Responsible feedback can and should influence management or operational decisions or actions, previous decisions or actions, or the implementation of the discipline itself. Key elements of RIDM and their interrelationships are depicted in Figure 1 below.

Figure 1 shows broadly the considerations to be integrated and evaluated given a need (the yellow box) for which a decision is needed. The figure shows each of the RIDM Key Elements—Defense-in-Depth, Safety Margins, Risk Assessment, Performance Monitoring, and Regulatory Compliance—to be considered (the blue boxes), and it shows that each Key Element has an impact on and is impacted by each of the other Key Elements (the green and orange lines). Next the Figure shows that the integrated output from the Key Elements is evaluated with respect to its impact on deterministic and probabilistic insights, considerations, guidance, or criteria. Finally the Figure indicates that the insights from such disciplined thinking are focused to reach a decision.
RIDM is a progressively developing discipline. Experience indicates that it is effective in refining and improving safe operations of nuclear installations. It has also proved to be effective in providing an appropriate balance among operational, maintenance, and design strategies and decisions.

3 GUIDANCE

The IAEA provides high level guidance through its three tier Safety Standards Series documents. The three tiers are fundamentals, requirements, and guides. In support of RIDM, the IAEA is working on four documents: One is a draft requirements document, Safety Assessment for Facilities and Activities [1]. Three are draft Safety Guides: Risk Informed Decision Making [2], Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants [3], and Development and Application of Level 2, Probabilistic Safety Assessment for Nuclear Power Plants [4].

Associated with the IAEA Safety Standards Series documents, an overall assessment mission document, the RIDM-International Review (RIDM-IR), is in preparation.

Recently, the WWER Regulators Forum PSA Working Group, representing nine nations, emphasized that the international nuclear community increasingly wants to and needs to look to IAEA guidance. Accordingly it encouraged the IAEA management to continue to support the issuance of the RIDM Safety Guide and its associated implementation program. Similarly it encouraged the management to support the issuance of the Safety Guides on PSA, Level 1 and Level 2. These documents and program are viewed as very important.

In addition to IAEA guidance, each nation or organization implementing RIDM should have its own high level guidance. Such guidance may be based on IAEA guidance;
however, it should include and conform to national or organizational vision, principles, policies, and regulations.

4 PROGRAM

As mentioned above, each nation or organization implementing RIDM needs programmatic controls. The highest level in the hierarchy of programmatic controls should be a detailed upper tier description of RIDM, principles involved, and national policies and requirements. Under this upper tier document, there should be more detailed guidance. The IAEA Requirements document and Safety Guides mentioned above could provide a solid foundation for this program.

Lower level program elements should provide for development, review, approval, control, maintenance, revision, and security of RIDM tools. Included among such tools is software. The software includes probabilistic safety assessment (PSA) programs, installation-specific PSA models, Configuration Risk Analyzers (CRA) [also known as a Risk Monitors, Safety Monitors, etc.], and deterministic safety assessment codes. Also included among such tools is the associated computer hardware. The program should provide controls to assure that the various models and codes analyze the actual installation configuration (real time) or the actual configuration under consideration. The program should also control access to the tools, their inputs, and their outputs.

The program should provide for personnel training and qualification in several categories as follows:

- Personnel who work directly with the tools, their inputs, and their outputs.
- Personnel who access and directly use the analytical outputs and results.
- Personnel who are not directly involved in the analytical outputs and results but who receive such outputs and insights to make recommendations or decisions.
- Management.

The program should provide for the development, review, approval, distribution, control, and use of procedures for specific program aspects.

A very important set of procedures would be for specific RIDM applications. Each application should be implemented through a specific procedure or set of related procedures. Current applications are shown in Table 1 below, and additional applications continue to emerge. While almost all RIDM applications are currently at nuclear power plants (NPP), a number of non-NPP applications appear to be under consideration.

Table 1: RIDM Applications

<table>
<thead>
<tr>
<th>Nuclear Power Plants</th>
<th>Configuration Management</th>
</tr>
</thead>
<tbody>
<tr>
<td>Plant Vulnerability Assessment</td>
<td>Configuration Risk Analyzer (CRA)</td>
</tr>
<tr>
<td>Design &amp; Design Review</td>
<td>Technical Specifications</td>
</tr>
<tr>
<td>Plant Improvement</td>
<td>Maintenance &amp; Outage Planning, Assessment, and Management</td>
</tr>
<tr>
<td>Program Improvement</td>
<td>Regulatory</td>
</tr>
<tr>
<td>In-Service Inspection</td>
<td>Regulation Development</td>
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<tr>
<td>In-Service Testing</td>
<td>Licensing</td>
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<tr>
<td>Quality Assurance</td>
<td>Inspection Planning</td>
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<tr>
<td>Special Treatment</td>
<td>Significance Determination</td>
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<tr>
<td>Security</td>
<td>Events Assessment</td>
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<tr>
<td>Training</td>
<td>Emerging Issues</td>
</tr>
</tbody>
</table>
A very significant part of the RIDM program should be an ongoing enrichment of the organizational culture. While some uses of the concept of culture address, for example, “safety culture,” an organizational culture has actually many aspects. Accordingly, in this context, organizational culture includes operational culture, safety culture, risk culture, security culture, management culture, etc.

With respect to merging RIDM into an existing culture, the organization should develop a perspective of what the desired “target culture” should be. Then the organization should have an evaluation of the existing culture. The evaluation should identify gaps between the existing culture and the target culture. Next the organization should develop and implement a plan to achieve the target culture. Such assessment and refinement should be performed or at least considered periodically. The goal, the target culture, of such efforts should be to help all levels of personnel—management and staff—to appreciate and respect the relative benefits, strengths, weaknesses, limitations, boundary conditions, and cautions of RIDM.

This cultural enrichment is critical for RIDM to succeed. RIDM has proved to be very beneficial to organizations in which the organizational culture fully understands the discipline, approach, and implementation.

5 SUPPORT & IMPLEMENTATION

Deterministic safety analyses as well as probabilistic safety analyses use calculations. Such calculations require the best available information: inputs, assumptions, boundary conditions, etc. One significant input is statistical data; e.g., initiating event frequencies; structure, system, and component failure probabilities; common cause failure probabilities; human error probabilities; etc.

The more applicable data that is available for statistical analyses, the more uncertainties can be understood and reduced. Accordingly, the IAEA is pursuing an approach to make international PSA and RIDM data, insights, references, and related information available through the secure IAEA Centre for Advanced Safety Assessment Tools (CASAT). Such information may be accumulated directly in data bases or it may be linked through the internet.

As mentioned above, a significant tool to independently assess an organization’s culture, program, and procedures to use RIDM is the RIDM-IR mission to be offered to Member States by the IAEA. In an RIDM-IR mission, a team of international experts would review a nation’s policies and regulations and the organization’s culture, programs, and procedures with respect to the guidance mentioned above and with respect to experience with other similar organizations’ use of RIDM. Accordingly, the team would provide valuable constractive comments for additional development and enrichment.

Based on the guidance mentioned above and accumulated experience, the IAEA intends to offer national, regional, and international workshops and other forums to build RIDM capacity. A key feature of such events would be an exercise to get the participants involved and to facilitate disciplined thinking.

Finally, a number of IAEA Member States have initiated or formally requested to be pilots to implement the RIDM program and further build their related capacity. A number of other Member States have orally expressed interest in being pilots or in having assistance to implement and enhance RIDM.

The IAEA intends to support this interest, to analyze experience from the above activities, to distil lessons learned and insights, and to make the lessons learned and insights available on the CASAT.
6 SELECTED INSIGHTS

To provide a flavour of RIDM good practices and insights discussed through various RIDM discussions at meetings, seminars, or actual RIDM applications, the following selected insights are provided.

Some nations strive to achieve risk “As Low As Reasonably Achievable” (ALARA) through cost effective measures. They focus more on insights than numerical results. Numerical results provide ranking insights, but absolute numerical results are not the primary focus.

One nation offered that it realizes improvements by implementing RIDM applications as a whole rather than implementing them item by item in a stand alone fashion. Such holistic application provides enhancements to procedures, overall infrastructure, and safety and risk culture.

The consensus of a significant discussion on “safety” and “risk follows: For RIDM to optimize decisions does not require that a decision maker be able, in an absolute sense, to maximize safety and minimize risk. In an absolute sense, to minimize the risk of using nuclear engineering technology — as would be true for most technology — would be to not use the technology at all. In a realistic sense, RIDM balances a spectrum of insights and inputs integrated with risk information to strive for an optimum and safe decision. Such inputs and insights would be from deterministic safety assessment, compensatory measures, mitigating measures, operational experience, probabilistic safety assessment, etc. While RIDM has the potential to offer cost effective options, economic considerations would not take precedence over safety considerations.

Definitions used in PSA can vary among PSAs and can be ambiguous. Terms for hardware, for dependencies, and terms used in the PSAs, per se, differ; e.g., initiating event, safe state, core damage. There are some definite differences in the initiating event frequencies used among similar NPPs. A goal is to have harmonious definitions internationally.

7 CONCLUSION

As discussed, the value of RIDM has been increasingly recognized in the international nuclear community for a number of years. Additionally, nations developing nuclear engineering technology programs and nations emerging into that arena are recognizing the value and potential of RIDM. At the same time, nations, organizations, and individuals express genuine concerns and cautions that need to be satisfied. Establishing a solid RIDM infrastructure and developing a conforming and supportive organizational culture are keys to allowing all members of the international nuclear community to increasingly benefit from this valuable discipline.

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Current Trend in Nuclear Safety in Belgium

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ABSTRACT

Nuclear phase out is legally in force in Belgium for the time being. The electricity demand kept however increasing and alternative power sources alone will not be able to match the difference between the offer and the demand. As a consequence, most of the Belgian nuclear power plants (NPPs) underwent a power uprating process these last years. In Belgium, the NPPs must be re-assessed each ten years to demonstrate that the level of safety is at least maintained throughout the years and that the plant can safely operate for the ten years to come. Amongst other decennial activities, this led us to set up and run a complete aging management program adapted to each NPP. The power uprates by themselves gave rise to extra concerns in terms of system solicitations and capacities and safety analyses. Extra concerns require extra resources but Belgium, alike other nuclearized countries, is facing a shortage of experimented human resources. Hence, there is a need for improving the efficiency of the nuclear safety activities and for enhancing the implementation of the safety culture in the operation of the nuclear sites, be it power reactors or other facilities. This paper presents the high-level top-down approach that has been proposed in order to streamline the training in safety culture and the operational safety efforts, pinpointing particular benefits of the method.
1 INTRODUCTION

The global decline of manpower resources is inescapable in Western Europe as a consequence of the post-World War II baby-boom leading to the current pappy-boom effect. To illustrate this, let’s have a look at Figure 1 showing the birthrate curves in Belgium and in France for the period ranging from 1940 to 1985 (source: ref.[1]). People born after 1985 are not likely to be of help as nuclear safety expert this year; this is the rationale for the end of the graphic at the one hand. At the other hand, 100 % has been chosen as reference for the birthrate that year in order for the numbers to be comparable for the two countries.

Figure 1: Evolution of the birthrates (1940-1985)

As you may see, the baby-boom clearly appears in 1946 with an echo 18 years later coupled with a golden-sixties effect. After that there is a decrease by 40% in Belgium to reach the situation in 1985. This means that all the jobs created by the baby-boomers since 1965 will hardly find successors after those will retire. This is not specific to the nuclear safety sector but it shows that any shortage in human resources for a particular sector will hardly benefit from resources released by another sector.

In Belgium the situation worsened when the Government stated in 2002 the close-down of the nuclear power plants after 40 years operation. Figure 2 shows the status of the nuclear electricity production at that time. Based on a 40-year lifetime for each plant, this decision intervened roughly when 50% of the installed capacity had been used. The last plants were connected to the grid in 1985 so they had still about 60% of their capacity to be produced over the next 23 years, which is about one generation time. To cope with this situation, fresh human resources are needed.

Now, imagine the dilemma for the students in the engineering schools who were decided to go for nuclear engineering when they hear about the limited future of the nuclear power plants in Belgium! Most of them reoriented their options for other sectors, thereby increasing further the gap between needed and available resources for the specific sector of nuclear safety. This explains why there is an urgent need for enhancing the working methods in the field of nuclear safety in Belgium.
2 KEEPING SAFE WITH LESS RESOURCES

Considering the usual steps of a project development one could classify the successive activities in a standard format where following phases appear:

- information to be gathered
- objectives to be set
- tasks definition
- project development
- implementation of the outcomes
- verification & validation
- start-up & operation

Let’s examine each of those phases in the context of a nuclear safety project.
2.1 Information

This phase is very crucial. It requires gathering information both about the status of the plant and about the applicable regulations and reference documents. Any shortcoming or wrong ranking of this kind of information might lead to improper orientation of the project. It is therefore a good candidate for improvement action in the light of previous experiences.

2.2 Objectives

Based on the current plant and safety cases documentation status on the one hand and the goals to be reached at the other hand, clear objectives should be set. Any ambiguity, lack of precision or shortcoming might have a disastrous effect on the process. A thorough review of this phase is therefore greatly advisable.

2.3 Tasks

The next step concerns the translation of the objectives into a specific set of tasks. Once more one must ensured that the process is complete and well documented. A good documentation will allow the right introduction of holding points and/or monitoring actions that will have to be conclusive when applied during the next phases.

2.4 Development

By far the most resources-demanding phase, it will be the largest beneficiary of the efforts invested in the previous phases through a precise definition of the required objectives to be achieved and an optimized fit of the tasks associated to each objective.

2.5 Implementation

This phase will benefit of the holding points and monitoring actions set forth during the task definition phase. Any deviation against the foreseen progress will be identified in due time so that appropriate corrective actions may be initiated to redirect a correct implementation.

2.6 Verification

The verification phase will include a check of the fulfillment of the objectives as they have been defined in the “Objectives” phase. In case of failure, the good documentation of the process should help to identify the root cause of the discrepancy – be it at the task definition level or in the course of the project development – allowing efficient corrections to be easily designed.

2.7 Operation

This last phase should not be impacted by the improved project development process as the aim of the improvement was to reach the same final goal but with a global reduced amount of manpower involved in the project.

2.8 Global evaluation of the needed resources

Enhancing the care for project definition through its precise positioning in the overall safety management process, and close monitoring of its development will lead to an improved efficiency of the whole process as illustrated by Figure 3.
Figure 3: Resources involved in the projects

- Phases:
  - Information
  - Objectives
  - Definition
  - Development
  - Implementation
  - Verification
  - Operation

- Standard Project Development

- Improved Project Development

Resources:
- Additional
- Reoriented
- Monitoring
3 APPLICATION TO DECIENNIAL SAFETY REVIEW

Decennial safety reviews are imposed by the Royal Decree for Authorization issued for each nuclear power plant (NPP) in Belgium. The purpose is to reassess the safety of the plant taking into account the evolution of the international regulatory context as well as the evolution of the knowledge databases including the experience feedback with the aim of ensuring the safety of the NPP for at least ten more years. Because of this provision, the lifetime of the Belgian NPPs has never been technically fixed and hence the concept of life extension does not apply.

The outcomes of the decennial safety reviews may nevertheless lead to the necessity of replacing equipments, even large ones such as a vessel head for instance or the need for setting up monitoring programs as for the follow-up of the ageing related degradation mechanisms.

3.1 Information

Since the last decennial safety review, new applicable reference documents have been issued: the IAEA issued the Safety Guide NS-G-2.10 "Periodic Safety Review of Nuclear Power Plants" and WENRA published its reference levels for NPPs. The ranking came in favour of the IAEA document because it was the better adapted to existing plants but the WENRA reference levels were kept in mind for the quantification of some proposals for modifications. This was an improvement versus the subject list of the previous decennial safety reviews where both the Utilities and the Authorized Body set up separate lists that were finally assembled in a common list of concerns that was approved by the Federal Agency for Nuclear Control.

3.2 Objectives

Although the objectives were clearly to ensure an acceptable treatment of all the issues listed in the Safety Guide, it was agreed that the scope of the safety review itself would be, when applicable, limited to the validation of the processes that were to be put in place and to the verification of the outcomes of the existing processes, for instance the ageing management program initiated during the previous decennial safety review.

This will allow the safety review project to be completed in a reasonable time schedule, say two or three years. The previous decennial safety reviews included the evaluation of the outcomes of such processes what imposed to keep the project teams alive a long period of time and delayed the official closure of the safety review.

3.3 Tasks

A special attention is to be paid to the final issue raised by the Safety Guide: the global assessment. This task aims at verifying the completeness and the consistency of the outcomes of the whole set of the tasks. To avoid findings about incompleteness or inconsistency late in the course of the project, holding points are included in the schedules of the tasks. They are designed to allow pinpointing any indication that the global assessment could reveal incompleteness or inconsistency when performed at the end of the project. Early corrections will minimize time losses should such indication appear.

3.4 Development

The development phase is presently on-going and proceeds in line with the monitoring actions have been programmed.
3.5 Implementation, verification and operation

These phases are not yet activated at the time this paper is being written. Updated information will be given during the topical meeting.

3.6 Global assessment

As this exercise has been anticipated along the tasks of the project, it is expected that the outcome will be satisfying without demanding any extra resource.

4 CONCLUSION

Keeping the nuclear safety level as high as reasonably achievable is no more a question of the amount of money one could spend but a question of the amount of experimented manpower available to optimize the safety management.

In line with the development of quality system, efficient use of the available expertise requires enhanced efforts for the definition of nuclear safety projects with a view of its final global assessment in order to streamline and monitor the subsequent development phases.

The most important benefits of the method described above are twofold. First, the enhanced efforts, devoted to enlarge the scope of the contextual information to be gathered, allow adjusting the definition of a project to what it should really produce in terms of safety. Second, the verification process and its hold points are designed with a special attention paid to the final impact of the outcomes of the project upon the global safety of the unit. This makes sure that the project will be effective and that it will be performed in accordance with an appropriate efficient method.

To present it shortly, one could summarize the spirit of the approach as being THE “Think High Early”.

ACKNOWLEDGMENTS

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Operational Safety
State-of-the-Art of the Ignalina RBMK-1500 Safety

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ABSTRACT

Ignalina NPP is the only nuclear power plant in Lithuania consisting of two units, commissioned in 1983 and 1987. Unit 1 of Ignalina NPP was shutdown for decommissioning at the end of 2004 and Unit 2 is to be operated until the end of 2009. Both units are equipped with channel-type graphite-moderated boiling water reactors RBMK-1500.

The paper summarizing the results of deterministic and probabilistic analyses, developed within 1991 – 2007 by specialists from Lithuanian Energy Institute. The main operational safety aspects, including analyses performed according the Ignalina Safety Improvement Programs, development and installation of the Second Shutdown System, Guidelines on Severe Accidents Management and results of the implementation of (SIP, SIP-2 and SIP-3) will are discussed. Also the phenomena related to the closure of the gap between fuel channel and graphite bricks, multiple fuel channel tube rupture, containment issues as well as implication of the external events to the Ignalina NPP safety, are discussed separately.

1 INTRODUCTION. HISTORICAL CONTEXT

Preparatory works of construction of the Ignalina NPP have been started in 1974, and the first unit of Ignalina NPP was commissioned in December 31, 1983. At the same time the second unit was under construction and construction of the third unit began. The second unit it was planned to start to operate in 1986 year, but because of accident in Chernobyl works on preparation to operate 2 unit have been rescheduled. Second unit was commissioned in August 31, 1987. At that time 60 % of the third unit have already been constructed, but later construction was suspended and terminated soon. Nowadays because of political reasons the first unit of Ignalina NPP is shutdown, the second unit is planned to shutdown at the end of 2009.

Ignalina NPP with RBMK-1500 reactors belongs to the second generation of RBMK type reactors (it means, that this is most advanced version of RBMK reactor design series in comparison with others RBMK type nuclear power plants). In comparison with infamous Chernobyl NPP, Ignalina NPP reactors are by a third more powerfully and already from the beginning of operation had substantially advanced emergency protection systems (e.g., emergency core cooling and accident localization systems) [1].

After 1990 Lithuania declared its independence, Ignalina NPP with two largest in the world RBMK-1500 reactors came under authority of the Lithuania Republic, however nobody in the world did not know about the real level of these reactors safety. The first Safety
Justification of Ignalina NPP has been prepared by Russian experts of Research and Development Institute of Power Energy (RDIPE), organization - designer and developer of RBMK reactors, after Chernobyl NPP accident. In this document the analysis of all design basis accidents (except partial breaks of pipes) is presented in sufficient details. The analysis is performed using at that time existing tool – quasistationary derivative approximation method, being based on conservative assumptions and existing experimental data. From the present-day viewpoint such safety justification [2] has lacks:

- it was limited only to the systems description and the analysis of design basis accidents;
- computer codes, developed in Russia, have been used for simulations, but these codes have not been verified;
- the independent expertise of safety analysis has not been performed.

Therefore, at the beginning of the 90-s of the last century reasonably there were doubts how such safety justification of Ignalina NPP, presented in the first safety justification, corresponds to the real situation. In 1992 at G7 Munich Summit the decision on closing of Soviet-design nuclear power plants, first of all the nuclear power plants with reactors of RBMK and VVER-440/230 types was accepted. In 1994 Lithuania has signed the agreement with the European Bank for Reconstruction and Development (EBRD) Account of Nuclear Safety on which has undertaken to perform in-depth safety analysis of the Ignalina NPP and to not change fuel channels in a reactor.

Right from the start, when Lithuania assumed control of the Ignalina NPP, the plant, its design and operational data has been completely open and accessible to Western experts. A large number of international and local studies have been conducted to verify the operational characteristics of the Ignalina NPP and analyze its level of risk. Ignalina NPP is unique nuclear power plant of RBMK type about which information it was collected, checked, systematized and accessible. Collected and verified data base has allowed:

- to assess present safety level of NPP,
- to compare it level with others RBMK type NPPs safety level,
- to plan improvements of plant equipment and operating procedures increasing safety of the NPP.

2 DETERMINISTIC AND PROBABILISTIC IGNALINA NPP SAFETY ANALYSES

In this chapter the main Ignalina NPP safety analyses, performed since 1991 till these days, are discussed:

- Ignalina NPP Unit 1 safety analysis report and its review,
- Modifications of activation algorithms for reactor shutdown and emergency core cooling systems,
- Second diverse reactor shutdown system development, safety justification and implementation,
- Studies of Ignalina NPP 1 and 2 levels of Probabilistic Safety Assessment (PSA),
- External events at Ignalina NPP analysis.

2.1 Deterministic Ignalina NPP safety justification

In 1995 – 1996 it has been prepared In-depth Ignalina NPP Unit 1 Safety Analysis Report, using USA and Western Europe methodology and computer codes for providing of safety analysis [3]. It was comprehensive international study sponsored by EBRD. The purpose of this international study was to provide a comprehensive overview of plant status...
with special emphasis placed on its safety aspects. Specialists from the Ignalina NPP, Russia (RDIPE), Canada and Sweden contributed. During implementation of the project it has been described more than 50 systems of normal operation, safety important systems and auxiliary systems. Also analysis of these systems has been performed, considering compliance of these systems to the Lithuanian standards and rules as well to practice of safety used in the West. Analysing systems the attention has been concentrated on their consistency to criterion of single failure, as well as to auxiliary safety aspects: maintenance, inspections and impact of external factors (fire, flooding by water). This analysis of systems has defined the main lacks of systems and has developed conditions for elimination of the deficiencies. The performed review on operation and safety has allowed to identify all possible malfunctions, which can potentially cause an emergency situation.

In the safety analysis report of the Ignalina NPP Unit 1 the comprehensive accident analysis, equipment assessment has been provided as well as discussed questions concerning equipment ageing, investigated topics related to operators action and power plant control, provided conclusions about safety of Ignalina NPP (NPP safety level was assessed realistically), main lacks has been defined and measures for elimination of the deficiencies has been foreseen. It is the first western type report on safety for nuclear power plants with RBMK reactors.

One of the basic conclusions in this safety analysis report was such that in this case there was no problem, which would demand immediate shutdown of the Ignalina NPP. Detail accident analysis (accidents because of different pipelines ruptures, reactivity initiating accidents, equipment failures, transients with additional failure of reactor shutdown system, fuel channel ruptures in the reactor cavity) has shown, that accident occurring because of equipment failures does not cause such condition of the plant station which would cause violation of acceptance criteria, as well as safety system ensures a safe condition of the plant even doing the assumption, that operator does not take any action for 10 minutes from the beginning of accident to mitigate an emergency situation. Because of reactivity initiating accidents (exactly such type of initiating event became the reason of accident on the Chernobyl NPP) acceptance criteria of power plant also are not violated, even postulating single failures additionally. It has been shown, that Ignalina NPP is reliably protected against loss of the coolant accidents if ruptures of pipelines do not cause local stagnation of flow. In case of one steam line rupture the acceptance criteria will not be exceeded. But there are two steam lines located in the shaft at the Ignalina NPP, thus rupture of one steam line can cause rupture of other steam line, and in this case radiological dozes can be exceeded. Being based on these results of accident analysis the recommendations for modifications of activation algorithms for reactor shutdown and emergency core cooling systems have been prepared.

It is necessary to note, that in parallel with the Ignalina NPP Unit 1 safety analysis report in 1995–1997 it was performed independent Review of the Ignalina Nuclear Power Plant Safety Analysis Report [4]. This studio was performed by experts from USA, Great Britain, France, Germany, Italy, Russia and Lithuania. Independent Review has confirmed the main conclusions of Safety Analysis Report.

In recommendations of Ignalina NPP Unit 1 safety analysis report it has been shown, that Ignalina NPP will be reliably protected from any ruptures of pipelines and steam lines after improving of activation algorithms for reactor shutdown and emergency core cooling systems. According to these algorithms the system will automatically activate on coolant flow rate decrease in single Group Distribution Header (GDH) and sharp pressure decrease in drum-separators. These modifications have been implemented in both Ignalina NPP units. Safety justification of these modifications have been performed in Lithuanian Energy Institute (LEI). Further discussed situation, when conditions for local flow stagnation because of GDH rupture in the fuel channels connected to this affected GDH are developed [5]. The reason of
such flow stagnation is the leak of the certain size, and at discharge of a part of the coolant through this leak the zero gradient of pressure is developed in fuel channels (7 – see Figure 1), i. e. pressure in a bottom of the channel is close to pressure in drum - separators (1). Coolant flow rate stagnation in fuel channels can be broken only in case of early activation of emergency core cooling system (ECCS) (see Figure 2 (a)). Thus if ECCS would operate according to design algorithm (reactor cooling water started to supply only after approximately 400 seconds from the beginning of accident), acceptance criteria for both fuel rod cladding and fuel channel walls temperatures in high power channel would be exceeded (see Figure 2 (b, c)). After implementation of ECCS activation algorithms according coolant flow rate decrease in separate group distribution headers, water from ECCS starts to supply already after 5–10 seconds from the beginning of flow stagnation. Thus stagnation is broken and fuel channels, connected to affected GDH, are reliably cooled (see Figure 2). These modifications of activation algorithms for reactor shutdown and emergency core cooling systems are installed in power plant unit 1 in 1999, and unit 2 – 2000 m.

Figure 1. Ignalina NPP reactor cooling circuit (one loop) and coolant flow diagram in case of partial GDH rupture: 1 – drum-separators, 2 – suction header, 3 – main circulation pumps, 4 – pressure header, 5 – group distribution headers, 6 – water supply from emergency core cooling system, 7 – affected fuel channels
In the Ignalina NPP Unit 1 safety analysis report have been investigated not only basic design accidents (discussed above), but also Anticipated Transients Without reactor Shutdown (ATWS). Investigations of such accidents are carried out at the licensing process for USA and Western Europe nuclear power plants, however for the NPPs with RBMK type reactors such analysis has been performed for the first time. Consequences of accident for RBMK-1500 reactor during which loss of preferred electrical power supply and failure of automatic reactor shutdown occurs [6] are presented in Figure 3. Due to loss of preferred electrical power supply all pumps are switched (see Figure 3 (a)) off therefore the coolant circulation through fuel channels is terminated. Because of the lost circulation fuel channels are not cooled sufficiently therefore temperature of the fuel channels walls starts to increase sharply. As it is seen from Figure 3 (b), already after 40 seconds from the beginning of the accident the peak fuel channel wall temperature in the high power channels reaches acceptance criterion $650^\circ$C. It means that because of the further increase of temperature in fuel channels plastic deformations begin – the channels because of influence of internal pressure can be ballooned and ruptured. On the first second of accident the main electrical generators and turbines are switched off as well. Steam generated in the core is discharged through the steam discharge valves, however their capacity is not sufficient Therefore the pressure in reactor cooling circuit increases and approximately after 80 seconds from the beginning of accident reaches acceptance criterion $10.4$ MPa (see Figure 3 (c)). The further increase of pressure can lead to rupture of pipelines.

Thus the analysis of anticipated transients without shutdown has shown that in some cases the consequences can be dramatic enough. Therefore the priority recommendation has been formulated: to implement the second, based on other principles of operation, diverse shutdown system. However development, designing and implementation of such system needed few years (in the Ignalina NPP unit 2 this system was installed in 2004), so the compensating means, which were used in transition period while second diverse shutdown system was developed, has been implemented. This temporary system was called according Russian abbreviation „DAZ“ („Dopolnitelnaja avariynaja začita“ – „Additional emergency

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**Figure 2. Analysis of partial GDH rupture considering modification of ECCS algorithm:**

- **a)** coolant flow rate through fuel channels,
- **b)** fuel rod cladding temperature in high power channel connected to ruptured GDH,
- **c)** behaviour of fuel channel wall temperature
This system used the same control rods as well as design reactor shutdown system, however signals for this system control were generated independently in respect of design reactor shutdown system. In Lithuanian Energy Institute for DAZ system has been selected not only set points of activation, but also the safety justification was performed. Performed analysis has shown, that after implementation of DAZ system the reactor is shutdown in time, cooled reliably as well acceptance criteria are not violated even in case of transients when design reactor shutdown system does not functioning. In Figure 3 is shown the behaviour of the main parameters of reactor cooling circuit in case of loss of preferred electrical power supply and simultaneous failure of design reactor shutdown system. In this case two signals for activation of DAZ system (reactor shutdown) are generated: on increase of pressure in drum - separators and on decrease in the coolant flow rate through the main circulation pumps. In Unit 1 DAZ system was installed in 1999, in Unit 2 – 2000.

![Figure 3. Analysis of loss of preferred electrical power supply and simultaneous failure of design reactor shutdown system, when DAZ system was installed: a) coolant flow rate through one main circulation pump, b) the peak fuel channel wall temperature in the high power channel, c) pressure behaviour in drum - separators, 1 – acceptance criterion, 2 – set points of DAZ system activation (reactor shutdown)](image-url)

The Diverse Shutdown System (DSS) has been designed and installed in Ignalina NPP Unit 2 in 2004. In the first unit of Ignalina NPP this system has not been installed because reactor has been shutdown in 2004. Therefore, nowadays Ignalina NPP reactor emergency protection (emergency shutdown) system consists of two independent shutdown systems: first – (BSM) controls manual control rods and shortened absorber rods, which are inserted into the core from bottom. This system performs the normal reactor shutdown function and can maintain a reactor in sub-critical state. Second system (AZ) controls 24 fast acting reactor shutdown rods as well additionally 49 rods, which belong both – BSM and AZ systems. AZ system performs emergency protection function. Also the Additional Hold-down System of the reactor is installed. This system allows to prepare and inject water and neutron absorber gadolinium mixture into control rods cooling circuit. Thus, the reactor remains in sub-critical state even in the case of failure of BSM system.

DSS justification was one of the main projects increasing a level of NPP safety. Specialists from LEI together with experts from the countries of the Western Europe checked and have assessed the design documentation, carried out independent calculations, thus helping Lithuanian regulatory body (VATESI) to make the appropriating decisions concerning implementation of mentioned system at Ignalina NPP [7]. In conclusions of
review it has been shown, that implementation of second, diverse reactor shutdown system protects a reactor in case of failure of design reactor shutdown system. Implementation of this system has ensured that any initiating event cannot cause accident with damage of the reactor core, as well as decreases core damage probability from $4 \cdot 10^{-4}$ up to $5 \cdot 10^{-6}$.

2.2 Ignalina NPP probabilistic safety assessment

The Ignalina NPP first level PSA “Barselina” project (1991–1996) was initiated in 1991 [8]. It was first PSA for nuclear power plants with RBMK type reactors. From the beginning this project was carried out by nuclear energy experts from Lithuanian, Russian and Swedish institutions, and since 1995 it was carried out by efforts of experts from Lithuania (Ignalina NPP, LEI) and Sweden. Main objective of deterministic analysis was to show, that nuclear power plant reliably copes with accidents, and basic purpose of PSA 1 level is to assess probability of reactor core damage, to create a basis for severe accident risk assessment and management. Performed Ignalina NPP PSA 1 level study is predicted by assumption, that the main radioactive source is reactor core. This PSA is performed for maximum permissible reactor operating power. Only internal initiating events have been analysed – transients, loss of the coolant accidents, common cause failure and internal hazards (fire, flooding, missiles). Results of the analysis have shown that after implementation of recommendations from BARSELINA [8], Safety analysis report and its independent review [3, 4], probability of Ignalina NPP core damage is about $6 \cdot 10^{-6}$. According to the international requirements this parameter for the operating nuclear power plants should not exceed $10^{-4}$ per year, and for new NPPs, which are in process of construction – $10^{-5}$. Therefore Ignalina NPP fulfils this requirement. Analysis has shown that in Ignalina NPP risk topography dominates transients, instead of loss of the coolant accidents. The risk of core damage most of all increases transients with loss of long-term core cooling. It is the positive fact meaning that up to consequences of severe accidents there is enough time. Thus operators supervising reactor operation can undertake corrective measures, and it means that Ignalina NPP has great potential opportunities for implementation of the program on management of severe accidents. It is necessary to note, that procedures and means on severe accident management are already implemented at Ignalina NPP Unit 2 [9, 10].

According to the international requirements probability of the large reactivity release outside nuclear power plant should not exceed $10^{-7}$ per year for new NPPs, which are in process of construction, and for NPPs in operation – $10^{-6}$. Scenarios and probabilities of the large reactivity release outside nuclear power plant are objects of investigations for PSA level 2. Ignalina NPP PSA level 2 project was performed in 1999–2001 [11] and it was the first project of such type for nuclear power plants with RBMK reactors. This project was carried out by efforts of experts from Lithuania (LEI) and Sweden. Performing PSA level 2 as initial data were used results of level 1. According in PSA level 1 investigated accident scenarios consequences and its similarity criteria on radioactive contamination the conditions of damage of the reactor have been developed and possibilities of accident management were assessed. Results of PSA level 2 have shown that barrier of the large reactivity release after core damage is 1.5. This barrier is smaller in comparison with modern nuclear power plants having function of containment, which reaches 10 and more. Being based conservative assumptions and estimation of parameters, in PSA level 2 was calculated that general estimation of large discharge frequency is $3.8 \cdot 10^{-6}$ per year. Therefore, Ignalina NPP according the probability of large reactivity release outside nuclear power plant is not the worst in comparison with the plants of the USA and the Western Europe, constructed in the same years.
Carrying out the complex analysis about influence on Ignalina NPP units safety [12] LEI the following external events have been investigated:

- aircraft crash;
- extreme wind and tornado;
- flooding and extreme showers;
- external fire.

**Aircraft or others flying objects crash** caused accidents in Ignalina NPP will have local character because of its big territory. According to the Lithuanian civil aviation data it has been assumed that average congestion - up to 50000 flights per one year within the 50 kilometres zone around NPP. Three zones have been defined by a radius up to 15, 50 and 85 meters around the reactor in the territory at Ignalina NPP (15 – according reactor dimensions, 85 – according reactor building size). Probability of air crash on a 85 metres zone around of the reactor center, assuming that aircraft weight is 5700 kg as well assuming that half of these flights carry out planes of the western manufacturers, and other half – Soviet is \(2.06 \times 10^{-9}\) 1/year. Even doing more conservative assumptions (heavy planes falling frequency equalized to easy planes falling frequency) probability of air crash on a 85 metres zone around of the reactor center will be \(1.64 \times 10^{-7}\) 1/year. The obtained heavy plane crash probabilities are less than the probabilities obtained in probability analyses for the majority of the West-European and American NPPs.

**Tornado** may cause huge damage and destruction. From all buildings of nuclear power plant the tornado is most dangerous for a technical water supply system building, because it is located on the open territory on a coast of lake. Tornado and hurricane winds do not create danger for buildings of reactor and technical systems. Contrariwise probability of tornado and hurricane winds is \(5.3 \times 10^{-6}\) 1/year. Therefore it is possible to approve, that their influence on reactor safety is insignificant.

**Rise of a water level in the lake Druksiai** represents the greatest danger to pump station on the lake, since the service water system is the nearest NPP construction to the lake. Water level elevation of the lake Druksiai up to a level 144.1 m is not possible practically; therefore there is no danger on flooding of pump station. The platform of the other Ignalina NPP construction is located at a level of 148 - 149 m above the sea level. Rise of a water level in the lake Druksiai up to such mark is impossible and flooding does not represent the direct danger for Ignalina NPP.

Besides lake, other external flooding source is **extreme showers**. In territory of Ignalina NPP there is drainage system and all compartments which are located below a critical mark of a level are connected to this system, therefore the water leaks in case of internal flooding. Thus, extreme showers do not cause external flooding of the reactor building. For probabilistic external flooding analysis the mathematical model to assess peak water level elevations of the lake Druksiai has been developed. Probabilistic assessment of water level elevation in the lake has been performed. Maximum amount of precipitation (not less than 279.7 mm in 12 hours) probability is \(1 \times 10^{-6}\) 1/year. Such event will not have influence on reactor safety.

**Probabilistic analysis of external fire.** Ignalina NPP is situated in the region, where 30 % of territory is occupied by forests (40 % are grassland and 30 % are occupied by lakes and swamps). The edge of the closest forest is less than one kilometres from territory of Ignalina NPP. On the territory of the NPP there are only separate trees and grass. The global fire of a forest with a high wind to the NPP side can cause the smoke cover on the territory of Ignalina NPP. The smoke does not influence work of reactor mechanisms, but will complicate work of the personnel. Fire probability of forest, which is in 10-kilometer zone around Ignalina NPP and there are more than 2000 ha woods, is \(2.7 \times 10^{-3}\) 1/year. It is a high probability, but any fire cannot affect safety of the reactor considerably.
3 IGNA LINA NPP SAFETY ASSESSMENT IN CASE OF SPECIFIC RBMK PROBLEMS

Discussing safety of RBMK type nuclear power plants three vulnerabilities more often are mentioned generally:

- problem of gas gap closing between fuel channels and graphite blocks;
- problem of multiple fuel channel ruptures;
- containment issue.

Below specificity of RBMK-1500 in respect of these problems is discussed.

3.1 Problem of gas gap closing between fuel channels and graphite blocks

The fuel channels of RBMK type reactor are separated from the graphite bricks by gaps maintained by graphite rings. These rings are arranged next to one another in such a manner that one is in contact with the channel, and the other with the graphite stack block. (see Figure 4). As a result of exposure to neutron radiation and temperature the diameters of graphite columns gaps decreases, and fuel channel tube – expands, thus the gap between them decreases.

The availability of the gap between graphite bricks and fuel channels is the main condition limiting the operation of RBMK type reactors. These graphite - fuel channel tubes gaps allow:

- unimpeded (axial and radial) thermal expansion and contraction of the fuel channels;
- predictable non-contacting heat transfer from graphite bricks (temperature higher than 500 °C) to fuel channels (temperature 300 - 320 °C) across the gaps;
- leakage of helium-nitrogen mixture, which provides heat transfer from graphite to coolant and protects graphite against oxidation. Furthermore helium-nitrogen mixture is part of fuel channel integrity monitoring system.

The control of gap between fuel channels and graphite blocks at Ignalina NPP Unit 1 & 2 is carried out from the beginning of its operation and now the largest database and experience of assessment of gap among all RBMK type reactors is saved. After gap closure not only some functions of the control are lost, but also worsens characteristics of the reactor. Increase probabilities of damage of the channel and deformations of graphite, withdrawing of the channel from a reactor if necessary becomes complicated, the temperature of graphite and...
the fuel channel changes. In Ignalina NPP Unit 1 reactor the average gap between fuel channels and graphite up to final shutdown of the reactor from an initial level (3 – 2.7 mm) has decreased three - four times. This decreasing in Unit 2 is insignificant. Estimation of such small gap is very sensitive to errors of measurements, uncertainties of used models and to strategy of selection of fuel channels for measurements.

As it is known, after signing the agreement with the EBRD Account of Nuclear Safety in 1994, Lithuania has undertaken to not change fuel channels and not operate Ignalina NPP reactor after closing even one gas gap between graphite stack and fuel channels. In Ignalina NPP in-depth safety report [3], which has been prepared by the international experts in 1996, it was predicted, that at Ignalina NPP unit 1 it happens not later than in the beginning of 1999.

In Lithuanian Energy Institute complex investigations on the problem of gap closure between fuel channels and graphite blocks at Ignalina NPP have been carried out. Assessment of the gap between graphite stack and fuel channels has very big importance because results of this problem are very important making of the decision on duration of Ignalina nuclear power plant operation. At development of a technique on assessment of gap and strategy of measurements the thermal-hydraulic, structural and probabilistic calculations have been performed. The detailed analysis [13] has shown, that in Ignalina NPP In-depth Safety Analysis Report [3] the assessment of the gap between fuel channels and graphite blocks at Ignalina NPP Unit 1 reactor has been performed using simplified deterministic calculations. Therefore obtained results were too pessimistic and conservative, predicting closure of the gap in set of channels in 1998 – 2000.

The specialists from LEI developed the integrated technique on assessment and control of risk of gas gap reduction. This allowed to develop strategy of measurement of holes diameters in graphite columns and replacement of fuel channels. This strategy has ensured existing of gap in Unit 1 reactor up to its final shutdown and by that has allowed considerably to prolong time of Ignalina NPP Unit 1 operation (until the end of 2004).

Change of a gas gap in the second unit of a reactor very much differs from the first Unit because in a reactor of Unit 2 it is used zirconium tubes of fuel channels having different hardened surfaces and the rate of their ballooning is two times slower in comparison with tubes in reactor of the Unit 1. Tendencies of change of graphite stack diameters in the second Unit are very similar to the first Unit.

### 3.2 Problem of multiple fuel channel ruptures

In case of fuel channel rupture a two-phase flow is discharged to gaps between graphite stack. Part of graphite blocks can be damaged cracked by coolant jet impingement, graphite columns can be displaced and coolant passes into the reactor cavity. Because graphite stack is hotter than the coolant, the pressure in tight reactor cavity increases. The leak tight Reactor Cavity (RC) performs the function of containment in the region immediately surrounding the nuclear fuel and graphite. The RC is formed by a cylindrical metal structure together with bottom and top metal plates. The reactor cavity confines the steam release in case of rupture of fuel channels. The steam-water-gas mixture from the reactor cavity is directed via Reactor Cavity Venting System (RCVS) pipelines to two steam distribution devices of the 5th (upper) condensing tray in the Accident Localisation System (Figure 5). Two pipelines d = 400 mm that come from a branch pipe d = 600 mm located above the top plate of RC are interconnected to a pipe d = 600 mm and which connects to one steam distribution device [1]. In the same way the other two pipelines d = 400 mm from the top plate of RC are connected to the second steam distribution device. On their way these pipelines have branches, which are interconnected in a leak-tight corridor and end up with three Membrane Safety Devices
The blowdown pipes from the bottom of RC pass directly to the leak-tight corridor and also end up with three MSD.

In the case of multiple fuel channel tube ruptures, if the RCVS does not assure relief of steam-water-gas mixture from RC, the pressure increase in the RC will lift top plate of the RC. Those, structural integrity of the RC and the rest fuel channels would be lost as well. Such event would cause very severe consequences similar to Chernobyl accident. Therefore it is important to maintain RC integrity, which is assured if pressure in the RC is below permissible pressure (314 kPa, abs) i.e. the pressure of upper plate of biological reactor shielding weight [14].

Rupture of one fuel channel is design basis accidents for RBMK-1500 reactors. Probability of such rupture – $10^{-2}$ 1/year. According design the reactor cavity venting system assured the integrity of RC in the case of up to 3 fuel channels ruptures. This system has been modernized in 1996 as shown in Figure 5.

![Figure 5. Simplified schematic of the reactor cavity venting system: 1 - reactor, 2- the fifth ALS suppression pool, 3 - suppression pools 1-4, 4 - steam distribution devices, 5 - membrane safety devices (350 mm diameter) ](image)

Moscow Research and Design Institute for Power Engineering (RDIPE), designer and developer of RBMK reactors, specialists in 1996 have analysed pressure behaviour in the Reactor Cavity in case of multiple fuel channel rupture [14]. Results of these calculations have shown that acceptance criterion – maximum permissible load (310 kPa) to upper reactor cavity plate will be exceeded in case of 9 fuel channels rupture (according RDIPE calculations). In RDIPE calculations the coolant discharge through the rupture conservatively was assumed equal 32 kg/s through one fuel channel. This flow rate has been selected as constant versus time. Because of such conservative assumptions amount of discharged coolant into reactor cavity is largest and number of channels, when permissible pressure in reactor cavity is not exceeded, will be minimal.
Such analysis is conservative with impact of uncertainties. The best estimate analysis of Ignalina NPP response to multiple fuel channels tubes rupture was performed at the Lithuanian Energy Institute. Sensitivity and uncertainty analysis was performed as well [15]. At performance of the analysis it has been considered, that results of calculations can be influenced by uncertainties such as the plant initial conditions, assumed at the modelling, as well as assumptions and correlations of CONTAIN code. Summarizing the results of the uncertainty and sensitivity analysis, it was concluded, that the capacity of RCVS comprises from 11 up to 19 ruptured fuel channels, i.e. 15±4 channels (Figure 7).

![Figure 6. Pressure in the reactor cavity as a function of a number of ruptured fuel channels (FCs)](image)

It is necessary to note, that the analysis was performed for the case, with RCS filled by coolant (the water levels in drum separators are nominal). Thus, after the fuel channels rupture the steam-water mixture is discharged into the gaps of graphite stack. If the “dropout” model is used in CONTAIN 1.1 code, it is assumed, that all the water released from the ruptured fuel channels in liquid fraction leaves from RC to the water drain. If the “dropout” model is not used in CONTAIN 1.1 code, it is assumed, that all not evaporated water remains in a dispersed condition, and it may be transferred into RC and through the pipelines into ALS. The last assumption leads to higher calculated pressure in the RC (see Figure 6).

It is necessary to note, that during operation of RBMK reactors there were only three cases of ruptures of separate fuel channels:
- at Leningrad NPP Unit 1 in 1975,
- at Chernobyl NPP Unit 1 in 1982,

In any of these cases adjacent channels have not been damaged. Thus, in reality there was no so-called “cascade rupture of fuel channels” when rupture of one channel causes ruptures of other channels. Experiments made on the large scale TKR-Test facility at Electrogorsk Research & Engineering center for NPP safety [16] have shown also, that cascade rupture of fuel channels are impossible.

### 3.3 RBMK reactor containment issue

In case of accident in nuclear power plant (rupture of reactor cooling circuit pipelines), the coolant with radioactive materials will spread into reactor and compartment enclosed reactor cooling circuit. In many (but not in all) reactors of the USA and the Western Europe function of containment carries out visible from afar, photogenic, semicircle form protection
enclosure. Usually non-existence of containment is treated as deficiency of RBMK reactors. However such containment as for vessel type reactors it is technically impossible to implement for RBMK reactors. In the Ignalina NPP the function of containing accidentally released radioactive material is accomplished by an extensive system of interconnected steel lined, re-enforced concrete compartments called the Accident Localization System (ALS). The ALS uses the “pressure suppression” principle employed by G.E. designed boiling water reactors. The ALS encloses the large Ignalina NPP reactor core, the coolant pumps and all of the piping providing coolant to the core. It is not necessary to enclose the pipes above the reactor core, which carry the exiting two-phase (steam-water) mixture to the drum - separators, because if one of them is breached, coolant flow to the fuel channels (which is provided by pipes entering the core from bellow) will not be interrupted. Significant amounts of radioactive material can escape only if fuel rods are over-heated. Breaches in the exiting pipes will not reduce coolant flow, therefore the fuel rods will not overheat.

The effectiveness of the ALS has been verified by extensive international analysis and experimental programs. They all show that even if events leading to release of radioactive materials are postulated, these materials will be contained by the ALS, thus the ALS performs the function of containment [17]. The minimal amounts (due primarily to non-condensable noble gases) which would eventually reach the environment, would not exceed the amounts that would be released by Western built reactors provided with the more familiar, prominently visible “dome containments”.

4 CONCLUSIONS

Requirements on nuclear power plants safety depends on the accumulated experience, a level of a technical society evolution, which always raises, and from position of the state. About safety level of Ignalina NPP it was worried after Chernobyl accident in 1986. The first modernizations of reactors have been implemented at that time. RDIPE, designer and developer of RBMK reactors, experts have prepared the first safety justification for operating power plant in 1989. When Lithuania assumed control of the Ignalina NPP in 1991, a large number of studies on safety level have been conducted. It is necessary to note Safety Analysis Reports for Ignalina NPP Units 1 & 2, Safety Justifications of Reactor Cooling System and Accident Localization System. The Ignalina nuclear power plant is distinguished from all RBMK type reactors for the matter of that many international studios to investigate design parameters as well level of its risk have been performed. Ignalina NPP, its design and operational data have been completely open and accessible to Western experts. At first the effective initial help in questions of nuclear safety has provided by Sweden, and after by other countries (Germany, United Kingdom, USA etc.), capable to perform expertises of the safety analysis.

The detailed analysis of accidents has shown, that design basis accidents do not cause such condition of the plant, which postulates violation of acceptance. As well safety systems of the plant ensures a safe condition of the plant even doing the assumption, that operator does not take any action for 30 minutes from the beginning of accident to mitigate an emergency situation.

The performed Probabilistic Safety Analysis of level 1 and 2 has allowed to compare safety level of Ignalina NPP with the reached level on other nuclear power plants and to plan, how to improve NPP safety systems and operational procedures. Investigations have shown that Ignalina NPP according the probability of large radioactivity release outside nuclear power plant is not the worst in comparison with the plants of the USA and the Western Europe, constructed in the same years.
On the basis of the performed investigations the recommendations on safety improvement were developed by efforts of local and foreign experts. These recommendations were brought into Ignalina NPP Safety Improvement Programs (SIP-1, SIP-2 ir SIP-3) which implementation strictly was checked by Lithuanian regulatory body VATESI. These means have allowed to improve safety level of the Ignalina NPP constantly. These works do not stop even on forthcoming final shutdown of the plant. In outcome of last significant project the Severe Accident Management Guide is developed. Now this guide is under implementation at Ignalina NPP. Severe Accident Management Guide will supplement Symptom-Oriented Emergency Operating Procedures and will provide safe elimination of accident consequences in all range of accidents.

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TOOLS TO SUPPORT IMPORTANT TECHNICAL DECISIONS DURING ACCIDENT CONDITIONS

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ABSTRACT

To handle design basis and beyond design basis accidents with an intact reactor core, Nuclear Power Plants are using Emergency Operating Procedures (EOP) that they may have developed based on the generic Westinghouse Emergency Response Guidelines.

Even though the EOPs are very directive, some questions are left to external support. In many Western NPP, a so-called Technical Support Center (TSC) is responsible for the engineering support. The Pressurized Water Reactor Owner Group (PWROG, previously Westinghouse Owner Group, WOG) has developed a generic TSC manual to support the TSC in their decision making processes.

Due to the specific and particular design of the Beznau NPP (KKB) Safety Systems, the development of a plant-specific TSC manual required a lot of additional issues compared to the generic PWROG material. The majority of considered issues are relevant for beyond design basis accidents and external events.

The new developed plant-specific TSC manual covers all technical issues the shift supervisor could ask the SED to address during a technical emergency. The background material for the SED’s decisions regarding issues of the EOPs is documented in fifty separate evaluations; one for each issue.

The plant specific TSC manual is a helpful tool for the SED of the KKB to better evaluate issues and potential concerns arising while executing the EOPs. Moreover, the manual can be used for the training of Pikett engineers and other technical specialists involved in technical emergencies.

This article gives an overview of the structure of the TSC manual and the necessary steps for the development of a plant-specific TSC manual on basis of the generic manual.
1 INTRODUCTION

NOK operates the Beznau NPP (KKB) with two 380 MW electrical power PWR units. First operation was in 1969 for unit 1 and in 1971 for unit 2. The nuclear island (NSSS) was designed and constructed by Westinghouse, the secondary systems by BBC. During 38 years of operation a lot of upgrades were carried out.

Following the TMI accident, the Westinghouse Owner Group (now PWROG) developed the Emergency Response Guidelines (ERGs) [1] as guidance for the control room operators to handle Emergency Conditions.

Within the ERGs, operators are sometimes instructed to consult the Technical Support Centre (TSC), a group of persons mobilized at the time an Emergency Situation is declared on the plant, and that remains outside of the Control Room to technically support the operators during their recovery actions. In the Westinghouse ERGs, there are seventeen issues for which the advice of the TSC is being asked. For some time, the TSC members had no specific tool to support their decision. To fill this gap, a TSC Manual [2] was developed to enhance the guidance available to the TSC for performing these 17 evaluations and making recommendations during implementation of the EOPs. This manual can be adapted into a plant-specific document in whatever format suits the utility the best (for example, as a job aid or as a training aid).

Like many other plants, NOK developed its plant-specific Emergency Operating Procedures (EOPs) based on the Westinghouse ERGs. The first KKB EOP package in German language was implemented in 1985 [3]. Because of the complex safety architecture of the KKB units, the actual EOPs [4] consists of two sets, one usable from the Main control Room and a second one usable for the Notstand Control Room when the plant safety is ensured by the Notstand system (which is described later in this document). However, as in the ERGs, there are several places in the KKB EOPs where the TSC guidance is requested.

In 2004, NOK decided to develop a plant-specific TSC manual in cooperation with Westinghouse.

2 PARTICULAR SAFETY FEATURES OF THE KKB PLANT

The procedures and TSC manual developed by the WOG are based on two reference PWR designs. Both units of the Beznau plant differ significantly from the two reference plants, especially in terms of safeguards systems which include lots of redundancies. Of course, these particular features had to be taken in account while developing the plant specific EOPs and the TSC Manual.

2.1 Notstand system

The Notstand system is a bunkered building designed to ensure plant safety following external events. The main safety features include:

- One feedwater train which is able to cool down the plant. The Notstand feedwater tank is fed by the Notstand well water system. During low pressure conditions in the SGs, the Notstand well water can also feed the steam generators.
- One Emergency Seal Injection train to ensure seal #1 cooling for the Main Coolant Pumps
- One high pressure safety injection train with throttling capability.
- One low pressure safety injection train with its own recirculation capability (dedicated jet pump).
- An Emergency Cold Shutdown System, which fulfills the same function as the residual heat removal system.
- A containment spray function ensured by the diversion of part of the Notstand safety injection to the containment spray nozzles.
An independent power supply by Diesel generator. The 6 kV Notstand bus system may also be fed by the Notstand Diesel generator of the other unit or by the electrical grid.

An independent well water system for component cooling and residual heat removal. The well water system may also be fed by the Notstand well water system of the other unit.

An Emergency control room (ECR) that allows taking control of all Notstand systems and some systems out of the Notstand building, as for example the main steam relief valves. The controls from the ECR have priority over those from the Main Control Room once the ECR has been activated.

2.2 Emergency Feedwater System

An additional bunkerized feedwater system (independent from the feedwater system available with the Notstand system) is installed on each unit. Each emergency feedwater system can feed the other unit.

2.3 Specific Containment features

The containment consists of a steel liner and an armed concrete structure. The free volume between the liner and the concrete wall is referred to as the annulus space. To ensure the containment isolation function, a back-up containment seal water system is installed.

In addition, the containment is equipped with passive autocatalytic hydrogen recombiners and a filtered containment venting system which are designed for Severe Accident conditions.

2.4 Emergency power supply

In addition to both external grids (50 kV and 220 kV), one emergency power bus of each unit is continuously supplied by the Beznau hydro power station. In case of loss of external power, one additional emergency power bus of each unit can be supplied by another hydro power supply.
3 PROCEDURES AND DOCUMENTS FOR EMERGENCY RESPONSE IN CASE OF DESIGN BASIS ACCIDENTS AND BEYOND DESIGN BASIS ACCIDENTS

The KKB has established a set of procedures in order to give the control room operators the best information for normal operation, anticipated operational transients, design basis accidents and beyond design basis accidents. Figure 1 gives an overview on procedure type applicability with respect to the different operation modes from normal operation to severe accidents.

![Figure 1: Overall system of procedures for the Beznau plant operation](image)

After a Reactor Trip or a Safety Injection (SI) signal occurs, control room operators have to enter the EOPs, starting with E-0 ("Reactor Trip or Safety injection") procedure. The main goals of the E-0 procedure are to verify the status of automatic actions resulting from the reactor protection and safeguards system actuation and to identify the appropriate optimal recovery procedure to handle the on-going event. If the Reactor has successfully shut down and at least one emergency electrical power bus is available, the control room operators move to the appropriate procedure, which is normally an Optimal Recovery Procedure (ORP) to recover from one of the following events:

- “No break” type transients
- Loss of Coolant Accidents (LOCA)
- Secondary Line Breaks (SLB)
- Steam Generator Tube Rupture (SGTR).

Within each optimal recovery procedure, a continuous diagnostic takes place (looking at symptoms) making it possible to address time-evolving events, including combination of events (beyond design basis).

In parallel to the implementation of the ORPs, the control room operators have to monitor the critical safety functions (CSF) with the following priority:

- Subcriticality
- Core Cooling
Heat sink  
Integrity (of the reactor vessel)  
Containment  
Primary Inventory.

The critical safety functions are monitored through their corresponding status trees. Based on the decision criteria (plant symptoms), at any moment, the possible status of each one of the CSF is one of the following:

Table 2: CSF status and requested operator actions

<table>
<thead>
<tr>
<th>CSF Status</th>
<th>Example: Core Cooling CSF</th>
<th>Color Coding</th>
<th>Operator Action</th>
</tr>
</thead>
<tbody>
<tr>
<td>Satisfied</td>
<td>Subcooling is guaranteed</td>
<td>Green</td>
<td>Remain in procedure in use</td>
</tr>
<tr>
<td>Not Satisfied</td>
<td>Subcooling is lost</td>
<td>Red</td>
<td>Optional transition to the specified Function Restoration (FR) procedure</td>
</tr>
<tr>
<td>Severe Challenge</td>
<td>$650^\circ C &gt; \text{Core exit temperature} &gt; 376^\circ C$</td>
<td>Yellow</td>
<td>Prompt transition to the specified FR procedure</td>
</tr>
<tr>
<td>Extreme Challenge</td>
<td>Core exit temperature &gt; $650^\circ C$</td>
<td>Red</td>
<td>Immediate transition to the specified FR procedure</td>
</tr>
</tbody>
</table>

4 KKB EMERGENCY ORGANISATION

4.1 Normal Operation and anticipated operational transients

During normal operation and anticipated operational transients, the Beznau plant is controlled by the operating crew directed by the shift supervisor in the MCR. In case of operational problems, the shift supervisor has to inform the head of operations during normal working hours or the Pikett engineer outside working hours. Depending on the situation, he has to ask for technical advice or a decision.

The Pikett engineer (Pikett is a Swiss German term for an individual or organisation on duty) remains on the plant area during his duty time such that he can reach the MCR in a reasonable time. In case of operational disturbances he serves as an advisor for the shift crew and the involved maintenance staff. The licensing process of a Pikett engineer requires an engineering degree, qualification as a shift supervisor and additional specific qualification.

4.2 Design basis accidents and beyond design basis accidents

After a reactor trip has occurred, the shift supervisor must call for the Pikett engineer into the control room. The Pikett engineer must evaluate whether a technical emergency has occurred or not based on the following decision criteria:

- Safety Injection or Containment Spray actuation (automatic or manual)
- Shut down and/or the following cooldown of the plant might be difficult or complicated
- Station Blackout
- Non explicable reactivity transient
- Long term spent fuel pit cooling is jeopardized.

In contrast to plants in other countries, the intervention criteria for a technical emergency in KKB are determined in a very early stage of an accident.

If one of the above conditions is satisfied, the Pikett engineer declares the technical emergency and calls for the activation of the plant emergency organization. The structure of this emergency organization is shown on Figure 2.
As long as the Site Emergency Director (SED) has not overtaken his responsibility by formal declaration, the Pikett engineer acts as the SED. In this function, the Pikett engineer is responsible for control of the emergency and directs first actions on the site.

As soon at least 3 members of the Emergency Staff including one individual for the position of SED are on the plant, the leadership for the emergency may be handed over. Outside of working hours this step is expected one hour after the begin of the event, during working hours within 30 minutes.

The Emergency Staff consists of the plant department managers and some technical experts. The designated persons for the SED position are the plant manager and his deputy. If both persons are not available, one suitable member of the Emergency Staff may function as the SED.

The SED is responsible for all important technical and organizational decisions in emergencies. The Emergency Staff support the SED in terms of preparation and realisation of the decisions. If the SED is asked for a decision on a complex issue, the member of Emergency Staff will involve additional experts from the engineering support teams. The SED decides on basis of a proposal of the Emergency Staff within the next Emergency Staff meeting.

5 PROBLEMS TO BE DEALT WITH IN THE PLANT SPECIFIC TSC MANUAL

The crew on shift operates the plant according to the EOPs independently to a large extent. However the EOPs in the KKB include many instruction steps where the shift supervisor has to inform the SED or refer to him for a decision.

The plant specific “TSC manual” [6] is a decision making tool with the objective to cover all technical issues the shift supervisor could ask the SED to address during a technical emergency. The same technical issue can appear in several EOPs. In the TSC manual, each technical issue is the subject of a separate Evaluation.
5.1 Example

Part of Westinghouse approach to recover from SGTR and SLOCA accidents is to continue RCP operation, if possible. If RCP seal injection was interrupted due to station blackout or other reasons, restart of RCP is permitted only if the seals are in good condition. The objective of the subsequent evaluation is to find out whether the RCP shall be restarted or not. This particular technical issue appears in 13 steps of 11 different EOPs.

In the Beznau specific TSC manual, 50 different evaluations are documented. The following presents the split of the evaluations as it applies to Beznau.

5.2 Classification of the problems with respect to the different EOP packages

The generic TSC manual contains only 17 evaluations. Due to the particular safety features of the Beznau units, 33 additional evaluations have been added to its plant specific TSC manual. Figure 3 illustrates that a large amount of evaluations are applicable for the ECR. The reason is that a lot of systems located outside of the Notstand building cannot be controlled from the ECR, whereas the EOP for ECR requires taking actions on components located outside of the Notstand building, which requires assessing their accessibility.

Problems related to operation from the MCR are in minority. Furthermore, some generic technical issues are considered not relevant for Beznau SED decision.

Figure 3: Distribution of evaluations for the different EOP packages

5.3 Types of required actions by the SED

The required actions of SED ranges from “take notice of information” up to complex decisions between two or more alternate solutions (see Figure 4). “Take notice of information” means that hard criteria are satisfied and the SED has to be informed. “Guidance” will be given if an EOP step was not successful and the control room operators need instructions to solve that problem with the help of a substitute.
Half of the evaluations are developed to decide YES/NO, e.g. “Should the RPV be vented or not?” In some evaluations, the SED has to decide which method to apply, i.e. which cooldown method to use following a SGTR (backfill, blow down or steam dump).

5.4 Severity of considered accident situations

The majority of the evaluations in the Beznau TSC manual are developed to handle problems arising in beyond design basis situations. Even in the ORPs, some evaluations are addressing conditions outside of the expected plant behavior during design basis accidents.

Just 10 of 50 evaluations are developed to decide on measures in the range of design basis accidents, e.g. decision on the long term plant status after an accident and the already mentioned decision on cooldown method following a SGTR.

Figure 5 gives an overview of the different sources of problems handled in the evaluations. Some particular evaluations (6) handle problems arising in the whole bandwidth of ORP, ECA and FRP procedures.
Figure 5: Distribution of accident severity covered by the evaluations of the TSC manual
6 STRUCTURE OF THE TSC MANUAL

The first part of the manual informs how to use the manual and gives an overview where in the EOP packages are the technical issues applicable.

Fifty separate evaluations are documented. The structure of each evaluation is similar to each other:

1. Concern: This is a brief statement of the underlying concerns prompting the operators to consult with the SED or to obtain a recommendation from the SED.
2. Applicable EOPs: This is a list of EOPs during which the SED might be called upon to perform this evaluation.
3. Plant specific Considerations: These are plant-specific issues and/or design characteristics (in particular compared to the Westinghouse ERGs "reference plant") pertaining to this particular evaluation.
4. Plant conditions: This is a list of the most likely plant conditions anticipated to exist at the time the SED is called upon to perform this evaluation.
5. Prevailing cautions and notes: This is a list of the EOP cautions/notes anticipated to be in effect/applicable at the time the SED is called upon to perform this evaluation, and that are in some way directly related to the evaluation.
6. Guidance
   a. Evaluation Objectives: Essential question to be answered or determination to be made by the NFS in this evaluation. In other words, it is the expected output of the evaluation or decision-making process.
   b. Points to Consider: This is a comprehensive survey of points the SED should consider in performing this evaluation. The documented facts inform the SED about the available options and their advantages, disadvantages and risks.
   c. Review of background documentation: This is the documentation of all information related to the evaluation. This documentation can be found in the Background Information Documents that were developed together with the EOPs. Each EOP has its own BID that includes information about analyses that were realized to develop the strategy of the procedure, some information about the physics of the accident the procedure is supposed to deal with, as well as detailed explanation of each step of the dedicated procedure.
   d. References: Any known document that brings additional information about topics covered by the evaluation, but was too big to be included directly into the Manual.

7 EVALUATION EXAMPLE: DECISION ON REFILL OF ONE DRY STEAM GENERATOR

The above explained content of an evaluation is demonstrated using an evaluation with simple structure.

Applicable EOPs: The issue is integrated in two procedures concerning countermeasures at low SG narrow level (YELLOW CSF condition) in the MCR EOP package as well as in the ECR EOP package.

Plant conditions: Although the Beznau plant is equipped with four trains of feedwater, the plant specific EOPs consider the total loss of feedwater. This may be a result of a station blackout or an external event. Consequently, the critical safety function “Heat Sink” will eventually not be satisfied. One of the strategies applied in this event is to remove residual heat with one SG and isolate the other SG. So, the following situation may occur: One SG is dry and the level of the other SG is yet within the Narrow Range. The feedwater supply has now been reestablished.
In order to reestablish the secondary heat sink completely, the SED has to decide whether to refill the dry SG or not.

The **objective** of the evaluation is to decide whether the dry SG should be fed with feedwater after re-establishing of any feedwater system.

The **points to consider** contain the necessary information for the decision process; it is summarized here below:

- At least one SG is active and has a level within the narrow range. A YELLOW PATH condition means no urgent need for dry SG refill.
- The refill process has to be delayed in order to cool down the structures of the dry SG by heat losses.
- Thermal shock phenomenon must be considered, a critical temperature for the SG shell is given
- The preferred method is to use one of both auxiliary feedwater trains.

8 WORK FLOW AND STATUS OF PROJECT

The plant specific TSC manual has been developed in co-operation between Westinghouse and NOK, stepwise:

1. Identify of all steps in the plant specific EOPs requiring a decision of the SED
2. Summarize the amount of steps and definition of the problems respective evaluations
3. Establish a draft version of the manual (Westinghouse)
4. Check the draft material by a group of NOK experts
5. Establish the final version (Westinghouse)
6. Final overall check by a NOK expert not involved before

A total of 15 experts in both organizations were involved in the project.

At the time the present article was written, step 7 was on-going. The objective is to implement the manual by mid 2008.

9 FUTURE WORK

After release of the manual, training measures are needed.

1. Brief training for the **Emergency Staff** and the Pikett engineers in the field of covered problems and methods of the manual
2. Detailed training of the Pikett engineers and some experts of the **Emergency Staff**

This training is to provide the practical skills on how to use the manual.

Some evaluations of the manual are very detailed and difficult to apply in urgent situations. For these evaluations, development of simplified decision schemes is undergoing.

Thanks to the new full scope simulator on the plant site, the TSC manual will often be used during the emergency trainings of Pikett engineers and other experts. It is to be expected, that these trainings will provide a verification/validation of the manual and give opportunities to improve it on a continuous basis.

10 CONCLUSIONS

In some extreme situations, the EOPs rely on the TSC to provide advices of “what to do” and “how to do”. So far, the Beznau **Emergency Staff** had no specific tool to provide the operating crew the requested advice. With the Beznau TSC manual which has now been developed, the **Emergency Staff** have the needed tool to provide that advice taking all relevant information into account.
Moreover, the manual can be used for the training of Pikett engineers and other technical specialists, because both categories “Know how” and “Know why” are contained in the manual.

In the framework of the manual's development, plant specific design and perceptions led to changes and supplements in the considered 17 issues of the original (generic) TSC manual developed for Westighouse PWRs. Before implementing the original TSC manual, the individual NPP should evaluate the confidence of the guidance in the issues with plant specific design and accident analysis.

11 LIST OF ACRONYMS

<table>
<thead>
<tr>
<th>Acronym</th>
<th>Signification</th>
</tr>
</thead>
<tbody>
<tr>
<td>CSF</td>
<td>Critical Safety Functions</td>
</tr>
<tr>
<td>ECA</td>
<td>Emergency Contingency Actions</td>
</tr>
<tr>
<td>ECR</td>
<td>Emergency Control Room</td>
</tr>
<tr>
<td>EOP</td>
<td>Emergency Operating Procedure</td>
</tr>
<tr>
<td>ERG</td>
<td>Emergency Response Guideline</td>
</tr>
<tr>
<td>LOCA</td>
<td>Loss Of Coolant Accident</td>
</tr>
<tr>
<td>MCR</td>
<td>Main Control Room</td>
</tr>
<tr>
<td>NOK</td>
<td>Nordostschweizerische Kraftwerke</td>
</tr>
<tr>
<td>NPP</td>
<td>Nuclear Power Plant</td>
</tr>
<tr>
<td>ORP</td>
<td>Optimal Recovery Procedure</td>
</tr>
<tr>
<td>PWR</td>
<td>Pressurized Water Reactor</td>
</tr>
<tr>
<td>PWROG</td>
<td>Pressurized Water Reactor Owners Group</td>
</tr>
<tr>
<td>RCP</td>
<td>Reactor Coolant Pump</td>
</tr>
<tr>
<td>RPV</td>
<td>Reactor Pressure Vessel</td>
</tr>
<tr>
<td>SED</td>
<td>Site Emergency Director</td>
</tr>
<tr>
<td>SG</td>
<td>Steam Generator</td>
</tr>
<tr>
<td>SGTR</td>
<td>Steam Generator Tube Rupture</td>
</tr>
<tr>
<td>SI</td>
<td>Safety Injection</td>
</tr>
<tr>
<td>SLB</td>
<td>Steam Line Break</td>
</tr>
<tr>
<td>TMI</td>
<td>Three Mile Island NPP</td>
</tr>
<tr>
<td>TSC</td>
<td>Technical Support Center</td>
</tr>
<tr>
<td>WOG</td>
<td>Westinghouse Owners Group</td>
</tr>
</tbody>
</table>

12 REFERENCES


[4] NOK, Notfall-Vorschriften (emergency operating procedures), series NV-B-ff, last release 01 July 2008


ABSTRACT

To estimate the success criteria time windows of operator actions the conservative approach was used in the conventional probabilistic safety assessment (PSA). The current PSA standard recommends the use of best-estimate codes. The aim of this study was to estimate the operator action success criteria time windows, which were needed for updated human reliability analysis (HRA). The RELAP5/MOD3.3 best estimate code calculations were performed for the three selected initiating events: small or medium loss of coolant accident (LOCA) requiring manual auxiliary feedwater (AFW) start, loss of normal feedwater requiring AFW start, and LOCA requiring manual actuation of safety injection (SI) signal. In these events human actions are supplement to safety systems actuations. For calculations the qualified RELAP5 input model representing a two-loop pressurized water reactor, Westinghouse type, was used. The results of deterministic safety analysis were examined what is the latest time to perform the operator action and still satisfy the safety criteria. The results showed that the time available to perform operator action is larger than the time needed to perform operator action. The results showed that uncertainty analysis of realistic calculation in general is not needed for human reliability analysis when additional time is available and/or the event is not significant contributor to the risk.

1 INTRODUCTION

To estimate the success criteria time windows of operator actions the results of a severe accident code such the MAAP has been used in the conventional probabilistic safety assessment (PSA). However, information from these is often too conservative to perform a realistic PSA for a risk-informed application. On the other hand, the PSA standard [1] recommends the use of best-estimate code to improve the quality of a PSA. Therefore the aim of this study was to estimate the operator action success criteria time windows needed for updated human reliability analysis by using RELAP5/MOD3.3 best-estimate computer code [2]. The specified time windows are important for human reliability analysis (HRA) to determine the likelihood of operator actions. The human error probability of certain action is lower if operators have more time available. In the control room of a nuclear power plant there is a team of operators, which is supervised by a shift supervisor. If operators have 10 or more minutes of additional time for action, it can be expected that colleagues or shift supervisor can observe and correct a possible error of their colleague. Consideration of recovery causes lower human error probability and may cause a different impact of human error to the overall probabilistic safety assessment results. The actual times needed for
performing the action were assessed based on real simulator scenarios, while the time windows were determined by deterministic safety analysis. In the present study RELAP5/MOD3.3 best estimate code calculations were performed for the three selected initiating events: establishing auxiliary feedwater in case of small or medium loss of coolant accident (LOCA), establishing auxiliary feedwater in case of transients, and manual actuation of safety injection (SI) signal at LOCA. In these events human actions are supplement to safety systems actuations. For calculations the qualified RELAP5 input model representing a two-loop pressurized water reactor, Westinghouse type, was used [3].

2 SAFETY ANALYSIS METHODOLOGY

The success criteria time windows are described first. Then the input model for the RELAP5/MOD3.3 is described. Finally, for each of the three selected events the scenario is described. The realistic code calculations were performed by RELAP5/MOD3.3 P03 computer code [2].

2.1 Description of success criteria time windows

The idea of the HRA method [4] was to use those deterministic safety analyses to perform sensitivity studies of human actions, which are supplement to safety systems actuations. Sensitivity studies include variations of timing of human action to determine the latest time, when operators have to perform the needed action in order that the main plant parameters are not exceeded their limits. The core cooling success criteria as defined in [5] were used. It is assumed if the hottest core fuel/clad node temperature in the reactor core exceeds 923 K for more than 30 minutes or if temperature of the core exceeds 1348 K, the core damage may occur, which may lead to accident state. Based on the core temperature the time windows were determined.

2.2 RELAP5 Input Model Description

To perform this analysis, Krško nuclear power plant (NPP) has provided the base RELAP5 input model, so called “Master input deck”, which have been used for several analyses, including reference calculations for Krško full scope simulator verification [3], [6]. A full two-loop plant model has been used for the analysis. It includes the new Siemens-Framatome (now Areva) replacement steam generators type SG 72 W/D4-2. The model consists of 469 control volumes, 497 junctions and 378 heat structures with 2107 radial mesh points. Besides, 574 control variables and 405 logical conditions (trips) represent the instrumentation, regulation isolation, safety injection (SI) and auxiliary feedwater (AFW) triggering logic, steamline isolation, and so on.

2.3 Scenarios Description

Three scenarios are described, which were needed for updated human reliability analysis. In these scenarios the human actions are supplement to safety systems actuations. In the first scenario the human action was establishing AFW in case of small or medium LOCA assuming that high pressure safety injection (HPSI) system fails. In the second scenario the human action was establishing AFW in case of loss of feedwater (LOFW) transient. In the third scenario the human action was actuation of SI signal for the most limiting accident (excluding large break LOCA), i.e. small and medium LOCA.

In the case of small or medium LOCA in a nuclear power plant with the assumption that HPSI system fails, one of the means to cool the reactor is through the secondary side
depressurization providing that AFW system is operating. Normally, AFW system is automatically put into operation when main feedwater is lost. If the AFW pumps would not start automatically, operators should intervene. The success criterion requires operation of one of three AFW pumps to maintain the flow in order to depressurize the primary system below the accumulator injection setpoint at 4.9 MPa. Besides passive accumulators it was assumed that low pressure safety injection (LPSI) is available too. The parameter to indicate depressurization was primary pressure and the parameter to indicate core cooling was rod cladding temperature. As larger breaks can depressurize through the break in any case below accumulator injection setpoint pressure after some time, AFW is not needed for depressurization. Therefore analysis was performed for a spectrum of break sizes from 1.27 cm (0.5 inch) to 15.24 cm (6 inch) to determine, for which break sizes is needed operation of one AFW pump and for them the time available to start AFW was determined based on the parametric study varying delay of AFW start. The break was located in the cold leg between the reactor coolant pump and the reactor vessel.

The most limiting transient requiring operation of AFW is LOFW. The success criterion is that capacity of one train of AFW is adequate to remove decay heat, to prevent overpressurization of primary system, and to prevent uncovering of the core resulting in core heatup. The time when the operator succeeds to start AFW pump was varied. When the AFW pump started to inject into the secondary side, cooling of the secondary side caused the pressurizer pressure to drop below the pressurizer PORV closure setpoint and then below the maximum pressure capacity of HPSI pump. The HPSI injection efficiently prevents further core uncovery.

The third considered initiating event was LOCA without automatic SI signal actuation. This means that none of the safety systems including HPSI system, LPSI system and AFW system was assumed available. The whole spectrum of LOCAs from 1.91 cm (0.75”) to 15.24 cm (case 6”) break size was evaluated and for the most critical break regarding the time available to the operator it was shown that with establishing safety injection with 20 minutes delay the core heatup could be prevented.

3 RESULTS

Results are shown in Figs. 1 through 5 showing the most important variables, based on which the time available to perform operator action was determined.

3.1 LOCA calculations with manual actuation of AFW

The spectrum of break sizes was analyzed. For the most limiting break regarding time available it was shown that operation of AFW is not enough if not supported by manual opening of steam generator (SG) power operated relief valve (PORV). These two actions were assumed to be performed with the same time delay. The results for a spectrum of break sizes are shown in Fig. 1. From Fig. 1(a) it can be seen that 5.08 cm (2 inch) and larger breaks depressurize (through the break) in any case below accumulator injection setpoint pressure at 4.9 MPa after some time and therefore AFW is not needed for depressurization. On the other hand, 2.54 cm (1 inch) equivalent diameter break size and smaller need depressurization. As reactor coolant system (RCS) mass depletion (see Fig. 1(c)) and core heatup (see Fig. 1(b)) are earlier for 2.54 cm (1 inch) break than for 1.91 cm (0.75 inch) and 1.27 cm (0.5 inch) break, the 2.54 cm break was identified as the most critical regarding the time available to start AFW. Fig. 1(d) shows that for break 2.54 cm (and smaller), the steam generators start to dry out as their inventory is lost through SG PORVs, what caused core heatup.
Figure 1: Calculated results for spectrum of break sizes: (a) RCS pressure, (b) Core cladding temperature, (c) RCS mass inventory, (d) SG no.1 wide range level

To establish the depressurization by cooling through the secondary side, AFW is needed. However, as shown in Fig. 2(a), just by operating AFW the RCS pressure could not be depressurized and the core heated up (see Fig. 2(b)). The reason is that the SG PORV is cycling. Once SG is filled to normal level, the AFW injected intermittently following cycling of the PORVs. Depressurization could be efficiently achieved by manual opening of SG PORV providing that SG level is maintained above the minimum level by AFW.

Figure 2: Scenario with AFW available: (a) RCS pressure, (b) Core cladding temperature

To achieve the depressurization for the 2.54 cm break two operator actions were assumed to be performed, manual opening of SG PORV and manual AFW start as shown in Table 1. To determine the time window available to perform these two actions, scenarios with different delays of performing operator actions were analysed.
Table 1: Operator actions delay

<table>
<thead>
<tr>
<th>Case</th>
<th>Operator action</th>
<th>SG PORV full opening delay (min.)</th>
<th>AFW start delay (min.)</th>
</tr>
</thead>
<tbody>
<tr>
<td>A</td>
<td></td>
<td>0</td>
<td>Not available</td>
</tr>
<tr>
<td>B</td>
<td></td>
<td>30</td>
<td>30</td>
</tr>
<tr>
<td>C</td>
<td></td>
<td>50</td>
<td>50</td>
</tr>
<tr>
<td>D</td>
<td></td>
<td>80</td>
<td>80</td>
</tr>
<tr>
<td>E</td>
<td></td>
<td>100</td>
<td>100</td>
</tr>
<tr>
<td>F</td>
<td></td>
<td>120</td>
<td>120</td>
</tr>
</tbody>
</table>

Fig. 3(a) shows that RCS depressurization with SG PORV is efficient in preventing core heatup (see Fig. 3(b)), when delay is not too large. After the RCS pressure depressurizes below the accumulator injection setpoint, the RCS systems starts to fill as shown in Fig. 3(c). For case A with immediate depressurization of RCS with one SG PORV without AFW available the SG emptied in 40 minutes and core started to heat up 25 minutes later. From Fig. 3(d) it can be seen that for cases B to D the SG level drops approximately linearly and that cooling is sufficient until SG is not emptied. In cases A, E and F the SGs emptied and the core heatup was therefore unavoidable. From case E it can be seen if operator actions are performed immediately after SGs emptying the further heat up could still be prevented. Based on the set criteria 100 minutes are available to the operators.

Figure 3: 2.54 cm break scenarios with two operator actions: (a) RCS pressure, (b) Core cladding temperature, (c) RCS mass inventory, (d) SG no.1 wide range level
3.2 LOFW Calculations with Manual Actuation of AFW

The delays of AFW pump start from 40 min. to 70 min. were simulated to determine the time window for AFW start. The pressurizer pressure shown in Fig. 4(a) does not exceed 18.95 MPa what means that primary system was not overpressurized. From Fig. 4(b) the core heatup could be determined, while Fig. 4(c) shows the RCS mass inventory. Finally, in Fig. 4(d) it is shown the SG no. 1 level, which starts to efficiently fill after AFW pump start thus enabling RCS depressurization.

At the time when one AFW pump was started to inject into the secondary side, cooling of the secondary side caused the pressurizer pressure to drop below the pressurizer (PRZ) PORV closure setpoint and then below the maximum pressure capacity of HPSI pump (see Fig. 4(a)). The closure of the pressurizer PORV and coolant injection into primary system resulted in recovering the RCS inventory as shown in Fig. 4(c) and quenching the core as shown in Fig. 4(b). From Fig. 4(c) it can be seen that the RCS mass depletion depends mainly on the delay of one AFW pump start. The parametric analysis showed that the core significantly heats up with start of AFW pump delayed for 60 minutes or greater. The case with start of one AFW pump delayed for 50 minutes cause small core heatup and with delay of 60 minutes the core temperature is still below criterion 1348 K for core damage, while in the case with delay of 70 minutes this value is exceeded. Finally, in Fig. 4(d) is shown the steam generator wide range level. As already mentioned the start of AFW caused filling of steam generator and RCS system depressurization. Also it can be seen that the steam generator fills in approximately one hour.

![Figure 4: LOFW transient](image-url)

- **(a)** RCS pressure
- **(b)** Core cladding temperature
- **(c)** RCS mass inventory
- **(d)** SG no.1 wide range level
3.3 LOCA Calculations with Manual Actuation of SI

The results are shown in Fig. 5. At breaks smaller than 5.08 cm the RCS was not sufficiently depressurized as shown in Fig. 5(a) to enable accumulator injection, while larger breaks depressurize the RCS. Figure 5(b) shows that the temperature criterion 1348 K is first exceeded for 15.24 cm (case 6’”), then for 10.16 cm break (case 4’’), 7.62 cm (case 3’’), 1.91 cm (case 0.75’’) and the last for 5.08 cm (case 2’’). The reason is that for 5.08 cm break the accumulators were sufficient to cool the core until they emptied. At breaks larger than 5.08 cm the core starts to significantly heat up after the accumulators emptied. In general it can be concluded, the larger is the break the faster is the core uncover. For the 15.24 cm break the core starts to heat up at 20 minute. For the 5.08 cm break the core cladding temperature could exceed criterion at first peak in the case of considering the uncertainty. When SI signal was actuated at that time further core heatup was prevented (case 6’’ SI). Similarly this was the case for 5.08 cm break (case 2’’ SI). Therefore, at least 20 minutes are available for the operator action. In the case of this scenario the treatment of uncertainty is not needed as the time window is the shortest for the largest break in the spectrum.

![Figure 5: LOCA with manual actuation of SI:](image)

3.4 Results discussion

The times needed for performing operator actions were determined based on the simulator experience [7]. For starting the AFW the operator needs from 1 to 10 minutes, while for SI signal actuation 2 minutes are needed. When the time window is large, much of the additional time is available and there is no need to very accurately determine the time window even if the human factor event is an important contributor to the risk. For example, the time needed to start SI signal is 2 minutes and there is additional 18 minutes to perform this action. Considering typical uncertainties in peak cladding temperatures of 200 K based on previous uncertainty evaluations [8] and adiabatic heatup rate for 15.24 cm break, the criterion would be reached 3 minutes earlier. Equally important is also time uncertainty of reaching maximum temperature which is approximately 2 minutes according to [9]. The additional time considering uncertainties is still sufficient.

In the case of small and medium break LOCAs with the assumption that HPSI is not available, the depressurization is needed for breaks smaller than 5.08 cm. The break 5.08 cm is limiting as for this and larger breaks the RCS depressurize by itself. However, when the pressure drops below the accumulator injection point, the core is already heated up for 5.08 cm break. Considering the typical cladding temperature uncertainty of the best estimate calculation to be 200 K [8] the criterion 1348 K could be exceeded. The recovery action
would be questionable because of short time window. The uncertainty analysis was not needed, as the risk contribution of this event to the plant risk is insignificant.

On the other hand, establishing AFW at LOFW event is significant contributor to the risk, but the calculated time window gives sufficient additional time, even if conservative time window is considered in the human reliability analysis.

For the case of LOCA with delayed SI signal actuation it was shown that the additional time available is sufficient, therefore uncertainty analysis is not needed in spite of the fact that event is significant contributor to the risk.

All these examples showed that uncertainty analysis was not needed, as additional time was available and/or the event was not significant contributor to the risk. If the event is significant contributor to the risk or not, it is answered by PSA. Based on this it can be concluded that uncertainty analysis may be valuable only for significant risk contributors, when additional available time is small. For the selected examples this was not the case. In reference [10] it is proposed to estimate the uncertainty for an operator’s action in the PSA work scope by considering conservative time windows.

4 CONCLUSIONS

The operator action success criteria time windows were estimated using RELAP5/MOD3.3 for updated human reliability analysis. For the three selected cases the results of deterministic safety analysis were examined in sense how late after the required human intervention the operator performs its action that the safety criteria are not exceeded. This gives available time for operator to act. The results of deterministic analyses showed that in some cases the treatment of uncertainty for variables compared with safety criterion could significantly change the time window. However, based on the information from PSA regarding the contribution to the risk, uncertainty analysis was not needed, what greatly support the use of best estimate codes for probabilistic safety assessment. It can be concluded that uncertainty evaluation of realistic safety analysis may be needed only when there is little time for recovery action and the affected human factor event is an important contributor to risk.

ACKNOWLEDGMENTS

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REFERENCES


AN EVALUATION ON THE EMERGENCY OPERATION STRATEGY FOR THE RECIRCULATION SUMP BLOCKAGE AT KORI UNITS 3&4

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ABSTRACT

The purpose of this paper is to evaluate the emergency operation strategy on the recirculation sump blockage to address the recommendation in USNRC Bulletin 2003-01, “Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors” for Kori Units 3&4 in Korea. This Bulletin requires that the licensees evaluate the ECCS and CSS performance in compliance with 10CFR50.46(b)(5) (Option 1) or implement interim compensatory measures to reduce the potential risk due to LOCA-generated debris (Option 2) before the design improvement and installation of the strainer to resolve the GSI 191, “Assessment of Debris Accumulation on PWR Sump Performance”. WOG has developed a guidance to prepare the plant-specific responses to the Bulletin and performed qualitative evaluations and quantitative analyses for selected COAs to be incorporated into a plant’s EOP. The results of these evaluations and analyses were summarized in WCAP-16204-NP, “Evaluation of Potential ERG and EPG Changes to Address NRC Bulletin 2003-1 Recommendations” for both Westinghouse and Combustion Engineering type plants. In this paper, the analyses for operator actions securing containment spray pump and safety injection pump prior to recirculation alignment were performed to evaluate the applicability of the recommended operator actions to Kori Units 3&4 using computer codes, RELAP5/MOD3 and CONTEMP4/MOD5. Based on the analysis result, it is concluded that the operator action for early termination of one containment spray pump is applicable to the current EOP for Kori Units 3&4 if three or four Containment Fan Coolers are operating.

1. INTRODUCTION

A major safety issue in relation to long-term recirculation cooling after a LOCA is that LOCA-generated debris may be transported to the recirculation sump screen, eventually resulting in the blockage in the sump screen and loss of ECCS and CSS pump’s NPSH margin. The USNRC issued Bulletin 2003-01[1] to inform licensees of the potential adverse effects due to sump blockage by LOCA-generated debris in the recirculation operation mode during long-term cooling. In addition, the Bulletin requested the verification of conformance to
the existing regulatory requirements or implementation of any interim compensatory measures to reduce the risk due to LOCA-generated debris before the sump design improvement.

WOG launched a research program to respond to the Bulletin 2003-01 and recommended COAs to be incorporated into a plant’s EOP based on the qualitative evaluation and quantitative analysis for both Westinghouse and Combustion Engineering type plants. The results of this program were summarized in Reference [2]. The reference provides generic support and guidance for those licensees that choose to include operational changes as part of their response to the Bulletin 2003-1. WOG selected eleven COAs from the Bulletin 2003-1 and operator input from Procedures Working Group of WOG. Selection of COAs was based on 1) Were they identified in USNRC Bulletin 2003-01, 2) Could they increase the time to automatic switchover to recirculation and 3) Could they reduce the velocity of recirculation through the sump. Eleven COAs are as follows:

1. Secure one containment spray pump before recirculation alignment,
2. Manually initiate one train of containment sump recirculation earlier,
3. Terminate one train of HPSI/high-head injection after recirculation alignment,
4. Terminate LPSI/RHR pump prior to recirculation alignment,
5. Refill refuelling water storage tank,
6. Inject more than one RWST volume from refilled/diluted RWST or by bypassing RWST,
7. Provide more aggressive cooldown and depressurization following a small break LOCA,
8. Provide guidance on symptoms and identification of containment sump blockage,
9. Develop contingency actions in response to containment sump blockage, loss of suction, and cavitation,
10. Terminate HPSI/high-head injection prior recirculation alignment,
11. Delay containment spray actuation for small break LOCA in ice condenser plants.

Some COAs require quantitative analysis to verify the benefit of their implementation to plant-specific EOP. However, some COAs need only qualitative evaluation like the review of relevant steps in the current EOP. In particular, operator actions securing containment spray pump and safety injection pump before the alignment of containment sump recirculation mode need to be analyzed quantitatively. These quantitative and qualitative evaluations should be performed for the justification and implementation of these COAs into plant-specific EOP.

The purpose of this paper is to evaluate the emergency operation strategy on the recirculation sump blockage to address the recommendation in USNRC Bulletin 2003-01 for Kori Units 3&4, Westinghouse type PWRs, which consist of 3 RCS loops. Kori Units 3&4 have two trains of safety injection system. Each train consists of one HPSI pump and one LPSI pump. The plants have also two trains of containment heat removal system to maintain the safety function of containment pressure and temperature. Each train includes one containment spray pump and two containment fan coolers. The system is designed such that two containment spray pumps, four containment fan coolers or one train of the system provide enough heat removal to maintain the safety function of containment pressure and temperature. At the time of recirculation actuation, ECCS and CSS pumps transfer their suction from the RWST to the containment sump.

In this paper, the analyses for operator actions securing containment spray pump and safety injection pump prior to recirculation alignment were performed to evaluate the applicability of the operator actions to Kori Units 3&4 using computer codes, RELAP5/MOD3 and CONTEMP4/MOD5.
2. EVALUATION METHODOLOGY

Two major operator actions of COAs which require quantitative analyses were selected to evaluate the emergency operation strategy on the recirculation sump blockage. The analyses of two major operator actions were performed to justify quantitatively the implementation of the operator actions into Kori Units 3&4 EOP[3] in terms of reducing the flowrate through the sump screen and delaying the time to the start of containment recirculation mode during LOCA.

One category is to secure containment spray pump(s) before the alignment of containment recirculation mode (COA 1). This category is to verify the RWST depletion time and containment pressure and temperature within the EQ curve limit when any containment spray pump is secured.

The other category is to terminate safety injection pump(s) prior to the alignment of containment recirculation mode (COAs 4 & 10). This category is to verify PCT of LOCA licensing requirements when any safety injection pump is terminated. Although this operation action is conflicted with the current operator action steps, the analysis for this category was performed to provide the bases in case it is required to modify the emergency operation strategy related to safety injection pump termination.

The methodology applied in this paper referred the same methodology provided in Reference [2] including the selection of accident scenarios, aspect of evaluation, etc.

2.1 ANALYTICAL TOOL

Best-estimate simulation tools were used in the analyses. Thermal hydraulic response of the NSSS to LOCA was simulated using RELAP5/MOD3[4] and the containment pressure and temperature response to mass and energy release was simulated by coupling RELAP5/MOD3 and CONTEMP4/MOD5[5].

Figure 1 shows the RELAP5/MOD3 nodalization for Kori Units 3&4. The nodalization consists of 296 hydraulic volumes and 342 junctions for modelling primary and secondary systems including steam generator, pressurizer, reactor coolant pumps and various safety systems.

2.2. INITIAL CONDITIONS AND ANALYSIS SCENARIO

Analyses for two categories above were performed to evaluate the emergency operator actions for securing containment spray pump and safety injection pump. Table 1 enumerates the initial conditions for key parameters used in the analyses.
Table 1: Initial Conditions for Key Parameters

<table>
<thead>
<tr>
<th>Key Parameter</th>
<th>Value *</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core Thermal Power, MWt</td>
<td>2,958</td>
</tr>
<tr>
<td>Pressurizer Pressure, MPa (psia)</td>
<td>15.51 (2,250)</td>
</tr>
<tr>
<td>Pressurizer Level, %</td>
<td>58.93</td>
</tr>
<tr>
<td>RCS LOOP Flowrate, kg/sec</td>
<td>4443.52</td>
</tr>
<tr>
<td>Cold-Leg Temperature, K(°F)</td>
<td>562.59 (553)</td>
</tr>
<tr>
<td>Hot-Leg Temperature, K(°F)</td>
<td>600.37 (621.0)</td>
</tr>
<tr>
<td>Core Average Temperature, K(°F)</td>
<td>584.54 (592.5)</td>
</tr>
<tr>
<td>S/G Pressure, MPa (psia)</td>
<td>6.38 (926)</td>
</tr>
<tr>
<td>S/G Level, m</td>
<td>12.75</td>
</tr>
<tr>
<td>RWST Inventory, m³ (gal)</td>
<td>1,639.8 (433,200)</td>
</tr>
<tr>
<td>Containment Spray flowrate (for 1 CSP), kg/sec (gpm)</td>
<td>173.34 (2,750)</td>
</tr>
</tbody>
</table>

* Best-estimate value
**Category I: Securing Containment Spray Pump**

Operator action securing containment spray pump is intended to delay the time up to the start of recirculation and to reduce the flow rate through the sump screen when recirculation begins. It is also expected to reduce the differential pressure across the containment sump screen if there is any debris buildup.

To evaluate the duration of RWST depletion time, it is assumed that one containment spray pump is secured for a small break LOCA when two containment spray pumps are operating. It is also assumed that one containment spray pump is secured on CSAS and at 10 minutes after 2 inch- and 6 inch-small break LOCA, respectively. Operator action time considered is 10 minutes for analysis of emergency operation strategy generally, and the representative smaller break sizes for LOCA are selected because it will have a negligible effect on large breaks. It is also assumed that two trains of SIS are operating. It is expected that this operator action delays the initiation of recirculation operation.

In addition, one remaining operating containment spray pump may also stop if its electric bus is lost. This case needs a quantitative analysis to justify that containment temperature and pressure will be bound by the current licensing basis. To demonstrate the conformance with environmental qualification requirements in case of securing all containment spray pumps, it is assumed that one of the two operating containment spray pumps is turned off by operator at 10 minutes after LOCA, and at the same time the remaining spray pump is stopped due to the loss of electric bus to this pump. This case was performed for large break LOCA at the discharge leg of RCP by varying operable CFCs to quantitatively support the implementation of this operator action. The cases for operator action securing containment spray pump are summarized in Table 2.

**Table 2: Cases of Securing Containment Spray Pump**

<table>
<thead>
<tr>
<th>Input Case</th>
<th>Break Size</th>
<th>Containment Spray Pump</th>
<th>CFC</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 2 in SB</td>
<td>1 Pump Stop At CSAS</td>
<td>N/A</td>
<td></td>
</tr>
<tr>
<td>2 6 in SB</td>
<td>1 Pump Stop after 10 min</td>
<td>N/A</td>
<td></td>
</tr>
<tr>
<td>3 LB</td>
<td>All Pumps Stop after 10 min</td>
<td>1 CFC Available 2 CFCs Available 3 CFCs Available 4 CFCs Available</td>
<td></td>
</tr>
</tbody>
</table>

**Category II: Securing Safety Injection Pump**

Operator action securing safety injection pump is aimed at reducing the total flow through the sump screen and the rate of debris transport. It is expected to reduce the risk of sump blockage following a LOCA. This analysis includes shutting down redundant safety injection pumps that are not necessary to provide the required flows for the heat removal of the reactor core.
To evaluate the impact on the LOCA consequences, it is assumed that 1) all safety injection pumps, 2) one HPSI pump and two LPSI pumps, and 3) one LPSI pump and two HPSI pumps are secured at 10 minutes after large break LOCA for each case. The case of securing all safety injection pumps simulates the scenario for which the operator turns off flow from one safety injection train (1 HPSI and 1 LPSI) with a single failure causing the loss of safety injection flow from the other safety injection train.

This analysis was performed to evaluate the conformance with LOCA licensing requirements for peak cladding temperature when any safety injection pump prior to recirculation alignment is early terminated to delay ECCS suction switchover from the refuelling water tank to containment sump. These cases are summarized in Table 3.

<table>
<thead>
<tr>
<th>Input Case</th>
<th>Break Size</th>
<th>HPSI Pump</th>
<th>LPSI Pump</th>
</tr>
</thead>
<tbody>
<tr>
<td>4</td>
<td>LB</td>
<td>All Pumps Stop after 10 min</td>
<td>All Pumps Stop after 10 min</td>
</tr>
<tr>
<td>5</td>
<td>LB</td>
<td>Only 1 Pump Available after 10 min</td>
<td>All Pumps Stop after 10 min</td>
</tr>
<tr>
<td>6</td>
<td>LB</td>
<td>All Pumps Stop after 10 min</td>
<td>Only 1 Pump Available after 10 min</td>
</tr>
</tbody>
</table>

### 3. EVALUATION OF ANALYSIS RESULTS

Figures 2 and 3 show the depletion of the RWST during 2 inch- and 6 inch-small break LOCA, respectively. As shown in the figures, RWST depletion time for securing one of two operating containment spray pumps after small break LOCA is longer by 11.5 to 16.4 minutes than the case of two operating containment spray pumps. This operator action is effective for the prolongation of the RWST depletion time and delaying the recirculation operation as one of interim compensatory measures in response to Bulletin 2003-01.

Figures 4 and 5 show the containment pressure and temperature, respectively, when all containment spray pumps are secured during LBLOCA. As shown in the analysis results, the CFCs maintain the containment pressure and temperature within environment qualification curve. However, in case of one or two running CFCs, the containment pressure and temperature increase slowly after securing two containment spray pumps and start to exceed environment qualification limit at about \(3 \times 10^5\) seconds and \(3 \times 10^4\) seconds, respectively. Therefore, at least three CFCs or more should be operated to maintain the safety function of the containment pressure and temperature control.
The results of this analysis show that the operator action securing one of two operating containment spray pumps is applicable to the current EOP for Kori Units 3&4 if three CFCs or more are operating. In addition, these results for Kori Units 3&4 are similar to the results of analysis for securing containment spray pump in Reference [2].
Figures 4 through 8 illustrate the peak cladding temperature for cases 4, 5 and 6, respectively. Figure 4 shows the variation of the cladding temperatures when all safety injection pumps are stopped at 10 minutes after LBLOCA. The temperature of higher fuel regions starts to increase due to the lack of SI flow from about 1,000 seconds. Since these temperatures are rising very rapidly, it is expected that the cladding temperature could exceed the acceptance criterion for licensing analysis within a few minutes.
Figures 7 shows the variation of the cladding temperature when all safety injection pumps except only one HPSI pump are stopped at 10 minutes after LBLOCA. As shown in the figure, the cladding temperature for all fuel regions is less than 300°F as the core region is fully covered by the safety injection flow. Figure 8 depicts the variation of the cladding temperature when all safety injection pumps except only one LPSI pump are stopped at 10 minutes after LBLOCA. As shown in the results, the cladding temperature for all fuel regions is also maintained at less than 300°F as in the previous case. These indicate that the safety injection flow of only one HPSI or LPSI pump is sufficient to keep the core covered and to remove decay heat.
The analysis results for securing one HPSI or one LPSI pump show that this operator action is effective for the prolongation of the RWST depletion time delay with LOCA licensing requirements satisfied. However, as described above, this operator action conflicts with the operation step for safety injection termination of the current EOP for Kori Units 3&4. The results of this analysis will provide the backgrounds in case it is required to modify the emergency operation strategy related to safety injection pump termination. Therefore, this operator action securing safety injection pump is not applicable to the current EOP for Kori Units 3&4.

4. CONCLUSIONS

The evaluation of the emergency operation strategy on the recirculation sump blockage was performed to address the recommendation in the Bulletin 2003-01 for Kori Units 3&4. The analyses for operator actions securing containment spray pump and safety injection pump prior to the alignment of containment sump recirculation mode were carried out to evaluate the applicability of the recommended operator actions to Kori Units 3&4. It is concluded that the operator action securing one of two operating containment spray pumps before recirculation alignment is applicable to the EOP for Kori Units 3&4 if three CFCs or more are operating.

In addition, the COA for RWST refill has already incorporated into the current EOP for Kori Units 3&4. The COA related to symptoms and contingency actions of containment sump blockage is considered for its incorporation into the EOP for Kori Units 3&4.

ACKNOWLEDGMENTS

This study was performed as a part of the “Analysis of Long-Term Cooling Performance and Development of Relevant Operational Strategies Project” sponsored by KHNP.
NOMENCLATURE

BL  Bulletin
CFC  Containment Fan Cooler
COA  Candidate Operator Action
CSAS  Containment Spray Actuation Signal
CSS  Containment Spray System
ECCS  Emergency Core Cooling System
EOP  Emergency Operating Procedure
EPG  Emergency Procedure Guideline
ERG  Emergency Response Guideline
GL  Generic Letter
GSI  Generic Safety Issue
HPSI  High Pressure Safety Injection
KHNP  Korea Hydro & Nuclear Power Company, Ltd.
LPSI  Low Pressure Safety Injection
LOCA  Loss of Coolant Accident
NPSH  Net Positive Suction Head
NPP  Nuclear Power Plant
PWR  Pressurized Water Reactor
RAS  Recirculation Actuation Signal
RCP  Reactor Coolant Pump
RCS  Reactor Coolant System
RWST  Refueling Water Storage Tank
SIS  Safety Injection System
USNRC  United States Nuclear Regulatory Committee
WOG  Westinghouse Owners Group.

REFERENCES

Safety of Forthcoming Reactors
AP1000: The PWR Revisited

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ABSTRACT

The distinguishing features of Westinghouse’s AP1000 advanced passive pressurized water reactor are highlighted. In particular, the AP1000’s passive safety features are described as well as their implications for simplifying the design, construction, and operation of this design compared to currently operating plants, and significantly increasing safety margins over current plants as well. The AP1000 design specifically incorporates the knowledge acquired from the substantial accumulation of power reactor operating experience and benefits from the application of the Probabilistic Risk Assessment in the design process itself. The AP1000 design has been certified by the US Nuclear Regulatory Commission under its new rules for licensing new nuclear plants, 10 CFR Part 52, and is the subject of several combined Construction and Operating License for USA utilities applications. Currently the AP1000 design is being assessed against the EUR Rev C requirements for new nuclear power plants in Europe.

1 BACKGROUND

For nearly two decades, Westinghouse has pursued an improved pressurized water reactor (PWR) design. The result of this commitment is the AP1000, a simpler and more economical PWR. The design began to develop in the late 1980s in conjunction with the development of the “Advanced Light Water Reactor Utility Requirements Document (URD).” The URD, drafted under the direction of the Electric Power Research Institute (EPRI), came to embody the policy and design requirements of US power utilities for the next generation of nuclear power plants in the US. These requirements were also endorsed by the US Nuclear Regulatory Commission (NRC). In Europe the corresponding body of design requirements and expectations developed as the European Utility Requirements (EUR).

The URD addresses evolutionary and passive light water reactors. The two classifications have different requirements. Expectations are much higher for passive designs. Indeed, more should be expected from designs that are not constrained to follow the existing models. For example, passive designs are expected to be able to achieve and maintain safe shutdown for 72 hours following the initiation of a design basis event without needing operator action. The corresponding expectation for an “evolutionary” plant is 30 minutes before the operator must take action to protect the core. As defined by the URD, a passive reactor is also “simpler, smaller and much improved...”

Simplification is a major requirement of the URD and a major characteristic of the AP1000.
2 THE AP1000 OVERVIEW

AP1000 is designed around a conventional 2-loop, 2 steam generator primary system configuration that is improved in several details. AP1000 is rated at 3400 MW(t) core power and, depending on site conditions, nominally 1117 MW(e). The core contains 157 fuel assemblies, similar to Doel 4 and Tihange 3. AP1000 features passive emergency core cooling and containment cooling systems. This means that active systems required solely to mitigate design basis accident conditions have been replaced in AP1000 by simpler, passive systems relying on gravity, compressed gases, or natural circulation to drive them instead of pumps. AP1000 also does not require safety-grade sources of ac power. Class 1E batteries provide for electrical needs during the unlikely scenario requiring the activation of the passive emergency system.

Compared to a standard plant of similar power output, AP1000 has 35% fewer pumps, 80% less safety-class piping, and 50% fewer ASME safety class valves. There are no safety-grade pumps. This allows AP1000 to be a much more compact plant than earlier designs. With less equipment and piping to accommodate, most safety equipment is installed within the containment. Because of this, AP1000 has approximately 55% fewer piping penetrations in the containment than current generation plants. Seismic Category I building volume is about 45% less than earlier designs of comparable power rating. Figure 1 depicts the compact AP1000 station. Figure 2 compares the essential nuclear island building footprints to a typical, currently operating PWR. Seismic Category I buildings are shown in bold outline.

Here is a comparison of AP1000 safety margins to those of a currently operating plant.

Table 1: AP1000 safety margins

<table>
<thead>
<tr>
<th>Margin to DNBR, Loss of flow, %</th>
<th>Watts Bar</th>
<th>AP1000</th>
</tr>
</thead>
<tbody>
<tr>
<td>SG tube rupture Operator action required in 15 minutes</td>
<td>14</td>
<td>No operator action required</td>
</tr>
<tr>
<td>Small break LOCA Peak clad temperature, C</td>
<td>10 mm break Core uncovered PCT = 608°C</td>
<td>20 mm break Core stays covered</td>
</tr>
<tr>
<td>Large break LOCA peak clad temperature, C</td>
<td>977</td>
<td>&lt; 871</td>
</tr>
</tbody>
</table>

With a relatively large pressurizer, the AP1000 is more accommodating to transients and is, therefore, a more forgiving plant to operate.

The AP1000 is designed in accordance with the principles of ALARA to keep worker dose As Low As Reasonably Achievable. Features such as an integrated reactor vessel head package for quicker removal reduce the time required to do the job, and, therefore, reduce worker exposure. Attention to shielding, establishing distance from radiation sources, using low cobalt alloys, and using remote tooling or controls, are among the approaches that will minimize exposure throughout the plant. This is an area that has greatly benefited from operating plant experience.

Before delving into the further details of the AP1000 and how it is constructed, let us first review the regulatory status of this design.
3 AP1000 LICENSING AND REGULATORY STATUS

Nuclear power plants currently operating in the US were licensed under Title 10 CFR Part 50. In 1989 the US Nuclear Regulatory Commission (NRC) established alternative licensing requirements under 10 CFR Part 52. Prior to 1989 and under Part 50, all aspects of licensing from the design of the nuclear steam supply system to site-related topics remained open until after the plant was constructed. This left all aspects of a plant license application unsettled – and at risk - until virtually the entire plant capital investment was made. The current regulations under Part 52 ensure that all significant licensing issues have been resolved early in the process and with a high degree of finality.

Under Part 52 regulations, a plant design can be submitted for NRC Design Certification. The applicant is the plant design organization and the certification is generic and independent of any particular plant site. NRC approved and certified the AP1000 design under 10 CFR Part 52 in December 2005. The certification is valid for 15 years. Westinghouse submitted the AP1000 application in March, 2002.

Similarly, individual plant sites can be generally approved for construction of a nuclear plant through the Early Site Permit process under 10 CFR Part 52. This approval covers all elements affecting site suitability except for the specific effects of a particular plant design. These permits are valid for 10 to 20 years and can be extended for an additional 10 to 20 years.

With a design approved and certified and with a site that has received a permit, it then remains to merge these in order to actually proceed to construct and operate a specific nuclear power plant design at a specific site. This marriage of the two is the combined Construction and Operating License (COL) application. This application is made to the NRC by the site owner. Once the COL is granted by NRC, construction at the site may proceed.

This leaves the final step in the licensing process which is a verification that the plant has been constructed and will operate in conformance with the previously issued COL. This is accomplished by the Inspection, Tests, And Acceptance Criteria (ITAAC). Specific requirements for ITAACs for a particular case are established along the way in conjunction with the Final Design Certification and the COL applications.
Figure 3 summarizes all of this and identifies the US utilities that have declared that they will pursue a COL application. With the design certified for AP1000, preparing applications for COLs based on the AP1000 design can proceed directly.

- 10 CFR Part 52 (operating plants licensed under earlier 10 CFR Part 50)
- Resolve licensing issues early in the process and with high degree of finality

Figure 3: Licensing and Regulatory Status

4 AP1000 PASSIVE SAFETY SYSTEMS

What is meant by passive safety systems, the major differentiating feature of the AP1000? Let us start with the emergency core cooling system. This system comes into play only during transients or accidents which cannot be handled by the first-line of defense: the non-safety grade systems. In the current Generation II plants, the emergency core cooling system consists of redundant trains of high pressure and low pressure safety injection systems driven by pumps. These pumps force water into the primary system to replace core coolant in the event of a loss of coolant accident. Such pump-driven systems are termed “active” systems. The pumps take suction from tanks of borated water, valves are opened, and water is sent to the reactor vessel to cool the fuel rods. To increase reliability, multiple redundant trains may be installed. The net result is a substantial amount of machinery standing by for a call to action that designers and operators work very hard to never need.

By contrast, the AP1000 passive core cooling system uses staged reservoirs of borated water that are designed to discharge into the reactor vessel at various threshold state points of the primary system. To begin the description, let us first see the configuration of the AP1000 reactor primary coolant system shown in Figure 4. Now we can attach the essentials of the passive emergency core cooling system, as illustrated in Figure 5. There are three sources of borated replacement coolant and three different means of motivating the injection in AP1000:

1) Two core makeup tanks (CMT). Each CMT is directly connected to a RCS cold leg by an open “pressure balance” line. The balance line enters the CMT at the top of the tank, as shown in the figure. With outlet valves closed, the system is static. When actuated and check valves opened, water is forced out of these tanks and into the reactor vessel depending on and motivated by conditions in the cold leg via the always open balance line. Water from the RCS cold leg, which is hotter than water in the CMTs, will force the injection by its expansion into the CMT. If the cold leg is full of steam, steam will force the injection. CMTs are the first to actuate for smaller primary system breaks.

2) Two accumulators (ACC). These spherical tanks are 85% full of borated water and pressurized to 700 psig with nitrogen. Check valves open when pressure in the reactor vessel drops below 700 allowing the water in the tanks to flow into the reactor vessel. Large break LOCA’s, which cause rapid system de-pressurization, will result in the accumulators being the first to respond.
3) The in containment refueling water storage tank (IRWST). Located above the RCS piping, the IRWST will discharge by gravity to the reactor vessel after the RCS has been de-pressurized by a break or by the automatic depressurization system, also shown in Figure 5. Flow is initiated by a depressurization signal which activates squib valves which open using an explosive charge. The squib valves are in series with check valves in the injection lines.

These injection sources are connected to two Direct Vessel Injection nozzles on the reactor vessel dedicated solely for this purpose. The passive emergency core cooling system components are all located within the containment vessel. Without pumps to run, there is no need for emergency ac electrical power to maintain operation during an event. Any electrical power needed for the few safety valves and actuators that require it comes from 1E dc power, backed up by 1E batteries.

The injection system is enabled by an automatic depressurization system which executes a staged depressurization of the primary system initiated from any actuation of the CMTs that reaches pre-set water levels in those tanks.

The IRWST is part of the passive decay heat removal system. A heat exchanger inside the IRWST has an inlet from the reactor coolant system (RCS) hot leg and an outlet into the RCS cold leg. In the event of loss of RCS heat removal from the steam generators, the IRWST will absorb heat from the heat exchanger while primary system coolant circulates through the exchanger by natural circulation. After several hours of operation, the IRWST water will begin to boil. Steam from IRWST will begin to condense on the containment walls. The condensate will then be directed by a safety-grade gutting system back to the IRWST to continue the cycle.

The steel containment vessel located inside the concrete shield building provides the heat transfer surface that removes heat from inside the containment and rejects it to the atmosphere. Heat is removed from the containment vessel by the continuous natural circulation of air within the shield building/containment vessel annulus. During a design basis accident, the air cooling is supplemented by evaporation of water. This cooling water drains by gravity from a tank located on top of the containment shield building. The water runs
down over the steel containment vessel, thereby enhancing heat transfer. This passive containment cooling system design eliminates the safety-grade containment spray and fan coolers required for a conventional plant.

Elements of this system were extensively tested and documented as part of the basis for receiving NRC’s Final Design Certification. Figure 6 indicates the kind of simplification that results from AP1000’s passive system versus a standard PWR emergency system.

5 SEVERE ACCIDENT MITIGATION

The AP1000 is designed to retain melted core debris within the reactor vessel. To start with, the reactor vessel has no penetrations in the bottom head. In case of a severe accident, cooling water from the large IRWST can be used to flood the reactor cavity and cool the outside of the reactor vessel. The arrangement is shown in Figure 7. Specially designed reactor vessel insulation forms an annulus that allows cooling water to directly contact the vessel. Vents are provided for steam to escape the annulus. To complete the description, the vented steam will condense on the containment walls and be directed back to the cavity.

**Figure 6: Reduced Complexity**

**Figure 7: Severe Accident Design**
6  PROBABILISTIC RISK ASSESSMENT

One of the advancements that benefit the AP1000 is the further development of probabilistic risk assessment tools (PRA) and the application of these tools to the design process itself. The result for AP1000 has been a more effective combination of redundancy and diversity. This includes the defense-in-depth design that utilizes non-safety controls and systems as the first line of defense. If the first line systems are not capable of handling the event, the passive safety systems come into play. As revealed by the PRA, the risk of core damage and large radioactive release for AP1000 is extremely low. Here are the results for combined conditions of power, shutdown, and internal events, as well as fire and flood events:

- Core damage frequency, 5x10^-7
- Large release frequency, 6x10^-8.

For some perspective, here are some comparative results for core damage frequency:

Table 2: Comparative results for core damage frequency

<table>
<thead>
<tr>
<th></th>
<th>US NRC requirement</th>
<th>Current plants</th>
<th>URD requirement</th>
<th>AP1000</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core damage frequency</td>
<td>1 x 10^-4</td>
<td>5 x 10^-5</td>
<td>&lt;1 x 10^-5</td>
<td>5 x 10^-7</td>
</tr>
</tbody>
</table>

The AP1000 PRA led to the following statement by the US Advisory Committee for Reactor Safeguards in their report on AP1000 certification:

“This PRA was well done and rigorous methods were used…The fact that the PRA was an integral part of the design process was significant to achieving this estimated low risk.”

7  AP1000 REACTOR COOLANT PUMPS

Among the improvements embodied in the AP1000 are the reactor coolant pumps. AP1000 employs four canned motor pumps, two in each loop, as can be seen in Figure 4. Although such pumps have been used for decades in naval nuclear power plants, commercial PWRs have not employed them recently because the capacities required for Generation II nuclear plants began to exceed the capacity range of canned pumps prevailing at that time. However, in the meantime, the capacity of canned motor pumps has increased. The advantages of the canned motor design over conventional reactor coolant pumps are:

- Elimination of the shaft seal and the system needed to maintain seal injection
- By eliminating this seal and seal injection, a potential leakage path of primary coolant and a source of small break LOCA are also eliminated
- Canned motor pumps require very little or no maintenance and thereby also help lower worker dose.

8  AP1000 INSTRUMENTATION AND CONTROL SYSTEMS

The Westinghouse AP1000 instrumentation and control (I&C) system is comprised of the following subsystems:

- Operation and control centers (OCS)
- Data display and processing (DDS)
- Protection and safety monitoring (PMS)
Plant control (PLS)
Main turbine control and diagnostics (TOS)
Incire instrumentation (IIS)
Special monitoring (SMS)
Diverse actuation (DAS)
Radiation Monitoring (RMS)
Seismic Monitoring (SJS).

Following are highlights of some of these systems:

The OCS provides the human interface control facilities: the main control room, the technical support center, the remote shutdown workstation, the emergency operations facility, local control stations, and the associated workstations for each of these centers. The main control room, for example, is environmentally controlled and designed in conjunction with a comprehensive human factors engineering program conducted at Westinghouse. This program included an extensive operating experience review. Figure 8 shows a representative main control room layout for the AP1000.

Figure 8: Control Room

The plant control system (PLS) provides for control rod motion and position monitoring and controls the transport of heat energy from the nuclear reactor to the main steam turbine by means of the following major control functions:
Pressurizer pressure and level
Steam generator water level
Steam dump (turbine bypass)
Rapid power reduction
Various component controls (pumps, motors, valves, breakers, etc.)

The system provides for automatic and manual control.

The special monitoring system (SMS) is a non-safety-related system comprised of subsystems that interface with the I&C architecture to provide specialized diagnostic and long-term monitoring functions for detection of metallic debris in the reactor coolant system, core barrel vibration, and reactor coolant pump monitoring.

The diverse actuation system (DAS) provides I&C functions necessary to reduce the risk associated with a postulated common-mode failure in the PMS. The types of common-
mode failures addressed by the DAS include software design errors, hardware design errors, and test and maintenance errors.

9 CONCLUSION

The AP1000 is a PWR design that offers power generating companies a clear and practical alternative for new generating capacity. It was designed to be competitive with fossil fuel plants and will be overwhelmingly so as actions are implemented to reduce greenhouse gas emissions. With decades of operating experience to draw on, AP1000 incorporates proven technologies in a new combination to consolidate the advantages of nuclear power units while reducing their cost and complexity. It is important to recognize that among all the advantages of AP1000, it is also a demonstrably safer plant and an advanced design that has already been certified by the US.
The Safety Concept of the SWR 1000 with Active and Passive Safety Systems

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ABSTRACT

The SWR 1000 blends years of experience in design, construction and operation of BWRs with new concepts to achieve an optimum blend of increased safety and reduced costs. It has been developed to provide a reliable source of economically competitive and safe electricity.

This is achieved through the use of passive safety systems which function solely according to basic laws of nature such as gravity, natural convection and heat transfer induced by temperature differentials. In keeping with their principles of operation, the passive systems are of simple design and perform their intended functions without any need for instrumentation and control (I&C) equipment or an active supply of energy, such as electric power.

A key feature of the safety concept of the SWR 1000 is the fact that all design basis accidents can be controlled using just passive systems alone. Nevertheless, service-proven active safety systems are still intended to operate, if possible, before passive safety equipment takes over. The functional scope and degree of redundancy of these active systems can, however, be reduced.

This safety concept is supplemented by systems and actions for controlling a postulated core melt accident; i.e. for retaining the molten core inside the reactor pressure vessel (RPV) by external cooling.

By this way the consequences of this type of severe accident will remain restricted to within the plant itself and no emergency response actions will be necessary in the plant environs. Compared to existing plants, this capability for core melt retention represents an additional level of safety within the overall concept known as "defense-in-depth".

1 INTRODUCTION

The overall safety concept of nuclear power plants is basically built up around three safety objectives:

1. Attainment and maintenance of sub-criticality
2. Assurance of core cooling, and
3. Confinement of radioactivity.

To meet these objectives both during normal operation and in the event of off-normal operating conditions and accidents, safety systems are required to perform certain protective functions such as:
- Reactor shutdown
- Reactor pressure relief
- Core cooling
- Removal of residual heat from the containment
- Isolation of the reactor coolant pressure boundary
- Containment isolation.

To guarantee the necessary degree of functional reliability, these systems must satisfy stringent requirements in terms of redundancy, diversity, physical separation and component quality.

Until now, the safety systems have mainly comprised active systems; i.e. systems that can only perform their designated safety functions with the aid of a secure external energy supply, whether in the form of electric power or actuating fluids. Of course, the auxiliary systems supplying this energy have to satisfy the same requirements as the safety systems themselves.

The aim pursued in developing a new safety concept for the SWR 1000 was to guarantee the functional capability and reliability of safety equipment by providing simple and robust system designs and thus to increase plant safety as well as economic efficiency even further.

This is achieved through the use of passive safety systems which function solely according to basic laws of nature such as gravity, natural convection and heat transfer induced by temperature differentials. In keeping with their principles of operation, the passive systems are of simple design and perform their intended functions without any need for instrumentation and control (I&C) equipment or an active supply of energy, such as electric power.

A key feature of the safety concept of the SWR 1000 is the fact that all safety functions can be performed by passive systems. Design basis accidents can be controlled using just passive systems alone. Nevertheless, service-proven active safety systems are still intended to operate, if possible, before passive safety equipment takes over. The functional scope and degree of redundancy of these active systems can, however, be reduced.

This safety concept is supplemented by systems and actions for controlling a postulated core melt accident; i.e. for retaining the molten core inside the reactor pressure vessel (RPV) so that even the consequences of this type of severe accident will remain restricted to within the plant itself and no emergency response actions will be necessary in the plant environs. Compared to existing plants, this capability for core melt retention represents an additional level of safety within the overall concept known as "defences-in-depth".

All new passive systems have been tested either in full scale or in a scaled configuration to ensure the proper function of these systems. They will be tested again with full-scale, prototype components. In addition the external cooling of RPV has been tested successful [3].

2 SAFETY SYSTEMS

In line with the philosophy underlying the new safety concept of the SWR 1000, both active and passive systems are available for all safety functions. The passive systems are – just like the active systems – of multiple redundancy. In addition, active and passive systems provide functional diversity for one another. In the postulated event of failure of all active systems, accident control is still possible using only passive systems that need no I&C signals or power supplies to operate.
An overview of the safety systems of the SWR 1000 is given below, broken down according to the various safety functions listed above. The individual systems are not, however, described in detail. The active safety systems planned for the SWR 1000 are functionally equivalent to the systems already familiar from existing plants such as Gundremmingen B+C.

2.1 Systems for Reactor Shutdown

Diverse systems are available for shutdown of the reactor. These include the control rods, with their diverse drive systems:

- Electric motor drive for operational shutdown processes (control rod drive (CRD))
- Hydraulic drive for reactor scram

The SWR 1000 is also equipped with a fast acting boron injection system, which causes reactor shutdown independent from the control rods and is completely independent of control rod operations (Fig. 1).

The scram system is based largely on the accumulator tank concept implemented in German BWR plants, whereby the energy required for fast control rod insertion by hydraulic means is stored in tanks under nitrogen pressure. The tank pressure for the SWR 1000 is provided by steam pressure, much like a PWR pressurizer, to provide the required driving head. The water-filled tanks’ steam pressure blanket is generated by electric heaters in the upper area of each tank. This modification enables the reduction of the tank size, and prevents nitrogen from entering the RPV in the event of a malfunctioning tank isolation valve.

The scram system of the SWR 1000 also differs from existing collector tank systems because the two ring lines are not interconnected. Each ring line is supplied by two scram tanks, and supplies half of the control rod drives. Insertion of half of the control rods is sufficient to bring the core rapidly to the ‘hot sub-critical’ condition. This means that the connections between the two ring lines are eliminated, and the system remains separated into two redundant subsystems. Each scram tank has the capacity to insert half of the control rods within the required time. During power operation the tanks are each isolated via a quick-opening valve. Tripping of the scram function is initiated via solenoid pilot valves or via parallel diaphragm pilot valves that are actuated by Passive Pressure Pulse Transmitters (PPPTs).

The fast acting boron injection system (FABIS) is also based on the pressure tank concept: The quantity of pentaborate solution required for hot and cold sub-criticality is stored in a tank and injected into the RPV.

![Figure 1: Shutdown systems](image-url)
2.2 Systems for Core Cooling

The active system provided for core cooling in the event of a loss-of-coolant accident (LOCA) is the two-train residual heat removal (RHR) system operating in the low-pressure coolant injection (LPCI) mode ("active core flooding"). The RHR trains are connected to the emergency power supply system. The RHR system is designed to pump water from the pressure suppression pool into the RPV once reactor pressure has dropped below 10 bar. The pumping capacity of just one RHR train is sufficient for controlling such events. During flooding, heat is also removed from the containment to the plant's ultimate heat sink via the RHR heat exchangers.

Reactor depressurization in the event of a LOCA is assured by eight safety-relief valves (SRVs), four of which operate according to the depressurization principle and four according to the pressurization principle. Each main valve is provided with redundant and diverse means of actuation in the form of battery-backed solenoid pilot valves as well as pilot valves operated by PPPTs. The main valves are latched in the open position to ensure that they stay open when the reactor has been depressurized. The high degree of redundancy and diversity of the safety-relief valve system together with its capabilities for active and passive actuation guarantee a high level of system availability.

Passive removal of residual heat from the reactor is performed by the emergency condenser system (Fig. 2). The emergency condensers (ECs) are tubular heat exchangers which are submerged in the core flooding pools and are connected to the RPV by non-isolatable inlet and outlet lines. They are arranged such that if reactor water level should drop, steam will flow into the heat exchanger tubes where it condenses. Thus they perform two functions at once: heat removal from the RPV (by transferring the residual heat to the water of the core flooding pools through condensation) and core flooding (by returning the resulting condensate to the RPV).

Figure 2: Emergency condenser

2.3 Systems for Removal of Residual Heat from the Containment

In the case of all accidents which lead to the reactor becoming isolated from the main heat sink (e.g. loss of the main heat sink, LOCAs outside containment or natural and external man-made hazards), the two-train RHR system operating in the "flooding pool/pressure suppression pool cooling" mode is used to remove heat from the containment to an ultimate heat sink.
The containment cooling condenser (CCC) system, which mainly consists of four tubular heat exchangers, is provided for passive heat removal from the containment (Fig. 3).

The CCCs condense steam released inside the containment in the event of an accident and return the condensate to the core flooding pools. Condensation is effected by water from the shielding/storage pool located directly above the containment which flows by natural circulation through the tubes of the CCCs, thus removing the heat from the containment to the pool above. The water inventory of the shielding/storage pool is sized such that passive containment heat removal by the CCCs can continue for a prolonged period of time (without any need for further actions). If makeup water is supplied to the pool – e.g. via connected systems or external pumps – the grace period available before the entire pool water inventory has evaporated can be extended indefinitely.

![Figure 3: Containment cooling condenser](image)

### 2.4 Systems for Reactor Pressure Relief

Following isolation of the RPV from the main heat sink, reactor pressure relief is performed by the eight SRVs (see below). Active actuation of the system-fluid-operated main valves is effected by the solenoid pilot valves. Passive actuation of the SRVs for the pressure relief function is performed by spring-loaded pilot valves (Fig. 4).

![Figure 4: Safety relief valve system](image)
2.5 Systems for Reactor Coolant Pressure Boundary and Containment Isolation

Active isolation of the reactor coolant pressure boundary (RCPB) and of the containment is initiated by I&C equipment according to a concept comprising containment isolation at the main steam line penetrations, isolation at the feedwater line penetrations, and isolation at the penetrations of auxiliary system piping.

Two containment isolation valves (main steam isolation valves (MSIV)) are provided in each of the three main steam lines: one inboard gate valve and one outboard globe valve. These system-fluid-operated isolation valves are each actuated by two pilot valves (one solenoid valve and one pneumatic valve operated by PPPTs). The two feedwater lines are each provided with two containment isolation valves: one non-piloted check valve (inboard) and one system-fluid-operated gate valve outside the containment. The gate valve is actuated by two pilot valves (one solenoid valve and one pneumatic valve operated by PPPTs). The inclusion of the system-fluid-operated gate valve – serving as a second isolation valve of diverse design – has enabled the probability of loss of reactor coolant inventory due to a feedwater line break occurring outside the containment to be considerably reduced even further. Finally, appropriate means are likewise provided to isolate the RCPB from all other systems that penetrate the containment, except for those required to perform safety-related functions.

According to the design concept of the SWR 1000, there are no other high-energy piping systems conveying reactor coolant that are situated outside the containment apart from the main steam and feedwater lines. Hence only these lines are equipped with passive isolation devices.

2.6 Systems for In-Vessel Core Melt Retention in the Event of a Severe Accident

The new safety concept of the SWR 1000 makes the probability of a core melt accident even lower than at existing plants. Provisions have nevertheless been made to ensure control of this hypothetical event.

If there should be an impending risk of core melt, water from the core flooding pools is allowed to flow down by gravity into the lower section of the drywell surrounding the RPV through a special drywell flooding line. The RPV then becomes surrounded by water up to the elevation of its support skirt. If a pool of molten core material should collect in the bottom head of the RPV, enough heat can be removed to this water through the RPV wall without causing melting of the wall [2]. Thus it is possible to retain the melt inside the RPV. The steam produced by evaporation of the water on the RPV exterior is condensed by the CCCs and the resulting condensate circulates back to the drywell by way of the core flooding pools (Fig. 5).

Figure 5: Severe accident control
3 REDUNDANCY CONCEPT

Active and passive systems together fulfil the requirements for redundancy according to the single failure concept. In other words, the specified safety functions will still be performed even if systems should be unavailable on demand as a result of:

- Consequential failure
- Single failure
- Maintenance.

For the passive systems, the scope of failures that need to be postulated can be reduced since they do not contain any active components for which a single failure would have to be assumed and also since unavailability due to maintenance work can be disregarded because they are located inside an inerted containment which is not accessible during power operation of the plant. Single failures in passive components such as piping can be ruled out due their high quality of fabrication. For these reasons, the redundancy concept presented below still ensures that the specified safety functions can be fulfilled by just passive systems alone.

The requirement for diversity, aimed at avoiding common cause failures, is met by employing different passive systems, or active systems together with passive systems of diverse design, for all safety functions. In addition, the systems are installed with physical separation in order to protect them against the effects of flooding or fire, etc.

Fig. 6 provides a summary of the active and passive systems provided to fulfil the designated safety functions.

![Table showing active and passive systems for accident control](image)

The redundancy provided for the safety functions of heat removal from the RPV, core flooding and heat removal from the containment, largely fulfilled by innovative equipment, is as follows:

![Table showing capacities for heat removal from RPV](image)
Figure 8: Capacities for core flooding in the event of a LOCA

It can thus be seen that the diverse and redundant safety systems planned for the SWR 1000 provide adequate functional capacity, i.e. ≥ 100%, even when failures postulated according to the single failure concept are taken into account. The degree of redundancy of the passive systems has been selected such that accidents can be controlled by just the passive systems alone, even upon loss of the active RHR systems due to a consequential failure combined with maintenance or a single failure.

4 ELECTRICAL AND I&C SYSTEM DESIGN

The design of the electrical and I&C systems for the SWR 1000 is based on the requirements associated with normal plant operation as well as on fulfillment of the safety objectives for the reactor and its auxiliary systems. Whereas the objectives applying to normal operation (e.g. feeding of power into the main offsite power system, supply of power to electrical loads, automation for reducing the workload on operating personnel, and monitoring and display of plant conditions) are centered on achieving maximum system and component availability, and thus maximum availability of the overall plant, different objectives are pursued in the case of safety-related tasks, as already described above.

A particularly unusual aspect encountered in designing the electrical and I&C systems was the fact that they have to perform their tasks in conjunction with passive systems – systems that do not actually exist from an electrical or I&C viewpoint because they require neither power supplies nor I&C signals to operate.

Since, according to the SWR 1000's safety concept, all postulated accidents can be controlled by passive systems alone, this means that it is possible to limit the degree of redundancy of the electrical and I&C systems to 2 x 100%, or dual redundancy. The degree of redundancy generally prescribed in codes and standards for accident control assuming single failures and repairs is thus provided jointly by active and passive systems.

Naturally, for safety-related tasks for which no passive systems are available (e.g. monitoring), the necessary degree of redundancy is provided only at the active equipment.

The configuration of the plant electrical systems can be described, in simple terms, as follows. The generator feeds power to the 400 kV main offsite power system connection via a
generator transformer. Auxiliary power is tapped off between the generator breaker and the
generator transformer and fed to the 10 kV switchgear via two auxiliary power transformers. 
In the event of loss of the normal power supply via the generator and main offsite power 
system, as well as loss of the standby offsite power system, certain electrical loads must 
remain in operation or come into operation in order for safety functions to be performed. In 
terms of power supply requirements, a distinction is made between two categories of 
electrical loads:

- Loads for which a period without power is permissible
- Loads which must remain in operation without interruption or which must be 
immediately started up.

The first group is connected to the three-phase AC distribution boards of the two-train 
emergency power supply system, while the second is supplied with power either from the 
220 V DC system, via inverters or from distributed uninterruptible power supply equipment.

The I&C concept planned for operator control, monitoring and closed- and open-loop 
control is based on the following principles.

The technology implemented in the control rooms of today's nuclear power plants as 
well as in control rooms of the future is and will be characterized to a large extent by screen-
based displays and screen-based control equipment. The design concept of the control room is 
therefore based on a powerful process information system for operating data analysis and 
storage, trend analyses, preventive maintenance, process optimization, support of the 
operating personnel in fault analysis and other tasks.

Widespread use of field bus systems can minimize cabling requirements. This is made 
possible by the availability today of bus systems with high data-transfer capacities, as well as 
distribution of I&C equipment throughout the plant (ensuring sufficiently short response 
times). Bus systems provided for safety systems are designed taking single failures into 
account. In non-safety-related I&C systems the bus systems can be configured to tolerate 
certain single failures.

The safety I&C with its programmable logic control systems for actuation of the active 
safety systems primarily comprises the reactor protection system which is configured in 
multiply redundant subsystems that are decoupled to provide non-interactive separation of the 
redundant trains at their interfaces. Measured data acquisition is performed with quadruple 
redundancy for limit signal generation in the data acquisition computers and for two times 
(2 out of 4) logic gating in the processing computers. The output signals are gated once more 
in 1 out of 2 logics and actuation signals are generated for controlling the active safety system 
components.

AREVA NP's digital safety I&C platform TELEPERM XS is to be used for the reactor 
protection system of the SWR 1000. Analyses of this safety I&C equipment's expected 
unavailability under SWR 1000 boundary conditions have shown, for example, that an 
unavailability for reactor scram of 0.5 x 10^-8 per demand can be achieved. This provides 
significant margins to the requirements specified in pertinent codes and standards.

5 PROBABILISTIC SAFETY ASSESMENT

By combining well-known active safety equipment with passive safety systems of 
diverse design [1], the effects of Common Cause Failures are significantly reduced and the 
frequency of core damage states caused by plant-internal events is two orders of magnitude 
lower than that of contemporary plants. In fact, the integral frequency of core damage states 
calculated by proven methods for initiating events occurring during power operation and plant 
shutdown is only 8.4 x 10^-8 per year.
6 PROTECTION OF BUILDINGS AGAINST NATURAL AND MAN MADE HAZARD

The plant is designed in accordance with the European Utility Requirements and the Finnish requirements to withstand the effects of natural and external man-made hazards such as seismic events, aircraft crash (military fighter and large passenger aircraft) and explosion pressure waves. One of the goals in designing the plant was to accommodate the systems and components that require protection against these hazards in such a way inside the plant buildings that as few buildings as possible would have to be designed to withstand the loads from such events. Since all safety-related systems and components as well as those containing a high activity inventory are housed in the reactor building – except for the redundant emergency diesels along with their switchgear and the two safety-related closed cooling water and service water systems – the concept implemented for building protection is as follows:

The reactor building is the only building protected against all three major postulated hazards (seismic events, aircraft crash and explosion pressure waves). The buildings containing the emergency diesels and safety-related cooling water systems are protected against the effects of aircraft crash through physical separation (diesel buildings 120 m) and are designed to safely accommodate the loads imposed by a seismic event or an explosion pressure wave. An emergency control room building (bunker) is protected in the same way and is physically separated from the reactor supporting systems building housing the main control room. Since none of the other buildings contains safety-related equipment or components with a high activity inventory, they are only designed to withstand seismic loading according to standard industrial practices. These simplifications in system and component design reduce plant construction cost as well as, of course, plant operating cost since any decrease in construction cost always means less expenditure on inspection and maintenance.

Figure 10: Protection against external events
7 CONCLUSION

Through the consistent deployment of passive safety systems to provide redundancy and diversity for active systems, the safety level of the SWR 1000 has been able to be significantly increased compared to existing plants. The advantages of the new safety concept are:

- Reduced susceptibility of safety systems to failures
- Larger safety margins
- Good plant behavior in the event of accidents due to the fact that conditions change at a slower rate
- Grace periods of several days after an accident before operator intervention is required
- Significantly reduced impact of operator error on reactor safety
- No need for large-scale emergency response actions such as temporary evacuation or relocation of the neighboring population following a core melt accident.

The use of passive systems of simple design combined with a corresponding reduction in the number of active safety systems has enabled not only plant safety but also the economic efficiency of the plant as a whole to be considerably increased.

The need to comply with nuclear codes and standards provided the framework for development of the new safety concept. In accordance with the original design development contract, the SWR 1000 was based on German codes and standards for nuclear power plant construction and operation. In the meantime, however, the design concept has also been assessed with respect to the European Utility Requirements (EUR), a set of guidelines issued by Europe's major nuclear power plant operators. Here, too, compliance has been verified. Furthermore, as part of preparations to submit a proposal to build Finland’s fifth nuclear power plant, the Finnish Radiation and Nuclear Safety Authority, STUK, has certified that the SWR 1000 is basically licensable according to the Finnish codes YVL [4].

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ACR-1000™: Advanced CANDU Based on Proven Safety of CANDU Reactors

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ABSTRACT

This paper provides an overview of the Advanced CANDU Reactor®-1000 (ACR-1000®) design focusing on the safety systems and functions that are based on decades of design development and R&D of different CANDU reactor designs in Canada.

The ACR-1000 developed by Atomic Energy of Canada Limited (AECL) is a 1200 MWe-class light-water-cooled, heavy-water-moderated pressure-tube reactor, which has evolved from the well-established CANDU® line of reactors. The ACR-1000 design retains the basic, proven, CANDU design features while incorporating innovations and state-of-the-art technologies to ensure fully competitive safety, operation, performance and economics. Improvements include greater operating and safety margins plus adherence and compliance with the latest safety thinking regarding external events and risk assessment.

The ACR-1000 design complies with all applicable Canadian Nuclear Safety Commission (CNSC) regulatory requirements. Although not mandatory in Canada, the ACR-1000 design takes into account all applicable international requirements as appropriate. Moreover, IAEA’s safety standard “Safety of Nuclear Power Plants: Design Requirements”, NS-R-1, has been used in the ACR-1000 design.

AECL has recently issued the ACR-1000 Generic Safety Case Report (GSCR) that provides a site-independent overview of the design, safety characteristics, and bounding safety analysis of the ACR-1000, which demonstrates design readiness and licensability in Canada and abroad.

1 INTRODUCTION

This paper provides an overview of the Advanced CANDU Reactor®-1000 (ACR-1000®) design, with a focus on the safety systems and functions. The paper demonstrates a good design balance by taking advantage of proven traditional CANDU features with a number of innovations that enhance the safety, operability and maintainability of the reactor. The ACR-1000 design is based on decades of design development and R&D of different CANDU reactor designs in Canada and internationally. The ACR-1000 complies with all applicable Canadian regulatory requirements and the IAEA safety guidelines.

1.1 Background

The ACR-1000 developed by Atomic Energy of Canada Limited (AECL) is a 1200 MWe-class light-water-cooled, heavy-water-moderated pressure-tube reactor, which has evolved from the well-established CANDU® line of reactors. CANDU 6 units have already been licensed since the early 1980s until 2007 in a number of countries around the world: Canada, Argentina, Republic of Korea, Romania, and China. There are 11 CANDU 6 units

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currently in successful operation, which have exceptional lifetime operating performance records. The ACR-1000 design retains the basic, proven, CANDU design features while incorporating innovations and state-of-the-art technologies to ensure fully competitive safety, operation, performance and economics. Improvements include greater operating and safety margins plus adherence and compliance with the latest safety thinking regarding external events and risk assessment.

With a 60-year design life, the ACR-1000 is based on the use of a modular horizontal fuel channel surrounded by a heavy water moderator, the same feature as in all CANDU reactors. Each unit has a nominal gross output of 1165 MWe with a net output of approximately 1085 MWe. The unit uses low enriched uranium fuel, and the CANFLEX® ACR fuel bundle, which has a lower linear rating and higher critical heat flux relative to the 37-element bundles used to date as the standard CANDU 6 fuel.

The major nuclear systems of the ACR-1000 are located in the Reactor Building (RB) and the Reactor Auxiliary Building (RAB). Safety enhancements made in the ACR-1000 encompass improved safety margins and reliability of Safety Systems, which include two Shutdown Systems, enhanced Emergency Core Cooling System, Emergency Feedwater System (defined as the Emergency Heat Removal System), Containment System, and the associated safety support systems.

The development of the ACR-1000 safety case is largely based on the ACR-1000 Design Program, which consists of three distinct project phases: the Product Definition phase, Basic Engineering Program, and Project Final Design phase. The Generic Safety Case Report (GSCR), that was issued in June 2008, reflects the Basic Engineering Program (BEP), with the major structures, systems and components designed, developed, refined and integrated to a more precise level of detail on a generic basis (meaning non site-specific, non project-specific) to allow speedy and timely adaptation for specific customers and multiple project implementations. Generic system documentation covers system design requirements and descriptions, detailed flow sheets, system assessments, etc. The ACR confirmatory R&D program is designed to support reactor development and licensing. The use of PSA in design has been extensive, including using the knowledge and experience gained from operating CANDU plants and in defining reliability and risk targets.

The ACR-1000 design complies with all applicable Canadian (CNSC) regulatory requirements [1], while taking into account all applicable international requirements, as appropriate, such IAEA’s safety standard “Safety of Nuclear Power Plants: Design Requirements”, NS-R-1 [2].

The major proven safety features of the CANDU are, for comparison with other Gen III reactor systems: four-quadrant separation; modular core design with replaceable and inspectable pressure tubes; negative reactivity coefficients backed by multiple, diverse shutdowns systems; inherent built-in cooling of the core by moderator, shield tank and reactor vault water; and a robust containment design with diverse cooling and heat rejection systems. These contribute not only to an ultra-low Core Damage Frequency (CDF), but also provide inherent physical barriers against accident progression.

1.2 **Generic Safety Case Report (GSCR)**

The Generic Safety Case Report (GSCR) [1] has been prepared to provide an integrated, site-independent overview of the ACR-1000 safety design requirements, a demonstration of the design compliance with Canadian regulatory requirements, and document the Level 1 Probabilistic Safety Assessment (PSA) and the comprehensive bounding deterministic safety

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analysis. This 20 chapter, 3000 page report follows the scope and content of the Preliminary Safety Case Report (PSAR), and it consolidates key design information from ACR-1000 support documents, and presents the basis of the ACR-1000 safety case based on the detailed design completed during the basic ACR engineering program.

2 OVERVIEW OF THE ACR-1000 SAFETY DESIGN

2.1 Safety Design Approach

Consistent with the overall safety concept of defence-in-depth, the design of the ACR-1000 plant aims to, as far as practicable: prevent, and reduce challenges to the integrity of physical barriers; maintain integrity of any barriers when and if challenged; and obviate failure of a barrier as a consequence of the failure of another barrier. The objectives of this approach are to provide adequate means to maintain the plant in a normal operational state, to ensure proper short-term response immediately following an initiating event, to facilitate the management of the plant in and following any DBA, and in certain defined accident conditions, beyond the DBAs. This includes the design of the core to have a negative power coefficient and a small negative coolant void reactivity under nominal design conditions, improved performance of safety systems, and provision for a robust containment design meeting Canadian and international practice for the siting and licensing of new nuclear plants.

2.2 Defense in Depth

The ACR-1000 design has evolved from a proven CANDU line of products that has always used the defence-in-depth principle as a basis for design. In addition, the ACR-1000 design includes additional inherent and engineered safety features and incorporates the five major classic physical barriers to the release of radioactive materials to the environment, as follows:

a) The fuel matrix. The bulk of the fission products generated in the fuel are contained within the fuel grains or on the grain boundaries, and are not readily available to be released even if the fuel sheath fails.

b) The fuel sheath. There are large margins to fuel sheath failure under normal operating conditions.

c) The heat transport system (HTS). Even if fission products are released from the fuel during an accident, they will be contained within the HTS, which is designed to withstand the pressure and temperature loading resulting from the accident conditions. Figure 1 presents a schematic of the HTS.

d) Containment. Designed to withstand major internal and external forces, and retain its integrity. In the event of a DBA, automatic containment isolation will occur, ensuring that any subsequent release to the atmosphere is extremely small. Figure 2 presents a schematic of the containment building.

e) The exclusion zone. To provide an additional physical mechanism to limit doses to the public; this siting requirement ensures that even if fission products were released from containment they are dispersed in the atmosphere limiting any harm to any member of the public.

As a part of the inherent safety features, the moderator system in all CANDU designs provides a key additional heat sink, providing another means of core cooling and maintaining the barriers to the release of radioactive materials to the containment. Also, even following in-core LOCAs in which fission products are released from a channel, the fission products must pass through the moderator water, where the majority are retained.
This classic concept of defence-in-depth for physical systems is also extended and applied to all management activities, whether organizational, safety, behavioural, or design-related, thus in effect providing a sixth and overarching safety management barrier. By ensuring that all safety-related activities are subject to overlapping provisions, even if a failure occurs, it is detected and compensated for or corrected by appropriate measures. Application of the concept of defence-in-depth throughout design and operation provides a graded protection against a wide variety of postulated transients, AOOs, and DBAs, including those resulting from equipment failure or human action within the plant (internal events), and events which originate outside the plant (external events). Application of the concept of defence in depth in the ACR-1000 design provides a series of levels of defence (physical barriers, quadrant separation, safety management, inherent features, equipment, and procedures) aimed at preventing accidents and ensuring appropriate protection in the event that prevention fails, and ensuring not only low probabilities of occurrence but utilizing the reactor’s redundant and diverse safety features as shown below.

2.3 Independence and Separation

Physical and functional separation of systems important to safety performing the same safety function provides independence to ensure that common cause events and functional interconnections do not impair performance of the required safety functions. The common cause events that tend to drive the requirements for independence typically include fires, flooding, earthquakes, explosions, missiles, pipe whip, electrical faults, electromagnetic or radio frequency interferences, and software errors.

ACR-1000 safety and support systems are designed in conformance with the philosophy and safety objective of physical and functional separation for SSCs important to safety required by References [2] and [3]. Basically, any two systems or system divisions or redundant components carrying out the same function need to be separated and treated as being independent in safety analyses or Probabilistic Safety Assessments (PSAs).

The philosophy is applied as follow:
- Separation of safety systems from process and control systems;
- Separation between safety systems;
- Separation of redundant SSCs important to safety.

The four fundamental nuclear safety functions (Control, Cool, Contain and Monitor) are generally provided by at least two totally redundant systems or subsystems, and the trip signals to actuate these systems are provided by four redundant instrumentation channels that feed redundant actuation logic circuitry. This is done to ensure high reliability in the execution of these essential safety functions. Independence must be ensured between redundant systems and between redundant parts of a system.

To address important common cause events such as a fire requires both physical and functional separation. In addition to limiting the direct damage by the fire, and the adverse environmental conditions due to heat and smoke, physical boundaries prevent fire damage causing adverse electrical effects between connected systems, because of functional separation of redundant systems or subsystems.

Different philosophical approaches can be taken in the design and layout of the plant to achieve the goal of physical and functional separation. For the ACR-1000 design, independence is provided between redundant systems and between redundant divisions and components within those systems to mitigate the consequences of common cause events. The “Four Quadrant (4Q) Separation Philosophy” consists of four separate areas or “quadrants” of systems important to safety, and the associated quadruplicated instrumentation channels and power supply divisions for safety system instrumentation. The loss of one quadrant due to a
common mode event still allows continued safe operation with the remaining three quadrants intact (but still only two are required to perform the fundamental safety functions, each generally being 4x50% capacity). This approach provides an added advantage with regard to on-line maintenance and helps achieve high capacity factors. The ACR-1000 quadrant layout is provided in Figure 2.

2.4 Safety Systems

The ACR-1000 safety systems are those designed to shut down the reactor, remove decay heat, and limit the radioactivity release subsequent to the failure of normally operating process systems. These consist of the Shutdown System 1 (SDS1), Shutdown System 2 (SDS2), Emergency Core Cooling (ECC) system, Emergency Feed Water (EFW) system, and Containment System. Safety support systems are those that provide services needed for proper operation of the safety systems (e.g., electrical power, cooling water, and instrument air).

Shutdown System 1: SDS1 is a mechanical rod design inserting into the low pressure moderator tank (so no rod ejection is possible) and quickly terminates reactor power operation and brings the reactor into a safe shutdown condition by dropping shut-off rods into the reactor core. Reactor operation is terminated when a certain neutronic or process parameter enters an unacceptable range. The measurement of each parameter is performed by channel and the system is initiated when any two of the four SDS1 trip channels are tripped by any parameter or combination of parameters.

Shutdown System 2: SDS2 provides a second diverse and independent chemical method of quickly terminating reactor power operation by injecting a strong neutron-absorbing solution (gadolinium nitrate) into the moderator when any two of the four SDS2 trip channels are tripped by any parameter.

Emergency Core Cooling System: The ECC system is designed to supply emergency cooling water to the reactor core to cool the reactor fuel in the event of a LOCA. The design basis accidents are LOCA events where ECC is required to fill and maintain the HTS inventory, and remove decay heat from the fuel. The ECC function is accomplished by two sub-systems:

- The emergency coolant injection (ECI) system, which immediately injects high-pressure coolant into the HTS after a LOCA.
- The Long Term Cooling (LTC) system provides long-term injection including coolant recovery after a LOCA. The LTC system is also used for LTC after reactor shutdown following other accidents, and to allow for routine maintenance activities.

Emergency Feedwater System: The emergency heat removal function is accomplished by the EFW system which is designed to provide cooling water to the secondary side of the steam generators to enable the steam generators to remove the decay heat to the ultimate heat sink on a loss of normal feedwater supply (main feedwater and start-up feedwater). The EFW system is designed to meet the CNSC licensing requirements for an emergency heat removal system.

Containment System: The basic function of the containment system is to provide a continuous, pressure-retaining envelope around the reactor core and HTS. Following an accident, the containment system minimizes release of resultant radioactive materials to the external environment to well below regulatory limits.

The containment system includes the steel-lined, prestressed concrete Reactor Building (RB) containment structure, main and auxiliary airlocks, containment cooling spray for pressure suppression, and a containment isolation system consisting of valves or dampers in the ventilation ducts and certain process lines penetrating the containment envelope. The
design ensures a low leakage rate and provides a pressure-retaining boundary for all DBAs causing high pressure and/or temperature inside containment.

The containment system automatically closes all penetrations open to the RB atmosphere when an increase in containment pressure or radioactivity level is detected. Measurements of containment pressure and radioactivity are quadruplicated and the system is actuated using two-out-of-four logic.

2.5 Safety Support Systems

Safety support systems provide services needed for proper operation of the safety systems for the ACR-1000 plant.

The ACR-1000 design includes a Reserve Water System (RWS) with a reserve water tank (RWT), located at a high elevation in the RB to provide an emergency source of water by gravity feed to the steam generators (back-up EFW), containment cooling spray, calandria vessel, reactor vault, and HTS, if required.

The Electrical Power Systems (EPS) supply all electrical power needed to perform safety functions under transient and accident conditions and non-safety functions for Normal Operation (NO). The essential (safety support) portions of the systems are seismically qualified and consist of redundant divisions of standby generators, batteries, and distribution to the safety loads.

The Essential Cooling Water System (ECW) system circulates demineralized cooling water to systems important to safety. The ECW system is seismically qualified and is comprised of four separated closed loops. All four loops operate during NO.

The Essential Service Water System (ESW) disposes heat from the ECW system to the ultimate heat sink. The ESW system is seismically qualified and comprised of four separated open loops. All four loops operate during NO.

The Compressed Air System (CAS) provides service air, instrument air, and breathing air to different safety systems and power production systems in the plant.

The Chilled Water System (CWS) supplies water to air conditioning and miscellaneous equipment, and provides sufficient cooling capacity during NO. The system includes two separate systems; one shared system (between two reactor units) serving loads during normal plant power production function, and one unitized system serving loads credited for safety support function.

3 ACR-1000 COMPLIANCE WITH CANADIAN AND INTERNATIONAL SAFETY REQUIREMENTS

The ACR-1000 design has been developed to ensure its compliance with the Canadian regulations and regulatory requirements, and with the nuclear series of standards that are prepared and issued by the Canadian Standards Association (i.e., the CSA N285 to N293 standards series, including the N286 series of standards on Quality Assurance), with the National Building Code of Canada subject to the exemptions identified in CSA N293, and with the National Fire Code of Canada. The design also follows relevant sections of the ASME Codes and Standards for boiler and pressure vessel nuclear components, with the sole exception of the pressure tubes which are subject to special inspection and licensing requirements (failure of a CANDU pressure tube is acceptable and manageable event with limited acceptable safety consequences in a CANDU reactor, whereas failure of a pressure vessel in an LWR is not acceptable).

AECL focus has been to design ACR-1000 to primarily meet the Canadian regulatory requirements, and thus place the initial emphasis on ACR-1000 design licensability and construction in Canada. However, as part of international marketing of the ACR-1000
designs, reviews that were carried out by the US NRC and in the United Kingdom, give high confidence that the ACR-1000 design is robust and that it will meet regulatory requirements in foreign jurisdictions. Both the US and the UK Nuclear Installations Inspectorate (NII) experience indicate that the ACR design is consistent with the regulators’ expectations of a robust design that provides adequate protection against potential accidents in a manner that meets modern international good practice.

AECL has conducted a thorough review of CNSC’s Generic Action Items (GAIs) and licensing-related OPEX issues for their applicability to and resolution by the ACR-1000 design. The improvements adopted for the current ACR-1000 reference design ensure that all of the issues raised in the past by the Canadian regulator have been addressed in the design.

In addition, AECL has completed a comprehensive review of the requirements in the IAEA document NS-R-1, “Safety of Nuclear Power Plants: Design Safety Requirements”, and determined compliance of the ACR-1000 design with the IAEA requirements. This provides confidence that the ACR-1000 design will be licensable in other international jurisdictions.

4 ACR-1000 DESIGN FEATURES – ENHANCED SAFETY ROBUSTNESS

4.1 Plant Design Objectives

4.1.1 Power Generation Objectives

Each unit of the two-unit ACR-1000 plant is designed to have a core thermal output of 3200 MWth, with a nominal gross electric output is 1165 MWe. The capability criteria for power cycling of the ACR 1000 design are:

- Controlling the plant automatically in either the turbine-following-reactor mode or the reactor-following-turbine mode.
- Continuously compensating for grid fluctuations of plus or minus 2.5% full power, while operating in the range 90% to 97.5% of full power.
- Reducing reactor power quickly from steady state 100% power operation to 75%, remaining at that reduced power level for any length of time, and then returning to full power.
- Periodic load following down to 60% of full power.
- On loss of line, a safe transition to continued operation at house load.

4.1.2 Reliability and Availability Objectives

The ACR-1000 plant is designed so that the plant is readily maintainable over its full 60 year operating life, including pressure tube (PT) replacement, to provide a high confidence in achieving a lifetime capacity factor of greater than 90%. Noting that world experience shows major overhauls are required in all plants during their life, the planned and unplanned incapability has been minimized allowing an operating capacity factor of 95% to be achieved on a year-to-year basis. Plant reliability is improved by ensuring the following are implemented on an integrated basis:

- High unit capability (which is a measure of a plant’s ability to stay on-line and produce electricity), by minimizing time lost due to unplanned outages and reducing duration of planned outages.
- 3-year duration between planned outages, with an outage duration time of 21 days.
- Major outages lasting less than 365 days for every 30 years.
- Annual forced outage rate of less than 1.5%.
• Low loss factor (a measure of the effects of equipment and component ageing), and unplanned shutdowns or outage extensions, by implementing equipment performance and material condition monitoring programs.
• System design changes incorporated to address lessons learned from operational experience, including ease of maintenance and operation.
• Time to enter and exit Guaranteed Shutdown State (GSS) is minimized.
• An enhanced maintenance program is used to maximize component life and minimize component replacement time, thereby minimizing radiation exposure, replacement costs, and the number of operating and maintenance personnel required.
• Information technology and configuration management systems and processes are used to support operations and maintenance (O&M).
• Retain proven technology and basic CANDU design features is used, including:
  o On-power refuelling.
  o Horizontal fuel channel design.
  o Heavy-water-moderated reactor core.
  o CANDU type fuel bundle (CANFLEX®-ACR fuel).
• Traditional CANDU design features enhanced to meet or exceed target reliability and performance of the systems important to plant availability.

4.1.3 Operability and Maintainability Objectives

Improved operability and maintainability in support of high plant reliability is assured by the following:
• Operational improvements allow increased on-line testing and maintenance. For example, the ACR-1000 plant uses four-quadrant separation concept for the safety and safety support systems to improve operability and safety.
• Modernized control room interface based on extensive feedback provided by licensed control room operators in current CANDU operating plants. The annunciation system is designed for improved event diagnosis.
• Systems, controls and procedures that minimize the potential for control room operator error. This is accomplished by providing a panel and console with careful assessment of the operator interface, automating frequent operations, validating normal and abnormal operating procedures, and providing a system that presents alarms in a sorted fashion to enable event diagnosis.
• Human factors principles and criteria are included in the design of systems, facilities, equipment, and procedures.
• To facilitate commissioning, performance tests are documented and provisions built into the design for ease of execution.
• Credited operator actions and procedures related to abnormal operations are practical to perform and validated during the design phase.
• The core physics and the fuel handling system are designed to allow and minimize maintenance of fuelling machines.
• Operational feedback and experience are incorporated into the design.
• Good access and lifting devices for maintenance purposes is provided in the design based on laydown and access needs.
• Facilities, specialized tools, flasks and other equipment, and procedures are included to allow for the inspection, disassembly and/or replacement of major pieces of

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equipment in the plant, including but not limited to fuel channels, fuelling machines, steam generators (SGs), large pumps and heat exchangers, and turbine generator.

- Equipment standardization is applied in order to reduce parts cost, and facilitate maintenance training and execution.

4.1.4 Safety Design Criteria

The safety design objective requires excellence in safety to protect the general public, plant personnel, and the environment, as well as plant investment.

As noted above, the ACR-1000 plant is designed based on the “defence-in-depth” safety philosophy applied to the all CANDU plants with enhancements to further improve the overall safety of the plant. This includes the core being designed to have a negative power coefficient of reactivity and a small negative coolant void reactivity (CVR) under nominal design conditions, improved performance of safety systems, and provision for a robust containment design to meet Canadian and international practice for new plants, as applicable.

The design basis events are defined, determined, and analyzed according to the internationally-applied safety analysis approach in which the worst single failure in a mitigating system is postulated along with each initiating event. The required analyses for the design basis events are performed with conservative computational methods and assumptions, and meet mandated regulatory acceptance criteria.

To ensure the safety design bases are met, a series of Safety Design Guides (SDGs) are applied in the ACR-1000 design process in order to ensure that the design complies with the appropriate safety and regulatory requirements. These SDGs identify systems important to safety, and provide requirements and guidance to designers for safety classification of structures, systems and components (SSCs); seismic qualification; environmental qualification; separation of systems and components; fire protection; and containment, radiation, and tornado protection. Appropriate design guides (DGs) are also developed and used in the system design and analysis.

Probabilistic Safety Assessments (PSAs) are conducted in parallel with the design to ensure that accident analysis frequencies meet regulatory acceptance criteria. Quantitative safety goals are defined and the PSAs demonstrate that they are met. The objective safety goal is a summed core damage frequency SCDF < 10^{-6} per reactor operating year, and a large release frequency LRF < 10^{-7} per reactor operating year. These goals are met with comfortable margin.

The ACR-1000 plant design ensures safety during construction, commissioning, start-up, and O&M by assuring the following:

- Ample thermal and safety margins for safe operation.
- Radiation exposure to plant personnel and the public is well below regulatory limits.
- Feedback from construction, commissioning, start-up, O&M experience is incorporated.
- The “defence-in-depth” safety philosophy is applied.
- Physical and functional separation of safety systems.
- Simplified, more reliable systems.
- Safety systems perform their required safety functions during design basis accidents (DBAs) without credit for active mitigation by the process systems.
- Human factors engineering principles and criteria are applied in the design of systems, facilities, equipment, and procedures.
### 4.2 Plant Design Features

#### 4.2.1 Plant Description

Each ACR-1000 unit consists of the nuclear steam plant (NSP) and the balance of plant (BOP). Table 1 provides a summary of the ACR-1000 design features compared to other selected CANDU designs.

The major nuclear systems, which comprise each NSP portion, are located in the RB and Reactor Auxiliary Building (RAB). These systems include, but are not limited to, the following (noting major changes from existing CANDU practice):

- The reactor assembly, consisting of 520 channels in a reduced square lattice pitch, with larger-diameter calandria tubes (CTs) than current CANDU designs, contained within a calandria vessel. Figure 3 presents the ACR-1000 reactor assembly.
- The moderator system with a reduced volume of D$_2$O as compared to CANDU 6 on a per MWe output.
- The HTS with light water coolant operating at higher temperatures and pressures than current CANDU designs, in a two-loop, figure-of-eight configuration with four steam generators, four HTS pumps, four reactor outlet headers, and four reactor inlet headers.
- The fuel handling system, which consists of two fuelling machines, each mounted on a fuelling machine bridge and columns, located at both faces of the reactor to allow for on-line refuelling.
- The main steam supply system, with higher pressure and temperature conditions than the current CANDU designs, for improved turbine cycle efficiency.
- Safety systems, specifically, two shutdown systems, the emergency core cooling (ECC) system, the emergency feedwater (EFW) system (defined as the emergency heat removal system), and the containment system, and associated safety support systems.

The BOP consists of the Turbine Building (TB), the steam turbine, the generator and condenser, the feedwater heating system with associated auxiliaries, and electrical equipment. The BOP also includes the plant cooling and service water systems, water treatment facilities, auxiliary steam facilities, service and breathing air systems, BOP pumphouses and optional cooling towers, main switchyard, and other systems, equipment, and components credited for the plant power production function in the ACR-1000 two-unit plant. The TB does not contain any safety or safety support systems or components.

#### 4.2.2 Fuel Channels and Calandria Assembly

The reactor assembly consists of 520 horizontally-aligned fuel channels arranged in a square pitch. The fuel channels are mounted in a calandria vessel containing the D$_2$O moderator. Each fuel channel assembly consists of a PT, two end fittings, and associated hardware. The PTs contain the LEU fuel and the high-pressure light water coolant. Individual CTs surround each individual PT. The end fittings are out of core extensions of the PT, and extend out of the end shields past the feeder cabinets. The end fittings provide connections to the fuelling machine head for on-line refuelling and to the feeder pipes.

The calandria vessel has end shields located at both ends. They are filled with shielding balls and water to provide shielding. The fuel channels are located by adjustable positioning assemblies on the two end shields and are connected by individual feeder pipes to the HTS. The calandria vessel is enclosed in a concrete vault (reactor vault) filled with light water for shielding. The reactor vault is closed at the top by the reactivity mechanisms deck. Both the
moderator in the calandria vessel and light water in the reactor vault provide additional heat sink capability for beyond design basis accidents (BDBAs).

4.2.3 Fuel

The CANFLEX-ACR fuel (see Figure 4) represents the next evolution in fuel design beyond what is currently used in the Pickering, Bruce, and all of the CANDU 6 reactors. The fuel design is a modified CANFLEX-type fuel bundle similar to that already demonstrated in the CANDU 6 reactor at Point Lepreau. The fuel consists of 42 elements containing uranium dioxide fuel pellets plus a central element containing burnable poison in a zirconia matrix. The uranium dioxide pellets are made with LEU. The fuel element sheaths are made from zirconium alloy. The 43 elements are assembled between end plates to form a fuel bundle. Each of the 520 channels contains 12 bundles. The fuel enrichment of the reference core is 2.4%, and the average fuel burnup is 20,000 MWd/t. A lower fuel enrichment is used for the intermediate core fuel compared to the reference core fuel, during the transition from the fresh core for initial reactor start-up to equilibrium fuelling with the reference fuel.

4.2.4 Reactor Control Units

The reactivity control units (RCUs) are comprised of the in-reactor sensor and actuation portions of reactor regulating and shutdown systems. RCUs include neutron-flux measuring devices (vertical and horizontal flux detector units, ion chamber units, and fission chamber units), reactivity control devices (zone control units and control absorber units), safety shutdown devices (shutdown units and liquid injection shutdown units), and GSS devices. RCUs are designed to be simple, rugged, highly reliable, and require little maintenance.

Flux detectors are provided in and around the core to measure neutron flux, and reactivity control devices are located in the core to control the nuclear reaction.

In-core flux detectors are used to measure the neutron flux in different zones of the core. These are supplemented by fission chamber and ion chamber assemblies mounted in housings on the calandria shell. The signals from the in-core flux detectors are used to adjust the location of the zone control unit assemblies to compensate for changes in power levels and distribution. By varying the absorber position in these assemblies, the local neutron absorption in each zone of the reactor changes, thereby controlling the local neutron flux level.

Control absorber unit elements penetrate the core vertically. These are normally parked out of the reactor core and are inserted to control the neutron flux level at times when a greater rate or amount of reactivity control is required than can be provided by the zone control units.

Slow or long-term reactivity variations are controlled by the addition of a neutron-absorbing poison to the moderator. Control is achieved by varying the concentration of this “neutron-absorbent material” (i.e., gadolinium nitrate and boron) in the moderator. For example, the liquid neutron-absorbent material is used to compensate for the excess reactivity that exists with a full core of fresh fuel at first start-up of the reactor.

Two independent safety-grade reactor shutdown systems are provided. Each shutdown system, acting alone, is designed to shut down the reactor and maintain it in a safe shutdown condition. The safety shutdown systems are independent of the reactor regulating system and are also independent of each other. The first shutdown system, shutdown system no. 1 (SDS1), consists of shut-off units (absorber element, guide assembly, and drive mechanisms), which drop neutron-absorbing elements into the core by gravity and spring assist on receipt of a shutdown signal. A long-term guaranteed shutdown state (GSS) can be achieved by the
reactor regulating system either by addition of poison to the moderator, or by insertion of
dedicated GSS rods into the core. The second shutdown system, Shutdown System 2 (SDS2),
uses injection of a strong neutron-absorbing solution into the moderator. The automatic
shutdown systems respond to both neutronic and process signals.
A dedicated system of GSS rods is provided to allow the reactor to be maintained in a
GSS without the use of moderator poisons during planned and unplanned maintenance
outages. During normal reactor operation, the GSS rods are withdrawn from the core. When
the reactor is in the GSS state, the GSS rods are kept inserted in the core, along with the SDS1
absorber elements.

4.2.5 Heat Transport System

The HTS (see Figure 1) circulates pressurized light water coolant through the reactor
fuel channels to remove heat produced by nuclear fission in the core. The fission heat is
carried by the HTS coolant to the steam generators, to produce steam on the secondary side
that subsequently drives the turbine generator.
The HTS is complemented by auxiliary systems, which support its operation and
maintain parameters within the HTS operating ranges. HTS auxiliary systems are the pressure
and inventory control (P&IC) system, HTS purification system, and HTS pump seal system.
The HTS and its auxiliary systems are similar to those in the CANDU 6 design. However, the overall design of these systems has been improved and optimized, based on
operational feedback from existing CANDU plants.
The major components of the HTS are the 520 reactor fuel channels and associated
feeders, four steam generators, four HTS pumps, four reactor inlet headers, and four reactor
outlet headers configured in two figure-of-eight loops with interconnecting piping. Light
water coolant is fed to the fuel channels from the inlet headers at each end of the reactor and
is returned to the outlet headers at the opposite end of the reactor. Figure 1-1 provides a
simplified illustration of the ACR-1000 nuclear systems.
The principal function of the HTS is to provide reliable cooling of the reactor fuel under
all operating conditions for the life of the plant with minimal maintenance.
The HTS also provides a barrier to radioactive fission products released during NO to
ensure that radiation doses to plant staff remain within acceptable limits. It is designed to
retain its integrity under normal and abnormal operating conditions.
The pressure and inventory of the coolant in the HTS are controlled by the P&IC
system. The long-term cooling (LTC) system is used to remove decay heat following a reactor
shutdown and to cool the HTS to a temperature suitable for maintenance of the heat transport
and auxiliary system components.

4.2.6 Steam Generator Design

Four identical steam generators with integral preheaters transfer heat from the HTS
coolant on the steam generator primary side to raise the temperature of, and boil, feedwater on
the secondary side of the steam generator. The steam generator consists of an inverted vertical
U-tube bundle installed in a shell. Steam-separating equipment is housed in the upper portion
of the shell. A venturi flow restrictor is installed at the outlet nozzle of each steam generator
to reduce the pressure inside the RB containment in the event of a main steam line break
(MSLB).
4.2.7 Heat Transport System Pump Design

The four HTS pumps are vertical, single-stage centrifugal pumps with single suction and double discharge.

When maintenance of the shaft seals or the pump internals is required, the coolant level in the HTS can be lowered to a level below the pumps. The LTC system cools and maintains the HTS after a reactor shutdown to a temperature suitable for maintenance.

A gland seal circuit supplies cooled and filtered water for lubricating and cooling the mechanical seals. A leakage recovery cavity takes the seal leakage to the light water leakage collection system.

Each pump is driven by a vertical, totally enclosed, air-to-water cooled squirrel cage induction motor. The motor has built-in inertia to prolong pump rundown on loss of power.

4.2.8 Moderator System

Neutrons produced by nuclear fission are moderated by the D$_2$O in the calandria. The D$_2$O moderator is circulated by the moderator pumps through the calandria at a relatively low temperature and low pressure, and cooled by the moderator heat exchangers. The moderator heat exchangers remove the nuclear heat generated in the moderator and the heat transferred to the moderator from the fuel channels. Helium is used as a cover gas over the D$_2$O in the calandria. Chemistry control of the moderator water is maintained by the moderator purification system.

The moderator system also acts as a back-up heat sink under certain postulated BDBA conditions. Moderator circulation and the calandria are shown in Figure 5.

4.2.9 Fuel Handling System

The fuel handling system is used to fuel the reactor on demand for the purpose of controlling the reactor power distribution. The fuel handling system stores and handles fuel, from the arrival of new fuel to the storage of spent fuel. The fuel handling system is divided into new fuel handling and storage, fuel changing, and spent fuel handling and storage.

Fuel changing is performed on-power and remotely, using two fuelling machines. One fuelling machine is connected to each end of the fuel channel being fuelled. A two-bundle shift scheme is used for the reference core and a four-bundle shift scheme is used for the transition from the initial core to the reference core. In the two-bundle shift operation, two new fuel bundles are inserted at the inlet end of the channel and two spent fuel bundles are removed from the outlet end of the fuel channel.

Fuel is cooled by the fuel handling system once it is removed from the fuel channel, and it remains in the fuel handling system until it is reinserted in the channel or discharged into the spent fuel bay. The normal refuelling sequence does not require reinsertion.

The spent fuel bundles remain fully submerged in water while being transferred from the fuelling machine in the RB, to the spent fuel reception bay in the RAB. The spent fuel bay has a storage capacity for more than 10 years of accumulated reactor operation plus a full core discharge. Equipment, including a series of tools and accessories, is provided for the physical handling of spent fuel in the spent fuel bay from the manbridge that operates above the bay. Special design provisions and administrative procedures protect against inadvertent criticality due to use of low enriched fuel.

The spent fuel bay cooling and purification system removes the decay heat generated by the fuel, removes the suspended activation products, and controls the water chemistry.
5 CONCLUSIONS

Based on decades of design development and R&D of different CANDU reactor designs in Canada, the ACR-1000 was developed by AECL is a 1200 MWe-class light-water-cooled, heavy-water-moderated pressure-tube reactor, which has evolved from the well-established CANDU® line of reactors. The ACR-1000 design retains the basic, proven, CANDU design features while incorporating innovations and state-of-the-art technologies to ensure fully competitive safety, operation, performance and economics. Improvements include greater operating and safety margins plus adherence and compliance with the latest safety thinking regarding external events and risk assessment.

The ACR-1000 design complies with all applicable Canadian Nuclear Safety Commission (CNSC) regulatory requirements. Although not mandatory in Canada, the ACR-1000 design takes into account all applicable international requirements as appropriate. Moreover, IAEA’s safety standard “Safety of Nuclear Power Plants: Design Requirements”, NS-R-1, has been used in the ACR-1000 design.

AECL has recently issued the ACR-1000 Generic Safety Case Report (GSCR) that provides a site-independent overview of the design, safety characteristics, and bounding safety analysis of the ACR-1000, which demonstrates design readiness and licensability in Canada and abroad.

6 ACKNOWLEDGEMENT

The authors of this paper acknowledge the major contributions from many AECL staff working on the ACR-1000 design and safety analysis, and on the production of the Generic Safety Case Report that was used as a key reference in this paper.

7 REFERENCES


Figure 1: Schematic of the ACR-1000 Heat Transport System
Figure 2: ACR-1000 Quadrant Layout

Figure 3: ACR-1000 Reactor Assembly
Figure 4: ACR-1000 Fuel Assembly

Figure 5: Calandria and Moderator Circulation
EPR SAFETY CONCEPT

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ABSTRACT

The purpose of this presentation is to present the safety concept of the European Pressurized Water Reactor (EPR).
Special emphasis is given on safety strategy, separation of functions and core melt mitigation.

1 INTRODUCTION

For the development of future Nuclear Power Plants (NPPs) there are two fundamental objectives:
- High safety level
- Competitiveness with other power generating plants,

These two objectives may be mutually contradictory because improvement of safety features is normally linked to additional investment cost; however, higher investment can also result in higher availability, and this enhances competitiveness. It requires intensive engineering effort and intelligent technical solutions to harmonize these at first sight contrary requirements. For example, higher availability can be achieved by increasing system redundancy; this in turn allows preventive maintenance during power operation and therefore shorter refuelling outages; also increasing component quality results in fewer component failures and thus avoids shutdowns.

Adopting an evolutionary reactor design was chosen because this is the best way to take advantage of the operating experience gained from existing plants and the research and development studies conducted for them. Having chosen the “evolutionary approach”, operating experience was a significant driving force in actually improving the defence-in-depth of the EPR.

2 TECHNICAL FEATURES

The rated thermal power of the Nuclear Steam Supply System (NSSS) is 4300 MW. The high efficiency of the plant has resulted in an electrical output of about 1600 MW.
The development of the EPR is an evolutionary approach which is based on the experience gained in the construction and operation of the existing plants in France and Germany. The reactor coolant system design, loop configuration and the design of the main components are close to those of existing designs and can therefore be considered as well proven.

The system organization fulfills the principle of enhanced efficiency and reliability as well as the principle of diversification since any safety-grade system function can be backed up by another system or group of systems.

3 SAFETY APPROACH

A twofold strategy was pursued in the EPR safety principles:
- To improve the prevention of accidents and core damage;
- To mitigate severe accident consequences, even if their probability has been further reduced. This is achieved by implementing features to ensure containment integrity. Thus, it can be demonstrated that the need for emergency response measures is restricted to the immediate vicinity of the plant.

The safety approach includes a rugged deterministic basis complemented by probabilistic analyses in order to improve the prevention of accidents, and also their mitigation. Representative scenarios are defined for both core melt prevention and the prevention of large releases in order to provide a design basis for risk reduction features.

Accident and core damage prevention measures are enforced by:
- Enhanced efficiency and reliability of the safety systems
- Elimination of common mode failures by physical separation and diverse backup for safety functions
- Increased grace periods for operator actions by designing components (e.g. pressurizer and steam generators) with larger water inventories to moderate transients
- Reduced sensitivity to human errors by an optimized man-machine interface, digital instrumentation and control systems and information supplied by modern operator information systems
- Low-probability events with multiple failures and coincident occurrences up to the total loss of safety-grade systems are considered beyond the deterministic design basis
- Design provisions for severe accident management are:
  - Reactor coolant system depressurization within the containment in case of total loss of secondary side cooling;
  - Features for corium spreading and cooling, for hydrogen recombination, and for containment heat removal in case of severe accidents.

4 PLOT PLAN

The configuration of the individual power plant buildings is shown in Figure 1.

The main Nuclear Island buildings of the plant unit are:
- Reactor building
- Safeguard buildings
- Fuel building
- Nuclear auxiliary building
Access building
Diesel buildings
Essential service water pump buildings
Vent stack
Radioactive waste building

The main turbine island buildings of the plant unit are:
- Turbine building
- Switchgear building
- Office building
- Switchyard, transformer structures
- Circulating water buildings and structures

Figure 1: Plot plan

The four train redundant safety systems are located in the physically separated safeguard buildings. The following examples show the location of the emergency core cooling system.
Figure 2: Location of safety system in physically separated buildings

The safety-related buildings are designed to withstand natural external hazards (earthquake). Furthermore, buildings containing systems for core and spent fuel cooling are designed for manmade external hazards (explosion pressure wave, airplane crash). The features to mitigate airplane crash consist in either physical protection of the buildings (double walls for reactor building, safeguard buildings 2/3 and fuel building) or separation of redundant subsystems by distance (e.g. diesel generators).
5 EPR SYSTEM CONFIGURATION

5.1 Nuclear steam supply system

The Reactor Building houses the following main components of the Nuclear Steam Supply System (NSSS):
- A low-alloy steel Reactor Pressure Vessel (RPV) with stainless steel inner cladding containing the core;
- Four reactor coolant loops filled with water at a pressure of 15.5 MPa provide heat removal from the core. Each loop consists of a reactor coolant pump, a steam generator and connecting piping;
- The reactor coolant system (RCS) inclusive of its pressurizing system;
- Safety systems and auxiliary systems providing supporting functions.
Figure 4: EPR primary circuit

Table 1: Data of pressurizer and steam generator

<table>
<thead>
<tr>
<th>PRESSURIZER</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Total volume</td>
<td>75 m³</td>
</tr>
<tr>
<td>Number of safety valves</td>
<td>3</td>
</tr>
<tr>
<td>Capacity of each safety valve</td>
<td>300 t/h</td>
</tr>
<tr>
<td>Diverse depressurization valve</td>
<td>900 t/h</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>STEAM GENERATORS (SG)</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Number</td>
<td>4</td>
</tr>
<tr>
<td>Heat transfer surface area per SG</td>
<td>~ 7960 m²</td>
</tr>
<tr>
<td>Tube outer diameter</td>
<td>19.05 mm</td>
</tr>
<tr>
<td>Water mass per SG on secondary side at full load</td>
<td>~ 80.9 t</td>
</tr>
<tr>
<td>Saturation pressure in the tube bundle</td>
<td>7.8 MPa</td>
</tr>
<tr>
<td>Pressure at hot zero power</td>
<td>9.0 MPa</td>
</tr>
</tbody>
</table>
5.2 Main characteristics of the reactor

The fuel assembly structure supports the fuel rod bundle. It consists of bottom and top nozzles plus 24 guide thimbles and 10 spacer grids. The spacer grids are vertically distributed over the fuel assembly structure. Inside the assembly, the fuel rods are vertically arranged in a square lattice with a 17x17 array. 24 positions in the array are occupied by the guide thimbles, which are joined to the spacer grids and to the top and bottom nozzles. The bottom nozzle is equipped with a debris filter that almost completely eliminates debris-related fuel failures.

The fuel rods are composed of a stack of sintered enriched uranium dioxide pellets, with or without burnable absorber (gadolinia), contained in a hermetically sealed cladding tube made of M5™ alloy.

Table 2: Data of core and reactor coolant system

<p>| | |</p>
<table>
<thead>
<tr>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>CORE</strong></td>
<td></td>
</tr>
<tr>
<td>Active height</td>
<td>420 cm</td>
</tr>
<tr>
<td>Number of fuel assemblies</td>
<td>241</td>
</tr>
<tr>
<td>Fuel rod lattice</td>
<td>17 x 17 -24</td>
</tr>
<tr>
<td>Type of fuel assembly</td>
<td>HTP X5</td>
</tr>
<tr>
<td>Average linear heat generation rate</td>
<td>156.1 W/cm</td>
</tr>
<tr>
<td>Number of RCCAs</td>
<td>89</td>
</tr>
<tr>
<td>Core outlet temperature</td>
<td>≈ 330°C</td>
</tr>
<tr>
<td><strong>REACTOR COOLANT SYSTEM</strong></td>
<td></td>
</tr>
<tr>
<td>Operating pressure</td>
<td>15.5 MPa</td>
</tr>
<tr>
<td>Design pressure</td>
<td>17.6 MPa</td>
</tr>
<tr>
<td>RPV inlet temperature</td>
<td>295.9 °C</td>
</tr>
<tr>
<td>RPV outlet temperature</td>
<td>327.2 °C</td>
</tr>
<tr>
<td>Coolant flow per loop</td>
<td>28330 m³/h</td>
</tr>
</tbody>
</table>
5.3 EPR system configuration

The EPR system configuration provides generally a four-fold redundancy of safety systems.

Figure 5: Survey of main fluid systems
5.4 Separation of functions

The principle of separation of functions applied in the EPR results in a system configuration in which the safety related tasks and the operational tasks are separated. Beyond these deterministic considerations for the safety systems, backup functions for the loss of one complete redundant safety system are provided.

Table 3: Safety system backup by diverse functions

<table>
<thead>
<tr>
<th>Safety-grade system</th>
<th>Diverse system functions</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>MHSI</strong> Medium Head Safety Injection System</td>
<td>Fast Depressurization via Secondary Side + Pressurizer Relief Valve + Accumulator Injection System + Low Head Safety Injection System</td>
</tr>
<tr>
<td><strong>LHSI</strong> Low Head Safety Injection System</td>
<td>Medium Head Safety Injection System + For small breaks: Secondary Side Heat Removal System</td>
</tr>
<tr>
<td><strong>FPC</strong> Fuel Pool Cooling System</td>
<td>Fuel Pool Water Heating with subsequent Steaming + Coolant make-up</td>
</tr>
<tr>
<td><strong>EFWS</strong> Emergency Feedwater System + Steam Relief</td>
<td>Primary side Bleed via the pressurizer safety valves + Primary side Feed with MHSI</td>
</tr>
<tr>
<td><strong>Diesels</strong></td>
<td>SBO Diesels</td>
</tr>
<tr>
<td><strong>TLOCC (Total Loss of Cooling Chain)</strong></td>
<td>RPV closed: Secondary Side Heat Removal System + RPV open: LHSI + Steaming Note: LHSI pumps cooling by chilled water</td>
</tr>
</tbody>
</table>

6 MITIGATION OF SEVERE ACCIDENTS

The EPR follows the deterministic approach with the objective of strengthening the design measures in such a way that “practical elimination” of large releases is achieved. This implies technical features to prevent early containment failure by transient events as well as features to ensure long-term integrity of the containment.

Despite their extremely low expected overall frequency, severe accidents and their consequences are taken into account in the design of EPR. In order to rule out severe consequences for the environment and for the population, the design objective is that containment integrity and leak tightness will be maintained during the entire course of potential severe accidents.

Potential processes which could jeopardize containment integrity during severe accidents and result in large fission product releases were identified in various risk studies.
They include:
- Failure of the reactor pressure vessel (RPV) at high pressure caused by the molten core with a risk of corium dispersal and direct containment heating (DCH);
- Energetic interaction of molten fuel and coolant: in-vessel and ex-vessel (MFCI);
- Interaction of molten corium with the basemat with the possibility of basemat melt-through;
- Fast hydrogen deflagration or detonation;
- Long-term pressure and temperature increase in the containment.

6.1 General design criteria

Certain general design criteria are used to design features to cope with the different phenomena to be considered. They are as follows:
- Use of passive components and means in the early phase of the accident appropriate to the plant state in case of severe accident conditions;
- Use of simple and robust designs;
- Use as far as possible of proven materials and technologies

The design criteria used for the specific design features for mitigation of individual phenomena are derived from the analysis of representative scenarios which are specified for the various phenomena and which are selected in a deterministic way.

6.2 Prevention of high pressure core melt sequences

High pressure failure of the RPV is eliminated by deliberate depressurization of the primary system to a pressure below 2 MPa using highly reliable dedicated valves in series which supplement the three pressurizer safety relief valves. Valve opening is manually actuated.

Depressurization also prevents early containment failure due to direct containment heating, and containment bypass due to creep failure of the steam generator tubes. Different scenarios have been analyzed in order to evaluate the pressure and temperature history within the reactor coolant system and the potential time frame and criterion for activation of the depressurization valves. The scenarios cover:
- Total loss of AC power with inoperability for 12 hours of all diesels;
- Total loss of feedwater with inoperability of RCS feed and bleed;
- Small-Break Loss-of-Coolant Accident (SBLOCA) with inoperability of the safety injection system.

Both power and shutdown states are considered. The analyses show that depressurization at a mass flow of 900 t/h before 650°C core outlet temperature is reached enables water injection by the accumulators to delay core melting and the reactor coolant system pressure to be reduced to well below 2 MPa by the time the RPV fails, in effect to around 0.5 MPa for many core melt scenarios. Sufficient time for activation is available and more than 1 h margin exists for latest activation to prevent vessel failure above 2 MPa and to prevent any risk of dispersal of corium debris which could induce direct containment heating.

Vessel support and cavity structures are designed to accommodate the loads resulting from a vessel failure at 2 MPa.

6.3 Hydrogen mitigation

The following design features are relevant for hydrogen control:
The large dry containment; the average hydrogen concentration will be below 10 vol-% in all events;
- Around 50 recombiners distributed mainly over the equipment rooms; these remove most of the hydrogen before RPV failure occurs and significantly enhance mixing of the atmosphere;
- Direct discharge of the reactor coolant system inventory via a relief tank into the lower equipment compartments; this provides a large amount of steam at the time of hydrogen release and improves mixing;
- Convection and rupture foils and mixing dampers that provide passive and failsafe transformation of the two compartment containment into a single compartment following temperature and/or differential pressure increase.

The justification of the hydrogen mitigation concept is based on representative scenarios (mainly SBLOCA scenarios with breaks at different locations) and bounding scenarios with additional aggravation (e.g. with reflood of the hot core at the most penalizing moment) selected to explore the limits of the concept and proceeds as follows:

- Calculation of mass and energy input into the containment, resulting in 500 to 800 kg hydrogen for representative scenarios and up to 1000 kg with a peak rate of 6 kg/s for bounding scenarios;
- Calculation of the gas and temperature distribution for the relevant phase (until mixing) with CFD (Computational Fluid Dynamics) method;
- Assessment of the risk of fast deflagration or DDT (Deflagration to Detonation Transition with experimentally based criteria and the risk from slow deflagration via AICC (Adiabatic Isochoric Complete Combustion) pressure);
- Calculation of the combustion process with a CFD code in case flame acceleration and fast combustion cannot be ruled out under these criteria;
- Assessment of thermal loadings from recombination, short deflagration and long-lasting combustion (“standing flames”).

The most important results are as follows:

- AICC pressure is always below design pressure (0.53 MPa) for representative scenarios and below the ultimate containment pressure, where still leaks are avoided 0.96 MPa for bounding scenarios;
- Flame acceleration occurs locally, mainly in the steam generator compartments, but the flame decelerates as it progresses in the 3 dimensional space within the dome; hence, no significant dynamic loads occur on the shell, and the slow pressure increase is enveloped by the AICC pressure;
- The recombination rate is largely independent of the arrangement of the recombiners.

Temperature loadings due to recombination on the internal walls are benign.

6.4 Core melt mitigation

Features are provided to retain the melt within the containment to prevent penetration of the basemat by corium concrete interaction and to prevent a significant release of fission products, including groundwater contamination, as a consequence of loss of containment integrity at the basemat.
The EPR melt retention concept is based on ex-vessel melt retention. The risk of ex-vessel MFCI during failure is prevented by the provision of a dry reactor cavity and a dry spreading compartment.

The basic concept of EPR for melt stabilization is spreading of the melt into a large lateral compartment; this is followed by flooding, quenching and cooling from the top and bottom with water drained passively from the Incontainment Refueling Water Storage Tank (IRWST). It is a characteristic feature of this concept that the corium is not directly discharged into the spreading compartment as it is released from the RPV but first temporarily retained in the reactor cavity. This feature results in spatial separation of the functions:

1. To withstand the thermal-mechanical loads during RPV failure with only the rugged concrete structures of the reactor cavity being affected; and
2. To transfer the melt to a coolable configuration and stabilize it in an area in which only the structure of the core catcher is affected.

This separation results in a clear definition of loadings on the involved structures (Figure 6) and in defined conditions for spreading and stabilization of the corium.

Figure 6: Reactor cavity section

The connection between reactor cavity and the spreading compartment is normally closed by a plug. This plug will be molten-through by the corium.

The wall of the reactor cavity is covered with a layer of sacrificial concrete which will be eroded by the corium. During the erosion time until the failure of the plug, the corium which has not been discharged during the first pour after RPV failure can accumulate within the reactor cavity. The accumulated melt will then relocate into the adjacent compartment. Spreading will occur under dry conditions.

The bottom and side structures of the core catcher are covered with sacrificial concrete to shield the steel structure from transient thermal loadings during spreading. Erosion of this
layer also lowers the temperature of the melt and the density of the oxidic phase. This in turn creates a layer inversion with oxidic melt located above the metallic melt; this enables significant fragmentation of the oxides on contact with the cooling water from above. The lower, metallic melt will cool down below its solidification temperature, and the corresponding formation of crusts will dampen the initial thermal impact on the cooling structure.

Underneath the sacrificial layer a cooling structure is provided consisting of an array of massive steel blocks which at the bottom form parallel channels of rectangular cross section for cooling. Water from the IRWST for melt cooling will pass through the cooling channels and then flood the melt from above. Thus, the melt will be cooled from above and below. The generated steam will escape into the containment via a specific steam exhaust channel. Heat removal from the corium to the containment atmosphere and the structures can be performed completely and continuously in a passive mode.

However, this passive mode will apply only during the short term of the accident. In the long term, the containment heat removal system can perform active cooling by direct water injection into the cooling structure instead of spray injection into the containment atmosphere. Additional advantages are that the spreading compartment and the reactor cavity can be fully flooded, and the overflowing water which is still subcooled flows back into the IRWST.

6.5 Containment heat removal and leak tightness

Long-term pressurization of the containment is averted by a dedicated active two-train containment heat removal system which does not need to operate earlier than 12 h after accident initiation. The building structures provide sufficient heat capacity for the design pressure not to be exceeded within the uncooled period. The external recirculation cooling loops are located in special ventilated and shielded compartments with provisions for decontamination and repair of the involved components.

The system has two modes of operation:
1. Spray within the containment dome for fast condensation of the steam and pressure reduction;
2. Water recirculation through the cooling structure of the core catcher and the IRWST for long-term prevention of steaming into the containment volume, thus maintaining ambient pressure conditions in the containment and zero leakage from the containment.

Figure 7 shows a schematic diagram of the containment heat removal system indicating the different modes of operation.
Containment heat removal system and hydrogen recombination can reduce the containment pressure to near atmospheric pressure. As a consequence of Finnish regulations containment filtered venting is provided for the EPR in Finland to permit release of non-condensable gases in the long term, if necessary. Venting is performed to finally depressurize the containment and terminate any releases. With the containment and severe accident management systems functioning as designed there is no need for use of the containment filtered venting system. The design pressure of the venting system is 1.1 MPa and the design temperature 200°C.

The double-wall containment is designed to prevent a direct path forming from the inner containment to the environment. Design provisions ensure the limitation of radiological releases:
- Collection of leaks from the mitigation system in the peripheral buildings and filtration;
- Collection of leaks from the penetrations in the annulus before filtration;
- Safe and tight isolation of systems which could form a containment bypass.

7 CONCLUSION

The EPR fulfils the fundamental objective as defined in INSAG-3:
- “To protect individuals, society and the environment by establishing and maintaining in nuclear power plants an effective defence against radiological hazards”.

Figure 7: Scheme of containment heat removal system
This objective lead to design the EPR according to deterministic criteria and verification by PSA.

In addition, probabilistic objectives and targets, associated with levels of radiological consequences, are as follows:

- the integral core melt frequency, considering all plant states and all types of events shall be less than 10^{-5}/r.y.,
- the risk of large releases shall be “practically eliminated”, large releases being defined as releases beyond the acceptable limits in terms of consequences: no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, and no long term restrictions in consumption of food.

The design of the EPR is such that the total core damage frequency 1.3 E-6/a (External hazards, internal hazards) and the total frequency to exceed 100 TBq CS is 7 E-8/a.

The external source terms are limited in a way that stringent countermeasures, such as relocation or evacuation of the population are restricted to the immediate vicinity of the plant and the restrictions of the use of food-stuff are limited to the first year harvest.
European Utility Requirements (EUR) Volume 3 Assessment for AP1000 and EPP Phase 2E Program

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ABSTRACT

The European Passive Pressurized Water Reactor (EPP) Program was initiated in 1994 by several European utilities, Westinghouse and its Industrial Partner ANSALDO Nucleare. The objective of the EPP Program is to develop a PWR Nuclear Island design based on the Westinghouse passive plant technology and ensuring compatibility of the plant design with the European Utilities Requirements (EUR), as well as key European licensing requirements. Since 2001, the EPP reference plant design is the AP1000. The AP1000 is two-loop 1100 MWe pressurizer water reactor (PWR). It uses passive safety systems to provide significant and measurable improvements in plant simplification, safety, reliability, investment protection and plant costs. The Westinghouse AP1000 development program is aimed at making available a nuclear power plant that is economical in the world-wide deregulated electrical power industry in the near term.

In 2004, the EUR organization launched a program for the preparation of an EUR Volume 3 Subset for the AP1000 NPP. The assessment of the AP1000 plant versus the EUR requirements was a key activity in the frame of EPP Phase 2D program and it is an important step for the evaluation of the AP1000 design for application in Europe. EUR results also provided input to the EPP design group that has selected the most significant deviation for performing detailed studies to quantify the degree of compliance and, if needed.

Ansaldo Nucleare is currently involved in the EPP Phase 2E program, started at beginning of 2007. The EPP Phase 2E program is intended to bring the AP1000 design into optimum compliance with the EUR. As a result, it is expected that the AP1000 plant design will either be shown to be adequate to meet the EUR or design changes will be identified to fully meet the EUR Requirements. The proposal also addresses initial licensing steps in the EPP member countries.

The purpose of this paper is twofold:
1. to provide an overview of the EUR program, with particular reference to the EUR Volume 3 Subset for the AP1000 NPP.
2. to provide an overview of the results obtained during the EPP Phase 2E.
1 INTRODUCTION

The purpose and main objective of the EUR is to produce a common set of utility requirements, endorsed by major European utilities for the next generation of LWR and BWR nuclear power plants. The aim of the requirements is to promote the harmonization of Safety, approaches, Targets, Criteria and assessment methods, Standardization of design conditions, Design objectives and criteria for the main systems and equipment, Equipment specifications and standards, Information required for assessment of safety, reliability and cost, thus allowing the development of standard designs that can be built and licensed in several European countries with only minor variations. The benefits of a common set of requirements are:

1. Improvement in the licensing of new nuclear power plants and in their public acceptance:
   - by setting common safety Targets which are consistent with the best European and international objectives,
   - by promoting within Europe common technical responses to safety problems,
   - by setting “good neighborhood” requirements like low targets for accidents and routine radioactive releases into the environment, and consideration of decommissioning aspects at the design stage.

2. Strengthening of nuclear electricity competitiveness:
   - by controlling construction costs and operating costs through standardisation, simplification and optimisation of maintenance at the design stage,
   - by establishing stable conditions for competition between the suppliers on the European Market,
   - by allowing low operation and fuel cycle costs, through flexible and efficient design features that allow easy adaptation to future plant operating and fuel management schemes,
   - by laying down ambitious (but achievable) availability and lifetime Targets.

The EUR document [1] is divided into four volumes. Each volume is divided into chapters that deal with a specific topic. Volumes 1, 2 and 4 provide the Main Policies and Top Tier Requirements, the Generic Nuclear Island Requirements and the Power Generation Plant Requirements for the generic European LWR and BWR nuclear power plants (NPP).

Volume 3 is intended to report the Plant Description, the Compliance Assessment to EUR Volumes 1 and 2, and finally, the Specific Requirements for each specific Nuclear Power Plant Design considered by the EUR.

As the result of the continuous updating program and making use of the experience and feedbacks from the development of Volume 3 detailed assessment programs conducted in the previous years, in April 2001, the EUR organization issued Revision C of the European Utility Requirements.

In 2004-2006 the EUR organization with the help of the European Passive Plant Program (EPP) Team performed a detailed compliance review of the AP1000 PWR design against the EUR Revision C Volume 1 and 2 requirements.

EUR results also provided input to the EPP design group that has selected the most significant deviation for performing detailed studies to quantify the degree of compliance and, if needed, to define possible design changes to fully meet the EUR Requirements. The EUR Volume 3 Subset for the AP1000 NPP has been issued in December 2007 [2].
2 EPP PROGRAM OVERVIEW

The European Passive Pressurized Water Reactor (EPP) Program was initiated in 1994 by several European utilities, Westinghouse and its Industrial Partner ANSALDO Nucleare. The objective of the EPP Program is to develop a PWR Nuclear Island design based on the Westinghouse passive plant (AP600, AP1000, EP1000 and SPWR) technology by ensuring compatibility of the plant design with the EUR as well as key European licensing requirements.

Through the middle of 1999, the EPP Program studied a 1000 MWe three-loop passive plant design, called EP1000. In 1999, in parallel with the EPP Phase 2B activities, Westinghouse, in response to U.S. market conditions, which indicated that new nuclear units must be competitive with natural gas fired generation alternatives, initiated a study to evaluate the feasibility of uprating the AP600 (2-loop passive plant design) to achieve better plant economics. Although the AP600 was the most cost-effective plant ready for deployment, it was more expensive than that needed to compete in the U.S. market. In order to develop a cost competitive nuclear power plant, Westinghouse undertook a program to develop and license a larger version of the AP600 with an increased power output of greater than 1000 MWe (3400 MWt), while maintaining the AP600 design configuration, use of proven components and licensing basis. The plant is called AP1000.

Before the end of EPP Phase 2B, the EPP utilities decided, to merge the EPP program with the U.S. AP1000 program. The new EPP reference plant design therefore became the two-loop AP1000 with minor design modifications for European application. The plant follows very closely the AP1000 U.S. design, that has implemented some of the design features developed during the EPP program, including Low Boron Capabilities, Auxiliary Systems Design, capability to operate with MOX fuel.

Merging the EPP and AP1000 programs provides a more cost effective way to achieve the final objective of the EPP Program which is to develop a 1000 MWe PWR design based on passive technology that meets the EUR and is licensable in Europe.

The AP1000 was granted Final Design Approval in September 2004 and Design Certification from the US NRC on December 30, 2005. Comparison studies of the AP1000 versus the EP1000 have shown that the AP1000 offers distinct advantages, particularly in generating cost and in level of design maturity, while retaining good compliance with EUR and European licensing requirements. The EPP Phase 2C and 2D activities supported refinement of the AP1000 design, with specific studies performed to identify and establish the design modifications needed for Europe.

Westinghouse and Ansaldo are currently involved in the Phase 2E program, started at beginning of 2007. The EPP Phase 2E program is intended to bring the AP1000 design into optimum compliance with the EUR. As a result, it is expected that the AP1000 plant design will either be shown to be adequate to meet the EUR or design changes will be identified to bring the AP1000 into compliance. The proposal also addresses initial licensing steps in the EPP member countries.

The objectives of Phase 2E of the European Passive Plant Program are:

- Continue to support progress in refining the AP1000 base design details for Europe.
- Address EUR non-conformances.
- Address initial licensing steps in the EPP member countries.
- Support the EUR organization to finalize EUR Volume 3 for AP1000.

The EPP program provides the participants with the opportunity to gain passive plant technology knowledge, participate in the final configuration of the European AP1000 design and define and resolve any European licensing issues.
2.1 AP1000 PLANT OVERVIEW

The AP1000 is a two-loops PWR (Figure 1), with a gross electrical power of 1117 MWe [3]. The AP1000 design includes advanced passive safety features and extensive plant simplifications to enhance the safety, construction, operation, and maintenance of the plant. The plant design utilizes proven technology, which builds on over 35 years of operating PWR experience.

On December 30, 2005 the United States Nuclear Regulatory Commission (NRC) approved Design Certification for the AP1000 standard nuclear plant design, making the AP1000 the first Generation III+ plant to receive such certification.

On 24th of July 2007 Westinghouse Electric Co. signed landmark contracts with China's State Nuclear Power Technology Corporation, to provide four AP1000 nuclear power plants in China. The four plants are to be constructed in pairs at the Sanmen (Zhejiang) and Haiyang (Shandong) sites. First concrete is expected to begin March 2009, with the first plant becoming operational in late 2013. The remaining three plants are expected to come on line in 2014 and 2015. In this framework, Ansaldo Nucleare in Joint Venture with Mangiarotti Nuclear, has signed a contract with Westinghouse for the design and the supply of innovative components to be installed in the first AP1000 unit at the Sanmen site.

The AP1000 uses passive safety systems to improve the safety of the plant and to satisfy safety criteria of regulatory authorities. The use of passive safety systems provides superiority over conventional plant designs through significant and measurable improvements in plant simplification, safety, reliability, and investment protection.

The passive safety systems require no operator actions to mitigate design basis accidents. These systems use only natural forces such as gravity, natural circulation, and compressed gas. Safety systems do not use active components (such as pumps, fans or diesel generators) and are designed to function without safety-grade support systems (such as electric power, component cooling water, service water, HVAC). The AP1000 design includes features such as simplified system design to improve operability while reducing the number of components and associated maintenance requirements. The AP1000 has 50%
fewer valves, 83% less piping, 87% less control cable, 35% fewer pumps and 50% less seismic building volume than a conventional plant of similar installed capacity. These reductions in equipment and bulk quantities lead to major savings in plant costs and construction schedules.

The AP1000 passive safety-related systems include, among the others, the Passive Core Cooling System (PXS) and the Passive Containment Cooling System (PCS).

The PXS (Figure 2) protects the plant against reactor coolant system (RCS) leaks and ruptures of various sizes and locations. The PXS provides the safety functions of core residual heat removal, safety injection, and depressurization.

The PXS uses three passive sources of water to maintain core cooling through safety injection. These injection sources include the core makeup tanks (CMTs), the accumulators, and the IRWST. These injection sources are directly connected to two nozzles on the reactor vessel so that no injection flow can be spilled for the main reactor coolant pipe break cases.

Long-term injection water is provided by gravity from the IRWST, which is located in the containment just above the RCS loops. Normally, the IRWST is isolated from the RCS by squib valves. The tank is designed for atmospheric pressure, and therefore, the RCS must be depressurized before injection can occur.

The depressurization of the RCS is automatically controlled to reduce pressure to about 12 psig (0.18 MPa) which allows IRWST injection. The PXS provides for depressurization using the four stages of the ADS to permit a relatively slow, controlled RCS pressure reduction.

The PXS includes a 100% capacity passive residual heat removal heat exchanger (PRHR HX). The PRHR HX is connected through inlet and outlet lines to RCS loop 1. The PRHR HX protects the plant against transients that upset the normal steam generator feedwater and steam systems. The PRHR HX satisfies the safety criteria for loss of feedwater, feedwater line breaks, and steam line breaks.

Figure 2 - AP1000 Passive core cooling system
The IRWST provides the heat sink for the PRHR HX. The IRWST water volume is sufficient to absorb decay heat for more than 1 hour before the water begins to boil. Once boiling starts, steam passes to the containment. This steam condenses on the steel containment vessel and, after collection, drains by gravity back into the IRWST. The PRHR HX and the passive containment cooling system provide indefinite decay heat removal capability with no operator action required.

The PCS (Figure 3) provides the safety-related ultimate heat sink for the plant. As demonstrated by computer analyses and extensive test programs, the PCS effectively cools the containment following an accident such that the pressure is rapidly reduced and the design pressure is not exceeded.

The steel containment vessel provides the heat transfer surface that removes heat from inside the containment and rejects it to the atmosphere. Heat is removed from the containment vessel by continuous natural circulation flow of air. During an accident, the air cooling is supplemented by evaporation of water. The water drains by gravity from a tank located on top of the containment shield building. Calculations have shown the AP1000 to have a significantly reduced large release frequency following a severe accident core damage scenario.

With only the normal PCS air cooling, the containment stays well below the predicted failure pressure for at least 24 hours. Other factors include improved containment isolation and reduced potential for LOCAs outside of containment. This improved containment performance supports the technical basis for simplification of offsite emergency planning.

Figure 3 - AP1000 Passive containment cooling system
3 AP1000 COMPLIANCE WITH EUR

Compliance with the EUR has been a key design objective for the various plants studied and developed under the EPP Program. Assessments have been made, throughout the design process, to define the impact on the Westinghouse passive plant designs in meeting the EUR requirements.

As part of the EPP Phase 2D Program, the EPP Utilities proposed collaboration with the EUR group to prepare an EUR Volume 3 assessment for the AP1000 plant design. Following agreement by the EUR Steering Committee, the AP1000 EUR assessment commenced in January 2004 and was completed in March 2006 with the so called EUR Administration Group Harmonization Meeting.

The EUR Volume 3 AP1000 Subset will include three main chapters:

- Chapter 1 – AP1000 Design Description
- Chapter 2 – Highlights of Results and Conclusions from the Analysis of Compliance
- Chapter 3 – Specific Requirements on the AP1000 Design by EUR

Overall scores for the AP1000 plant are to be considered the most positive in consideration of the reduced number of requirements that were not addressed, as a result of the complete documentation submitted that also included the AP1000 DCD approved by US NRC, and due to the decision taken at the beginning of the program to address the US plant with minor adaptation from EPP program.

In fact, while some of the NPPs addressed by the EUR have been designed taking the EUR as design requirements (e.g., EP1000 plant developed at the end of the '90 by the EPP organisation), the AP1000 plant has been designed following the EPRI URD requirements, integrating some major design feature to make the plant licensable in the largest number of countries around the world.

Nevertheless, EUR requirements have been considered in the development of the AP1000 plant design. In some areas, EUR requirements have even driven specific AP1000 design features (e.g., low boron core). In other areas specialized studies have been performed or are to be performed to specifically evaluate AP1000 compliance with European requirements in those areas that are felt the most important by the European Utilities. Examples include:

- the capability of the plant to accommodate at least 50% MOX fuel,
- a liquid radwaste system that incorporates boron recycle,
- heat removal systems designs (e.g., RHR, CCW, SWS) that can accommodate the EUR rapid cooldown requirements,
- assessment of the design to comply with the EUR DBA and DEC dose targets,
- the capability of the plant to mitigate European specific design basis accidents (e.g., fast boron dilution, multiple steam generator tube ruptures).

4 EPP PHASE 2E OVERVIEW

Westinghouse and Ansaldo, in cooperation with two sponsor utilities (EdF and Swissnuclear) and with the thorough review of the AP1000 EUR Coordination Group (composed of the two aforementioned utilities and Tractebel, Iberdrola, TVO) completed the compliance analysis of the AP1000 Plant versus the EUR requirements. The dedicated EUR Volume 3 subset for AP1000 is scheduled to be issued at beginning of 2007. This represents the conclusion of Phase 2D of the EPP Program.

Some non-conformances found during this process have already been dealt with and other ones will have to be analyzed and solved in order to reach an AP1000 configuration
suitable for construction in Europe while maintaining a close contact with the development and eventual construction of the AP1000 in the USA.

The new Phase of the EPP program, called Phase 2E started at the beginning of 2007. The program is supported by five European Utilities, namely EdF, Eon, RWE, SwissNuclear and Tractebel.

Based on direction provided by the Utilities members and by the EPP Steering Committee, Phase 2E activities are devoted to address the EUR non-conformances and the initial licensing steps in the EPP member countries.

In the following sections some major EPP Phase 2 activities are described.

4.1 Decommissioning Study

The purpose of this activity is to provide a preliminary Decommissioning Study for the AP1000 plant; to identify features which shall be considered at the design stage, as they may become relevant to the future decommissioning work.

The overall objective of the activity is to give evidence of the fact that the dismantling and decommissioning of the AP1000 reactor is feasible and, with respect to the operating reactor, the decommissioning costs and overall impact on the environment and personnel are minimized.

The activities to be performed include the following:

- Task 1: Outline of Decommissioning Objectives and Strategies
- Task 2: Evaluation of AP1000 Wastes and Overview of Waste Treatment
- Task 3: Review of Plant Layout Configuration for Decommissioning

In the first task an overview of the European strategies and policies for the decommissioning of nuclear installations has been provided. In particular, the regulating and licensing requirements in the various European countries have been analysed, as well as the decommissioning objectives for specific sites (green field, research centres, other plant locations), the approach relating to radiological protection to minimise doses to the workers and the public and other specific factors that may influence the selection of a decommissioning strategy for the AP1000 plant.

On the basis of the previous study, the identification and analysis of the factors influencing the selection of strategies for the decommissioning of nuclear installations in the European Union (EU) Member States has been carried out.

In addition to safety and the availability of practical decommissioning techniques, the following issues were identified to be particularly relevant to the selection of a decommissioning strategy:

- The basic decommissioning options; the scope of the decommissioning activities.
- The reactor type; the reactor size; the number of units on a site; the operational history.
- Project planning; analysis of material flow.
- Regulatory and policy requirements (timing; release criteria).
- Socio-economic issues.
- Waste management provisions.

A summary report has been prepared that defines the main decisions and plant specific parameters that may influence the decommissioning decisions and strategies and the overall decommissioning costs.

Task 2 objective is to provide an evaluation of AP1000 decommissioning materials/volumes, an overview of the wastes categorization and waste treatment.
The AP1000 passive plant provides a distinctive and measurable advantage with respect to active plants. In fact, the use of passive safety systems limits the use of support systems and permits an overall simplification of the plant auxiliary systems, see Figure 4. The use of passive systems and the resulting RCS compact design, also result in a major reduction of NI volumes and masses.

![Figure 4: Reduction Bulk Quantities and Equipment Number](image)

The evaluation of the AP1000 plant volumes is performed making a large use of data from the 3D PDS plant model. This tool has also been used to provide examples of dismantling of critical areas using the so called 4D Model that links the 3D geometrical model to a project tool (e.g., Primavera).

The evaluation (decommissioning material and radiological inventory) will be the basis for future decommissioning cost estimate.

Task 3 activities have focused the plant layout features that may impact the decommissioning activities.

In lessons learnt from actual decommissioning operations, areas have been identified and options formulated that should be considered when designing new facilities, the objective being to reduce worker exposure, to minimise waste generation and to simplify dismantling procedures, which will also result in cost saving.

Meeting these objectives requires that structures and equipment be designed in such a way that:

- Activation of materials is limited as much as possible;
- Contamination of plant and equipment can be avoided as much as possible;
- Contaminated or active areas can be easily separated from non-contaminated areas;
- Adequate space and access points are provided to allow the use of special tools and equipment for remote operation and handling, and also to allow the installation of appropriate shielding;
- Plant and equipment items can be easily dismantled, handled and transported, and that adequate openings are provided to allow for easy removal of components and materials from active areas;
- Equipment and buildings can be easily decontaminated;
AP1000 design has been analyzed according to the above lesson learned and major applicable requirements (e.g., EUR).

The overall results of the activities can be summarized as follow:

- The AP1000 plant can be safely dismantled and decommissioned;
- The AP1000 simplified plant design minimize decommissioning waste;
- Activation of materials is limited both by using stringent materials specification (e.g., limit of cobalt in primary components) and proper design of neutron shielding; this minimizes both operation and decommissioning doses;
- Separation between radioactive and non-radioactive equipment and areas is enforced
- Special features are implemented to facilitate decontamination of equipment and buildings (e.g., Structural modules);
- Adequate space, staging area and access are provided for easy dismantling and transportation of equipment

4.2 Aircraft Crash Study

The EUR states that “protection against aircraft crash shall be based on probabilistic approach unless the authorities require a deterministic approach”, with the following comment: “In many European countries only a probabilistic demonstration is requested for aircraft crash. In other countries the demonstration must be based on a deterministic approach i.e. against a regulatory loading function and associated criteria.”(EUR 2.1.5.3.4)
The purpose of this study was to define and evaluate enhancements to the AP1000 plant design to protect against an aircraft crash (ACC) with the goal of minimizing impacts on the existing plant layout and preventing perforation of the steel containment vessel (SCV). As a result of the studies, the AP1000 standard plant design includes provisions to address ACC scenarios. The external event of a terrorist initiated aircraft impact is considered for the AP1000 according to recently adopted U.S. NRC requirements. The plant resilience is demonstrated as a beyond Design Basis issue. The AP1000 ACC methodology was reviewed and found to be acceptable by an industry expert group which included several European Utility members. A technical report [4] describing the ACC evaluations has been submitted for review by the US NRC as part of their ongoing review of AP1000 for US utility combined license (COL) applications.

The AP1000 ACC assessment performed under the EPP Program considered both military and commercial aircraft impact. The ACC evaluations demonstrate that the AP1000 plant is able to provide adequate protection of the public health and safety with respect to aircraft impact. The aircraft impact would not inhibit AP1000’s core cooling capability, containment integrity, spent fuel pool integrity, or adequate spent fuel cooling.

In this frame, Ansaldo has developed a new concept for the shield building roof and the air inlet opening (Fig. 5).

Figure 5: Redesign of Shield Building Roof

4.3 Secondary Containment Study

Per the EUR (Vol. 1, Appendix B), secondary containment is the final fission product confinement envelope which surrounds the following:

- Primary Containment
- Primary Containment penetrations and isolation valves
- Part of systems and components, connected with the reactor pressure boundary or the Primary Containment atmosphere, which may transport highly contaminated fluids outside Primary Containment.

The AP1000 primary steel containment vessel is characterized by a design leak rate of 0.1% vol/day at design pressure of 59 psig.

According to EUR “The optimal combination of the performance of the Primary Containment and the extent and the performance of the Secondary Containment, may be
different for each type of confinement. In particular the requested extent of the Secondary Containment will appear as a consequence of the performance specified for the Primary Containment.

Based on the above, AP1000 Policy, for the definition of the secondary containment boundary and performance is an optimization process that takes into account several parameters (for instance the ranges of leakrate of the Primary Containment, the range of efficiency of the Secondary Containment and its bypass rate). In this process Primary and Secondary containments are not considered individually but as a whole.

The first task of the activity consisted in the evaluation of the leak rate and its partition between the various leak paths.

AP1000 containment isolation is significantly improved by means of a large reduction in the number of penetrations (40 vs. 93 of conventional PWRs) and the number of normally open penetrations (11 vs. 38 of conventional PWRs)

The results demonstrate that the current design of the primary containment meets with margins both the leak rate design value and the 10 CFR50 Appendix J requirements.

On the basis of the above evaluation and of the requirements set in EUR a Secondary Containment boundary has been established.

In addition, following a realistic approach, leak rates have been neglected from closed systems, filled with water in post accident conditions (e.g., Component Cooling System, Fuel Transfer Tube) or equipped with filtering system. The above process allowed to reduce and simplify the rooms to be included in the SC boundary.

The secondary containment has been defined as the annulus between the shield building and containment vessel below the operating deck, the containment isolation valve penetration area, the Spent Fuel System penetration area, the RNS rooms located in the A/B and the staging areas of the equipment hatches.

The calculated Secondary Containment bypass (i.e. the sum of the leakages from the Primary Containment boundary released outside the Secondary Containment) is acceptable (largely below the EUR value of 10 %).

The SC efficiency, in terms of % of SC volume exchanged with the environment, can be assured without active support systems, but:

- Doors equipped with interlock/alarms system and appropriate gasket;
- Isolation dampers on ventilation ducts and penetrations in order to guarantee proper isolation in case of accident requiring Secondary Containment isolation.

Finally, Table 1 reports the results of the release evaluation following DBA according to the EUR methodology described in EUR Chapter 2.1.

It can be noted that the release requirements set by EUR for DBA, except for Iodines, can be meet without taking credit for Secondary Containment.

Table 1 – Releases evaluation with and without Secondary Containment following a DBA.

<table>
<thead>
<tr>
<th>Design Target</th>
<th>EUR limit (TBq)</th>
<th>With secondary containment</th>
<th>Without secondary containment</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>External release (TBq)</td>
<td>Compliance margin (%)</td>
</tr>
<tr>
<td>No action beyond 800 m (Cat. 4)</td>
<td>5 E-03</td>
<td>5.02 E-04</td>
<td>90</td>
</tr>
<tr>
<td>Limited economic impact for I-131</td>
<td>10</td>
<td>5.55</td>
<td>44.5</td>
</tr>
<tr>
<td>Limited economic impact for Cs-137</td>
<td>1.5</td>
<td>0.3384</td>
<td>77.5</td>
</tr>
</tbody>
</table>
4.4 Core Design Study with 50% MOx Loading

The purpose of this study was to demonstrate the ability of the AP1000 reactor to meet the EUR MOx requirements without significant changes to the AP1000 plant design.

An important reactor physics consideration when designing a core with a combination of MOx and UO2 fuel is the impact of the differing neutron spectrum of the adjacent assemblies on the calculation of basic nuclear data for core modeling. Traditional modeling of an infinite lattice of a single assembly type in a higher order code is not sufficient for generating cross-sections and pin power reconstruction data for mixed MOx / UO2 cores. Instead, nodal code cross-section and pin power reconstruction data are calculated by modeling representative “mini-cores” of MOx and UO2 in the higher order lattice code.

This interaction between different assembly types also impacts the intra-assembly power distribution. Power peaking within the MOx assemblies can be controlled by selective enrichment zoning of the rods within the lattice, with the lowest enrichments on the assembly periphery to counteract the thermal flux current from the adjacent UO2 assemblies.

A key concern with MOx from a fuel performance perspective is the high fuel temperatures relative to UO2 fuel resulting in an increase in fission gas release. This issue with MOx fuel can result in a lower fuel burnup limit with respect to the no clad lift-off rod internal pressure limit, potentially limiting the number of cycles that MOx fuel can operate. The approach taken to mitigate this effect is to design the MOx fuel rod with annular pellets. The annular fuel decreases the peak pellet temperature which reduces fission gas release from the pellet, and also provides an increase in the fuel rod internal volume. Such a design can permit MOx fuel assemblies to operate for four annual cycles up to maximum rod burnups consistent with high burnup UO2 fuel.

A typical equilibrium annual cycle loading pattern with a 50% MOx fuel loading is shown in Figure 6. The feed region of this core design consists of 24 UO2 fuel assemblies containing various numbers and loadings of gadolinium rods for peaking factor control, and 24 MOx assemblies without any burnable absorbers. This core loading yields an annual cycle energy output of 338 EFPD at 3400 MWt in an AP1000 reactor.

The MOx fuel assemblies in this design consist of full length annular fuel rods with an annulus diameter of approximately 4 mm. This fuel rod design permits the MOx assemblies to operate for four annual cycles while maintaining the fuel rod internal pressure below the reactor coolant system operating pressure (≤15.5 MPa). This design, however, reduces the heavy metal loading of the MOx assemblies by approximately 23% relative to a solid rod design. As a result, a total of 48 fresh fuel assemblies are loaded each cycle to maintain the MOx lead rod burnups within the range of high burnup UO2 fuel experience. This is in comparison to a 100% UO2 core, which would feed 40 fresh assemblies for the same cycle energy output.

The peaking factor behavior of the core loading shown in Figure 1 is very well behaved over the duration of the cycle, with the cycle maximum integrated rod power being very similar to that of a 100% UO2 core as shown in Table 1.

The impact of the harder neutron spectrum on the 50% MOx core is illustrated by the critical boron concentration and moderator temperature coefficient (MTC) differences listed in Table 2. From Table 2, it can be seen that the MOx core design results in beginning of cycle critical boron concentrations 300 to 400 ppm greater than those in the 100% UO2 core design. However, even though the 50% MOx design has much higher BOC critical boron concentrations, the corresponding MTC is approximately 1.7 pcm/°C more negative than the
MTC for the 100% UO2 core. In a 100% UO2 core, this amount of boron concentration increase would typically increase the MTC by approximately 2 pcm/°C. Therefore, the neutron spectrum effect is quite significant on the BOC MTC.

Another impact of the harder neutron spectrum is a reduction in the control rod worth and available shutdown margin in the 50% MOx core as shown in Table 2. While the 50% MOx core has less available shutdown margin than the 100% UO2 core, the available shutdown margin still exceeds the 1.6% $\Delta\rho$ assumed in the AP1000 design bases.

The study illustrates that the AP1000 reactor is capable of operating with a core design consisting of both UO2 and MOx feed fuel, meeting the EUR requirements for up to 50% MOx feed regions. The MOx assembly designs are capable of operating for four annual cycles while still meeting the no clad lift-off fuel rod design criteria. Peaking factor margins of the 50% MOx core are consistent with those from a 100% UO2 design. While some excess shutdown margin is lost in the 50% MOx design, the available shutdown margin still exceeds the design requirements with comfortable margins without having to change the design of the AP1000 control rods.

![Typical Loading Pattern for Annual Equilibrium Cycle with 50% MOx Loading](image)

Figure 6 Typical Loading Pattern for Annual Equilibrium Cycle with 50% MOx Loading

<table>
<thead>
<tr>
<th>Parameter</th>
<th>100% UO2</th>
<th>50% MOx</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cycle Max. Integrated Rod Power, FΔH</td>
<td>1.494</td>
<td>1.48</td>
</tr>
<tr>
<td>BOC, HFP, ARO, EQXE CB (ppm)</td>
<td>1133</td>
<td>1412</td>
</tr>
<tr>
<td>BOC, HZP, ARO, No Xe CB (ppm)</td>
<td>1789</td>
<td>2187</td>
</tr>
<tr>
<td>BOC, HZP, ARO, No Xe MTC (pcm/°C)</td>
<td>-7.0</td>
<td>-8.7</td>
</tr>
<tr>
<td>EOC Shutdown Margin (%$\Delta\rho$)</td>
<td>3.18</td>
<td>2.52</td>
</tr>
</tbody>
</table>

Table 2 Summary of Key Physics Parameters for 100% UO2 and 50% MOx Annual Equilibrium Cycles
5 CONCLUSIONS

The AP1000 program activities performed under the EPP Program further confirm the potential capability of passive PWR technology in meeting the safety standards established by the EUR while keeping a cost competitiveness. The EUR requirements are being considered in the development of the AP1000 plant design. In some areas, EUR requirements have driven specific AP1000 design features.

The EPP Program and AP1000 EUR compliance assessment are valuable elements for sustaining a long-term positive view of nuclear power contributions to Europe's energy supply mix. It has become increasingly clear that nuclear power generation additions are most competitive with other energy choices when a standard plant design can be applied in multiple locations.

EPP Phase 2E activities further established that the AP1000-Europe plant design complies with the latest European Utility Requirements while retaining the standard plant economic advantage of being largely the same AP1000 plant design as for the U.S.

ACKNOWLEDGMENTS

The authors thank the European utilities currently participating in the EPP Phase 2E program, SwissNuclear (Switzerland), Electricité de France (France), Suez-Tractebel S.A. (Belgium), E.ON Kernkraft GmbH (Germany) and RWE Power AG (Germany), along with Westinghouse and Ansaldo Nucleare. Special thanks to Kathy Demetri (Westinghouse) for her precious contribution.

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DEVELOPMENT OF NATIONAL INFRASTRUCTURE FOR SAFE AND RELIABLE NUCLEAR POWER

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ABSTRACT

The development of an appropriate national infrastructure is an essential task for countries interested in introducing nuclear power or expanding the nuclear power programme. The infrastructure includes both software (legal, regulatory, training, etc) and hardware (grid, facilities, etc) that fulfil the conditions necessary for the safe, economic and reliable nuclear power plant (NPP) operation. Countries need to consider the whole spectrum of (safety, security, environment, finance, industrial etc) issues and the commitments and resources involved over the lifecycle of the programme.

The activities to prepare the infrastructure can be split into three progressive phases of development. The duration of these phases will depend upon the degree of commitment and resources applied in the country. The IAEA published a guide outlining 19 infrastructure issues that are expected to be addressed in each phase. The completion of the infrastructure conditions of each of these phases is marked by a specific milestone at which the progress and success of the development effort can be evaluated and a national decision made to progress to the next phase:

Milestone 1: Ready to make a knowledgeable commitment to a nuclear programme.
Milestone 2: Ready to invite bids for the first NPP.
Milestone 3: Ready to commission and operate the first NPP.

The initial development of the nuclear infrastructure can be facilitated by the establishment of an interdisciplinary team assigned to develop policies and make recommendations. This study group is identified as a Nuclear Energy Programme Implementing Organization (NEPIO). The role of the NEPIO particularly in the achievement of Milestone 1, addressing Government commitment and authorities, is explained. Also the responsibilities and functions of NEPIO, its composition including Government agencies, industry, and other stakeholders, and the competencies and resources necessary to complete the tasks of the NEPIO, are briefly described.

Many countries have some, if limited, infrastructure to support a nuclear power programme. To understand where a country is in its development, a methodology for the evaluation of the national nuclear infrastructure development status was elaborated based on the milestone approach. The envisaged guidance can be used either for the country self-evaluation or for an external evaluation. This will allow countries to evaluate or ask others to help them evaluate the level of their present readiness to introduce nuclear power and to determine those issues in which they need to make additional commitments.
1 INTRODUCTION

Many parts of the world have a pressing need for sustainable development from reducing poverty and raising living standards to improving health care, and industrial and agricultural productivity. Nearly every aspect of development requires reliable access to energy sources. Nuclear power can play a role in providing improved access to affordable energy. Careful planning in the early stages of a programme across a wide range of national infrastructure issues can help instil confidence in the country’s ability to legislate, regulate, construct and safely and securely operate a nuclear power plant (NPP).

Many countries have approached the IAEA regarding support for the introduction of nuclear power. Under its Statute, the IAEA is authorized to assist any Member State that is considering or has decided to “go nuclear” to meet its energy needs. The Agency has considerable experience in doing this through its assistance programmes. The IAEA has recently prepared a number of guidance publications on infrastructure development for countries planning to launch a nuclear power programme consisting of the construction of one or more NPPs and all or any of the necessary supporting facilities and stands ready to provide expert assistance in this area if requested.

1.1 Development of national infrastructure

The IAEA publication “Milestones in the development of a National Infrastructure for Nuclear Power” [1] describes 19 infrastructure issues to support the infrastructure development. Early attention to all of these issues will facilitate the efficient development of a successful national nuclear power programme. Equally, lack of appropriate attention to any of the issues is likely to result in future difficulties that may significantly delay or otherwise affect the successful introduction of nuclear power.

The national infrastructure covers a wide range of issues, from the physical facilities and equipment associated with the distribution of electricity, the transport of the material and supplies to the site, the site itself, and the facilities for handling the radioactive waste material, to the legal and regulatory framework within which all of the necessary activities are carried out, and the human and financial resources necessary to implement the required activities. In short, infrastructure issues, as discussed in this paper, include all activities and arrangements needed to set up and operate a nuclear power programme.

The decision by a State to embark on a nuclear power programme should be based upon a commitment to use nuclear power for peaceful purposes, in a safe and secure manner within a stable political, economic and social environment. A viable NPP project and nuclear power programme will require the establishment of a sustainable national infrastructure that provides governmental, legal, regulatory, managerial, technological, human and industrial support for the nuclear power programme throughout its life cycle. This infrastructure is relevant whether the nuclear power programme is planned for the production of electricity, for seawater desalination or for any other peaceful purpose.

The development of a nuclear power programme entails sustained attention to many interrelated activities over a long duration and involves a commitment of at least 100 years throughout NPP operation, decommissioning and waste disposal. As with any major project, the commitment of resources for the NPP project needs to be phased and decisions to move to subsequent phases, where the commitment of resources will increase significantly, need to be made with a full understanding of the requirements, risks and benefits. Many of these issues are at all times the responsibility of the State authorities, others may also be State controlled. However, in some countries the responsible organisation for the construction and operation of a NPP may be in private or partially private ownership. Irrespective of the ownership, the
utility/ operator will need to establish the necessary infrastructure to manage and control the NPP, including the establishment of human resources, and the long term arrangements. The infrastructure issues are presented as a coherent whole, recognising that eventually many different organisations may have the direct responsibility for developing and implementing the infrastructure.

Experience has shown that the time frame from the initial policy decision by the State to the operation of the first NPP may well be 10–15 years. For a country with a little-developed technical base the implementation of the first NPP would be expected to take closer to 15 years. A country with a strong technical base could take 10 years if it makes a significant and concerted effort to implement a programme. Even countries with existing nuclear power programmes may take about 10 years to approve and construct a new NPP.

1.2 Fundamental Importance of Safety, Security and Safeguards

The fundamental nuclear safety objective is to protect people and the environment from the harmful effects of ionizing radiation. A comprehensive safety culture needs to be developed that permeates all infrastructure development activities. The IAEA publication “Fundamental Safety Principles” [2] contains ten safety principles that represent the international consensus on the high level of safety required for the sustainable use of nuclear power. The first principle establishes that the ultimate responsibility for safety must rest with the operator. It is incumbent on the leadership and management of the State and the operator to develop awareness, encouragement and enforcement of a safety culture throughout the entire programme. It cannot be overemphasized that everyone involved in a NPP project carries a responsibility for safety.

In addition to nuclear safety, and no less significant, are the issues associated with the control of nuclear material, either to ensure the security of the material, or to ensure that all of the activities in a State can be demonstrated to ensure that there is no risk of proliferation of nuclear weapons and that all the materials are adequately accounted for and protected. This also requires the development of a culture, system and practices that ensure that all staff are aware of their responsibilities and the importance of their actions.

2 INFRASTRUCTURE PHASES AND MILESTONES

2.1 Overview

The IAEA publication “Milestones in the Development of a National Infrastructure for Nuclear Power” [1] makes the distinction between a nuclear power programme and a NPP project. A nuclear power programme encompasses all of the elements of a national infrastructure that would support the first NPP as well as any planned expansion. It would include consideration of strategic decisions such as human resource development, industrial support and fuel cycle planning. The nuclear power programme is thus the responsibility of the Government itself. The NPP project is defined as being related to the plant itself, and is likely to be managed by the owner-operator. Before a NPP project can proceed, a nuclear power programme must establish the infrastructure to support that NPP project during its planning, construction, operation and eventual decommissioning.

The milestones approach for implementation of the national infrastructure is depicted in Figure 1 taken from [1] that shows the activities split into three progressive phases.

The duration of these phases will depend upon the degree of commitment and resources applied in the State. The duration, especially the bidding process and construction phases, may also be influenced by national requirements, such as public consultation during the
licensing process. The term “infrastructure milestone” refers to the point at which it can be demonstrated that the preceding phase has been successfully completed and that the State is fully prepared to embark on the subsequent phase. The “infrastructure milestone” is thus a set of conditions and does not necessarily have specific time based implications.

Figure 1: Nuclear infrastructure development programme

For each milestone, 19 issues that need to be considered are shown schematically in Table 1. The order of the issues does not indicate an importance or hierarchy. Each issue is important and requires careful consideration.

The development of the infrastructure necessary to support a nuclear power programme would be expected to proceed through Phases 1–3, leading to the achievement of the corresponding milestones, while at the same time many other specific activities are progressing in order to ensure implementation of the first NPP project. The three programme phases of development are:

—Phase 1: Considerations before a decision to launch a nuclear power programme is taken;
—Phase 2: Preparatory work for the construction of a NPP after a policy decision has been taken;
—Phase 3: Activities to implement a first NPP.

The completion of the infrastructure conditions of each of these phases is marked by a specific milestone at which the progress and success of the development effort can be assessed and a decision made to move on to the next phase. These milestones are:

—Milestone 1: Ready to make a knowledgeable commitment to a nuclear power programme;
—Milestone 2: Ready to invite bids for the first NPP;
—Milestone 3: Ready to commission and operate the first NPP.
Table 1: Nuclear infrastructure issues and milestones

<table>
<thead>
<tr>
<th>ISSUES</th>
<th>MILESTONE 1</th>
<th>MILESTONE 2</th>
<th>MILESTONE 3</th>
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<td>National position</td>
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<td>Nuclear safety</td>
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<td>Management</td>
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<td>Funding and financing</td>
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<td>Legislative framework</td>
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<td>Safeguards</td>
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<td>Electrical grid</td>
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<td>Human resources development</td>
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<td>Stakeholder involvement</td>
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<td>Site and supporting facilities</td>
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<td>Environmental protection</td>
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<td>Emergency planning</td>
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<td>Security and physical protection</td>
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<td>Nuclear fuel cycle</td>
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<td>Industrial involvement</td>
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2.2 **Conditions to reach Milestone 1**

This is the point at which the State would be in a position to make an informed decision on whether it is appropriate to introduce a nuclear power programme. In order to achieve this milestone the State will not only have assessed that it needs additional energy and included nuclear power as a possible option to meet some of these needs, but will also have carried out the first phase of the programme, which would culminate in the attainment of Milestone 1.

The initial phase in the development of a nuclear power programme involves the considerations and planning before a firm decision to develop a nuclear power programme is taken. During this phase the responsible organizations are the Government and the government formed multi-disciplinary group charged with the initial investigation and promotion of nuclear power programme, here referred to as the “Nuclear Energy Programme Implementing Organization” (NEPIO). The NEPIO should be appropriately staffed and resourced and be given the authority to carry out its work. It should develop for the State a complete understanding of the commitments associated with the use of nuclear power. Their report at the end of Phase 1 should clearly show an understanding of the infrastructure that needs to be developed and demonstrate viable plans for its introduction, identifying resource requirements and timescales. It should include plans for the development of organizations to
undertake the role of regulator, owner, operator and technical support. It is also essential even at the earliest phases that recognition of an appropriate safety culture is developed in each organization and their responsibilities for ongoing safe operation.

2.3 Conditions to reach Milestone 2

At this point the infrastructure in the State would be sufficiently developed so that the responsible organizations, State owned or private, would be in a position where inviting formal bids for the first NPP would be possible with confidence that all national and international infrastructure issues have been resolved.

Following the policy decision to proceed with the development of a nuclear power programme, substantive work for achieving the necessary level of technical and institutional competence will have been undertaken. This phase requires a significant and continuing commitment from the Government, which continues to have a role as an advocate and guiding organization for the State’s programme. The necessary legal framework will be in place and separate regulator and operator organizations will have been established.

The State will have carried out the work required to prepare for the necessary infrastructure for construction of a NPP. The regulatory body will need to be developed to a level at which it can fulfill all of its oversight duties. The necessary infrastructure should be developed to the point of complete readiness to request a bid or enter into a commercial contract. The owner/operator (or utility), which may or may not be State owned, has a key role at this time, ensuring that it has developed the competence to manage a NPP project, to achieve the level of organizational and operational culture necessary to meet regulatory requirements, and the ability to demonstrate that it is an adequately informed and effective customer.

2.4 Conditions to reach Milestone 3

At this point the State will be in a position to commission and operate the first NPP. The completed work programme will have brought the Government to the point of having established a nuclear power programme that will bring the benefits of energy security and economic development envisioned in the initial policy decision. The owner/operator will have developed from an organization capable of ordering a NPP to an organization capable of accepting the responsibility for commissioning and operating one. This will require significant development and training for all levels of staff, and the demonstration that the owner/operator can manage the NPP project throughout its life.

While achieving the third milestone is a major accomplishment, it should be remembered that it is only the beginning of a lasting commitment to the safe, secure and effective application of nuclear power. Confidence to move beyond Phase 3 will be enhanced by the use of existing international assessment and review methodologies.

3 RESPONSIBILITIES AND COMPETENCIES OF NEPIO

3.1 Overview

A State generally needs information to make a knowledgeable decision about a major policy initiative such as the launch of a nuclear power programme. For convenience, the milestones publication [1] labels the organization charged with developing such information NEPIO. The important aspects of the NEPIO are that it be given the appropriate authority and resources to prepare the material necessary for a State decision and that it be able to work
closely with all of the relevant stakeholders internal and external to the Government. As a pre-decisional body, its leadership by a trusted and respected person is vital to the credibility of the effort.

The initial purpose of the NEPIO is to compile the information necessary for a knowledgeable policy decision to proceed with the development of a nuclear power programme so that this decision can be made with full realization of all that it entails. During Phase 1, the NEPIO researches, studies and makes policy and strategy recommendations for the Government with respect to the 19 infrastructure issues called out in [1]. While the resolutions of all 19 issues do not have to be finalized by Milestone 1, their implications and the approach to their resolution must be considered. It is the responsibility of the NEPIO to see that this is done. If the decision is made to proceed with a nuclear power programme, specific areas of responsibility may eventually migrate from the NEPIO to other organizations such as the regulatory body and the owner/operator.

During Phase 2, the NEPIO may become the champion and advocate for seeing that the policies and strategies are turned into firm action plans for each of the 19 issues. As it changes roles from policy-formation to coordination, it may also see that the corresponding responsibilities are assigned to those institutions which will become a permanent part of the overall programme infrastructure. As these organizations assume their responsibilities, the NEPIO should assume an oversight roll to assure that the overall programme is proceeding as envisioned. The NEPIO’s functions are most important to the long-term success of the nuclear programme in Phase 1 and 2. In an advocacy role, it may continue in Phase 3, although by then other institutions may take over this role.

3.2 Description of the NEPIO

The purpose of the NEPIO is to lead the national effort to come to a firm decision with respect to the introduction of nuclear power. It should be a specifically assigned body with clear leadership. Appropriate authority needs to be provided to develop the staff and to include other government organizations as may be necessary. The establishment of the NEPIO should be at a senior government level so that this authority is understood by all. It should have available to it the appropriate expertise to address all of the infrastructure issues identified in [1]. Some of this expertise may be temporarily seconded from other Government organizations, or filled by consultants.

During Phase 2 of the development effort, the NEPIO should oversee the transition of infrastructure development to specific, permanent organizations which will carry on the long term responsibilities for regulating, constructing and operating a fully functional nuclear energy programme. The staffing of the NEPIO may have to be adjusted as the State proceeds through Phase 2 to the achievement of Milestone 2.

With the achievement of Milestone 2, the responsibilities for the various aspects of a nuclear power programme should have been assumed by permanent entities.

3.3 Responsibilities

During Phase 1, the NEPIO will be responsible for compiling all the information necessary for the Government to make an informed decision on whether or not to proceed with the development of a nuclear power programme. If a positive decision to do so is taken, the NEPIO will be responsible during Phase 2 for coordinating and overseeing the development of the necessary infrastructure to bring the country to a point of issuing a bid for

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1 The IAEA is developing a Technical Report providing practical guidance on the responsibilities, competencies and interfaces needed by NEPIO, scheduled to be issued by end 2008.
the first NPP project. The NEPIO will study the 19 issues discussed in [1] and produce a comprehensive report clearly laying out the commitments and processes necessary to undertake a nuclear power programme. It may also produce the implementation plan for Phase 2, with resource requirements to which the Government would have to commit to launch this Phase 2.

The outcome of the NEPIO’s work will be a National Position on the introduction of nuclear power which demonstrates an understanding of the nation’s energy needs and the alternative options available for meeting those needs. Along with the energy needs, the existing size and design of the electrical grid should be thoroughly understood. In addition, the international commitments, bilateral relationships, and implications for technology strategy should be understood. Available, existing NPP designs can then be studied to assess their compatibility with those needs and the present grid structure.

During Phase 2, the NEPIO will assume a coordination and oversight role as the owner-operator, the regulator and other organizations take on permanently the functions necessary to carry out the national strategy.

### 3.4 Structure, Competencies and Life Span of NEPIO

A possible structure for the NEPIO organisation is shown in Figure 2.

![Diagram of NEPIO structure](image)

**Figure 2: Example of a Nuclear Energy Programme Implementing Organization (NEPIO)**

To successfully accomplish its responsibilities, the NEPIO must be staffed with individuals capable of exploring and understanding each of the 19 infrastructure issues. Broad use of consultants or interactions with international organizations is strongly encouraged, especially where domestic expertise may not be available. However, the leadership and decision making should remain with national authorities.

During Phase 1 of a nuclear programme development, the NEPIO should be the lead organization guiding the State to Milestone 1. It has to completely define the commitments and requirements necessary to employ a safe, secure and peaceful nuclear power programme. Equally important, during Phase 2, it will have to see that those commitments and
requirements are assumed and carried out by the permanent organizations designated to do so. According to experience, it is likely that the members of the NEPIO will become leaders of the respective institutions and organizations designated to implement the first and subsequent NPP projects.

4 EVALUATION OF INFRASTRUCTURE DEVELOPMENT STATUS

4.1 Overview

The publication "Milestones in the Development of a National Infrastructure for Nuclear Power" [1] provides guidance on the timely preparation for a nuclear power programme through a sequential development of the necessary infrastructure issues. The development is described through a framework of milestones that mark the completion of infrastructure conditions during the progressive phases. The guidance can be applied to evaluate the progress and to aid in planning the further steps necessary for a consistent development of the national infrastructure. The practical implementation of the evaluation requires a suitable means to determine the status of the infrastructure conditions\(^2\).

The evaluation methodology is intended to complement the information presented in [1] by providing a criteria for appraising the status of a country against each of the infrastructure issues. It is focused on the early two phases of the NPP project (up to ready to begin construction) for two key reasons:

- it is important in any major national project to invest wisely and effectively in the early preparatory stages,
- existing IAEA assessment tools and methodologies already provide a sound basis for assessing the status of infrastructure during Phase 3 and beyond, i.e. construction and operation.

The intent of the evaluation is to allow a holistic review of a State readiness to move forward to the next phase of introducing a nuclear power programme. It is also a means for identifying gaps and focusing resource allocation. It is vital to evaluate readiness across all 19 infrastructure issues because each and every one is essential and because there are significant interactions. The management of each infrastructure issue and the human and financial resources required to support them need to be fully integrated.

This holistic evaluation tool based on the IAEA milestones approach can be used either as a self evaluation tool by any State wishing to review its readiness to proceed to the next phase of a nuclear power programme or as a peer review tool where the State can invite others to carry out an independent evaluation of their readiness.

The aim of the evaluation tool is:
- To ensure that all relevant infrastructure issues are reviewed
- To ensure consistency between infrastructure issues
- To ensure a consistent approach between countries thereby providing other countries and potential vendors with an internationally recognized evaluation of a country’s readiness
- To bring the results together in order to identify actions required to move into a subsequent phase of the project to establish a nuclear power programme.

\(^2\) The IAEA is developing a Technical Report providing a means of evaluation based on the milestones approach, scheduled to be issued by the end 2008.
4.2 Evaluation Approach

The basis of the evaluation is a review against the criteria developed for each condition of each infrastructure issue. The scope of this evaluation includes both the ‘hard’ (grid, facilities, etc.) and ‘soft’ (legal, regulatory, training, etc.) infrastructure needed for a NPP. Criteria are defined for each issue of Milestones 1 and 2 in the development of a nuclear power programme. The purpose of the criteria is twofold:

- Firstly to check that all the work required in the phase leading up to the milestone has been adequately completed
- Secondly to help ensuring that the plans for the following phase are comprehensive and realistic

Operation, decommissioning, spent fuel and waste management are addressed to the degree necessary prior to NPP commissioning. All the issues, including those for operation and decommissioning, as well as for spent fuel and waste management, should be considered by the time the bid request is issued. Having reached the point of readiness to commission a NPP, the Member State should have developed an understanding of the commitments required for a successful nuclear power programme and be able to uphold those commitments throughout the NPP life.

In general the evaluation at Milestone 1 is looking at the proposed work programme for Phase 2 and beyond in order to establish if the requirements have been fully understood, scoped and resourced. It is important to look at what infrastructure and operations already exist. For example, countries contemplating a nuclear power programme will already have in place an operational, legal and regulatory system for the safe use and transport of radioactive material and may have a research reactor in operation. One of the key inputs to the overall evaluation will be the results of national and international evaluations of existing activities.

At Milestone 1, there is no nuclear safety risk related to nuclear material; the evaluation is mostly about programme risk management. A State can do less planning in Phase 1 but then may carry a much greater risk of delays or unexpected outcomes because the necessary issues have not been properly scoped. The wide international experience of how best to control these project risks have been taken into account in the development of the evaluation methodology. The activities that need to be undertaken in Phase 1 in order to manage project risks include some activities that need to be started in Phase 1 because of the very long lead time. The typical example of this is related to siting where unless some activities towards site selection and characterisation are carried out in Phase 1, their long lead time will introduce significant delays into the programme.

One of the elements that the evaluation methodology looks for is a clear national commitment. A State will want to make use of international experience and cooperation in the introduction of a nuclear power programme. The use of partnership agreements with vendors and/or countries with experience of NPP operations and the use of recognised experts as consultants are encouraged by the IAEA. However, any evaluation of readiness to proceed to a further phase will want to ensure that a full ownership and understanding of the key issues is with the Member State wishing to implement the nuclear power programme.

4.3 Criteria for Evaluation and their Use

The methodology provides detailed criteria to obtain evidence that a particular condition has been met. They are presented for each of the 19 infrastructure issues and specific criteria are proposed for each condition. An example is provided in Table 2.
Table 2: Example of criteria for evaluating the issue National Position/Milestone 1

<table>
<thead>
<tr>
<th>Conditions</th>
<th>Criteria</th>
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<tbody>
<tr>
<td>Safety, security and non-proliferation needs recognized</td>
<td>1. A document clearly demonstrating the Governments commitment to the safe, secure and peaceful implementation of nuclear energy for the long term.</td>
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<td></td>
<td>1. The charter showing that the NEPIO has been established by and reports to a Senior Government Minister</td>
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<td>2. The basis of the charter is known by other Government ministries and key members of NEPIO</td>
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<td>3. The NEPIO charter clearly charges and authorizes the preparation of a comprehensive report to identify the commitments and conditions necessary to establish a national nuclear power programme. It defines an adequate scope of investigations and clear definition of objectives and timescales. It should identify how its mandate and activities fit with overall plan for implementing nuclear power option</td>
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<td>4. A clear description of how NEPIO operates in terms of funding, office accommodation and equipment, reference material</td>
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<td>5. Evidence showing adequate interactions between and support from appropriate ministers such as those responsible for Energy, Environment, etc</td>
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<td>6. A documented budget planning and reporting process showing appropriate funding is provided to and expended by NEPIO to fulfil its charter in the scheduled time</td>
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<td>7. Organisation chart; job descriptions and CVs of members demonstrating appropriate skills, qualifications and experience to address all the infrastructure issues based on requirements in IAEA-TECDOC–1513 [3]. This includes appropriate use of consultants and the demonstration of national staff as “intelligent customer”</td>
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<td>8. Comprehensive report produced by NEPIO covering all areas identified in [1] and recognising the resources and timescales required for the activities required for Phase 2. A demonstration that the Member State can provide the overall resources required integrated across all areas.</td>
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<td>9. Executive summary of comprehensive report is based on detailed report, contains estimates of total resources and timescales and has been properly reviewed by senior government officials</td>
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For each issue, there should be a clear work programme for the next phase of the project which states the objectives of the work programme, the detailed activities, the funding and resources required, how it will be provided and the timescales for each activity.

Several of the 19 issues apply to any major project and need to be evaluated in a similar way as to any other project. However, there are often additional “nuclear” project features and these are identified under the appropriate issue.

Strong evidence of a holistic approach to information gathering, resource development and decision making is needed. It is, for example, no use having a small team fully aware of the nuclear safety requirements if there is not a clear plan to develop a competent operating organization with a strong safety culture, or a plan to ensure the capability to produce components with an assurance of the required integrity or reliability. This view will be obtained by looking at each of the 19 issues and then drawing the detailed evaluation together. For example, to be assured that Milestone 1 has been reached, it is necessary to see that:

- the State has the knowledge that is required;
- sufficient resources and attention of senior officers has been given to the analysis;
- the overall strategy and objectives are sound;
- programme risk is adequately managed;
- the plans for work and resources required for Phase 2 are sound;
- the existing activities associated with radiation sources are adequately managed; controlled and regulated and that has been benchmarked;
- there is strong government commitment to the programme.

The criteria proposed often refer to obtaining “evidence” and “plans”. Evidence can include reports, meeting notes, correspondence, talks and presentations, conferences attended with meeting report, discussions, CV’s, organisation descriptions, job descriptions etc. Plans need to have clear actions with associated timescales, resources required and evidence that they are available. In all cases there should be evidence that the documents have been approved by a person/organisation with the appropriate authority.

ACKNOWLEDGMENTS

The authors are the Scientific Secretaries responsible for the development of the IAEA publications related with infrastructure and planning for nuclear power programmes. The material of this paper was taken from several IAEA publications developed with the contributions from many experts from inside as well outside IAEA.

REFERENCES


Safety of Future Reactor Designs
A Critical Overview of Heat Transfer Phenomena through Different Solutions of In-Pool Condensers for Passive Safety Systems

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ABSTRACT

Safety systems for advanced LWRs often rely on natural phenomena like the natural circulation, in order to increase simplicity and improve their safety. Emergency heat removal is a fundamental safety function which in some designs of advanced light water reactors is guaranteed by means of a passive circuit. A typical passive safety system for that purpose is the two-phase flow, natural circulation, closed loop system, where the heat is removed by means of a steam generator or heat exchanger, a condenser and a pool.

Various solutions of condenser tubes arrangement have been implemented for applications to the next generation NPPs. In this paper two possibilities, which represent the simplest and most used configurations, are analyzed. Horizontal U-tubes have been chosen e.g. in the emergency condenser of the SWR1000, whereas a vertical pipes solution has been adopted in the GE-SBWR1000 Isolation Condenser (IC), as well as in the IRIS Emergency Heat Removal System (EHRS).

The horizontal solution with a U-tubes disposition offers the advantage to inherently allow the thermal expansions, while a straight single pass vertical tube could create dangerous thermal stresses due to prevented dilatations. Nevertheless, the better thermalhydraulic performance, mostly with the meaning of higher heat transfer coefficients, provided by vertical tubes is noticeable. In a horizontal tube, the HTC is strongly influenced by the flow pattern, which can be very different (ranging from annular to stratified) without avoiding intermittent regimes; in a stratified regime the HTC is negatively affected by the sump accumulation on the bottom of the pipe. The two-phase flow path is instead well defined in a vertical tube. An annular film of condensate forms on the wall and is driven by gravity reaching higher velocities: resulting higher HTCs for the vertical solution.

This paper deals with a critical comparison between condensation in horizontal and vertical tubes, focusing on the thermalhydraulic aspects and describing the best-estimate correlations, with the aim to point out the global advantages of the vertical tubes arrangement. At the end, an experimental validation of the correlations proposed for the vertical tube, is provided relying on PERSEO facility (SIET labs, Piacenza), showing a good accordance with the model discussed.

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### Acronyms

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<th>Acronym</th>
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<tr>
<td>ECT</td>
<td>Emergency Cooldown Tank</td>
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<tr>
<td>EHRS</td>
<td>Emergency Heat Removal System</td>
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<td>ESBWR</td>
<td>European Simplified Boiling Water Reactor</td>
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<td>GDCS</td>
<td>Gravity Driven Cooling System</td>
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<tr>
<td>HTC</td>
<td>Heat Transfer Coefficient</td>
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<td>HX</td>
<td>Heat eXchanger</td>
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<tr>
<td>IC</td>
<td>Isolation Condenser</td>
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<tr>
<td>IRIS</td>
<td>International Reactor Innovative and Secure</td>
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<tr>
<td>KNGR</td>
<td>Korea Next Generation Reactor</td>
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<tr>
<td>LWR</td>
<td>Light Water Reactor</td>
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<tr>
<td>PCCS</td>
<td>Passive Containment Cooling System</td>
</tr>
<tr>
<td>PERSEO</td>
<td>in-Pool Energy Removal System for Emergency Operation</td>
</tr>
<tr>
<td>PRHRS</td>
<td>Passive Residual Heat Removal System</td>
</tr>
<tr>
<td>RPV</td>
<td>Reactor Pressure Vessel</td>
</tr>
<tr>
<td>SBWR</td>
<td>Simplified Boiling Water Reactor</td>
</tr>
<tr>
<td>SIET</td>
<td>Società Informazioni Esperienze Termoidrauliche</td>
</tr>
<tr>
<td>SMART</td>
<td>System integrated Modular Advanced Reactor</td>
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<tr>
<td>SWR1000</td>
<td>Siede Wasser Reaktor – 1000 MW_e</td>
</tr>
<tr>
<td>VISTA</td>
<td>experimental Verification by Integral Simulation of Transient and Accidents</td>
</tr>
</tbody>
</table>

### Nomenclature

#### Greek Symbols

- $\alpha$: void fraction [-]
- $\Delta h_{lg}$: latent heat of vaporization [J/kg]
- $\delta$: film thickness of annular ring [m]
- $\Phi_i$: heat flux related to internal diameter [W/m$^2$]
- $\Gamma$: mass flowrate [kg/s]
- $\Gamma_i$: condensate mass flowrate [kg/s]
- $\mu$: liquid dynamic viscosity [Pa s]
- $\nu_l$: liquid cinematic viscosity [m$^2$/s]
- $\theta$: falling film stratification angle [-]
- $\rho_g$: steam density [kg/m$^3$]
- $\rho_l$: liquid density [kg/m$^3$]

#### Subscripts

- $\text{cond}$: condensation
- $\text{e}$: external
- $i$: internal
- $\text{lam}$: laminar regime
- $\text{lam}_\text{wavy}$: laminar wavy regime
- $\text{pool}$: pool boiling
- $\text{sat}$: saturation
- $\text{tube}$: tube metal
- $\text{turb}$: turbulent regime
- $w$: wall
1 INTRODUCTION

The nuclear advanced water reactors design is primarily focused on the achieving of innovative safety characteristics. The main goal of a safety design is to establish and maintain core cooling and ensure containment integrity, for any transient situation, so to minimize the core damage and the fission product release probabilities. This is permitted in practice by means of an integral layout, where the vessel contains all the principal components of the primary circuit (reactor core, cooling pumps, steam generators, pressurizer etc.).

Another important strategy in an advanced safety concept is the large utilization of passive systems. They need no operator action, nor AC power sources, being based only on natural forces, such as gravity and natural circulation, for their continued operation.

A typical passive safety system is a natural circulation loop able to transfer core decay heat and sensible heat from the reactor coolant to the environment during transients, accidents or whenever the normal heat removal paths are lost. Usually, such a system consists of a steam generator (behaving as hot well), a hot leg (circuit riser), a heat sink composed by a heat exchanger bundle submerged in a pool, and a cold leg (circuit downcomer). Its operation is based on the high heat transfer capability offered by the steam generator, which is able to remove decay heat producing steam which rises up in the loop and then condenses rejecting this heat into the pool; the downcomer closes the circuit by bringing back the cold condensed water to the steam generator.

This paper is based on a critical analysis of a component of the circuit discussed above, which is the condenser submerged in the pool. The aim is to compare and assess two different approaches for realizing the condensation, i.e. using a horizontal tubes arrangement or instead a vertical tubes arrangement. The comparison is mainly focused on the thermalhydraulic aspects, reproducing by means of simple Matlab/Excel codes the most updated and reliable modelling of condensation found in literature.

2 LITERATURE REVIEW OF PASSIVE SAFETY SYSTEMS FOR DECAY HEAT REMOVAL

A first concept of in-pool immersed heat exchangers deals with a parallel arrangement of horizontal U-tubes between two common headers. This configuration has been provided in the emergency condenser of a SWR1000, as Fig.1 shows. The top header is connected via piping to the reactor pressure vessel steam plenum, while the lower header is connected to the reactor vessel below the water level. The heat exchangers are located in a lateral pool filled with cold water, forming a system of communicating pipes with the vessel. At normal reactor water level, the emergency condensers are flooded with cold water; if a reactor trip occurs, the level can drop so that the heat exchanging surfaces inside the tubes are gradually uncovered. The incoming steam condenses on the cold surfaces, and the condensate is simply returned to the reactor vessel, avoiding any core uncovering [1].

Other advanced water reactors chose a vertical tubes arrangement. This has been followed, for example, in the Isolation Condenser (IC) of a SBWR and in the Passive Containment Cooling System (PCCS) of a KNGR. In both cases, decay heat is removed passively from the containment by the condenser submerged in a suitable pool. In a SBWR the system ICs are designed to passively limit the reactor pressure under accident conditions, whereas the containment ICs have to remove the decay heat after a LOCA. The physical locations of the ICs are above the RPV, Suppression Pools and Gravity Driven Cooling System (GDCS), in order to utilize the gravity induced return flow of the condensate from the ICs to the respective ports. The reactor layout is reported in Fig.2 [2].
The PCCS of a KNGR (shown in Fig.3) is perfectly similar to the passive containment cooling system in the Simplified Boiling Water Reactor. It is provided with two heat exchangers, relevant lines and water tanks, and guarantees a way of external containment cooling through a natural circulation circuit. Heat transferred from the containment atmosphere to the coolant through the primary heat exchanger tube is, in the same way described above, removed by the condenser tubes to the water tank, which is outside the containment [3].

Also the innovative IRIS reactor has its passive safety system for removing the decay heat, which is the Emergency Heat Removal System (EHRS), reproduced in Fig.4. Firstly a horizontal U-tubes heat exchanger should have been provided; then the choice fell on a vertical tubes arrangement, adopting the same condenser used for the IC of the SBWR [4]. The Korean SMART reactor is conceived according to the same philosophy of modularity and integral layout; the decay heat removal function is accomplished by the natural circulation in the PRHRS, based again on steam generators for heat extraction and on submerged condensers, precisely located in the Emergency Cooldown Tank (ECT), for heat discharging. A vertical tubes arrangement has been proposed [5].

The different solutions developed in the Generation III+ reactors pointed out a dominance of vertical condensers, whose better performance can be demonstrated with a careful study of the different phenomena involving the condensation process.

3 THERMALHYDRAULIC PHENOMENA IN HORIZONTAL AND VERTICAL TUBES CONDENSERS

The main advantages offered by the vertical tubes solution concern the thermalhydraulic features. The power that should be rejected in a passive system condenser is roughly 20 - 30 MW, which brings to the establishing of low mass fluxes (surely $G < 100$ kg/sm$^2$) in the loop.

The condensation in horizontal tubes is strongly influenced by the flow pattern, which can be very different (ranging from annular to stratified) without avoiding intermittent regimes. In a defined interval of mass flux and qualities, in fact, the possibility of pulsation is concrete.
At low flowrates and at the medium-high pressures characterizing a natural circulation loop for decay heat removal (typically 60 - 100 bar), a fully stratified/stratified wavy regime occurs. A thin condensate film drains down from the upper part of the tube under the influence of gravity force, whereas a water sump accumulates and flows on the bottom. The HTC is of course negatively affected by this sump, especially when it reaches subcooling conditions towards the end. The effect of interface waves, increasing with the flowrate, does not overcome this disadvantage. Average heat transfer coefficients around 4000 - 6000 W/m²K result (evaluated considering G equal to 60 kg/sm²). Condensation in a horizontal tube deals thus with complex phenomena, which deserve a careful analysis.

The two-phase flow path is instead well defined in a vertical tube. An annular film of condensate forms on the wall, and falls down driven by gravity, filling up the cross section only at the end of the tube and assuring thus a better condensation process. Usually, the condensate flows at higher velocity, and this reduces fouling and corrosion effects, besides assuring higher HTCs. At fixed quality, the bigger velocities (pulled by gravity) lead to a thinner film, with a smaller thermal resistance, a smaller laminar boundary layer and thus higher turbulence. Average heat transfer coefficients are strongly enhanced with respect to the horizontal solution, leading to values around 8000 - 11000 W/m²K. In literature, a lack of high pressure steam condensation data for large diameter condenser vertical
tubes, which is the case of the passive heat removal systems in KNGR and SBWR, has been noticed. A comprehensive investigation of how condensation is provided in a vertical tube appears thus useful. The different physical phenomena interesting horizontal and vertical pipes can be easier understood considering the drafts in Fig. 5 and Fig. 6.

The better thermalhydraulic performance of a vertical tube (assuring HTCs even doubled in comparison with a horizontal pipe) is anyway modulated by the strong influence that the tube metal thermal conductance has on the heat transfer. The material revealing as the most suitable for in-pool condensers is INCONEL 600, thanks to its better mechanical properties at high temperature (≈300 °C) and its higher resistance to corrosion phenomena.

4 CONDENSATION MODELS

4.1 Modelling of condensation in a horizontal tube

In early models, the flow patterns were classified just under two categories, i.e. stratified or annular. The first, dominated by gravity forces, considers a thick condensate layer flowing along the bottom of the tube, while a thin liquid film forms on the wall in the upper portion. The latter, dominated by shear effects, leads to an annular ring of condensate flowing uniformly along the tube. Actually, different flow regimes can be induced. When the stratified condensate layer in the tube sump reaches medium-high velocities, often ripples or waves are generated at the phase surface (stratified wavy). If these waves become so large to wash the top of the tube, an intermittent flow pattern (slug – plug) can establish, which has a very complex flow structure. At very high velocities, instead, a mist (spray) regime can occur, characterized by impinging droplets on the thin unsteady liquid film.

Two models are proposed as the most updated and reliable concerning the condensation in a horizontal tube, each one based on a proper flow pattern map for the identification of the flow regime (reported in Fig. 7 and Fig. 8).

In Thome’s model the intermittent (both slug and plug) and mist flows are considered and evaluated as annular flow. For annular flow, a uniform film thickness is assumed and the actual larger thickness of the film at the bottom than the top due to gravity is ignored. Stratified and stratified wavy are instead characterized by the so called stratification angle, which subtends the cross sectional area occupied by the liquid, assumed as truncated annular ring of uniform thickness. Fig. 9 is useful to understand the simplifications provided.

---

**Fig.7-** Thome’s flow pattern map for condensation in a horizontal tube.

S → stratified  SW → stratified wavy  A → annular  I → intermittent  MF → mist flow

**Fig.8-** Tandon’s flow pattern map for condensation in a horizontal tube.
Two different heat transfer mechanisms are considered within the tube: convective condensation and film condensation. The first refers to the axial flow of the condensate along the channel due to the imposed pressure gradient; the second refers to the flow of condensate from the top of the tube towards the bottom due to gravity. The convective condensation heat transfer coefficient $h_c$ is applied to the perimeter wetted by the axial flow of liquid film, which is the entire perimeter in annular flow, but only part of the perimeter in stratified wavy and stratified one. The film condensation heat transfer coefficient $h_f$, characterizing only stratified – stratified wavy regime, is obtained by applying the Nusselt’s falling film theory to the inside of the horizontal tube, assuming the falling film laminar [6]. Thus, the condensation HTC is given combining these two coefficients according to:

$$h = \frac{h_c}{\pi D_i} \left[ \frac{2\pi - \theta}{2} \right] + h_f \left[ \frac{2\pi - \theta}{2} \right]$$

(1)

The convective condensation heat transfer coefficient $h_c$ can be obtained from the following film equation, assuming the axial flow as turbulent:

$$h_c = 0.003 \cdot Re_i^{0.74} \cdot Pr_i^{0.5} \cdot \frac{k_i}{\delta} \cdot f_i$$

(2)

where $\delta$ is the film thickness, considered uniform in all the cross section. This is an only liquid correlation type, i.e. just the liquid part of the two-phase flowrate has to be considered. Resulting:

$$Re_i = \frac{4G(1-x)\delta}{(1-\alpha)\mu_i}$$

(3)

The interfacial roughness correction factor $f_i$ takes into account the shear effects that dominate an annular flow regime, leading to a remarkable increasing of the heat transfer.

The film condensation heat transfer coefficient $h_f$, instead, is given from a modification of the Nusselt’s theory for laminar flow of a falling film outside a horizontal tube (around the perimeter, from top to bottom), applied to the condensation on the inside. Any effect of axial shear on the falling film is ignored. Since heat exchanger design codes are typically implemented assuming the heat flux in each incremental zone along the exchanger, the heat flux version of the Nusselt’s equation is preferable to be considered:

$$h_f = 0.655 \left[ \frac{\rho_l (\rho_l - \rho_s) \varphi \Delta h_y k_i^3}{\mu_i D_i \Phi_i} \right]^{1/3}$$

(4)

The heat transfer coefficient given by equation (4) has to be intended as average HTC if referred to the tube average heat flux, whereas represents a local value if the tube length is divided into different cells and the local heat flux for each cell is taken into account. In the same way, equation (2) gives a local HTC if the mean quality at half cell is considered. All the computational details to calculate $h_c$ and $h_f$ can be anyway found in literature [7] [8].

Schaffrath’s model was developed for the investigation of the operation mode of the SWR1000 emergency condenser (NOKO test facility) [1] [9]. It uses the flow regime map of Tandon for the determination of the actual flow regime and switches to flow regime semi-
empirical correlation for the calculation of HTC: Soliman for spray flow [10], Nusselt for laminar annular flow, Kosky and Staub for turbulent annular flow, Rufer and Kezios for stratified flow, Breber for slug-plug flow. The parameters chosen to identify the flow pattern are the dimensionless vapour velocity $\dot{j}_D$ and the volume ratio of liquid and gas in a cross-sectional area $(1-\alpha)/\alpha$.

### 4.2 Modelling of condensation in a vertical tube

In a vertical pipe condensation takes place with a condensate film growing up in contact with the wall, while vapour phase flows along the bulk of the channel; according to Nusselt’s approximation, the condensation occurs just at the interface (and only conductive processes are considered within the film) [6]. Many methods have been proposed for predicting the film condensation heat transfer coefficient; these range from empirical or semiempirical correlations to highly sophisticated analytical treatments of the transport phenomena.

In this paper the distinction of film condensation into three different regimes has been provided, according to the different physical phenomena which can occur: laminar, laminar wavy and turbulent. The main parameter which governs the process is the condensate velocity, expressed by its Reynolds number $Re_t$. Resulting:

$$Re_t = \frac{4\Gamma_l}{\pi D_f \mu_l}$$

where $\Gamma_l$ is the liquid flowrate, to be expressed according to an only liquid approach:

$$\Gamma_l = \Gamma(1-x)$$

- **Laminar** $Re_t < 30$:

  In the laminar region, condensation HTCs decrease with increasing film thickness due to the increased thermal resistance. Nevertheless, the laminar condensation occupies a very short portion of the tube and can be so neglected in the calculations. Nusselt correlation (i in Tab.1), which can be derived analytically by solving the momentum and the energy equations inside the condensate layer, is valid in case of laminar film.

- **Laminar wavy** $30 < Re_t < Re_{tr} = 4658 Pr_t^{-0.05}$:

  When the film becomes wavy, HTCs (again a decreasing function of film thickness) grow up because the waves promote turbulence in the film and increase heat exchange surface [11]. $Re_{tr}$ represents the transition value between laminar (wavy) and turbulent film condensation. Kutateladze correlation (ii in Tab.1) is recommended.

- **Turbulent** $Re_t > Re_{tr} = 4658 Pr_t^{-0.05}$:

  Once the film has become turbulent, the highest HTCs are provided; the trend is now different from previous case (i.e. an increasing function of film thickness) due to the fact that mixing effects exceed the greater thermal resistance. Two different semiempirical correlations due respectively to Labuntsov (iii in Tab.1) and Chen (iv in Tab.1) can be applied [12] [13].

  All these correlations, listed in the first column of Tab.1, give a local HTC value. Sometimes, especially in design applications, a general expression giving an average heat transfer coefficient is more useful. The easiest way to evaluate the average heat transfer coefficients is to integrate the equations for local coefficients along the tube length, resulting:

$$\bar{h}_L = \frac{1}{L} \int_0^L h(s)ds$$  \(7\)
The correlations proposed for the evaluation of a mean HTC, according to the different flow regimes, are listed in the second column of Tab.1. Also the local and the average versions of Shah correlation are reported. This is an empirical correlation based on a wide range of experimental data, and mostly considered as the best correlation for the turbulent film condensation heat transfer both in horizontal, vertical and inclined pipes. It is based on a liquid only approach, referring the two-phase HTC to the single-phase coefficient computed with Dittus-Boelter correlation assuming all the flowrate being liquid \[14\]. Nevertheless, the validity of Shah correlation is questionable for high pressure and large diameter tube applications with water, as has been recently experimentally confirmed by Kim \[15\]. Furthermore, this correlation predicts a decreasing HTC with decreasing quality, exactly the opposite trend of film condensation theory. It is also questionable how the same correlation could be suitable for all the flow orientations, since it has been widely seen how the physical phenomena involved are different.

### Tab.1- List of the correlations presented for the evaluation of local and average HTCs during condensation inside a vertical tube.

<table>
<thead>
<tr>
<th>Correlation</th>
<th>Local heat transfer coefficient</th>
<th>Average heat transfer coefficient</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Nusselt</strong></td>
<td>( h_{\text{local}} = k \left( \frac{v_f^2}{g} \right)^{-1/3} ) (1.1 Re_f^{-1/3}) ([i])</td>
<td>( \overline{h}_{\text{local}} = k \left( \frac{v_f^2}{g} \right)^{-1/3} ) (1.47 Re_f^{-1/3}) ([vi])</td>
</tr>
<tr>
<td><strong>Kutateladze</strong></td>
<td>( h_{\text{local - wavy}} = k \left( \frac{v_f^2}{g} \right)^{-1/3} ) (0.756 Re_f^{-0.22}) ([ii])</td>
<td>( \overline{h}_{\text{local - wavy}} = k \left( \frac{v_f^2}{g} \right)^{-1/3} ) (\frac{Re_f}{1.08 \left( Re_f^{0.22} - 5.2 \right)}) ([vii])</td>
</tr>
<tr>
<td><strong>Labuntsov</strong></td>
<td>( h_{\text{turb}} = k \left( \frac{v_f^2}{g} \right)^{-1/3} ) (0.023 Re_f^{0.25} Pr_f^{0.5}) ([iii])</td>
<td>( \overline{h}_{\text{turb}} = k \left( \frac{v_f^2}{g} \right)^{-1/3} ) (\frac{Re_f}{8750 + 58 Pr_f^{-0.5} \left( Re_f^{0.75} - 253 \right)}) ([viii])</td>
</tr>
<tr>
<td><strong>Chen</strong></td>
<td>( h_{\text{turb}} = k \left( \frac{v_f^2}{g} \right)^{-1/3} ) (0.00402 Re_f^{0.4} Pr_f^{0.65}) ([iv])</td>
<td>( \overline{h}_{\text{turb}} = k \left( \frac{v_f^2}{g} \right)^{-1/3} ) (\frac{Re_f^{-0.44} + 5.82 \cdot 10^{-6} Re_f^{0.8} Pr_f^{0.5}}{Re_f^{0.75} - 253} \rho_f^{-0.25}) ([ix])</td>
</tr>
<tr>
<td><strong>Shah</strong></td>
<td>( h_{\text{turb}} = h_{\text{local}} \left( 1 - x \right)^{0.8} + \frac{3.8x^{0.76} \left( 1 - x \right)^{0.04}}{Pr_f^{0.38}} \rho_f^{0.38}) ([v])</td>
<td>( \overline{h}<em>{\text{turb}} = h</em>{\text{local}} \left( 0.55 + 2.09 Pr_f^{-0.38} \right) ) ([x])</td>
</tr>
</tbody>
</table>

The correlations proposed for the evaluation of a mean HTC, according to the different flow regimes, are listed in the second column of Tab.1. Also the local and the average versions of Shah correlation are reported. This is an empirical correlation based on a wide range of experimental data, and mostly considered as the best correlation for the turbulent film condensation heat transfer both in horizontal, vertical and inclined pipes. It is based on a liquid only approach, referring the two-phase HTC to the single-phase coefficient computed with Dittus-Boelter correlation assuming all the flowrate being liquid \[14\]. Nevertheless, the validity of Shah correlation is questionable for high pressure and large diameter tube applications with water, as has been recently experimentally confirmed by Kim \[15\]. Furthermore, this correlation predicts a decreasing HTC with decreasing quality, exactly the opposite trend of film condensation theory. It is also questionable how the same correlation could be suitable for all the flow orientations, since it has been widely seen how the physical phenomena involved are different.

### 5 CALCULATION RESULTS

In order to test the proposed models of condensation, the data of SBWR Isolation Condenser (IC) mean tube design have been taken as reference. The aim is to show with simple calculations (without the utilization of a code) how a vertical pipe can guarantee the highest HTCs during condensation, as well as the best trend (i.e. increasing along the tube abscissa).

The considered mean tube (with an inner diameter of 46.2 mm and an outer diameter of 50.8 mm) is submerged in a pool of boiling water at 100°C; it is made up of INCONEL 600 (17.4 W/mK of thermal conductivity), and its inlet conditions are represented by saturated steam at 72.4 bar. The utilization of this HX in the EHRS of IRIS has been taken into account. Thus, following design prescriptions, two units of the condenser (with 120 tubes each one) have to exchange altogether a thermal power of 35 MW; fouling effects for both internal and external side (with an additional thermal resistance of \(5 \times 10^{-5} \text{ m}^2\text{K}/\text{W}\)), plus a possible plugging of the 5% of the tubes, are also considered. Resulting a flowrate per tube of 0.103 kg/s, giving a mass flux \(G\) equal to 61.47 kg/sm².
Working out from the real contest to which these data are reported and evaluating just a flow in a horizontal tube, such value of mass flux leads to a stratified pattern. Heat transfer coefficient trend along the pipe has been reconstructed, applying first Thome’s analysis and then Schaffrath’s model, switching to the proper correlation (Rufer and Kezious formula). Also an indication of the tube length needed to achieve saturated liquid at outlet has been given. A total tube length of about 2.50 m results. As it is shown in Fig.10, a very good agreement can be observed between the two different models: an average heat transfer coefficient of around 5500 W/m²K occurs along almost the entire length, except for the end, where the sump effects become dominant and oppose the heat transfer. Besides, in the final part of the pipe (corresponding to low qualities), the Schaffrath’s analysis suggests the possibility of a slug-plug flow (when x<0.1), which could induce instabilities. Thus, the condensation in a horizontal tube is not a so confident phenomenon, since these instabilities could create some problems during the condenser working.

The other condensation regimes, which can occur in a horizontal tube according to the different values of flowrate, have been investigated with Fortran simple codes. Two different values for inlet flowrate have been considered: 0.7 kg/s (giving a dominant annular flow) and 1.5 kg/s (inducing a mist flow at the beginning of the pipe). The outputs of the implementations are reported in Fig.11 and Fig.12. The calculations have been provided considering all the water properties at 72.4 bar.

The condensation models proposed for the vertical tube have been then tested. Resulting a pipe length of 2.17 m as regards the film condensation, and of 3.04 m for the Shah correlation. Since the first model is surely more trustworthy, being based on physical principles, it appears evident how for the IC conditions of pressure, flowrate and tube diameter (globally representative of the typical conditions for the working of an in-pool immersed HX in a natural circulation loop), the Shah’s model broadly under-predicts the

![Fig.10- Comparison between Thome’s analysis and Schaffrath’s analysis in stratified flow \[G=61.5 \text{ kg/sm}^2\].](image)

![Fig.11- Comparison between Thome’s analysis and Schaffrath’s analysis with dominant annular flow \[G=417.6 \text{ kg/sm}^2\].](image)

![Fig.12- Comparison between Thome’s analysis and Schaffrath’s analysis at very high flowrates \[G=984.8 \text{ kg/sm}^2\].](image)
condensation HTC. Furthermore, an opposite trend of turbulent heat transfer coefficient with quality decrease has been confirmed between the two models (increasing for film condensation, but decreasing for Shah’s model). The comparison is given by the graphs reported in Fig.13 and Fig.14. In film condensation, laminar zone is absolutely negligible, and laminar wavy portion (with its decreasing trend of HTC with tube abscissa) lasts up to a quality value of 0.95. As regards turbulent zone, both Labuntsov and Chen correlations have been considered; the latter leads to lower HTCs appearing more conservative (and thus it has been used in the sizing calculations). Chen correlation proposed for the mean HTC along the tube (ix in Tab.1), gives a value of about 8000 W/m²K. It is clear how the vertical tube is in position to assure better performance; a direct consequence of this is the greater length required by horizontal tubes for a complete condensation of the steam (i.e. 2.50 m against the 2.17 m of the vertical model).

6 SHORT REVIEW ON EXPERIMENTAL FACILITIES

A critical discussion on the best solution for a decay heat removal HX cannot go without a short description of the experimental work found on this matter in literature.

For the experimental investigation of the SWR1000 emergency condenser effectiveness, the NOKO test facility has been constructed at the Forschungszentrum Jülich. The condenser operation conditions were determined as a function of the primary and pool pressure, characterizing the tube surface available for condensation by studying the geodetic pressure drops variation as a consequence of vessel water level decrease. A single tube test, besides, was provided to clarify the condensation flow regimes inside a horizontal tube and to validate Tandon’s flow map condensation model (reproduced in Fig.8). This work led to the improvement of ATHLET thermal-hydraulic code, implementing the correlations discussed at the end of paragraph 4.1 [9].

The large scale PANDA facility was constructed at the Paul Scherrer Institute (PSI) for the investigation of both overall dynamic response and the key phenomena of passive containment systems. The major aspect of the tests dealt with the evaluation of the ESBWR PCCS performance in case of main steam line break. As regard the condensation issues, PANDA facility was very useful for studying the degradation of HTC in presence of non-condensable gasses, both lighter than steam (hydrogen, simulated by helium), and heavier (nitrogen or air). The provided tests permitted furthermore to extend the data base available for containment analysis code qualification [16] [17].

![Fig.13- Condensation HTC along the tube abscissa, Fig.14- Condensation HTC along the tube abscissa, according to condensation film theory \( G=61.5 \text{ kg/m}^2 \text{s} \). according to Shah correlation \( G=61.5 \text{ kg/m}^2 \).]
The VISTA facility was instead developed by the KAERI to simulate the primary and secondary systems as well as the PRHRS of the SMART reactor. An attractive objective was to confirm the capability of MARS code to predict the overall thermalhydraulic behavior of the PRHRS. An important outcome was that the code under-predicted the heat transfer at the heat exchanger [5]. The condensation model used in the code, exactly the same of that implemented in RELAP5 code, considers the maximum value between the Nusselt correlation (7) for the laminar flow and the Shah correlation ($v$ in Tab. 1) for the turbulent flow. The most suitable heat transfer correlations proposed in the paper are thus required to be implemented in these codes.

The reliability of the active valves involved in the passive systems actuation is fundamental. A new concept of valve liquid side, instead of steam side in the primary system, located on a line connecting two pools at the bottom, has been proposed in the PERSEO facility, developed in SIET labs (Piacenza) for testing a full scaled module for the GE-SBWR in-pool heat exchanger. The valve (named triggering valve) is closed during normal operation and the pool containing the heat exchanger (HX pool) is empty; the other pool (Overall pool) is full of cold water. In emergency conditions the valve is opened and the heat exchanger is flooded, with consequent heat transfer from the primary side to the pool. The effectiveness of the actuation valve movement from the high pressure primary side of the reactor to the low pressure pool side, has been tested during the experimental campaign, both in steady and in unsteady conditions [18].

### 6.1 Experimental results from PERSEO facility

The strong collaboration between POLIMI and SIET made possible the analysis of the experimental data collected during the PERSEO campaign. In particular, the greater interest focused on the condenser performance. Some tubes were monitored with wall thermocouples (K-type, nominal accuracy of ±1.5°C), mounted on the outer surface at three different axial positions: at the top, at the middle and at the end of the pipe. The circuit pressure (equal to 7 MPa) was measured with a pressure transmitter installed inside the HX upper header, whereas primary flow rate (on average 0.1 kg/s per tube) was measured by an orifice differential pressure transmitter (with an uncertainty of ±0.25% of the instrument full scale). Due to the lack of accuracy of fluid thermocouples, saturated steam has been considered at condenser inlet and saturated liquid at outlet. This assumption permitted to calculate the exchanged thermal power, and then to obtain the local heat flux needed for the computation of the heat transfer coefficients. The saturation temperature at the circuit pressure has been adopted as fluid bulk temperature, while the inner wall temperature has been calculated by the external tube value (measured) considering the thermal jump in the metal. Resulting:

$$ T_{w,\text{int}} = T_{w,\text{ext}} + \frac{D_i \ln(D_e/D_i)}{2k_{\text{tube}}} \Phi_i $$

The local heat flux value $\Phi$ and the local HTC $h_{\text{cond}}$ have been obtained by solving the following equation system, where $T_{w,\text{int}}$ must be calculated according to equation (8):

$$ \left\{ \begin{array}{l} U_i = \frac{\Phi}{T_{\text{sat,\text{int}}} - T_{\text{sat,\text{ext}}}} = \left( \frac{1}{h_{\text{cond}}} + \frac{D_i \ln(D_e/D_i)}{2k_{\text{tube}}} + \frac{1}{h_{\text{pool}} D_i} \frac{D_e}{D_i} \right)^{-1} \\ \frac{\Phi}{T_{\text{sat,\text{int}}} - T_{w,\text{int}}} = h_{\text{cond}} \end{array} \right. $$

$$ (9) $$
The reference is the situation depicted in Fig. 15. The results are presented in Fig. 16, where the HTC trend according to film condensation theory (in particular, Kutateladze correlation for the laminar wavy zone and Chen correlation for the turbulent zone) and to a semi-empirical model proposed by Kim et al. [15] are also reported. It is pointed out that the experimental work of Kim was the only one found in literature dealing with a large diameter vertical tube used for condensing high pressure steam; in particular, the test section consisted in a tube with the same geometrical features of the SBWR IC tested in the PERSEO facility (i.e. outer diameter of 50.8 mm, thickness of 2.3 mm and length of 1.8 m). A turbulent film condensation model based on the similarity between the single-phase turbulent convective heat transfer and the annular film condensation heat transfer has been developed. For any computational detail, refer to [15].

The graph in Fig. 16 reveals that only at the end of the tube a remarkable shifting between theory and experimental data is present; this is maybe due to the fact that when the void fraction falls down (exactly at the bottom of the pipe), the liquid film at the wall cannot be considered thin anymore, and thus the conditions for the applicability of the presented theory fail. Fig. 17 shows the comparison of the PERSEO data with the ones found in literature; the model proposed by Kim and the Shah correlation (v in Tab. 1) are considered in order to predict the experimental values. The new data are slightly bigger than Kim ones; this should be due to the higher condensate Reynolds numbers reached in PERSEO facility. The order of magnitude is however well captured. Anymore, both the data bases confirm that the Shah model is not accurate to deal with the condensation phenomena at high pressure, whereas the turbulent annular model proposed gives more reasonable results.

At the end, the film condensation model based on Kutateladze correlation (ii) and Chen correlation (iii) is considered to predict PERSEO data. The results are reported in Fig. 18. The set of correlations proposed in this paper gives the best agreement with PERSEO data, while the turbulent annular condensation model proposed by Kim fails at low qualities. Except for the tube inlet (high qualities), the Shah model broadly under-predicts the condensation HTC.
A review on the concept of in-pool condenser for passive safety systems has been carried out in the first part of the paper. A preponderance of vertical tubes arrangement solutions has been pointed out; the main advantages offered by a vertical configuration are:

- higher heat transfer coefficients (even doubled with respect to the horizontal solution);
- better and more predictable in-tube flow distribution.

In order to prove the better thermalhydraulic performance of condensation in a vertical pipe, the most updated and reliable models for condensation in horizontal and vertical tubes have been found in literature and tested. As regards the horizontal pipe, two methods of analysis have been focused. The first is due to Thome and, distinguishing between annular, stratified and stratified wavy regimes, considers two heat transfer mechanisms: convective condensation and film condensation. The second method has been proposed by Schaffrath, and it is based on Tandon’s map. Condensation in a vertical pipe is instead dealt according to a physical principle analysis based on the film condensation theory. In this paper the distinction of film condensation into three different regimes has been provided: laminar, laminar wavy and turbulent. Suitable correlations for each regime have been proposed.

A comparison between all the models has been provided by means of simple Excel/Matlab codes, referring to the SBWR IC mean tube data, considering the utilization of this condenser for the IRIS EHRS. A stratified flow is induced in case of horizontal tube; resulting a quite constant value of HTC (around 5500 W/m²K), except for the end, when the sump effects become dominant. Film condensation theory have been then applied for a vertical pipe. The establishing of turbulence is responsible for the dominant increasing trend. Chen correlation indicates a mean HTC clearly bigger (around 8000 W/m²K). A direct consequence of this fact is the minor length required for completely condensing the entering steam (resulting 2.17 m against the 2.50 m of a horizontal tube).

A strong corroboration of the model proposed in this paper for the condensation inside a vertical tube came from the analysis of PERSEO experimental campaign results. The trend of HTC along the tube abscissa has been reconstructed for the condenser tubes provided with wall thermocouples. The results, in good accordance with a previous experimental work due to Kim et al., indicate an increasing trend going down along the tube, definitely confuting Shah model predictions. The correlations selected in the paper as regards the vertical tubes configuration, appear finally the most reliable, proposing themselves for a possible future implementation in the best-estimate thermalhydraulic codes, as RELAP.
REFERENCES


ABSTRACT

The Generation IV International Forum (GIF) member countries, identified the six most promising advanced reactor systems and related fuel cycle as well as the R&D necessary to develop these concepts for potential deployment. Among the promising reactor technologies for fast reactors (Sodium and Lead Fast Reactors) being considered by the GIF, the LFR has been identified as a technology with great potential to meet the needs for both remote sites and central power stations.

Ansaldo Nucleare, with its past experience on fast reactors, is promoting research and development of a pure Lead cooled fast reactor as the coordinator of the ELSY project (European Lead SYstem) funded by the European Commission in the frame of the sixth framework program. Activities are being carried out by Ansaldo Nucleare also on waste transmutation as part of another sixth framework program funded project: IP-EUROTRANS. The project aims to the conceptual design of an European Facility for Industrial Transmutation (EFIT), and the detailed design of a prototype of a smaller eXperimental Transmutation in an Accelerator Driven System (XT-ADS), an irradiation facility to be constructed in the short term for demonstration of key features of the larger EFIT.

The paper presents a summary of the two projects, with the description of the main components of ELSY reactor and EFIT and XT-ADS facilities, with a particular reference to the parts of projects developed by Ansaldo Nucleare.

1 INTRODUCTION

Concerns over energy resource availability, climate change, air quality, and energy security suggest an important role for nuclear power in future energy supplies. While the current Generation II and III nuclear power plant designs provide an economically and publicly acceptable electricity supply in many markets, further advances in nuclear energy system design can broaden the opportunities for the use of nuclear energy (Figure 1). To explore these opportunities, the U.S. Department of Energy’s Office (DOE) of Nuclear Energy has engaged governments, industry, and the research community worldwide in a wide-ranging discussion on the development of next-generation nuclear energy systems known as “Generation IV.”

In January 2000, the Generation IV International Forum (GIF) was established to investigate innovative nuclear energy system concepts for meeting future energy challenges. GIF members included Argentina, Brazil, Canada, Euratom, France, Japan, South Africa,
South Korea, Switzerland, United Kingdom, and United States, with the OECD-Nuclear Energy Agency and the International Atomic Energy Agency as permanent observers. China and Russia signed the GIF Charter on 15 November 2006, bringing GIF membership to thirteen. The forum serves to coordinate international research and development on promising new nuclear energy systems for meeting future energy challenges.

![Figure 1: Evolution of nuclear energy systems](image)

In 2001, the GIF agreed to proceed with the development of a Technology Roadmap [1] for Generation IV nuclear energy systems. The purpose of the roadmap was to identify the most promising nuclear energy systems (consisting of both a reactor and fuel cycle) for meeting the challenges of safety, economics, waste, and proliferation resistance.

The Generation IV roadmap process culminated in the selection of six most promising Generation IV systems: Gas-Cooled Fast Reactor System (GFR), Lead-Cooled Fast Reactor System (LFR), Molten Salt Reactor System (MSR), Sodium-Cooled Fast Reactor System (SFR), Supercritical-Water-Cooled Reactor System (SCWR), Very-High-Temperature Reactor System (VHTR).

Among the above promising reactor technologies, the Lead Fast Reactor (LFR) has been identified as a technology with great potential to meet needs for both remote sites and central power stations. The LFR system features a fast-neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium and management of actinides. A full actinide recycle fuel cycle with central or regional fuel cycle facilities is envisioned. The LFR can also be used as a burner of actinides from spent fuel by using inert matrix fuel. A burner/breeder could use thorium matrices. The system considered by GIF would use either lead or Lead/Bismuth Eutectic (LBE) as the liquid-metal coolant for the reactor.

A major step in favour of the LFR, occurred when EURATOM decided to fund ELSY [2] (the acronym for the European Lead cooled System) - a Specific Targeted Research Project of the 6th European Framework Program (FP6) – proposed to investigate the economical feasibility of a lead-cooled, critical reactor of 600 MWe power for nuclear waste transmutation. The ELSY project, scheduled to last three years, aims at demonstrating the possibility to design a competitive and safe Lead-cooled fast power reactor using simple engineered features. The use of compact, in-vessel steam generators and a simple primary circuit with all internals possibly being removable are among the reactor features needed for competitive electric energy generation and long-term protection of investment.

Ansaldo Nucleare has gained considerable expertise on heavy liquid metal coolants (initially on LBE and, currently, also on Lead) in the frame of R&D activities on
transmutation of Long Lived radioactive waste (Minor Actinides - MA - as well as some Long Lived Fission Products -LLFP).

Following a preliminary design developed in 1998, which was based on the Energy Amplifier concept proposed by CERN [3], a first configuration of a Lead-Bismuth Eutectic cooled Experimental ADS (LBE-XADS) [4] was worked out in the period 1999-2001 by a group of Italian organizations led by Ansaldo Nucleare, with the aim of assessing the feasibility of a small-sized (80 MWth) ADS.

The activity continued within the 5th Framework Program of the European Commission, in the context of the research on Fission Reactors Safety, that funded a project named PDS-XADS (Preliminary Design Studies of an Experimental Accelerator Driven System) [5] with a three-year contract (2002-2004) involving the participation of 25 European partners (industries, research organizations and Universities). The PDS-XADS Project was the first major step of a joint European effort that, as a key-milestone, did allow the detailed design of an XADS for demonstration of the transmutation technology by means of a subcritical reactor.

In the frame of the project IP EUROTRANS [6] of the 6th FP of the EU, 51 European Organizations have the strategic R&D objective to pursue forward an ETD in a step-wise manner. The aim of the 4-year (from April 2005 to March 2009) IP-EUROTRANS project is twofold: develop the conceptual design of an European Facility for Industrial Transmutation (EFIT) [7] with a pure lead-cooled reactor; carry out the detailed design of the smaller eXperimental Transmutation in an ADS (XT-ADS) [8] as irradiation facility and for demonstration of key features of EFIT.

In the following sections, more details on ELSY and EUROTRANS Projects are reported and the description of reference configuration of ELSY reactor and of EFIT and XT-ADS facilities is provided. Particular relevance is given to the parts of projects developed by Ansaldo Nucleare.

2 THE ELSY PROJECT

The goal of the Specific Targeted Research Project "European Lead-cooled System (ELSY)" performed in the frame of the 6th Framework Programme (FP) is to investigate the potential of the lead-cooled fast critical reactor to transmute nuclear waste, assessing its advantages and disadvantages, and assess the critical scientific issues and the technical feasibility of the lead-cooled fourth generation reactor system and the fuel cycle.

The project is structured to confirm the compliance with the main GEN IV requirements. Beside the goals of sustainability, safety and Proliferation Resistance and Physical Protection, emphasis is given to economics in order to always keep the capital cost under control.

Ansaldo Nucleare is the project coordinator, it has in charge the design of the main components/systems (such as the Reactor Vessel, the Steam Generators, the Primary Pumps, the Secondary System and the Decay Heat Removal System). Moreover, Ansaldo Nucleare contributes to the Design objectives, to the cost estimates, to the future R&D needs and to the compliance with the waste transmutation in a critical reactor and with the more general goals of Generation IV, to the system integration (Reactor Assembly configuration, Plant Layout, and the Application of Digital Engineering Technology) and to the Safety and to the transient analysis.

2.1 ELSY preliminary configuration

The configuration of the primary system is pool-type, similar to the design adopted for most sodium-cooled reactors and of the XADS design. The pool design has important
beneficial features, verified by the experience of design and operation of sodium-cooled fast reactors. These include a simple low temperature boundary containing all primary coolant, the large thermal capacity of the coolant in the primary vessel, a minimum of components and structures operating at the core outlet temperature.

The primary coolant is pure lead, characterized by good nuclear properties and no fast chemical reaction with water and air, instead of Lead-Bismuth Eutectics (LBE). Since lead is much more abundant (and less expensive) than bismuth, in case of deployment of a large number of reactors, pure lead as coolant offers enhanced sustainability. Furthermore, the use of lead strongly reduces the production of the highly radioactive, and hence decay-heat generating polonium in the coolant with respect to LBE.

The ELSY power plant is sized at 600 MWe because only plants of the order of several hundreds MWe are expected to be economically affordable on the existing, well-interconnected grids of Europe.

The choice of a large reactor power suggests the use of forced circulation to shorten the reactor vessel, thereby avoiding excessive coolant mass and alleviating mechanical loads on the reactor vessel. Thanks to the favorable neutronic characteristics of lead, the fuel pins of a lead-cooled reactor, similarly to LWRs, can be spaced apart, resulting in a lower pressure drop across the core. As a consequence, in spite of the higher density of lead, the pump head can be kept low (on the order of one to two bars) with a reduced requirement for pumping power.

A possible primary-side thermal cycle of 400°C/480°C in lead, without an Intermediate Cooling System, offers reduced risk of steel creep and milder thermal transients, while providing the thermal efficiency above 40% with a supercritical Rankine steam cycle at 240 bar, 450°C. The reactor vessel is designed to operate at the cold temperature of 400°C, which would be a safe condition even if oxygen control in the melt is temporarily lost. All reactor internals will have to operate at higher temperatures, at which it is necessary to rely on oxygen control, whereas fuel cladding could be surface-treated (aluminization seems to be a promising) for a greater safety margin. An improved primary-side thermal cycle at higher core outlet temperature could be adopted in the longer term, as new materials become available.

According to the predicted low primary system pressure loss and the favorable thermodynamic characteristics of lead, decay heat can be removed with lead in natural circulation in the primary system. A simple system for decay heat removal is the Reactor Vessel Air Cooling System (RVACS), which consists basically of an annular pipe bundle of U-pipes arranged in the reactor pit with atmospheric air flowing pipe-side in natural circulation. RVACS is a passive system, but its use without other systems can only be considered for small-size reactors since the vessel outer surface is relatively large in comparison with the reactor power. In the case of ELSY, the RVACS performance is sufficient only in the long term (after about one month after shut down) and two additional systems (under conception) are needed and will be implemented according to the stringent safety requirements of redundancy and diversification. A Reactor Pit Cooling System (RPCS) is additionally included for use during in-service inspection of the reactor vessel.

Figure 2 shows the cylindrical inner vessel concept, a scheme evaluated as a starting point for the primary system design of ELSY. The steam generator (SG) and primary pump (PP) assembly, consisting of two SG Units (SGUs) and one PP arranged between the SGUs and casing, is an integral part of the primary loop, i.e. from PP suction to SGU exit. Hot lead is pumped into the pool above the PP and SGU and driven shell-side downwards through the SGU tube bundle into the cold pool. The free level of the hot pool inside the casing is higher than the free level of the cold pool outside that is higher, in turn, than the free level of the hot pool above the core enclosed by the inner vessel.
A free level difference of cold and hot collectors at normal operating condition of only 1-2 m is sufficient to feed the core, eliminating the complicated, pressurized core feed system typical of the pool-type, sodium-cooled reactors. Simplification of the internals will offer the possibility of removable in-vessel components, a provision for investment protection. Compactness of the reactor building is the result of reduced footprint and height. The reduced footprint is allowed by the elimination of the Intermediate Cooling System, the reduced elevation is the result of the forced circulation and of the design approach of reduced-height components.

### 2.2 Ansaldo contribution to ELSY Project

The following activities have been carried out by Ansaldo Nucleare in the frame of the preliminary design of ELSY reactor:

- **Proposed Decay Heat Removal System:**
  
  A high reliability Decay Heat Removal function is requested, since the complete function failure has to be practically eliminated. The PRA method (in future) will be used to confirm the overall architecture, today a more simple Line Of Defence (LOD) method is used: 2 strong and one medium LOD are requested to consider the event in the Residual Risk. On this basis, two Decay Heat Removal Systems are proposed, diverse, independent and characterized by high reliability.
  
  - **DHR N° 1:** 4 Water Loops-Dip Coolers in the Cold Collector of the Primary System (improvement of the concept proposed by the Elsy partner Del Fungo). The total Heat Removal capacity is equal to 24 MW, so that 3 loops are sufficient to fulfill the Decay Heat removal function.
  
  - **DHR N° 2:** 4 Isolation Condenser connected to the Steam and Feed-water Lines of the Secondary System (same concept of EFIT, see Section 3.2.3)
A preliminary functional sizing of primary pumps has been performed on the basis of an analytical approach and the resulting configuration has been verified with a CFD calculation using the CFX code (Figure 3).

The thermo-mechanical verification of the vessel has been performed.

3 THE EUROTRANS PROJECT

Among the prior research and development topics of EURATOM 6th Framework Programme is the management of high-level nuclear wastes. A reference approach to reduce the burden on a geological repository is to remove the long-term radiotoxic isotopes occurring in the nuclear waste (Partitioning) and to burn them in dedicated systems (Transmutation). A promising system for the transmutation of the long-lived radioactive isotopes, as Pu and Minor Actinides (i.e. Am, Np), is a sub-critical system driven by an accelerator (ADS).

The implementation of partitioning and transmutation of a large part of the high level nuclear wastes in Europe needs the demonstration of the feasibility of several installations at an “engineering” level. This is the general objective of the integrated project EUROTRANS (European Research Programme for the Transmutation of High Level Nuclear Waste in an Accelerator Driven System).

To demonstrate the technical feasibility of an ADS, a two step approach has been developed. This approach foresees the definition of an advanced design of an eXperimental facility, which has the aim to demonstrate the technical feasibility of Transmutation in an Accelerator Driven System (XT-ADS). In parallel a conceptual design of a modular European Facility for Industrial Transmutation (EFIT) realisation in the long-term. Both designs bear the same fundamental system characteristics in order to allow

Ansaldo Nucleare is the coordinator of the Work Package related to the development and assessment of XT-ADS and EFIT Design, it has in charge the design of the main components/systems (such as the Reactor Vessel, the Steam Generators, the Primary Pumps, the Decay Heat Removal System for both XT-ADS and EFIT, the Secondary System and the Target of EFIT) and to the system integration (Reactor Assembly configuration, Plant Layout of EFIT). Moreover, Ansaldo Nucleare contributes to the design objectives, to the cost estimates, to the future R&D needs, and to the safety and transient analysis.
3.1 EFIT Primary System

The configuration of the primary system is pool-type, similar to the design solution adopted for most sodium-cooled reactors and of the previous XADS design. The pool concept allows to contain all the primary coolant within the Reactor Vessel, thus eliminating all problems related to out-of-vessel transport of the primary coolant. The pool design has important beneficial features, including a simple low temperature boundary containing all primary coolant, the large thermal capacity of the coolant in the primary vessel, a minimum of components and structures operating at the core outlet temperature. The primary coolant is molten lead, which is characterized by higher melting point than LBE or sodium.

The operating temperatures are: 400°C at core inlet (to have sufficient margin from the risk of lead freezing) and 480°C at core outlet. The core outlet temperature is chosen taking into account, on the one hand, the corrosion risk of structures in molten lead environment that increases with temperature (the current limit for candidate structural materials protected from corrosion by oxygen dissolved in the melt is about 500°C) and, on the other hand, considering that the outlet temperature cannot be too low because the associated increase of coolant flow rate would bring about unacceptable erosion of the structures. The proposed operating temperatures are, hence, a compromise between corrosion/erosion protection and performance.

The primary circuit is designed for effective natural circulation, i.e. relatively low pressure losses and driving force brought about by the core mid plane elevation arranged wide below the mid plane of the steam generators or, in case of emergency decay heat removal, the mid plane of the DHR dip coolers.

The speed of the primary coolant is kept low by design (less than 2 m/s), in order to limit the erosion. Wherever this cannot be complied with, e.g. at the tip of the propellers blades, the relative speed is kept lower than 10 m/s and appropriate construction materials are selected for qualification (among them, the machinable Ti₃SiC₂).

Protection of structural steel against corrosion is ensured, in general, by controlled activity of oxygen dissolved in the melt and additional coating for the hotter structures. Wherever stagnation of the primary coolant is predicted, e.g. within dummies, provisions ensure a minimum coolant flow.

The EFIT primary system (Figure 4) is described in the following sections, mainly referring to the systems designed by Ansaldo.

3.1.1 Reactor Vessel and Internal Structure

The Reactor Vessel is a welded structure without nozzles, made of a cylindrical shell with hemispherical bottom head and top Y-piece, both branches of which terminate with a flange. The conical, outer branch is flanged to, and hangs from, the Annular Structure anchored to the civil structure of the Reactor Cavity, whereas the inner branch supports the reactor roof. A cylindrical structure, welded inside on the vessel bottom head, is a radial restraint for inner vessel and core. All welds can be accessed for in-service inspection by means of remotely-operated vehicles.

The Internal Structure is located inside the Reactor Vessel and hangs from the Fixed Plate of the Reactor Roof. The functional parts welded together are the following:

- **Bottom Support**: The Internal Structure ends at the bottom with a short cylindrical support, which engages with the support welded to the Reactor Vessel Bottom during installation and maintenance operations. In normal operation no connection between Internal Structure and Reactor Vessel exists.
Diagrid: It is a very thick perforated plate with a large hole on its axis to centre the bottom of the target unit. The holes allow positioning all the fuel assemblies, and they are shaped to receive the subassembly spike with a little clearance. The diagrid holes have a conical shape to facilitate the assembly spike insertion, and to accommodate the assembly transition cone.

Inner Vessel with upper core restraint plate: it has the following main functions: separates the Core Region and the above Core Volume from the Downcomer, houses the Core, the Above Core Structure and the In Vessel Fuel Handling System, limits the Core Assemblies oscillations due to seismic accelerations. The Inner Vessel is welded to the upper side of the diagrid. It ends with a flange by which it is connected to the fixed plate of the Reactor Roof. The Inner Vessel delimitates two distinct plena: a lower plenum between diagrid and core restraint plate (and assembly heads), where the main primary coolant flow path takes place and an upper plenum of almost stagnant primary coolant, formed by an upstanding of larger cross section than the lower part of the Inner Vessel, welded to the core restraint plate and connected to the Reactor Roof by means of a flange.

Elbow connections: they are welded on the upper part of the cylindrical inner vessel and allow the connection with the suction pipe of the primary pumps that are engaged in the piston seal.
3.1.2 Steam Generator and Primary Pump Sub-Assembly

The SG-PP_SA (Steam Generator and Primary Pump Sub-Assembly), is an integral part of the primary loop, i.e. from pump suction to steam generator outlet. It is made of two Steam Generator Units (SGU) and one Primary Pumps (PP) arranged between the SGU, all included in a casing supported by, and hung from, the reactor vessel roof. It is designed to carry out both functions of hot coolant circulation and power heat transfer.

The casing cross section is arranged in the annular space between cylindrical inner vessel and reactor vessel. It is immersed in the cold pool. The only connection with the reactor internals is by the suction pipe of the PP that is engaged in the piston seal at the upper end of the elbow welded to the inner vessel. Thus, the whole sub-assembly can be easily put in and out of the reactor vessel with relatively short handling time.

The lead coolant flow path is illustrated by arrows in Figure 4. The hot lead is pumped into the enclosed pool above the PP and SGUs and driven shell-side downwards across the SGU helical-tube bundles into the cold pool. The Steam Generator Unit is a contra-flow heat exchanger, whose size is 52 MW, giving eight units per station to achieve the nominal Thermal Power of EFIT (=416 MW).

The SGU is a vertical unit with an inner and an outer shell. The primary coolant flows downwards shell-side through the inner shell and the annulus between inner and outer shell. The tube bundle is made of U-tubes, the inlet legs of which are straight inside the inner shell. After the U-turn, the outlet tube legs are helical inside the annulus and become straight again at the exit of the annulus.

The total length of tubes is less than the current maximum length of commercially available tubes (=28 m). Therefore the tube bundle can be fabricated without welds in the tubes. The tube ID of 14.2 mm allows the visual inspection of the inner surface.

The Primary Pump is to operate at 480°C and to supply the coolant at a rate of 8625 kg/s. Its most important part is the impeller, the circumferential speed of which reaches 9 m/s. The speed of flowing lead relative to the rotating propeller blades shall be limited by design to about 10 m/s as order of magnitude. Such operating conditions can give rise to problems associated with erosion-corrosion and cavitation wear as well as with the high stresses in the working parts of the pump. These problems are to a certain extent obviated by an engineering solution found in the development of the pump design. It reduces the stresses involved and ensures cavitation-free operation of the pump, which allows drawing on the operational experience of facilities with lead under erosion conditions in selection of materials.

3.1.3 The Target Unit

The coupling of the accelerator to the sub-critical core takes place via the Target Unit. This has been designed as a removable component, because its service life is anticipated to be shorter than the reactor lifetime, owing to the intense irradiation and local high thermal stresses.

The Target Unit is a slim component of cylindrical form, positioned co-axially with the Reactor Vessel and hung from the Reactor Roof. Because it serves also as inner radial restraint of the core, the outline of its outer shell fits the inner outline of the core. Its main component parts are the Proton Beam Pipe, the Heat Exchanger and the two axial-flow pumps arranged in series in the vertical legs of the loop upstream and downstream the horizontal target region.

3.1.4 In-vessel Fuel Handling System

The In Vessel Fuel Handling System provides the means to transfer the absorber assemblies from their storage positions in the outer rows of the Diagrid (Reactor in operating...
conditions) to the inner rows location (Reactor in shutdown conditions), and to transfer the core assemblies to and from all in-vessel positions.

Functionally, this system consist of:
- the Rotating Plug
- the extendable Above Core Structure
- the Transfer Machine (pantograph type)
- the Rotor Lift Machine

Access to any core position is achieved by rotation of both Rotating Plug and Transfer Machine.

### 3.2 EFIT Decay Heat Removal System

Three systems contribute to the DHR function in EFIT: the non safety-grade water/steam system, the safety–related DHR N1 (Direct Reactor Cooling - DRC) system and the DHR N2 system (Isolation Condenser). Following reactor shutdown, the non safety-grade water/steam system is used for the normal decay heat removal. In case of unavailability of the water/steam system, the DRC system is called upon.

The DRC System is composed of four identical loops (3 loops out of 4 are sufficient to perform the DHR intended functions), one of which is shown schematically in Figure 5. The main components of the loop are: a molten lead-diathermic oil heat exchanger (dip cooler, DHX); an air-diathermic oil vapor condenser (AVC) with stack chimney.

![Figure 5: EFIT Direct Reactor Cooling System](image)

The dip coolers DHX are placed in the reactor vessel in the upper part of the cold collector, where the primary coolant presents a mild thermal gradient, owing to convection streams brought about by the heat losses from the hotter Internals. Both oil vapors separator
and condensed oil drum in the lower part of the AVC are half filled of oil and half of a mixture of oil vapors and nitrogen, while tube bundle, upper header and nitrogen header are filled with nitrogen at the temperature of the atmospheric air. Since the oil in the DHX is prevented from vaporizing at temperatures lower than 400°C by the superimposed pressure and the temperature difference between oil and lead in the DHX is a few K, oil is kept circulating at the flow rate brought about by the small density difference between hot and cold leg. Vaporization take place because of increasing lead temperature due to the heat losses from the hot internals. Condensation of oil vapor takes place in a short portion of the tube bundle entrance of the AVC because the heat losses are a few hundred kW. As a consequence, the DRC loops are almost idle during normal operating conditions.

In case of unavailability of the water/steam system, the DRC system is called upon to passively enhance its performance and remove decay heat to specification. At start of the emergency condition, lead enters the DHX at higher temperature and flow rate, driven by the larger density difference between the cold shell-side leg and the hot outside leg, and oil starts to vaporize massively, speeded to circulate at higher flow rate by the large hydrostatic head between vaporizing and return legs of the loop. A recirculation ratio of more than four times the once-through vapor rate is achieved, having installed the oil vapor separator and condensate drum at the appropriate level above the DHX. The recirculation ratio is defined as the flow rate of liquid leaving the DHX compared with the vapor rate alone.

The vapor requires, to be condensed at the nominal rate, the whole AVC surface. In fact, owing to the increasing pressure, vapor floods the finned tubes while displacing the nitrogen gas. Vapor condensation takes place on the tube inner walls, and condensate runs down by gravity into the lower header and condensate drum, and mixes up in the separator with the oil rising from the DHX dip cooler. The sub-cooled oil returns the DHX via the cold leg, closing thereby the natural recirculation loop of the oil. The displaced nitrogen gas enters the extra tubes placed in front of the tube bundle, and hence cooled by fresh air, rises up, being lighter than the oil vapor, and eventually, at steady state, is confined in the nitrogen header connected to the upper ends of the tubes. Any vapor entrained by nitrogen or evaporating from the oil free surface below, would condense and fall before reaching the header. The system pressure has increased, at steady state, because of the added mass of vapor, but the pressure increase has been kept limited by the large-volume header, and, hence, also the oil boiling point has remained almost unchanged. Thermal expansion of nitrogen, that would further increase system pressure and the oil boiling point, an occurrence that would affect the thermal stability of the oil and is therefore undesirable, does not occur, because the nitrogen header is kept at the temperature of the ambient air being hydraulically connected, but thermally isolated from the loop.

3.2.1 Isolation condenser

In 1992 Ansaldo Nucleare designed the so called “Isolation Condenser” as part of the cooperation for the development of the SBWR design. Recently, GE used the component developed by Ansaldo for the ESBWR design. Ansaldo Nucleare successfully proposed the same type of arrangement for the IRIS Westinghouse reactor. The same arrangement can be proposed for EFIT, taking advantage of the fact that the condenser has been already tested in Italy by SIET (Ennea) at full scale SBWR conditions.

The inlet piping of the system is attached to the main steam line. Steam is then routed to a vertical condenser immersed in a pool and the outlet of the condenser is connected to the Vessel. In normal operation the isolation valve below the condenser is closed, the condenser is full of water and no heat exchange takes place. To put in operation the component, the feed
water line and steam line must be isolated and the condenser isolation valve opens injecting water in the vessel and starting the steam condensation process.

For EFIT, respect to the original configuration, the condenser has been down-sized. The sizing is such that with three units in operation the DHR is able to remove the decay power.

A scheme of the EFIT Isolation Condenser is reported in Figure 6. From the MSL a 12” piping is routed to the upper collector of the condenser. Inclination of the inlet line provides the filling of the condenser during normal operation when the lower isolation valve is closed. A Hot Storage Tank is located below the isolation valve (the piping connection to the top of FW line is such to promote natural convection of the FW fluid to the tank). The storage tank is used to provide additional water mass in case of isolation valves leakage and to limit the thermal shock to SGs at DHR start-up. Safety Relief Valves located on MSL are used to limit to 150 bars the system pressure rise due to system isolation (MSV and FWV closure) and water injection into the SGs. Condenser pools have to be sized to provide the required design grace period (3 days)

Figure 6: Isolation condenser scheme

Preliminary simulations (performed with Relap5 code) indicate that the system can be successfully operated: with 4 DHRs operation and pump on the margin to lead solidification is of one hour (at this point it is possible to rely on operator action to isolate one of the DHRs); with 3 DHRs in operation and pumps off, the maximum lead temperature reached at the SG inlet is below the design Lead Temperature at the SG inlet.

3.3 XT-ADS design

Already at the beginning of the EUROTRANS project, SCK•CEN offered to use the MYRRHA Draft II design file as a starting basis for the XT-ADS design. This allowed optimizing an existing design towards the needs of XT-ADS and within the limits of the safety requirements. Although XT-ADS has maintained the design scope of the former MYRRHA Draft II project, namely a multipurpose ADS for R&D activity and medical radioisotopes production, it has also the function of demonstration facility of important design elements of the larger EFIT ADS. XT-ADS differs from MYRRHA Draft II, therefore, on several topics for each field of the design and for the general configuration of the primary system.
Ansaldo Nucleare participated to the development of the present XT-ADS design, illustrated by Figure 7.

The configuration of the primary system is pool-type. The primary coolant is the lead-bismuth eutectic (LBE). Using the eutectic instead of pure lead allows the thermal cycle in the range of relatively low $T_{in} = 300^\circ C$ inlet and $T_{out} = 400^\circ C$ outlet temperatures, with the associated benefit of using known structural steels that can be employed without risk of large corrosion. The maximum velocity of the LBE coolant is also limited to 2 m/s in order to keep erosion under control.

Figure 7: XT-ADS reactor assembly – Vertical section

The XT-ADS heat transfer system, that ultimately dissipates the heat to the atmosphere, is made of the Primary Circuit and of two Secondary Circuits. Within the main vessel, the hot and cold pools are separated by a diaphragm, which is basically an inner tank, spanning the reactor vessel. Two mechanical axial-flow pumps and four conventional PHX (an evaporator shaped similar to the classical straight-tube IHX of the sodium-cooled reactors) penetrate the diaphragm wall.

Controlled dissolved oxygen activity in the LBE melt is the technique employed to prevent corrosion/erosion. The activity has to be controlled at the level that promotes the formation and self-healing of a stable, compact oxide layer on the surface of the structural steels in order to prevent the excessive corrosion and the penetration of lead into the fuel cladding and other structures materials.

In the primary loop the pumps draw the LBE from the PHXs bottom window and pass it to the mass of the cold plenum and provide the head required for the LBE to flow through the core, where it is heated up again. The pump flow does not pass entirely through the core, so that a significant core by-pass flow is present.
When leaving the core, the LBE mixes with the mass of the hot plenum and with the colder by pass flow (the temperature drop from the core exit to the PHX’s at steady state, due to thermal exchange between the hot and cold plenum, has temporarily not been taken into account, because predictably of a few degrees). The LBE flows shell-side through the four PHX’s and exits into the cold plenum.

Each of the two secondary circuits is made of two PHXs serving as the evaporators, a steam separator of the water/steam mixture rising from the PHXs, a steam condenser, interconnecting piping, and a stack chimney.

In each circuit, saturated water enters tube-side the two PHXs, operating in parallel, and leaves as a steam-water mixture. Ferruling is provided at tube inlet in order to stabilize the flow. Steam rising from the Separator is condensed into the AC condenser, from which the condensate falls into the separator where it adds to the water plenum and feeds the PHXs, thereby closing the water circuit.

The water flows in natural circulation in the loop owing to the hydrostatic head provided by the density difference between saturated water in the inlet pipe and the steam/water mixture in the outlet pipe.

The driving force for the natural circulation is provided by the hydrostatic head difference between the hot leg and the cold leg of the circuit, achieved by arranging the separator at a sufficient elevation with respect to the PHXs.

The Reactor Vessel is a welded structure without nozzles, made of a cylindrical shell with elliptical bottom head. It is closed by the reactor roof. The top end is an Y-piece, both branches of which terminate with a flange. The conical, outer branch is flanged to, and hangs from, the Annular Support anchored to the civil structure of the Reactor Cavity, whereas the inner branch supports the reactor roof.

The outer Safety Vessel has the same general shape as the reactor vessel. It is hung from the top to the support structure. The connection between the safety vessel and the supporting structure is obtained by a bolted flange. This vessel is the inner wall of an annular inter-space, surrounding the reactor. The purpose of the inter-space is to collect the reactor primary coolant in case of reactor vessel rupture.

The Inner Vessel separates the cold lower plenum from the hot upper plenum of the primary coolant. It has a flat bottom since this will simplify fabrication, while offer more efficient penetrations (PHX, PP, spallation loop, fuel manipulator, in-vessel inspection manipulators, core barrel, and LBE conditioning system) reducing the diaphragm by-pass flows. The diaphragm is supported by the reactor vessel. It serves as the support structure for the two in-vessel suspended fuel storage racks and the in-vessel recovery manipulators.

The four Primary Heat Exchangers (PHX) are grouped in two subsystems of reactor block that include two PHX and one PP each. The PHX is a one-tube-pass, vertical natural-circulation evaporator connected with a disengaging drum elevated above the evaporator. The main features are shell-side LBE flow, straight-tube bundle, and the upper tube plate immersed in the hot pool.

The pumps, installed vertically with fluid fed from the top, draw the LBE from the hot plenum at about the level of the PHX’s outlet window and pass it to the mass of the cold LBE providing the static head required to feed the core. Their most important part is the impeller, the circumferential speed of which is limited to 10 m/s. The speed of flowing lead relative to the rotating propeller blades shall be limited by design. The pump is specified to operate at the temperature of the cold plenum and to provide the total static head, that makes up most of the driving force of the natural circulation of the primary loop, at steady-state and during transients, when the temperature of the hot plenum changes with the core power. After looking at a pump coverage chart available in handbooks, based on normal ranges of operation of commercially available pumps, the type that meets the head-capacity operating
point of Table B.1 is the axial flow pump, the same type of the EFIT primary pump. This pump is a high-capacity, low-head pump, normally designed for flows in excess of 450 m³/h against heads of 15 m or less. The pump sizing performed by Ansaldo resulted in an 3-vaned 480 mm O.D. axial impeller. The vane profile being chosen by engineering judgement as the 5-digits N.A.C.A 23012 profile, adjustments of hub ratio, the vane inlet and discharge angles have been combined and simulated iteratively by means of CFD runs in order to make the design point the best efficiency point.

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ABSTRACT

As there is a need to deal with the problem of radioactive waste of nuclear plants, prototypes of subcritical reactors are being developed at the moment. They are known as ADS and they have as their main goal, the actinides transmutation. The neutron source, which keeps the reactions in a subcritical reactor, is very intense, that is why a high energy proton accelerator is required. For many applications, it is important to design methods to measure the intensity of the source. ADS are expected to work in a subcritical range between 0.92< $k_{\text{eff}}$ <0.97 and because of that, control rods have a limited use and many designers are considering not to include them. They can be substituted by reflector or combustible elements displacements. In order to calibrate in reactivity these displacements, one can either take advantage of the pulsing structure of the neutron source in microseconds time scale, or can design methods for the minute time scale wherein the source is constant, that is the objective we want to achieve here. If the reactivity control is made with the reflector, in between the extreme states of open reflector and closed reflector, both $k_{\text{eff}}$ are calculated and a reactivity ramp is estimated appropriate to the reflector movement velocity. This calibration should be made empirically by measuring the response to the ramp with neutron detectors. In the response to the ramp, the specific parameters are: the source intensity, the initial $k_{\text{eff}}$ and the ramp slope, which defines the objective to be reached in the calibration process. Measurements are available from experimental subcritical reactors in the operation range of the ADS, which allow getting ready this calibration method based on the reactivity ramps. However, in order to fit the measurements to a theoretical model, the point kinetics equations need to be solved for this case. With the Prompt-Jump approximation and the consideration of slow ramp compared to the time constants of delayed neutrons, the integration of these equations and the non-linear fit of the measurements to the resulting method are detailed. The static approximation is enough when the reactor is in a typical subcritical state. However, when close to a critical state, the reactivity is no longer a ramp but a parabola which lead us to estimate the neutron source by means of a non-linear fit of the experimental values and assuming a known reactivity initial value.

1 INTRODUCTION

For the sake of actinide transmutation, Accelerator Driven Systems (ADS) are actually under design [1],[2]. ADS are subcritical reactors operating in the approximate range 0.90 < $k_{\text{eff}}$<0.97, so they require an intense neutron source for maintaining the neutron chain. Such a neutron source is driven by a high energy proton accelerator [3]. In the microsecond time...
scale, the source is pulsed; but in the minute time scale, it can be considered as a constant intensity source [4].

Taking advantage of both time scales, a method for reactivity calibration is developed. It is based on a quasi-static change of reactivity. Firstly, a theoretical model is built by using the prompt jump approximation of the one group point kinetic equations; later, the model is tested in a subcritical experimental reactor by changing continuously the reactivity in the range that ADS are expected to operate.

2 THEORETICAL DEVELOPMENT

The operation range of a transmutator is between $-14$ and $-4$, (referred to $^{235}$U); it is a very subcritical state wherein the prompt jump approximation is valid for solving the kinetics equations. The spallation neutron source comes from an accelerator pulsed at a high frequency (above 1 KHz), so that for times below 1 ms, the source is not constant, but when the characteristic unit time is the minute, the source can be considered to be constant, in the sense that a neutron source would always indicate the same counts per second if reactivity does not change.

A typical calibration process would consist in going from the initial state (for instance, open reflector) to the final state, closed reflector, at a very slow and constant velocity. If the initial and final subcritical $k_{\text{eff}}$ are estimated through a calculation code, the velocity can be fitted to an average ramp of 1 $$/\text{min}$. These ramps are slightly slower than the delayed neutrons whose characteristic time (inverse of the disintegration constant $\lambda$) is 12 seconds. From now on, we will name them as slow ramps.

If $k_{\text{eff}}(0)$ is calculated in the most subcritical state and the neutron population, $N_0$ is measured, then [5]:

$$\frac{N_0}{l} = \frac{S}{1 - k_{\text{eff}}(0)} \Rightarrow \frac{N_0}{\Lambda} = \frac{S}{-\rho} \quad (1)$$

where $l$ is the average neutron lifetime, $\Lambda$ is the average neutron generation time; $\rho$ the reactivity (negative) and $S$ the intensity of the neutron source. The source can be obtained from the right hand side of the equation (in counts per second). The reactivity is obtained at any subcritical configuration from the source and the measurement of the neutron population. If the calculation of $k_{\text{eff}}(0)$ is not accurate enough, then the reactivity can be measured through a pulse of the source in the microseconds time scale as it can be seen in Fig.1.
Figure 1: Response to the pulse for the initial reactivity measurement

It may be observed that the response to the pulse is only required for the initial reactivity and the calculation of the source, the rest of the calibration can be achieved with the instrumentation of the Plant.

For the case of a continuous reactivity insertion, expressed in dollars, the kinetic equation is:

\[
\frac{dN}{dt} = \frac{\lambda \rho(t) + \rho'}{1 - \rho(t)} + \frac{\lambda S \Lambda / \beta}{1 - \rho(t)}
\]  

(2)

where \( \beta \) is the delayed neutrons fraction and \( \lambda \) the time constant of one precursor group of delayed neutrons, and \( \rho' \) the reactivity time derivative. In the case of a ramp, this equation has an analytical solution; if the ramp is slow as well, \( \lambda \rho >> \rho' \), which simplifies the calculation considerably.

Writing the kinetics equation as:

\[
\frac{1}{\lambda} \frac{dN}{dt} = \frac{\rho N}{1 - \rho} + \frac{S \Lambda / \beta}{1 - \rho}
\]  

(3)

due to the slow ramp condition, the derivative term turns out negligible; therefore, it can be derived that, as a first approximation, the solution is the static equation:

\[
N_e(t) = \frac{S \Lambda / \beta}{-\rho(t)}
\]  

(4)

Considering that the derivative term is a perturbation, \( N(t) \) can be substituted for \( N_e(t) \) and \( N \) is obtained, leading to:

\[
N(t) = N_e(t) \left[ 1 + \frac{1 - \rho(t)}{(-\rho)(\rho')} \right]
\]  

(5)
It may be observed that if the ramp is slow, the static approximation is correct until close to a critical state, that is, out of the operation range of transmutators.

3 EXPERIMENTAL VALIDATION

As the transmutators are still under design, the measurements were taken from the literature. In the CORAL-I reactor, with $\beta = 0.0068$, the evolution of the neutron population is measured from an initial state with reactivity -13.5 $\Sigma$ to a final state with reactivity -0.5 $\Sigma$. The movement lasted 15 minutes, so that the average ramp was around 0.86 $\Sigma$/min, what can be considered as “slow ramp”. The term $S/\beta = (8.6 \pm 0.2) \times 10^3$ (c/s)$\Sigma^{-1}$ was measured as well. The evolution of the neutron population $N(t)$ is represented in Fig.2.

![Figure 2](image)

Figure 2: Example of response to the ramp of a subcritical reactor

At first, it can not be stated that the inserted reactivity is exactly a ramp, to verify this is enough representing the inverse of the static approximation respect to time; if a negative slope straight line is obtained, the hypothesis of the ramp is correct. In Fig.3, $1/N$ versus time is represented:

![Figure 3](image)

Figure 3: The dotted line is the experimental data, the continuous one, the fit to the static approximation.
In figure 3, it can be observed that at instant $t=10$ min the ramp approximation is correct, but from 10 min up to 14.5 min there is a clear change of slope. That is why a parabolic reactivity insertion is supposed, and the static approximation has been used to fit the data. The obtained fitting parabola is:

$$\rho(t) = -13.8 + 0.288t + 0.0365t^2 \quad (6)$$

Then, Fig.4 is obtained:

![Graph showing the inserted reactivity. Ramp between -13.4 $ and -8 $](image)

$$\text{Figure 4: Inserted reactivity. Ramp between -13.4$ and -8$}$$

It is convenient to take into account that the static approximation needs the value of the source to obtain the reactivity, or the initial reactivity value to obtain the source and the rest of the parameters. Besides, above subcritical values of -2 $ (t=14 \text{ min})$ the static approximation must be corrected.

It is necessary to go through a non-linear fit to obtain the source from the corrected solution of the kinetics equation. Such a fit must be done assuming that the reactivity follows a parabolic equation like the following:

$$\rho(t) = a_0 + a_1t + a_2t^2 \quad (7)$$

Where initial reactivity(-13.8 $) is supposed to be $a_0$. In this way, the solution $N(t)$ can be written as a non linear function (rational type) with three undetermined coefficients: the source value, $a_1$ and $a_2$. By fitting the neutron population experimental data used in Figure 3, we have obtained Figure 5.

The value obtained for the neutron source is $(8.6 \pm 2.5) \times 10^3 \left(c/s\right)S^{-1}$. This value is exact within an error margin of 30 %. (the exact determination of the source was made with an error of 3 %).
4 CONCLUSIONS

Data coming from a continuous reactivity insertion have been analysed in an experimental reactor at the typical subcritical conditions of the transmutators. In this range of reactivity and with a constant source, we have deduced and proved that the static approximation is enough to calibrate the movement in reactivity.

We have also derived the correction when the static approximation is not enough, as it happens in the subcritical states close to critical ones($\Delta R > -2$ $\%$). The reactivity is no longer a ramp and it is necessary to approximate it to a parabola.

A known initial reactivity value, which can be obtained from the response to a pulse of the spallation source, is required. The rest of the cases need to go through a non-linear fit.

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CONCEPT OF A FUTURE  
HIGH PRESSURE-BOILING WATER REACTOR, HP-BWR

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ABSTRACT

Some four hundred Boiling Water Reactors (BWR) and Pressurized Water Reactors (PWR) have been in operation for several decades. The presented concept, the High Pressure Boiling Water Reactor (HP-BWR) makes use of the operating experiences. HP-BWR combines the advantages and leaves out the disadvantages of the traditional BWRs and PWRs by taking in consideration the experiences gained during their operation. The best parts of the two traditional reactor types are used and the troublesome components are left out. HP-BWR major benefits;

1. Safety is improved;
   - Gravity operated control rods
   - Large space for the cross formed control rods between fuel boxes
   - Bottom of the reactor vessel without numerous control rod penetrations
   - All the pipe connections to the reactor vessel are well above the top of the reactor core
   - Core spray is not needed
   - Internal circulation pumps

2. Environment friendly;
   - Improved thermal efficiency, feeding the turbine with ~340 °C (15 MPa) steam instead of ~285 °C (7MPa)
   - Less warm water release to the recipient and less uranium consumption per produced kWh

3. Cost effective, simple;
   - Direct cycle, no need for complicated steam generators
   - Moisture separators and steam dryers outside the reactor vessel, inside the containment
   - Simple dry containment

1 INTRODUCTION

Now the time has come to move a step further and develop an improved type of power reactors. Common sense, public confidence and economic considerations demand that this new design should not be a big leap from the presently functioning devices; however it should be a significant improvement. Therefore it is important to avoid those parts of the older designs which have caused trouble in the past e.g. PWR steam generators, BWR perforated reactor vessel bottoms and instead rely only on a stable construction with proven components which served well in the past. The High Pressure – Boiling Water Reactor, HP-BWR (Figure 1) attains these goals, by partly using the PWR concept, i.e. the pressure vessel, the electromagnetic control rod operator, and partly the BWR concept, i.e. core internals, internal circulation pumps and steam and moisture separators. All the figures here are made by the combination of the CAD codes of existing BWRs and PWRs. The subject was introduced by the ENS as is given in the References
2 SAFETY IS IMPROVED

The control rods are gravity operated instead of being operated by an intricate hydraulic system. The gravity operated control rod system has served well in PWRs. The stems are introduced into the reactor vessel via the vessel head (Figure 2). The control rods themselves are in the form of a cross, as it is in the BWRs. This assures large space for the cross formed...
rods between the BWR type fuel boxes. Also the neutron measurement sounds are introduced via the reactor pressure vessel head the way it is used in BWRs.

Figure 2: Reactor vessel head and reactor internals

The bottom of the reactor vessel (Figure 3) now is without numerous control rod penetrations, a great advantage compared with the previous design.
All the pipe connections to the reactor vessel are well above the top of the reactor core. This means that a major pipe break will not empty the reactor vessel. Therefore core spray is not needed. In Sweden for example the core spray in all BWRs with internal pumps are disconnected after the approval of the safety authority. Detailed studied of this subject is available at the Swedish Radiation Safety Authority.

Internal circulation pumps are used to assure hydrodynamic stability. In this way the orifices at the fuel channel inlets are chosen so that the one phase pressure drop will dominate over the two phase pressure drop to avoid hydrodynamic oscillations. By utilizing natural circulation one could omit the circulation pumps. However the margin to avoid hydrodynamic oscillations may be reduced. This is an experience gained at several Boiling Water Reactors and a phenomenon studied at thermal hydraulic loops at research institutes universities and manufacturers. There is a wealth of literature on this subject which can easily be fined e.g. on the Web.
Compared to the traditional BWR the HP-BWR has further advantages, namely improved thermal efficiency due to the higher temperature and further improved inherent stability due to the increased negative power reactivity coefficient (see Figure 4). Table 1 below shows a comparison calculated - with the RELAP code - between a BWR and a HP-BWR. Using presently available PWR pressure vessel and presently available BWR fuel boxes the approximate power output would be some 1200 MWe

<table>
<thead>
<tr>
<th></th>
<th>BWR</th>
<th>HP-BWR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Feedwater temperature</td>
<td>486.6 K</td>
<td>486.6 K</td>
</tr>
<tr>
<td>Outlet void temperature</td>
<td>559 K</td>
<td>617.8 K</td>
</tr>
<tr>
<td>Pressure in the steam dome</td>
<td>7 MPa</td>
<td>15.5 MPa</td>
</tr>
<tr>
<td>Inlet temperature to the core</td>
<td>550.29 K</td>
<td>582.3 K</td>
</tr>
<tr>
<td>Inlet core quality</td>
<td>-3.909E-02</td>
<td>-0.254</td>
</tr>
<tr>
<td>Outlet quality from the core</td>
<td>0.128</td>
<td>0.323</td>
</tr>
<tr>
<td>Total Mass Flow Rate from the core</td>
<td>13634 [kg/s]</td>
<td>5955 [kg/s]</td>
</tr>
<tr>
<td>Total Mass Flow Rate in the steam lines</td>
<td>1795 [kg/s]</td>
<td>2026 [kg/s]</td>
</tr>
<tr>
<td>Total Mass Flow Rate through the pumps</td>
<td>13634 [kg/s]</td>
<td>5955 [kg/s]</td>
</tr>
<tr>
<td>Total Power Coefficient</td>
<td>-1.64e-4 [Δk/%]</td>
<td>-4.4e-4 [Δk/%]</td>
</tr>
</tbody>
</table>

Table 1: Comparison between BWR and HP-BWR calculated with the RELAP code
Figure 4: Long term stability without the use of any control system calculated with the MATLAB code
Inherently stable reactor

3 ENVIRONMENT FRIENDLY

Improved thermal efficiency is attained by feeding the turbine with ~340°C (15.5MPa) steam instead of ~286°C (7MPa). The Carnot cycle theoretical efficiency \((T_{\text{Hot}} - T_{\text{Cold}}) / T_{\text{Hot}}\) is for BWR ~46% and for HP-BWR ~51% at \(T_{\text{Cold}} = 300 \, ^\circ\text{K}\), i.e. an increase by a factor of 1.109. Assuming the same improvement ratio, today’s efficiency of ~33% would increase to ~37%. This demonstrates the advantage of the HP-BWR which utilizes the fuel more efficiently and releases less warm cooling water to the environment per produced kWh. There are several conventional thermal power plants with 15.5 MPa turbines. Though to use dry saturated water might need some development work.
4 COST EFFECTIVE, SIMPLE

The HP-BWR operates in direct cycle mode, with no need for complicated and expensive PWR steam generators and also instead the rather complicated BWR reactor pressure vessel bottom a simplified one is used. The main steam separators are inside the pressure vessel and secondary separators and dryers can be installed outside the reactor vessel, inside or outside the containment. The containment (Figure 5) is a simple dry containment which allows easy entrance and inspections and also minor repairs during operation.

Figure 5: HP-BWR in a dry containment
The flow scheme of the reactor is clear and simple. The straightforward flow scheme is shown in Figure 6.

![Flow scheme diagram](image)

**Figure 6: Flow scheme**

5 CONCLUSIONS

As a reactor inspector on leave from Sweden I participated in IAEA’s OSART and ASSET missions. Also due to my engagement at the International Electrotechnical Commission (ICE) I visited nuclear installations in Europe, Asia and America. This way I gained insight of the operational experiences of most reactor types. As a result now I can contribute to the advancement of nuclear energy independently. As I have no obligation to any vendor or reactor type I can suggest an optimal reactor construction which hopefully will lead to further detailed studies at some vendors, power companies, research institutes and universities, especially after this conference. In Sweden already some universities expressed interest to make further studies of this concept.
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ABSTRACT

In the present study it is shown that choosing the Pebble Bed (PB) concept for the VHTR is not only a very effective way to supply all the world energy needs, it is also one of the safest nuclear reactor concept. Depending on the fuel cycle chosen, it is possible to reduce significantly the transuranic radioactive waste (TRU) normally produced in light water reactors and thus further reduce the environmental concerns of long living FP. Due to the very high coolant temperature, in the vicinity of 1000°C, this reactor can provide energy to produce hydrogen for motor transportation, electricity at efficiencies close to 50%, and process heat for desalination and other industrial applications.

A conceptual 600MWt High Temperature Pebble Bed reactor is proposed, and its safety characteristics are analyzed by simulating various hypothetical accidents, using the DSNP (Dynamic Simulator for Nuclear Power-plants) simulation system. Analyzing various LOFA (Loss of Flow Accidents), LOCA (Loss of Coolant Accident), and load rejection accidents, it is shown that the maximum temperatures reached in the fuel will not result in significant release of radioactive fission products. In addition, the transients are extremely slow due to the core large heat capacity, and will proceed for hours and days rather than seconds and minutes as in LWRs. It is concluded from the present study that the VHTR can provide sustainable, safe and emission free energy for the next generations, thus complying with the proposed Generation IV reactor requirements.

1 INTRODUCTION

For the energy hungry world there is a single power plant solution which has the potential to supply most of the present and future energy needs, with almost zero pollution at high thermal efficiency. The Very High Temperature Reactor (VHTR) can produce Hydrogen for automotive needs to replace the polluting gas and oil; it can produce electricity at very high efficiency with almost no pollution, and provide clean process heat for the industry and the energy needed for desalination plants to provide fresh water.

In the present study it is shown that choosing the Pebble Bed concept for the VHTR is not only a very effective way to supply all the energy needs, it is also one of the safest nuclear reactor concepts[1]. Depending on the fuel cycle chosen, it is possible to reduce significantly the TRU waste[2] normally produced in light water reactors and thus further reducing the environmental concerns of long living FP. Appropriate design and fuel cycle optimization can enhance the burning of minor actinides. Past studies have shown that the fuel kernels used in HTGRs can be subjected to much higher BU[3] and consequently produce much lower waste.
An extensive experimental program is underway in the Osiris reactor (the SIROCCO program) \cite{4}, to test FP retention capability of newly developed HTR fuel under normal and accidental conditions. A conceptual 600MWt High Temperature Pebble Bed\cite{5} reactor is proposed, and its safety characteristics are analyzed by simulating various hypothetical accidents, using the DSNP\cite{11} simulation system.

2\hspace{1cm} VHTR GEN-IV REACTORS

Various Generation IV reactor concepts are being studied world-wide with the objective to develop inherently safe, pollution free, superior economics, sustainable and proliferation resistance energy supply for the 21\textsuperscript{st} century and beyond. The countries participating in the VHTR development project include: US, UK, Japan, Russia, China, France, South-Africa, Germany and Switzerland. Presently two major HTGR construction projects are underway. The GT-MHR\cite{12}, developed in Russia for burning military Pu, with the support of an international consortium (General Atomics and DOE (USA), Minatom (Russia), Framatome ANP (France) and Fuji Electric (Japan); and the PBMR\cite{13}, developed by ESKOM in South Africa, with the involvement of several European organisations (BNFL, FZJ, HTR-GmbH, NRG, AEA Technology etc.) and a utility (Exelon). China is also planning to extend its HTR-10 technology to build a ~200MW version funded partly by China’s largest electricity generator, Huaneng\cite{14}, and to start construction in 2009 in Rongcheng City.

Superior economics can be obtained by using very high temperatures; hence the various VHTR concepts are planned to have coolant exit temperatures from 900 to 1000°C. The various designs include the Fast Breeder Reactor (FBR)\cite{6} either liquid metal or gas cooled\cite{7}, the Lead or Lead-Bismuth-cooled\cite{8} Fast Reactor, the Molten Salt Reactor (MSR), and the gas cooled graphite moderated reactor using either prismatic fuel, or spherical fuel elements – the pebble bed reactor. The reactors using Helium as coolant can reach very high temperatures, particularly in combination with graphite. Lead and Molten salt reactors have boiling points above 1400ºC which enables them also work with coolant temperatures above 1000ºC if corrosion and other technical problems can be resolved. The safety merits of PB-VHTR concept is the subject of the present study. One of the major advantages of the PB-VHTR is that its construction can be based on proven technology. Several countries have\cite{9} or are operating pebble-bed reactors with exit temperatures of up to 950ºC.
Fig. 1 demonstrates the superior thermal efficiency of the HTGR compared to the presently operating light water reactors. The superior safety characteristics of the PB-VHTR are assured by four major factors. Large thermal inertia - very slow transients, single phase coolant – no phase change during power excursions, large negative feedback-mitigating any accident and limiting the temperature increase, and a very low power density.

3 CONCEPTUAL DESIGN OF THE PEBBLE BED HIGH TEMPERATURE REACTOR

A conceptual design of the proposed Pebble-bed High-temperature Gas-cooled power plant is shown in Fig. 2. The PB core heats up the He gas to 1000°C or above. The He pressure is 60 bars. The gas flows inside a concentric hot duct (preheating the incoming cold He gas) through a series of heat exchangers, in each giving up part of the heat. Optionally, the hot He is transported directly to the hydrogen production plant or the turbine.

The energy between 960°C and 540°C is used for hydrogen production in one of the chemical processes being developed for this purpose. The most probable candidate is the Sulphur-Iodine (S-I) process\(^{[10]}\). The energy between 540 and 350°C is used to drive the turbine-generator to produce electricity. This part of the energy can be driven either through a steam generator producing high pressure steam to drive a conventional steam turbine, or using the direct cycle to drive a gas turbine. Finally the energy between 350 to 250°C is used via appropriate heat exchangers as industrial process heat or to provide heat to a desalination plant.

The fuel used in the pebble bed core is made up of 6cm graphite spheres in which the TRISO fuel particles are
embedded as shown in Fig. 3. The TRISO fuel particle includes a UO₂ fuel kernel surrounded by four layers of protective coatings, including two pyrolytic carbon layers, a silicon carbide layer, and an additional buffer zone of PyC to improve the fission product retention capability of the fuel even at very high temperatures. The fuel enrichment is about 10%, but depends on the particular fuel cycle chosen. The particles are dispersed in a graphite matrix sphere of 5cm, which is then surrounded by a denser graphite shell of 0.5cm. The TRISO particles can achieve very high burn-up, and experimental irradiations have shown that up to 747 Gwd/t, no detrimental effects were observed.

4 THE COMPUTATIONAL MODEL

The DSNP[11] simulation package was used to study the proposed PB VHTR response to various accident conditions. A schematic description of the core radial cross section and its various enclosures are shown in Fig. 4. Each of the shown elements is represented by a two dimensional cylindrical DSNP module surrounded by a one-dimensional cylindrical axial flow path. The axial flow schematics are presented in Fig.5.

The core is represented by a cylindrical element with 6 axial and 6 radial subdivisions. Each axial segment can be of arbitrary length, in the present case equal segments were assumed, and the radial sections have equal flow areas. Each cylindrical segment contains an
average spherical fuel element, the coolant flow is from top to bottom. The radial reflector which is cooled by incoming cold He also isolates the steel pressure vessel from the hot He. From the outside the pressure vessel is cooled by natural air flow. The reactor is enclosed in a concrete silo which is water cooled on its inner side by a special liner. The silo is cooled by the ambient air.

In addition to the reactor cylindrical elements and the flow hydraulics, the various heat exchangers, the pipes and valves, the steam generator, turbine and other component models were included in this simulation. The core neutronics was represented by the kinetic equations and the various control and safety features were also included. The thermal-hydraulic model is based on the three conservation equations, namely the conservation of mass, energy and momentum as shown below. In addition, the neutron kinetic equations resulting from spatial and energy integration of the neutron balance equations is used.

The conservation of mass equation,

$$\frac{D\rho}{Dt} = - \rho (\nabla \cdot \vec{v})$$

The conservation of energy equation,

$$\rho \frac{Dh}{Dt} = - \nabla \cdot q''' + q''' + \frac{Dp}{Dt} + \Phi$$

The conservation of momentum equation,

$$\rho \frac{D\vec{v}}{Dt} = - \nabla p + \mu \nabla^2 \vec{v} + \rho \vec{f}$$

The exact approximation used depends on the module level and application

5 RESULTS OF THE SIMULATIONS

The hypothetical accidents investigated in the present study include loss of flow as a result of a breach in the main coolant duct, which also includes depressurization of the core and a loss of heat sink by failure of one of the thermal loads, the turbine, the hydrogen production plant or the plant using the process heat. Three specific cases are presented below, namely, depressurized loss of flow with scram, depressurised loss of flow without scram, and a partial loss of load.

5.1 Depressurised Loss of Flow with Scram

A small part of the results are reproduced in the figures below. Figures 6 to 8 show the loss of coolant accident due to a break in the cold duct. As the safety system detects a change in the pressure the reactor is shut down. The He transient temperatures in the core hottest region - the 6th axial core region are shown in Fig. 6 for the 6 consecutive radial segments, with the highest temperature in the core central region. As can be seen there is an initial fast temperature drop, following the power reduction, to the level of the coolant cold gas temperature. This is then followed by a very slow temperature increase due to the presence of the decay heat and the absence of sufficient heat removal capability, reaching a maximum value after about 5 days and then a very slow temperature decrease following the decay heat reduction curve. The transient calculation is terminated after 1.5x10⁶s (about 17days) at which
time some of the pebbles are still above 1000°C. Fig. 7 shows the 6 fuel spheres central temperatures for the same core regions. The maximum coolant temperature is 1360°C, while the fuel temperatures are a few degrees higher. It should be noted that the coolant temperatures are calculated for the section entrance and exit, while the fuel temperatures are computed for the centre of each axial and radial section under LOCA conditions. The heat is dissipated mainly by radiation and conduction.

Figure 6: Coolant transient exit temperatures in the 6 radial regions following a depressurized LOCA accident in a 600MW PB-VHTR.

Figure 7: Fuel sphere centre temperatures in the 6th axial region, in 6 radial regions following a depressurized LOCA accident in a 600MW PB-VHTR.
The side reflector temperatures for this transient are shown in Fig. 8 for the 6 axial segments of the innermost reflector region. In this study the reflector is modelled by 6 axial and 10 radial cylindrical shells using a two-dimensional conduction model. As expected, the temperature decrease and the following increase are much slower and the peak temperature is reached after about 6 days.

![Figure 8: Temperature distribution in the radial reflector innermost region.](image)

5.2 Depressurised Loss of Flow without Scram

Figures 9 to 12 show the same LOCA accident but a malfunction of the safety system, i.e. no scram is assumed. As in the previous case, Fig. 9 shows the coolant temperatures at the exit of the core last region along the 6 radial segments modelled. Comparing this figure to the previous case, fig 6, a very different transient behaviour can be observed. The temperatures start increasing following the flow reduction. The transient shape results from the changing nonlinear power-flow mismatch which is taking place during this event. The power is shutdown due to the negative reactivity coefficient of this core. The maximum temperature of the coolant is only about a 100°C higher than in the scrammed case.

Fig. 10 presents the fuel temperatures in the centre of the fuel spheres in the core last axial region along the 6 radial segments. Somewhat higher temperatures (about 100°C) than in the previous cases and a different transient behaviour can be observed. However, the maximum temperature is reached at about the same time as in the previous case.

Figure 11 shows the reactor power in MW on a logarithmic time scale. As can be observed, the power starts decreasing at about 100s, and decreases slowly to the decay heat levels. The power is reduced in this case due to the negative feedback reactivity inserted into the core by the increase in the fuel temperature. As can be observed, in all cases the power is reduced, either by the safety system or by the negative feedback, and no damage to the fuel will occur. Figure 12 shows again the reflector temperatures in the reflector innermost 6 axial segments. The reflector temperatures in this case are about a 100°C higher than in the previous case, but no significant differences between the scrammed and un-scrammed case can be observed.
Figure 9: Coolant transient exit temperatures in the 6 radial regions following a depressurized LOCA accident in a 600MW PB-VHTR with malfunction of the safety system.

Figure 10: Fuel sphere centre temperatures in the 6th axial region, in the 6 radial regions following a depressurized LOCA accident in a 600MW PB-VHTR with malfunction of the safety system.
Figure 11: Reactor total power in MWt following a depressurized LOCA accident in a 600MW PB-VHTR with assumed malfunction of the safety system.

Figure 12: Temperature distribution in the radial reflector innermost region following a depressurized LOCA accident in a 600MW PB-VHTR with assumed malfunction of the safety system.

5.3 Partial Loss of Load.

In this accident a part of the load is lost. As a result, a core heat up is initially expected. Such an event can occur if either the turbine, the hydrogen plant or the process heat utilization
malfunctions. The immediate consequence is an increase in the He temperature as seen in Fig. 13. As can be observed the coolant temperatures will undergo a transient during about 1000s, and afterward will stabilize on a new somewhat lower temperature level. The core negative feedback will adjust the new power level according to decreased demand as shown in Fig. 14.

The resulting fuel temperatures in the centre of 6 spheres located along the axis in the second core radial region are shown in Fig. 15. A mild transient in the fuel temperatures is observed until the system is adjusted to the new power and temperatures.
CONCLUSIONS

The major conclusions that can be deduced from the above accidents simulations is that the proposed VHTR is inherently safe and all major accidents will be mitigated by naturally occurring phenomena regardless of the availability of the safety system.

- None of the accidents resulted in core damage
- The reactor is shut down due to the existing negative feedback
- Decay heat can be easily removed by radiation and conduction to the ambient air
- No excessive temperatures are observed
- No fission products will be released in any of the modelled accidents

From the simulations of various hypothetical accidents of the PB-VHTR and on author experience in the safety analyses of different nuclear power plants, it is obvious that this reactor is safer than most presently operating reactor concepts. It can generate fuel, electricity and process heat to cover all energy needs by a single system. Further design studies are needed to resolve economic issues and technological concerns of the very high temperatures.
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ABSTRACT

Loss of coolant in a fast reactor is a crucial, safety-related issue that has two distinctive consequences on a reactor operation: first of all, heat removal ability is lost leading to overheating and possibly to loss of core integrity, standing for one of the main reasons for core melting accidents. Secondly, it can introduce reactivity into the system. Void effect profile for ELSY fast reactor has been evaluated by means of MCNP code, simulating the gradual voiding of the active region, while keeping the lead reflector around. As expected for liquid metal reactors, a positive void reactivity effect (about 5200 pcm) has been calculated; the value corresponds nearly to the amount of reactivity coverable by the twelve B$_4$C control rods designed for the whole core. This fact leads to an important consequence: in case of such unbelievable complete voiding of the core, no more reactivity worth of the absorbers would be available.

In order to investigate a possible reduction of positive void feedback in case of hypothetical loss-of-flow (LOF) scenario, an internal blanket of some proper absorbing material has been inserted in the mid-plane of each fuel rod with consequent increase of core height. MCNP criticality calculations have been performed for different blanket thicknesses and for three kinds of materials: natural uranium in oxide form, and natural thorium in both metallic and oxide forms.

Results have shown that none of the three materials is able to decrease the void effect for small thickness: more than 300 mm of thorium (and even more than 400 for uranium) are necessary to start reducing the positive reactivity insertion.

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1 INTRODUCTION

ELSY is a Generation IV power plant whose aim is the production of electricity in a competitive and safe design [1].

Many factors can cause perturbations to the reactor operation, due to feedback effects. Temperature of materials, fuel properties change during operation, pressure and occurrence of void are all aspects related to neutrons balance in the reactor, and they must be carefully estimated to assure effective compensating action in case of reactivity swings.

In particular, accidental void occurrence in the system is a delicate issue concerning safety; the main cause for this accidental scenario is a rupture in the fissile rods which leads to fission gas release with consequent formation of void. Void reactivity effect - a measure of how the nuclear system reacts to accidental void occurring in the core - needs then to be core-widely evaluated for the promising ELSY fast reactor, and solutions for its mitigation pursued.

2 VOIDING IN LIQUID METAL FAST REACTORS

Several causes may give rise to extensive voiding in a liquid-metal reactor. Usually, LMR plant designs are arranged with backup protection to mitigate the impact of vessel leakage or rupture, to the degree that large-scale loss-of-coolant accidents (LOCA) are extremely unlikely. Pool systems typically have a second guard vessel, and loop systems are normally double pipe and tank designs. Since the liquid-metal coolant is not pressurized under normal operation, a leak in the primary system will not automatically result in coolant boiling due to the depressurization (as in LWRs). In a sodium-cooled reactor, voiding may arise due to boiling out of coolant. This is prevented in a lead system. In order for the lead to get hot enough to boil (Tb=2023 K), temperatures have to be above the melting point of steel (Tm=1700 K). In that case, much larger antireactivity becomes available due to fuel floating.

Another possible mechanism for coolant voiding, without the precondition of steel melting, is the possibility of entrance of air into the core from the cover gas region as well as steam during a failure in the steam generator, i.e., a so-called steam generator tube rupture (SGTR) event. In sodium plants, intermediate sodium-loops are introduced as a second physical barrier to minimize the consequences of SGTRs and to avoid violent chemical reactions between water and sodium in the primary system. Because lead is chemically inert with water/steam, two-circuit designs are suggested, with the steam generators located in the primary system. In such designs, there will only be one barrier to fail in order to get high-pressure steam into the primary system; the pressure on the steam side can be as high as 100-150 bars and low pressure on the metal side, about 1 bar.

Because of multiple coolant entries preventing blockage in fuel subassemblies (wrapperless assemblies) and avoidance of gas entrance in lead, voiding of the core in ELSY plant is rather unbelievable. The fatal context which is realistically taken into consideration is the gas release from failed pins.

ACRONYMS

| ELSY | European Lead System |
| IMPB | Internal Mid-Plane Blanket |
| LFR  | Lead Fast Reactor |
| LMR  | Liquid Metal Reactor |
| LWR  | Light Water Reactor |
| MCNP | Monte Carlo N-Particle |
Void fraction, i.e. the fraction of void in a certain “total volume” which is normally filled with coolant, can be written as:

$$\alpha = \frac{\text{Void volume}}{\text{Total volume}} = \frac{V_v}{V}.$$  

**Void reactivity effect** estimates how much the reactivity changes as voids form in the coolant. It is defined as the relative change in reactivity per change in void fraction:

$$\alpha_v = \frac{1}{k_{\text{eff}}} \frac{\partial k_{\text{eff}}}{\partial \alpha}.$$  

The total coolant void worth is just the difference in the $k$-eigenvalue between the flooded and voided state. The effect can be calculated by considering the effect of changes of void fraction on the different contributions to $k_{\text{eff}}$.

The magnitude and sign of the reactivity effect due to void is a complex function of core design, void location and void volume. Inherent stability of such a nuclear reactor, as any dynamic system, can be achieved only by negative feedbacks acting sufficiently fast so that the integrity of the reactor core is not compromised.

The coolant density change or **occurrence of the void** in the core can be a result of the following events:
- temperature increase of the coolant due to pump failure, inadequate instrumentation and prediction of hot-channel factors, crud deposition and plugging of the coolant channels; ultimately, this leads to a coolant phase change. Accident scenarios assuming coolant boiling are rather unbelievable for systems cooled with lead due to its high boiling temperature.
- blocking of the coolant circulation as a consequence of coolant freezing in the steam generator, i.e. overcooling.

Additionally, void cavities can appear in the core due to:
- gas leakage from ruptured pins,
- steam ingress from ruptured steam generator,
- blow-up of bubbles from gas injection system,
- coolant leakage caused by brittle failure of the reactor vessel.

The coolant acts in the reactor not only as a neutron absorber, but also as moderator, affecting the neutron spectra. The reactivity change when decreasing the density of the coolant in fast neutron cores is mainly due to three effects:

- reduction of neutron moderation (spectral hardening), increasing fission probabilities of even neutron number actinides, reduction of absorptions in fuel (and consequent increase of $k_{\text{eff}}$);
- reduction of neutron parasitic absorption in coolant (consequent increase of $k_{\text{eff}}$);
- increase of neutron leakage (consequent decrease of $k_{\text{eff}}$).
3 ELSY VOID REACTIVITY EFFECT EVALUATION

According to what has been recently done at ENEA national research laboratories [2], ELSY reactor has been modelled following the square wrapperless design option (Figure 1) in the preliminary configuration defined for a 1530 MW thermal power.

The core is made of three fuel zones of different enrichment (13.4, 15 and 18.5% v.f. of Pu enrichment). Externally, suitable elements and lead surround the core as reflector and shielding. Under active fuel a plenum zone is designed for fission gases, while structural parts surround on the top and at the basis. Absorbers are of two kinds: twelve control rods with a volumetric fraction of 70% in boron, and an upper cloak with v.f. of 15% in boron, which has the main purpose of regulation and compensation.

Criticality calculations performed by MCNP with use of JEFF 3.1 nuclear data library have been run to obtain a first estimate of $k_{eff}$ for ELSY reactor.

Results show that the present configuration has $\rho = 193$ pcm, and this is taken as reference value (std. dev. = 64 pcm).

A method for calculating the liquid metal loss effect is to use perturbation theory and determine the contributions separately [3]. This method provides useful insight into the physical processes, but the calculation is quite difficult. Therefore the method generally used to calculate liquid metal reactivity loss is to perform successive calculations (one with coolant present and the second one without coolant from the zone of interest) and to compare the criticalities.

Void reactivity effect has been studied applying the void into the core for successive steps. The Monte Carlo input file has been modified to simulate the situation in which void penetrates progressively into the core: a void plenum region, which gradually replaces the...
coolant, has been added (Figure 2) as far as a complete void condition is reached. At each step the lead level in the core has been lowered of 10 cm each time to the end of the active region (90 cm).

![Figure 2: Scheme of the process of progressive voiding (left) and MCNP plot of the voided core (right).](image)

$k_{\text{eff}}$ and relative standard deviation have been reported at each step of voiding. As expected for liquid metal reactors, reactivity progressively increases to an amount of slightly more than 5000 pcm, which represents a large reactivity worth.

The profile of the resulted void reactivity effect has been plotted along with its differential with respect to the voided height (Figure 3).

Differential analysis gives information about local void: the maximum is located very close to the core mid-plane. Hence, the reactivity insertion rate is highest at core midlevel.

![Figure 3: Void Effect and derivative function.](image)

As shown in the graphs, reactivity worth due to void effect has resulted in a great swing which must be deeply analysed and controlled. Although this aspect is typical of liquid metal
fast reactors and cannot be avoided [4], many measures have recently been investigated for reducing it.

4 OPTIMIZATION OF IMPB (INTERNAL MID-PLANE BLANKET) FOR VOID EFFECT REDUCTION

It was shown that in reactor systems cooled by liquid metals in a configuration consisting of steel pins acting as absorbers immersed in the coolant, void worth is significantly lowered. Several attempts have been already made to reduce void effect by design; most design modifications have focused on increasing the leakage component. The most promising appears to be the heterogeneous core concept, in which blanket assemblies (containing pure fertile material) are distributed through the core region to achieve a high neutron leakage rate from the active core to the blankets [5]. There are radial and axial heterogeneous cores, according to the option of arranging the blanket fuel assemblies among (radial blanket) or within (internal blanket) the seed fuel assemblies [6]. Another option concerns the combination of both types. In addition, other axial blankets can be added on and beneath the seed fuel. This design reduces void reactivity effect, while yields higher breeding ratios but consequently requires higher fissile inventories.

In the present work the concept of adding a blanket acting as absorbing material in the core has been considered; three different material solutions have been tested. A blanket of varying thickness for each material has been inserted in the mid-plane core, and void worth has been evaluated.

Analyses have been carried out for natural uranium in dioxide form and for thorium in both metallic and oxide forms.

4.1 Mid-Plane Blanket Design (Internal Mid-Plane Blanket configuration)

Since lower void reactivity can be attained through the leakage of neutrons from the active region, this effect may be enhanced by shortening the height of the core. This option is obviously limited by reactor design and constraints. However, a great challenge is represented by the possibility of reducing void effect inserting a blanket of some proper material as depleted UO₂ (for its absorbing power) inside the core.

The coolant loss (in the region around the inserted blanket) has two effects:

- an increase of the core transparency, so that a higher blanket absorption rate will occur; considering that the ratio $\sigma_{\text{fission}}/\sigma_{\text{absorption}}$ of the blanket is lower than the one of the fissile fuel, this will act in the sense of reducing the reactivity;
- because of the lack of the coolant moderating contribution, neutrons are kept to higher energies where the ratio $\sigma_{\text{fission}}/\sigma_{\text{absorption}}$ of both the blanket and the fissile fuel is higher; this will act in the sense of increasing the reactivity.

Moreover, the spectral contribution to void reactivity effect (usually positive) is proportional to the flux intensity, and therefore is important near the centre of the core, while the leakage component (negative) becomes important near the edges where the flux gradient is stronger. As a result, expulsion of coolant from the central region results in a much more positive reactivity gain.

Figure 4: ELSY R-z view with IMPB.
Due to the flux importance in the central axial position in the core, the internal blanket has been placed in the mid-plane of each fuel rod. If this solution succeeds, and void reactivity worth gets lower, the burden on control rods in any accidental case in which void fills the core causing a fatal increase of criticality will substantially decrease; as a consequence, a great improvement in reactor safety would be gained.

Criticality calculations have been performed for the three selected materials and then compared. For the case of natural uranium, MCNP tests have been carried out for layers of different increasing thickness starting from 4 to 400 mm of material, increasing therefore the total active height of the core. Figure 5 shows the results of criticality calculations. Reactivity, before and after any void occurrence, is plotted in the curve referenced to the left scale, showing clearly that reactivity drops after insertion of a blanket material in the core. This fact indicates that adding such an amount of absorbing material leads necessary to a higher enrichment of the fresh fuel as to assure criticality at starting. In the curve referenced to the right scale, the reactivity variation (the difference between reactivity after voiding and reactivity before voiding) is plotted for increasing blanket thickness.

![Figure 5: Reactivity and Void Effect in the UO2 IMPB insertion.](image)

The same calculations have been made for thorium, and all the results, normalized to the configuration without blanket, have been plotted together (Figure 6), coherently with the expression:

\[
\Delta \rho = (\text{Void Worth})_{\text{blanket}} - (\text{Void Worth})_{\text{blanket free}}
\]

where:

\[
\text{Void Worth} = \rho_{\text{void}} - \rho_{\text{no void}}
\]
4.2 Results and flux analysis

As shown in Figure 5, the adding of an IMPB lowers consistently the criticality of the reactor. The contribution to void effect due to uranium insertion (blue curve in Figure 6) increases by about 1500 pcm with increasing thickness until 20 cm of material: this point represents the maximum positive variation to void effect for the present configuration. Then, void effect variation gets lower again, but actually uranium IMPB doesn’t reduce void effect unless an amount of more than the upper tested limit (40 cm) is added.

Thorium metal and oxide appear as better solutions: both the curves show a maximum before the one related to uranium addition, between 10 and 15 cm of thickness, that means less amount is required to get similar values. Moreover, void effect gets finally lower then the reference value as at least 30 cm of thorium oxide are added.

Creation and death of neutrons have been considered as discriminative aspects for explaining the reasons of the curves of Figure 6. In particular, MCNP output files have been analysed to get information about number of fissions, captures and escapes normalized with respect to one neutron history. Figure 7 show that production rate due to fission increases after voiding independently from the blanket thickness, whereas the sum of escape and capture rates related to the whole system
slightly decreases. Figure 8 compares singularly the various event types related to the active region for the three tested materials.

Figure 8: Contribution to rates probabilities referred to the active region due to the IMPB configuration.

Increase of fission rate and decrease of captures seem to be mainly responsible for reactivity gain after insertion of small uranium or thorium blankets; these variations in the rates become indeed less relevant with greater thicknesses.

The occurring of the maximum void reactivity worth at different blanket thicknesses (after 10 cm for thorium and after 20 for uranium) is probably due to a greater $\sigma_{\text{fission}}$ for uranium and a slightly greater $\sigma_{\text{capture}}$ at high energies for thorium (Figure 9).
Last, MCNP neutron flux distributions have been analysed. With regard to neutron flux in the blanket-free configuration before void occurrence, the curve has the expected cosine trend (Figure 10); after voiding, it shows a flattening due to the increase of the neutron path.

Figure 9: fission and capture cross-sections for U-238 and Th-232.

Figure 10: MCNP Neutron Axial Fluence of ELSY reactor.
All the fluxes show a reasonable symmetry; greater values of the flux are in the upper part of the core: the reflector function of the upper lead zone influences strongly the neutron distribution, increasing probability of avoiding neutron escapes. Thorium curve shows a greater flux depression in the blanket zone in comparison to uranium curve, due to the worse neutron multiplication characteristics.

Internal mid-plane blanket tends to give axial decoupling, that is spatial separation of the neutron flux, as clearly shown in the middle of Figure 11. Flux distributions in a decoupled core are very sensitive to perturbations, and local reactivity insertion tends to cause very high local power peaking or unstable transients. Decoupling effects are thought to be undesirable, as the stability under transient conditions is affected: this is a strong drawback for the IMPB configuration.

The decoupling effect justifies the void effect reduction for great blanket thicknesses: 40 cm of IMPB lead to a two-separated-cores configuration (Figure 12), where each core is
characterized by half the initial H/D ratio. In this configuration the increased leakage component, as shown from the flux slope at the edges, is fundamental in reducing void effect.

5 SUMMARY AND CONCLUSIONS

With reference to the European design, a preliminary void effect study of ELSY reactor has been carried out. Reactivity worth due to void effect in the active region has resulted in a great swing, equal to about 5200 pcm. Although large void effect is a typical feature of liquid metal fast reactors and cannot be annulled, many measures have been studied in the past in order to mitigate it; in the present work the concept of adding some fertile through three different material solutions (natural uranium in oxide form, and thorium in metallic and oxide forms) has been considered. A blanket of varying thickness for each material has been inserted in the core mid-plane and void worth evaluated. Results have shown that uranium blanket doesn’t reduce void reactivity effect unless great amounts are added, while thorium provides an actual improvement if at least 300 mm of blanket are inserted in the mid-plane of the core.

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ABSTRACT

The BORAX accident (Boiling Water Reactor Experiment) is the Mastered Severe Accident reference to be taken into account by the containment design of JHR. Two major stakes are concerned: the radiological consequences with respect to released activities and the mechanical consequences with verification of acceptable behaviour for the pool system. A reasonably conservative approach was developed in 2007-2008 to simulate the global phenomenon of this accident. A “BORAX calculation scheme” based on existing codes and specifically adapted or developed for the JHR application, was set up and will provide quantitative results.
1 INTRODUCTION

Beyond the four classical safety categories, the French safety methodology defines several situations integrating the Mastered Severe Accident (MSA) for which it is necessary to demonstrate that the consequences are controlled.

The BORAX accident (Boiling Water Reactor Experiment) is the MSA reference to be taken into account by the containment design of JHR [1]. Two major stakes are concerned: the radiological consequences with respect to released activities and the mechanical consequences with verification of acceptable behaviour for the pool system.

This conventional accident consists in a fast ejection of a control rod, leading to a nuclear power excursion and the fusion of a part of the nuclear fuel (actually aluminium alloy only). This is supposed to lead to a steam explosion generated by the violent interaction between liquid aluminium and the cooling water. This phenomenon consists of a complex and fast sequence of physical-chemical processes with different time and space scales.

This accident has been treated in the past for all French research reactors on the basis of a hypothetical thermal energy deposition close to 135 MJ and a mechanical efficiency of 9%. Consensual values for radionuclide transfer coefficients were chosen for the design phase of the reactor. Afterwards, experimental validations were performed, with real explosions on low scale mock ups to verify the safe behaviour of the second safety barrier.

The methodology proposed at the end of 2005 in the Preliminary Safety Report of the JHR consists in a more realistic approach. A thermal-dynamical approach of the accident scenario was particularly expected to improve our understanding of phenomenon.

A multi-disciplinary workshop was set up in mid-2007 with the goal of identifying and modelling the different phenomena involved in the Borax sequence. It led to the setting-up of a BORAX calculation scheme, enabling to quantify the different stages of the accident.

A reasonably conservative approach was thus developed in 2007-2008, based on existing codes and specifically adapted or developed for the JHR application. Code chaining or coupling are set up where necessary. A detailed assessment on accidents and experiments has been consecutively performed on research reactors in the world since the beginning of the nuclear era.

The accident has thus been chronologically divided into a few stages with characteristic durations varying from 1 to some 100 ms. For each stage, conservative hypothesises have been made in order to reasonably aggravate the consequences, related to the stakes.

2 THE INITIATING EVENT AND POWER TRANSIENT

The hypothetical failure of a control rod leads to a fast ejection of the absorber and a reactivity injection of about 3 $ in 0.1 second. Then the nuclear power rises and decreases very quickly by Doppler and void effect, leading to a thermal energy deposit in the fuel meat and the fuel cladding in a few hundredths of a second. A molten fuel crucible (T>T_{f,Al}) appears inside the plate, expanding and projecting a spray of fuel particles under pressure into the water volume close to the plates. One can imagine the extremely quick cascading propagation to neighbouring plates in the hotter area of the core.
The physics of this phase simultaneously combines neutron physics and thermal-hydraulics, and justifies the development of a 3D kinetics calculation scheme with coupling between CRONOS2 [2] and FLICA4 [3] codes, based on a simplification of the HORUS3D scheme [4,5].

The 3D neutronics simulation is performed with the CRONOS2 code, in an approach homogeneous on assembly level. The 6 group cross sections libraries, elaborated by the APOLLO2 code [6], have been parameterized for a fresh core, in fuel temperature, void ratio and absorber rod insertion level.

The assembly neutronics parameters are homogenised over the whole core for the resolution of the kinetics equations.

The coupled resolution of the coolant thermal-hydraulics and the fuel thermal problem is accomplished with the FLICA4 code. It’s based on the modelling of the HORUS3D calculation scheme with the following modifications:

- adaptation of the thermal exchange and post dry-out vaporisation models,
- nucleate boiling is not taken into account,
- the fusion of aluminium taken into account in the modelling of the fuel materials physical properties,
- modelling of hydrogen production by radiolysis.

The fuel thermal modelling is based on the hypothesis that the U₃Si₂-Al meat fusion is limited to the simple fusion of the aluminium matrix: the U₃Si₂ particles dispersed within this aluminium matrix stay at solid state. We model the physical properties of the fissile meat by a homogenisation of the properties of the Uranium Silicide dispersed phase and the Al matrix.

The data exchange between CRONOS2 and FLICA4 is performed by the ISAS [7] coupling code.

The fast dilatation of the fuel plate, leading to a decrease of the size of the gap between the plates and to a negative reactivity feedback due to the under-moderation, is not considered in the model. The calculations are thus penalizing. This dilatation effect is probably significant and would minimize significantly the transient consequences.

FLICA4 and DULCINEE [8] codes are also separately used for 0D-kinetics sensitivity studies. The legitimacy of the use of 0D-kinetics model is based on the small size of the JHR core and the small deformation of the power distribution during the reactivity accident.

The objective of the CRONOS2-FLICA4 calculations is to obtain a realistic molten fuel ratio and the associated fusion kinetics, as well as the quantity of thermal energy deposited in the fuel, potentially transferable into the coolant.

The choice of the “end of accident” criteria is an important parameter of the setting up of the calculation scheme, as the geometrical variations of the fuel and its rapid loss of integrity are not being considered in the modelling.
3 FUEL - WATER INTERACTION

The MC3D code [9] simulates this key-phase with the injection of the molten fuel particles into the interaction volume and explicitly reproduces the thermal exchange between the fuel particles and water in a time of less than 30 ms. The water pressure rises in a rather isochoric manner up to some hundred bar within less than 1 ms. Then the water volume begins to warm up, decreasing its pressure at the same time. The relaxation of the vapour bubble pressure leads to a violent expansion of the bubble (steam explosion) and to an impulse of some ten bar, acting for several ms on the wall of the nuclear unit’s pool.

The MC3D code is adapted for simulating this steam explosion phase and, in a second time, for calculating the impulse on the pool wall and ground after numerous rebounds and interferences.

Considering the rapidity of the phenomenon, the heat exchange by conduction is quiet insignificant and the fuel meat fusion leads to a mechanical strain on the clad with an increase of internal pressure.

We took the hypothesis of an instantaneous fragmentation of the molten aluminium drops, down to a final diameter close to 40 µm (BORAX specific injection mode). These aluminium drops originate from the clad and the aluminium matrix of the fuel meat. The U$_3$Si$_2$ particles, swept along in the spray of the molten aluminium, are supposed to stay at solid state and keep their 60 µm diameter.

The aluminium droplets and the U$_3$Si$_2$ grains have thus different diameters and temperatures and the MC3D code has been modified to describe the two populations of fragments.

Two injection scenarios are considered on the base of the coupled neutronics/thermalhydraulics calculations, which give the fuel melting kinetics: the first one assumes that the fuel particles are injected as soon as the fusion begins. The second supposes an instantaneous injection of the fuel into the water at the end of the fuel meat fusion, thus simulating a quick increase of the internal pressure in the meat until a violent failure of the clad.

In addition to the thermal energy of the molten fuel, the supplementary energy of the interaction of water with NaK, contained in irradiation devices, and the oxidation of aluminum is taken into account simultaneously and in a conservative manner, by increasing the mass of molten fuel at constant temperature.

Two types of calculation are considered:

- "Vessel" calculations, modelling the pressure vessel and the primary circuit piping with an adapted meshing; the points of interest are the pressure strains on the structures and the presence of water steam.
- "Pool" calculations with the hypothesis of an unrestricted steam explosion.

The first mentioned MC3D calculations, more representative, make possible the analysis of the vessel's failure mode and, in particular, the resistance of the vessel closure head (cannonball effect). They enable furthermore to estimate a potential amplification of the effect, due to the "confinement" during the short phase of the pressure rise.
The second type of calculations leads to a reasonably conservative assessment of the mechanical effects on the walls of the water block, a potential ejection of debris and the kinetics of the steam and incondensable gas bubble.

4 THE EVOLUTION OF THE VAPOUR AND INCONDENSABLE GAS BUBBLE

Immediately after the fuel and water interaction, the brutal expansion of the hot water drop leads to the formation of a bubble containing a mixture of steam and incondensable gas (fission gas, $\text{H}_2$ formed by aluminium oxydation and NaK- water reactions).

Several risks are considered:

- The formation of a water spray at the pool surface, with a potential impact on the containment, with entrainment of fuel debris into the containment hall, thus contributing to the dispersal of radioactive products and accelerating the pressure rise within the containment by direct heating,
- The ascension of bubbles to the surface, directly transporting fuel aerosols to the containment hall, in an analogous manner to phenomenon mentioned before but with microscopic particles.
- The amplification of the radiological consequences outside of the containment, because of a higher instantaneous source term and an accelerated pressure rise.

The structures inside the reactor pool do certainly contribute significantly to the fragmentation of important bubble, but this assumption is neglected in a conservative manner. Likewise, the presence of a considerable height of water above the core (9 m) prevents the entrainment and the ejection of macroscopic fuel or structure debris. The body of experience in accidents and experiments effectuated in the past confirms this point, provided that the water height above the core is significant.

The expansion of the vapour bubble is followed by a phase change at the vapour/liquid frontier. When the pressure significantly decreases, the water condenses and the bubble contracts itself to a new pressurized form before moving on to a new cycle. These oscillations continue, and the bubble rises up in its contracted phases, like in a submarine explosion.

The analysis of underwater explosions, carried out in the military domain in equivalent energetic conditions (equivalent TNT), showed that a chemical explosion does not produce a spray below 9m of water; merely a "dome", consisting of vertical water jets due to the cavitation phenomenon, is observed. As in a chemical explosion the totality of the bubble is formed by incondensable gas, the effects are intensified with respect to the steam explosion in terms of mechanics (no phase change), in terms of bubble cycle time (pulsations more frequent) and regarding the risk of entrainment of aerosol fuel particles (higher volume of incondensable gas).

This phase was also illustrated with the EXCOBULLE experiment [10] performed in the 1980’s for breeder reactor needs. It aimed to study the behaviour of a hot drop in contact with a cold fluid, which takes place in three phases:

- a first, instable phase shows the exchange of hot liquid jets between the two phases (Rayleigh-Taylor instabilities develop at the exterior of the hot drop),
• a mixing phase, where the mixing zone becomes thin and can be describes as homogenous,
• a stable phase, where the condensing film is integrated in the steam bubble.

The bubble quickly disappeared after some oscillations, and change itself into a final smaller incondensable gas bubble.

Only the mechanical effects, essentially linked to the first expansion of the bubble (convective component of the shock wave), and the ascension of fuel aerosols via the incondensable gas are to be dreaded.

5 THE FUEL AND INCONDENSABLE GAS BUBBLE FUTURE

The fuel is ejected in the form of fine particles (some micrometers to some hundred micrometers), cooling down quickly in the water and falling down to the pool’s floor. Thus a fraction of the fuel may be guided to the surface in aerosol form (diameter lower than 200 micrometers) within the incondensable gas bubble.

The analysis of experiments, performed in Japan at NSRR on RIA test [11] on comparable fuel plates and taking into account the statistical analysis of the fuel fabrication foreseen for the JHR, showed that less than 2% of the particles will have a size inferior to 70 µm after the accident. Only 12% of all particles can conservatively be considered as aerosols (d < 200 µm).

The gas bubble is essentially constituted of hydrogen (Al-water and NaK devices–water interactions), immediately formed during fuel-water interaction, and is quickly split into little fragments because of the structures within the core. The fuel particles are thus subjected to various physical phenomena during their slow rising within the instable cap-bubble form (figure 1): thermal-phoresis, diffusion-phoresis, coagulation, washing out, sedimentation (figure 2)…

Initially, and in a conservative approach, a model of aerosol sedimentation in a stable bubble (spherical of cap shaped) enabled a first analytic approach of the transport of aerosols ([12]). Using Stocke's law, the particles velocity can be expressed as a function of their size.
The key results show that:

- Only bubbles with a diameter greater than 10 cm can transport aerosols to the surface, according to this simplified model; with a diameter in the range of 1 meter, a bubble can transport aerosol particles with a size up to 10 µm, thus carrying along a fraction of 0.2% of the transportable fuel aerosol particles (with 100% of the particles produced by the core in a lonesome bubble, and without washing out effect).

- No aerosol particle with a size of 20 µm or more will reach the surface of the pool (for bubbles with a diameter from 2 mm to 1.2 m or equivalent diameter for cap bubbles).

In conclusion, after first analysis, the sedimentation and the washing out are the essential phenomena because of the particle size distribution of the aerosols and the existing vapour in the bubble. These effects show a very significant return of the aerosols in the water.

In a second time, considering the convective effect in a bubble in contact with water, it appeared to be necessary to characterise better the behaviour of this kind of bubble (with a relatively significant size), in order to know if it would ascent to the pool surface without being fragmented underway.

A conservative on-going calculation with the TRITON code [13] simulates incondensable gas bubble hydrodynamics, combined with a DSMC code [14] enabling to “sow” this bubble with a known particle population.

6 THE RADIONUCLIDES’ PROCESS

Except this potential phenomenon of a direct rising of fuel particles to the pool surface, the core radionuclides ejected during the explosion may migrate from the fuel either directly to the confinement, or indirectly through the pool water. This is particularly the case of the rare gas and volatile fission products (iodine family) which are the main contributors to the effective dose outside the facility.

A first step is related to the pressure increase in the containment after the explosion and due to the residual power, but also with potential existence of fission gas directly expelled into the containment, constituting a direct and immediate thermal source.

This pressure excursion is calculated with the HORUS/Sys calculation route, based on the CATHARE code [15].

At the same time, the ASTEC ([16] - figure 3), CERES [17] and GAZAXI [18] codes are respectively used to calculate the iodine transfer process to the containment and the contribution of the main radionuclides to the effective dose during their migration outside the containment. They are used to verify that the containment design covers the BORAX risk.
7 THE POOL SYSTEM AND VESSEL MECHANICAL CONSEQUENCE

The pressure wave created by the explosion propagates into the pool water up to the wall and ground, generating a characteristic impulse on the surfaces after multiples rebounds. This impulse may damage the material constituting the pools system (iron pool liner and wall concrete).

The EUROPLEXUS [19] and RADIOSS [20] codes (figures 4 and 5) are used for a quantitative evaluation of the reasonably conservative consequences of the BORAX accident in terms of the mechanical behaviour of the reactor structures.
8 CONCLUSION – THE BORAX CALCULATION SCHEME FOR JHR

On the base of an active and constructive collaboration between the CEA and external units, a calculation scheme of the BORAX accident (figure 6) is under development, and will lead to:

- better understand the complex physical phenomena involved in very short time frames,
- obtain a reasonably conservative quantification of the different parameters at all stages of the accident,
- support safety assessment.

![Figure 6: BORAX calculation scheme](image)

REFERENCES


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ABSTRACT

Irradiation of target materials for research purposes is an everyday activity in material testing reactors. The estimation of the gamma heating expected to be deposited on irradiated samples is basic safety issue. The GHRRC (Gamma Heating in Research Reactor Cores) code developed in NCSR Demokritos, is based on a point-kernel parameterization. It includes the photons produced from U235 thermal fission and from \((n, \gamma)\) reactions in the core materials. It uses empirical correlations for the dose build-up in the core and the energy absorption build-up in the irradiated sample. The dose build-up factor, as well as the macroscopic cross sections of U235 fission and \((n, \gamma)\) reactions are determined assuming a homogenized core. In this work, GHRRC is used to estimate the relative importance of the mechanisms contributing to the total gamma heating of the irradiated material, for two different fuel enrichments in U235. Comparison is made between the gamma heating components produced in a core fuelled with a) Low Enrichment Uranium (LEU, 19.75% U235, 62.48g U per fuel plate) and b) High Enrichment Uranium (HEU, 93% U235, 10.8g U per fuel plate), with 13% burn-up. In both cases a critical core is considered, with the configuration of the Greek Research Reactor (slab geometry, pool type, light water moderated and cooled, beryllium reflectors, 34 fuel assemblies). The gamma heating of a small Fe sample, located in the middle of a central irradiation channel of the core is examined and the heating components considered are those due to a) the prompt and delayed fission gammas, b) capture gammas originated from heavy nuclides, fission products, structural materials and water and c) the gamma dose build-up in the core. For the capture gammas from fuel plate, indicative isotopes were examined (i.e. isotopes of U, Pu and Sm), based on the relative importance of their absorption macroscopic cross section and the data availability for their photon capture spectra. It was found that the higher thermal neutron flux of the HEU core causes higher heating from fission gammas than in the LEU core. However, the higher uranium content in LEU fuel makes the dose build-up more important in the LEU core. Also, the higher U238 content in LEU fuel induces more significant heating by capture gammas from U and Pu isotopes. The Si contribution (existing only in LEU) is found of small importance while the contributions of water, structural materials and fission products are found higher in the HEU core.
1 INTRODUCTION

Heating from gamma radiation of irradiated sample materials is an issue of primary importance for the safety and the radiation protection of research reactors. Designing of the optimum conditions for a sample irradiation requires calculation of the energy that will be deposited on the target material.

The GHRRC (Gamma Heating in Research Reactor Cores) is a “home-made”, three-dimensional numerical code, developed to estimate the gamma heating of small samples inside a research reactor core. The code, based on a point-kernel parameterization, was found to give reasonable gamma heating estimations within reasonable error margins, which allow the Reactor Operator to pre-determine the irradiation conditions so that the sample temperature will safely remain below the melting point during irradiation [1].

In the present work, GHRRC is used to estimate the relative importance of the mechanisms contributing to the total gamma heating of an irradiated Fe sample, for two different fuel enrichments in U235, with 13% burn-up. The results show that the higher thermal neutron flux in the high enrichment (HEU) core causes higher heating from fission gammas than in the low enrichment (LEU) core. However, the higher uranium content in LEU makes the dose build-up more important in the LEU core. Also, the higher U238 content in LEU induces more significant heating by capture gammas from U and Pu isotopes. The Si contribution (existing only in LEU) is found of small importance while the contributions of water, structural materials and fission products are found higher in the HEU core.

The aim of this work is a) to present the capability of an easily handled model to reasonably assess the relative importance of the components of the gamma heating deposited in a sample irradiated in a research reactor core and b) to contribute to the studies performed within the framework of the RERTR (Reduced Enrichment for Research and Test Reactors) Program [2].

2 THE GHRRC CODE

As mentioned above, GHRRC code is a three-dimensional numerical code, based on a point-kernel parameterization.

The developed model includes the prompt and delayed photons produced from the U235 fission and the gammas produced by neutron capture (\((n,\gamma)\) reactions) in the core materials. Empirical correlations are adopted for the dose build-up in the core and the energy absorption build-up in the irradiated sample. The required neutron fluxes are calculated using the neutronics code system XSDRNP [3] and CITATION-LDI2 [4] in a three-dimensional representation of the Greek Research Reactor (GRR-1) core. For the determination of the macroscopic cross sections for the U235 fission and the \((n,\gamma)\) reactions in the core materials, a homogenization of the core is performed. The attenuation coefficient of the monoenergetic \(\gamma\)-rays is also derived for a homogenized core, as a with-respect-to-density weighted sum of the individual attenuation coefficient values of the core materials [5]. The same approximation is used for the derivation of the core dose build-up factor based on the values tabulated for each core material.

Thus, the rate of the total gamma energy deposited per unit volume of the sample is computed from:

\[
W = \int \int w(\vec{r}, E) dE d\vec{r} (1)
\]
Where \( E \) is the photon energy, \( V_c \) is the core volume and \( w(\vec{r}, E) \) is the heat deposited per unit volume of the sample irradiated at position \( \vec{r} \), from the monoenergetic gamma rays \( E \) released at core position \( \vec{r}_0 \). In the GHRRC \( w(\vec{r}, E) \) is computed from:

\[
    w(\vec{r}, E) dE = dE \frac{\mu_{ab}(E)}{\mu_a(E)} \left(1 - e^{-\mu_a(E)\ell} \right) \\
    B_s(\mu_a(E)\overline{\ell}, E)B_r(\mu(E)|\vec{r} - \vec{r}_0|, E)E \frac{e^{-\mu(E)|\vec{r} - \vec{r}_0|}}{4\pi|\vec{r} - \vec{r}_0|^2} d\vec{r}_0 \sum_n A_n(\vec{r}_0, E)\Phi_n(\vec{r}_0)
\]

(2)

Where \( \mu_a(E) \) and \( \mu_{ab}(E) \) (in \([\text{cm}^{-1}]\)) are respectively the attenuation and the absorption coefficient of the monoenergetic photons of energy \( E \) in the sample material, \( \overline{\ell} \) is the mean chord length of the sample defined as \( \overline{\ell} = 4V_s/S_s \) with \( V_s \) and \( S_s \) being respectively the volume and total surface of the sample [6], \( B_s \) is the build-up factor for the energy absorption in the sample material, \( B_r \) is the dose build-up factor in the homogenized core, \( \overline{\mu}(E) \) is the attenuation coefficient of the photons of energy \( E \) in the homogenized core, \( \Phi_n(\vec{r}_0) \) is the neutron flux at core position \( \vec{r}_0 \) for neutron energy group \( n \) and \( A_n \) is given from the relationship:

\[
    A_n(\vec{r}_0, E) = \sum_j \Sigma_{f,n}(\vec{r}_0)X_n(\ell_{\gamma})(E) + \sum_{j,n}(\vec{r}_0)X_n(E)
\]

(3)

Where, \( \Sigma_{jn}(\vec{r}_0) \) (in \([\text{cm}^{-1}]\)) is the macroscopic cross section of \((n,\gamma)\) reaction for nuclide ‘\( j \)’, with neutrons of the energy group ‘\( n \)’ at core position \( \vec{r}_0 \). \( Y_{jm}(E) \) (in \([\text{J}^{-1}]\)) is the spectrum of gamma rays of energy \( E \) due to \((n,\gamma)\) reactions in nuclide ‘\( j \)’, with neutrons of the energy group ‘\( n \)’. \( \Sigma_{f,n}(\vec{r}_0) \) (in \([\text{cm}^{-1}]\)) is the fission macroscopic cross section of neutron energy group ‘\( n \)’ at the core position \( \vec{r}_0 \) and \( X_n(\ell_{\gamma})(E) \) is the probability that a photon of energy between \( E \) and \( E+dE \) results from fission-produced neutron at the energy group ‘\( n \)’. In GHRRC, for \( X_n(\ell_{\gamma})(E) \) exponential fits are used [7], [8] while for \( Y_{jm}(E) \) the discrete values of PGAA-IAEA and NNDC databases have been included [9], [10].

It should be noted that in the present model application only the gamma rays produced from reactions (fission and capture) with thermal neutrons have been considered, due to lack of gamma rays yield data from epithermal neutrons reactions.

In GHRRC, the energy integration is performed using the trapezoidal method, while a 21-Point, 5th-degree of accuracy formula for triple integrals is used for the volume integration [11].

The code is capable of calculating the gamma heating components separately, with respect to the different reaction types, i.e. fission, core built-up and capture. The code can be very easily handled, even by poorly experienced users.

### 3 MODEL APPLICATION TO THE GRR-1 CORE

GRR-1 is a pool type, light water moderated and cooled reactor, using beryllium reflectors and fueled by MTR-type fuel elements. The reactor is normally operating at 5MW power. The active core dimensions in \( x \), \( y \) (horizontal) and \( z \) (vertical) directions are 45.66cm, 47.74cm and 62.55cm respectively. There are five control blade locations in the core where shim/safety rods are placed. Two critical core configurations were used in this work. In the
The horizontal core configuration is shown in Figure 1, using x (letters) and y (numbers) coordinates. The grid position D4 hosts a control fuel assembly without control rod and is used as a flux trap. Grid positions D4, A7 and F7 are used for material irradiation.

**Figure 1:** Horizontal cross section of the GRR-1 Core. The notation is: CR for control fuel assemblies with control rods inserted, W for water and Be for beryllium reflectors.

The gamma heating of a Fe cylindrical sample of 5cm height and 0.7cm diameter, placed in the middle grid channel D4 was calculated using GHRRC. The heating components considered in the computations include (i) prompt and delayed fission gammas, (ii) capture gammas originated from heavy nuclides, fission products, structural materials and water and (iii) the gamma dose build-up in the core and the energy absorption build-up in the sample. For the capture gammas from fuel plate, indicative isotopes were examined (i.e. isotopes of U, Pu and Sm), based on the relative importance of their absorption macroscopic cross section and the data availability for their photon capture spectra.

It should be noted that the beryllium blocks and the surrounding pool water were not considered, since their homogenization with the active core that intervenes between the considered volume and the irradiated sample in D4, would introduce more significant error than their omission. Also, the \(\gamma\)-rays produced from thermal capture in several nuclides that are present in the irradiated fuel plates were not taken into account, since their \((n,\gamma)\) spectra were not available. Thus, less than 25% of the above nuclides was taken into account, while nuclides with significant \((n,\gamma)\) cross section in the thermal range, such as Xe135, Sm151, Pu241, Pm isotopes and others, were omitted.
The computational domain includes the core shown in Figure 1, with 20 cm of surrounding pool-water in all six sides. The three-dimensional group-averaged neutron flux in the GRR-1 core as well as the densities of the nuclides contained in the irradiated fuel inventory were calculated using the neutronics code system XSDRPM and CITATION-LD12 with the NDF5 238-group library; a slab geometry with the actual nuclide distribution was considered for the above calculations, i.e. separate homogenized zones were defined in the core, as in [12], [13]. Five neutron energy groups were considered, the thermal threshold being at 0.5 eV. The macroscopic cross sections of the U235 fission and the \((n, \gamma)\) reactions in the core materials, \(\Sigma_f\) and \(\Sigma_j\) respectively, were determined assuming a homogenized core, through the relationship:

\[
<\Sigma_c> = \frac{\sum_{i=1}^{n_z} \sigma_{j,i} N_{j,i} V_i}{V_c}
\]

(4)

where \(<\Sigma_c>\) stands for \(\Sigma_f\) or \(\Sigma_j\), \(n_z\) is the number of homogenized zones that include the nuclide \(j\) (e.g. for a nuclide of the fuel meat, \(n_z\) is equivalent to the number of fuel assemblies), \(\sigma_{j,i}\) is the equivalent microscopic cross section (fission or capture) of the nuclide \(j\) in the zone \(i\), \(N_{j,i}\) is the number density of the nuclide \(j\) in the zone \(i\), \(V_i\) is the volume of zone \(i\) and \(V_c\) is the volume of the active core, i.e. without beryllium blocks and surrounding pool water.

4 RESULTS AND DISCUSSION

The results are shown in Table 1. As can be seen, the higher thermal neutron flux of the HEU core causes higher heating from fission gammas than in the LEU core. However, the higher uranium content in LEU makes the dose build-up factor more important in the LEU core, thus resulting to a higher total power density deposited in the sample from fission in the LEU than in the HEU core. Also, the higher U238 content in LEU induces more significant heating by capture gammas from U and Pu isotopes. The Si contribution (existing only in LEU) is found of small importance while the contributions of water, structural materials and fission products are found higher in the HEU core. Aluminium contribution, in particular, is higher in HEU due to its higher content in the HEU meat.

It should be noted that the results in Table 1 are expected to be underestimated, due to (a) the omission from calculations of heating mechanisms such as inelastic scattering, activation of nuclides, epithermal capture and thermal capture in several core compartments and fuel plate nuclides and (b) the core homogenization at least for the core materials that are not really distributed in the core, such as the control rod constituents. The omission of the pool water and the beryllium blocks may have caused higher underestimation of the total gamma heating in the LEU core, since the build-up of the gamma rays that travel towards the sample is expected more significant in LEU. On the other hand, the omission of \((n, \gamma)\) reactions in several nuclides of the irradiated fuel may have induced higher underestimations either in the LEU or the HEU case, depending on the omitted nuclide. For example, the omission of Pu isotopes is expected to cause more significant underestimation to the LEU result, while the omission of \((n, \gamma)\) in fission products, such as Xe, Pm and some Sm isotopes, is expected to cause gamma heating underestimation in HEU. However, it should be noted that the results are considered comparable in the two fuel enrichments.
Table 1: Gamma Heating Power Density deposited in a Fe sample from several mechanisms, for two different fuel enrichments in U235

<table>
<thead>
<tr>
<th>Gamma heating component</th>
<th>( W_c (\text{W/cm}^3) ) LEU</th>
<th>( W_c (\text{W/cm}^3) ) HEU</th>
</tr>
</thead>
<tbody>
<tr>
<td>Non Built-up Fission</td>
<td>8.89</td>
<td>10.85</td>
</tr>
<tr>
<td>Built-up Fission</td>
<td>15.56</td>
<td>15.08</td>
</tr>
<tr>
<td>Core-buildup factor</td>
<td>1.75</td>
<td>1.39</td>
</tr>
<tr>
<td>Capture U</td>
<td>0.035</td>
<td>0.008</td>
</tr>
<tr>
<td>Capture Al</td>
<td>0.732</td>
<td>0.825</td>
</tr>
<tr>
<td>Capture Pu</td>
<td>0.454</td>
<td>0.00012</td>
</tr>
<tr>
<td>Capture H</td>
<td>0.542</td>
<td>0.595</td>
</tr>
<tr>
<td>Capture Sm</td>
<td>0.0308</td>
<td>0.0324</td>
</tr>
<tr>
<td>Capture Si</td>
<td>0.0152</td>
<td>0.000</td>
</tr>
<tr>
<td>Total ( \gamma )-Heating</td>
<td>17.369</td>
<td>16.540</td>
</tr>
</tbody>
</table>

5 CONCLUSIONS

GHRRC is a three-dimensional numerical code for gamma heating computations, based on a point kernel parameterization, which was developed in NCSR “Demokritos”. The code, which is very flexible and easily handled, is capable of calculating the gamma heating components separately, with respect to the different reaction types, i.e. fission, core built-up and capture.

In this work, GHRRC was applied to two different core configurations (LEU, 19.75% U235, 62.48g U per fuel plate and HEU, 93% U235, 10.8g U per fuel plate) of the Greek Research Reactor (GRR-1). The gamma heating power density deposited in a Fe sample located in the middle of a central irradiation channel of the core was computed. Comparison of the gamma heating between the two configurations showed that the higher thermal neutron flux of the HEU core causes higher heating from fission gammas than in the LEU core. However, the higher uranium content in LEU makes the dose build-up more important in the LEU core. Also, the higher U238 content in LEU induces more significant heating by capture gammas from U and Pu isotopes. The Si contribution (existing only in LEU) is found of small importance while the contributions of water, structural materials and fission products are found higher in the HEU core, due to higher neutron fluxes. The omission from the code of surrounding materials, such as the reactor pool water and the beryllium blocks, may have caused underestimation of the total gamma heating, which is higher in the case of LEU core, since the build-up of the gamma rays that travel towards the sample is expected more significant in LEU. On the other hand, the omission of \((n, \gamma)\) reactions in several nuclides of the irradiated fuel have induced underestimations which might be higher either in the LEU or in the HEU core, depending on the omitted nuclide. However, the results are considered comparable for the two fuel enrichments, a finding that can be very useful in cases of core conversion studies, since the transition from a HEU to a LEU core does not seem to yield safety issues rising from the gamma heating.

REFERENCES


Data Collection Computerized System for TRIGA Research Reactor

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ABSTRACT

The data collection for TRIGA SSR 14 MW reactor started in the frame of co-ordinated IAEA research project “Up-date and expand IAEA Reliability Database for research reactors for PSA use”. The necessity to develop a raw data collection and processing computerized system arose due to the need to:

- Store all the information regarding the events produced in the operation of TRIGA SSR reactor, whether these are systems or components failures, events due to test or maintenance or information about reactor power, time intervals, number of scrams, etc.;
- Identify, retrieve, select and group information from raw data sources in a time interval period;
- Calculate reliability data, failure data and confidence interval limits, which are used as input data in the Probabilistic Safety Analysis for TRIGA Research Reactor;
- To study the failure rate evolution for the components.

The paper presents the Computerized System called “PSARelData”, which is used to manage raw data for the history of failures, to obtain reliability data in the PSA analysis and to give information about failure trends. The system was developed in the Visual Basic 6.0 programming environment. The interfaces of Visual Basic 6.0 with Windows Access and Windows Excel allowed to develop the database and to calculate the failure rates and confidence interval limits (95%, 5%) using statistical functions.

The computerized system includes operation events for TRIGA SSR 14 MW reactor during 1979 – 2000, covering three data sources: Shift Supervisor Reports, Reactor Logbooks, Work Authorizations.

1 INTRODUCTION

The data collection for TRIGA Steady State Reactor 14 MW reactor started in the frame of coordinated IAEA research project “Up-date and expand IAEA Reliability Database for research reactors for PSA use” [1]. In the frame of the CRP mentioned above more than 40 components were analysed and processed according to the boundaries and failure modes selected. Generally, the components investigated belong to different systems of TRIGA SSR reactor. More than 6,000 failure and related maintenance records were considered during data collection. Not only independent failure but multiple components failures susceptible to CCF were also collected. In case of multiple components, the events collected are analyzed with respect to component type, failure mode and failure degree. These events involved pumps, control rods and control rod drives, funs, valves. Qualitative analysis of root causes, coupling factors, corrective actions and quantitative analysis of the events were performed. The
information regarding raw data was stored in MS EXCEL worksheets. The period of data collection was chosen between 1979 and 2001 involving three data sources: Reactor Log Books, Shift Supervisor Reports and Maintenance Work Authorizations. Due to necessity to store all the information regarding events in the operation of TRIGA 14MW reactor and to perform reliability data analysis, a software application was created after the completion of the IAEA CRP mentioned. The system for data collection and processing offers a software which may be used in raw data collection, in making queries in the database and in reliability data calculation.

The paper presents a brief description of the Computerized System, developed for TRIGA SSR 14 MW reactor and some conclusions referring to this system.

2 DESCRIPTION OF COMPUTERIZED SYSTEM

2.1 Brief description of logic diagram for data computerized system

The Computerized System called “PSARelData” is used to manage raw data for the history of failures, to obtain reliability data and use them further, in the PSA analysis. Also, the PSARelData system, developed in the Visual Basic 6.0 programming environment gives information about failure trends for different reactor components and structures in different failure modes. The interfaces of Visual Basic 6.0 with Windows Access and Windows Excel allow to develop the database and to calculate the failure rates and confidence interval limits (95%, 5%) using statistical functions [2]. The logic diagram used for the computerized data system is presented in the Figure 1. Information regarding the failure data, test and maintenance data, number of scrams, etc, is collected from the three above mentioned raw data sources of TRIGA research reactor and is available for processing. By the processing action one obtains a visualization of all the failure records ordered in time, or just a selection of these. In addition, the processing of data may go on with the calculation of failures rates and confidence intervals limits. The visualization is possible on the screen but paper reports can be produced, too.

![Figure 1: Logic diagram for data Computerized System](image)

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2.2 Brief description of the computerized application

The computerized application contains five screens (forms). The main form of the application (Figure 2) gives the view of the whole database and offers the possibility to navigate inside it. By means of the main form it is possible to introduce new data and to edit the already existing records using the corresponding buttons.

The component type and failure modes (including critical and degraded failure modes) were taken from IAEA TECDOC 930 and in connection with available information from TRIGA data sources. The event type criteria include the following possibilities: functioning, repairable, revision, replaced, verification.

Also, from the main form one can switch to the queries form (enlarged main form), which allows one to impose different simultaneous criteria for data grouping and selection. The selection criteria are:

- name of the component
- starting and ending dates of the requested failure time interval
- component type
- system to which component belongs
- failure mode
- operation mode (run or stand-by).

![Figure 2. Main form of the application (with example records in Romanian)](image-url)
From the enlarged main form is also possible to write a report containing the result of selection process and to calculate the failure rate or failure probability (depending on the operation mode: run or stand-by) according to statistical formulas. This calculation is accompanied by the calculation of confidence interval limits and the results are displayed in a new view of the datagrid in the enlarged main form. The computerized system calculates failure rate and confidence interval limits for the components in two operation modes: run and stand-by.

To calculate the interval limits, Excel functions are used for F distribution and chi-square distribution according to [3]. The formulas are as follows:

**Stand-by**

If \( f \) is the number of failures and \( d \) is the number of demands, then:

Probability: \( p = \frac{f}{d} \) \hspace{1cm} (1)

Lower 5% limit: \( \frac{1}{1 + (d - f + 1)/f} \ast FINV(0.05, 2 \ast (d - f + 1), 2 \ast f) \) \hspace{1cm} (2)

, where \( FINV \) is probability fraction of cumulative F-distribution from EXCEL

Upper 95% limit: \( \frac{1}{1 + (d - f)/(f + 1)} \ast FINV(0.05, 2 \ast (f + 1), 2 \ast (d - f)) \) \hspace{1cm} (3)

Note that \( f \) and \( d \) are sums of “Failures” and respectively “Demands” fields content over all records appearing in the queries datagrid for that stand-by component.

**Operating**

If \( n \) is the number of failures, \( dt \) is the effective (operation) time interval and \( N \) is the number of identical components in group then:

Failure rate: \( \lambda = \frac{n}{dt/N} \) \hspace{1cm} \( \text{hr}^{-1} \) \hspace{1cm} (4)

Lower 5% limit: \( \frac{CHIINV(0.95, 2 \ast n)}{2/dt/N} \) \hspace{1cm} (5)

, where \( CHIINV \) is inverse of chi-square (\( \chi^2 \)) distribution from EXCEL

Upper 95% limit: \( \frac{CHIINV(0.05, 2 \ast (n + 1))}{2/dt/N} \) \hspace{1cm} (6)

Note that \( n \) is taken automatically as the number of failure records appearing in the queries datagrid for that operating component.

The application is asking for the ratio \( \frac{R}{C} \) between the operation time and calendar time \( C \). The later is taken as a date difference function between ending date and starting date. The failure rate is calculated using the effective (operation) time interval in hours:

\[ dt = R \times C \]
The time ratio \((R)\) is less or equal with 1.00 (pre-defined value = 1.00).

The output of the calculation is displayed as a new grid view containing the six or seven fields (depending on the operation mode) bearing the following column captions:

- “Component”
- “Component type”
- “Failure mode”
- “Failure rate (hr\(^{-1}\)) for operating components or probability for stand-by components”
- “Lower limit for confidence interval limit 5\%”
- “Upper limit for confidence interval limit 95\%”
- “Time used to calculate lambda (h)” (only for operating components).

To restore the queries datagrid view from the lambda calculation output view, the “Show selected records” button can be used.

The figures 3, 4 and 5 show the results of the selection process and calculation of failure rates for two components (centrifugal cooling fan and control rod drive).

![Figure 3. Selection process results for a component in stand-by (control rod drive) for failure rate calculation](image)

The operation mode for the selected failure examples is different for the two cases: centrifugal fan – run mode and control rod drive – stand-by mode.
Figure 4. Results of calculation of failure rate and confidence interval limits for control rod drive (stand-by mode)

Figure 5. Calculation results of failure rate and confidence interval limits for the centrifugal fan (operating mode)
The application has two auxiliary forms that deal with supplementary data in case the “Data source” field in the database is either “Work Authorizations” or “Reactor Logbooks” (figure 6 and figure 7). In these cases, the auxiliary forms, appearing automatically, allow the recording and editing of the particular data that are found for that failure event in each of two data sources.

The “Work Authorization” form appears automatically at click event on Validate data button in order to record additional data when the data source is “Work Authorizations”.

The data can be edited by means of text boxes, except for the first two fields which are displayed here for the sole purpose of identifying the failure record and consequently they cannot be modified at this stage. Thus only the Starting data, Starting hour, Ending date and Ending hour are to be introduced and/or modified using the controls in this form.

This additional form includes a data control that can be used to navigate through all the records in the database having the data source “Work Authorizations” and to determine the position of the current record inside this particular collection.

There is also a “Write Report File” button which causes the ‘Report_File1.txt’ to be written in the working directory.

The “OK” button will update the database and send the application back to the current edited record in the main form datagrid.

Figure 6. Auxiliary form for “Work Authorizations” data source

The “Reactor LogBooks” form appears automatically at click event on Validate data button in order to record additional data when the data source is “Reactor LogBooks”.

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The data can be edited by means of the text boxes. In the case of “Reactor LogBooks” form, the first two fields cannot be modified at this stage. Thus only the reactor starting date, starting hour, number of control drives fail up, reactor shutdown date, shutdown hour, reactor shutdown mode, number of control drives fail down, time interval are to be introduced and/or modified using the controls in this form.

This additional form includes a data control that can be used to navigate through all the records in the database having the data source “Reactor LogBooks” and to determine the position of the current record inside this particular collection.

There is also a Write Report File button which causes the ‘Report_File2.txt’ to be written in the working directory.

The application has the capability to represent graphically the time evolution of number of failures and failure rate by dividing the chosen time interval in an equal number of segments also defined by the user. The form appears on click event on “Trends” button. The time interval, between the starting date and ending date, is divided according to user defined number of time division which is introduced in the coresponding form. Selection can be made using the “Line/Bar” check box for the representation of either 2D bars chart type or line chart type.

“Copy” button permits the copy of chart type or line chart type in a word file, using the “Paste special” from the menu of Word.

Example of time evolution of “Fail to run” failure mode for TRIGA main pumps failure rate is shown in the figure 8.

![ Auxiliary form for “Reactor Logbooks” data source ](image)

Figure 7. Auxiliary form for “Reactor Logbooks” data source
CONCLUSIONS

The necessity to develop a raw data collection and processing computerized system arose due to the need to:

- Store all the information regarding the events produced in the operation of TRIGA SSR reactor, whether these are systems or components failures, events due to test or maintenance or information about reactor power, time intervals, number of scrams, etc.;
- Identify, retrieve, select and group information from raw data sources in a time interval period;
- Calculate reliability data, failure data and confidence interval limits, which are used as input data in the Probabilistic Safety Analysis for TRIGA Research Reactor;
- To assist maintenance, test activity in order to have a schedule of these activities, to optimize the test intervals, repair times;
- To study the failure rate evolution for the components.

The PSARElData System can be applied successfully with minor modification not only for the others Research Reactors but also for data collection in the NPPs.
ACKNOWLEDGMENTS

The author is grateful to the TRIGA Reactor Department in INR for their collaboration in raw data collection for TRIGA SSR 14 MW Reactor.

REFERENCES


Safety Barriers for the HANARO Research Reactor

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ABSTRACT

The HANARO, a 30MW pool type research reactor in KOREA, has many safety barriers to prevent accidents or to mitigate the consequences of accidents. Multiple barriers are incorporated into the design to avoid core damage, irradiation accidents, and a release of radioactive material and a leakage of pool water. Safety barriers designed by considering a defence-in-depth concept should always be operable. The integrity of these barriers are verified periodically by tests and inspections. The reactor operation is restricted according the operating limiting conditions if any barrier is broken. In this paper, hazards and safety barriers are discussed.

1 INTRODUCTION

The HANARO is a pool type research reactor which produces a 30MW thermal power and a maximum 5E14 thermal neutron flux maximum. The reactor assembly consists of an inlet plenum, a heavy water tank with a honeycomb shape core, and a hexagonal chimney. The reactor uses light water as a coolant for all cooling system and heavy water for the reflector cooling system. There are three water pools in the reactor hall of the reactor building. The reactor pool and the spent fuel pool are 13 meters deep to attain a sufficient shielding and cooling capacity and the service pool is 6 meters deep. The reactor hall acts as a confinement which allows a limited leakage of air. Many safety barriers are incorporated into the design of the HANRO for a health, physical, industrial, radiation, and nuclear safety.

2 SAFETY BARRIERS

2.1 Physical and Industrial Safety

The environment in the reactor hall is not void of the accidents which are common in industrial plants. There are water pools, deep rooms with removable hatches, high walk ways, and cranes as shown in figure 1.
Figure 1 View of the reactor hall

For the potential dangers in the reactor hall, the countermeasures are provided in the table 1.

<table>
<thead>
<tr>
<th>Hazards</th>
<th>Area</th>
<th>Barriers</th>
<th>Verification</th>
</tr>
</thead>
<tbody>
<tr>
<td>Falling(human, things)</td>
<td>-Pool</td>
<td>-Safety hand rail</td>
<td>-Surveillance</td>
</tr>
<tr>
<td></td>
<td>-Walkway</td>
<td>-Access control</td>
<td>-Supervising</td>
</tr>
<tr>
<td></td>
<td>-Crane</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>-Deep room</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Dropping(things)</td>
<td>-Pool</td>
<td>-ditto</td>
<td>-ditto</td>
</tr>
<tr>
<td></td>
<td>-Walkway</td>
<td>-deposit</td>
<td>-training</td>
</tr>
<tr>
<td>Sabotage</td>
<td>-Hall and building entrance</td>
<td>-Safety door</td>
<td>-Periodic tests</td>
</tr>
<tr>
<td></td>
<td></td>
<td>-Finger print identification</td>
<td>-Supervising</td>
</tr>
<tr>
<td></td>
<td></td>
<td>-Multiple check points</td>
<td>-Audit</td>
</tr>
<tr>
<td>Dangerous work</td>
<td>-Work place</td>
<td>-Personal protective equipment</td>
<td>-Work procedure</td>
</tr>
<tr>
<td></td>
<td></td>
<td>-Proper tools</td>
<td>-Training</td>
</tr>
<tr>
<td></td>
<td></td>
<td>-Human error protection</td>
<td></td>
</tr>
</tbody>
</table>

2.2 Radiation Safety

All the nuclear facilities in HANARO have mitigation provisions for the relevant postulated accidents to protect humans, the environment, and equipment. Because HANARO is a pool type reactor, shielding and cooling is maintained by the inventory of the pool water. And the heavy water is managed strictly to avoid the risks from a tritium leakage. The air contaminated in the reactor hall is filtered and exhausted through the dedicated ducts and stacks to reduce risks to the employee, the public, and the environment. Also the solid and
liquid wastes are collected in a defined way and moved to a waste management facility in a controlled manner. Barriers provided for the radiation protection are presented in table 2.

### Table 2 Barriers against radiation hazards

<table>
<thead>
<tr>
<th>Hazards</th>
<th>Risks</th>
<th>Barriers</th>
<th>Verification</th>
</tr>
</thead>
<tbody>
<tr>
<td>-Pipe break</td>
<td>-Loss of pool inventory</td>
<td>-High connection pipe</td>
<td>-Periodic inspection</td>
</tr>
<tr>
<td>-Seal leakage</td>
<td></td>
<td>-Siphon hole</td>
<td></td>
</tr>
<tr>
<td>-Pool liner leakage</td>
<td>-Loss of pool inventory</td>
<td>-Concrete shielding</td>
<td>-Periodic inspection</td>
</tr>
<tr>
<td></td>
<td></td>
<td>-Stainless steel liner</td>
<td>-Real time monitoring</td>
</tr>
<tr>
<td></td>
<td></td>
<td>-Leak detector</td>
<td></td>
</tr>
<tr>
<td>-Beam tube leakage</td>
<td>-Loss of pool inventory</td>
<td>-Leak-tight joint</td>
<td>-Periodic inspection</td>
</tr>
<tr>
<td></td>
<td></td>
<td>-Leak detector</td>
<td>-Real time monitoring</td>
</tr>
<tr>
<td></td>
<td></td>
<td>-Leak-tight tube cover</td>
<td></td>
</tr>
<tr>
<td>-Failure of dampers, filters,</td>
<td>-Loss of confinement function</td>
<td>-Redundant devices</td>
<td>-Periodic tests</td>
</tr>
<tr>
<td>and ducts</td>
<td></td>
<td>-Filter performance</td>
<td>-Performance tests</td>
</tr>
<tr>
<td></td>
<td></td>
<td>-Monitoring sensors</td>
<td>-Real time monitoring (position, pressure difference)</td>
</tr>
<tr>
<td>-Waste spread</td>
<td>-Failures of waste management</td>
<td>-Administrative control</td>
<td>-Work permit</td>
</tr>
<tr>
<td></td>
<td></td>
<td>-Proper equipments</td>
<td>-Periodic inspection</td>
</tr>
<tr>
<td></td>
<td></td>
<td>-Check points</td>
<td>-Surveillance</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>-Supervising</td>
</tr>
<tr>
<td>-Contamination</td>
<td>-Failure of work control</td>
<td>-Administrative control</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>-Work procedure</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>-Monitor and survey</td>
<td></td>
</tr>
<tr>
<td>-Irradiation</td>
<td>-Accidents</td>
<td>-Biological Shied</td>
<td>-Barrier verification</td>
</tr>
<tr>
<td></td>
<td></td>
<td>-Administrative control</td>
<td>-Work permit</td>
</tr>
<tr>
<td></td>
<td></td>
<td>-Work procedure</td>
<td>-Safety evaluation</td>
</tr>
<tr>
<td></td>
<td></td>
<td>-Human error protection</td>
<td>-Safety culture</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>-Supervising</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>-Real time monitoring</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>-Sampling and analysis</td>
</tr>
</tbody>
</table>

2.3 Nuclear Safety

Nuclear safety, such as a core damage protection is a basic and important feature for nuclear installations including the HANARO reactor facility. Safety barriers for the nuclear hazards of the HANARO reactor are provided by considering the relevant postulated initiating events. The following events are analysed and verified as manageable events within a safe shutdown state:

- Loss of a coolant flow
  - Loss of a primary coolant flow
  - Loss of electric power
  - Failure of a bypass flow control
  - Loss of a secondary coolant flow
  - Loss of a reflector coolant flow
- Reactivity accidents
  - Start-up accident
Withdraw of control rods
Reactivity insertion from the experimental facility
Introduction of cold water

**External events**
- Earthquake
- Fire
- Flooding

**Other failures**
- Fuel handling
- Equipment failure

All the relevant postulated events or accidents can be mitigated to prevent core damage or a severe accident by the following multiple barriers

- Fuel design
- Fuel cladding
- Upward and passive cooling system
- Reactor concrete island
- Pool water inventory
- Biological Shield
- Confinement building
- Emergency ventilation system
- Emergency water supply system
- Reactor protection system
- Fail safe design

### 3 CONCLUSIONS

Hazards in the HANARO research reactor facility are investigated. There are risks to a physical, industrial, radiation, and nuclear safety in the HANARO plant. Multiple barriers for the various hazards are provided in the form of physical installations, work procedures, and administrative measures. In addition to the barriers, many training programs cover the potential human errors.

### REFERENCES

Transient Behaviour of Low Enriched Uranium Silicide Plate-Type Fuel for Research Reactors during Reactivity Initiated Accident

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ABSTRACT

The pulse irradiation tests were conducted on un-irradiated silicide mini-plate fuels. The principal aim is to study a failure threshold and its mechanism as a function of deposited energy and peak cladding surface temperature. It is revealed that the fuels were intact at energy depositions <82 cal/g but failed at energy depositions of >94 cal/g. A failure threshold must be existed between these two values. Two failure modes, that is, a through-plate cracking occurred below the melting point of Al cladding (640 deg.C) and Al cladding melt above 640 deg.C are revealed. The cause of the former is estimated to be a thermal stress occurred during fuel quench.

1 INTRODUCTION

To understand a transient behaviour of a low enrichment uranium silicide mini-plate fuel for material testing and research reactors, the experiment was conducted on un-irradiated mini-plate fuels at Nuclear Safety Research Reactor (NSRR) in JAEA (The former Japan Atomic Energy Research Institute, JAERI). In 13 experiments, 8 mini-plate fuels were damaged at the temperatures ranged from 174 deg.C to 970 deg.C. Fuel failure threshold and failure mechanism as well as dimensional stability of the mini-plate fuel were studied by means of in-core instrumentations and post-pulse irradiation examination (PIE). The results obtained in this study should be useful as a database for safety evaluation of water cooled research reactors existed in the world [1].

2 EXPERIMENT

2.1 Test Mini-Plate Fuel

The test mini-plate fuel used in this study were designed by JAERI and fabricated by two foreign vendors; CERCA in Romans, France and B&W in Lynchburg Virginia., the U. S. The outline of it is shown in Fig. 1. Similar plate-type fuels are fabricated for the cores of the Japan Materials Testing Reactor (JMTR) and Japan Research Reactor-3 (JRR-3). The fabrication processes for these mini-plate fuels were described elsewhere [2,3]. Characteristics of the test mini-late fuel are summed up in Table 1. The test mini-plate fuel consists of the fuel core (25 ÷ 70 ÷ 0.51mm) sandwiched by Al-3wt%Mg based alloy cladding (35 ÷ 130 ÷ 0.38mm), hereinafter abbreviated as “Al cladding”.

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Figure 1: Schematic presentation of tested silicide mini-plate fuel having enrichment by 19.89 wt% $^{235}$U and density by 4.8 g/c.c.

Table 1: Physical and mechanical parameters of the tested silicide mini-plate fuel fabricated by CERCA and B&W.

<table>
<thead>
<tr>
<th>1. Silicide core (U-21wt%Al-7.5wt%Si)</th>
</tr>
</thead>
<tbody>
<tr>
<td>(1) Dimension (mm)</td>
</tr>
<tr>
<td>70 (length) x 25 (width) x 0.51 (thickness)</td>
</tr>
<tr>
<td>(2) Enrichment (wt%)</td>
</tr>
<tr>
<td>19.89 $\pm$ 0.84 $-$ 0.86 gU-235 per plate</td>
</tr>
<tr>
<td>(3) Element</td>
</tr>
<tr>
<td>Si (wt%)</td>
</tr>
<tr>
<td>7.5 (CERCA), 7.7 (B&amp;W)</td>
</tr>
<tr>
<td>U (wt%)</td>
</tr>
<tr>
<td>92.3, U density: 4.8g / c.c.</td>
</tr>
<tr>
<td>Void fraction</td>
</tr>
<tr>
<td>5.0$\pm$0.9 (CERCA), 6.3$\pm$0.9 (B&amp;W)</td>
</tr>
<tr>
<td>(4) Composition</td>
</tr>
<tr>
<td>Fuel</td>
</tr>
<tr>
<td>U3Si2+USi, U3Si2 density:12g / c.c., U3Si2 &gt; 97wt%</td>
</tr>
<tr>
<td>Matrix</td>
</tr>
<tr>
<td>A5NE (CERCA), A6061-0 (B&amp;W)</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>2. Aluminum alloy cladding</th>
</tr>
</thead>
<tbody>
<tr>
<td>(1) Dimension (mm)</td>
</tr>
<tr>
<td>130 (length) x 35 (width) x 0.38 (thickness)</td>
</tr>
<tr>
<td>(2) Composition</td>
</tr>
<tr>
<td>Al-2.8wt%Mg-0.04wt%Mn-0.01wt%Cr (AG3NE)</td>
</tr>
<tr>
<td>Al-1.0wt%Mg-0.67wt%Si-0.25wt%Cu-0.25wt%Cr (A6061-0)</td>
</tr>
<tr>
<td>(3) Density (g/c.c.)</td>
</tr>
<tr>
<td>2.67</td>
</tr>
<tr>
<td>(4) Mechanical properties at room temp.</td>
</tr>
<tr>
<td>Tensile strength (MPa)</td>
</tr>
<tr>
<td>CERCA: 240, B&amp;W: 114</td>
</tr>
<tr>
<td>0.2% proof strength (MPa)</td>
</tr>
<tr>
<td>CERCA: 130, B&amp;W: 62</td>
</tr>
<tr>
<td>Elongation (%)</td>
</tr>
<tr>
<td>CERCA: 25, B&amp;W: 29</td>
</tr>
<tr>
<td>(5) Blister test</td>
</tr>
<tr>
<td>No blister at annealing temperature of 475$^\circ$, &gt;1h</td>
</tr>
</tbody>
</table>

### 2.2 Instrumentation and Irradiation Capsule

The in-core instrumentation was Pt/Pt-13%Rh bare wire thermocouples (0.2mm outer diameter), hereinafter abbreviated as “T/C’s”. Of which melting point was 1,780
deg.C. These were, as shown in Fig. 1, spot welded directly to the external surface of the mini-plate fuel at different locations. The maximum numbers of welded T/C’s a mini-plate fuel were 9. In most experiments, however, T/C’s used were 5. After assembling mini-plate fuel in the supporting jig with electric cables, it was loaded into irradiation capsules as shown schematically in Fig. 2. All the irradiation tests with those instrumentations were conducted in stagnant water at room temperature about 20 deg.C and one atmospheric pressure inside the sealed irradiation capsule [4].

Figure 2: Schematic drawing of NSRR irradiation capsule for experimental series of 508 for the silicide mini-plate fuel.

2.3 Pulse history

The half-width of power of NSRR pulse irradiation is a minimum of about 4.4ms at a maximum integral power of 110MW · s. The value of this width varies from 4.4 to 20ms depending on the magnitude of inserted reactivity. The effect of pulse width variation in this experiment is, however, negligible since the pulse-width is far below the thermal time constant of the mini-plate fuel (approximately 0.1s). The integral value of the reactor power $P$ (MW · s) measured by micro fission chambers was used to estimate deposited energy $E_g$ (cal/g · fuel plate) in each test mini-plate fuel. Hence, $E_g = kg \cdot P$, where the power conversion ratio $kg$ (cal/g · fuel plate per MW · s), is the ratio of mini-plate fuel power to reactor power. This ratio was determined through fuel burn-up analysis [5], taking the radial and axial power skew into consideration.
Table 2: Summary of the results of in-core measurements and PIEs for the tested silicide mini-plate fuels

<table>
<thead>
<tr>
<th>Experiment</th>
<th>508-12</th>
<th>508-1</th>
<th>508-2</th>
<th>508-10</th>
<th>508-7</th>
<th>508-9</th>
<th>508-6</th>
<th>508-8</th>
<th>508-11</th>
<th>508-13</th>
<th>508-3</th>
<th>508-4</th>
<th>508-5</th>
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<tr>
<td>Miniplate fuel number</td>
<td>CS514837</td>
<td>CS514815</td>
<td>CS514816</td>
<td>12907020</td>
<td>12907010</td>
<td>CS514834</td>
<td>CS514831</td>
<td>CS514832</td>
<td>CS514836</td>
<td>12907030</td>
<td>CS514819</td>
<td>CS514829</td>
<td>CS514830</td>
</tr>
<tr>
<td>Deposited energy (cal/g fuel plate)</td>
<td>32</td>
<td>62</td>
<td>77</td>
<td>82</td>
<td>94</td>
<td>95</td>
<td>96</td>
<td>97</td>
<td>98</td>
<td>115</td>
<td>116</td>
<td>116</td>
<td>116</td>
</tr>
<tr>
<td>Peak cladding surface temperature (°C)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>#1</td>
<td>133</td>
<td>x(a)</td>
<td>200</td>
<td>216</td>
<td>198</td>
<td>279</td>
<td>270</td>
<td>309</td>
<td>No T/C</td>
<td>391</td>
<td>350</td>
<td>971</td>
<td>779</td>
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<tr>
<td>#2</td>
<td>136</td>
<td>177</td>
<td>179</td>
<td>180</td>
<td>210</td>
<td>315</td>
<td>229</td>
<td>261</td>
<td>-</td>
<td>-</td>
<td>372</td>
<td>387</td>
<td>893</td>
</tr>
<tr>
<td>#3</td>
<td>136</td>
<td>216</td>
<td>183</td>
<td>227</td>
<td>199</td>
<td>284</td>
<td>202</td>
<td>211</td>
<td>-</td>
<td>-</td>
<td>414</td>
<td>652</td>
<td>x</td>
</tr>
<tr>
<td>#4</td>
<td>134</td>
<td>234</td>
<td>178</td>
<td>173</td>
<td>237</td>
<td>285</td>
<td>x</td>
<td>244</td>
<td>-</td>
<td>-</td>
<td>393</td>
<td>881</td>
<td>918</td>
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<tr>
<td>#5</td>
<td>139</td>
<td>178</td>
<td>195</td>
<td>204</td>
<td>191</td>
<td>305</td>
<td>261(b)</td>
<td>205</td>
<td>330</td>
<td>-</td>
<td>-</td>
<td>424</td>
<td>544</td>
</tr>
<tr>
<td>#6</td>
<td>139</td>
<td>(c)-</td>
<td>-</td>
<td>182</td>
<td>-</td>
<td>280</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
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</tr>
<tr>
<td>#7</td>
<td>140</td>
<td>-</td>
<td>-</td>
<td>173</td>
<td>-</td>
<td>282</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>#8 (No active fuel core region)</td>
<td>60</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>#9</td>
<td>135</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
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</tr>
<tr>
<td>Average±σ</td>
<td>137±3</td>
<td>201±28</td>
<td>187±10</td>
<td>194±22</td>
<td>207±18</td>
<td>290±14</td>
<td>227±31</td>
<td>271±48</td>
<td>-</td>
<td>391</td>
<td>418±74</td>
<td>871±128</td>
<td>761±117</td>
</tr>
<tr>
<td>Coolant temperature; prepulse (°C)</td>
<td>21</td>
<td>20</td>
<td>22</td>
<td>21</td>
<td>24</td>
<td>22</td>
<td>21</td>
<td>24</td>
<td>21</td>
<td>16</td>
<td>18</td>
<td>17</td>
<td>22</td>
</tr>
<tr>
<td>; peak (°C)</td>
<td>26</td>
<td>24</td>
<td>26</td>
<td>28</td>
<td>53</td>
<td>58</td>
<td>43</td>
<td>45</td>
<td>39</td>
<td>42</td>
<td>47</td>
<td>35</td>
<td>34</td>
</tr>
<tr>
<td>Temp.drop ΔT(Tmax-Tp); min (°C)</td>
<td>26(#1, #5)</td>
<td>72(#5)(d)</td>
<td>77(#2)</td>
<td>73(#1,#4)</td>
<td>74(#5)</td>
<td>149(#7)</td>
<td>96(#3)</td>
<td>105(#3)</td>
<td>-</td>
<td>391(#1)</td>
<td>240(#1)</td>
<td>540(#3)</td>
<td>463(#2)</td>
</tr>
<tr>
<td>; max (°C)</td>
<td>31(#9)</td>
<td>128(#4)</td>
<td>98(#5)</td>
<td>129(#3)</td>
<td>122(#4)</td>
<td>202(#2)</td>
<td>159(#1)</td>
<td>214(#5)</td>
<td>-</td>
<td>391(#1)</td>
<td>440(#5)</td>
<td>882(#1)</td>
<td>853(#4)</td>
</tr>
<tr>
<td>Cladding wall; min (mm)</td>
<td>0.256</td>
<td>0.320</td>
<td>0.261</td>
<td>0.355</td>
<td>0.340</td>
<td>0.347</td>
<td>0.330</td>
<td>0.365</td>
<td>0.347</td>
<td>0.154</td>
<td>0.000(e)</td>
<td>0.010</td>
<td></td>
</tr>
<tr>
<td>Cladding wall; max (mm)</td>
<td>0.412</td>
<td>0.419</td>
<td>0.388</td>
<td>0.413</td>
<td>0.408</td>
<td>0.416</td>
<td>0.412</td>
<td>0.427</td>
<td>0.394</td>
<td>0.540</td>
<td>0.671</td>
<td>0.682</td>
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<tr>
<td>Fuel meat thickness; min (mm)</td>
<td>0.433</td>
<td>0.426</td>
<td>0.427</td>
<td>0.462</td>
<td>0.450</td>
<td>0.423</td>
<td>0.442</td>
<td>0.427</td>
<td>0.491</td>
<td>0.447</td>
<td>0.565</td>
<td>0.578</td>
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</tr>
<tr>
<td>Fuel meat thickness; max (mm)</td>
<td>0.556</td>
<td>0.610</td>
<td>0.587</td>
<td>0.543</td>
<td>0.554</td>
<td>0.555</td>
<td>0.560</td>
<td>0.549</td>
<td>0.554</td>
<td>0.520</td>
<td>1.078</td>
<td>1.125</td>
<td></td>
</tr>
<tr>
<td>Fuel plate thickness; min (mm)</td>
<td>1.235</td>
<td>1.270</td>
<td>1.148</td>
<td>1.240</td>
<td>1.206</td>
<td>1.233(f)</td>
<td>1.224(f)</td>
<td>1.247</td>
<td>1.261</td>
<td>1.289</td>
<td>1.440</td>
<td>1.515</td>
<td>1.549</td>
</tr>
<tr>
<td>Fuel plate thickness; max (mm)</td>
<td>1.253</td>
<td>1.330</td>
<td>1.21</td>
<td>1.260</td>
<td>1.274</td>
<td>1.256</td>
<td>1.261</td>
<td>1.267</td>
<td>1.289</td>
<td>1.440</td>
<td>1.440</td>
<td>1.440</td>
<td></td>
</tr>
<tr>
<td>Maximum bowing (mm)</td>
<td>0.225±14</td>
<td>None</td>
<td>None</td>
<td>0.1120±7</td>
<td>0.1230±11</td>
<td>0.3230±16</td>
<td>0.1430±7</td>
<td>0.1130±3</td>
<td>0.4321±3</td>
<td>1.20±85</td>
<td>6.42±18</td>
<td>2.7±12</td>
<td></td>
</tr>
<tr>
<td>Failure(F) / No Failure (NF)</td>
<td>NF</td>
<td>NF</td>
<td>NF</td>
<td>NF</td>
<td>NF</td>
<td>F</td>
<td>F</td>
<td>F</td>
<td>NF</td>
<td>F</td>
<td>F</td>
<td>F</td>
<td>F</td>
</tr>
</tbody>
</table>

(a) Thermocouple (T/C) malfunctioned
(b) Two temperature peaks
(c) No thermocouples(T/C's) welded
(d) Temperature drop of 72° occurred due to quench at thermocouple location #5
(e) No cladding wall due to significant aluminum agglomeration and denudation
(f) Thickness reduction due to hot apot was not taken into consideration
(g) IC: Incipient crack(number of observations), PT: Through-plate crack, HS: Hot spot, CS: Fuel core separation, CM: Aluminum cladding melt

Mechanical cracking due to thermal stress
Cladding melt

Failure mode

Findings in PIE(g)

IC(2),PT(2) | PT(3) | IC(1),PT(1) | IC(2),PT(1) | IC(3) | PT(2) | CS, CM | IC(1), PT(1) | CS, CM
3 RESULTS AND DISCUSSION

3.1 Transient Temperature

Table 2 is the summary of the results of in-core measurements and PIE. Technical terms used in the table are explained in the following sections.

In Fig. 3, a typical transient temperature measured by T/C #4 that received an energy deposition of 97 cal/g · fuel plate is shown with the pulse power, indicated by dotted line. It can be seen from the figure that cladding surface temperature (hereinafter abbreviated as CST) exceeded the boiling temperature, T_i (154 deg.C), beyond the saturation temperature, T_sat (100 deg.C), due to the pulse irradiation. Commencement of coolant boiling at temperature T_i was determined by data from capsule water level sensor. Namely, the timing of coolant boiling was detected by the movement of water free surface, and the timing of water free surface movement is detected by floating buoy having a magnetic sensor. The CST continued to increase to an overshoot temperature, T_ov (203 deg.C). It then decreased to 194 deg.C and remained <10ms. This CST was thought to be the commencement of film boiling. The author signify it as T_DNB and denote here as the departure from nucleate boiling (DNB) temperature. A signal of DNB temperature can be detected from a temperature plateau should be appeared after T_ov. The DNB value was found to be 174±6 deg.C from the average of 31 data points.

![Figure 3: Typical example of cladding surface temperature (solid line) and reactor power (dotted line), showing boiling temperature (T_i), DNB temperature (T_DNB), maximum overshoot temperature (T_ov), peak cladding surface temperature (T_max), quench temperature (T_q), temperature drop (∆T), and time to quench (t_q). These are from T/C #4 of the mini-plate fuel used in experiment 508-8 (97 cal/g · fuel plate, failure).](image-url)
Above $T_{DNB}$, the increase in CST terminated at temperature $T_{\text{max}}$ (244 deg.C). The CST was then quenched to temperature $T_p$ (116 deg.C) during an interval of $t_p$ (0.135s). The magnitude of the temperature difference is given by $\Delta T = T_{\text{max}} - T_p$ and is denoted here as the “temperature drop (128 deg.C for this case)”. Note that peak CSTs measured in the course of experiments are above $T_{DNB}$ except one which is performed intentionally to have peak CST<$T_{DNB}$ (Experiment 508-12, 32 cal/g·fuel plate, see Table 2).

In two experiments 508-4 and 508-5, all peak CSTs further exceeded the melting point of Al cladding. Figure 4 shows a typical transient temperature around melting point observed in the former. Measured melting point of the Al cladding was found to be $579\pm36$ deg.C, an average from 10 T/C’s. It was lower than that (640 deg.C) given by the binary phase diagram due to the fin effect of the T/C’s [6].

3.2 Failure Threshold and Mechanism

Figure 5 summarizes the relation between the measured peak CST and the given deposited energy. Note again that all peak CSTs are above $T_{DNB}$ except one case (32 cal/g·fuel plate). The tested mini-plate fuel are intact at energy depositions <82 cal/g·fuel plate, while they are damaged at energy deposition >94 cal/g·fuel plate except one mini-plate pulsed at 98 cal/g·fuel plate without T/C’s. The cause of this exception is not clear. The possible explanation is that the mini-plate having no T/C had rather uniformly quench because of no fin effect. The author consider from experimental facts that between 82 and 94 cal/g·fuel plate, a failure threshold must exist.

Figure 4 Experimentally observed solidus-liquidus transformation temperatures of Al-3wt%Mg alloy (AG3NE) by T/C’s welded directly to the mini-plate fuel surface, where physical solidus-liquidus transformation temperatures cited from binary phase diagram of the Al-3wt%Mg alloy are shown by hatched area for comparison.
The failure mode was dependent significantly on deposited energy, that is, CST. It is revealed from the figure that there are two failure modes. One is either a through-plate cracking (<400deg.C) or an incipient cracking occurred between 400deg.C and 640deg.C. The other is apparently Al cladding melt. Detail discussion about failure mode is as follows;

(1) Through-plate cracking failure
For through-plate cracking, a typical example is shown in Photo 1 from B&W (94 cal/g • fuel plate, peak CST: 237deg.C). Two major through-plate cracks propagated perpendicularly from a cladding external surface to fuel core. These cracks are intergranular and rather tight, and they existed locally. This occurred without accompanying significant dimensional changes to the tested mini-plate fuel. The observed damage is likely to be a hardening crack led by a thermal stress due to the temperature drop \( \Delta T \). The calculated thermal stress caused by \( \Delta T \) is ranged between 156MPa and 216MPa, which is greater than the tensile stress (120MPa) and 0.2% proof strength (85MPa) of B&W mini-plates. It implies that the local stress arising from the temperature drop \( \Delta T \) during the quench is enough to affect on test mini-plate fuel cracking. On the other hand, the calculated thermal stress for CERCA fuel ranged between 175 and 394MPa, which is close to or greater than the tensile stress (230MPa) and the 0.2% proof strength (125MPa) of the CERCA mini-plate fuels.
A3-035.8

Photo 1: (a) Overview of the test mini-plate fuel at 94 cal/g · fuel plate (B&W, peak CST: 237 deg.C), where two through-plate cracks occurred locally. (b) The polished longitudinal section cut from the through-plate crack at the plate top region.

(2) Incipient cracking failure
As shown in Photo.2, at temperatures between 400 and 640 deg.C, the test mini-plate fuels failed by incipient cracking, accompanied by a significant plate deformation. This seems to be due to annealing of Al cladding. Namely, crack propagation from Al cladding surface during quench might be ceased at annealed (softened) Al material.

Photo2. Cross section of pulsed mini-plate fuel at energy deposition 116cal/g · fuel plate in experiment 508-3, where peak CST was about 544deg.C. Increase and decrease of meat thickness due to cladding melt is clearly observed.

(3) Al cladding melt
At temperature beyond the Al cladding melt, the test mini-plate fuel failed accompanied with significant formation of molten Al holes, molten Al agglomeration, fuel core separation, and through-plate cracking. This is shown representatively in Photo 3 and Photo 4.
Photo 3: Cross section of pulsed mini-plate fuel at energy deposition 164 cal/g · fuel plate in experiment 508-5, where peak CST was about 918 deg.C. Formation of molten Al holes and molten Al agglomeration is seen.

Photo 4: Cross section of pulsed mini-plate fuel at energy deposition 154 cal/g · fuel plate in experiment 508-4, where peak CST was about 957 deg.C. Through-plate cracking and fuel core separation are seen.

In Fig. 6, in-core data for no cladding melt condition and for cladding melt one are shown. In both cases, neither detectable increase of capsule pressure nor movement of water column was observed. Hence, in spite of drastic damage, the test mini-plate fuel did show neither fragmentation nor destructive force that would be expected from interaction of molten fuel with coolant.
Figure 6: In-core measurement of capsule pressure, water front movement, cladding surface temperature and reactor power as a function of time. These are from (a) experiment 508-6 (96 cal/g · fuel plate, no cladding melt) and (b) experiment 508-5 (164 cal/g · fuel plate, cladding melt). In the latter, a little variation immediately after pulse occurred at capsule pressure and water front movement due to a natural convection of coolant.

In Photo 5, SEM/XMA (scanning electron microscope combined with x-ray micro analyzer) photographs obtained from experiment 508-5 (peak CST>746 deg.C) and that obtained from experiment 508-6 (peak CST<270 deg.C) are shown. In the latter, a microstructure composed of fuel elements U, Si and Al did not change significantly. In the former, however, a reaction between aluminium matrix and silicide particle did occur due to the diffusion of the composed elements. As a result, two additional new phases at outermost of the silicide particles were formed.

Photo 5: SEM/XMA examined along line l shown in the central part of the picture, where (a) specimen from experiment 508-6 (96 cal/g · fuel plate, peak CST<270 deg.C) and (b) specimen from experiment 508-5 (164 cal/g · fuel plate, peak CST>746 deg.C).

A relative magnitude of detected elements was

\[ \text{Al} : \text{Si} : \text{U} = 1 : 1 : 5 \] for case (a) and

\[ \text{Al} : \text{Si} : \text{U} = 1 : 2.5 : 5 \] for case (b).
3.3 Dimensional Stability

JRR-3 safety analyses principally performed by the EUREKA-2 computer code predicted that the maximum PCST after water channel closure was about 228deg.C for silicide fuel. From the viewpoint of verification, a dimensional stability was studied by data obtained from PIE. Hence, the magnitude of bow, that is, the magnitude of water channel closure was determined by specimens cut longitudinally or transversally from pulsed mini-plate fuel. These data are also summarized in Table 2. During PIE, either longitudinal or transversal cut was made along to the T/C. Therefore, a dimensional stability of the plate could directly be related to the measured peak CST.

In Fig.7, a maximum bowing of the silicide mini-plate fuel is shown. Up to the mini-plate fuel temperature of 400deg.C, the bowing was less than 1mm (42% gap closure in maximum case), still remaining a safety margin for the coolant flow. When the temperature was exceeded 400deg.C, however, bowing became greater and closed the water gap. The magnitude pf bow was enhanced significantly by occurrence of necking, that is, a marked thinning of the plate wall thickness at the end peak locations. Such condition is shown in Photo.6. Strictly speaking, the magnitude of bow determined by a single plate configuration may not enough for discussing the safety margin of water channel closure because in conventional research reactor many fuel plates are assembled together. To simulate such multi-bundle condition, at least a pulse irradiation by triplet configuration is necessary. The author has done such kind of experiment. The results are another topics of a separated report.

Figure 7: Observed maximum bowing of silicide mini-plate fuel at PIE, where cuttings a plate were made either of longitudinal (T/C #5) or of transversal sections (T/C’s except #5) in order to contain at least one T/C in a cut specimen.
Photo.6  Cross section of pulsed mini-plate fuel at energy deposition 154 cal/g · fuel plate in experiment 508-4, where peak CST was about 957 deg.C. The marked thinning of the plate wall thickness at the end peak locations (necking) occurred locally and enhanced the magnitude of plate bowing.

4  CONCLUSIONS

The conclusions reached in the present study are summarized as follows:
(1) The tested silicide mini-plate fuels were intact at energy depositions <82 cal/g · fuel plate but were damaged at energy deposition>94 cal/g · fuel plate. A failure threshold must exist between these two values. Departure from nucleate boiling about 174 deg.C and temperature drop ΔT>72 deg.C during quenching occurred in all tested fuel plated at energy deposition >62 cal/g · fuel plate.
(2) The failure mechanism was dependent on the energy deposition, in turn was strongly associated with the peak CST of the test fuel plate. Failure at temperature below 640deg.C (Al melting point) is caused by the thermal stress caused by the temperature drop during the quench. Several local intergranular cracks perpendicular to the axial direction of the plate have propagated from the cladding external surface to the fuel core. Test mini-plate under this situation showed little dimensional changes. Failure at temperature above 640deg.C is caused by the Al cladding melt. Test mini-plate under this situation showed large dimensional changes.
(3) Below the temperature 400deg.C, the fuel plate bow was less than 1mm (42% channel closure in maximum), which remained the safety margin for coolant flow. When the temperature was exceeded 400deg.C, however, the fuel plate bow became greater and caused the closure of water gap. The bow was enhanced significantly by occurrence of necking, that is, a marked thinning of the plate wall thickness at the end peak locations.
(4) Within this experimental scope (temperature <970 deg.C), no destructive force as a result of interaction between molten Al and coolant was observed.

REFERENCES


Reevaluation Of BDBA Consequences Of Research Reactors

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ABSTRACT

Up to now, the French aluminium plate-type, water-moderated research reactors have been designed accounting for the consequences of a core disruptive RIA, assuming an envelope bounding thermal energy release of 135 MJ during the power transient, and a mechanical energy delivery in the thermodynamic interaction between molten aluminium and the liquid water of 9% of the whole thermal energy. According to the IRSN, both BORAX-I, SPERT-I destructive tests and SL-1 accident do not show restrictive phenomena on the thermal energy release, which mainly depends on reactivity insertion and core features. Consequently, in the framework of "Beyond Design Basis Accident" (BDBA) analysis, IRSN has decided to study scenarios representative of large reactivity insertion sequences using a semi-empirical simplified model and extending the application of the coupled neutronics-fluid dynamics code SIMMER, originally developed for LMFR, to the BDBA field in water-moderated research reactor. An innovative method to improve the treatment of resonance self-shielding in heterogeneous media has been developed; a model to treat fuel plate geometry has been implemented and new clad-to-coolant heat transfer coefficients suitable for extremely fast transient conditions, have been adopted. For the large reactivity insertion sequences tested, it was found that the geometry of the core immediate surroundings (including the narrow coolant channels and the reactor vessel, with a coolant inlet and coolant outlet separated from the main pool), has a major impact on the transients. The investigation of structure failure model can finally answer questions about the mechanical energy release, the deformation potential, the influence of failure on mechanical loads elsewhere, and the maximum local pressures.

1 ACCOUNTING OF RIA SEVERE ACCIDENT FOR FRENCH RESEARCH REACTORS

Core destructive tests carried out in the United States in BORAX-I reactor in 1954 and SPERT-I in 1962, as well as the accident which occurred on January 3, 1961 in SL-1 reactor in the United States (Idaho), have emphasized that water-moderated with aluminium-type fuel reactors could be, in case of a fast and large reactivity insertion, dramatically damaged by violent excursions of power involving degradation, even the fast melting of a part of the core, as well as the partial or total degradation of reactor structures of the. The thermal energy released generates a water steam bubble, by fuel-coolant interaction (FCI), which expands in the primary loop and in the reactor pool, with shock waves. This accident can particularly induce:

- the destruction of experimental devices, which can contain non condensable gas,
• the damage to the reactor pool walls,
• the weakening containment lower part, by thermal effects of not dispersed melted materials,
• a water spray to the hall of the reactor,
• a production of hydrogen by the oxidation of aluminium by the steam water,
• a damaging of the upper part of the containment (hall of the building), due to the increase of both temperature and pressure, and, maybe, a hydrogen explosion,
• the transfer of rare gases and volatile fission products to the hall of the reactor building, as well as a possible drive of fragments or fuel particles in this hall, etc.

In France, this type of accident, named BORAX, is taken into account in the design of research reactors. Although considered as a “Beyond Design Basis Accident” (BDBA), which induces strong arrangements to prevent its occurrence, this accident actually is an extension of the “Design Basis Accidents” (DBA) domain through design of important safety related equipments (containment buildings, walls of the pools, post-accidental heat removal systems, filtration devices, etc.); accordingly, these important equipments have functional requirements in order to allow mitigating the BORAX.

In the case of the French research reactors, this accident has been historically accounted for through an energetic approach. The main assumptions were, for all the reactors, a thermal energy of 135 MJ delivered in fuel during the transient of power and a mechanical energy, from the FCI as much as 9% of thermal energy. These assumptions have been adopted for the last research reactors built in France, namely the High-Flux Reactor in Grenoble and the reactor ORPHEE in the Saclay Nuclear Centre.

In 2003, during the assessment of the JHR (Jules Horowitz Reactor) safety options, IRSN wondered upon these a-priori assumptions, because several main features reduce the representativeness and the transposition of destructive tests such as BORAX-I, SPERT-I to the RJH case. IRSN mainly noticed that:

• the fuel of those reactors was high enriched in $^{235}\text{U}$ (93%), which is not the case for many current research reactors, and, in particular, the JHR;
• no absolute limit seem existing on the thermal energy release: this energy depends strongly on the introduced reactivity and kinetics. The three reports BORAX-I, SPERT-I and SL-1 do not provide any element likely to justify that the value of 135 MJ is an absolute maximum.

Consequently, IRSN considered that it was advisable adopting another approach for this type of accident, based on the study of scenarios representative of the sequences of reactivity introduction, to be taken into account in the BDBA scope, accounting for all available knowledge and adopting up-to-date modelling to study the phenomena brought into play. Following the request formulated in this direction by the French Safety Authority, the studies carried out by the CEA for the JHR preliminary safety report lead to a thermal energy in case of BORAX higher than 135 MJ. For the expertise needs, the IRSN undertook, in collaboration with various international partners and mainly FzK, an adaptation of the code SIMMER-III, originally designed for the fast reactors sodium-cooled. This paper displays some present aspects of this important work in progress.
2. SIMMER-III MODIFICATIONS FOR MODELLING RIA IN RESEARCH REACTORS

SIMMER consists of three modules: for space-time neutron kinetics, for reactor structures, and for multiphase multi-component transient compressible fluid dynamics. SIMMER is dedicated to LMFR Safety studies [1].

![SIMMER overview diagram]

It is built on two superposed Eulerian meshes with cells that are consistent with the needs of both, the neutronics and fluid dynamics. For each of the eight major components, a full beyond-van-der-Waals equation of state is used [2], and each mobile component can have its own velocity.

### 1.1 Neutronics

The implementation of the neutronics part of the code has been carried-out with FZK, which has extended available and generated new cross section data libraries for SIMMER thermal reactor application. It has also improved the code with respect to neutron up-scattering and to the heterogeneity effect treatment in thermal reactors.

A set of codes for cross-section processing available at FZK has been extended in order to provide data in the JNC-extended CCCC format, and has been used for including data (from the ENDF 6.8 data library) for Be-9 (metal) in the 18-group library that is currently employed at IRSN for RHJ studies. Data for Be-9 are required for modelling of the research reactor reflector.

A new 40-group library for SIMMER that may be adopted in future studies on research reactor has also been implemented. Test calculations - performed by now at FZK - show that this 40-group data library provides more accurate results (compared to the 18-group one) for a set of thermal reactor models proposed by IRSN for benchmarking of SIMMER neutronics capabilities.

It has been extended the cross-section processing part of SIMMER in order to include a newly developed technique for taking into account heterogeneity effects. The preliminary results show that this technique improves the accuracy of calculations for the reactor models proposed by IRSN. Additional efforts on validation of the mentioned technique will probably be performed during this year.
From the initial runs for simplified geometric arrangements it was obvious that the currently available version of SIMMER was not particularly well suited for the treatment of thermal reactors. Especially the neutron up-scattering during transients was not dealt with efficiently. As we already knew about this shortcoming from own experience, we were able to implement corresponding improvements fairly quickly which helped appreciably to improve the performance of the so-called gamma-iteration. Some unexpected difficulties with convergence performance, which may be related to rather coarse spatial meshes and correlated negative flux fix-ups had to be investigated in detail and suitable improvements have been implemented in an updated version of SIMMER. At present, it cannot be completely excluded that further modifications will be needed e.g. for a much more refined model of research reactor with smaller meshes and/or adoption of more energy groups.

1.2 Reactor structures

The description of the interior of the reactor structures can deal with undestructed geometry in which three characteristic temperatures are given in each cell in order to model heat conduction, a state of destruction inception where a part of the structures remains intact, and a fully destructed state where all liquid and solid debris move and interact with the liquid or gaseous coolant. Destruction inception is based on threshold temperatures, melted volume fractions, and/or pressures in the interior of the structures.

Rising fuel temperatures have two main effects: first, the Doppler broadening reduces reactivity, and second, the fuel dilatation reduces the presence of water next to the fuel thus lowering the level of neutron slowing-down. The water that is pushed out of a given cell flows to its neighbours. As will be described below, the fuel structure can exchange energy and momentum at its surface with the coolant water. SIMMER possesses a heat transfer correlation based upon transient overpower experiments in the NSRR reactor [3] with conditions very similar to those of the present study (see figure 2).

![Figure 2: SIMMER heat transfer coefficient for transient boiling](image-url)
Beside the fuel elements, the code describes the transient behaviour of left and right subassembly outer structures (in research reactor the casier) each of which having two characteristic temperatures. These structures can also melt or mechanically fail.

1.3 Fluid dynamics

The model describes the movement of continuous and discontinuous fluids including solid particles in a staggered Eulerian mesh with state values defined at the cell centres, and velocities at the cell boundaries. The solution of the conservation equations of all components and states is of second order in space and time. A predictor-corrector method guarantees a smooth incorporation of the two other modules with the objective of conserving mass, momentum, and energy. The interfaces between all mobile components and between them and the intact structures are interfacial areas which depend on space and time. These areas are calculated using a simplified convection equation with built on source terms based on correlations from the multiphase fluid literature. A several year long verification program of this code part has been performed at CEA [4].

In the research reactor, the fuel channels are very narrow. It would be prohibitive to model all 296 channels individually. Instead, one SIMMER ensemble of fuel element and adjacent channel represents several ten channels. During a very rapid heat-up of the fuel, the temperatures in the water may vary substantially, from close to evaporation at the fuel element surface to slightly heated up in the channel centre. Because the heat is predominantly flowing into narrow layer at the structure surface, this results in an early water evaporation which, in turn, reduces coolant densities and lowers reactivity. Therefore, the coolant channel has to be divided into four radial sub-channels. The thickness of the sub-channel next to the fuel is chosen so that results of very fast heat-up transients in the PATRICIA experiment are well represented (Bessiron, private communication).

Figure 3: Comparison of PATRICIA results with a SIMMER calculation

Although second order solutions of the conservation equations can be adopted, numerical diffusion may extend over three adjacent fluid cells. The choice of four sub-channels is based upon the assumption that numerical diffusion and turbulent diffusion are of the same magnitude.
Upon destruction of fuel structures, liquid material and solid particles are injected into the coolant channel. The SIMMER calculations show that they penetrate to the farthest radial sub-channel. This is equivalent to the radial penetration of debris all over the research reactor channel. During a core-disruptive accident, fuel will fail first in the core centre, and then failure will propagate radially and axially to the core periphery. Under these conditions, the thermal interactions in the channels are first limited by the short penetration length. Water vapour is rapidly generated and superheated so that quite high pressures build up at the site of the first failure. Consequently, water is pushed out axially to the top and the bottom of the core. The kinetics of this movement is superposed to that of the propagation of fuel failure which advances in the same two directions. SIMMER can answer the question whether the voiding of the channels is slower or faster than destruction propagation, and whether the hot debris has a chance to efficiently interact with liquid water and thus increase pressures. The scenario of fuel coolant interaction has been verified for SIMMER [5].

In the SIMMER calculations, mechanical deformation beyond a representative channel cannot be taken into account. However, in research reactor, the radial deformation will be limited any way by the solid block of the casing.

The propagation of the pressure wave out of the narrow channels into the adjacent water plenum is limited by the sound velocity. Each code cell has a representative pressure. In case of propagation of the wave to the neighbour through a two-phase mixture, the derivative of density with the pressure is calculated. The derivatives of the mixture of components which is proportional to the mixture sound velocity are dependent on the local temperatures. The two-phase sound velocities are substantially lower than those of the single phase components, thus slowing down propagation.

At very high released thermal energies, the vapour may exit the core periphery and enter the water inlet and outlet plenum of the primary water cycle, which are carefully modelled by the code. Pressure drops are calculated using standard correlation for turbulent steady state flow. Orifice coefficients can be added to each cell. The perforated cylindrical flow distributor below the core is described by such coefficients.

While some structures outside the core may be rigid enough to withstand high pressures, others may fail early. In SIMMER, this is modelled by defining a threshold pressure at which the wall of the vessel is changed from a solid immobile structure to mobile particles. Although this may reduce inside-vessel pressures, the early part of the pressure waves may penetrate unmitigated to the bottom of the vessel.

Finally, the big vapour bubble enters the pool where it meets the bulk of cold water. At the bubble periphery, entrainment of cold water may take place. SIMMER has been verified on entrainment rates measured at FZK [6]. SIMMER also possesses the capability to take into account diffusion-limited condensation on the bubble surface in the case that non-condensable gases are present. This model has been verified on experiments performed at the Kyushu University, Japan [7]. Since the whole pool with its open surface to the ambient air is modelled, pool surface displacements and pressure loads are also results of the SIMMER calculations. The pool surface is influenced by two phenomena, first if the expanding vapour bubble is forced by structures to expand predominantly in axial direction an early doming of the surface becomes visible. Second, during bubble condensation, water of the upper pool starts to move downwards. This initiates a surface wave that can lead to increased doming if the wave is reflected at the pool wall and moves backwards to the centre of the pool.
2 RESULTS OF THE STUDY

Figure 4 outlines the research reactor geometry in SIMMER format, where the numbers at the side denote the axial and radial cell numbers.

In the left part is shown the cylindrical core model of research reactor with reflector (grey), in the right the closed cylindrical vessel model of research reactor, dashed lines = perforated structures with orifice pressure drop, inside the reactor pool with an open water surface. The measures in the vicinity of the core were read from copies of artist’s views of the
Axial and radial scales were not always consistent. The SIMMER geometry should be checked in the future. Two calculations are presented:

- a core within an indestructible vessel where the vessel lid has been removed (open vessel),
- a core within a detailed vessel including in-vessel perforated structures and a closed lid, where those axial parts than consist only of an aluminium wall of about 20 mm fail when the adjacent pressure exceeds a threshold value (closed vessel).

Both cases are calculated assuming a zero water velocity through the core. The initial pressure inside the closed vessel is about the same as outside. The transient starts before a steady state hydrostatic head inside and outside the vessel can be established. Therefore, pressure oscillations are still present when the core starts to be destructed. In a final study, a SIMMER pre-run with steady state power needs to be made until the hydrostatic equilibrium has been reached. The final state of this run will be used as the initial condition of the transient run. The closed vessel case has a threshold pressure of 11 bar. This value is probably below the failure pressure of the wall but it produces early vessel failure, and thus a reduction especially of pressure below the core. The following figure 5 shows the integral energetic results of both cases.

Figure 5: Comparison of PATRICIA results with a SIMMER calculation
They show little difference. The power in the case of an open vessel increases a very little bit later. The lower red curve represents the reactivity inserted (an input for the code). It is a linear ramp from zero to 2000 pcm at 100 ms. While the reactivity increases linearly until 100 ms, feedback become important afterwards. At 130 ms, the core starts melting, but the Doppler broadening and the moderation effects have already decreased the reactivity to close the initial value. Core destruction runs out at 160 ms. First the fuel temperatures rise, but the Doppler becomes effective only at higher values. Then water is leaving the core region pushed out by fuel dilatation which is also effective only a substantial temperature increase. Finally, as a consequence of FCI, the water vapour mass increases at the time the power is already down to half the peak value. A detailed picture of the core destruction is shown in the following figure 6.

![Figure 6: Side views of destroyed core](image)

At 180 ms (left figure), the vapour bubble has reached its maximum size and starts to re-condense. Core destruction has progressed only little coming to an end at this time. At 240 ms (right figure), the vapour bubble has almost completely collapsed, and the core debris is cooled down effectively by the water.
Figure 7: Fuel plates heat transfer coefficients (standard and NSSR)

Figure 7 shows that just before the fuel failure, which happens at 130 ms, the heat transfer coefficients of SIMMER-research reactor are significantly higher than those of standard correlations, which depend on fluid velocities. There is a negligible fraction of water vapour in the channel until the fuel plates fail and the subsequent FCI rapidly evaporates the water in the channel. Because the peak pressures are generated by FCI, it is important to study the condition under which FCI occurs. First, as already mentioned in the description of the code, the movement of hot debris from the fuel plates into the water channel is limited by the channel width. Then the generated vapour escapes axially upwards and downwards. In the following figure 8, the kinetics of this process will be demonstrated.

Figure 8: Vapour volume fraction close to the fuel plate

Figure 8 above here shows the volume fractions of the cells with fuel plates of centre core ring. It takes about 5 ms to void all cells of the core containing fuel plates.
Whether the voiding of the channels can influence local interaction must be studied by looking at two different radial cuts of the centre fuel ring of the core.

![Vapour volume fractions](image)

The figure 9 shows the volume fractions of the first two cells in the fuel plate and the volume fraction of water vapour. For the axial core centre, the four water rings adjacent to the fuel show voiding only 1 ms to 2 ms later than the simultaneous disruption of outer fuel and clad at 126 ms. This indicates that the debris are injected into a channel filled with water. For the right figure of the lower end of the core, there is less a time lag between disruption and voiding. The void profiles for the second and third cell show an earlier voiding than that of the cell of the fuel surface indicating that the bubble movement is predominantly in the channel centre.

To conclude, it is probable that the emerging bubble does not affect the efficiency of FCI in the core centre, but at its axial peripheries, the early presence of the bubble should decrease FCI.

The description above has enabled understanding the mechanisms that drive the pressures in the channels and outside of the core. The results show substantial difference between the open and the closed vessel. We need to keep in mind that the closed vessel is subjected to an early vessel failure below the core which results in a sort of open vessel. Simplifying the pressure-relevant conditions, the closed vessel is actually a vessel open below the core, and the open vessel is a vessel open above the core.

The figure 10 shows a comparison between pressure of the closed and open vessel cases. It demonstrates the time scales of the oscillations after peak power.
Figure 10: Open and close vessel pressure oscillations

The figure 10 shows in black the bottom pressures adjacent to the lower vessel wall which is of specific interest because its failure may lead to leakage. For the open pool at the maximum pressure, the radial distribution of this pressure is shown in the figure at the right. The value of the closed vessel is lower because of early vessel failure below the core. It shows high frequency oscillations. The maximum bottom pressure of the open vessel is 14 bar, that of the open vessel 57 bar. Oscillations prior to 120 ms are due to a start of the transient from a non-steady state of hydrodynamics.

To demonstrate the long time behaviour of the open vessel, the following figure 11 shows the vapour bubble (yellow) emerging from the core centre. It moves out of the core in both the upward and downward direction. Because the lower plenum is closed, the lower bubble does not penetrate further while the upper rises until almost half of the upper plenum is voided. Water displacement out of the upper plenum increases the pool surface (yellow).
3 CONCLUSION

The coupled neutronics-fluid dynamics code SIMMER has been up-graded to treat Design Basis Accidents in experimental reactors with thermal spectra. The research reactor has fuel plates and water cooling channels of millimetre-size, so that effects of core heterogeneity on neutron flux distribution are smaller compared to conventional LWRs. Due to off-the-shelf SIMMER models, this allows defining fluid dynamics and neutronics computational meshes convenient to the specific needs. The code was improved including models which increase its representativeness. However, further validation of neutronics and fluid dynamics models adopted is still necessary and is underway. The objective of the improvements is to calculating the different reactivity feedbacks, the deposited thermal energy, and the sequence of core disruption, transfer of thermal to mechanical energy, and finally pressures at sensitive locations.

It was found that the geometry of the region surrounding the core has a dominant influence on the transient. The structure failure model can finally answer questions about mechanical energy releases, deformation potential, the influence of failure on mechanical loads elsewhere, and maximum local pressures.

REFERENCES


Simulating a Partial LOCA in a Narrow Channel Using the DSNP Simulation System

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ABSTRACT

A partial LOCA accident in a pool type research reactor was investigated. A new MTR type fuel channel model for the DSNP simulation system was developed, permitting the calculation of detailed axial and radial temperature distributions. New and older heat transfer correlations were incorporated in the model. Simulations of accidental water levels of 14 and 35 cm during a partial LOCA in a 62cm narrow fuel channel were performed. The resulting maximum temperatures remain significantly below the aluminium melting point, and no damage to the core will take place under these conditions.

1 INTRODUCTION

A partial LOCA (Loss of Coolant Accident) is an extremely rare event in a research reactor. The purpose of this study was to investigate the consequences to the MTR type fuelled core having very narrow coolant channels, when the water level in a pool type research reactor remains at a level of between 20 to 55% above the fuel channel entrance. The water level is reduced due to a hypothetical guillotine break in either the primary coolant loop or in an experimental beam tube. A schematic description of the IAEA generic pool type reactor[4] is shown in Fig. 1, emphasizing the location of possible pipe breaks which may cause a partial LOCA event. The reactor structure includes an open concrete cavity filled with demineralised water up to a level of 10m above the pool bottom. The core is placed on a grid plate located 1m above the pool bottom. Six to eight beam tubes can penetrate the pool concrete wall reaching the core edge and providing neutron beams for experiments.

Following power shut-down, upon the detection of the water leak through the severed pipe, the decay heat remaining in the core will heat-up the fuel, and water boiling in the narrow channels might take place. The boiling process, depending on the residual power level, will force some of the water out of the channel, and the boiling process may be considered as boiling under very low Reynolds number or some kind of pool boiling or percolation condition.

A model for these conditions was developed for the DSNP[1] (Dynamic Simulator for Nuclear Power-plants) system using a recently proposed heat transfer correlation by Zhang, Hibiki and Mishima[2] for small diameter channels, and several older correlations given partly

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in Ref. [3]. The model was applied to study the behaviour of the IAEA generic research reactor[4] under partial LOCA conditions. This reactor is a pool type, light water, 10MW reactor, schematically presented in Fig. 1, using MTR type fuel elements. It is shown in the present study, that if the water level remains at or above 20% of the fuel channel length, that is, a partial rather than a full LOCA condition, no damage to the core will occur. This was also shown by the experiments performed at Livermore[5] and reproduced in this study.

2 THE LOCA ACCIDENT

The accident starts with a reactor operating at full power. In the case of a major break in one of the pipes, the water level in the pool starts decreasing. The reactor safety system will scram the reactor and further decrease in the water level will also stop the pumps. With no forced circulation a shutter valve will be opened by gravitational force and natural circulation through the core will be established removing the decay heat. Further decrease in the water level will eventually uncover the core and natural circulation ceases. The water in the core will start heating up and boiling will occur. A stable condition is achieved once the water level reaches equilibrium condition with the surrounding structures at some level along the core height. Most research reactors of this type are designed for the water level to stabilize at about mid-core or above.

A guillotine break in the primary loop or in a beam tube is modelled by starting an outflow of water from the pool via the broken pipe. The flow rate is determined by solving the momentum balance equations using the appropriate DSNP[1] models. The pumps are stopped after a short time, and the reactor is shut down. The power decreases rapidly until it reaches the decay heat level as shown in Fig. 2. The decay heat is calculated from the ANS standard curves for 235U. Calculations have shown, that it will take about 21 min for the water flow to reach equilibrium conditions, as can be seen in Fig. 3, and the out-flow is reduced to zero. When the water reaches its equilibrium level, about mid core (according to the exact reactor and its surrounding geometric design features), this part of the simulation is terminated.

Figure 1: Schematic drawing of an open pool type research reactor

Figure 2: Hot channel power reduction during a partial LOCA accident
As can be seen from the above figures, the power at time of reaching equilibrium conditions in the hot channel is 600W and the water out-flow ceases at this time. The simulation of the core condition is stopped at this point and the prevailing conditions serve as input to the next step of simulating the hot channel under partial LOCA conditions with the water level at mid core.

Figure 3: Water flow rate from the reactor pool following a break in the main coolant loop.

3 RESULTS OF THE SIMULATIONS

The results are presented here for a partial LOCA in which the equilibrium water level reached the mid-core level. Actually a level of 35cm from the coolant channel entrance was chosen (mid-core + 4cm), which is the equilibrium level following the LOCA event in the main coolant loop, for the geometry selected as shown in Fig. 1. In addition the low 20% core water level condition is also simulated in order to compare the results with the Livermore experiments[5]. All the results presented in this chapter are based on the simulations performed with the DSNP models presented in the next chapter of this paper. In Fig. 4 the power distribution in the hot channel and the actual power transmitted to the coolant are presented at the time when the equilibrium water level in the reactor pool is reached. The difference in the two distributions results partly from the changing heat transfer correlations due to the changes in the flow regime, and partly due to axial conduction of the heat along the fuel and the side plate. As can be observed part of the heat from the hot upper part of the fuel is conducted to the lower colder part of the fuel and the side plate. This can be concluded from the observation, that in the fuel plate lower part, more heat is transmitted to the coolant than the heat produced in this region.

Figure 5 shows the axial temperature distribution in the core hot channel in the fuel, the side plate, and the coolant channel. As can be seen the coolant enters the channel at 30°C, and starts boiling at about 18cm. The water coolant mixture flows up the channel cooling it as it flows with the steam fraction increasing continuously but remaining at saturation temperature until exiting the channel. The flow characteristic changes along the channel, however the most appropriate description will be a dispersed film boiling or slug flow. The maximum metal temperature is 135°C, which is significantly below the softening temperature of Al - about 400°C. Fig. 6 shows the steam and water enthalpies along the channel. The steam and
water flow rates are shown in Fig 7. At this stage both the water and steam are still at saturation conditions. The water enters the channel at 30°C, and starts boiling at 18cm. It continues to boil along the channel and exits with a quality of 57%. Under more realistic conditions slugs of water will be moved up the channel, some of the water is expelled from the channel while part might flow back into the channel.

![Graph showing power distribution along the hot channel (ZWIJ) and the power transmitted to the coolant (QHFC) at time of equilibrium water level conditions.](image)

**Figure 4:** Power distribution along the hot channel (ZWIJ) and the power transmitted to the coolant (QHFC) at time of equilibrium water level conditions.

![Graph showing fuel (ZTFP), side plate (ZTSP) and water-steam (ZTST) temperatures during a partial LOCA in the IAEA generic research reactor, equilibrium water level is at 35cm above the channel entrance.](image)

**Figure 5:** Fuel (ZTFP), side plate (ZTSP) and water-steam (ZTST) temperatures during a partial LOCA in the IAEA generic research reactor, equilibrium water level is at 35cm above the channel entrance.

Fig. 8, shows the axial temperature distributions for the case in which only 14cm of the core remains under water. This level was artificially chosen in order to compare the results with an experiment presented in Ref. [5] for the Livermore MTR fuel experiment. The distribution is presented using two different heat transfer correlations[^3]. As can be seen the fuel temperatures are much higher than in the previous case reaching 270°C and 310°C, and the steam gets superheated in this channel.
Some experiments with mock-up fuel channels performed at the Paul Scherer Institute\cite{8}, indicate that higher temperatures might result with the pool water level at 25cm. These results were not reproduced in the present study, and further theoretical and experimental investigations are underway for different pool water elevation, and different power ratings.

Figure 6: Water (ZHCP) and steam (ZHST) enthalpies along the coolant channel following a partial LOCA condition.

Figure 7: Water (Wli) and steam (Wsi) flow rates along the hot fuel channel, following a partial LOCA with equilibrium water level at mid-core.
4 THE FUEL CHANNEL MODEL

In the literature there is very little information on boiling and dry-out in parallel channels of the MTR type fuel elements, as boiling in this type of research reactor is not expected. The closest model representing these conditions was developed by Hochreiter\[9\], and is presented below in Fig. 9. This model shows the fuel channel rewetting after dry-out with a very low flow rate entering the channel. The model was actually developed for high pressure conditions, while in the present case atmospheric pressure prevails.

The transition between the single phase liquid and the DFFB – Dispersed Flow Film Boiling condition is very rapid as even a small amount of steam requires a volume 1600 times larger than the liquid. At high pressure, for example 40 bar, this ratio is only about 40. Consequently one expects rapid creation of large bubbles pushing droplets or slugs of water in the upward direction, may be even out of the channel. This phenomenon was designated as “Percolation” in the Livermore SAR from 1974\[5\]. However, this phenomenon was not indicated or modelled in any subsequent publications.

Figure 9: Flow regime description in a coolant channel after dry-out assuming very low re-wetting flow rate\[9\].
The physical principle which was used in the development of the present model to calculate the flow through the channel assumes that the hydrostatic pressure at the channel entrance must be equal to the static and the dynamic pressure drop created by the flow through the channel. This principle can be expressed by

\[ \rho L_y g = \Delta P_{st} + \Delta P_f + \Delta P_a + \Delta P_{io} \]

Where:
- \( \rho \) – coolant density
- \( L_y \) – pool water level
- \( g \) – gravitational constant
- \( \Delta P_{st} \) – the hydrostatic pressure drop in the channel obtained by integrating the coolant density along the channel: \( g \int \rho (y) \, dy \)
- \( \Delta P_f \) – frictional pressure drop in the channel
- \( \Delta P_a \) – Pressure drop due to accelerating the fluid along the channel
- \( \Delta P_{io} \) – entrance and exit pressure drops

The above equation serves as the basis for the momentum equation. In addition, the mass and energy balance equations were solved for the fuel channel to calculate all the system relevant state variables.

The computational fuel channel model developed for this study is presented schematically in Fig. 10. It is a two-dimensional model having an arbitrary number of nodes in the axial direction and three nodes in the radial direction, namely the fuel, the side plate and the water in the coolant channel. The model describes in detail the axial and radial heat conduction in the fuel, and from the fuel to the side plate and to the bottom grid plate and pool water, and convective heat transfer to the coolant flowing in the fuel channel. This model was included in the DSNP \( \text{\textsuperscript{1}} \) (Dynamic Simulator for Nuclear Power-plants) library, which also includes many power plant components, material properties, heat-transfer and flow correlations \( \text{\textsuperscript{6,7}} \) to be used with the simulation model. Full details of the mathematical model are subject to a detailed report \( \text{\textsuperscript{10}} \). Here only the principles are presented.

![Figure 10: Schematic description of a MTR type fuel channel model.](image-url)
Where:
T – Temperature
H - Enthalpy
w – Flow rate (Erroneously w is shown in the figure as ω)
w – Water flow rate
wd – Flow entering the channel to maintain an equilibrium pressure drop (ω, in the figure)
w – Evaporation rate from water slugs or drops
W – Decay heat power generated in the fuel
Qw,j+1 – Heat transfer between the j-th and the j+1-th axial region
Qw,j-1 – Heat transfer between the j-th and the j-1-th axial region
Qws – Heat convection between the fuel plate and the steam
Qwd – Heat convection between the fuel plate and the water slugs/drops
Qsd – Heat convection between water slugs/drops and the steam
Qpd – Heat convection between side plate and water slugs/drops
Qps – Heat convection between side plate and the steam
γ1 – Part of the decay heat released in water - γ radiation
γ2 – Part of the decay heat released in the side plates
γ = 1 - γ1 - γ2 – Part of the decay heat released in the fuel
Indices
f – fuel
s – steam
sw – saturated water
ℓ - water
w – wall
sp – side plate
j-1, j, j+1, node indexing

The basic heat balance is presented by the two equations below (with the symbols given in Fig. 10)

\[ γW_f - Q_w^{j+1} - Q_w^{j-1} - Q_{sp}^j = Q_{sd} + Q_{ws} \]
\[ γ_2W_f - Q_s^{j+1} - Q_s^{j-1} + Q_{sp}^j = Q_{pd} + Q_{ps} \]

In the present model all radiative heat transfer between the wall and steam/droplets and between adjacent fuel plates is neglected. The water slugs and droplets drifting along the channel are assumed to be at saturation temperature, while the steam is being super-heated. The transfer between the steam and droplets results in additional evaporation along the channel.

Part of the heat generated in the fuel does not reach the coolant in the channels, rather it is conducted to the bottom and top end of the fuel rods. At the bottom it is conducted to the massive aluminium grid plate, and from there to the pool of water. At the top of the channel the heat is conducted to the aluminium part of the fuel rod entrance and then transmitted to the steam escaping from the channel. All these phenomena are included in the present model[10].

5 CONCLUSIONS

A two-dimensional model for a MTR type fuel rod was successfully developed and incorporated into the DSNP simulation package. From the simulation of the partial LOCA event in the MTR fuelled IAEA generic 10MW reactor, it can be concluded that the temperatures during the accident remains significantly below the Al softening temperature and no damage to the core is expected. These results are supported by the experiment performed in Livermore (5).
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Fuel Cycle Safety
CRISTAL: A French Criticality Code Package to Assess AREVA NP Nuclear Installation Criticality Safety

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ABSTRACT

For more than thirty years, CEA (Commissariat à l’Energie Atomique, the French Atomic Energy Agency), IRSN (Institut de Radioprotection et de Sûreté Nucléaire, the Institute for Radioprotection and Nuclear Safety) and the French nuclear industry have been combining their efforts to finance, develop and validate computer codes to assess the criticality safety concerns of nuclear installations, transport casks, and reprocessing facilities. As one of the major world fuel vendors, AREVA NP is deeply involved in defining code developments which incorporate feedback from both users and customers. The result of these continuous efforts is the evolutionary CRISTAL code.

The CRISTAL package was developed as an easy-to-use system using cross-section libraries (JEF 2.2 and CEA93), well-established computer codes (APOLLO2, MORET 4 and TRIPOLI-4) and including a Graphical User-Friendly Interface. The APOLLO2 computer code, a spectral code used for evaluating the basic characteristics of fuel assemblies, has been upgraded to perform criticality safety calculations. The MORET 4 computer code is a neutron simulation code in three dimensions which uses the multigroup formalism for cross-sections and the Monte Carlo method to solve the Boltzmann equation. Through the years, the CRISTAL package has been improved to take into account both the growth of its validation database and the increasing user requirements. Today, CRISTAL V0 is an up-to-date computational tool incorporating the comprehensive APOLLO2 and MORET 4 computer codes; CRISTAL V0 is the result of more than five years of development work focusing on theoretical approaches and on the implementation of user-friendly graphical interfaces.

Thanks to its broad validation database, CRISTAL V0 provides outstanding accuracy of criticality evaluation for configurations covering the entire fuel cycle life (i.e. from fuel enrichment, pellet/assembly fabrication and transport casks to fuel reprocessing). With more than a thousand benchmark/calculation comparisons, uncertainties can be deduced for various file media, fissile shapes, fissile process interactions, neutron-poisoning screens and material reflectors. These uncertainties, combined with suitable modelling features, ensure confidence in the CRISTAL results when justifying sufficient safety margins.

After a brief description of the calculation scheme and the physics algorithms used in the codes, various industrial applications encountered in a UO2 fuel fabrication plant will be discussed.
1 INTRODUCTION

Criticality safety is a concern for fissile material during all stages of the fuel fabrication process. In this paper, typical processes for a UO2 fuel fabrication plant (including UF6 cylinder storage, UF6-UO2 conversion, powder storage, pelletizing, rod loading, and assembly fabrication) are investigated.

Safety implementation is based on criticality analyses and existing subcriticality margins with consideration of actual safety assessments and process requirements. The accuracy of a criticality calculation produces confidence when addressing safety concerns. The CRISTAL code package [1], with its large qualification database, can provide this high level of confidence.

2 THE CRISTAL CODE PACKAGE

The CRISTAL code package was developed during the late nineties by CEA (Commissariat à l'Energie Atomique) and IRSN (Institut de Radioprotection et de Sûreté Nucléaire), the advising body of the DGSNR (Direction Générale pour la Sûreté Nucléaire et la Radioprotection). This project was founded by the French nuclear industry together with the CEA and IRSN.

Figure 1:

The functional architecture of the CRISTAL code package, presented on Figure 1, is organized around two calculation routes:

- A "Standard route" or "Industrial route" with a multigroup formulation of cross-sections of the CEA93 library based on APOLLO2 computer code [2], a spectral computer code, and MORET 4 [3] computer code, a three dimensional Monte Carlo computer code.

These two calculation routes are using JEF 2.2, a basic microscopic cross library, via the CEA93 library with 172 energy group structure for the “standard route” and directly for the “reference route”.

This paper deals with the "Standard" route, the industry's preferred route for criticality safety evaluations.

The "Standard" route uses an input preprocessing Graphical User Interface, CIGALES [5]. CIGALES provides an efficient method to prepare APOLLO2 input decks and simplifies QA activities. Only basic "physical" data are used (fissile and structural media, shape, enrichment, dimension …). The APOLLO2 computer code performs cell or assembly spectrum calculations accounting for sophisticated self shielding process flux and macroscopic cross section determination. The resulting homogenized 172 group energy
structures are directly linked with the three-dimensional Monte Carlo code, MORET 4, which provides the capability to model simple or complex geometries.

The CIGALES computer code
The CIGALES computer code, a data generator used to provide in an interactive way the APOLLO2 assembly code data files for Pij and Sn calculations, was developed with Visual Basic and is available on Windows environment. The CIGALES computer code allows the atomic composition calculations of fissile materials using dilution laws and the generation of data for APOLLO2 calculations:
- for Pij calculations assigned to create macroscopic cross-sections (self-shielded, homogenised and/or collapsed) representing the chemical media used by the MORET 4 computer code (equivalent to media in homogeneous or heterogeneous geometries),
- for 1D and 2D calculations using the Sn method,
- for calculations of “criticality standards” (1D calculation with four predefined reflectors) with Sn method.

Moreover, the CIGALES computer code makes it possible to establish a coupling with the CESAR computer code (which is a simplified computer code for depletion calculations applied to the reprocessing).

The APOLLO2 computer code
The APOLLO2 computer code, developed since 1983 by CEA as a joint development effort with Electricité de France (EDF) and AREVA NP, makes it possible to solve the Boltzmann equation either with the integral form by the collision probability method (Pij, probability for a neutron created in volume Vi to have its first collision in volume Vj) or with the differential form by the Sn method. It also contains physical models to represent self-shielding effect, neutron leakage, double heterogeneity effect, non linear homogenization effect, etc. This modular computer code handles a great number of geometries (from the elementary cell in infinite medium to complex assemblies), and calculates in one and two dimensions the characteristic neutron parameters (such as cross-sections, buckling ...).

The MORET 4 computer code
The MORET computer code, developed since 1970 in the Criticality studies division at IRSN, is a neutron simulation code in three dimensions which uses the multigroup formalism for the cross-sections and the Monte-Carlo method to solve the Boltzmann equation. It allows us to determine the effective multiplication factor (k eff) of any configurations more or less complex in three dimensions as well as reaction rates in the different volumes of the geometry and the leakages out of the system.

In the CRISTAL framework, the MORET computer code has benefited from intensive development and validation work. The improvement of the computing structure quality and the integration of new physical functionalities (anisotropic diffusion of neutrons and loosely coupled fissile units) led to the MORET 4 computer code.

3 APPLICATIONS TO A LOW ENRICHED URANIUM (LEU) UO2 FUEL FABRICATION PLANT

While an LEU fuel fabrication plant is processing enriched uranium lower than 6.6% 235U, the known enrichment bounding criticality risk for non-moderated fissile material, the various processes and production areas may not always remain water-free. Therefore, due to
moderation concerns, processes involving fissile material must be evaluated for criticality safety (normal and accident conditions).

The CRISTAL code package is well suited to handle criticality analyses.

The global fuel assembly fabrication route includes the following processes:

- UF6 → UO2 conversion.
- UO2 powder storage.
- Pelletizing processes.
- Fuel assembly loading.
- Fuel assembly storage.

As far as criticality evaluations are concerned, some of these processes are controlled either by mass, concentration, geometry, storage pitch, neutron poison, or any combination of these listed controls.

Two examples of criticality control methods are provided on the following photos:

![Figure 2: Pitch (Spacing) control](image1)

Figure 2: Pitch (Spacing) control

![Figure 3: Safe layer (Safe slab) of fuel rods](image2)

Figure 3: Safe layer (Safe slab) of fuel rods

**Criticality Standards**

Basic criticality data (i.e., criticality standards) are used as a first step in the determination of sub critical margins for all processes involving fissile materials. There are several worldwide criticality standards, e.g., ARH-600, which the US nuclear industry relies
on. Since 2003, the CRISTAL code package has been used to produce these basic safety values. Some important standard values for a UO2 fuel fabrication plant are given in the following tables where they are compared with older standards.

Table 1: UF6 – HF mixture

<table>
<thead>
<tr>
<th>Minimal criticality parameter</th>
<th>Cylinder diameter (cm)</th>
<th>Spherical volume (l)</th>
<th>Uranium mass (kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td>UF6-HF SEC/T/77.70</td>
<td>90.4</td>
<td>200</td>
<td>115</td>
</tr>
<tr>
<td>CRISTAL (Se)</td>
<td>46.2</td>
<td>156</td>
<td>98</td>
</tr>
</tbody>
</table>

Table 2: UO2F2 – H2O mixture

<table>
<thead>
<tr>
<th>Minimal criticality parameter</th>
<th>Cylinder diameter (cm)</th>
<th>Spherical volume (l)</th>
<th>Slab thickness (cm)</th>
<th>Uranium mass (kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td>UO2F2– H2O SEC/T/74.201</td>
<td>23.9</td>
<td>40</td>
<td>13.7</td>
<td>39</td>
</tr>
<tr>
<td>CRISTAL (Se)</td>
<td>27.4</td>
<td>33</td>
<td>13.4</td>
<td>37.5</td>
</tr>
</tbody>
</table>

Table 3: UO2 – H2O mixture

<table>
<thead>
<tr>
<th>Minimal criticality parameter</th>
<th>Cylinder diameter (cm)</th>
<th>Spherical volume (l)</th>
<th>Slab thickness (cm)</th>
<th>Uranium mass (kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Homogeneous UO2 – H2O CEA-N-2551</td>
<td>25.9</td>
<td>29.1</td>
<td>12</td>
<td>35.4</td>
</tr>
<tr>
<td>CRISTAL (Se)</td>
<td>25.4</td>
<td>27</td>
<td>12.1</td>
<td>35.9</td>
</tr>
</tbody>
</table>

**Actual configuration**

As a consequence of the 1999 Tokai Mura criticality accident, the French safety authority implemented requirements for the reevaluation of several criticality safety items within the nuclear industry. One reevaluation included overloaded storage devices which were examined using CRISTAL. The same configuration was reproduced using MCNP and the results compared.

The configuration modeled is a typical infinite array of powder storage devices (Gemini) containing approximately 500 kg of low moisture content LEU UO2 per Gemini. Calculations demonstrate that even in the case of overloading these storage devices with 582 kg of LEU UO2 per Gemini, the configuration maintains a reactivity (Keff) lower than 1.0.

The computed configuration is shown Figure 2:
Figure 4:

Table 4 shows the main results of the CRISTAL code package compared with the MCNP results for the upper bounding cases.

Table 4: CRISTAL vs. MCNP results

<table>
<thead>
<tr>
<th>Concrete water content (%)</th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Water layer (cm)</td>
<td>CRISTAL</td>
<td>MCNP</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Keff+3σ</td>
<td>σ (pcm)</td>
<td>Keff+I+</td>
</tr>
<tr>
<td>8.9</td>
<td>0.5</td>
<td>0.903</td>
<td>232</td>
</tr>
<tr>
<td></td>
<td>1</td>
<td>0.918</td>
<td>234</td>
</tr>
<tr>
<td></td>
<td>1.5</td>
<td>0.916</td>
<td>245</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>0.897</td>
<td>231</td>
</tr>
<tr>
<td>4.7</td>
<td>0.5</td>
<td>0.951</td>
<td>221</td>
</tr>
<tr>
<td></td>
<td>1</td>
<td>0.972</td>
<td>275</td>
</tr>
<tr>
<td></td>
<td>1.5</td>
<td>0.962</td>
<td>245</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>0.936</td>
<td>243</td>
</tr>
<tr>
<td>3.0</td>
<td>0.5</td>
<td>0.958</td>
<td>214</td>
</tr>
<tr>
<td></td>
<td>1</td>
<td>0.981</td>
<td>264</td>
</tr>
<tr>
<td></td>
<td>1.5</td>
<td>0.979</td>
<td>256</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>0.960</td>
<td>248</td>
</tr>
</tbody>
</table>

* Uncertainty I = 1.645 * sqrt(0.0065²+σ²) where 0.0065 is derived from MCNP – JEF2.2 calculations
4 CONCLUSION

This paper demonstrates the capability of the CRISTAL code package to handle criticality safety concerns for the typical operation of a LEU UO2 fuel fabrication plant. The package uses sophisticated algorithms to solve the transport equation coupled with an up-to-date microscopic cross section set (JEF 2.2 level) produced by the APOLLO2 spectrum code. The three dimensional Monte Carlo computer code MORET4 allows the simulation of complex geometries encountered in production facilities. Finally, the Graphical User Interface, CIGALES, provides efficient preparation of the APOLLO2 input data and improves the QA process for the entire package.

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ABSTRACT

The transport security is very important for AREVA group and the nuclear business in general. Because the transports are on the public field they are more sensitive for the persons and the environment.

In order to optimize the AREVA group transport security and the effective implementation of nuclear safety charters the Business Unit [BU] Logistics puts in place in 2006 a specific organization with a reinforced team and some new processes.

The scope is the transportation of nuclear materials and contaminated equipment representing a specific risk to the group (safety, physical protection, industrial and media)

The main BU LOGISTICS assignments are:
- to provide customized services to every BU of the group
- to certify and control external subcontractors chosen with BU’s consent
- to supply all support for BU’s in the field of:
  - Transportation preparation and organization
  - Emergency management
  - Monitoring of regulations
  - Technical expertise associated with transportation

This paper will introduce the approach of the risk analysis. The approach is in three steps:

- The transportation flow risk analysis,
- The Implementation priority
- The proposal for corrective actions.

For risks analysis, we evaluate, for each transportation flow, the occurrence or probability of an event during transport and the consequence of any potential event. From this, we get a risk level for each transport flow.

For the implementation priority, based on the risk level already evaluated in step 1, we evaluate and grade the current management of the risk. So, for each transportation flow, we have the risk level and the current level of the risk management.

For the last part we propose corrective actions for priority flows.
INTRODUCTION

The safe transport of radioactive materials has always been a priority concern for AREVA; today, and to meet tomorrow’s challenges, working toward transport related risk reduction has become a major enterprise for AREVA which has entrusted its Logistics Business Unit to conduct this effort: AREVA’s Transport Securization.

This paper describes in a first paragraph the context which has led the Logistics Business Unit to conduct transport securization. In a second part, the paper outlines the general principle that governed the designing of the organization in place supporting this effort. The last part describes the organization and methodology of the Logistics Business Unit transport securization.

CONTEXT

Active in every stage of the fuel cycle, with actual production and manufacturing activities in 41 countries, and with two-third of its sales revenues depending on outside of France activities, the AREVA group manages numerous international streams of class 7 material between its plants as well as to and from other industrial plants worldwide.

Today, over 600 different steams of transports have been identified and are being part of the Logistics Business Unit transport securization perimeter. This number is increasing as nuclear renaissance springs around the globe. Indeed, AREVA faces increased demand for transportation, reaching out to new countries and regulations, involving new routes, new transport means, new actors within the logistical chain, and eventually new types of transport packages.

It is in this context, and to contribute to AREVA’s sustainable worldwide growth, that the Logistics Business Unit efforts are being conducted.

GENERAL PRINCIPLES OF THE ORGANIZATION: CREATING A GLOBAL CENTRE OF EXCELLENCE

The predominant element that governed the designing of the organization in place for transport securization relates to human resources and their allocation to specific tasks at specific times. In an emergency situation, the organization shall allow for experts to be mobilized at any time of the day, any day of the year, and to be driven to efficiently work on their field of expertise. For that organization to be effective, it must be tested, feedback must be analyzed, and implemented back in the organization as needed. Simulation and training is therefore essential: training programs and large scale emergency response drills are specifically designed to improve methodology and effectiveness of experts involved in emergency situations. Good communication and harmonization of terminology are key factors of success when decisions are to be made in restricted timeframe.

Initiative was taken to create a dedicated organization implementing a series of actions aimed at reducing risks related to specific transports. Such organization is not only using existing local Logistics Business Unit resources and experience, but also seeking to use the advanced skills present in the nuclear market including AREVA entities worldwide.
The organization as described in the following paragraph structures the existing skills and expertise forming a harmonized global centre of excellence for nuclear logistics supporting safe, reliable, and efficient transportation of nuclear material and extending the actual safety records to face the challenges of nuclear renaissance.

4 ORGANIZATION AND METHODOLOGY: PREVENTING EMERGENCY SITUATIONS, MANAGING EMERGENCY SITUATIONS

Backed-up by over 30 years experience and comprised of a worldwide staff of over 800 employees fully dedicated to transport and package engineering, the Logistics Business Unit has designed and implemented its own methodology for preventing and managing emergency situations.

Four pillars support such methodology:

a) Control of Sub-contractors’ Chain of Transports
b) Risk Analysis and Risk Management
c) Studies and Support
d) Emergency response

The first three pillars relate to preparedness for preventing any emergency situation, whereas the last one describes the organization in place to perform operational emergency response. All four are complementary and interdependent.

Control of Sub-contractors’ Chain of Transports

Sub-contractors’ control is achieved by implementing tight surveillance and an inspection program during transport on pre-qualified sub-contractors. Audits for qualification of sub-contractors are conducted by the Logistics Business Unit certified experts operating an active surveillance and backed-up by an inspection program of physical on-site checks at specific stages of the transport chain. Determination of the specific stages to be inspected and the periodicity of inspections are derived from the risk analysis performed for each transport stream. Inspection reports and their follow-ups feed back the risk analysis (described in the next paragraph); the risk analysis updates the inspection program, thus closing the cycle for a live continuous surveillance.

Risk Analysis and Risk Management

The Logistics Business Unit conducts a risk analysis on every AREVA transport stream. Following the risk analysis, proper recommendations are elaborated for reducing such risks.

The risk analysis establishes the probability of occurrence of an event during transport and the impact of such events on AREVA activities. Transport related events can be attributed to impacting safety, security, the media, or industrial streams.

Combining occurrence and impact provides ground for defining the risk level and the priority of action to engage on recommendations for lowering the risk. Recommendations range from technical solutions involving the development of new transport routes, packages, transport means, tie-down systems, modes of transport, but also improved fleet management, logistics, industrial partnership, processes, training, etc.
Studies and Support

The Logistics Business Unit’s organization for transport securization provides support and expertise in fields such as:
- Alternative transport routes
- Public acceptance
- Emergency procedures and operations
- Sustainable development in the field of transport
- Large scale logistical studies
- Package licensing strategy
- Package fleet management
- Public acceptance
- Training,
- Tracking systems
- Tie-down systems
- etc.

Emergency Response

One of the principles among those developed by the IAEA for the safe transport of radioactive material is that emergency preparedness enables to reduce the radiological consequences to persons and the environment, in case of accident.

The IAEA Regulations for the Safe Transport of Radioactive Material require that emergency provisions, as established by relevant national organizations, shall be observed to protect persons, property and the environment.

To help public authorities in charge of emergency response to establish adapted emergency plans, the IAEA published a Safety Guide. This Safety Guide was published for the first time in 1988. The current edition was published in 2002 under the reference TS-G-1.2.

In France, the prefect of the department where the accident occurs is responsible for decisions and measures required to ensure the protection of both population and property at risk.

During an accident, the ministers concerned provide the prefect with recommendations and information, in order to help him make the proper decisions.

The nuclear industry and transport companies also have to be prepared to intervene and to support the authorities at their request, depending on their specialities and their capacities.

The Logistics Business Unit emergency response organization is aimed at reducing transport related risks for the AREVA Group as well as at supporting authorities in the fields of packaging, transport means, contamination and irradiation risk evaluation, proximity expertise and evaluation (with its on-site mobile technical team), and communication.
Emergency preparedness is about providing decision makers with timely and reliable information. For that purpose, the organization provides for the joint effort of both the technical team and the communication team. Making the right decision shall be based on a very quick estimation of the potential consequences of the accident. To be able to evaluate such consequences, a good knowledge of the particular transport, accidental conditions, and material being transported is essential (drop, fire, immersion, duration of the accident, area with impact on the population and the environment).

The organization efficiency is measured against its capacity to achieve the objective of returning under safe conditions. At an early stage, scenarios for recovery of the damaged packages are elaborated by the technical team. Confirming the safety of the packages is a prerequisite to finalizing the recovery scenarios.

Data must be checked prior to release and communication. Communication should allow for simple and clear messages utilizing a harmonized terminology which must be established in advance and tested.

To prepare the emergency teams properly and acquire effective emergency plans, the Logistics Business Unit has been actively participating to regular training exercises with various ministerial department, the nuclear industry, members of the public and the media. Feedback from such training exercises is taken into account to improve the emergency procedures.

The table 1 below summarizes the last large scale national drills performed with the Logistics Business Unit.

<table>
<thead>
<tr>
<th>Materials transported</th>
<th>2002</th>
<th>2003</th>
<th>2004</th>
<th>2005</th>
<th>2006</th>
<th>2007</th>
</tr>
</thead>
<tbody>
<tr>
<td>Research used fuel</td>
<td></td>
<td></td>
<td>Used fuel and MOX used fuel</td>
<td>Liquid waste</td>
<td>Enriched UF6</td>
<td>Alpha technological wastes</td>
</tr>
<tr>
<td>Low level waste</td>
<td>Road</td>
<td>Road</td>
<td>Rail</td>
<td>Road</td>
<td>Road</td>
<td>Road</td>
</tr>
<tr>
<td>Road</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Packaging</td>
<td>IU 04</td>
<td>DV 78</td>
<td>TN 12</td>
<td>TN CIEL</td>
<td>30B cylinder</td>
<td>RD26</td>
</tr>
<tr>
<td>LEMARECHAL CELESTIN (LMC)</td>
<td>LEMARECHAL CELESTIN (LMC)</td>
<td>SNCF</td>
<td>LEMARECHAL CELESTIN (LMC)</td>
<td>LMC</td>
<td>LMC</td>
<td></td>
</tr>
<tr>
<td>Shipper</td>
<td>CEA Saclay center</td>
<td>AREVA NC La Hague plant</td>
<td>EDF Chinon NPP</td>
<td>EDF Paluel NPP</td>
<td>EURODIF</td>
<td>MELOX</td>
</tr>
<tr>
<td>AREVA NC Cadarache</td>
<td>Centraco incineration plant</td>
<td>AREVA NC La Hague</td>
<td>Centraco incineration plant</td>
<td>GNF</td>
<td>LANL</td>
<td></td>
</tr>
<tr>
<td>Consignee</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Transport agent (commissionn ing)</td>
<td>TN International</td>
<td>TN International</td>
<td>TN International</td>
<td>TN International</td>
<td>TN International</td>
<td>TN International</td>
</tr>
<tr>
<td>Area</td>
<td>Yonne (Auxerre)</td>
<td>Eure-et-Loire (Chartres)</td>
<td>Indre-et-Loire (Tours)</td>
<td>Val d’Oise (Cergy-Pontoise)</td>
<td>Roanne (Loire)</td>
<td>Montoir (Loire Atlantique)</td>
</tr>
</tbody>
</table>
Feedback from each training exercise or actual emergency situation often relates to improving communication and reactivity. Feedback has been very beneficial and has largely influenced the actual transport securization organization.

CONCLUSION

AREVA has engaged in a major enterprise for transport related risk reduction to sustain its worldwide growth and properly manage increasing transport activities induced by nuclear renaissance.

A dedicated organization was created, tested, and improved, leading to an effective centre of excellence for nuclear logistics supporting safe, reliable and efficient transportation of nuclear material.
ABSTRACT

The isothermal and transient oxidation behaviour of the four widely used cladding materials, the Zr-Sn alloys Zry-4 and DUPLEX and the Zr-Nb alloys E110 and M5 in steam, oxygen and air were investigated. The oxidation kinetics at temperatures above 1050°C is similar for these materials. Parabolic time dependences of mass increase and oxide layer growth were found. The oxidation rates depend on temperatures by Arrhenius functions. The activation energies differ for the different zirconium oxide crystal structures. At lower temperatures the behaviour can strongly differ between the materials due to the so called “breakaway effect”. In the temperature range between 800 and 1000°C this effect was found for oxidation of E110 in steam and oxygen. At 1000°C the oxidation of Zry-4 shows this effect, too. It results in an enhanced oxidation. The time dependence changes from parabolic to nearly linear. For the D4 layer of the DUPLEX material and for M5 no breakaway takes place under the conditions applied. The effect was also found for transient oxidation of E110 with heating rates below 0.1 K/s. For these low heating rates the time in which the materials are in the temperature range where the breakaway effect takes place is long enough for enhancing oxidation. The strong influence of spalling of oxide layers on the severe accident behaviour of fuel rod bundles can be seen by comparison of the large scale bundle simulation tests QUENCH-6 and QUENCH-12. The hydrogen release of the QUENCH-12 bundle during reflooding, where massive spallation of oxide parts took place, was six times higher than of the QUENCH-6 bundle under comparable conditions. Oxidation in air shows faster kinetics than in pure oxygen or steam. The reaction depends nearly linear on time. It is caused by the formation of a very porous oxide scale mixed with zirconium nitride which is formed under (local) oxygen starvation conditions, e.g. at the phase boundary between oxide and metal.
1 INTRODUCTION

Advanced cladding materials were developed for longer operation times in nuclear power plants and extended burnup of the fuel elements. They are optimized regarding their corrosion behaviour under operational conditions and were also tested for LOCA (loss of coolant accident) and RIA (reactivity-initiated accident) conditions by the manufacturers. However, also the oxidation behaviour at severe accident conditions (accidents beyond the loss of coolant accident LOCA) has to be known to prove the safety under these circumstances and to improve models used in severe accident simulation codes. Basis of the investigations are loss of coolant and reflooding scenarios.

The overheated cladding material reacts rapidly with steam at high temperatures. Simplified it can be described by:

\[
\text{Zr} + 2 \text{H}_2\text{O} \rightarrow \text{ZrO}_2 + 2 \text{H}_2 + 596 \text{kJ} \quad (1)
\]

It is a strongly exothermic reaction which causes additional increase of temperature in the reactor. The steam oxidation results in degradation of the metallic cladding material and in release of a large amount of hydrogen.

In contrast to the classical Zircaloy-4 (Zry-4) which is extensively investigated over a wide temperature range from operational conditions to temperatures beyond design basis accident [1], the publicly available data on high temperature oxidation of the various advanced cladding materials is scarce.

2 MATERIALS AND EXPERIMENTS

2.1 Materials

The investigations comprise two Zr-Sn alloys (D4 layer at the DUPLEX cladding and Zry-4) and two Zr-Nb alloys (E110 and M5). Whereas Zry-4, M5 and E110 are homogeneous materials, the DUPLEX material consists of a Zry-4 bulk and a D4 protection layer (thickness 150 µm) at the outer surface. The D4 layer differs from Zry-4 mainly by a reduced tin content and a higher concentration of iron and chromium. The main difference between the both Zr-Nb alloys is the higher iron content in M5. The concentrations of the main alloyed elements in the investigated materials are given in Table 1.

<table>
<thead>
<tr>
<th>alloy</th>
<th>Sn</th>
<th>Nb</th>
<th>Fe</th>
<th>Cr</th>
<th>O</th>
</tr>
</thead>
<tbody>
<tr>
<td>E110</td>
<td>&lt; 0,04</td>
<td>1,00</td>
<td>&lt; 0,01</td>
<td>&lt;0,003</td>
<td>0,05</td>
</tr>
<tr>
<td>D4</td>
<td>0,50</td>
<td>0,0001</td>
<td>0,50</td>
<td>0,20</td>
<td>0,14</td>
</tr>
<tr>
<td>M5</td>
<td>&lt; 0,03</td>
<td>1,00</td>
<td>0,34</td>
<td>0,04</td>
<td>0,14</td>
</tr>
<tr>
<td>Zry-4</td>
<td>1,50</td>
<td>0,0001</td>
<td>0,21</td>
<td>0,10</td>
<td>0,14</td>
</tr>
</tbody>
</table>

Table 1: Chemical composition of the cladding alloys, Zr - balance

2.2 Separate-Effect Tests

For the separate-effect tests different facilities were used: a thermo-balance with two furnace frames, one for reaction in oxygen or air with maximal temperature of 1600°C and one for steam oxidation with maximal temperature of 1100°C, and a horizontal tube furnace
for isothermal oxidation in steam. For the tests segment with a length of 10 or 20 mm were cut from original cladding tubes.

The specimen has to put into the thermo-balance at room temperature, then it was heated in flowing Ar atmosphere. When the test temperature (isothermal tests) or start temperatures (transient test) were reached the injection of the oxidizing gas and the mass registration was started. After the pre-defined oxidation time the injection of the reaction gas was finished and the specimen was cooled down to room temperature in flowing Ar.

The horizontal tube furnace provides the possibility of loading the specimen into the furnace at test temperature in flowing Ar atmosphere. After some seconds for temperature homogenisation the test gas injection starts. Reducing (H₂), inert (Ar, He) or oxidizing (air, O₂, steam and mixtures) atmospheres can be applied. The off-gas composition was measured with the mass spectrometer “GAM 300”. After the test the specimen can be cooled down to room temperature in Ar within about five minutes.

2.3 Large-Scale Bundle Tests

Quench experiments with overheated nuclear fuel rod bundle simulators were applied to study the behaviour of the reactor core during reflooding in various LOCA or severe accident scenarios. The vertical mounted fuel rod simulators have a length of 2500 mm. A part of them are electrical heated. Up to now 13 QUENCH-experiments were performed; differing in their accident scenario (e.g. air ingress, boil-off, slow cooling), in the applied reactor geometry and in the applied cladding materials including different types of control rods.

The QUENCH-12 experiment [2] was carried out to investigate the effects of VVER materials (niobium-bearing alloys) and bundle geometry on core reflood, in comparison with test QUENCH-06 using Western European PWR simulator bundle (Zircaloy-4) [3]. QUENCH-12 was conducted with largely the same protocol as QUENCH-06 given in Fig. 1, such that the effects on the VVER characteristics could be observed more easily.

While the PWR bundle uses a single unheated rod, 20 heated rods, 1 unheated central rod and 4 corner rods arranged on a square lattice, with a heated length of 1000 mm, the VVER bundle uses 13 unheated rods, 18 heated rods and 6 corner rods, arranged on a hexagonal lattice. The coolant channel area ratio between two bundles is QUENCH-12/QUENCH-06 = 1.09, therefore the fluid flow rates were preset 9 % higher for the QUENCH-12 bundle than for the QUENCH-06 bundle to provide the same flow velocity.

![Figure 1: Temperature at the 950 mm elevation and electric power vs. time together with an indication of test phases](image-url)
The bundle material mass ratio is QUENCH-12/QUENCH-06 ~ 0.97, however the metallic surface ratio is QUENCH-12/QUENCH-06 = 1.22. In this connection the electrical power for the QUENCH-12 bundle was installed lower than for the QUENCH-06 bundle to compensate the higher chemical energy production due to exothermic steam-metal reaction.

The steam, ascending from the bundle bottom, is heated and produced a pronounced axial temperature profile. The axial temperature distribution during the pre-oxidation phase given in Fig. 2 shows that the greater part of the QUENCH-12 bundle was oxidised long period at temperatures between 1000 K and 1300 K, i.e. under conditions typical for breakaway oxidation (see below) of the E110 alloy.

3 RESULTS AND DISCUSSION

3.1 Isothermal Oxidation Behaviour

3.1.1 Steam oxidation

The oxidation of all materials can be described by the parabolic time dependences expected for diffusion controlled reactions for all temperatures investigated at least for short oxidation periods:

$$\Delta m_{oxide} = \delta_{m,D} \cdot \sqrt{t} \quad (2)$$

$\Delta m$ is the mass increase per surface area, D the oxide layer thickness. $\delta_m$ and $\delta_D$ are the mass increase rate and the oxide layer growth rate, respectively. As an example, Fig. 3 gives the time dependence of the relative mass increase during steam oxidation at 900, 1000, 1100 and 1400°C for the four materials investigated. Because the oxidation occurs at both, inner and outer surfaces, the mass increase of the DUPLEX material is a result of oxidation of D4 and Zry-4 at outer and inner surface, respectively.

In Fig. 4 the temperature dependences of the mass increase rate and of the oxide layer growth rates are given for the investigated materials. The time dependences of oxidation rate $\delta$ for each material can be described by two Arrhenius functions, valid for the monoclinic (lower temperatures) and tetragonal (higher temperatures) ZrO$_2$, respectively:

$$\delta_{m,D} = \delta_{m,D}^* \cdot e^{\frac{Q}{RT}} \quad (3)$$
$Q$ is the activation energy and $R$ the universal gas constant ($8.314 \text{ J mol}^{-1} \text{ K}^{-1}$). For comparison the Leistikow – Schanz and the Cathcart – Pawel correlations [4] are given in the diagrams. They better describe the behaviour of the Zr – Nb alloys than of the Zr – Sn alloys.

Tab. 2 gives the activation energies determined for all investigated materials at the lower and the higher temperature ranges. The transition temperature between the monoclinic and tetragonal ZrO$_2$, is about 50 K higher for the both Zr – Nb alloys than for the Zr – Sn alloys. Due to the low oxide layer thicknesses at temperatures below this transition (only several microns), the uncertainties in the metallographic determination of the thicknesses are high. Therefore a certain determination of the activation energies for the oxide layer growth is not possible.

At 1400°C the oxidation behaviours of the four materials are quite similar. The oxidation of the Zr – Nb alloys is slightly faster than of the Zr – Sn alloys. Due to the higher activation energies of the Zr – Nb alloys the situation changes at temperatures between 1000 and 1200°C. In this temperature range the oxidation rate is lower for the Zr - Nb alloys than Zr - Sn alloys. With increasing temperature the oxidation behaviour of the D4 layer converges towards the higher oxidation rate of Zry-4. Reason for this convergence is the diffusion of tin from the Zry-4 bulk into the D4 layer which is relative fast, at least at temperatures of 1200°C and higher.

Figure 3: Comparison of the time dependence of the relative mass increase during steam oxidation of the materials Zry-4, Duplex, E110 und M5 at 900, 1000, 1100 and 1400°C.
Figure 4: Temperature dependence of mass increase rate $\delta_m$ and oxide layer growth rate $\delta_D$ of the investigated materials

Fig. 5 shows the tin concentration profile after 3 h annealing at 1100°C in flowing argon. The profile was determined by X-ray fluorescence measurements with a beam size of 8 µm at the FLUO/TOPO beamline of the synchrotron source ANKA of FZ Karlsruhe. The diffusion rates in the DUPLEX material determined for temperatures between 1000 and 1400°C are significantly higher than the values given in [5] for pure zirconium. On the other hand the activation energy for tin diffusion is in the DUPLEX material significantly lower than the value given in [5] (DUPLEX: 138 kJ/mol, zirconium [5]: 212 kJ/mol). A detailed description and discussion of the results will be given in [6].

At 1000°C the oxidation rates of D4 and Zry-4 differ significantly. Reason is the breakaway effect, discussed below, which occurs in Zry-4 but not in the D4 layer. At lower temperatures the D4 layer shows its protective behaviour against oxidation. At these temperatures tin diffusion does not affect the oxidation rate significantly.

The slowest oxidation takes place in M5. The oxidation in E110 is enhanced between 800 and 950°C also at early times for which the parabolic kinetics is valid. Possibly, the beginning of the breakaway effect results in a faster oxidation.

<table>
<thead>
<tr>
<th>material</th>
<th>temperature range</th>
<th>mass increase</th>
<th>oxide layer growth</th>
</tr>
</thead>
<tbody>
<tr>
<td>Zry-4</td>
<td>800°C ≤ T &lt; 950°C</td>
<td>$\delta_m^{Zry-4} = 2.857 \cdot e^{-12655 K/T}$</td>
<td>not to be determined</td>
</tr>
<tr>
<td></td>
<td>1000°C ≤ T ≤ 1400°C</td>
<td>$\delta_m^{Zry-4} = 0.298 \cdot e^{-9033.3 K/T}$</td>
<td>$\delta_D^{Zry-4} = 0.213 \cdot e^{-9277.1 K/T}$</td>
</tr>
<tr>
<td>Duplex/D4</td>
<td>800°C ≤ T &lt; 950°C</td>
<td>$\delta_m^{Duplex} = 1.114 \cdot e^{-11718 K/T}$</td>
<td>not to be determined</td>
</tr>
<tr>
<td></td>
<td>1000°C ≤ T ≤ 1400°C</td>
<td>$\delta_m^{Duplex} = 0.409 \cdot e^{-9614.4 K/T}$</td>
<td>$\delta_D^{Duplex} = 0.398 \cdot e^{-10472 K/T}$</td>
</tr>
<tr>
<td>E110</td>
<td>800°C ≤ T &lt; 1000°C</td>
<td>$\delta_m^{E110} = 0.240 \cdot e^{-9945.6 K/T}$</td>
<td>not to be determined</td>
</tr>
<tr>
<td></td>
<td>1050°C ≤ T ≤ 1400°C</td>
<td>$\delta_m^{E110} = 1.263 \cdot e^{-11332 K/T}$</td>
<td>$\delta_D^{E110} = 1.149 \cdot e^{-12070 K/T}$</td>
</tr>
<tr>
<td>M5</td>
<td>800°C ≤ T &lt; 1000°C</td>
<td>$\delta_m^{M5} = 0.080 \cdot e^{-8770.6 K/T}$</td>
<td>not to be determined</td>
</tr>
<tr>
<td></td>
<td>1050°C ≤ T ≤ 1400°C</td>
<td>$\delta_m^{M5} = 0.623 \cdot e^{-10170 K/T}$</td>
<td>$\delta_D^{M5} = 1.149 \cdot e^{-12070 K/T}$</td>
</tr>
</tbody>
</table>

Table 2: Parameter of the Arrhenius temperature dependence of mass increase and oxide layer growth
The parabolic oxidation kinetics is typical for reactions were oxygen has to diffuse through a growing oxide scale. The oxidation time dependences of E110 in the temperature range between 800 and 1000°C and of Zry-4 at 1000°C (see as examples Fig. 6) differ from the parabolic behaviour. At later times (4h for E110 at 800°C, 1 h for Zry-4 at 1000°C) the oxidation is enhanced. The oxidation kinetics changes from parabolic to nearly linear.

Differences from the parabolic time dependence of the mass increase of the DUPLEX material are caused by the oxidation of the inner (Zry-4) surface. For M5 only parabolic behaviour was found at all temperatures investigated.

The reason is the so called "breakaway effect". Due to lattice coherence and fitting stresses to the metal the oxide growth starts with an under-cooled tetragonal micro-structure. After a certain oxide layer thickness is reached a martensitic transformation from tetragonal to monoclinic oxide occurs. This transformation is connected with volume change and formation of residual stresses, micro cracks in hoop direction (see Fig. 7) and spalling of oxide parts, as shown in Fig. 6. The oxide layer looses its protective effect against further oxidation. Steam can penetrate into the cracks and the oxidation is no longer controlled by the diffusion of oxygen through the ZrO₂ scale.

Breakaway of oxide layer parts was found for E110 specimens after oxidation at a wide temperature range of 800 to 1000°C, for Zry-4 only at 1000°C. For M5 and D4 no breakaway is found at the temperatures and times investigated. Here the higher content of iron and chromium containing second phase particles increases the mechanical stability of the oxide layers.

Figure 5: Tin concentration profile in DUPLEX cladding in the as-received state and after 3 h annealing at 1100°C in inert atmosphere

Figure 6: Comparison of oxide appearance after steam oxidation with and without breakaway effect
3.1.2 Air oxidation

Actually, various scenarios have attracted interest in which the fuel rods are prone to exposition in air-containing atmospheres. Presence of air increases to the risk of an accelerated escalation basically due to stronger heat released in Zr oxidation; and additionally causes the formation of uranium oxide phases with lower melting temperature and of more volatile oxidic fission products (such as ruthenium oxides). An exemplary scenario could arise under shutdown conditions when the reactor coolant system is open to the containment atmosphere.

Systematic investigations on air oxidation under such conditions were performed by parametric separate-effects tests in the temperature range 800-1500°C [7]. These findings were cross-checked for its applicability to the outcome of the large-scale bundle test QUENCH-10 on air ingress performed in 2004 [8].

Fig. 8 shows the typical parabolic oxidation kinetics of Ziraloy-4 in oxygen caused by the growing compact oxide scale which acts as a diffusion barrier. The reaction kinetics in nitrogen is also of the parabolic type, but by orders of magnitude lower as compared to
oxygen.
The mixture of both gases, i.e. air, gives a much faster kinetics of a linear character. This is caused by the formation of a very porous oxide scale mixed with zirconium nitride (gold-coloured phase) which is formed at the phase boundary oxide-metal.

Similar studies on the oxidation of Zircaloy-4 in mixed air-steam and nitrogen-steam atmospheres and on the influence of pre-oxidation in steam on subsequent reaction in air and nitrogen confirmed the significant effect of nitrogen onto cladding oxidation and degradation under the conditions of a severe accident, although oxide formation is thermodynamically favoured.

This can be explained by the role of nitride phases which are formed under oxygen starvation conditions. The nitride is re-converted into oxide under returning oxidising conditions. The strong degradation seen in many of the experiments is due to the significantly different densities of ZrO₂ and ZrN.

Regarding modelling of air ingress in severe accident computer codes, it is suggested that parabolic correlations for oxidation in air should be applied only for high temperatures (>1400 °C) and for pre-oxidised cladding (≥ 1100 °C). For all other conditions, faster reaction kinetics is more appropriate.

3.2 Transient Oxidation Behaviour

In order to prove the possibility of the breakaway effect under transient conditions a limited number of oxidation tests in steam with varying heating rates between 0.05 - 0.3 K/s were performed. Start and end temperature were 400 and 1100°C, respectively.

Oxide breakaway was found in the transient tests only for E110. Spalled oxide layer parts are visible for all applied heating rates as Fig. 9 shows. At heating rates < 0.1 K/s the oxidation is accelerated significantly inside breakaway temperature region, as demonstrated in Fig. 10. The broken line indicates the time dependence to be expected for parabolic behaviour.

Transient tests in oxygen up to 1580 °C have been additionally conducted in a thermal balance. Starting temperatures of oxidation were 1100 °C and room temperature to examine transient oxidation excluding and including the breakaway region, respectively.

Fig. 11 compares the oxidation rates of four alloys in transient tests from 1100 °C. Heat up to 1100 °C was in inert atmosphere. At a first glance the curves look similar. As was seen in the 1100 °C isothermal tests, the niobium-bearing alloys show a slightly favourable behaviour during the initial period of the transient tests.
Figure 9: Post test appearance of E110 specimens after transient oxidation at various heating rates.

Figure 10: Relative mass increase during transient oxidation tests in steam

At about 1300 °C the situation changes and the tin alloys exhibit slower oxidation rates. The reaction rates of the tin and niobium alloys among themselves are very similar. At about 1480 °C the reaction rates significantly increase for all alloys. The reason for this behavior will be discussed later.

The picture is slightly different for the test series starting with oxidation from room temperature as can be seen in Fig. 12. Although the temperature program was identical for the four experiments (shown by the dotted curves in Fig. 12), the resulting TG curves look less similar to each other than those from in the isothermal series. Again, the tin alloys reveal very similar oxidation kinetics over the whole temperature range. The E110 curve starts to show irregularities at 930 °C, but recovers at 1100 °C. The highest oxidation rates for the mid temperature region are measured for M5, the lowest for E110. This relation reverses above 1480 °C.

Figure 11: Reaction rate vs. time during transient oxidation of zirconium alloys in oxygen from 1100 to 1580 °C
Generally, three domains can be distinguished in the diagram. The increase in reaction rates with temperature reveals inhomogeneities, i.e. bends in the TG curves, at about 1000 °C and identically to the first series at 1480 °C, as indicated by the dotted lines in Fig. 12. The three domains are related to the three crystallographic modifications of the zirconium oxide scale, namely monoclinic, tetragonal, and cubic.

3.3 Bundle Simulation Tests

The examination of the VVER bundle QUENCH-12 with E110 cladding revealed significant differences compared to the reference test QUENCH-06 with Zircaloy-4 cladding.

The first corner rod D of the QUENCH-12 bundle, which was withdrawn at the end of the pre-oxidation phase, revealed an extensive breakaway oxidation along the complete hot zone. It was not possible to measure the oxide layer thickness due to spalling of the oxide scales (Fig. 13). The second corner rod F was withdrawn during the transient phase before starting the moderate temperature escalation. This rod also exhibited an extensive spalling of oxide scales.

The QUENCH-12 bundle was investigated in detail by videoscope before filling with epoxy resin (Fig. 14). Axial differences in the surface morphology were observed. The lowest elevation, where breakaway oxidation of Zr1%Nb-cladding surface took place, was at 400 mm. The maximum temperature at this bundle position was about 850°C. The formation of typical breakaway oxidation at the relatively cooler Zr2.5%Nb-shroud took place at higher elevations.
It should be mentioned that the findings on breakaway effect of the VVER-type cladding material reported in this section refer to long term oxidation at temperatures between 730 and 1030°C. Such scenarios are more applicable for accident conditions for spent fuel pool than for reactor accidents.

The initially coarse shroud surface revealed thicker spalled oxide scales, but the oxide sub-layer showed the regular dark structure similar to the oxide inner sub-layer on the cladding surface.

The shroud surface showed at the higher hottest elevations a nodular kind of breakaway oxidation, whereas there is no evidence of breakaway on the cladding surface at these elevations. However the formation of longitudinal and circumferential cladding cracks in the hot bundle zone (0.70-1.00 m) is typical for Zircaloy-4 cladding as well.

Post-test metallographic investigations of the QUENCH-12 bundle showed an influence of the breakaway effect with extensive spalling of oxide scales at rod claddings and shroud at elevations higher than 400 mm. No influence of the breakaway effect was observed for the QUENCH-06 bundle (Fig. 15).

It is interesting to note that the surface of claddings of the QUENCH-12 bundle showed more regular and homogeneous structure of the oxide layer than the surface of the solid corner rods of this bundle. Both surfaces show breakaway oxidation being more pronounced at the corner rods. One possible reason for it could be the different mechanical properties of cladding tubes and solid corner rods.

Measurements of hydrogen production during the QUENCH-12 test are as follows: 34 g were released during the pre-oxidation and transient phases and about 24 g in the quench phase. The amount released in the quench phase is six times higher than in QUENCH 06 with about 4 g (Fig. 16). The reasons for the increased hydrogen production may be extensive damaging of the cladding surfaces due to the breakaway oxidation, local melt formation with subsequent melt oxidation, and the release of hydrogen previously absorbed by the metal. The hydrogen absorbed by claddings was measured by the hot extraction method (long heating of 2 cm probes by 1500 °C). This measurement showed higher concentration of absorbed hydrogen for the Zr1%Nb claddings in comparison to the Zry-4 claddings (Fig. 17).
The cracked oxide layer, which was intensively damaged on the surface of VVER claddings due to breakaway effect, has not more protective function and provides a high hydrogen penetration rate.

4 FUTURE PROGRAMME QUENCH-ACM

A new series “QUENCH-ACM” investigating test bundles with advanced cladding materials, i.e. M5, Duplex D4, ZIRLO, has been defined to be tested at elevated temperatures with subsequent quenching.

Besides the precursor VVER-type experiment QUENCH-12 which was already conducted in 2006, the QUENCH-ACM test series comprises three experiments, i.e. QUENCH-14 through 16 (see Tab. 3).

The test bundle arrangement for experiments QUENCH-14 (M5) and QUENCH-16 (Duplex) is identical to the standard one but different for QUENCH-15 (ZIRLO) due to a rod diameter of 9.5 mm and a pitch of 12.6 mm. The latter test bundle comprises 24 fuel rod simulators (heated rods), no unheated rod, and eight corner rods.
### Table 3: Cladding material and diameter of the fuel rod simulators and pitch for the QUENCH-ACM test series.

<table>
<thead>
<tr>
<th>Cladding</th>
<th>Vendor</th>
<th>Reactor type</th>
<th>Dimensions, mm</th>
<th>Pitch, mm</th>
</tr>
</thead>
<tbody>
<tr>
<td>M5</td>
<td>Areva</td>
<td>PWR</td>
<td>Ø 9.3 / 10.75</td>
<td>14.3</td>
</tr>
<tr>
<td>ZIRLO</td>
<td>Westinghouse</td>
<td>PWR</td>
<td>Ø 8.347 / 9.5</td>
<td>12.6</td>
</tr>
<tr>
<td>Duplex Zry-4/D4</td>
<td>Areva</td>
<td>PWR</td>
<td>Ø 9.3 / 10.75</td>
<td>14.3</td>
</tr>
<tr>
<td>E110</td>
<td>Russia</td>
<td>VVER</td>
<td>Ø 7.73 / 9.13</td>
<td>12.75</td>
</tr>
</tbody>
</table>

As in the Zry-4 experiments, fuel is represented by ZrO₂ pellets. The test section instrumentation will be as usual, i.e. thermocouples will be attached to the cladding, shroud, and cooling jacket at elevations between 50 mm and 1350 mm. The QUENCH-ACM test series is scheduled to be performed in the period of 2008-2010. Co-operations with respect to pre-test predictions and post-test calculations with severe accident codes, provision of material properties, and model development are welcome.

### 5 SUMMARY, CONCLUSIONS AND OUTLOOK

The isothermal and transient oxidation behaviour of the three advanced cladding materials DUPEX, E110 and M5 in steam, oxygen and air were compared with the behaviour of the classical Zry-4. For all materials a parabolic time dependence of mass increase and oxide layer growth was found at least in the early phase. The oxidation rates depend on temperatures by Arrhenius functions. The activation energies differ for the different zirconium oxide crystal structures.

In the temperature range between 800 and 1000°C the breakaway effect was found for oxidation of E110 in steam and oxygen. At 1000°C the oxidation of Zry-4 shows this effect, too. The effect results in an enhanced oxidation. The time dependence changes from parabolic to nearly linear. For the D4 layer of the DUPEX material and for M5 no breakaway takes place. The morphology of the spalled oxide layer parts differs significantly between E110 and Zry-4. They are much finer for E110 than for Zry-4. The intensive breakaway effect was also found for transient oxidation of E110 with heating rates below 0.1 K/s. For these low heating rates the time in which the materials are in the temperature range where the breakaway effect takes place is long enough for enhancing of oxidation. The strong influence of the breakaway effect on the severe accident behaviour of fuel rod bundles can be seen by comparison of the large scale bundle simulation tests QUENCH-6 and QUENCH-12. The hydrogen release of the QUENCH-12 bundle during reflooding, where massive spallation of oxidised parts takes place, was six times higher than of the QUENCH-6 bundle under comparable conditions.

Oxidation in air shows faster kinetics than in pure oxygen or steam. The reaction depends nearly linear on time. This is caused by the formation of a very porous oxide scale mixed with zirconium nitride which is formed at positions where (local) oxygen starvation occurs and sub-stoichiometric oxides are formed, i.e. at the oxide-metal phase boundary.
ACKNOWLEDGMENTS

The Zr1Nb (E110) and Zr2.5Nb (E125) material used in the VVER-type experiment QUENCH-12 for fuel rod simulators, grid spacers, and shroud was provided by Russian institutions in the context of the ISTC 1648.2 program. M5 and Duplex D4 rod claddings were delivered by AREVA. The ZIRLO cladding and pertinent grid spacer material was supplied by Westinghouse Electric Sweden AB.

The authors thank Ms. U. Stegmaier and Ms. P. Severloh for the metallographic investigations, Dr. R. Simon for the support in the XRA measurements of the tin distribution in the Duplex material. The author is also grateful to Dr. G. Schanz for the fruitful discussions.

REFERENCES


ABSTRACT

Since 2004, and in response to a specific request of the Spanish Regulatory Authority (CSN), ENUSA is undertaking the Juzbado Plant’s Integrated Safety Analysis (ISA) project. An ISA is a systematic examination of plant’s processes, equipment, structures and personnel activities to ensure that all relevant hazards that could result in unacceptable consequences have been adequately evaluated and the appropriate protective measures have been identified. The relevant hazards considered are all those which can lead to radiological consequences, basically, nuclear criticality, contamination & irradiation, fire & explosion, chemical or environmental. The applied methodology foresees to split the plant into basic pieces of analysis (nodes) and carefully peer at each of them identifying hazards. This allows the identification of all credible accident scenarios which are then analyzed in order to assess the risk (high, intermediate or low) of their consequences. The set of appropriate safeguards is then identified, and from it, the so called Items Relied On For Safety (IROFS) are selected depending on the rank of risk of the accident scenario. These IROFS are those safeguards which ensure safety even in the case that the rest of safeguards have been lost. The project implies a number of different activities which must be performed by the plant staff at all the levels and with different skills and qualifications, which means to implement a specific organizational structure to manage the project and training of all the personnel involved. This paper shows the key ideas related to ISA implementation and the approach chosen to deal with it, along with the conclusions reached so far. ENUSA considers that this project will qualitatively increase the degree of knowledge of the potential initiating events, which will lead to an optimization of the safety management of the plant.

1 INTRODUCTION

In 2004 and following a request of the Spanish Regulatory Authority (CSN), ENUSA started at the Juzbado LEU Fuel Fabrication Plant the development and implementation of a risk-informed methodology for accident analysis, which was called the Juzbado Plant Integrated Safety Analysis (ISA) project. By that time, ENUSA was involved in renewal of the Juzbado Plant Operating License and ISA was one of the items open to discussion with CSN. Several profitable meetings were held to clarify positions in issues such as regulatory basis, scope of the analysis, and time-schedule for the project before CSN issued in May 2004 the Official Request establishing the regulatory basis for the ISA and convening ENUSA to
submit a planning for the project. The decision was taken not to issue any specific Spanish regulation on ISA but instead refer to the US 10 CFR Part 70.

As established in the CSN’s Official Request, an ISA is a systematic examination of a plant’s processes, equipment, structures and personnel activities to ensure that all relevant hazards that could result in unacceptable consequences have been adequately evaluated and the appropriate protective measures have been identified. The relevant hazards considered are all those which can lead to radiological consequences, basically, nuclear criticality, contamination & irradiation, fire & explosion, chemical or environmental, along with natural phenomena (floods, quakes, hurricanes, etc.) and any other external event which could be safety related.

It took ENUSA almost one year to prepare the infrastructure needed to go through the project. Upon receipt of the Official Request, a functional organization was built-up. The first task of this organization was to identify a methodology which assured compliance with CSN’s requirements while fitting as much as possible with the Juzbado Plant specific idiosyncrasy. In this context, and taking into account the fact that ISA was already under implementation in USA following NRC requirements, the decision was taken to contact GNF-Wilmington in order to ascertain whether its ISA methodology would be applicable to the Juzbado Plant. Indeed it was, so a team was sent to Wilmington in order to imbed the methodology, adapt it the Juzbado Plant and produce the action plan to comply with the CSN’s request. The outcome of this process resulted in a number of different activities which must be performed by the plant staff at all the levels and with different skills and qualifications, which implied enhancement of the initial functional organization involving personnel from different areas in the plant. Thus a multi-disciplinary organizational structure was designed to manage the project and train all the personnel involved.

The following paragraphs show the key ideas related to ISA implementation and the approach chosen to deal with it, along with the conclusions reached so far.

2 ORGANIZATION & TRAINING

2.1 Organizational Setup

The functional organization which runs the project is shown in Figure 1. It consists of three levels: ISA teams, ISA experts and Project Manager (PM).

The primary level and real heart of the project are the ISA teams. These are constructed so that every single node (i.e. smallest piece of analysis) is analysed by a specific ISA team. Accordingly, team members are selected in such a way that they are able to bring to the group all the operational and technical experience necessary to conduct the job and in this sense, the participation of personnel directly related with shop-floor operations is absolutely essential. Thus, the teams are comprised, at least, by 2 ISA experts (personnel trained on the ISA methodology), 1 expert from each of the safety disciplines (that is, Nuclear Criticality, Health Physics & Industrial Safety), 1 plant engineer and 1 shop-floor operator. Of course, these teams can be complemented with members experienced on the node subject to analysis, including maintenance persons.

ISA experts are intended to coordinate the team meetings, lead the discussions of the group and document the whole process. They are specifically trained on the ISA methodology and in particular, on the hazard analysis techniques being applied. ISA experts are qualified according to internal procedures. The decision was taken to include 2 ISA experts in each team, so that one of them could lead the meetings, focusing mainly on the discussions, whilst the other could take care of the administrative tasks of the process.
The project is coordinated by a PM to whom the ISA experts in charge of the different teams report.

![ISA Organizational Setup](image)

**Figure 1: ISA Organizational Setup**

### 2.2 Training activities

The training program for the personnel involved in the ISA Project has been designed on a “cascade” basis, so that it started involving a small group of persons who would deliver the knowledge downstream when required.

Thus, by the end of 2004 the selected personnel was subjected to a comprehensive training course devoted to hazard analysis methodologies at the Juzbado Plant’s premises, which was addressed, basically, to engineers and safety people intended to later become ISA experts. This group constituted the seed of the actual functional organization.

The training at the PM level was complemented with a 1-week specific visit to GNF-Wilmington held by the end of January 2005. This visit allowed developing the ENUSA’s ISA approach to be later applied. Once this task was completed, the final training of the ISA experts was conducted by early May 2005 with a 3-day course held at the Juzbado premises, to which CSN staff members were invited to attend. After this six-month period, the initial organization was ready to begin the job.

Training at ISA team level (if needed) is foreseen after selection of the members and in any case, prior to the first meeting of each group. In this training, team members are subjected to an ISA methodology induction along with the specificities of the hazard analysis technique chosen for the node under consideration.

By mid-May 2005 and upon completion of the initial training program, the ENUSA’s ISA procedure and planning was submitted to the CSN fulfilling the requirement of the Regulator’s Official Request.

### 3 RISK ASSESSMENT

Figure 2 shows the front end of the chosen approach for the Juzbado Plant’s Integrated Safety Analysis. The first step was to split the Plant into areas of analysis following process or functional criteria and then, split the areas into nodes, which are the smallest piece to look...
upon and thus, the basic unit for the study. 20 areas and 117 nodes were identified in a first approximation, though it is likely that these numbers change as the project evolves.

The ISA team for the specific node is then selected and trained (when necessary). The work begins with the identification of hazards, which is done by means of methods as What if or HAZOP (for more complex nodes). This allows the identification of all credible accident scenarios and for each of which, evaluating all possible causes and consequences. At this stage, it is important to highlight that the idea is to identify all the consequences, no matter whether they are radiological or not. Once every individual consequence is identified, the Severity and Unmitigated Likelihood (UML) of the sequence are assessed and combined to derive the Unmitigated Risk (UMR) associated with the considered accident scenario.

3.1 Severity

The severity matrix adopted by ENUSA is shown in Table 1. This matrix is based upon the criteria established in Appendix A of NUREG-1520 [1]. The first two columns of Table 1 refer to Nuclear Criticality Safety (NCS) related events which are not included in reference [1] but that have been considered for completeness. Therefore, NCS events are categorised in terms of the remaining number of Independent Control Parameters (ICP) which remain in place after the initiator so that the rank will depend on whether there are Multiple Parameters being controlled (MCP) or just a Single Parameter with several controls (OCP).

Regarding purely radiological-related events, the figures included in Table 1 slightly differ from those included in reference [1]. In this case, the rational behind the figures was to conservatively accommodate their values to the current Spanish regulations on Radiological Protection. Furthermore, the severity of events related to effluent discharges has been related to the Juzbado Plant’s operational license and notification limits.

Remarkably enough there is no reference to chemical doses in Table 1, as this is very low risk at the Juzbado Plant and any case with purely industrial safety related consequences, but not radiological. As all the consequences are recorded in the ISA, the decision was taken to rank as level 1 all the events that could result in conventional (non-radiological) damage to the workers, although they fall out of the severity matrix.
Table 1: Severity Matrix

<table>
<thead>
<tr>
<th>Rank</th>
<th>MCP</th>
<th>OCP</th>
<th>Workers</th>
<th>Public</th>
</tr>
</thead>
<tbody>
<tr>
<td>3</td>
<td>Lost of all ICP</td>
<td>Lost of all controls</td>
<td>D &gt; 1Sv</td>
<td>D &gt; 250mSv</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Ingestion of more than 30mg of U</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Effluent release above License limits</td>
</tr>
<tr>
<td>2</td>
<td>Lost of one or several ICP such as only ONE ICP remains intact</td>
<td>Lost of one or several controls such as only ONE control remains intact</td>
<td>50mSv ≤ D ≤ 1Sv</td>
<td>5mSv ≤ D ≤ 250mSv</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Effluent release above Notification limits</td>
</tr>
<tr>
<td>1</td>
<td>Lost of any ICP such as the Double Contingency Principle remains intact</td>
<td>Lost of any control such as the Double Contingency Principle remains intact</td>
<td>D &lt; 50mSv</td>
<td>D &lt; 5mSv</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Effluent release below Notification limits</td>
</tr>
</tbody>
</table>

3.2 Unmitigated Likelihood

Table 2 shows the Unmitigated Likelihood table applied by ENUSA. It is important to note that this is an unmitigated value and thus, stands for the likelihood of the accident sequence without taking into account the specific safeguards already implemented (or to implement). A 50 year Plant lifetime was assumed to distinguish between credible and incredible sequences. On the other hand, a term of 2 years was considered as a reasonable value to rank the UML of credible sequences.

Table 2: Unmitigated Likelihood

<table>
<thead>
<tr>
<th>Rank</th>
<th>Frequency</th>
<th>Likelihood</th>
</tr>
</thead>
<tbody>
<tr>
<td>3</td>
<td>More frequent than once every 2 years</td>
<td>Likely to occur in the immediate future</td>
</tr>
<tr>
<td>2</td>
<td>Every 2 to 50 years</td>
<td>Likely to occur during the plant lifetime</td>
</tr>
<tr>
<td>1</td>
<td>Less than once every 50 years</td>
<td>Unlikely to occur during the plant lifetime</td>
</tr>
<tr>
<td>0</td>
<td>Incredible</td>
<td>Non-distinguishable from zero</td>
</tr>
</tbody>
</table>

3.3 Unmitigated Risk

The combination of the previously assessed values for Severity and UML leads to the Unmitigated Risk, as shown in Table 3. This matrix is more conservative that the one included in reference [1], which ranks as acceptable all risks below index 6. We consider as non-acceptable the intermediate risk scenarios, with risk index values of 3 and 4.
As we soon see, this risk assignment is used by the ISA team to assess the appropriate level of assurance for the safeguards.

4 SAFEGUARDS, IROFS AND OVERALL LIKELIHOOD

The back end of the process is shown in Figure 3. All the safeguards already implemented (or to implement) to either prevent or mitigate the effects of every single accident scenario are identified by the team members, including those related to low risk events.

Then, from this set of safeguards, the Items Relied On For Safety are selected for those intermediate and high risk accident scenarios. The IROFS are those safeguards which ensure safety even in the case that the rest of safeguards have been lost. At least two independent IROFS are identified for every intermediate and high level risk accident scenario. IROFS can

Table 3: Unmitigated Risk

<table>
<thead>
<tr>
<th>S</th>
<th>0</th>
<th>1</th>
<th>2</th>
<th>3</th>
</tr>
</thead>
<tbody>
<tr>
<td>3</td>
<td>Null</td>
<td>Intermediate</td>
<td>High</td>
<td>High</td>
</tr>
<tr>
<td>2</td>
<td>Null</td>
<td>Low</td>
<td>Intermediate</td>
<td>High</td>
</tr>
<tr>
<td>1</td>
<td>Null</td>
<td>Low</td>
<td>Low</td>
<td>Low</td>
</tr>
</tbody>
</table>
be either passive or active engineered controls, or enhanced administrative controls, but not purely administrative controls.

4.1 Overall Likelihood

Once the set of IROFS applicable to a specific scenario has been identified, the Overall Likelihood ($L_T$) index is computed. This takes into account the individual IROFS failure’s frequency, the time period (duration) of IROFS failed condition prior to detection, and the existence of (other) independent IROFS.

Let us assume that $N$ IROFS have been identified for mitigating a given sequence. The $L_T$ index is then calculated by means of Eq. (1) below, derived following the criteria established in reference [2]:

$$L_T = \sum_{i=1}^{N-1} \left( \lambda_{f,i} + \lambda_{d,i} \right) \lambda_{mcf,i} + \lambda_{f,N} \quad (1)$$

Where, $\lambda_{f,i}$ is the frequency index of failure per year of IROFS $i$th; $\lambda_{d,i}$ is the failure duration index of IROFS $i$th; and $\lambda_{mcf,i}$ is the common mode failure factor of IROFS $i$th. Both $\lambda_{f,i}$ and $\lambda_{d,i}$ are calculated by taking Log$_{10}$ to the appropriate values of frequency and duration of failure as tabulated in the existing industry data bases, whereas $\lambda_{mcf,i}$ can either equal to 1 if IROFS $i$th is completely independent from all other IROFS being considered, or to 0 providing IROFS $i$th is subject to common mode failure with any other IROFS. Finally, $\lambda_{f,N}$ represents the frequency index of failure for the last IROFS in the string. Conservatively, it is assumed that this last IROFS is always that with the smallest duration index.

The so computed $L_T$ index is then compared with the acceptance criteria shown in Table 4. $L_T$ index must always be smaller than the corresponding clearance criteria. Should $L_T$ index fall out of the limits, the IROFS selection is considered no longer valid and then, a new set of IROFS must be selected and newly subjected to the acceptance process.

<table>
<thead>
<tr>
<th>$L_T$</th>
<th>Likelihood per year</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>-6</td>
<td>1.00E-06</td>
<td></td>
</tr>
<tr>
<td>-5</td>
<td>1.00E-05</td>
<td></td>
</tr>
<tr>
<td>-4</td>
<td>1.00E-04</td>
<td>Acceptable limit for High Consequence sequences</td>
</tr>
<tr>
<td>-3</td>
<td>1.00E-03</td>
<td>Acceptable limit for Intermediate Consequence sequences</td>
</tr>
<tr>
<td>-2</td>
<td>1.00E-02</td>
<td></td>
</tr>
<tr>
<td>-1</td>
<td>1.00E-01</td>
<td></td>
</tr>
</tbody>
</table>

Once IROFS is validated according to the above described acceptance criteria, a specific management program is implemented according to established procedures. The scope of the program depends on the type of IROFS (passive or active, engineered or administrative,
etc.) and their importance (level of UMR they protect against to). This program is intended to ensure the reliability and availability of IROFS to perform its function at any time.

5 ISA SUMMARIES

All the above described process is recorded and documented using software specifically developed for process hazard analysis applications (Hazard Review LEADER™). The software allows maintaining a data base with the information generated by the ISA team meetings and drafting reports to document the process.

These reports are then reviewed by the ISA experts who leaded the meetings and subjected to management approval. Once all the nodes belonging to an area have been analyzed and documented, an ISA summary for the corresponding area is to be issued and submitted to CSN for approval.

6 FINAL REMARKS

The initial planning submitted to CSN on 2005 foresaw a minimum of 5 years for the whole project to be finalised. After almost 3 years of work, 51 nodes have been analysed with about 1000 accident sequences being identified, out of which 33 where ranked as intermediate risk, none as high risk.

The decision was taken to implement a design modification in the equipment to get rid of one of those sequences and thus, to avoid the implementation of IROFS. Therefore, 32 sets of IROFS have been identified up to the time this paper was written. In most cases, IROFS were selected from the safeguards currently available, but some of them have led to design modifications in the Plant to comply with the ISA requirements for IROFS.

Although the project is still ongoing, ENUSA considers that the implementation of the ISA program is increasing qualitatively the degree of knowledge of the potential initiating events and then, leading to an optimization of the safety management of the plant.

ACKNOWLEDGMENTS

The authors gratefully acknowledge the work done by all the ENUSA staff involved in the ISA project for the last 3 years. Also, we would like to thank the help of the GNF-Wilmington’s ISA team during these years and in particular, the work done by James W. Reeves.

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Safety Improvements on Fuel Rod Behavior Because of Axial Fission Gas Transport Modeling

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ABSTRACT

Fission gas release (FGR) to the gap during irradiation is not uniform along the fuel rod length. The axial profiles of temperature and neutron flux cause higher releases at high temperature and high power regions. Such lack of uniformity could be even enhanced in transients where FGR at specific regions are fostered. The importance of these scenarios lies in the analytical approach adopted by most present fuel performance codes, where axial gradients are neglected and transport processes are assumed to occur just along the radial direction.

This paper investigates the major features of axial transport of fission gases and critically assesses the potential impact on fuel rod behaviour. Experimental evidences are gathered and bases of existing models are reviewed. Estimates of characteristic transport times in postulated scenarios are presented.

According to this study, consideration of the axial fission gas transport in fuel performance codes might be necessary at the beginning of irradiation, as the gap is still open and the ingap gas content has not been extensively contaminated by Xenon (Xe) and Krypton (Kr). Both gases degrade the thermal properties of the gas mixture with respect to Helium (He) ones and, as a consequence of the thermal feedback of FGR, release is even further enhanced at those regions with higher Xe and Kr content. Hence, the present assumption of instantaneous gas mixture in the fuel rod gap might not be conservative.

The models proposed in the literature can be grouped in two categories: diffusion models and diffusion-convection models. The latter offers the most comprehensive description; however, some of them may be regarded as too complex to be implemented in present fuel performance codes.

1 INTRODUCTION

The fission gases released from the fuel get mixed with the gases in the gap. This mixing transient takes time depending on specific conditions of the fuel rod and FGR. Anyhow, mixing time is bounded between two asymptotic situations: instantaneous mixing and fully segregated emission (i.e., released gases push away the pre-existing gas mixture at the given location). The former involves a much lighter modification of the gap thermal conductivity than the latter (Xe and Kr thermal conductivities are roughly 20 times lower than He one), because of which the feedback on FGR can be significant.

Experimental evidences show that as the gases are released from the pellet, there is an initially local accumulation. Then gases start dilution in the existing gas mixture with a finite kinetics driven by the local concentration and pressure spikes, so that gases are transported
due to concentration and pressure gradients until reaching an equilibrium situation in the rod. As a consequence, the temperature evolution at the location where FGR occurs can be described as in Figure 1.

![Figure 1: Comparison of different gas transport assumptions on fuel thermal behaviour.](image)

Despite experimental evidences of the outlined dilution kinetics, most of the fuel rod performance codes do not consider any axial transport phenomenon, so that this mixing transient is not accounted for. Instead, they assume a non-conservative instantaneous mixing. Namely, fuel temperature at the release location is underestimated with respect to the real situation and, given the Temperature-FGR feedback, FGR as well.

Overall, the transport scenario is drastically affected by fuel and, more specifically, by the closure status of gap. As long as the gap is open, there will be a perfect axial communication in the gap all along the fuel stack. Under these conditions, the gas transport mechanisms inside the fuel rod do have a diffusive nature (i.e., axial concentration gradient) and a convective one (i.e., axial pressure gradient). However, as the gap becomes closed the transport is severely hindered. In addition, the higher the burnup the higher contamination of the rod filling gas by Xe and Kr and, as a result, the effect of any additional FGR becomes less significant progressively. Moreover, gap thermal resistance relevance in the global heat transfer turns out less significant as its thermal resistance decreases due to gap shrinkage.

2 STATE OF THE ART

The theoretical approaches of axial gas transport can be grouped in: diffusive models [1, 2] and diffusive-convective models [3, 4 and 5]. Coupling of diffusion and convection processes makes the latter group rather more complex. In order to overcome such complexity, Nakajima [5] assumes that convection is much faster than diffusion, so that he imposes an instantaneous convective transport, diffusion starting once pressure has been equalized along the fuel rod. This simplicity turns Nakajima’s model into a good alternative to be implemented into fuel performance codes. All the models neglect radial transport of the gas.

Some of the models consider the main fission gas release (Xe and Kr), while others just take conservatively into account Xe as the fission gas species (its fission yield being higher and lower diffusion velocity than Kr ones). Moreover, the models just consider diffusion between binary species, He-Xe or He-Kr, i.e., between one of the fission gas and the initial filling gas, but never between fission gases or in ternary diffusive processes.
2.1 Gas Transport Model [5]

The transport model proposed by Nakajima [5] has been reviewed and their equations developed and implemented in a FORTRAN stand-alone code. The model considers two gas species (He and Xe) and splits the fuel rod in several axial nodes (up to twelve), which may have different void volumes and temperatures (within each node gases distribute uniformly).

2.1.1 Convective transport

The convective flow due to pressure gradient causes a macroscopic gas flow toward rod plenum. Such flow is considered much faster than FGR and interdiffusion phenomena, so that instantaneous gas flow and pressure equalization are assumed.

The total amount of gas moles in each axial node:

\[ n_j = \sum_x n_{x,j} \quad (x=\text{He, Xe}) \]  \hspace{1cm} (1)

And the total gas moles in the rod:

\[ n = \sum_j n_j \]  \hspace{1cm} (2)

So, the rod internal pressure after a FGR spike can be estimated by means of the ideal gas law:

\[ P = nR \frac{1}{\sum_j (V/T)_j} \]  \hspace{1cm} (3)

Hence, the amount of gases in each axial node after pressure equalization is given by:

\[ n_j' = \frac{P}{R} \left( \frac{V}{T} \right)_j \]  \hspace{1cm} (4)

The difference \( n_j \) and \( n_j' \), \( \Delta n_j \), corresponds to the number of gas moles before and after pressure equalization. The pressure gradient induced by the xenon release will promote the convective transport to the upper and lower axial nodes. At the location of the release, \( j_{\text{FGR}} \), before any convective transport, the amount of gas in such node corresponds to adding the gas moles released (\( n_{\text{FGR},j_{\text{FGR}}} \)) to the previous gas content (\( n_{\text{Xe},j_{\text{FGR}}} + n_{\text{He},j_{\text{FGR}}} \)):

\[ n_{j_{\text{FGR}}} = n_{\text{Xe},j_{\text{FGR}}} + n_{\text{He},j_{\text{FGR}}} + n_{\text{FGR},j_{\text{FGR}}} \]  \hspace{1cm} (5)

Then, the gas fractions at the release node are updated as:
Estimate of the convective mole transport from the FGR node to upward and downward adjacent nodes proceeds in a stepwise fashion. It means that from release node, the gas exceeding the \( n_{jGR} \) will be pushed up and down to the adjacent nodes according to:

\[
\begin{align*}
    n_{jGR} \rightarrow j+1 &= \sum_{i=j+1}^{\text{plenum}} \Delta n_i \\
    n_{jGR} \rightarrow j-1 &= \sum_{i=j-1}^{1} \Delta n_i
\end{align*}
\]

Once the gas lumps move to the upper and lower nodes, they are mixed with the gases inside the node. Therefore, these receiving nodes turn into the new source nodes. The gas fraction will change accordingly to the amount and fraction of gases transferred and the previous gases in the axial node:

\[
\begin{align*}
    f_{He,jGR} &= \frac{n_{He,jGR}}{n_{He,jGR} + n_{Xe,jGR}} = \frac{f_{He,jGR}^b \cdot n_{jGR}^b}{n_{He,jGR} + n_{Xe,jGR}} \\
    f_{Xe,jGR} &= \frac{n_{Xe,jGR}}{n_{He,jGR} + n_{Xe,jGR}} = \frac{f_{Xe,jGR}^b \cdot n_{jGR}^b}{n_{He,jGR} + n_{Xe,jGR}}
\end{align*}
\]

2.1.2 Diffusive transport

Once the gas moles are distributed by instantaneous convective flux the diffusive transport starts governed by the gas concentration gradient between adjacent nodes. The diffusion equation is formulated by means of the Fick’s first law. The molar gas flux, \( G_x \), of the \( x \) gas species from \( j \) to \( j+1 \) is given by:

\[
G_x(j \rightarrow j+1) = \frac{f_{He,j} \cdot n_{jGR}^{He} + f_{He,jGR} \cdot n_{jGR}^{j+1}}{n_{He,j} + n_{He,jGR}}
\]

\[
\begin{align*}
    f_{He,j} &= \frac{n_{He,j}}{n_{He,j} + n_{Xe,j}} = \frac{f_{He,j}^b \cdot n_{j}^{He} + f_{He,jGR} \cdot n_{jGR}^{j+1}}{n_{He,j} + n_{He,jGR}} \\
    f_{Xe,j} &= \frac{n_{Xe,j}}{n_{He,j} + n_{Xe,j}} = \frac{f_{Xe,j}^b \cdot n_{j}^{Xe} + f_{Xe,jGR} \cdot n_{jGR}^{j+1}}{n_{He,j} + n_{He,jGR}}
\end{align*}
\]

The transport scheme, described above between FGR node and \( j+1 \) and \( j-1 \), is repeated in \( j+1 \) and \( j-1 \) in the upward and downward direction, respectively, up to achieve the plenum or the first (bottom) axial node.

\[
G_x(j \rightarrow j+1) = F_{j \rightarrow j+1} \left[ (C_{s,j} - C_{s,j+1}) - (\bar{C}_{s,j} - \bar{C}_{s,j+1}) \right]
\]
\[ F_{j,j+1} = \frac{A_j D_{\text{He-Xe},j} A_{j+1} D_{\text{He-Xe},j+1}}{A_{j+1} D_{\text{He-Xe},j+1} \Delta z_j + A_j D_{\text{He-Xe},j} \Delta z_{j+1}} \] (13)

\[ C_{x,j} = \text{concentration of gas species } x \text{ in axial node } j \]

\[ \overline{C}_{x,j} = \frac{P \cdot (f_{i,j} + f_{i,j+1})}{RT_j} \] (14)

\[ \overline{C}_{x,j+1} = \frac{P \cdot (f_{i,j} + f_{i,j+1})}{RT_{j+1}} \] (15)

\[ \Delta z_j = \text{axial length of segment} \]

\[ D_{\text{He-Xe},j} = \text{diffusion constant (m}^2/\text{s) of He-Xe mixture in axial node } j \]

\[ D_{\text{He-Xe},j} = \frac{3}{8} \left( \frac{\pi RT_j}{2M^*} \right)^{1/2} \frac{RT_j}{P \pi (d_{\text{He-Xe}})^2} \] (16)

where

\[ M^* = \frac{m_{\text{He}} \cdot m_{\text{Xe}}}{m_{\text{He}} + m_{\text{Xe}}} \] (17)

\[ m_i = \text{molecular weight of species } i \]

\[ d_{\text{He-Xe}} = \frac{1}{2} (d_{\text{He}} + d_{\text{Xe}}) = \text{atomic diameter average of gas species} \]

The diffusion constant is fitted with a \( d_{\text{He-Xe}} \) value of \( 3.45 \cdot 10^{-10} \) m to fit with the value of \( D_{\text{He-Xe}} \) experimentally observed of \( 3.58 \cdot 10^{-5} \) m\(^2\)/s at 0ºC and 1 atm [5].

3 SENSITIVITY ANALYSIS

In order to explore the gas transport sensitivity to parameters such as fuel length, initial rod pressure and FGR location, a set of calculations have been carried out. The scenario simulated is a fuel rod with characteristic pellet and clad dimensions of a commercial rod, a linear power rate of 10 kW/m and cooled by water at 240 ºC and 3.4 bar. The release is supposed to occur just in one node (out of a total of 9) at the beginning of the analysis.

3.1 Effect of the Rod Length

Short rods used in experimental reactors may behave differently from commercial ones. Two studies have been performed. The former simulates a fuel rod 0.36 m high with a plenum volume of \( 3.0 \cdot 10^{-6} \) m\(^3\) and a cold radial gap of 50 µm. The latter is a similar rod, but the length is 10 times longer, 3.6 m, and the plenum volume is increased up to \( 20 \cdot 10^{-6} \) m\(^3\). In the two cases, a Xe/(Xe+He) mole fraction of 58% was imposed in the middle axial node at the beginning of the analysis in a rod initially pressurized to 1 bar.
Figure 2 shows the normalized (i.e., initial Xe concentration at $t=0$ s is assumed to be 1.0) evolution of the Xe fraction at the release node in both rods. As expected, the mixing transient is faster in the short rod and the equilibrium is attained in a rather shorter period. This is consistent with the asymptotic solution of the Fick’s law for a continuous FGR with zero Xe concentration at an infinite distance [6]. Under these boundary conditions, the time at which fission gas concentration at the plenum reaches 50% of the concentration at the release node, $\tau_{1/2}$, (assumed to be constant along time) can be approximated by:

$$\tau_{1/2} \approx \frac{z^2}{D}$$

Eq. (18) yields 0.72 h for the short rod and around 72 h for the long one.

These findings emphasize that extrapolation of the fast axial mixing observed in experimental fuel rods after a FGR spike is far away to be straight and it should be considered with utmost caution.

![Figure 2: Effect of fuel rod length on fission gas transport](image)

### 3.2 Effect of the Initial Rod Pressure

Two new simulations with an initial pressure of 25 bar have been conducted and compared to those presented above. As noted, an increase of the initial pressure results in longer mixing transients (consequently with the effect of pressure on gas diffusivities) in both rod lengths. Even further, the response time of short rods is notably shorter than for the long ones. An indirect measure of the potential effect of FGR spikes is the final equilibrium fraction, given the substantial weight of the initial He content in the total gas amount in the gap at 25 bar the Xe equilibrium concentration is reduced to a 5%.

In short, the higher the initial rod pressures the longer the mixing transients, due to changed gas diffusivities, and the less the effect of any FGR spike, due to higher initial HE content.
Quantitatively speaking, an initial high pressure rod (i.e., 25 bar) would undergo mixing transients that, if similar to the one simulated, would be 10 times longer than at low pressure.

3.3 Effect of the release location

FGR during irradiation mainly takes place in the peak power zone. Experimentally [6], it has been observed that the longer the distance between releasing node and plenum the slower fission gases dilution. Such behaviour is attributed to the milder gradients set up. This is encapsulated in Eq. (18) where it can be noted that the diffusion time increases proportionally with the square of the diffusion distance.

Two new cases similar to the previous ones have been studied with the only difference being the FGR location, either at the top (node 8) or at the bottom (node 2) of the fuel stack. Consistently with previous discussion concerning diffusion times and gradient intensity (i.e., distance between concentration differences), Figure 4 shows that the higher the release location the shorter the time to reach equilibrium at the rod plenum. As shown in this example (high pressure, long rod), the delay resulting from a downward displacement of the FGR location can be n times the shorter diffusion time estimated, but within the same order of magnitude.
4 PROSPECTIVE INFLUENCE ON FUEL TEMPERATURE

A fission gas release is faster than the subsequent axial gas mixing and, particularly, a FGR spike, results in axial differences of gap thermal conductance. As a consequence, axial fuel temperature gradients would not be just a result of the power profile but also a consequence of the different gap thermal resistance at different rod locations. This scenario is away from most of codes that assume axial gas transport as an instantaneous phenomenon.

A prospective analysis of the potential influence of the gas transport on fuel temperature has been indirectly conducted. A PWR-UO2 fuel rod design has been simulated with FRAPCON-3 fuel rod performance code [7]. The main characteristics of the fuel rod at cold and under operating conditions are presented in Table 1.

Table 1: Main characteristic of simulated fuel rod

<table>
<thead>
<tr>
<th>Item</th>
<th>Cold State</th>
<th>Hot State</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of axial nodes</td>
<td>9(+1 for plenum)</td>
<td></td>
</tr>
<tr>
<td>System pressure, bar</td>
<td>155.1</td>
<td></td>
</tr>
<tr>
<td>Inlet coolant temperature, °C</td>
<td>292.4</td>
<td></td>
</tr>
<tr>
<td>Fuel density, %TD</td>
<td>95.5</td>
<td></td>
</tr>
<tr>
<td>U-235 enrichment, %</td>
<td>4.5</td>
<td></td>
</tr>
<tr>
<td>Linear power, kW/m</td>
<td>10 (flat profile)</td>
<td></td>
</tr>
<tr>
<td>Initial He moles</td>
<td>0.018719</td>
<td></td>
</tr>
<tr>
<td>Active fuel rod, m</td>
<td>3.6576</td>
<td>3.6746</td>
</tr>
<tr>
<td>Cold free fuel rod volume, m³</td>
<td>19.121E-06</td>
<td>16.236E-06</td>
</tr>
<tr>
<td>Cold plenum volume, m³</td>
<td>9.5536E-06</td>
<td>8.8728E-06</td>
</tr>
<tr>
<td>Outer clad diameter, mm</td>
<td>9.5</td>
<td>9.50996</td>
</tr>
<tr>
<td>Inner clad diameter, mm</td>
<td>8.357</td>
<td>8.421</td>
</tr>
<tr>
<td>Cold Fuel-Clad Radial Gap, µm</td>
<td>82.5</td>
<td>68.6627</td>
</tr>
<tr>
<td>Cold Gap volume, m³</td>
<td>9.5675E-06</td>
<td>7.3635E-06</td>
</tr>
<tr>
<td>Fuel pellet diameter, mm</td>
<td>4.096</td>
<td>4.1418</td>
</tr>
<tr>
<td>Initial He rod pressure, bar</td>
<td>23.5</td>
<td>59.86</td>
</tr>
<tr>
<td>Estimated Gap Temperature, K</td>
<td>--</td>
<td>629.8</td>
</tr>
<tr>
<td>Plenum Temperature, K</td>
<td>--</td>
<td>591.6</td>
</tr>
</tbody>
</table>

The effect of the Xe dilution kinetics has been approximated by a set of runs at different initial gap compositions. They have been taken from simulations with the model presented in previous sections, so that each gas concentration is associated to a specific time of a mixing transient. A low fuel power (10 kW/m) has been chosen to avoid any further FGR. The Xe injection was simulated to occur at the beginning of the analysis at the middle axial node. Detailed information concerning Xe injection is given in Table 2.

Table 2: Simulated Xe injection

<table>
<thead>
<tr>
<th>Item</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Initial He moles</td>
<td>0.018719</td>
</tr>
<tr>
<td>Initial He rod pressure, bar</td>
<td>23.5</td>
</tr>
<tr>
<td>Cold Rod Pressure after Xe injection, bar</td>
<td>24.11</td>
</tr>
<tr>
<td>Xe injection at t=0, mol (at node 5)</td>
<td>4.8665E-04</td>
</tr>
<tr>
<td>Asymptotic Xe/(Xe+He)</td>
<td>2.53%</td>
</tr>
</tbody>
</table>
Figure 5 presents the estimated fuel centre temperature according to the gap fission gas composition evolution at the injection position. As can be observed, the assumption of an instantaneous mixing, results in a non-conservative temperature underestimate of about 100 °C (greater than 10%) at the beginning of the transient and stays at those level for nearly 5 h. Given the link between fuel temperature and FGR, it can be foreseen that FGR will be fostered accordingly. This prospective study, however, cannot quantify this process. However, it has been shown that the temperature difference is high enough to recommend the axial mixing model implementation in fuel rod performance codes.

Figure 5: Fuel centre temperature evolution with and without gas transport model at the middle of the fuel stack

The estimated fuel centre temperature taking into account the axial gas transport is about 80-100 °C higher for more than 5 hours. This temperature difference is enough to promote further FGR at the gap and even increase the temperature difference. This further feedback can not be observed in this prospective approach since the axial gas transport model is not yet implemented in the fuel rod code. However, the fuel temperature difference and dilution times found are high enough to consider the implementation of an axial gas transport model in safety analysis.

5 CONCLUSIONS AND FINAL REMARKS

This paper analyzes the potential effects of a prompt FGR at a specific location of a fuel rod during reactor operation. Some of the main outcomes of this work can be summarized as follows:

- The time window of interest is bounded between a certain fuel irradiation that allows for a significant inter-granular fission gas accumulation and gap closure.
- The axial gas transport in pressurized commercial fuel rods happens to take longer than in non-pressurized experimental rods: Nonetheless, its effect is anticipated to be milder since under the same conditions the fraction of poisoning gases (i.e., Xe and Kr) is forcefully lower so that gap mixture properties would be closer to those of pure helium.
- The potential significance of not considering the axial gas transport during this effect can affect substantially the fuel rod behaviour during transients. By an indirect assessment based on using FRAPCON-3 under specific gap compositions, it has been
estimated that underestimations even higher than 10% could take place when the axial gas motion during transients is neglected. Such estimates are seen as approximate and, given the temperature-FGR feedback, significance is expected to be even more substantial.

- Modelling of axial convection and diffusion is complex and its implementation in current fuel performance codes are far from being straightforward. Nevertheless, by assuming that convection is an instantaneous process a remarkable simplification is gained. Presently, a stand-alone code has been built-up based on the concept proposed by Nakajima regarding instantaneous convection.

Further work foreseen is the implementation of the model into a fuel rod performance code such as FRAPCON-3.

ACKNOWLEDGMENTS

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REFERENCES


AREVA-CERCA 10 years licence for fuel fabrication

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ABSTRACT

Every ten years, each French Nuclear Installation (referred here after as INB for “Installation Nucléaire de Base”) shall be subject to a safety evaluation review in order to obtain the operating licence for the next ten years period. The licence is delivered during a so called “Factory Permanent Group” review whose participants are a group of experts from the French Safety Authority (ASN), the French Institute for Radiation protection and Nuclear Safety (IRSN) and the User of the plant. The safety evaluation is conducted by both the User and the IRSN during at least a one year period before the Permanent Group review. During this period, the User shall demonstrate the conformity with regards to applicable standards of all the safety issues related to the factory operation such as criticality, radioprotection, seismicity, fire, external risks, etc…

After more than one year of study, CERCA factory in Romans (France) referred as INB # 63 has succeeded its safety evaluation review in late 2006 and is now licensed to operate safely till end of 2016.

The aim of this talk is to present the content of this project that has been conducted since end of 2005 and whose purpose is to ensure the sustainability of CERCA fuel fabrication factory in Romans (France), at least for the next ten years period.

1 PURPOSE

Issue

Every ten years, each French Nuclear Installation shall be subject to a safety evaluation review in order to obtain the operating licence for the next ten years period.

As known, AREVA / CERCA is yearly manufacturing many types of Fuel Elements for Research Test Reactors & Material Test Reactors as well as thousands of molybdenum targets for the nuclear medical market. The factory is located in Romans (France) and is referred as INB 63 (Installation Nucléaire de Base # 63). The site is shared with FBFC as LWR plants type fuel factory through INB 98.

To operate, the INB 63 is subject to the authorization of the French Nuclear Safety Authority (ASN).
“ASN is tasked, on behalf of the State, with regulating nuclear safety and radiation protection in order to protect workers, patients, the public and the environment from the risks involved in nuclear activities. It also contributes to informing the citizens.”

By end of 2006 and after a long preparatory period, CERCA was licensed by the ASN for ten years.

The purpose of this paper is to present the stakes of such an authorization and to highlight the main issues to address during the project.

- **Be authorized**

  The authorization to run is subject to the prescriptions of the “Arrêté du 10 août 1984” (August 10th 1984 decree) related to the quality for the design, the construction and the operation of nuclear installations.

  It is the responsibility of the operator to conform to the regulations. In front of the population, the ASN must guarantee the conformance of the Nuclear Installation (INB) operation to the decree.

  CERCA no more authorized to run would deprive many research reactors of fuel and would significantly disrupt the production of molybdenum for medical exams. Therefore, be authorized is the challenge.

- **Show the ability to operate safely**

  So, it is CERCAs everyday responsibility to maintain a high level of safety and security in its facilities. For this, a complete Safety, Security & Environment (SSE) system is deployed in order to ensure that all the practices conform to the safety regulations requirements.

- **Be safe**

  The Nuclear Safety covers all the actions taken to prevent a nuclear accident or to limit its consequences.

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**Picture and map of the CERCA / FBFC site**
Establishing and developing a strong safety organization is our priority for whole of our activities such as design, fabrication, storage & shipment of nuclear products.

Particularly, this organization must take into account all the equipment changes.

2 THE MAIN STEPS OF THE AUTHORIZATION PROCESS

- General project organization and planning
  - French State side

The Nuclear Safety Authority is in charge of validating the authorization to run. This authorization may be delivered on the basis of a technical analysis which is conducted by the Institute for Radiation protection and Nuclear Safety.

“The IRSN is the expert in research and specialised assessments into nuclear and radiological risk serving public authorities”. The IRSN is appointed by the Safety Authority.

During the safety evaluation period, the IRSN has constituted a project organization with a project manager and a team of experts on each discipline.

- AREVA / CERCA Side

CERCA has also constituted a project type organization in order to prepare whole of the documentation and to answer to the questions of the IRSN experts.

The team is lead by the Safety, Security and Environment Management department, and is also composed of personnel from the operation department of CERCA and from personnel from different engineering departments of AREVA.

Both teams always wanted to work closely in order to avoid any kind of misunderstanding. This spirit was a key factor of success.

The overall schedule of the project was as follow:

Overall schedule of the project
International Topical Meeting on Safety of Nuclear Installations, Dubrovnik, Croatia, 30.9. – 3.10. 2008

- **Internal preparation period (Internal studies - FSAR revision)**

  The first step is to conduct internally a global safety analysis of the current situation in order to update the Final Safety Analysis Report (FSAR) and the Operating Guidelines. These documents must be an accurate picture of the factory at the beginning of the project in order to allow both parties to make their own diagnostic.

  Doing the studies and updating the FSAR took about 1 ½ year. Obviously, the ideal would be to demonstrate safe people with safe processes on safe machines in a safe building. But the regulation always changes in a safer way and is more and more demanding. So, even if our level of safety is continuously upgraded, it remains still a little gap between what is required and what is in place.

  The CERCA FSAR is divided in 3 volumes
  - 1st volume : General description of the site and associated facilities
  - 2nd volume : Detailed description and safety analysis of each workshop and facility
  - 3rd volume : Global safety analysis

  This structure allows anyone to easily access to the safety issues, either on the factory or at any work post.

  The detailed evaluation review of each workshop and each process has permitted to show the strong points and the weak points of our way to operate. So it was easy to draw up an improvement program that could be submitted to the IRSN and implemented gradually.

  Previously to the formal project start meeting, the revised FSAR as well as an improvement program proposal was transmitted to the IRSN.

- **Project Start**

  The Safety evaluation review of the CERCA Nuclear installation is driven by the IRSN which scheduled a formal “project start meeting” that took place on Wednesday December 5th 2005 in Fontenay-aux-Roses (IRSN head office).

  During this meeting, it was reminded the duties of each party, the way to work together and the main milestones:
  - Project organization on both sides (IRSN & CERCA)
  - IRSN experts assignments in CERCA
  - Discussions
  - Safety files delivery by CERCA to IRSN
  - Safety evaluation by the IRSN experts
  - Factory Permanent Group meeting preparation

  - **Evaluation by IRSN**

    This period took place between the project start and the safety files delivery to the IRSN by CERCA. It was a favourable period for technical exchanges between IRSN and AREVA/CERCA.
In ten months we had about 30 technical joint meetings.

As decided before, the relationship between the people was maintained very open in order to avoid any misunderstanding.

The following subjects were addressed:

- Criticality

Product sub-criticality follow-up during fabrication:

It is to demonstrate that, in any normal situation, the fabrication conditions allow to maintain $K_{eff} < 0.950$ and in any accident situation, $K_{eff} < 0.975$.

No accident occurrence in case of single failure:

Specific sketches have been elaborated in order to ensure that a double check is systematically done in case of a single criticality control mode

Example of specific sketch established to verify the presence of double check in case of single criticality control mode – case of a part of the uranium alloy elaboration process

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- Human factor (Tokaï-Mura accident experience feedback)

  Consequences of high constraints on the safety during fabrication:

  The purpose of this study is to identify the risk of overstepping the red line by the operator in case of constraints in his work.

  An investigation program has been launched in order, first, to determine the sensitivity of CERCA to the human factor, second, to evaluate whether or not, specific measures should be taken. The methodology is based on an interview of the operators.

  Work post experience feedback evaluation

  Establishing the safety / criticality basic requirements & rules applicable to the work post

  Operator interviews

  Analysis

  Validation

  Action plan (if any)

  Current conclusions are that CERCA is quite sensitive to the human factor (indeed, there is one operator on each machine) and that the safety instructions are well understood and well observed.

- Radioprotection (internal exposure)

  In CERCA, the internal exposure of the operators is very low. Every handling of material is done under glove boxes or with the protection of a mask. Nevertheless, a few improvements are on going on some work posts organisation.

- Radiological cleanliness / Material dissemination

  An evaluation was made on the safety of containments breaks during normal operation such as opening of a glove box airlock. A few minor improvements may be implemented.

- Seismicity

  The main seismicity issues were addressed during the previous evaluation review of the installation. A few equipments like storage compartments, tables, etc. remain to be fixed in order to fit with the current rules.

- Fire

  A complete fire risks evaluation has been conducted and ends up in a calorific load clearance which is on-going. Finally, the purpose of this study is to demonstrate that the local occurrence of a fire could not spread everywhere so as to set fire to a large part of the workshop.

- Equipment ageing
Each automated machine was analysed in order to identify if a loss or a defect of the control system could have consequences on the safety of the installation. The conclusions were that the safety is not sensitive to our automatisms.

- **External risks (rain, snow, wind, storm, …)**

  Series of risk evaluation have been requested by the IRSN to be conducted in the next 2 years. Those evaluations are on-going now.

- **Aggression risks (gas explosion, truck explosion, plane crash, …)**

  Same as above.

  A gas delivery cabinet will be moved away from the CERCA building in order to remove any accident due to a gas pipe breakdown.

- **Hydrogeology**

  A survey plan has been initiated in order to improve our capability to detect a potential contamination of the ground.

- **Waste management**

  This issue is managed at the site level. A global project is in charge of evacuating the wastes to the specialized sites of the ANDRA in conformance with the applicable rules.

  ANDRA is the National Radioactive Waste Management Agency. “ANDRA operates independently from the waste producers. …. It is responsible for the long term management of the waste produced in France.”

  A selective sorting leads to direct the waste, either directly to the storage sites, or to the compacting facility of AREVA.

All those subjects were discussed with, and evaluated by the IRSN. Some of them where addressed during the preparation period of the Factory Permanent Group of Experts meeting. Some others require more time and so, a commitment from the INB 63.

The IRSN requested CERCA to produce nearly 20 safety analysis technical documents that were transmitted in due time. The IRSN was satisfied with the quality of those documents.

- **Preparation of the Factory Permanent Group**

  It is the custom to organize a joint meeting between the IRSN and the operator in order to find acceptable solutions for the items that have not been agreed during the safety evaluation period.

  This meeting is very important as it states on most of the issues.

  The meeting took place on October 17th 2006. Its base of work was the IRSN report of INB 63 safety evaluation.

  During the meeting, we confirmed the commitment of AREVA/CERCA to precise and improve the safety system of reference of the installation where necessary. Also, we agreed together on several pending issues.
Factory Permanent Group meeting

The Factory Permanent Group of Experts meeting took place on November 29th 2006 and was preceded one week earlier by a visit of the installation by all the members (40 persons).

The purpose of this meeting is clearly to state on the “authorization to operate” renewal.

The expert members must be convinced by both the IRSN and CERCA that the installation and its organization are in condition to allow a safe operation. Also, it is to ensure that the tool will be improved and maintained during the next ten years.

During this meeting, the IRSN presented the conclusions of the INB 63 safety evaluation as well as the commitment of the operator as discussed during the preparatory meeting. There were some discussions between the members of the Permanent Group, the IRSN and AREVA/CERCA about pending issues. CERCA proposed an improvement plan with regard to the recommendations of the Permanent Group. This improvement plan is in progress now ad is very carefully followed by the ASN.

Finally:

« A l’issue de l’examen des documents que vous avez transmis à l’ASN et ses appuis techniques, …, je n’émet aucune objection à la poursuite de l’exploitation mentionnée en objet. »

The authorization to operate is delivered to CERCA.

Factory Permanent Group pursue

The project does not end. It is continuing!

Our authorization to proceed is bound with our wish to make progress.

For this, the CERCA project team has been maintained in order to perform all the improvements required by the conclusions of the FPG. Whole of the actions, recommendations and commitments have been assessed and scheduled with milestones to return to the ASN.

The top management of AREVA / CERCA is very committed.

Studies and works are on-going on line with the schedule. The ASN is in charge of checking the progress of the project through regular inspections on the basis of the IRSN ratification of the CERCA files and works.

3 CONCLUSION

Getting the ASN authorization to proceed was a major issue for CERCA.

CERCA is authorized to operate till end of 2016. We were able to fit with the very high requirements level of the ASN, provided some improvements and investments.

The key factors of success of this project were mutual comprehension, confidence, full transparency and commitment between both parties.

The continuity of CERCA production is a reality in France but, why not anywhere else?
Poster Summary
Safety Assessment and Analysis
VALIDATION OF TECH-M-97 CODE AGAINST RESULTS OF BEMUSE PROGRAM

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ABSTRACT

International BEMUSE program (BEST-ESTIMATE METHODS-UNCERTAINTY AND SENSITIVITY EVALUATION) was organized in 2003 by the Organisation for Economic Co-operation and Development Nuclear Energy Agency (OECD/NEA). The main purpose of the BEMUSE program is to evaluate the capability of modern computer codes to simulate processes in pressurized water reactors during Large Break Loss Of Coolant Accident (LBLOCA). Thirteen organizations from nine countries with six computer codes are the participants of BEMUSE program. The program consists of two main stages. At the first stage participants of the program performed the post-test calculations of experiment with Large Break Loss Of Coolant Accident, implemented about 20 years ago on the large-scale facility LOFT (USA) nuclear-powered. Earlier the experiment was the basis for international standard problem ISP-13. The main result of the stage was a comparison of the calculated parameters of the installation with the parameters obtained in the specified experiment. Within the framework of the first stage the estimation of uncertainty and sensitivity of the calculation results was also carried out. The description of the results of all participants is submitted in detail in the special OECD/NEA report that was published in 2006. The calculation results obtained by Russian design organization OKB “Gidropress” are in good agreement with the results of LOFT experiment and correlate with the results calculated by well known Western codes (like RELAP, ATHLET, CATHARE). The second stage of BEMUSE program envisages activities similar to those of the first stage, but with reference to Zion nuclear power plant (4-loop pressurized water reactor of 3250 MWth). Each participant is supposed to provide a calculation of main coolant pipeline break (at reactor inlet), and then estimate of uncertainty and sensitivity of the results obtained. The results obtained by OKB “Gidropress” with TECH-M-97 code in the framework of the second stage of BEMUSE program are presented in this paper.

1 INTRODUCTION

At present overwhelming majority of thermal-hydraulic safety analyses of NPP with WWER type reactor are being performed using TRAP-97 code package. TRAP-97 code package whose constituent part is TECH-M-97 code [1-4], was developed in OKB “Gidropress”. TECH-M-97 code is used for analysis of changes of primary coolant parameters and temperature conditions in the core during accidents caused by loss of integrity of the primary circuit, guillotine break of the main coolant pipeline included. The program was validated against plenty of domestic and foreign experiments, including international standard problems arranged by IAEA and OECD/NEA [4-7]. TECH-M-97 code was certified by Gosatomnadzor of Russia in 1999. Russian design organization “Gidropress” took part in international BEMUSE program for an additional validation of TECH-M-97 code. The results obtained at the first stage by OKB “Gidropress” are in good agreement with the results of LOFT experiment [7]. E.g., the peak cladding temperature (about 1057 K in experiment) was predicted with an error of about 10 K and the time of peak cladding temperature (about 12.6 s in experiment) with an error of about 1 s. The calculation results obtained with TECH-M-97 code correlate with the results calculated by well-known western codes (like RELAP, ATHLET, CATHARE).
2 CALCULATION MODEL OF ZION NPP

TECH-M-97 code solves the discontinuity, energy and momentum equations written down in one-dimensional approximation and the equation of state for calculation of coolant parameters. One-dimensional equation of thermal conductivity is used for determination of temperature field in the fuel rod and metalwork. Neutron kinetics equation written down in point approximation with account for six groups of delayed neutrons is used in the calculation of reactor power. The computer code makes provision for possible application of different procedures and correlations intended to determine the heat exchange conditions, pressure loss coefficients, modelling of coolant phase-separation processes in the reactor chambers and critical discharge of water, steam and steam-water mixture.

Structurally TECH-M-97 computer code is a set of interconnected modules and computer codes [4, 7].

2.1 Fuel rod model

TECH-M-97 computer code uses TVEL program for determination of temperature field in a fuel rod. Transient temperature field in a fuel rod is determined in TVEL code by solving one-dimensional thermal conductivity equation by difference method with a known variation of thermal-and-physical parameters of coolant, coolant flow rate and heat rate in the fuel. The procedure involves calculation of thermal conductivity between fuel pellet and cladding and calculation of heat generated during reaction between the cladding material and coolant. Possibility of deformation of fuel rod cladding due to difference between pressure of gaseous medium inside the fuel rod and external pressure is taken into account. Time histories of the heat generation rate in the fuel rod and axial power distribution are considered to be known for the whole transient conditions under consideration. Power distribution over the fuel pellet cross-section is supposed to be uniform. The thermal-and-physical parameters of coolant (pressure, enthalpy, temperature) and coolant flow rate at each moment are supposed to be known.

The centreline hole surface, external surface of fuel pellet, internal and external surfaces of cladding are supposed to be presented in any fuel rod cross-section as concentric circumferences (Figure 1). The temperature values in the equidistant points from the centre of these circumferences are equal.

![Figure 1 Nodalization of fuel rod](image)

Dimensions of the fuel pellet and cladding and gap pressure are used in the input file. All the indicated parameters correspond to the “cold” condition (20 °C). Then TVEL program automatically defines the geometrical sizes of pellet/cladding and gap pressure taking into
account their thermal expansion. The gaseous medium inside the fuel rod is supposed to conform the equation of state for ideal gas. Pressure of the gaseous medium filling the centreline hole, the gap between the fuel pellet and cladding, and gas plenum is taken into account. Fuel, cladding and gas properties are considered to be the known functions of the temperature.

The reaction between the material of the fuel rod cladding (zirconium alloy or zirconium-base alloy) and coolant (steam) is taken into account in accordance with a predetermined correlation.

To determine the temperature field, the fuel rod is broken down into several sections in axial and radial directions. The thermal-and-physical parameters of coolant and heat generation rate and, consequently, the radial temperature distribution are assumed to be the same in all cross-sections within one axial section except for the deformed cladding section which is considered separately.

Variation of the gap thickness and heat conductance during the LB LOCA is presented in Figures 2 and 3. It is seen in Figure 2 that gap thickness in steady-state condition (t = 0 s) is essentially smaller than in the "cold" condition. The gap thickness tends to the value, which corresponds to the "cold" condition (0.09 mm) after scram when the fuel and cladding temperatures are decreasing. Variation of the pellet and cladding temperatures, gap thickness, pressure and temperature of the gaseous medium determines the gap heat conductance during the accident (Figure 3).

The dependence of heat conductance on linear heat generation rate for different channels in the steady-state condition is presented in Figure 4.
2.2 Nodalization of reactor plant

Nodalization of reactor plant for TECH-M-97 code is shown in Figure 5 and Table 1 summarizes the code nodalization resources used. Nodalization includes the following main components:
- reactor;
- circulation loops;
- pressurizer;
- emergency core cooling system.

![Nodalization Diagram](image)

**Figure 5: Nodalization of reactor plant for TECH-M-97 code**

**Table 1 - Nodalization Resources**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Total number of hydraulic nodes</td>
<td>87</td>
</tr>
<tr>
<td>Total number of mesh points (Heat Structures)</td>
<td>811</td>
</tr>
<tr>
<td>Number of core heated channels (without bypass)</td>
<td>5</td>
</tr>
<tr>
<td>Number of axial core nodes per channel</td>
<td>12</td>
</tr>
</tbody>
</table>

*) The fuel column is simulated by 10 axial nodes.

Three components are singled out to describe the reactor: reactor core, pressure chamber and collection chamber. The pressure and collection chambers are presented by 7 and 4 control volumes, respectively. The core is simulated with a system of parallel channels.
(six parallel channels) combining the fuel rods with close power level and differing in power: five channels are heated and one channel is not-heated to simulate core bypass flow. The core channels along the height are divided into 12 sections, 10 of which simulate the fuel column, and the other two simulate the core inlet and outlet. Power of the heated channels is determined as a product of power of the medium-powered channel by the factor considering the core radial power peaking. The code makes the automatic comparison of the sum of power of the heated channels and the total core thermal power also assigned in the code. In case of disagreement of these powers, the code uniformly distributes this difference among all channels.

In determining the core radial power peaking factors of Zion NPP the data were borrowed from [8]. Table 2 gives the relative power peaking factors for the heated channels. Axial power distribution in the core is assumed similar for all channels and is given in Table 3. Table 4 contains the maximum linear powers considered in the axial discretization of the linear power profiles for the six temperature zones of the core.

Table 2: Relative Power Peaking Factors for the Heated Channels

<table>
<thead>
<tr>
<th>Channel</th>
<th>Number of fuel rods in the channel</th>
<th>Power peaking factor</th>
</tr>
</thead>
<tbody>
<tr>
<td>Peripheral channel</td>
<td>13056</td>
<td>0.80</td>
</tr>
<tr>
<td>Average channel</td>
<td>13056</td>
<td>1.0</td>
</tr>
<tr>
<td>Hot channel</td>
<td>13056</td>
<td>1.20</td>
</tr>
<tr>
<td>Hot FA in hot channel</td>
<td>203</td>
<td>1.40</td>
</tr>
<tr>
<td>Hot rod in hot FA</td>
<td>1</td>
<td>1.50</td>
</tr>
</tbody>
</table>

Table 3: Axial relative power distribution in the core

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>$H_{core}$, %</td>
<td>5  15  25  35  45  55  65  75  85  95</td>
</tr>
<tr>
<td>Power, rel. units</td>
<td>0.60  0.92  1.10  1.18  1.23  1.22  1.18  1.08  0.91  0.58</td>
</tr>
</tbody>
</table>

Table 4: Maximum Linear Power

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Hot rod in hot FA (Zone 5)</th>
<th>Average rod in average channel (Zone 2)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Bottom Level (below 1.22 m)</td>
<td>2/3 Core Height (between 1.22 - 2.44 m)</td>
</tr>
<tr>
<td></td>
<td>Bottom Level (below 1.22 m)</td>
<td>2/3 Core Height (between 1.22 - 2.44 m)</td>
</tr>
<tr>
<td>Maximum Linear Power (KW/m)</td>
<td>37.2</td>
<td>41.6</td>
</tr>
<tr>
<td>Elevation (m)</td>
<td>0.915</td>
<td>1.647</td>
</tr>
</tbody>
</table>

Each circulation loop consists of the hot leg, steam generator, cold leg and RCP. Each intact loop is divided into 16 calculated volumes, whereas the damaged loop is divided into 20 calculated volumes. The steam generator is described by seven calculated volumes. Five of them are the SG tubing, the other two are the SG inlet and outlet collectors. Pressurizer and connecting pipeline are represented by one and two volumes, respectively.

Coolant discharge is modelled out of volumes 67 and 68 for the cold leg of calculated loop 4.
Emergency core cooling system (low-pressure pumps, accumulator) is connected to volumes 2, 18, 34.

3 STEADY-STATE ACHIEVEMENT

Power plant geometry, initial and boundary conditions were assumed according to the data presented in [8]. TECH-M-97 code simulates equipment characteristic of the WWER plant, therefore it was necessary to replace the vertical steam generator of Zion NPP with horizontal steam generator keeping the same full volume, heat transfer surface area, steam and water volume relation in accordance with the data from [8].

Variation of some reactor plant main parameters under steady-state conditions (primary and secondary side pressure, primary system total loop coolant mass flow, pressurizer collapsed level) is presented in Figures 6 - 9. Parameter variation is not above 0.6% within 100 s of the process.

4 REFERENCE CALCULATION

At the onset of process the maximum flow rate through the break from the reactor coolant pump is 16744 kg/s and from the vessel side is 29130 kg/s (Figure 10). Loss of primary side integrity leads to a sharp pressure decrease (Figure 11), coolant boil-up in the core, in the upper chamber and MCP hot legs (Figure 12). Further pressure decrease leads to departure from nucleate boiling on the fuel rod surface and beginning of their heating at 0.184 s of the accident (Figure 13). Coolant parameters in the cold leg reach the condition of saturation at 3.52 s of the process (Figure 12).
As a result of leak, primary coolant inventory decreases. At 18.6 s of the transient the pressurizer is completely empty (Figure 14).

Figure 10: Coolant flowrate through the break on RCP side (1) and vessel side (2)

Figure 11: Coolant pressure in MCP hot legs (1) and pressurizer (2)

Figure 12: Coolant void fraction in MCP hot (1) and cold (2) legs

Figure 13: Minimum departure from nucleate boiling ratio

Figure 14: Coolant level in the pressurizer

Figure 15: Primary coolant pressure (1) and accumulator pressure (2)

Figure 15 provides primary and accumulator pressure variation. As the primary pressure decreases to 4.1 MPa (at 11.8 s) water injection out of the accumulators into the primary circuit begins and after pressure decrease to 1.4 MPa (19.8 s) low-pressure pumps begin to inject into cold leg of MCP.

Approximately at 60 s of the transient the accumulators get completely empty and core cooling is realized by three low-pressure pumps (Figures 16, 17).
Figure 18 provides fuel rod cladding temperature at different elevations for the average channel and hot rod in hot FA. Maximum values of fuel rod cladding temperature are reached in the sixth and seventh axial nodes approximately at 13 and 40 s of the accident and amount to 1306.44 and 1326.15 K, respectively. When temperature of the fuel rod cladding exceeds 700 °C, a zirconium-steam reaction takes place on the fuel rod surface (Figure 20).

Time of partial top-down rewet initiating was defined by significant clad temperature decrease in the topmost core node after the first PCT. This event takes place approximately at 6.5 s of the accident process (Figure 19). Time of partial top-down rewet ending was defined by stable clad temperature increase in the topmost core node. This event takes place approximately at 26.0 s of the accident process (Figure 19).

After 136.1 s of the transient the core has been quenched (Figure 18).

An important parameter that characterizes the amount of heat accumulated in the core is fuel temperature. Figure 21 provides the variation of centreline fuel temperature for the average channel and hot rod in hot FA.

5 SENSITIVITY CALCULATION

The list of varied parameters and range of their change was accepted according to [9]. All varied parameters are immediately set in the input file of TECH-M-97 code.

Sensitivity calculations for parameters 1 through 9 are performed by increasing or decreasing the value of base case parameter.

For sensitivity calculation 10 (hot/cold conditions for pellet radius) the diameter of the fuel pellet in a steady-state condition was defined from a requirement of thermal expansion of the fuel pellet at change of its temperature by ±75 K. It corresponded to a change of initial diameter of the fuel pellet by 0.007 mm.

Table 6 and Figures 23, 24 present the analysis of sensitivity of the results to the change in the input data. Parameter \( \Delta \text{PCT} \) was defined as the difference between the PCT obtained from the sensitivity run and the PCT of the base case calculation. Judging by the results, according to their influence on the peak cladding temperature the parameters can be divided into two groups:

1. Parameters, the change of which does not affect the maximum value of the fuel rod cladding temperature. The change in the moment of activation of the emergency core cooling systems (change of initial pressure in accumulators and increase of LPIS delay), change of initial accumulators level and change of containment pressure does not influence on the peak cladding temperature;

2. Parameters, the change of which results in visible quantitative and qualitative change of the fuel rod cladding temperature. The change of following parameters caused change of peak cladding temperature more than 20 K:
   - fuel conductivity;
   - gap conductivity;
   - decay power;
The change of the initial size of fuel pellet results in the change of peak cladding temperature by approximately ± 10 K.

Figure 18: Fuel rod cladding temperature at different elevations.

a) – c) Average channel; d) – f) Hot rod in hot FA
a), d) - Bottom Level (below 1,22 m);
b), e) - 2/3 Core Height (between 1,22 – 2,44 m);
c), f) - Top Level (above 2,44 m)
Figure 19: Fuel rod cladding temperature at level 3,477 m for different channels: average channel (1); hot channel (2); hot FA in hot channel (3); hot rod in hot FA (4)

Figure 20: Mass of oxidized zirconium.

Figure 21: Fuel temperature in the center of the fuel pellet for average channel (1) and hot rod in hot FA at level 2,013 m.

Table 6: Results of sensitivity calculation

<table>
<thead>
<tr>
<th>№</th>
<th>Parameter</th>
<th>ΔPCT, K</th>
<th>Notice</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Base case</td>
<td>1326,15</td>
<td>-</td>
</tr>
<tr>
<td>1</td>
<td>Fuel conductivity</td>
<td>LC 91,81</td>
<td>( \text{value}_{\text{BC}} - 0.4\ W/(m \cdot K) ) *</td>
</tr>
<tr>
<td></td>
<td></td>
<td>UC -49,3</td>
<td>( \text{value}_{\text{BC}} + 0.4\ W/(m \cdot K) ) *</td>
</tr>
<tr>
<td>2</td>
<td>Gap conductivity</td>
<td>LC 37,22</td>
<td>( \text{value}_{\text{BC}} \cdot 0.8 ) *</td>
</tr>
<tr>
<td></td>
<td></td>
<td>UC -30,44</td>
<td>( \text{value}_{\text{BC}} \cdot 1.2 ) *</td>
</tr>
<tr>
<td>3</td>
<td>Decay power</td>
<td>LC -43,5</td>
<td>Table 2 [9]</td>
</tr>
<tr>
<td></td>
<td></td>
<td>UC 41,27</td>
<td>Table 3 [9]</td>
</tr>
<tr>
<td>4</td>
<td>Initial power</td>
<td>LC -30,38</td>
<td>( \text{value}_{\text{BC}} \cdot 0.967 )</td>
</tr>
<tr>
<td></td>
<td></td>
<td>UC 43,54</td>
<td>( \text{value}_{\text{BC}} \cdot 1.033 )</td>
</tr>
</tbody>
</table>
Table 6 continued

<table>
<thead>
<tr>
<th>№</th>
<th>Parameter</th>
<th>ΔPCT, K</th>
<th>Notice</th>
</tr>
</thead>
<tbody>
<tr>
<td>5</td>
<td>Hot rod power</td>
<td>LC</td>
<td>-102.43, K</td>
</tr>
<tr>
<td></td>
<td></td>
<td>UC</td>
<td>95.68, K</td>
</tr>
<tr>
<td>6</td>
<td>Accumulator liquid volume</td>
<td>LC</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td></td>
<td>UC</td>
<td>2.31, K</td>
</tr>
<tr>
<td>7</td>
<td>Accumulator pressure</td>
<td>LC</td>
<td>-3.3, K</td>
</tr>
<tr>
<td></td>
<td></td>
<td>UC</td>
<td>15.31, K</td>
</tr>
<tr>
<td>8</td>
<td>Containment pressure</td>
<td>LC</td>
<td>-11.03, K</td>
</tr>
<tr>
<td></td>
<td></td>
<td>UC</td>
<td>10.85, K</td>
</tr>
<tr>
<td>9</td>
<td>Hot/cold conditions for pellet radius</td>
<td>LC</td>
<td>36.72, K</td>
</tr>
<tr>
<td></td>
<td></td>
<td>UC</td>
<td>-</td>
</tr>
<tr>
<td>10</td>
<td>Accumulator liquid volume</td>
<td>LC</td>
<td>10.79, K</td>
</tr>
<tr>
<td></td>
<td></td>
<td>UC</td>
<td>-7.79, K</td>
</tr>
</tbody>
</table>

* - Parameter was changed core wide

![Figure 22: Sensitivity No5. Linear power for hot rod in hot FA](image1)

![Figure 23: Analysis of sensitivity of results to input data](image2)
6 CONCLUSION

Calculations in the frame of BEMUSE program have confirmed the capability of TECH-M-97 to simulate the processes in reactor plants with WWER and PWR type reactor. In particular, calculations have shown that the core differences (square and triangular grid of fuel rods arrangement) and SG differences (vertical and horizontal) are not very important for core cooling during a LBLLOCA.

Participation of OKB “Gidropress” in BEMUSE program allows evaluating the adequacy of computer codes to simulate processes in pressurized water reactors during LBLOCA and to perform additional validation of TECH-M-97 code used in OKB “Gidropress” for safety assessment of WWER NPPs.

The calculation results obtained with TECH-M-97 code correlate with the results calculated by well-known Western codes (like RELAP, ATHLET and CATHARE).

REFERENCES


CATHARE Assessment of PACTEL LOCA Experiments with Accident Management

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ABSTRACT

This paper summarizes the analysis results of three PACTEL experiments, carried out with the advanced thermal-hydraulic system computer CATHARE 2 code as a part of the second work package WP2 (analytical work) of the EC project “Improved Accident Management of VVER nuclear power plants” (IMPAM-VVER). The three LOCA experiments, conducted on the Finnish test facility PACTEL (VVER-440 model) represent 7.4% cold leg breaks with combination of secondary bleed and primary bleed and feed and different actuation modes of the passive safety injection. The code was used for both defining and analyzing the experiments, and to assess its capabilities in predicting the associated complex VVER related phenomena. The code results are in reasonable agreement with the measurements and the important physical phenomena are well predicted, although still further improvement and validation might be necessary.

1 INTRODUCTION

This study was carried out in the framework of the EC project “Improved Accident Management of VVER nuclear power plants” (IMPAM-VVER) with participation of Finland, France, Germany, Hungary, Czech Republic, Slovakia and Bulgaria. The objective of the project was to gain experimental and analytical results in order to improve the safety management practices and provide information for both utilities and safety authorities. In some VVER small break LOCA scenarios it has been found out that there may be problems to depressurize the primary system in order to allow the emergency core coolant injection from the low-pressure system. The main objective of this project was to investigate which means and criteria for starting depressurization measures, like feed and bleed, would be most efficient. Also it had to be assessed the capability of computer codes like APROS, ATHLET, CATHARE and RELAP to predict the associated complex VVER related phenomena. The experiments have been performed on the Finnish test facility PACTEL and Hungarian rig PMK-2. This paper presents the modeling and the results of CATHARE calculations, compared to the three PACTEL experiments. More details and results are provided in [1].
2 DESCRIPTION OF THE PACTEL EXPERIMENTAL FACILITY

The PACTEL experimental facility was designed to model the thermal-hydraulic behavior of VVER-440 type pressurized water reactors (PWR). These reactors have several unique features that differ from other PWR designs. PACTEL simulates all the major components and systems of the reference VVER-440, making it a realistic tool to examine a broad range of postulated accidents and operational transients [2].

PACTEL is a volumetrically scaled (1:305) facility including core, cold and hot legs, steam generators, main coolant pumps, pressurizer, high and low pressure emergency core cooling systems and hydro-accumulators. The maximum operating pressures on the primary and secondary sides are 8 MPa and 4.6 MPa, respectively. The corresponding values in VVER-440 are 12.3 MPa and 4.6 MPa. The reactor vessel is simulated with separate downcomer and core sections. The core itself consists of 144 full-length, electrically heated fuel rod simulators with a heated length of 2.42 m. The axial power distribution is a chopped cosine with a peaking factor of 1.4. The maximum total core power output is 1 MW, 20% of scaled full power. The fuel rod pitch (12.2 mm) and diameter (9.1 mm) are identical to those of the reference reactor. The rods are divided into three roughly triangular-shaped parallel channels representing the intersection of the corners of three hexagonal VVER rod bundles.

Component heights and relative elevations correspond to those of the full-scale reactor to match the natural circulation gravitational heads in the reference system. The hot and cold leg elevations of the reference plant have been maintained, including the loop seals. To preserve flow regime transitions in the horizontal sections of the loop seals under two-phase flow conditions the Froude number has been applied to select the diameter and length of the hot and cold legs. Three coolant loops with double capacity steam generators are used to model six loops of the reference power plant. The steam generators (SG) have vertical primary collectors and horizontal heat exchanging tubes. The external and internal SG tube diameters are 16 mm and 13 mm as in real NPP. The scaled heat transfer area of the tubes is preserved. Secondary side steam production is vented through control valves directly to the atmosphere.

3 PACTEL MODELING BY CATHARE

The calculations have been performed with the system thermal-hydraulic code CATHARE 2, version V1.3L_1.
The input data deck has been prepared on the basis of the CATHARE nodalization [3] of the PACTEL facility, which was used for ISP-33 analysis [4]. The input model has been modified in order to correspond to the PACTEL state [2] of the experiments.

The main modifications are as follows:

- The full-length steam generators have been replaced by the model of Large Diameter SGs with shorter heat exchange tubes but with real SG collectors;
- Main coolant pumps have been added;
- ECCS has been modeled (Hydro-Accumulators and LPSI pump)

The three real loops of PACTEL are modeled separately because of some differences in the lengths, elevations etc.

The core vessel (Fig.3-1) is modeled by an average core channel with 11 axial meshes and weight 144 and a bypass with 11 axial meshes. The model of the upper plenum consists of a volume with 2 core and bypass inlet junctions and 3 outlet junctions (hot legs).

![CATHARE core model of PACTEL](image)

The pressurizer presents a volume with an external wall and 3 internal walls, modeling the heaters. For the modeling of the steam generators a multitube approach is applied. The heat exchange tubes of every SG, primary side, are presented as 9 axial elements, located at different horizontal elevations. Every axial element is divided in 10 meshes. The pressurizer is connected in the hot leg of Loop 1. The break is located in the cold leg of Loop 3 close to the reactor vessel.
As a whole the primary side contents 92 junctions, 1 tee element, 10 volumes and 40 axial elements with 539 segments. Fig.3-2 illustrates the modeling of the primary circuit of PACTEL.

The secondary side of the SG is presented by recirculation model. Every one of the secondary circuits comprises 4 junctions, 1 volume for the steam dome and 2 axial elements, modeling the SG liquid pool and the steam line with 22 segments.

The heat losses to the environment are modeled based on the information of the previous PACTEL configuration with some corrections taking into account the PACTEL heat losses test [5] (for example 9.5 kW per RCP etc.).

In the junction between the core and upper plenum the CATHARE Kutateladze model for CCFL has been applied. The CATHARE CCFL operator allows the user to specify the parameters $M$, $C$, $E$ and $X$ in the flooding equation:

$$
\left[ J_G^* Bo^* \right]^X + M \left[ J_L^* Bo^* \right]^X = C \tag{1}
$$

where $Bo$ is the Bound number and $J_G^*$ and $J_L^*$ are the dimensionless superficial velocity of gas and liquid respectively.

The peak cladding temperature is very sensitive to CCFL model and plays an important role in the considered scenario of the transient.
4 RESULTS OF THE CATHARE CALCULATIONS AND COMPARISON WITH THE EXPERIMENTS

4.1 Test T2.1 analysis

The test T2.1 represents a 7.4% (7.8 mm) cold leg break with secondary bleed and primary bleed and feed. The bleed and feed occur if predefined core heat-up takes place. Regarding the ECCS configuration, the hydro-accumulators and the LPSI pump are available [6].

The objective of Test T2.1 was to investigate whether the primary pressure can be reduced to the LPSI delivery pressure without high pressure injection in LOCA scenario.

The initial primary pressure in the experiment was close to the maximum operating pressure of the facility and the lower maximum power was compensated by decreasing the primary mass flow so that temperature distribution in the initial phase in the facility is as close as possible to the nominal temperature distribution in the plant.

The main conditions of the test are the following:
- the test is started from nominal conditions of the loop by opening the break in cold leg and initiating simultaneously:
  - scram
  - steam line and feedwater isolation
  - switch off pressurizer heaters
  - pump coast down
- injection of 1 accumulator to upper plenum, 2 accumulators to downcomer
- secondary bleed starts at Twall > 350 °C
- primary bleed starts if Twall > 400 °C
- LPSI starts at P < 0.7 MPa
- test is terminated if Twall > 450 °C

The sequence of the main events of the pre-test and post-test calculations and comparison with the measured parameters are provided in the Table 4.1.

Table 4.1: Test T2.1: Timing of the main events, CATHARE vs. experiment

<table>
<thead>
<tr>
<th>EVENT</th>
<th>TIME [s] Exp.</th>
<th>TIME [s] Pre-test</th>
<th>TIME [s] Post-test</th>
<th>COMMENT</th>
</tr>
</thead>
<tbody>
<tr>
<td>Start of calculation</td>
<td>- 1500</td>
<td>- 5000</td>
<td>- 5000</td>
<td>Stabilization</td>
</tr>
<tr>
<td>Opening break valve</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>7.4 % cold leg break (Ø 7.8 mm)</td>
</tr>
<tr>
<td>Reactor scram</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td></td>
</tr>
<tr>
<td>Pumps coast-down</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>Coast-down linear (0- 150 s)</td>
</tr>
<tr>
<td>Isolation of feedwater and steam lines</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>Closing time is 3 s and 10 s, respectively</td>
</tr>
<tr>
<td>Pressurizer heaters off</td>
<td>18.9</td>
<td>18.9</td>
<td></td>
<td>Level in PRZ &lt;2.7m</td>
</tr>
<tr>
<td>ACCU injection initiated to Downcomer to Upper plenum</td>
<td>50</td>
<td>60</td>
<td>60</td>
<td>Primary pressure &lt; 5.5 MPa</td>
</tr>
<tr>
<td>EVENT</td>
<td>TIME [s] Exp.</td>
<td>TIME [s] Pre-test</td>
<td>TIME [s] Post-test</td>
<td>COMMENT</td>
</tr>
<tr>
<td>-------------------------------------------</td>
<td>--------------</td>
<td>-------------------</td>
<td>-------------------</td>
<td>----------------------------------------------</td>
</tr>
<tr>
<td>END of HA injection to UP to DC</td>
<td>450</td>
<td>582</td>
<td>586.4</td>
<td>HA empty</td>
</tr>
<tr>
<td>Increase of fuel cladding temperature</td>
<td>940</td>
<td>900</td>
<td>920</td>
<td>Core uncovery and heat-up start</td>
</tr>
<tr>
<td>Secondary bleed</td>
<td>1230</td>
<td>1166</td>
<td>1202</td>
<td>Cladding temperature &gt; 350 °C,</td>
</tr>
<tr>
<td>Primary bleed</td>
<td>--</td>
<td>--</td>
<td>--</td>
<td>Cladding temperature &gt; 400 °C,</td>
</tr>
<tr>
<td>LPSI start</td>
<td>1250</td>
<td>1205</td>
<td>1244</td>
<td>Primary pressure &lt; 0.7 MPa</td>
</tr>
<tr>
<td>Maximal fuel cladding temperature</td>
<td>1261</td>
<td>1215</td>
<td>1267</td>
<td>Tclad, exp=379 °C</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Tclad, calc=394 °C post-test</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Tclad, calc=391 °C pre-test</td>
</tr>
<tr>
<td>LPSI pump switched off</td>
<td>2000</td>
<td></td>
<td>2000</td>
<td></td>
</tr>
<tr>
<td>End of test</td>
<td>2750</td>
<td>3600</td>
<td>2400</td>
<td></td>
</tr>
</tbody>
</table>

Regarding the boundary conditions the post-test calculations are based on the specification and measurements of test T2.1. In the post-test analysis also some modification of the singularity in the hydroaccumulator line modeling has been introduced in order to get better timing in the prediction of the maximal fuel cladding temperature, although even in the pre-test calculations the timing and the amplitude of the core heat up were predicted quite well.

Due to the coolant leakage a rapid primary pressure drop takes place (Fig.4.1.1). The pressure calculations are in good agreement with the experiment. Some overprediction can be observed during the HA injection phase.

**Fig.4.1.1 Primary and secondary pressures**

The primary pressure decrease below 5.5 MPa (after 50 sec.) leads to Hydro-Accumulators injection (Fig.4.1.2), which lasts until 450 sec. In the calculations the HA injection is relatively well predicted, but is slightly longer compared to the experiment.
Intensive core boiling takes place (Fig.4.1.3). The core liquid mass is going down and steam mass is increasing. The liquid flow in the core, downcomer and loops is stagnating around zero.

After the emptying of the HA the further decrease of the primary mass inventory leads to core uncovery and core heat up starts at t=940 sec. in the test, at t=920 sec. in the post-test calculations (Fig.4.1.4) and 900 sec in the pre-test. The timing and the temperature peak are very well predicted by the calculations.
According to the scenario when the maximal cladding temperature exceeds 350°C the operator starts the secondary bleed by opening the steam dump device to the atmosphere BRU-A (t=1230 sec. experiment, t=1202 post-test, t=1166 sec. pre-test). The calculated secondary pressure is decreasing in good agreement with the experiment (Fig.4.1.1).

The further decrease of the primary pressure leads very soon after the operator intervention to LPSI pump injection, (Fig.4.1.2, t=1250 sec. test, t=1244 sec. post-test and t=1205 sec. pre-test). With the LPSI the core heat up is stopped, core quenching occurs and the Tmax are going down (Fig.4.1.4). So Tmax does not reach the criterion for primary bleed (400°C), neither in the test nor in the post-test and pre-test calculations. A stable cool down of the reactor vessel and primary circuit is achieved without primary bleed. It should be noted that the threshold for LPSI (0.7 MPa) could be reached even without operator intervention.

4.2 Test T2.3 analysis

The test T2.3 is similar to test T2.1, but the pressure set-point for hydroaccumulators injection is lower: 3.5 MPa instead of 5.5 MPa and the water volume is increased.

The objective of Test T2.3 was to investigate whether the primary pressure can be reduced to the LPSI pressure without high pressure injection in LOCA scenario and if delayed hydroaccumulators injection with lower pressure set-point is more favorable for core cooling (to reach LPSI before core overheating occurs).

Until 500 seconds the primary pressure is very well predicted (Fig.4.2.1). Between 500 sec and 900 sec. the pressure drop is faster in the experiment than in the calculations. Probably this is related to the start of HA injection and stronger condensation in the experiment than in the calculations in the upper plenum. The cold water penetration into the core is quite sensitive to the CCFL modeling in CATHARE (strong dependence on the geometry and corresponding relationships).
The primary pressure decrease below 35 bars (after 422 sec. in the test and 441 sec. in the calculations) leads to Hydro-Accumulators injection.

Intensive core boiling takes place. In the calculations a small core uncovery occurs between 287 sec. and 441 sec., which is not observed in the experiment. The core liquid mass is going down and steam mass is increasing. The liquid flow in the core and in the down comer is stagnating around zero. The maximal fuel cladding temperature (Fig.4.2.2) remains below the threshold values to begin operator actions of secondary and then primary bleed (350°C and 400°C respectively).

The further decrease of the primary pressure leads to LPSI pump actuation. It should be noted that the threshold for LPSI (0.7 MPa) has been reached without operator intervention. A stable cool down of the reactor vessel and primary circuit is achieved (Fig.4.2.2, Fig.4.2.3).
without secondary and primary bleed. The delayed HA injection (reduced HA pressure set-point) had a favorable effect on the core cooling: no overheating occurred.

4.3 Test T3.2 analysis

The test T3.2 is similar to test T2.1, but secondary bleed is not actuated even if there are conditions to start it. The primary bleed and feed occur if predefined core heat-up takes place. Based on the experience from the earlier tests the temperature criterion to start the primary bleed was increased.

The objective of Test T3.2 was to investigate the effect of low pressure injection in the conditions that the secondary side pressure remains high.

The main conditions of the test are similar to T2.1 with exception of the following:
• no secondary bleed starts even if $T_{wall} > 350 \, ^\circ C$
• primary bleed starts if $T_{wall} > 500 \, ^\circ C$
• test is terminated if $T_{wall} > 550 \, ^\circ C$

The calculated primary and secondary pressures are in good agreement with the measurements (Fig.4.3.1). Small overprediction of the primary pressure can be observed during the HA injection period. It is due probably to the modeling of HA and some underestimation by CATHARE of the condensation effects.
Fig.4.3.1 Primary and secondary pressures

The primary pressure decrease below 55 bars leads to Hydro-Accumulators injection (Fig.4.3.2), which is well predicted by the pre-test and post-test calculations.

Fig.4.3.2 ECCS mass flows

With the LPSI start the break flow is increasing again (Fig.4.3.3). The comparison of the calculated and measured break flows shows a good agreement.
After the emptying of the HA the further decrease of the primary mass inventory leads to core uncover and core heat up starts at $t=1082$ sec. in the test, at $t=900$ sec. in the pre-test calculations and 1055 sec in the post-test (Fig.4.3.4). The maximal fuel cladding temperature is achieved at 1476 sec. in the experiment and 1470 sec. in the post-test. So the post-test results have been largely improved in the timing and amplitude and a very good agreement can be observed. It should be pointed out that $T_{\text{max}}$ is an extremely sensitive parameter.

The further decrease of the primary pressure ($P_1 < 0.7$ MPa) leads to LPSI pump injection, which is well predicted in the post-test calculations (Fig.4.3.2). With the LPSI the core heat up is stopped, core quenching occurs and the cladding temperature is going down (Fig.4.3.4). So $T_{\text{max}}$ does not reach the criterion for primary bleed (500°C), neither in the test nor in the post-test and pre-test calculations. A stable cool down of the reactor vessel and primary circuit is achieved without primary bleed. The threshold for LPSI was reached without operator intervention.
CONCLUSIONS

The main objective of the IMPAM-VVER project was to investigate experimentally and analytically the means and criteria in case of SB LOCA to depressurize the primary circuit to the value of the LPSI pump head without high pressure injection before core heat up takes place. The available measures for cooldown and pressure reduction are the hydroaccumulator injection and operator actions of secondary bleed and primary feed and bleed.

Correct definition of the initial and boundary condition of the tests is important for the proper code predictions. Global parameters as pressures, mass inventory etc. are less sensitive compared to fuel cladding temperature, which is a key criterion in the safety studies and in the test scenarios.

The investigated break size of 7.4% is close to the spectrum of intermediate break LOCAs. Because of the relatively big break size, it was observed relatively fast primary pressure decrease and the value of LPSI pump head (0.7 MPa) was reached even without operator actions in the code calculations as in the tests.

In the code calculations of test T2.1 and T3.2 (higher HA pressure set-point actuation) as in the tests, boiling crisis and core heat up took place, but the maximal heater rod wall temperatures did not exceed the predefined criteria for primary bleed. Timing and value of Tmax are well computed by CATHARE code.

With the start of LPSI the core heat up is stopped, core quenching occurs and the maximal cladding temperature starts to decrease. So Tmax does not reach the criterion for primary bleed, neither in the tests nor in the calculations.

Test T2.3 was carried out with delayed HA injection (reduced HA pressure set-point). This measure had a favorable effect on the core cooling: no overheating occurred. Tmax remained below 350 °C. This effect was reproduced well by CATHARE code calculations. No secondary and no primary bleed took place.

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HEMERA: a 3D Coupled Core-Plant System for Accidental Reactor Transient Simulation

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ABSTRACT

In the framework of their collaboration to develop a system to study reactor transients in “safety-representative conditions”, IRSN and CEA have launched the development of a fully coupled 3D computational chain, called HEMERA (Highly Evolutionary Methods for Extensive Reactor Analyses), based on the French SAPHYR code system, composed of APOLLO2, CRONOS2 and FLICA4 codes, and the system code CATHARE. It includes cross sections generation, steady-state, depletion and transient computation capabilities in a consistent approach. Multi-level and multi-dimensional models are developed to account for neutronics, core thermal-hydraulics, fuel thermal analysis and system thermal-hydraulics, dedicated to best-estimate simulations and sensitivity analysis. Currently Control Rod Ejection (REA) and Main Steam Line Break (MSLB) accidents are investigated. The HEMERA system is presently applied to French PWR. The present paper outlines the main physical phenomena to be accounted for in such a coupled computational chain with significant time and space effects. A selection of results is presented along with a comparison of the available levels of 3D simulation, ranging from assembly-wise to pin-wise in the core.

1 INTRODUCTION

Safety accident analyses must demonstrate the respect of the safety criteria. The demonstration was traditionally performed assuming the most penalizing initiator. To do this, one has to set up neutronics, thermal and thermal-hydraulics modelling to simulate normal and accidental transients. In principle, one should make the analysis for the three fields at the same time because:

- The cross-sections are dependent on the fuel temperature and the moderator density,
- The fuel temperature depends on the nuclear power and the thermal exchange with the moderator fluid,
- The thermal-hydraulic depends on the source term corresponding to the power released by convection and by γ radiation.
In the past, in the methods adopted for safety reports, the three fields have been more or less decoupled. Incorporating full three-dimensional (3D) models of the reactor core into system transient codes enables a “best-estimate” calculation of the interactions between the core behaviour and the plant dynamics. In the last years, progress in computer technology has made the development of coupled thermal-hydraulic (T-H) and neutron kinetics code systems feasible and a lot of effort has been made in the reactor physicist community to develop and validate coupled codes systems [see, for example, ref. 1, 2, 3].

The objectives of the HEMERA (Highly Evolutionary Methods for Extensive Reactor Analyses) system are to perform coupled (neutronics and thermal hydraulics) 3D calculations and to develop calculation schemes for safety analysis, in association with uncertainty and sensitivity study capability and penalization techniques.

The first part of this paper is dedicated to the description of the new HEMERA chain, based on the French SAPHYR code system, including APOLLO2, CRONOS2 and FLICA4 codes, as well as the system code CATHARE.

The second part of the paper presents two PWR applications of the HEMERA system, the Rod Ejection Accident (REA) and the Main Steam Line Break (MSLB).

Finally, a conclusion presents the main perspectives of this work.

2 DESCRIPTION OF THE HEMERA SYSTEM

The Figure 1 gives an overview of the HEMERA system [4]. Codes assembled in HEMERA are:

- CRONOS2 for the neutronics core calculation,
- FLICA4 for the thermal-hydraulics core calculation
- CATHARE for primary and secondary circuits modelling.

![Figure 1: Description of the HEMERA system](image_url)

The CRONOS2 code [5] is used with the neutrons diffusion approximation, on homogenized (locally heterogenized) assembly-type geometry, a limited number of energy groups (typically two) is chosen, 4 meshes per assembly are classically defined and the cross sections are read from the multi-parameters libraries computed by lattice calculations with the APOLLO2 code [6].
The FLICA4 code [7] solves the fuel thermal equation on one-dimensional geometry and the two-phase flow in 3 dimensions. The two-phase mixture is modelled by a set of four balance equations—mass, momentum and energy of mixture, and mass of steam. The velocity disequilibrium is taken into account by a drift flux correlation. The user can choose the closure laws for wall friction, drift flux and heat transfer and the correlations for critical heat flux, depending on the fluid, the geometry and operating conditions (e.g. the pressure). The numerical method is finite volume, based on an extension of Roe's approximate Riemann solver to define convective fluxes and on the VF9 scheme to estimate the diffusive fluxes. To go forward in time, a linearized conservative implicit integrating step is used, together with a Newton iterative method.

The CATHARE2 code [8] is a best-estimate system code developed by CEA, EDF, AREVA-NP and IRSN for PWR safety analysis, accident management, definition of plant operating procedure and for research and development. Two-phase flows are modelled using a two-fluid six-equation model. There are several modules for 0D, 1D or 3D.

Each of these codes can be run separately or coupled in the framework of stationary and transient calculations.

For the coupling, an explicit technique which consists in solving the neutronics and thermal-hydraulic equations separately has been adopted in the system. Driving codes and data exchanges are managed by the ISAS software [9], based on PVM (Parallel Virtual Machine).

The HEMERA system is currently adopted to simulate two accidental transients: the Rod Ejection Accident (REA) and the Main Steam Line Break (MSLB). Two types of coupling are carried out:

- The first one occurs between CRONOS2 and FLICA4 (neutronics feedbacks updating) and is used for REA and MSLB modelling. It consists of: i) the power distribution calculated by CRONOS2 is transferred to FLICA4 to be used as a source term in the energy balance equation of fluid and fuel; ii) the thermal-hydraulic parameters for the evaluation of cross-sections are provided to CRONOS2 (for interpolation in the cross-sections libraries);
- The second one occurs between FLICA4 and CATHARE2 (thermal hydraulics updating) and is specifically used for MSLB modelling. CATHARE2 provides mass flow, temperature and boron concentration at core inlet and pressure at core outlet for FLICA4, while FLICA4 sends back the pressure at the core inlet and the mass flow, temperature and boron concentration at the core outlet. The flow mixing between loops in the Reactor Pressure Vessel (before the core inlet and after the core outlet) is modelled by user-defined mixing coefficients.

A specificity of the HEMERA system is the possibility to make a pin by pin (neutronics)/sub-channel (thermal-hydraulics) description in a specific assembly of the core. This functionality permits to access to the main parameters of interest for REA and MSLB accidental situations which are local parameters: respectively the power peak and the DNBR. This functionality is based on a hybrid description for neutronics (CRONOS2) and primary/zoom description for the thermal-hydraulics (FLICA4).

In CRONOS2, the hybrid modelling consists of using homogeneous cross-sections everywhere except in the refined assembly where heterogeneous cross-sections are applied (cf. Figure 2).

For FLICA4, there are two levels modelling: i) a core description at fuel assembly level (or quarter of assembly); ii) a hot fuel assembly description at the sub channel level. The two
levels are coupled together through hydraulic boundary conditions: mass flow, enthalpy and pressure (cf. Figure 3).

2.1 REA transient type

The Rod Ejection Accident (REA) is a very quick expulsion of a control assembly out of the core that leads to a fast reactivity insertion and a rapid increase of the reactor power with a huge deformation of its shape. As a consequence of the sharp power increase, fuel temperature increases too, and fuel rod can be damaged with a resulting possibility of radioactive material dispersion. The increase of power is limited by the feedbacks, mainly Doppler and moderator effects, and definitively stopped by the control rods emergency shutdown (the ‘scram’).

The approach adopted to cope with these kinds of accidents is conservative, i.e., accidents are simulated starting from the worst possible initial conditions... Following this approach, the most important physical effects that can temperate a REA (control rod reactivity worth, feedback effects: Doppler and moderator, kinetic parameters) are penalized in conformity with the known experimental uncertainty. In such a way, ejected rod is the heaviest (in reactivity) of the all the groups, feedback effects are decreased and kinetics parameters are modified so as to make the reactor more reactive. In the analysis of REA, one must incorporate all possible reactor states and all possible control rod positions allowed by the operational limits and conditions of the nuclear power plant.
In the following, computations are made on a 3-loops PWR core (157 assemblies with 17x17 rods by assembly). Assemblies are UO2 (nearly 70%) and MOX (nearly 30%) at intermediate core power (40% nominal power). $\beta_{eff}$ value is around 450 pcm (1 pcm = 1. E-5) and rod worth after Xe profile modification is around 220 pcm.

Among the parameters that, nowadays, HEMERA is able to penalize, we choose, as just an example, to illustrate the effects of two penalized parameters: moderator effect and Doppler Effect. Both of them can be discussed from two different points of view: from a point of view that can be qualified as ‘Direct Approach’ and by a simulation of a ‘Point Kinetics’ type penalization.

### 2.1.1 Penalization: Direct Approach

As said in the previous paragraph, two of the parameters of the cross sections library are moderator density and fuel temperature and CRONOS2 needs to know a mesh moderator density and a fuel temperature to be able to assign cross sections set to a mesh. Moderator density and fuel temperature are issued by FLICA, which has computed them by using the power distribution computed by CRONOS in the previous time step.

A reactivity effect (Doppler or Moderator) is computed comparing two reactivities: starting reactivity and core reactivity when only moderator density or fuel temperature have been changed, all the others parameters remain exactly the same of the transient starting state.

What we quote as ‘Direct Approach’ consists of an evaluation of the parameters (fuel temperature for Doppler effect and moderator density for Moderator effect, one at a time) in such a way to give to CRONOS the cross sections set that make it able to compute the target reactivity, goal being to have the wanted reactivity effect. We emphasize that the fuel temperature and moderator density are artificially modified in order to allow to CRONOS to retrieve, in the library, the suitable cross sections set to compute the target Doppler and Moderator effects. We remind that, in CRONOS, cross sections are identified by a set of parameters (like a point in a phase space), fuel temperature and moderator density being two of them.

The research of the parameter to retrieve the suitable sections set is made by an algorithm, which works as:

- a) At the time step ‘t’, the reactivity effect is computed,
- b) A penalization is applied to the parameter, and a new, penalized, reactivity effect is computed,
- c) A research (by a Newton method, for instance) is done to compute the value of the parameter that allows to have the target reactivity effect,
- d) A convergence test is carried out; if the test is satisfied, the computation is stopped; otherwise it loops back to c) point.

### 2.1.2 Penalization: Point Kinetics Approach

Point Kinetics Approach consists of a simulation of reactivity effect computed by 0D kinetics. As matter of fact, in a point kinetics model reactivity effects are computed via a product by a coefficient (Doppler or moderator) and a parameter variation (fuel temperature or moderator density). Hence, to make use of a Point Kinetics type approach means, adopting a suitable algorithm, as follows:
a) At the time step ‘t’, a reactivity coefficient is computed (by doing a slight fuel temperature or moderator density variation around the actual values at the time ‘t’),
b) The reactivity coefficient is penalized (i.e., multiplied by a coefficient)
c) The penalized reactivity effect is computed as the product by the penalized reactivity coefficient and the parameter variation (always with all the other parameters unchanged compared to the beginning state ones)
d) A search for the parameter value (Newton method) is made, allowing to obtain the target reactivity effect,
e) A convergence test is made, should it be satisfied, the computation stops, otherwise the calculation loops back to the d) point.

2.1.3 Comparison of Results

In the following, we present two figures with the results of the two tests aforementioned: the comparisons of the core average power (in % of the Nominal Power) when Direct and Point Kinetics Approach for Doppler and Moderator effects are applied. Both the figures (4 and 5) show the reference case (Refer) and the reference case plus the Direct approach (‘Dopp’ or ‘Mod’) and the reference case plus Point Kinetics approach (‘Dopp_PK’ or ‘Mod_PK’).

Difference between Direct and Point Kinetics approaches is quite similar for both Doppler and Moderator effects:

- At the beginning of the transient, both the impact of the two different approaches is hardly measurable: the reactor has to warm up and any kind of penalization is ineffective;
- At peak power, Point Kinetics approach is already more effective (effectiveness means here ability in engendering more power in the core): average power is nearly 1% higher than for Direct approach;
- After 1 second, Point Kinetics approach continues to push the average power towards higher values. Our computation was stopped here but, probably, this behaviour should have continued even after the scram.

![Figure 4: Doppler Effect: Direct and Point Kinetics Approach](image-url)
Reasons for this bigger impact of the Point Kinetics approach reside in the non linearity of the reactivity coefficients: in our case, a reactivity coefficient computed during the transient at time ‘t’ (Point Kinetics approach) is smaller than the reactivity coefficient computed at any time before ‘t’ (and even smaller than the mean value of the reactivity coefficient between the beginning and the time ‘t’). That has the consequence that the reactivity loss is smaller with the Point Kinetics approach, hence power is bigger.

As a conclusion of this quick analysis, we can say that the PK approach is more effective (compared to the direct approach) as regards to the goal to increase the power level and that these penalization techniques are good control levers to drive power variation.

2.2 MSLB transient type

The HEMERA system has also been used for Main Steam Line Break (MSLB) studies and, more specifically, for a four-loop French PWR transient.

2.2.1 General concern

The Main Steam Line Break is a DBA (Design Basis Accident) in PWRs, which involves coupled physical phenomena such as the thermal-hydraulics of the secondary circuit, the thermal exchange between primary and secondary circuits (through the steam generator), the thermal-hydraulics of the primary circuit and both the neutronics and thermal-hydraulics of the core.

The steam release as a consequence of the rupture of a main steam line results in an initial increase in steam flow, which decreases during the accident as the steam pressure falls down. The energy removal from the RCS generates a reduction of coolant temperature and pressure. Due to the negative moderator coefficient, the RCS cool-down results in positive reactivity insertion. The reactor goes critical with a power excursion. The Doppler Effect and the boron insertion via the SI either limit or stop the power increase.

The MSLB is a dissymmetric accident because the loop corresponding to the break behaves differently from the others loops after the closure of all Main Steam Isolation Valves (MSIV). The cooling of the core isn’t uniform, which generates disequilibrium in the power distribution amplified by the control rod stuck.
2.2.2 Nodalisation

The nodalisation of the primary circuit (except for the core) with its 4 distinct loops and the secondary has been performed using 0D-1D elements of the CATHARE code as shown in the Figure 6 (only two loops out of four are illustrated).

![Figure 6: Nodalisation for the MSLB simulation]

The vessel is subdivided in four “channels”, related to each loop. The core is described in 3 dimensions (see Figure 6) with CRONOS2 and FLICA4 codes, with 4 nodes per assembly for neutronics and 1 mesh per assembly for thermal-hydraulic calculations, as well as 32 meshes on z-axis. A fine description is adopted for the hot fuel assembly (one mesh per rod). A matrix, derived from LACYDON-experiment, results simulates the mixing between the four loop flow rates and temperatures.

2.2.3 Initial state and boundaries conditions

This analysis therefore assumes a non-isolable Main Steam Line Break at hot zero power. The core (UOX fuel) is in End of Cycle configuration without Xenon. The most penalizing single failures, with regard to the DNBR (Departure from Nucleate Boiling Ratio), is a rod cluster control assembly RCCA having the highest reactivity-worth, stuck in its fully withdrawn position after the reactor trip. The Reactor Coolant Pumps (RCP) are assumed as not stopped.

2.2.4 Results

The double-ended guillotine break of the main steam line leads to a quick depressurization of the secondary side and the primary side. The energy removed from the RCS causes a reduction of coolant temperature. Reactor becomes critical and hence thermal power is increasing at 16 seconds. Figure 7 shows the evolution of the core power during the accident. The Doppler Effect limits the thermal power excursion but does not stop it. The thermal power increase is limited when the boron is injected in the core at 75 seconds. The boron propagation in the primary via safety injection lines is in the form of a front at the beginning and leads to power oscillation in the core. The time step corresponds to the time necessary for the boron front to cover all the primary circuit. Due to diffusion in the CATHARE code (mixing), this behaviour quickly disappears.
After a quick stabilization of the thermal-hydraulic parameters, a stable state (core just critical with the core power removed via the leak and Emergency Feed Water in the affected steam generator) is then reached. The maximum core power is 5.3 % NP reached at 145 seconds. The effect of a local pin description to the DNBR (French acronym is RFTC) response appears on the Figure 8.

Because of the localization of the hot assembly (between the reflector and the stuck rod; see Figure 9), the local power distribution is particularly heterogeneous that leads to a lower DNBR compared to the response issue from the primary description (coarse meshing). This result shows all the interest to carry out such a modelling.
3 CONCLUSION AND PERSPECTIVES

The current scope of neutronics and thermal-hydraulics coupling enables perform to best-estimate and penalized calculations for PWR safety analysis, in association with uncertainty and sensitivity studies.

For this purpose, CEA and IRSN are developing the HEMERA, a coupled neutronics and thermal-hydraulics computational system, based on CATHARE, CRONOS2, FLICA4 and adopting APOLLO2 as a server for data. HEMERA is now used by IRSN for PWR safety assessment with application to two main accidental transients: the Main Steam Line Break, involving the coupling between core and system, and the Rod Ejection Accident.

Taking advantage from the current experience, several main axis of improvement have already been identified by IRSN and CEA and stressed, such as:

- Necessity to use the best available models inside the coupled system (neutronics, thermal-hydraulics…).
- Modelling improvement of the impact of the thermal-mechanics of the fuel on the thermal feed-back,
- Validation of the coupled system with international benchmarks, if possible with actual plant data (e.g. Peach Bottom, Kozloduy…),
- To continue an in-deep examination of the penalizations techniques together with an analysis of uncertainties,
- New coupling techniques, including interpolation and unified data structures, definition and share of common data between coupled models, supervision of calculations.

Those improvements either are underway or will be addressed in a near future.
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APPRAOCH AND METHOD TO EVALUATE THE UNCERTAINTY IN SYSTEM THERMAL-HYDRAULICS CALCULATIONS.
KEY APPLICATIONS BY CIAU METHOD

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ABSTRACT

The evaluation of uncertainty constitutes the necessary supplement of Best Estimate (BE) calculations performed to understand accident scenarios in water cooled nuclear reactors. The needs come from the imperfection of computational tools on the one side and from the interest in using such tool to get more precise evaluation of safety margins. In the present paper the approaches to uncertainty are outlined and the CIAU (Code with capability of Internal Assessment of Uncertainty) method proposed by the University of Pisa is described including ideas at the basis and results from applications.

Two approaches are distinguished that are characterized as “propagation of code input uncertainty” and “propagation of code output errors”. For both methods, the thermal-hydraulic code is at the centre of the process of uncertainty evaluation: in the former case the code itself is adopted to compute the error bands and to propagate the input errors, in the latter case the errors in code application to relevant measurements are used to derive the error bands.

The CIAU method exploits the idea of the “status approach” for identifying the thermal-hydraulic conditions of an accident in any Nuclear Power Plant (NPP). Errors in predicting such status are derived from the comparison between predicted and measured quantities and, in the stage of the application of the method, are used to compute the uncertainty.

KEYWORDS
Uncertainty, CIAU, Assessment, Accuracy, RELAP5
1. INTRODUCTION

Deterministic safety analysis frequently referred to as accident analysis is an important tool for confirming the adequacy and efficiency of provisions within the defense in depth concept for the safety of Nuclear Power Plants. Typical upgraded international licensing environments offer two acceptable options for demonstrating that the safety is ensured with sufficient margin: use of best estimate computer codes either combined with conservative input data or with realistic input data but associated with evaluation of uncertainty of results. The second option is particularly attractive because it allows for more precise specification of safety margins and their potential use for higher operational flexibility. This constitutes the framework for the present paper.

Thermal-hydraulic system codes are needed to perform deterministic safety analyses and are suitable to calculate complex accident scenarios expected in water cooled nuclear reactors. The outputs of those codes are affected by unavoidable errors that are referred as uncertainties, notwithstanding extensive qualification programs carried out in the last three or four decades. In the current situation it can be said that the experimental programs have not been capable to prevent those errors but to identify and to characterize them.

The present paper aims at discussing the major source of errors or uncertainties, at characterizing approaches for performing uncertainty studies and at presenting one successful uncertainty method proposed by the University of Pisa.

2. THE ORIGIN OF UNCERTAINTY

Application of best-estimate (realistic) computer codes to the safety analysis of nuclear power plants implies the evaluation of uncertainties. This is connected with the (imperfect) nature of the codes and of the process of codes application. In other words, ‘sources of uncertainty’ affect the predictions by best-estimate codes and must be taken into account. Three major sources of uncertainty are mentioned in the Annex II of the IAEA guidance Accident Analyses for Nuclear Power Plants, ref. [2]:

- Code or model uncertainty.
- Representation or ‘simulation uncertainty’.
- Plant uncertainty.

A more detailed list of uncertainty sources can be found in ref. [1], where an attempt has been made to distinguish ‘independent’ sources of ‘basic’ uncertainty. The list includes the following items:

A) Balance (or conservation) equations are approximate:
   - not all the interactions between steam and liquid are included,
   - the equations are solved within cylindrical pipes: no consideration of geometric discontinuities, situation not common for code applications to the analysis of Nuclear Power Plants transient scenarios

B) Presence of different fields of the same phase: e.g. liquid droplets and film. Only one velocity per phase considered by codes, thus causing another source or uncertainty.

C) Geometry averaging at a cross section scale: the need “to average” the fluid conditions at the geometry level makes necessary the ‘porous media approach’. Velocity profiles happen in
the reality: These correspond to the ‘open media approach’. The lack of consideration of the velocity profile, i.e. cross-section averaging, constitutes an uncertainty source of ‘geometric origin’.

D) Geometry averaging at a volume scale: only one velocity vector (each phase) is associated with a hydraulic mesh along its axis. Different velocity vectors may occur in the reality (e.g. inside lower plenum of a typical reactor pressure vessel, at the connection between cold leg and down-comer, etc.). The volume-averaging constitutes a further uncertainty source of ‘geometric origin’.

E) Presence of large and small vortex or eddy. Energy and momentum dissipation associated with vortices are not directly accounted for in the equations at the basis of the codes, thus introducing a specific uncertainty source. In addition, a large vortex may determine the overall system behaviour (e.g. two-phase natural circulation between hot and cold fuel bundles), not necessarily consistent with the prediction of a code-discretized model.

F) The 2nd principle of thermodynamics is not necessarily fulfilled by codes. Irreversible processes occur as a consequence of accident in nuclear reactor systems. This causes ‘energy’ degradation, i.e. transformation of kinetic energy into heat. The amount of the transformation of energy is not necessarily within the capabilities of current codes, thus constituting a further specific energy source.

G) Models of current interest for thermal-hydraulic system codes are constituted by a set of partial derivatives equations. The numerical solution is approximate, therefore, approximate equations are solved by approximate numerical methods. The ‘amount’ of approximation is not documented and constitutes a specific source of uncertainty.

H) Extensive and unavoidable use is made of empirical correlations. These are needed ‘to close’ the balance equations and are also reported as ‘constitutive equations’ or ‘closure relationships’. Typical situations are:

- The ranges of validity are not fully specified. For instance, pressure and flow rate ranges are assigned, but void fraction, or velocity (or slip ratio) ranges may not be specified.
- Relationships are used outside their range of validation. Once implemented into the code, the correlations are applied to situations, where, for instance, geometric dimensions are different from the dimensions of the test facilities at the basis of the derivation of the correlation. One example is given by the wall-to-fluid friction in the piping connected with reactor pressure vessel: no facility has been used to derive (or to qualify) friction factors in two phase conditions when pipe diameters are of the order of one meter. In addition, once the correlations are implemented into the code, no (automatic) action is taken to check whether the boundaries of validity, i.e. the assigned ones, are over-passed during a specific application.
- Correlations are implemented approximately into the code. The correlation, apart from special cases, are derived by scientists or in laboratories that are not necessarily aware of the characteristics or of the structure of the system code where the correlations are implemented. Furthermore, unacceptable numeric discontinuities may be part of the original correlation structure. Thus, correlations are ‘manipulated’ (e.g. extrapolated in some cases) by code developers with consequences not always ascertained.
- Reference database is affected by scatter and errors. Correlations are derived from ensembles of experimental data that unavoidably show ‘scatter’ and are affected by errors or uncertainties. The experimentalist must interpret those data and achieve an ‘average-satisfactory’ formulation.
I) A paradox: shall be noted: ‘Steady State’ & ‘Fully Developed’ (SS & FD) flow condition is a necessary prerequisite or condition adopted when deriving correlations. In other terms, all qualified correlations must be derived under SS & FD flow conditions. However, almost in no region of the Nuclear Power Plant those conditions apply during the course of an accident.

J) The state and the material properties are approximate. Various materials used in a NPP are considered in the input deck, including liquids, gases and solids. Thermo-physical properties are part of the codes or constitute specific code user input data. These are of empirical nature and typically subjected to the limitations discussed under item H). A specific problem within the current context can be associated with the derivatives of the water properties.

K) Code User Effect (UE) exists. Different groups of users having available the same code and the same information for modelling a Nuclear Power Plant do not achieve the same results. UE (see also below) is originated by:
   • Nodalisation development, see also item N), below.
   • Interpreting the supplied (or the available) information, usually incomplete, see also item M) below.
   • Accepting the steady state performance of the nodalisation.
   • Interpreting transient results, planning and performing sensitivity studies, modifying the nodalisation and finally achieving “a reference” or “an acceptable” solution.

The UE might result in the largest contribution to the uncertainty and is connected with user expertise, quality and comprehensiveness of the code-user manual and of the database available for performing the analysis.

L) Computer/compiler effect exists. A computer code is developed making use of the hardware selected by the code developers and available at the time when the code development starts. A code development process may last a dozen years during which profound code hardware changes occur. Furthermore, the code is used on different computational platforms and the current experience is that the same code with the same input deck applied within two computational platforms produces different results. Differences are typically small in ‘smoothly running transients’, but may become noticeable in the case of threshold- or bifurcation-driven transients.

M) Nodalisation (N) effect exists. The N is the result of a wide range brainstorming process where user expertise, computer power and code manual play a role. There is a number of required code input values that cannot be covered by logical recommendations: the user expertise needed to fix those input values may reveal inadequate and constitutes the origin of a specific source of uncertainty.

N) Imperfect knowledge of Boundary and Initial Conditions (BIC). Some BIC values are unknown or known with approximation: the code user must add information. This process unavoidably causes an impact on the results that is not easily traceable and constitutes a specific source of uncertainty.

O) Code/model deficiencies cannot be excluded. The system code development started toward the end of the sixties and systematic assessment procedures were available since the eighties. A number of modelling errors and inadequacies have been corrected or dealt with and substantial progress has been made in improving the overall code capabilities. Nevertheless, deficiencies or lack of capabilities cannot be excluded nowadays. Examples, not applicable to all thermal-hydraulic system codes, are connected with the modelling of:
• the heat transfer between the free liquid surface and the upper gas-steam space,
• the heat transfer between a hotter wall and the cold liquid down-flowing inside a steam-gas filled region.
Those deficiencies are expected to have an importance only in special transient situations.

3. THE APPROACHES TO CALCULATE THE UNCERTAINTY

An uncertainty analysis consists of identification and characterization of relevant input parameters (input uncertainty) as well as of the methodology to quantify the global influence of the combination of these uncertainties on selected output parameters (output uncertainty). These two main items are treated in different ways by the various methods. One approach is to evaluate the ‘propagation of input uncertainties’, Fig. 1: uncertainty is derived following the identification of ‘uncertain’ input parameters with specified ranges or/and probability distributions of these parameters, and performing calculations varying these parameters. The propagation of input uncertainties can be performed either by deterministic or by probabilistic methods. The other approach, Fig. 2, is the ‘extrapolation of output uncertainty’: uncertainty is derived from the (output) uncertainty based on the comparison between calculation results and significant experimental data.

3.1. The propagation of code input uncertainty

The GRS is selected as the prototype method, ref. [3], for the description of the “propagation of code input uncertainty” approach. In these methods, the state of knowledge of each uncertain input parameter within its range is expressed by a subjective probability distribution. The word “subjective” expresses the state of knowledge rather than stochastic variability. Dependencies between uncertain input parameters should be identified and quantified. Peculiarities of the GRS method are:
• The uncertainty space of input parameters (defined by their uncertainty ranges) is sampled at random according to the combined subjective probability distribution of the uncertain parameters and code calculations are performed by sampled sets of parameters.
• The number of code calculations is determined by the requirement to estimate a tolerance/confidence interval for the quantity of interest (such as peak clad temperature). The Wilks formula is used to determine the number of calculations needed for deriving the uncertainty bands.
• Statistical evaluations are performed to determine the sensitivities of input parameter uncertainties on the uncertainties of key results (parameter importance analysis).
• There are no limits for the number of uncertain parameters to be considered in the analysis and the calculated uncertainty has a well-established statistical basis.

For the selected plant transient, the method is applied to an integral effects test simulating the same scenario prior to the plant analysis. If experimental data are not bounded, the set of uncertain input parameters has to be modified. Experts identify significant uncertainties to be considered in the analysis, including the modeling uncertainties, and the related parameters, and identify and quantify dependencies between uncertain parameters. Subjective Probability Density Functions (PDF) are used to quantify the state of knowledge of uncertain parameters for the specific scenario. The term “subjective” is used here to distinguish uncertainty due to imprecise knowledge from uncertainty due to stochastic or random variability.
Uncertainties of code model parameters are derived based on validation experience. The scaling effect has to be quantified as model uncertainty. Additional uncertain model parameters can be included or PDF can be modified, accounting for results from the analysis of Separate Effects Tests. Input parameter values are simultaneously varied by random sampling according to the subjective PDF and dependencies. A set of parameters is provided to perform the required number n of code runs. For example, the 95% fractile and 95% confidence limit of the resulting subjective distribution of the selected output quantities is directly obtained from the n code results, without assuming any specific distribution. No response surface is used or needed.

Sensitivity measures by using regression or correlation techniques from the sets of input parameters and from the corresponding output values allow the ranking of the uncertain input parameters in relation to their contribution to output uncertainty. Therefore, the ranking of parameters is a result of the analysis, not of prior expert judgement. The 95% fractile, 95% confidence limit and sensitivity measures for continuous-valued output parameters are provided. Upper statistical tolerance limits are the upper β confidence for the chosen α fractile. The fractile indicates the probability content of the probability distributions of the code results (e.g. α = 95% means that PCT is below the tolerance limit with at least α = 95% probability). One can be β % confident that at least α% of the combined influence of all the characterized uncertainties are below the tolerance limit. The confidence level is specified because the probability is not analytically determined. It accounts for the possible influence of the sampling error due to the fact that the statements are obtained from a random sample of limited size. The smallest number n of code runs to be performed is given by the Wilks formula:

$$(1 - \alpha/100)^n \geq \beta/100$$

and is representing the size of a random sample (a number of calculations) such that the maximum calculated value in the sample is an upper statistical tolerance limit. The required number n of code runs for the upper 95% fractile is: 59 at 95% confidence level, 45 at 90% confidence level, 32 at 80% confidence level. Two-sided statistical tolerance intervals can be adopted. As a consequence, the number n of code runs is independent of the number of selected input uncertain parameters, only depending on the percentage of the fractile and on the desired confidence level percentage. The number of code runs for deriving sensitivity measures is also
independent of the number of parameters. As an example, a total number of 100 runs is typical for the application of the GRS method.

3.2. The propagation of code output errors

The UMAE is the prototype method, ref. [4], for the description of “the propagation of code output errors” approach. The method focuses not on the evaluation of individual parameter uncertainties but on the propagation of errors from a suitable database calculating the final uncertainty by extrapolating the accuracy from relevant integral experiments to full scale NPP. Considering integral test facilities of a reference water cooled reactor, and qualified computer codes based on advanced models, the method relies on code capability, qualified by application to facilities of increasing scale. Direct data extrapolation from small scale experiments to reactor scale is difficult due to the imperfect scaling criteria adopted in the design of each scaled down facility. So, only the accuracy (i.e. the difference between measured and calculated quantities) is extrapolated. Experimental and calculated data in differently scaled facilities are used to demonstrate that physical phenomena and code predictive capabilities of important phenomena do not change when increasing the dimensions of the facilities.

Other basic assumptions are that phenomena and transient scenarios in larger scale facilities are close enough to plant conditions. The influence of user and nodalisation upon the output uncertainty is minimized in the methodology. However, user and nodalisation inadequacies affect the comparison between measured and calculated trends; the error due to this is considered in the extrapolation process and gives a contribution to the overall uncertainty. The method utilizes a database from similar tests and counterpart tests performed in integral test facilities that are representative of plant conditions. The quantification of code accuracy is carried out by using a procedure based on the Fast Fourier Transform characterizing the discrepancies between code calculations and experimental data in the frequency domain, and defining figures of merit for the accuracy of each calculation. Different requirements have to be fulfilled in order to extrapolate the accuracy.

Calculations of both Integral Test Facility experiments and NPP transients are used to attain uncertainty from accuracy. Nodalisations are set up and qualified against experimental data by an iterative procedure, requiring that a reasonable level of accuracy is satisfied. Similar criteria are adopted in developing plant nodalisation and in performing plant transient calculations. The demonstration of the similarity of the phenomena exhibited in test facilities and in plant calculations, accounting for scaling laws considerations, leads to the Analytical Simulation Model (ASM), i.e. a qualified nodalisation of the NPP. The flow diagram of UMAE is given in Fig. 3. The bases of the methods and the conditions to be fulfilled for its application, including the use of the FFTBM can be found in refs. [5] to [9].

4. THE CIAU METHOD

All of the uncertainty evaluation methods are affected by two main limitations:
• The resources needed for their application may be very demanding, ranging to up to several man-years;
• The achieved results may be strongly method/user dependent.
Figure 2: Uncertainty approach: propagation of code output errors.

Figure 3: UMAE flow diagram (also adopted within the process of application of CIAU).

The last item should be considered together with the code-user effect, widely studied in the past, e.g. ref. [5], and may threaten the usefulness or the practical applicability of the results achieved by an uncertainty method. Therefore, the Internal Assessment of Uncertainty (IAU) was requested as the follow-up of an international conference jointly organized by OECD and US NRC and held in Annapolis in 1996. The CIAU method, refs. [1, 10], has been developed with the objective of reducing the above limitations.
The basic idea of the CIAU can be summarized in two parts, Fig. 4:

- Consideration of plant status: each status is characterized by the value of six relevant quantities (i.e. a hypercube) and by the value of the time since the transient start.
- Association of an uncertainty to each plant status.

In the case of a PWR the six quantities are: 1) the upper plenum pressure, 2) the primary loop mass inventory, 3) the steam generator pressure, 4) the cladding surface temperature at 2/3 of core active length, 5) the core power, 6) the steam generator downcomer collapsed liquid level. A hypercube and a time interval characterize a unique plant status to the aim of uncertainty evaluation. All plant statuses are characterized by a matrix of hypercubes and by a vector of time intervals. Let us define Y as a generic thermal-hydraulic code output plotted versus time. Each point of the curve is affected by a quantity uncertainty (Uq) and by a time uncertainty (Ut). Owing to the uncertainty, each point may take any value within the rectangle identified by the quantity and the time uncertainty. The value of uncertainty, corresponding to each edge of the rectangle, can be defined in probabilistic terms. This satisfies the requirement of a 95% probability level to be acceptable to the NRC staff for comparison of best estimate predictions of postulated transients to the licensing limits in 10 CFR (Code of Federal Regulation) Part 50.

The idea at the basis of CIAU can be made more specific as follows: the uncertainty in code prediction is the same for each plant status. A Quantity Uncertainty Matrix (QUM) and a Time Uncertainty Vector (TUV) can be set up including values of Uq and Ut derived by an uncertainty methodology, Fig. 5. At the moment the UMAE constitutes the ‘engine’ for the rotation of the CIAU shaft. The QAM and TAV, respectively Quantity Accuracy Matrix and Time Accuracy Vector in Fig. 5, are derived from an UMAE like process and are the precursor of QUM and TUV. However, within the CIAU framework, any uncertainty method can be used to derive directly QUM and TUV.

5. KEY APPLICATIONS OF THE CIAU METHODOLOGY

Three main applications of the CIAU methodology with relevance to the nuclear industry are presented hereafter. More details may be found in refs 14, 15 and 16.

5.1. Uncertainty Analysis of the LBLOCA-DBA of the Angra-2 PWR NPP

Angra-2 is a 4 loop 3765 Mwth PWR designed by Siemens KWU. The NPP is owned and operated by the ETN utility in Brazil. The NPP design was ready in the ‘80s, while the operation start occurred in the year 2000 following about ten-year stop of the construction. The innovation proposed to the licensing process by the applicant consists in the use of a Best Estimate tool and methodology to demonstrate the compliance of the NPP safety performance with applicable acceptance criteria set forth in the Brazilian nuclear rule.

In this study [11], the CIAU application aimed at performing an ‘independent’ best-estimate plus uncertainty analysis of the LBLOCA-DBA of the Angra-2 PWR NPP. The analysis is classified as ‘independent’ in the sense that it was carried out by computational tools (code and uncertainty method) different from those utilized by the applicant utility. The main results are summarized in Fig. 6 and 7, where PCT and related uncertainty bands obtained by the CIAU and by the computational tools adopted by applicant, are given. The following comments apply:
- Continuous uncertainty bands have been obtained by CIAU related to rod surface temperature (Fig. 6), pressure and mass inventory in primary system. Only point values for PCT are considered in Fig. 7.
- The CIAU (and the applicant) analysis has been carried out as best-estimate analysis: however, current rules for such analysis might not be free of undue conservatism and the use of peak factors for linear power is the most visible example.
- The conservatism included in the reference input deck constitutes the main reason for getting the ‘PCT licensing’ from the CIAU application above the acceptability limit of 1200 °C.
- The amplitude of the uncertainty bands is quite similar from CIAU and applicant. Discrepancies in the evaluation of ‘PCT licensing’ outcome from the way of considering the
‘center’ of the uncertainty bands. In the case of CIAU, the ‘center’ of the uncertainty bands is represented by the phenomenological result for PCT obtained by the reference calculation (1100 °C in Fig. 6). In the case of applicant the ‘center’ of the uncertainty bands is a statistical value obtained from a process where the reference calculation has a role (796 °C in Fig. 7).
- The results of the CIAU method are supported by a number of ‘finalized’ sensitivity studies as large as about 150 (i.e. about 150 LBLOCA calculation have been performed to confirm the CIAU uncertainty results).
- The reference best estimate PCT calculated by the applicant (result on the left of the Fig. 15) plus the calculated uncertainty is lower than the allowed licensing limit of 1473 K.
- The reference best estimate PCT calculated by CIAU (central result in the Fig. 7) is higher than the PCT ‘proposed’ by the applicant and the upper limit for the rod surface temperature even overpasses the allowed licensing limit of 1473 K thus triggering licensing issues.
It is shown that a lower best estimate PCT is calculated by CIAU (result on the right of the Fig. 7): however, user choices leading to such a result could not be justified, so far, based upon the available expertise including the supporting evidence from experimental data.

Figure 6: Result of CIAU application to Angra-2 LBLOCA analysis: uncertainty bands for rod surface temperature at ‘axial level 9’ of the hot rod realistic (reference run).

Figure 7: Angra-2 LBLOCA uncertainty evaluation: final result from the CIAU study and comparison with results of the applicant.
5.2. Kozloduy-3 200 mm Break to Show Similarity of Code Results

Results of independent safety evaluations [12] of the transient behaviour of the Kozloduy unit 3 VVER 440/230 NPP (675 MWth) following Large Break LOCA is discussed in the following. The considered LOCA is originated by a 200 mm single ended break in cold leg, and conservative boundary and initial conditions were assumed. A comprehensive analysis of the ‘LBLOCA 200 mm’ transient was carried out. The specific purposes of the analysis include:

- the assessment of the results obtained by Bulgarian applicant,
- the demonstration that the use of the Cathare code (not applied by EGP) provides quantitatively and qualitatively similar predictions as the Relap5,
- the execution of an independent safety analysis supported by uncertainty evaluation (use of a method available at UNIPI).

The following comments apply:

- The application of the uncertainty method to the results of the ‘LBLOCA 200 mm’ might be not justified owing to the use of some conservative input data. However, within the present context, the uncertainty evaluation of the Relap5 UNIPI analysis allows the quantitative evaluation of the Bulgarian applicant results and of the Cathare results predicted by DIMNP.
- Uncertainty results related to the rod surface temperature, obtained from the application of CIAU to the reference UNIPI-Relap5 calculation, are summarized in Fig. 8.
- The ‘PCT licensing’ predicted by CIAU (1062 °C) lies within the licensing acceptability threshold (1200 °C). The available safety margin is close to 150 K. The uncertainty results obtained by CIAU are supported by the outcome of the sensitivity study. The removal of the conservatism considered in the process, that could not be justified within the present context, is expected to bring the predicted ‘PCT licensing’ below 1000 °C.
- The demonstration that the results of predictions by RELAP5 and CATHARE are not in contradiction has been obtained through the uncertainty bands calculated by CIAU having as reference the RELAP5 calculation. Fig. 8 shows that the CATHARE results are embedded within the uncertainty bands of the RELAP5, when the same transient is calculated with the same boundary and initial conditions, thus allowing a successful solution to the assigned problem.

5.3. Best Estimate and Uncertainty Evaluation of LBLOCA 500 mm for Kozloduy-3

The analysis of the ‘LBLOCA 500 mm’ (DEGB in CL) transient [13] was carried out by adopting the Relap5 code. The specific purposes of the analysis include the assessment of the results and the execution of an independent safety analysis supported by uncertainty evaluation. A BE transient prediction of the ‘LBLOCA 500 mm’ was performed. Evaluation of the uncertainty was performed by CIAU for the RPV upper plenum pressure, the mass inventory in primary system and the hot rod cladding temperature. Only the last parameter is shown in Fig. 9 together with the uncertainty bands. The most relevant result is the demonstration that the PCT in the concerned hot rod is below the licensing limit.

In the same Fig. 9, bounding results from two conservative calculations (i.e. obtained by a BE code utilizing conservative input assumptions) are given: one is the conservative calculation (DC, ‘Driven’ conservatism in Fig 9), the other is the conservative calculation performed by University of Pisa (RC, ‘Rigorous’ conservatism). The following can be noted:
Figure 8: Uncertainty analysis of the ‘200 mm’ LOCA-DBA of VVER-440 NPP: main result from CIAU application.

6. CONCLUSIONS

The start of the development of uncertainty method in thermal-hydraulic system code calculations can be dated as in the early 80’s. Much before (even in the 60’s) similar activities were in progress in different technological areas like meteorology and neutron kinetics. A pioneering effort in the area of thermal-hydraulics was made by the US NRC with the publication of the CSAU method at the beginning of 90’s. However, background activities were carried out in the previous decade within the umbrella of OECD/CSNI.

Mature methods exist nowadays that are capable ‘of fixing the boundaries’ for the error of thermal-hydraulic system codes. Two main approaches, characterized as “propagation of code input uncertainty” and “propagation of code output errors”, have been discussed in the paper. These approaches are pursued by two reference methods ready for applications, i.e. the GRS method and the CIAU. The last method has been described with more detail and three main applications (the so-called key applications) of the CIAU methodology have been briefly introduced. A recent extension of the CIAU code to the uncertainty evaluation of the 3D Neutron-kinetics/Thermal-hydraulic coupled code calculation may be found in Ref. 10.
All the working methods to estimate the uncertainty derive from complex pictures of a complex reality that is constituted by the transient scenarios of water cooled NPP. Even though extensive documentation exists and (in most cases) is available, the level of common understanding about the capabilities and the drawbacks of the methods is not sufficient for achieving a full acceptability of the method. Therefore, rather than additional qualification of the methods, training and communication are needed for spreading the application of coupled best-estimate calculation and uncertainty evaluation.

REFERENCES


Addressing Boron Dilution Scenario Through RELAP5/3.3 Analysis of VVER-1000 SB LOCA

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ABSTRACT

Reactivity accident scenarios could originate from internal boron dilution in the primary system of a nuclear pressurized water reactor type (PWR or VVER). In essence the phenomena is caused by boron dilution following vaporization and condensation within the primary coolant system as a result a reduction in primary system mass inventory, e.g. during a loss of coolant accident (LOCA). When the liquid level in the reactor vessel falls below the hot leg elevation, steam begins to flow to the steam generators and condenses there. This steam carries no boron, thus boron concentration in the cold leg loop seals begins to decrease. If for some reason this water plug with low boron concentration enters the core without any major mixing with the borated coolant, the result is a positive reactivity insertion. The paper presents an analysis using the RELAP5 Mod 3.3 for a small break LOCA in one cold leg pipe of a Russian-type VVER-1000 nuclear reactor. The boundary conditions consider, as far as possible, a realistic situation in which the four accumulator tanks (SIT) and the three low pressure injection systems (LPIS) are available for core cooling but one of the three high pressure injection systems (HPIS), not serving the damaged loop, is not available. A range of break sizes (from 30 cm\(^2\) to 120 cm\(^2\)) have been investigated in order to identify a range of potentially adverse conditions. From the results obtained, in some calculations boron dilution is observed in some loop seals. The worst situation was found to occur at the 40cm\(^2\) break size. However, due to ECCS injection and mixing an increase of boron concentration is observed in the upper parts of the downcomer. Additionally RELAP codes have been used to perform analyses of the balance of boron mass within the reactor circuit.

1 INTRODUCTION

Reactivity accident scenarios could originate through internal boron dilution in the Primary System (PS) of a nuclear pressurized water reactor type (PWR or VVER) [1]. The potential for this scenario is more relevant for the reactor core at beginning of life (BOL),
when boron added to the coolant has its maximum effect and therefore any dilution is significant.

The problem is caused when the liquid level in the reactor vessel falls below the hot leg elevation, consequently steam begins to flow to the steam generators and condenses there. This steam carries no boron and thus its concentration in the cold leg loop seals begins to decrease. If for some reason this water plug with low boron concentration begins to flow towards the core and enters it without any major mixing with the borated coolant, the result is a positive reactivity insertion. Boron dilution is the process where a local decrease in boron concentration in the PS occurs with (nearly) constant average boron concentration; Deboration is the process where a net loss of boron from the PS occurs. During special small break LOCA scenarios dilution and de-boration processes occur simultaneously.

When boron dilution analysis are performed several other aspects could be distinguished like:
- formation of the diluted boron plug;
- transport of the diluted boron plug;
- mixing of the of the diluted boron plug;
- de-boration and boration (i.e. net boron gain in the primary circuit associated with ECCS actuation);
- reactivity feedback, necessarily associated with a three-dimensional performance of the neutron flux and coolant distribution in the vessel and in the core region.

Some of the aspects are mandatory associated with the use of specific type of tools, e.g. Computational Fluid Dynamics (CFD) codes to analyse the mixing of the diluted boron plug; 3D Neutron Kinetic codes for the reactivity feedback.

It should be pointed that the study here presented refers only to the formation and transport of the diluted boron plug and boron dilution, boration and deboration aspects, all associated with the use of system Thermalhydraulic tools. Other studies performed at the University of Pisa have analysed the problem of boron dilution from the mixing (CFD) [1] and neutron kinetics point of view [2], [3].

The main objective of the study was to investigate using the RELAP5 Mod 3.3 patch03 code the formation of significant (of a certain amount of volume) non-borated plugs of water in the loop seals of a VVER-1000 due to reflux-condensation mode after a Small Break (SB) LOCA. And from the break areas calculated for the SB LOCA to identify for which areas this situation is more evident. The range of break sizes selected for the analysis results from current international activities available in the field [2].

2 INPUT DECK AND BOUNDARY AND INITIAL CONDITIONS

The nodalization for RELAP5 Mod 3.3 patch03 [4] input-deck used in the analysis represents a generic VVER-1000 NPP of nominal power 3000 MWth (Figure 1).

The input-deck used has been widely used and validated at the University of Pisa [5]. In particular, the specific horizontal and slightly inclined shape of the steam generator U-tubes in VVER reactors requires particular attention in its modeling. This was extensively addressed by University of Pisa in some studies [6], in the PSB-VVER facility post-test activity (15 experiments) [7] and against PKL Integral Test Facility experiments [8].

The RELAP5 Mod 3.3 code has been qualified for boron mass transport against recent boron transport experiments in the PKL Integral Test Facility [9].
The analysis performed is a SB LOCA, a rupture in the cold leg of loop #2 between the pump and the reactor vessel. Several break locations were tested both in the hot leg and in the cold leg (before and after the loop seal and the pump). This revealed that the most adverse situation (for which loop seal boron dilution is more evident) is in the cold leg between the pump and the reactor vessel.

The boundary conditions consider, as far as possible, a realistic situation, taking into account Probabilistic Safety Assessment (PSA) ([10], [11]), in which the four accumulators (SIT) tanks and the three low pressure injection systems (LPIS) are available for core cooling but one of the three high pressure injection systems (HPIS), not serving the damaged loop, is not available. The Emergency Core Cooling Systems (ECCS) comprises four accumulators, two of them injecting into the upper plenum of the reactor vessel, with the other two injecting into the downcomer. Two HPIS inject into the cold legs of loop #2 (close to the break) and loop #4, whilst the third HPIS serving loop #3 is assumed to be unavailable, as previously mentioned. Of the three LPIS, one injects into the upper plenum, another into the downcomer and the third into cold leg and hot leg of loop #1. The injection of ECCS water directly into the vessel is one characteristic of the VVER-1000 NPPs, which differs from PWR. This feature results in differences in the boron dilution transients between both reactors.

3 PERFORMED ANALYSES

A range of break sizes (from 30 cm$^2$ to 120 cm$^2$) have been investigated in order to identify a possible range of potential adverse conditions from unanticipated boron dilution. A break area 40 cm$^2$ has been selected as the Reference Calculation, since this is the case in which boron dilution occurs more readily in the loop seals (see Chapter 4). Two sensitivity studies have been performed: one based on the Reference Calculation, increasing maximum
time step from 0.001 to 0.1. Another based on calculation of 60 cm$^2$ area break and using the option w11=1 in kinetic card 30000001, which indicates that the method of multiplying the boron worth times the mixture density will be used to obtain the boron density. In the other calculations boron worth was multiplied by liquid density to obtain boron density, in the light of boron acid being transported only in the water liquid phase. Table 1 summarizes all the calculations performed.

Table 1: Calculations performed

<table>
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<tr>
<th>No</th>
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<th>SB LOCA Break Area (cm$^2$)</th>
<th>Notes</th>
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<tr>
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<td>&quot;</td>
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Table 2 shows the correspondence between Case ID Label shown in the Figures and all calculations.

Table 2: Correspondence between calculations performed and Case ID Label

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<td>11</td>
<td>120CM2_345_NOHPIS3_BINT</td>
</tr>
</tbody>
</table>

4 MAIN RESULTS OF THE VVER-1000 SB LOCA CALCULATIONS

The sequence of relevant events resulting from the VVER-1000 NPP SB LOCA is given in Table 3 for the Reference Calculation (40 cm$^2$ break area). The break is imposed at time 0 s of the transient initiation after 100s of steady-state.

Figures 2 and 3 show PS pressure and normalized (to the steady-state value) PS coolant mass for all calculations performed. Initial depressurization is rather fast leading to emptying of the pressurizer and to saturated conditions in the hottest parts of the PS. The HPIS initiates
on PS pressure trip. In some of the calculations it could be observed how the increased rate of depressurization due to break conditions changes from single phase to two-phase flow. The accumulators (SIT) stop in all calculations when PS pressure stabilizes. PS coolant mass reflects the balance of mass lost from the break and various ECCS injections, i.e. HPIS, SIT and LPIS in that order. The loss of mass is larger at the beginning of the transient for calculations with larger area breaks, but in these cases LPIS injection occurs before, thus restoring borated water to the system.

Table 3: Resulting sequence of events

<table>
<thead>
<tr>
<th>EVENTS</th>
<th>CALCULATED TIME AFTER TRANSIENT INITIATION (s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Break opening</td>
<td>0.0</td>
</tr>
<tr>
<td>SCRAM power curve enabled</td>
<td>8.9</td>
</tr>
<tr>
<td>Start of main coolant pumps coast down</td>
<td>9.9</td>
</tr>
<tr>
<td>Main steam line valve closure</td>
<td>11.0</td>
</tr>
<tr>
<td>Feedwater valve closure</td>
<td>40.0</td>
</tr>
<tr>
<td>HPIS starts (at PS pressure &lt; 11MPa)</td>
<td>140.0</td>
</tr>
<tr>
<td>Pressurizer emptied</td>
<td>228.0</td>
</tr>
<tr>
<td>Upper plenum in saturation condition</td>
<td>412.0</td>
</tr>
<tr>
<td>Accumulators injection starts (at PS pressure &lt; 6MPa)</td>
<td>2711.0</td>
</tr>
<tr>
<td>Minimum boron in loop seals</td>
<td></td>
</tr>
<tr>
<td>Broken loop #2</td>
<td>3159.0</td>
</tr>
<tr>
<td>Intact loop #1</td>
<td>3152.0</td>
</tr>
<tr>
<td>Intact loop #3</td>
<td>3228.0</td>
</tr>
<tr>
<td>Intact loop #4</td>
<td>2988.0</td>
</tr>
<tr>
<td>Loop seal clearing (first occurring)</td>
<td></td>
</tr>
<tr>
<td>Broken loop #2</td>
<td>3164</td>
</tr>
<tr>
<td>Intact loop #1</td>
<td>4700</td>
</tr>
<tr>
<td>Intact loop #3</td>
<td>3160</td>
</tr>
<tr>
<td>Intact loop #4</td>
<td>3011</td>
</tr>
<tr>
<td>Occurrence of minimum PS coolant mass</td>
<td>3146.0</td>
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<td>Primary-Secondary pressure reversal</td>
<td>4322.0</td>
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<td>LPIS starts (at PS pressure &lt; 2.6MPa)</td>
<td>4547.0</td>
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<td>Accumulators injection stop</td>
<td>5158.0</td>
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<tr>
<td>Break two phase flow</td>
<td>Not occurring</td>
</tr>
<tr>
<td>End of calculation</td>
<td>9400.0</td>
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</table>
In order to observe local boron decrease (boron dilution) in the liquid water, boron mass and water mass were calculated along the loop seal pipe for the four loops seals. The correspondence of control variables that take into account the boron mass present in the loops and the loop number in the Figures is the following:

- loop #1 (cntrlvar26)
- loop #2 (cntrlvar25)
- loop #3 (cntrlvar24)
- loop #4 (cntrlvar23)

The correspondence of control variables that take into account the water mass present in the loops and the loop number the Figures is the following:

- loop #1 (cntrlvar30)
- loop #2 (cntrlvar29)
- loop #3 (cntrlvar28)
- loop #4(cntrlvar27)

Boron dilution in the loop seals is not observed for very small area breaks, lower than $35 \text{ cm}^2$ (Figure 4). It is observed in the case of $40 \text{ cm}^2$ area break (Figure 5). In this case (Figure 6) three of the four loop seals (loop #1-cntrlvar30, loop #2-cntrlvar29 and loop #4-cntrlvar27) clearly have a certain mass of water (about $5000 \text{ kg}$) after $3200\text{s}$. Boron mass is low in these loops (Figure 5) from $2500\text{s}$ to $4200\text{s}$, and especially in loop #4 (cntrlvar23), where around $4000\text{s}$ the boron mass is almost zero. In loop #3 (cntrlvar24) (Figure 5) there is a low mass of boron, in addition water mass is low in this loop #3 (cntrlvar28) (Figure 6), which indicates that there is vapour in the loop (which carries no boron), thus no presence of liquid water free of boron, or boron dilution occurrence.

In Figure 5, the big increase of boron mass in loop #2 (cntrlvar25) after $4000\text{s}$ is due to the slight decrease in loop pressure that provokes an increase of massflow of borated water coming from the HPIS in this loop at that time. And even that part of this flow is going out through the small break. It can be also confirmed by the water mass increase in loop #2 (cntrlvar29) (Figure 6). In loop #3 the third HPIS was assumed to be unavailable, so no increase in water mass or boron is observed in the loop till LPIS starts.

![Figure 4: Boron mass in loop seals in case 35 cm² break area](image-url)
Boron concentration in the downcomer upper parts, for the 40cm break area case, is shown in Figure 7 (four vertical nodes, in the radial part entering from loop #4). It seems that water free of boron, seen in previous figures in loop #4 at around 4000s, it mixes going downwards in these upper parts of the downcomer with borated water coming from ECCS and thus an increase in boron concentration from top (volume 136-01) to bottom (volume 136-04) is observed.
Figure 7: Boron concentration in downcomer upper part (coming from loop #4) from top to bottom in case 40 cm² break area

As the area break is increasing no boron dilution occurs in the loop seals. Boron mass distribution in loop seals is shown in Figure 8 in the case 60 cm² break area. Low mass of boron is observed in loop #1 (cntrlvar26) for a short period and in loop #2 (cntrlvar25) between 1000s and 5000s, but also water mass is low in these loops for the same period (Figure 9). There is the presence of vapour which carries no boron, and there is no boron dilution occurrence.

Figure 8: Boron mass in loop seals in case 60 cm² break area
Figure 9: Water mass in loop seals in case 60 cm$^2$ break area

Boron concentration in lower plenum top is shown in Figure 10 for all calculations. It can be observed that boron concentration is always at values equal or above the initial concentration value.

Figure 10: Boron concentration in lower plenum top in all cases

A summary of all the calculations is shown in Figure 11. Values for minimum boron concentration are plotted against break flow area in four parts in the vessel namely the central downcomer upper part; lower plenum bottom and lower plenum top and core inlet. All parts are full of liquid water.

Roughly, boron concentration in core inlet is the same for all area breaks and with high values close to the steady-state one, there is no indication of boron dilution in this part of the vessel. The minimum boron concentration in the other parts represented in Figure 11 follows...
the trend already observed i.e. water free of boron mixes with borated water going downwards in the downcomer and enters the core inlet and thus an increase in boron concentration occurs. In the upper part of the downcomer, as already observed for the loop seals, the break areas leading to low boron concentration are around 40 cm$^2$.

![MINIMUM BORON CONCENTRATION IN DOWNCOMER UPPER PART, LOWER PLENUM AND CORE INLET](image)

To put into perspective the issue of an unwanted plug of liquid water free of boron it is useful to note the following volumes of the different components considered in this study.

- the volume of one loop seal is about 11 m$^3$
- the volume of the downcomer is about 24 m$^3$
- the volume of the lower plenum is about 8 m$^3$
- the volume of the active core is about 13.5 m$^3$

Thus the situation in which one or more loops seals contain a significant amount of diluted water can lead to a potentially hazardous situation, particularly, if these plugs are transported to the core without mixing with other borated water.

The first sensitivity analysis was performed based on the Reference Calculation (40 cm$^2$ break area, case 3 in Table 1) but with higher maximum time step in the calculation (case 4 in Table 1). Compared to the Reference Calculation, the same results were obtained for PS pressure (Figure 12), and almost the same results were obtained in ECCS injection flows (PS mass), but there were differences in water (Figure 13) and boron mass distribution (Figure 14) in the four loop seals.

Boron distribution in the loop seals (Figure 15) for the other sensitivity analysis, based on the 60 cm$^2$ break area case but with $w_{11}=1$ in kinetic card 30000001 (case 8 in Table 1) is slightly different compared to the 60 cm$^2$ break area case (case 7 in Table 1, Figure 8), due to different water distribution in the loop seals. PS pressure and PS mass gave nearly the same results for these two cases (Figures 2 and 3).
Figure 12: Primary System Pressure in Reference Calculation and Reference Calculation with increased max time step

Figure 13: Water mass in loop #3 for Reference Calculation and Reference Calculation with increased max time step
Figure 14: Boron mass in all loop seals for Reference Calculation and Reference Calculation with increased max time step

Figure 15: Boron mass in loop seals for case 60 cm$^2$ break area and w11=1 in kinetic card 30000001
5 BORON MASS CONSERVATION IN A SIMPLIFIED SYSTEM

Other analyses were carried out in order to understand the basic treatment of boric acid mass transport by RELAP5 thermalhydraulic codes [12]. The codes used in the exercise were: RELAP5 Mod 3.3 patch03, RELAP5/SCDAPSIM Mod3.4 (bi7) [13] and RELAP5/3D© 2.4.2 [14]. The simplified nodalization used is shown in Figure 16 (reactor, pressurizer, primary system and containment). Water flow is simulated to inject from a tmdpvol into the lower plenum and vapour exits from the upper head (vol. 130).

Figure 16: Reactor, pressurizer, primary system and containment simplified nodalization for boron mass calculations

Two types of calculations were performed with RELAP5 Mod 3.3 patch03 code:
1. Increasing and decreasing boron concentration in single liquid phase conditions (i.e. water flow injected with more or less boron concentration) after 10s of transient initiation (100s of steady-state).
2. A large break (LB) LOCA in the upper plenum is imposed after 10s of transient initiation (100s of steady-state). Four sensitivities on this calculation were performed: increasing time step to 0.1 instead 0.001, using the option w11=1 in kinetic card 30000001 (as explained in Chapter 3), using RELAP5/SCDAPSIM/Mod3.4(bi7) code and using RELAP5/3D© 2.4.2 code.

A total of seven calculations were performed. They are summarized in Table 4. Table 5 shows the correspondence between Case ID Label shown in the Figures and all calculations.
### Table 4: Calculations performed

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<td>RELAP5 Mod 3.3 patch03</td>
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<td>RELAP5 Mod 3.3 patch03</td>
<td>Base Case Calculation</td>
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<td>&quot;</td>
<td>&quot;</td>
<td>Base Case Calculation above and increased max time step</td>
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<td>5</td>
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<td>Base Case Calculation and w11=1 in kinetic card 30000001</td>
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<td>6</td>
<td>&quot;</td>
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<td>7</td>
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### Table 5: Correspondence between calculations performed and Case ID Label

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Total Boron mass was calculated in the reactor, pressurizer, and containment volumes (cntrlvar 401).

Total Boron mass was increasing or decreasing in calculations in single liquid phase conditions (Figure 17). In all LB LOCA calculations the reactor core is “voided” (Figure 18) and total boron mass is not conserved. It was observed a decrease of about 10 kg (of an initial 50 kg) in the calculations run with RELAP5/3.3 patch03 and RELAP5/3D© 2.4.2. It was observed almost no influence in the results changing time step (Figure 19). In the calculation run with RELAP5/SCDAP Mod3.4 (bi7) a decrease of about 40 kg (of an initial 50 kg) is observed (Figure 19). This could be associated to a total system mass decrease also bigger in case with RELAP5/SCDAP Mod3.4(bi7) code compared to the total system mass decrease obtained with the other codes (Figure 19).
Figure 17: Total boron mass calculated in simplified system when increasing or decreasing inlet boron concentration in single phase conditions

Figure 18: Void fraction in core region in simplified system in LB LOCA Base Case

Figure 19: Total boron mass calculated in simplified system in all LOCA calculations
6 CONCLUSIONS

The paper presents a study to investigate using RELAP5 Mod 3.3 patch03 the formation of significant non-borated plugs of water in the loop seals of a VVER-1000 due to reflux-condensation after an SB LOCA. From the results obtained, in some calculations boron dilution is observed in one or more loop seals. The most adverse condition was found to be that for 40cm$^2$ break size calculation. However due to ECCS injection and mixing an increase of boron concentration is observed in the upper parts of the downcomer.

It should be pointed out that for RELAP5, full boron mixing is assumed in the single nodes, and nodalizations are also coarser than in other approaches, such as using Computational Fluid Dynamics (CFD) codes. These assumptions could lead, in the case of system codes, to non realistic predictions of full mixed zones of borated and non-borated water, which could be non-conservative. Thus CFD analysis starting from the end of the cold legs at the inlet of the vessel, downcomer and lower plenum is another approach that could be adopted in analyzing this problem in certain type of analysis, unless with system codes a conservative situation is found.

Simulations with RELAP5 codes for a simplified system showed that boron mass is not conserved in two-phase situations.

ACKNOWLEDGMENTS

Acknowledge is given to the EC (Sixth Framework Programme (Euratom), Intra-European Fellowship) for the financial support to the study here presented.

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THE “CODE USER EFFECT” AND THE INTERNATIONAL TRAINING SEMINAR 3D S.UN.COP

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ABSTRACT

Thermal-hydraulic system computer codes are extensively used worldwide for analysis of nuclear facilities by utilities, regulatory bodies, nuclear power plant designers and vendors, nuclear fuel companies, research organizations, consulting companies, and technical support organizations. The computer code user represents a source of uncertainty that can influence the results of system code calculations. This influence is commonly known as the ‘user effect’ and stems from the limitations embedded in the codes as well as from the limited capability of the analysts to use the codes. Code user training and qualification is an effective means for reducing the variation of results caused by the application of the codes by different users. This paper describes a systematic approach to training code users who, upon completion of the training, should be able to perform calculations making the best possible use of the capabilities of best estimate codes. In other words, the program aims at contributing towards solving the problem of user effect. The 3D S.UN.COP (Scaling, Uncertainty and 3D COuPled code calculations) seminars have been organized as follow-up of the proposal to IAEA for the Permanent Training Course for System Code Users [1]. Eight seminars have been held at University of Pisa (two in 2004), at The Pennsylvania State University (2004), at the University of Zagreb (2005), at the School of Industrial Engineering of Barcelona (January-February 2006), in Buenos Aires, Argentina (October 2006), requested by Autoridad Regulatoria Nuclear (ARN), Nucleoelectrica Argentina S.A (NA-SA) and Comisión Nacional de Energía Atómica (CNEA), at the College Station, Texas A&M, (January-February 2007), in Hamilton and Niagara Falls, Ontario (October...
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2007) requested by Atomic Energy Canada Limited (AECL), Canadian Nuclear Society (CNS) and Canadian Nuclear Safety Commission (CNSC). It was recognized that such courses represented both a source of continuing education for current code users and a mean for current code users to enter the formal training structure of a proposed ‘permanent’ stepwise approach to user training. The 3D S.UN.COP - CANDU 2007 in Canada was successfully held with the attendance of 33 participants coming from 8 countries and 16 different institutions (universities, vendors and national laboratories). More than 30 scientists (coming from 13 countries and 23 different institutions) were involved in the organization of the seminar, presenting theoretical aspects of the proposed methodologies and holding the training and the final examination. A certificate (LA Code User grade) was released to participants that successfully solved the assigned problems. A ninth seminar is currently holding (October 2008) at the Institute for Energy, Joint Research Centre of the European Commission in Petten (The Netherlands), involving more than 30 scientists between lecturers and code developers (http://dimnp.ing.unipi.it/3dsuncop/2008/index.html)

1 INTRODUCTION

The best estimate thermal-hydraulic codes used in the area of nuclear reactor safety have reached a marked level of sophistication. Their capabilities to predict accidents and transients at existing plants have substantially improved over the past years as a result of large research efforts and can be considered satisfactory for practical needs provided that they are used by competent analysts.

Some recognized inadequacies in code calculation results are due to the limitations embedded in the codes. These range from some model deficiencies to approximation in the numeric solution. The transformation of the actual reference system geometry into an approximate noding scheme constitutes an additional limitation. Nodalization imperfections, insufficient knowledge of initial and boundary conditions, and ‘user effects’ add to the limitations of the code prediction. User effects [2] lie at the origin of most of the inaccuracies for the following reasons:

• Fully detailed, comprehensive code user guidelines do not exist.
• The actual (three dimensional) plant is modeled with several one dimensional approximations.
• Engineering knowledge has to be applied in the preparation of the input deck in order to deal with some of the code limitations.
• Certain problems are inherent in the approaches used in the modeling process such as: use of local pressure drop coefficients, critical flowrate multipliers, application to transient conditions of models qualified for steady state, application of the fully developed flow concept for different nuclear reactor conditions, etc.
• The fact that an increasing number of users without adequate qualification have access to the system codes and nodalizations may produce diverging results and lead to the diffusion of erroneous evaluations.
• Experimental data, including the values of initial and boundary conditions that are used as a basis for comparisons are, in the large majority of cases, supplied without error bands.
• Clear criteria for the acceptability of the results have not been agreed upon among experts in the area.

A wide range of activities have recently been completed in the area of system thermal-hydraulics as a follow-up to considerable research efforts. Problems have been addressed, solutions to which have been at least partly agreed upon on international ground. These include: the need for best-estimate system codes [3] and [4], the general code qualification process [5] and [6], the proposal for nodalization qualification and attempts aiming at qualitative and quantitative accuracy evaluations [7]. Complex uncertainty methods have been proposed, following a pioneering study at USNRC [8]. This study attempted, among other
things, to account for user effects on code results. An international study aiming at the comparison of assumptions and results of code uncertainty methodologies has been completed [9]. More recently, the IAEA developed a Safety Report on Accident Analysis of Nuclear Power Plants containing a set of practical suggestions based on best practice worldwide [10].

2 IAEA SAFETY REPORT ON ACCIDENT ANALYSIS

During the period 1997-1999, the IAEA developed a document consistent with its revised Nuclear Safety Standards Series [10] that provides guidance on accident analysis of nuclear power plants (NPPs). The report includes a number of practical suggestions on the manner in which to perform accident analysis of NPPs. These cover the selection of initiating events, acceptance criteria, computer codes, modeling assumptions, the preparation of input, qualification of users, presentation of results, and quality assurance (QA). The suggestions are both conceptual as well as formal and are based on present practice worldwide for performing accident analysis. The report covers all major steps in performing analyses and is intended primarily for code users.

Within the framework of the IAEA guidance the important role of the user effects on the analysis is addressed. The need for user qualification and training is clearly recognized. The systematic training of analysts is emphasized as being crucial for the quality of the analysis results. Three areas of training, in particular, are specified:

- practical training on the design and operation of the plant;
- software specific training; and
- application specific training.

Training on the phenomena and methodologies is typically provided at the university level, but cannot always be considered as sufficient. Furthermore, training on the specific application of system codes is not usually provided at this level. Practical training on the design and operation of the plant is, however, essential for the development of the plant models. Software specific training is important for the effective use of the individual code. Application specific training requires the involvement of a strong support group that shares its experience with the trainees and provides careful supervision and review.

Training at all three levels ending with examination is encouraged for a better effectiveness of the training. Such a procedure is considered as a step in the direction of establishing a standard approach that could be applicable on an international basis.

A significant number of the suggestions made by the IAEA relate to the preparation of input decks and to the collection of the relevant plant data as well as to the presentation and evaluation of the results and to QA. In addition, the report specifies a procedure for performing accident analysis that covers all important steps needed for this task.

3 CODE USERS

Best estimate codes are used by designer/vendors of NPPs, by utilities, licensing authorities, research organizations including universities, nuclear fuel companies, and by technical support organizations. The objectives of using the codes may be quite different, ranging from design or safety assessment to simply understanding the transient behavior of a simple system. In view of the current computing capabilities, a system code (e.g. RELAP, TRAC, CATHARE, or ATHLET) can be put into operation in a few days. In the same time span, results can be obtained for a complex system provided that there is a nodalization available. An unqualified input deck related to a complex system such as an NPP can be set up in time periods of a few weeks using the available code manuals. However, these periods can be shortened if the analyst is a ‘qualified’ code user. Qualified code user groups already
exist; scientists who have been working with system codes for more than thirty years belong to such groups.

The most sensitive use of the code deals with situations in which the results obtained have an effect on the design or safety assessment of the NPP. In this context, the code validation process, nodalization qualification, qualitative or quantitative accuracy evaluation, and the use of the code by a qualified code user have been recognized as necessary steps to reduce the possibility of producing poor code predictions [11].

4 PERMANENT USER TRAINING COURSE FOR SYSTEM CODE: THE PROPOSAL

As a follow-up to the Specialists Meeting held at the IAEA in September 1998, the Universities of Pisa and Zagreb and the Jožef Stefan Institute, Ljubljana, jointly presented a Proposal to IAEA for the Permanent Training Course for System Code Users [1]. It was recognized that such a course would represent both a source of continuing education for current code users and a means for current code users to enter the formal training structure of a proposed ‘permanent’ stepwise approach to user training.

Before finalizing the main outcomes in relation to the proposed user training, the following can be emphasized:
- the user gives a contribution to the overall uncertainty that unavoidably characterizes system code calculation results;
- in the majority of cases, it is impossible to distinguish among uncertainty sources like 'user effect', 'nodalization inadequacy', 'physical model deficiencies', 'uncertainty in boundary or initial conditions', 'computer/compiler effect';
- 'reducing the user effect' or 'finding the optimum nodalization' should not be regarded as a process that removes the need to assess the uncertainty;
- in general, it is misleading to prepare guidelines that focus codes predictions into a narrow part of the uncertainty.

As a follow up of the massive work conducted in different Organizations, the need was felt to fix criteria for training the code user. As a first step, the kind of code user and the level of responsibility of a calculation result should be discussed.

4.1 Levels of User Qualification

Two main levels for code user qualification are distinguished in the following:
- Code user, level "A" (LA);
- Responsible of the calculation results, level "B" (LB).

A Senior grade level should be considered for the LB code user (LBS). Requisites are detailed hereafter for the LA grade only; these must be intended as a necessary step (in the future) to achieve the LB and the LBS grades. The main difference between LA and LB lies in the documented experience with the use of a system code; for the LB and the LBS grades, this can be fixed in 5 and 10 years, respectively, after achieving the LA grade. In such a context, any calculation having an impact in the sense previously defined must be approved by a LB (or LBS) code user and performed by a different LA or LB (or LBS) code user.
4.2 Requisites for Code User Qualification

4.2.1 - LA Code User Grade

The identification of the requisites for a qualified code user derives from the areas and the steps concerned with a qualified system code calculation: a system code is one of the (four) codes previously defined and a qualified calculation in principle includes the uncertainty analysis. The starting condition for LA code user is a scientist with generic knowledge of nuclear power plants and reactor thermalhydraulics (e.g. in possession of the master degree in US, of the 'Laurea' in Italy, etc.).

Areas for code user qualification: The requisites for the LA grade code user are in the following areas:
A) Generic code development and assessment processes;
B) Specific code structure;
C) Code use -Fundamental Problems (FP);
D) Code use -Basic Experiments (BETF);
E) Code use -Separate Effect Test Facilities (SETF);
F) Code use -Integral Test Facilities (ITF);
G) Code use -Nuclear Power Plant transient Data
H) Uncertainty Methods including concepts like nodalization, accuracy quantification, user effects.

Area A)  


Area B)  
Sub-area B1): Structure of the system code selected by the LA code user: thermalhydraulics, neutronics, control system, special components, material properties, numerical solution.

Sub-area B2): Structure of the input deck; examples of user choices.

Area C)  
Sub-area C1): Definition of Fundamental Problem (FP): simple problems for which analytical solution may be available or less. Examples of code results from applications to FP; different areas of the code must be concerned (e.g. neutronics, thermalhydraulics, and numerics).

Sub-area C2): The LA code user must deeply analyze\footnote{1}{- to develop a nodalization starting from a supplied data base or problem specifications;
- to run a reference test case;
- to compare the results of the reference test case with data (experimental data, results of other codes, analytical solution), if available;
- to run sensitivity calculations;
- to produce a comprehensive calculation report (having an assigned format).} at least three specified FPs, searching for and characterizing the effects of nodalization details, time step selection and other code-specific features.

Area D)  
Sub-area D1): Definition of Basic test facilities and related experiments (BETF): researches aiming at the characterization of an individual phenomenon or of an individual

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quantity appearing in the code implemented equations, not necessarily connected with the NPP. Examples of code results from applications to BETF.

Sub-area D2): The LA code user must deeply analyze at least two selected BETF, searching for and characterizing the effects of nodalization details, time step selection, error in boundary and initial conditions, and other code-specific features.

Area E)

Sub-area E1): Definition of Separate Effect Test Facility (SETF): test facility where a component (or an ensemble of components) or a phenomenon (or an ensemble of phenomena) of the reference NPP is simulated. Details about scaling laws and design criteria. Examples of code results from applications to SETF.

Sub-area E2): The LA code user must deeply analyze at least one specified SETF experiment, searching for and characterizing the effects of nodalization details, time step selection, errors in boundary and initial conditions and other code-specific features.

Area F)

Sub-area F1): Definition of Integral Test Facility (ITF): test facility where the transient behavior of the entire NPP is addressed. Details about scaling laws and design criteria. Details about existing (or dismantled) ITF and related experimental programs. ISPs activity. Examples of code results from applications to ITF.

Sub-area F2): The LA code user must deeply analyze at least two specified ITF experiments, searching for and characterizing the effects of nodalization details, time step selection, errors in boundary and initial conditions and other code-specific features.

Area G)

Sub-area G1): Description of the concerned NPP and of the relevant (to the concerned NPPD and calculation) BoP and ECC systems. Examples of code results from applications to NPPD.

Sub-area G2): The LA code user must deeply analyze at least two specified NPP transients, searching for and characterizing the effects of nodalization details, time step selection, errors in boundary and initial conditions and other code-specific features.

Area H)

Description of the available uncertainty methodologies. The LA code user must be aware of the state of the art in this field.

4.2.2 - LB Code User Grade

A qualified user at the LB grade must be in possession of the same expertise as the LA grade and:
I) he must have a documented experience in the use of system codes of at least 5 additional years;
J) he must know the fundamentals of Reactor Safety and Operation- and Design having generic expertise in the area of application of the concerned calculation;
K) he must be aware of the use and of the consequences of the calculation results; this may imply the knowledge of the licensing process.

4.2.3 - LBS Code User Grade

A qualified user at the LBS grade must be in possession of the same expertise as the LB grade and:
L) he must have an additional documented experience in the use of system codes of at least 5 additional years.
4.3 Modalities for the achievements of the LA, LB and LBS Code User grades

**LA grade:** Two years training and "Home Work" with modalities defined in Table 1, are necessary to achieve the LA grade, following an examination.

**LB grade:** The steps and the time schedule needed to achieve the LB code user grade are summarized in Tab. 1. An examination is needed (5 years after the LA grade).

**LBS grade:** The steps and the time schedule needed to achieve the LBS code user grade are summarized in Tab. 1. The LBS code use grade can be obtained (5 years after achieving the LB grade) following the demonstration of performed activity in the 5 years period.

4.4 Course Conduct

The training of the code user requires the conduct of lectures, practical on-site exercises, homework, and examination while, for the senior code user, only a review of documented experience and on-site examination is foreseen.

The code user training, including practical exercises, which represent an essential part of the course, lasts two years and covers the following areas:

A) Generic code development and assessment processes:
   - general structure of a system code;
   - conservation (or balance) equations in thermal-hydraulics;
   - conduction and radiation heat transfer;
   - neutron transport theory and neutron kinetics approximation;
   - constitutive (closure) equations including convection heat transfer;
   - special components (e.g. pump, separator);
   - material properties;
   - constitutive (closure) equations including convection heat transfer;
   - special components (e.g. pump, separator);
   - material properties;
   - simulation of NPP and balance of plant (BoP) related control systems;
   - numerical methods;
   - developmental assessment;
   - independent assessment including the separate effect test code validation matrix [5], and integral test code validation matrix [6]; and
   - examples of specific code validation matrices.

B) Specific code structure:
   - structure of a system code selected by the code user: thermal-hydraulics, neutronics, control system, special components, material properties, and numerical solution; and
   - structure of the input deck, examples of user options.

C) Fundamental problems or simple problems for which analytical solution may be available:
   - definition of fundamental problems; and
   - examples of code results from applications involving different areas of the code concerned (e.g. neutronics, thermal-hydraulics, numerics).

D) Basic test facilities and related experiments for the characterization of an individual phenomenon or of an individual quantity appearing in the code equations.

E) SETFs where a component (or an ensemble of components) or a phenomenon (or an ensemble of phenomena) of the reference NPP is simulated:
   - details of scaling laws and design criteria; and
   - examples of code results from applications.
F) ITFs where the transient behavior of the entire NPP is addressed:
   • details of scaling laws and design criteria; and
   • details of ITFs and related experimental programs;
   • International Standard Problem activity; and
   • an example of code results from applications to ITFs.

G) Applications to nuclear power plants:
   • description of the NPP concerned and of the relevant BoP system and emergency core
     cooling system;
   • an example of code results from applications to an NPP;
   • practical exercises in the use of the code for NPP accident analysis highlighting the
     detection of errors in boundary and initial conditions and other code specific features;
   • use of NPP simulators/analyzers.

H) Uncertainty methods including accuracy quantification:
   • description of the available uncertainty methodologies; and
   • state of the art and future prospects in this field.

In addition to the aforementioned areas, senior code user training also covers:

I) The use of accident analysis in reactor design and safety assessment.

J) Effects of analysis results on the licensing process.

4.5 Training Exercises

Practical exercises foreseen during the training include development of the nodalization
from the pre-prepared database with problem specifications. To this end, didactic material and
presentations/lectures on the exercise will be provided with a detailed explanation of the
objectives of the work that the trainee must perform. Extensive application of the code by the
trainee at his own institution following detailed recommendations and under the supervision
of the course lecturers is foreseen as 'homework'. The use of the code at the course venue is
foreseen for the following applications:
   • fundamental problems including nodalization development;
   • basic test facilities and related experiments including nodalization development;
   • SETFs and related experiments including nodalization development;
   • ITF experiments with nodalization modifications; and
   • NPP transients including nodalization modifications.

For each of the above cases, the trainee will be required to:
1. develop (or modify) a nodalization starting from the database or problem specifications
   provided;
2. run the reference test case;
3. compare the results of the reference test case with data (experimental data, results of other
   codes, analytical solution);
4. run sensitivity calculations;
5. produce a comprehensive calculation report following a prescribed format whereby the
   report should include, for example:
   – the description of a particular facility;
   – the description of an experiment (including relevance to scaling and relevance to
     safety);
   – modalities for developing (or modifying) the nodalization;
   – the description and use of nodalization qualification criteria for steady state and
     transient calculations;
   – qualitative and quantitative accuracy evaluation;
– use of thresholds for the acceptability of results for the reference case;
– planning and analysis of the sensitivity runs; and
– an overall evaluation of the activity (code capabilities, nodalization adequacy, scaling, impact of the results on the safety and the design of NPP, etc.).

4.6 Examination

On-site examination at different stages during the course is considered a condition for the successful completion of the code user training. The homework that the candidate must complete before attempting the on-site examination includes:
A) Studying the material/documents supplied by the course organizers.
B) Solving the problems assigned by the course organizers. This also involves the preparation of suitable reports that must be approved by the course organizers.

The on-site tests consist of four main steps that include the evaluation of the reports prepared by the candidate, answering questions on the reports and course subjects and demonstrating the capability to work with the selected code. Each step must be accomplished before proceeding to the subsequent one.

The completion of all the steps of the examination requires that the candidate spend one full week at the course venue.

<table>
<thead>
<tr>
<th>Code User Grade</th>
<th>WEEKS</th>
<th>LECTURES</th>
<th>SPECIF FOR HOMEWORK</th>
<th>HOMEWORK</th>
<th>ON-SITE TEST</th>
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<tr>
<td>LA</td>
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<td></td>
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<tr>
<td>1-2</td>
<td></td>
<td>A1, A2^, B1, B2^, C1, D1</td>
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<td>3</td>
<td></td>
<td>C2, D2</td>
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<td>4-25</td>
<td></td>
<td></td>
<td>A, B, C2*, D2*</td>
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<td></td>
<td></td>
<td>A1, B1, C, D, C2°, D2°</td>
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<tr>
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<td></td>
<td>A2, E1</td>
<td>E2</td>
<td>E2*</td>
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<td>F2</td>
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<td>G2*</td>
<td>G2*</td>
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<td>103</td>
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<td></td>
<td>G, H, G2°</td>
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<tr>
<td>LB (5 yrs after LA)</td>
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<td>I*, J, K, K°</td>
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<tr>
<td>LBS (5 yrs after LB)</td>
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<td></td>
<td></td>
<td>L*</td>
<td></td>
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</tbody>
</table>

^ Fundamental, * Report necessary, ° Solution of submitted problems and discussion
5 3D S.UN.COP SEMINARS: FOLLOW-UP OF THE PROPOSAL

5.1 Background Information about 3D SUNCOP Trainings

The 3D S.UN.COP (Scaling, Uncertainty and 3D COuPled code calculations) training aims to transfer competence, knowledge and experience from recognized international experts in the area of scaling, uncertainty and 3D coupled code calculations in nuclear reactor safety technology to analysts with a suitable background in nuclear technology.

The training is open to research organizations, companies, vendors, industry, academic institutions, regulatory authorities, national laboratories, etc. The seminar is in general subdivided into three parts and participants may choose to attend a one-, two- or three-week course. The first week is dedicated to the background information including the theoretical bases for the proposed methodologies; the second week is devoted to the practical application of the methodologies and to the hands-on training on numerical codes; the third week is dedicated to the user qualification problem through the hands-on training for advanced user and include a final exam. From the point of view of the conduct of the training, the weeks are characterized by lectures, code-expert teaching and by hands-on-application. More than thirty scientists (including the organizers and the external lecturers) are in general involved in the organization of the seminars, presenting theoretical aspects of the proposed methodologies and holding the training and the final examination. A certificate of qualified code user is released to participants that successfully solve the assigned problems during the exams.

The framework in which the 3D S.UN.COP seminars have been designed may be derived from Figure 1, where the roles of two main international institutions (OECD and IAEA) and of the US NRC (and the regulatory bodies of other countries) in order to address the problem of user effect are outlined together with the proposed programs and produced documents. Figure 2 depicts how the 3D S.UN.COP ensures the nuclear technology maintenance and advancements through the qualification of personnel in regulatory bodies, research activities and industries by mean of teaching of very well known scientists belonging to the same type of institutions.

At present, three institutions are planning and managing the 3D SUNCOP: 1- the Department of Mechanical, Nuclear and Production Engineering (DIMNP) of the University of Pisa, Italy (UNIPI, the group was the pioneer in the organization of the initial 3D SUNCOP trainings), 2-the group of Dynamic Analysis of Energy Systems of the Department of Physics and Nuclear Engineering, Technical University of Catalonia (UPC), at the premises of the School of Industrial Engineering of Barcelona, Spain (ETSEIB), and 3- the Department of Power Systems (ZVNE) of the Faculty of Electrical Engineering and Computing of Zagreb, Croatia (FER), University of Zagreb (UNIZG).

Eight Training Courses have been organized up to now and were successfully held at:

- The University of Pisa (Pisa, Italy), 5 – 9 January 2004 (6 participants).
- The Pennsylvania State University (University Park, PA, USA), 24 – 28 May 2004 (15 participants).
- The University of Pisa (Pisa, Italy), 14 – 18 June 2004 (11 participants).
- The University of Zagreb (Zagreb, Croatia), 20 June – 8 July 2005 (19 participants).
- The Technical University of Catalonia (Barcelona, Spain), 23 January – 10 February 2006 (33 participants).
- The Autoridad Regulatoria Nuclear (ARN), the Comisión Nacional de Energía Atómica (CNEA), the Nucleoelectrica Argentina S.A (NA-SA) and the Universidad Argentina De la Empresa (Buenos Aires, Argentina), 2 October – 14 October 2006 (37 participants).
- College Station, Texas A&M, (USA), 22 January – 9 February 2007 (26 participants).
- Hamilton & Niagara Falls (Ontario, Canada), 8 – 26 October 2007 (33 participants).
5.2 Objectives and Features of the 3D S.UN.COP Seminar Trainings

The main objective of the seminar activity was the training in safety analysis of analysts with a suitable background in nuclear technology. The training was devoted to the promotion and use of international guidance and to homogenize the approach to the use of computer codes for accident analysis. Between the main objectives are:

- To transfer knowledge and expertise in Uncertainty Methodologies, Thermal-Hydraulics System Code and 3D Coupled Code Applications;
- To diffuse the use of international guidance;
- To homogenize the approach in the use of computer codes (like RELAP, TRACE, CATHARE, ATHLET, CATHENA, PARC, RELAP/SCDAP, MELCOR, IMPACT) for accident analysis;
- To disseminate the use of standard procedures for qualifying thermal-hydraulic system code calculation (e.g. through the application of the UMAE <Uncertainty Methodology based on Accuracy Extrapolation> [12]);
- To promote Best Estimate Plus Uncertainty (BEPU) methodologies in thermal-hydraulic accident analysis through the presentation of the current industrial applications and the description of the theoretical aspects of the deterministic and statistical uncertainty methods as well as the method based upon the propagation of output errors (called CIAU <Code with the capability of Internal Assessment of Uncertainty> [13, 14]);
- To spread available-robust approaches based on BEPU methodology in Licensing Process;
- To address and reduce User Effects;

Figure 1: 3D S.UN.COP Framework to address the user effect problem

Figure 2: 3D S.UN.COP Loop of benefits
To realize a meeting point for exchanges of ideas among the worlds of Academy, Research Laboratories, Industry, Regulatory Authorities and International Institutions. Other two fundamental goals to achieve are:

- To ensure of a suitable Quality Assurance (QA) for the training. Higher Education in Europe is nowadays involved in a process of change as focus has to be set on the student’s workload and on his significant learning. This is a wide subject with many implications that are leading to important changes. Different initiatives have been already carried out in many European universities looking for this new approach in teaching organization and in the methodology used. Some essential aspects to be taken into account are the definition of the learning objectives and results, the planning of activities necessary to reach these objectives, the use of active learning methodologies (cooperative learning, problem-based learning) and the use of continuous learning measurement. To fulfil this main goal, some other objectives have been established:
  - To ensure the teaching quality at the following levels
  - To ensure an adequate learning measurement
  - To establish a procedure for admission of participants
  - To ensure adequacy of teachers
  - To consider the tools for preserving the knowledge
  - To follow the international developments

- The connection with EC objectives and framework. To connect the 3D SUNCOP training with EC objectives and framework. This includes:
  - Experience and dissemination from past and present EC projects which have links with the 3D SUNCOP training subjects (CRISSUE-S, VALCO, CERTA…)
  - Consideration of key results of EC Framework Programs.
  - Consideration of transfer of knowledge inside EC TACIS and Phare projects.
  - Consideration of any individual EC program that may have any connection with the 3D SUNCOP subjects.
  - Consideration of ENEN network initiatives.
  - Consideration of (new) relevant political areas for the EC.
  - To establish a permanent contact with EC offices (Bruxelles, JRC, etc...)

The following main features of the seminar-course may be identified and outlined:

- The idea of practical use of the code: a course without practical code application has (much) lower validity.
- The idea to mix different codes: the use of different code is worthwhile also to establish a common basis for code assessment and for the acceptability of code results.
- The need of exam: exams were in the past courses (very) well accepted by code users. The exam gave them the possibility to show their expertise and to demonstrate the effort done during the course.
- The practical use of procedures for nodalisation qualification that can be directly applied in the participants institutions.
- The practical use of procedures for accuracy quantification that are demonstrated at the qualitative and the quantitative level.
- The “joining” between BE codes and uncertainty evaluation that shows the full application of uncertainty methodologies and the worth of these within a licensing process.
- The establishment, promotion and use of international guidance through large participation of very well known international experts.
5.3 3D S.U.N.COP Training Structure

The seminar is subdivided into three main parts, each of one with a program to be developed in one week. The changes between lectures, computer work and model discussion showed up useful to maintain a steady high level of participant’s attendance. The duration of the individual sessions varied substantially according to the complexity of the subjects and the training needs of the participants:

- The first week (titled “Fundamental Theoretical Aspects”) is fully dedicated to lectures describing the concepts of the proposed methodologies. The following 8 technical sessions (with more than 30 lectures) are presented covering the main topics hereafter listed:
  - **Session I: system codes: evaluation, application, modelling & scaling**
    - models and capabilities of system code models
    - development process of generic codes and developmental assessment
    - scaling of thermal-hydraulic phenomena
    - separate and integral test facility matrices
  - **Session II: International Standard Problems**
    - lesson learned from OECD/CSNI ISP
    - Characterization and Results from some ISP
  - **Session III: best estimate in system code applications and uncertainty evaluation**
    - IAEA safety standards
    - origins of uncertainty
    - approaches to calculate uncertainty
    - user effect
    - evaluation of safety margins using BEPU methodologies
    - international programs on uncertainty (UMS [11] and BEMUSE [12])
  - **Session IV: qualification procedures**
    - qualifying, validating and documenting input deck
    - the feature of UMAE methodology
    - description and use of nodalization qualification criteria for steady state and transient calculation
    - use of thresholds for the acceptability of results for the reference case;
    - qualitative accuracy evaluation
    - quantitative accuracy evaluation by Fast Fourier Transform Based Method (FFTBM)
  - **Session V: methods for sensitivity and uncertainty analysis**
    - GRS statistical uncertainty methodology
    - CIAU Method for Uncertainty Evaluation
    - ASAP and GASAP procedures for Sensitivity Analysis
    - Comparison of Uncertainty Methods with CSAU Methodology
  - **Session VI: relevant topics in best estimate licensing approach**
    - best estimate approach in Brazilian, German, and US licensing
  - **Session VII: industrial application of the best estimate plus uncertainty methodology**
    - Westinghouse realistic large break LOCA methodology
    - AREVA realistic accident analysis methodology
    - GE Technology for Establishing and Confirming Uncertainties
    - BEAU for CANDU reactors
    - UMAE/CIAU application to Angra-2 DEGB licensing calculation

- The second week (titled “Practical Applications and Hands-on Training”) is devoted to lectures on the practical aspects of the proposed methodologies and to the hands-on training on numerical codes like ATHLET, CATHARE, CATHENA, RELAP5 USNRC, RELAP5-
3D ©, TRACE, PARCS, RELAP/SCDAP and IMPACT. The following 4 technical sessions are presented covering the main topics hereafter listed:

- **Session I: coupling methodologies**
  - Cross section generation: models and applications
  - coupling 3D neutron-kinetics/thermal-hydraulic codes (3D NK-TH)
  - uncertainties in basic cross-section
  - CIAU extension to 3D NK-TH

- **Session II: coupling code applications**
  - PWR-BWR-WWER analysis
  - BWR stability issue
  - WWER containment modelling
  - system boron transport, boron mixing and validation

- **Session III: CIAU/UMAE applications**
  - key applications of CIAU methodology
  - example of code results from application to ITF (LOFT, LOBI, BETHSY) and to a NPP (PWR-Type and VVER-Type)
  - PSB counterpart test
  - bifurcation study with CIAU
  - CIAU software

- **Session IV: Computational Fluid Dynamics Codes**
  - The role and the structure of the CFD codes
  - CFD simulation in nuclear application: Needs and Applications

Each of the parallel hands-on trainings on numerical codes consists of about 20 hours and covers the following main topics:

- Structure of specific codes
- Numerical methods
- Description of input decks
- Description of fundamental analytical problems
- Analysis and code hands-on training on fundamental problems (e.g. for RELAP5 fundamental proposed problems deal with boiling channel, blow-down of a pressurized vessel, pressurizer behaviour)
- Example of code results from applications to ITFs (LOFT, LOBI, BETHSY)

The third week (titled “Hands-on Training for Advanced Users and Final Examination”) is designed for advanced-users addressing the user effect problem. The participants are divided in group of three and each group receive the training from one teacher. The applications of the proposed methodologies (UMAE, CIAU etc.) are illustrated through the BETHSY ISP 27 (SBLOCA) and LOFT L2-5 (LBLOCA) tests. Applications and exercises using several tools (RELAP5, WinGraf, FFTBM, UBEP, CIAU, etc…) are considered. The following main topics are covered:

- Modalities for developing (or modifying) the nodalization
- Plant accident and transient analyses
- Examples of code results from application to a NPP (PWR-Type and VVER-Type)
- Code hands-on training through the application of system codes to ITFs (LOFT and BETHSY)

A final examination on the lessons learned during the seminar is designed and consists of three parts:

1) **Written Part:** Questions about the topics discussed during the seminar are proposed and 20 questions are assigned both to each participant and to each group. At least 14 questions must be correctly answered by the group and 14 by each participant.
II) **Application Part:** Two types of problems are proposed to the single participant and to the group:

- **Detection of Simple Input Error:**
  Each participant receives the experimental data of the selected transient, the correct RELAP5 nodalization input deck and the restart file of the wrong input deck containing one simple input error. Each participant shall identify the error.

- **Detection of Complex Input Error:**
  Each group receives the experimental data of the selected transient, the correct RELAP5 nodalization input deck and the restart file of the wrong input deck containing one complex input error. Each group shall identify the error.

Evaluation reports are submitted in a written form containing short notes about the reasons for the differences between results of the reference calculation and results from the ‘modified’ nodalization. At least one problem over two shall be correctly solved to obtain the certificate.

III) **Final Discussion:** Each participant takes an oral examination of about 15-20 minutes, discussing own results (or results obtained by own group) with the examiners. General questions related to lectures presented during the three-weeks seminar are asked to the participants.

A certificate of type “LA Code User Grade” (see Table 1) like the one depicted in Figure 3 is released to participants that successfully solved the assigned problems.

![Figure 3: 3D S.UN.COP “LA Code User Grade” Certificate](image)

5.4 **3D S.UN.COP CANDU 2007 in Hamilton & Niagara Falls (Ontario, Canada)**

The 3D S.UN.COP CANDU 2007 was successfully held in Hamilton & Niagara Falls (Ontario, Canada) from October 8th to October 26th with the attendance of 33 participants coming from 8 countries and 16 different institutions (universities, vendors and national laboratories). 32 scientists (13 countries and 23 different institutions) were involved in the organization of the seminar, presenting theoretical aspects of the proposed methodologies and holding the training and the final examination.

All the participants achieved a basic capability to set up, run and evaluate the results of a thermal-hydraulic system code (e.g. RELAP5) through the application of the proposed qualitative and quantitative accuracy evaluation procedures.
At the end of the seminar a questionnaire for the evaluation of the course was distributed to the participants. All of them very positively evaluated the conduct of the training as can be derived from the charts in Figure 4.

I. WEEK

II. WEEK (LECTURES)

III. WEEK (CODES)

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**Figure 4: Design and conduct of the seminar-training**

- **II WEEK**
  - Participants: 4
  - Scales: Excellent/Thoroughly, Good/Fairly Well, Average/Somewhat, Disappointing/Not at All
  - Topics: Overall satisfaction, quality of lectures, discussions, seminar's organization, relevance of seminars, overall impression of lecturers, balance of hands-on training, hands-on training's quality, balance between lectures and hands-on training, balance between theoretical and practical elements, balance between seminar and organization, overall impression of seminar, balance between seminar and organization, balance between theoretical and practical elements, overall impression of seminar, balance between seminar and organization, balance between theoretical and practical elements.

- **III WEEK**
  - Participants: 5
  - Scales: Excellent/Thoroughly, Good/Fairly Well, Average/Somewhat, Disappointing/Not at All
  - Topics: Same as II WEEK.

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International Topical Meeting on Safety of Nuclear Installations, Dubrovnik, Croatia, 30.9. – 3.10. 2008

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In Table 2 a list of lecturers and participants organizations of all 3D S.UN.COP organized so far is given.

Table 2: Organizations of participants and lecturers to all 3D S.UN.COP (2004-2007)

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<th>Participants and Lecturers Organizations</th>
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CONCLUSIONS

An effort is being made to develop a proposal for a systematic approach to user training. The estimated duration of training at the course venue, including a set of training seminars, workshops, and practical exercises, is approximately two years. In addition, the specification and assignment of tasks to be performed by the participants at their home institutions, with continuous supervision from the training center, has been foreseen.

The 3D S.UN.COP seminars constitute the follow-up of the presented proposal. The responses of the participants during the training demonstrated an increase in the capabilities to develop and/or modify the nodalizations and to perform a qualitative and quantitative accuracy evaluation. It is expected that the participants will be able to set up more accurate, reliable and efficient simulation models, applying the procedures for qualifying the thermal-hydraulic system code calculations, and for the evaluation of the uncertainty.

The ninth 3D S.UN.COP is currently holding (from October 13th through October 31st 2008) at the Institute for Energy, Joint Research Centre of the European Commission in Petten (The Netherlands), with the cooperation of University of Pisa, University of Zagreb, Technical University of Catalonia and the Institute for Energy of the European Commission and involves more than 30 scientists between lecturers and code developers (http://dimnp.ing.unipi.it/3dsuncop/2008/index.html).

REFERENCES


Technical Support in the Field of Thermal-Hydraulics Performed by the JRC-IE in the Framework of the TACIS and PHARE Programmes

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ABSTRACT

The European Commission has been supporting both the European Union candidate countries and the Commonwealth of Independent States by providing financed technical assistance to enhance their ongoing economical transition processes via the PHARE and TACIS programmes.

One of the areas of cooperation is the support for nuclear safety. The Institute for Energy of the Joint Research Centre of the European Commission has been actively providing technical & scientific expertise in this domain to the various Directorates General of the European Commission involved in these programs.

More specifically expertise has been provided in the fields of: On-Site Assistance, Assistance to the Regulatory Authorities, Design Safety and Radioactive Waste Management.

In the Design Safety domain a set of projects were specifically aimed at addressing thermal-hydraulic issues related to RBMK and VVER plants, which ranged from transfer of Western know-how methodology and training Eastern experts to specific accident and severe accident analysis.

The main objectives were to improve the safety of the plants via both deterministic Design Base Accident analysis and by Beyond Design Base Accident analysis, including also the validation of specific thermal-hydraulic codes to be used by the beneficiary organisations and by the regulatory bodies for licensing purposes.
This paper will address the concerned projects of the PHARE and TACIS programmes identifying the objectives of the projects and the main results that have been achieved.

1 INTRODUCTION

The Nuclear power stations in the European Union (EU) currently produce around a third of the Community's electricity. The foundations for nuclear energy in Europe were laid in 1957 by the European Atomic Energy Community (EURATOM) whose main functions consisted of furthering cooperation in the field of research, protecting the public by establishing common safety standards, ensuring an adequate and equitable supply of ores and nuclear fuel, monitoring the peaceful use of nuclear material and cooperating with other countries and international organizations. The tool with which the European Commission (EC) promotes the convergence with its extensive legislation (the acquis communautaire) to the candidate or potentially candidate countries for accession is the PHARE (Poland and Hungary: Assistance for Restructuring their Economies) program. Furthermore it provides grant-financed technical assistance to 12 countries of Eastern Europe and Central Asia via the TACIS (Technical Assistance to the Commonwealth of Independent States) program in order to enhance and support the transition to market economies.

The PHARE program is one of the three pre-accession instruments financed by the European Union to assist the applicant countries of Central and Eastern Europe in their preparations for joining the European Union. Originally created in 1989, it has rapidly expanded from Poland and Hungary to include six more countries of the 2004 accession (the Czech Republic, Estonia, Latvia, Lithuania, Slovakia, and Slovenia) as well as those countries that entered the union in 2007 (Bulgaria and Romania).

The TACIS program is the EU tool to promote the transition to a market economy, reinforce democracy and the rule of law and to observe the democratic principle and human rights in the Commonwealth of Independent States. It was originally launched in 1991 as a technical assistance program and has developed as an instrument of cooperation based on Partnership and Cooperation Agreements. It includes the following countries: Armenia, Azerbaijan, Belarus, Georgia, Kazakhstan, Kyrgyzstan, Moldova, Russia, Tajikistan, Turkmenistan, Ukraine and Uzbekistan. Nuclear safety is a priority sector having as major objectives: to promote an effective safety culture; to support the establishment of strategies in the management of spent fuel, decommissioning and waste management; and to contribute to international activities (such as the G7/EU initiative for the closure of Chernobyl).

Within the EC the TACIS and PHARE programmes are entrusted to different Directorates General. The Directorate General Enlargement (ELARG) is responsible for the PHARE programming and implementation. For TACIS, the Directorate General External Relations (RELEX) is responsible for the strategy while the programming and implementation are dealt with by Directorate General EuropeAid (AIDCO).

The Joint Research (JRC) of the European Commission is providing its technical & scientific expertise to these Directorates General via the Nuclear Operation Safety Unit of the Institute for Energy. The JRC carries out work in the following areas of nuclear safety: On-Site Assistance, Design Safety, Industrial Radwaste Management, Decommissioning and Safeguards and Regulatory Authorities Assistance.
The expertise is provided to the entire project cycle ranging from the programming phase to the assessment of the results. Examples of activities are: drafting or review of Project Description Summaries (PDS), procurement Technical Specifications and Terms of Reference (TOR); participation in tender Evaluation Committees; technical follow-up of projects; and reviews of project deliverables.

A large number of TACIS and PHARE projects have been financed in the field of Thermal-Hydraulics, either specifically centred on T-H issues or implying T-H calculations within specific tasks of the project. These have been successfully implemented and the quality of the work has been recognized.

This paper reports on the projects performed in the accident analysis area where there has been stand-alone use of T-H codes or coupled with N-K or fuel rod mechanical codes. Thirty-two projects have been identified and each is briefly presented in this paper.

Most of the information in this paper originates from Ref [1] and the JRC Dissemination website (Ref [2]). The remaining references are restricted documents that have been consulted mainly for clarification purposes.

2 PROJECTS SUMMARY

Since 1991, 14 PHARE and 18 TACIS projects have covered a wide range of T-H issues. Assistance was provided to the utilities, design organisations and the Regulatory Authorities and their Technical Support Organisations. The general objective was to improve the safety of the RBMK (7 projects) and VVER (25 projects) reactors by performing Design Base Accident (DBA) or Beyond Design Base (BDBA) analysis. The validation of codes used for licensing purposes, and the training of the local staff to use them with adequate hardware were also a major topic of investigation.

In the TACIS framework 14 T-H related projects were performed for the Russian Federation and 4 for Ukraine whereas in the PHARE framework Bulgaria had 2 projects, the Czech Republic had 3, Hungary had 2 and Lithuania had 3. The Czech Republic, Slovakia and Hungary participated in 4 joint multi-country projects.

2.1 PHARE

BG/TS/01(F) – Transfer of Western Methodology to Bulgarian Nuclear Safety Authority

The BG/TS/01(F) project was part of the 1991 PHARE programme in support to Technical Support Organisations. The beneficiary country was Bulgaria and the main plants concerned were the VVER 440-230 of Kozloduy.

The objective of the project was to provide technical support to the Bulgarian Nuclear Safety Authority (BNSA) and to ensure the transfer of know how and Western methodology to its staff and technical support.

Accident analysis assessment was one of the specific objectives of the project.

PH/91(C) - Review of accident analysis units 1-4 (Item C)

The PH/91(C) project was part of the 1991 PHARE programme in support to the Technical Support Organisations. The beneficiary country was Bulgaria and the main plants concerned were the VVER 440-230 of Kozloduy.

The objective of the project was to review the accident analysis prepared for Units 1-4 establishing setpoints for actuation of the unit Safety Systems. The following accidents were considered: (steam system piping failure, loss of normal feedwater, feedwater system pipe
break, boron dilution, rod ejection, inadvertent opening of a pressurizer relief valve and steam generator tube rupture).

For each accident, a set of conservative assumptions was established and the single failure criterion was applied throughout the analyses. The RELAP5/MOD2 computer code was used to analyze most of the thermal and hydraulic transients.

The results of the analyses indicated that the Kozloduy units 1 and 2 could cope with the accidents analyzed with sufficient margin. A few recommendations for hardware modifications and performance of additional analyses were also made.

4.2.6A/93 - VVER 440/213 Accident Analysis - DBA Analysis Improvement
The 4.2.6A/93 project was part of the 1993 PHARE programme in support to Design Safety. The beneficiary countries were Czech Republic, Slovakia and Hungary and the main plants concerned were the VVER 440-213 of Dukovany-2, Bohunice 3-4, Paks 1-4.

The objective of the project was to improve and to extend the present capability to analyse DBA of VVER 440-213 reactors with the aid of advanced TH codes and using Western methodology. Containment analysis and reactivity accidents were excluded.

This project was carried out in parallel with PHARE project 4.2.6B/93 in order to incorporate the experimental results into the assessment and the validation of the ATHLET code. Computer hardware was also supplied to Dukovany (the main beneficiary). Bohunice NPP and Paks NPP were co-beneficiaries.

It was recommended that two LB LOCA calculations, one Intermediate Break LOCA and three ATWS calculations be performed for Dukovany 2.

4.2.6B/93 – VVER 440-213 Accident Analysis DBA Code Verification
The 4.2.6B/93 project was part of the 1993 PHARE programme in support to Design Safety. The beneficiary countries were Czech Republic, Slovakia, Hungary and the main plants concerned were the VVER 440-213 of Dukovany-2, Bohunice 3-4, and Paks 1-4.

The objectives of the project were to identify the experimental support needed for validation of T-H codes applied to the analyses of DBA in VVER-440/213 reactors, to provide limited additional experimental data by integral loop testing in the PMK-2 test facility and finally to assess the predictive capability of code(s) through code validation using these available experimental results. Paks was the main beneficiary and Dukovany and Bohunice were co-beneficiaries.

An integral test matrix covering phenomena relevant to VVER-440/213 DBAs was established and two experiments were defined and performed in the PMK-2 facility: a) Inadvertent opening of the pressurizer relief valve and b) break of the pressurizer surge line. Pre-test analyses were performed with the ATHLET MOD 1.1 and RELAP5/MOD 3.2 codes to establish suitable boundary conditions for the tests and to confirm expected phenomena. Code validation was performed using PMK-2 experimental results. A series of post-test analyses with ATHLET MOD 1.1 code showed good agreement between calculation and the key measured parameters.

4.2.7A/93 - VVER-440/213 Beyond Design Basis Accident Analysis and Accident Management
The 4.2.7A/93 project was part of the 1993 PHARE programme in support to Design Safety. The beneficiary countries were Czech Republic, Slovakia and Hungary and the main plants concerned were the VVER440-213 of Dukovany-2, Bohunice 3-4, Paks 1-4.

The objectives of the project were: the identification of relevant scenarios and assessment of VVER-440/213 behaviour in case of BDBA and SA; the definition and validation of preventive and mitigative AM strategies coping with BDBA; the identification
of possible mitigation strategies and evaluation of their efficiency in case of SA scenarios and the transfer of Western technology relative to AM and SA mitigation strategies after adaptation to VVER-440/213 reactors.

The code used within the project was the MAAP4 VVER version. The code was used to perform an integrated simulation of accident conditions in VVER type reactors, including severe accidents involving core melt, vessel failure and ex-vessel challenges to containment.

The main results were 20 BDBA sequence analyses performed with MAAP4/VVER. Code transfer and training were included in the scope of the project along with the identification of areas for potential improvement which led to recommendations for improvements to preventive AM. The final recommendation was that the plants should proceed with equipment upgrades to improve mitigation of SA and the implementation of SAMG.

**HU/TSO/VVER02 - Topical Issues Concerning Accident Analysis: Methodologies and Management (PAKS Units 1, 2, 3 and 4)**

The HU/TSO/VVER02 project was part of the 1994 PHARE programme in support to Design Safety. The beneficiary country was Hungary and the plants concerned were the VVERs 440-213 of Paks.

The objective of the project was to assist and to cooperate with the Hungarian Nuclear Regulatory Authority and its Technical Support Organisation in the licensing assessments of the improvement programme of the Paks NPP Units 1 to 4 in the following fields of development of accident management procedures: development and validation of 3D coupled codes, evaluation of ATWS analysis methods and development of methodologies for the application of best estimate computer codes for licensing purposes. A set of comparison calculations using the ATHLET, CATHARE and RELAP codes was performed analysing boron dilution and steam line break. An integral type ATWS test for the PMK facility was defined and executed and post-test calculations were performed using European codes.

Finally a methodology applicable to VVER type reactors was elaborated. The project deliverables also included a discussion of applied codes and models, the status of code validation and the assessment of code results for VVER 440-213 specific tests and basic experiments, discussion of the reasons for the selection of the cases, initial and boundary conditions and uncertainties in the plant data, questions of quality assurance as well as accuracy and reliability of quantitative results.

**PH2.06/94- Need and Alternatives for Filtered Venting of the Containment**

The PH2.06/94 project was part of the 1994 PHARE programme in support to Design Safety. The beneficiary country was Hungary and the plants concerned were the VVER440-213 of Paks.

The main objective of this project was to develop safety goals and functional requirements for filtered venting system, to evaluate a potential system for VVER 400-213 application, and finally to provide recommendations for the selection of a system that satisfies safety goals and meets the functional requirements.

In order to achieve the goal a set of analyses were performed by VEIKI Budapest, Hungary and NRI Rez, Czech Republic, using the MAAP4/VVER code. The following severe accident sequences were studied: Loss of all AC power with operator action to depressurize the system at core damage, Small break LOCA (ID 25mm) with failure of safety injection, Large LOCA (double ended, ID 500mm) with failure of safety injection.

The T-H results of the project are a large set of LOCA analyses with various break sizes. The transients were performed to study important severe accident phenomena to
investigate possible solutions to limit radioactive releases, prevent late containment failure and cope with severe accident hydrogen.

SRR3/95- Experimental and Calculational Investigation of System Behaviour during Accident and Accident Management in VVER Reactors

The SRR3/95 project was part of the 1995 PHARE programme in support of the Technical Support Organisations. The beneficiary countries were Czech Republic, Slovakia and Hungary and the plants concerned were the VVER 440-213 of Dukovany-2, Bohunice 3-4 and Paks.

The main objective of this project was to contribute to the establishment of a VVER specific database by generation of experimental data from the PMK-2 and the ISB integral test facilities representing, respectively, the Russian-design VVER-440 and VVER-1000 nuclear power plants.

The second objective of the project was a contribution to the validation of European Thermal-hydraulic System Codes for VVER applications by performing post-test calculations and verifying the correct simulation of specific thermal-hydraulic phenomena with respect to the experiments.

In the experimental area an upgrading by advanced instrumentation of the PMK-2 test facility was performed as well as three experiments on the PMK-2 (stationary test addressing energy transport in the steam generator with lowered secondary side water level, 0.5% cold leg break overfed by HPIS, 7.4% cold leg break without HPIS, with secondary bleed and primary B&F) and one on the ISB test facility (0.5% cold leg break overfed by HPIS).

In the analytical area the validation matrix was extended to include AM procedures and scaling effects were investigated by means of data and results from tests on the ISB-VVER (according to VVER-1000) and PMK-2 /PACTEL (according to VVER-440) facilities. The European ATHLET and CATHARE codes were verified on the basis of these data and results.

CZ/TS/01 – Licensing Related Assessments for design and operational safety of VVER 213, Dukovany 1,2,3 and 4

The CZ/TS/01 project was part of the 1998 PHARE programme in support of the Technical Support Organisations. The beneficiary country was the Czech Republic and the plants concerned were the VVER 440-213 of Dukovany.

The main objectives were to enhance the effectiveness of the Czech Nuclear Regulatory Authority and its TSO in the accident analysis and ten year operational safety assessment domain.

The main activities were the identification of DBA analyses requiring improvement, performing selected DBA analyses and assessment of the representativeness of the thermal-hydraulic analyses.

CZ.01.14.02- Assessment and Validation of Computer Codes Based on PSB-VVER Experimental Data

The CZ.01.14.02 project was part of the 2001 PHARE programme in the Design Safety domain. The beneficiary country was the Czech Republic and the plants concerned were the VVER1000-320 of Temelin.

The main objectives were to define a set of initiating events and a set of computer codes to be verified and validated using experimental data from the Russian integral facility PSB-
VVER-1000 in order to increase confidence of Thermal-hydraulic calculations of VVER-1000 reactors.

The work consisted of reviewing relevant system computer codes with regard to their suitability in predicting VVER-1000 relevant accidents and transients, to determine the dominant phenomena, which needed additional validation (the white spots) and to check the capabilities of the PSB-VVER-1000 facility to provide experimental data for those validations, and finally to derive from the identified white spots, experimental data from the Electrogorsk Russian PSB-VVER-1000 facility.

**LI.01.18.01- Support to VATESI and their TSOs during Review of the Safety Analysis Report of Ignalina Unit 2**

The LI.01.18.01 project was part of the 2001 PHARE programme in the Design Safety domain. The beneficiary country was Lithuania and the plant concerned was the RBMK of Ignalina-2.

The main objectives were to provide EU expertise to support the Lithuanian safety authorities (VATESI) and its TSOs in their review of the SAR-2 by gaining insight into the Lithuanian review process and enhancing it by providing feedback and supporting the training of the Lithuanian review team.

A specific task within the project was to produce basic regulatory documents for VATESI licensing activities related to DBA and BDBA and to provide basic training courses on reactor core physics and on severe accident phenomenology and on-the-job training on the use of coupled codes ATHLET and QUABBOX/CUBBOX

**LI.01.18.03- Support to VATESI for important tasks relevant to the licensing activities of Ignalina Nuclear Power Plant**

The LI.01.18.03 project was part of the 2001 PHARE programme in the Design Safety domain. The beneficiary country was Lithuania and the plant concerned was the RBMK of Ignalina-2.

The main objectives were to support the development of the VATESI regulatory guide on implementation of requirements for accident analysis, to improve skills on use of DBA coupled computer codes to perform regulatory analyses and to support the development of requirements on BDBA assessment and management for RBMK-1500 reactor.

The results of the accident analyses were used to evaluate the reactor safety level during operational transients and the effectiveness of the safety functions during DBA accidents.

**5812.02.02 - Assessment and Validation of Computer Codes Based on PSB-VVER Experimental Data. Computer Codes Validation**

The 5812.02.02 project (follow-up of CZ.01.14.02) was part of the 2003 PHARE programme in the Design Safety domain. The beneficiary country was the Czech Republic and the plants concerned were the VVER1000-320 of Temelin.

The main objectives were to support the activities of the Czech Regulatory Authority in defining the methodology of validation of best estimate system computer codes and thermal-hydraulic analyses evaluation. This was done by demonstrating improvements in code
simulation (validation of the numerical simulation of certain thermo-hydraulic events through comparison with experimental results).

Several computer codes were validated and a methodology was developed for utilisation of experimental data from a large-scale integral test facility for the different initial events specific for VVER 1000 reactor. The results were evaluated, including statistical evaluation as well as accuracy/uncertainty for selected initial events. This project has provided necessary support for the application of advanced system computer codes, which will be used in the process of safety evaluation of a VVER-1000 NPP (Temelin NPP).

5812.04.02 - Support to VATESI and its TSOs in assessment of beyond design basis accidents for RBMK-1500 reactors

The 5812.04.02 project was part of the 2003 PHARE programme in support of the Technical Support Organisations. The beneficiary country was Lithuania and the plants concerned were the RBMKs of Ignalina.

The main objectives were to support and increase the competence of VATESI and its TSOs in assessing the BDBA in the following areas: accident scenarios regarding behaviour of the defence-in-depth barriers (fuel matrix, fuel rod, main coolant circuit, and reactor confinement), behaviour of the spent nuclear fuel in the pools of Ignalina NPP, radionuclide transfer and discharge to the environment, including realistic evaluation of source-terms during BDDB.

The main result of the project was the increased competence of the Lithuanian nuclear safety authority and strengthened technical expertise resources of VATESI and its TSOs in the area of the beyond design basis accidents for Ignalina NPP.

This work included the analysis of transients that potentially could develop into BDDB with core damage, development with verification, validation (based on data from experimental measurements in situ) of input decks of fuel behaviour, neutronics and thermo-hydraulic computer codes, adapted for RBMK-1500, development of Regulatory Guides on the assessment and management of BDDB for RBMK-1500;

2.2 TACIS

R1.3/91 – Accident analysis

The R1.3/91 project was part of the 1991 TACIS programme in the Design Safety domain. The beneficiary country was the Russian Federation and the main sites concerned were Kola and Novovoronezh.

The objective of the project was to review the existing Accident Analysis done by Russian Experts for the VVER 230 projects at Kola 1-2 (VVER-230) and Novovoronezh 3-4 (VVER-179) and, if necessary, to complete this analysis in order to provide a systematic and consistent "state of the art" safety analysis, in accordance with western practice. A Western code transfer and training on new hardware and software (RELAP5/ MOD2.5, CATHARE, WAVCO and HEXTIME/COBRA) were also included together with a wide set on LOCA, Non-LOCA, Confinement and 3D Core Physics calculations.

In conclusion the results of the accident analysis were positive and both Western and Russian experts gained valuable experience in application of Western methodology and tools to VVER-type reactors.

R2.1/91 – RBMK Safety Assessment
The R2.1/91 project (R/TSO/RBMK/B2.1/91) was part of the 1991 TACIS programme in the support to Technical Support Organisations domain. The beneficiary country was the Russian Federation and the plants concerned were the Smolensk-3 and Chernobyl-3 RBMKs.

The objective of the project was to review the safety of the RBMK reactors in nine technical areas. In two of the nine areas, “Accident Progression” and “Core Physics”, Thermal-Hydraulic and Neutron-Kinetic analyses were performed using both Western and Eastern codes (CATHARE, RELAP5/MOD1, MOD2 and MOD3, RDIPE, RAZRYV VK, PHENIX, KRITIKA, PIPPELLA, TRANS-8 and TRIADA, RAPTA, MAKET-2 and PTETNETS).

About 50 accident sequences were reviewed in detail by Western experts including LOCAs, Reactivity Transients, Loss of Flow Transients and Loss of Electric Power Supply.

In conclusion, the western experts gained knowledge in the state of the art of Russian TH accident analyses and the review highlighted the importance of improving management systems and practices at the plant and throughout the Eastern Nuclear Industry. Over 150 recommendations were made covering: improvements to existing systems, computer codes, safety analyses and proposals for further work.

R3.8/91 – Severe Accidents and Accident Management Technology
The R3.8/91 project was part of the 1991 TACIS programme in the support to Technical Support Organisations domain. The beneficiary country was the Russian Federation and the plants concerned were all the VVER. The reference plant was Balakovo Unit 4 (VVER-1000).

The primary objectives of the project were the transfer of Western technology and experience on Severe Accident (SA) analysis to the Russian counterpart, and to assist and collaborate with the experts in the review of SA tests performed internationally (modelling of physical phenomena) and in the frame of this project. Other objectives were to transfer the Western technology and experience on accident management to the Russian counterpart and to assist and collaborate with the Russian experts in the analysis of the reference VVER plant to develop AM measures for prevention of core damage and mitigation of accident consequences.

In conclusion, the selected codes, RELAP5, ESCADRE and MELCOR, were transferred to Russian experts together with deep training in SA scenarios (LBLOCA, SBLOCA, Station Blackout) phenomena and in the use of the code (detailed model of the plant). Areas for improvements to code modules were also identified.

R5.1.1/91 – Transfer of accident analyses codes and their application
The R5.1.1/91 project was part of the 1991 TACIS programme in the Design Safety domain. The beneficiary country was the Russian Federation and the plants concerned were all the VVER (VVER-1000 and VVER-440)

The aim of the project was to provide transfer, application and verification of Western codes to the Safety Authorities of Russia (GAN-RF) and the Technical Safety Organizations (TSOs) and to provide assistance to the verification and validation of Russian codes used for safety analyses of Russian pressurized water reactors (VVER type reactors).

The activities consisted mainly in the definition and installation of the technical infrastructure (computers and network) and implementation of the French and the German codes dealing with thermal-hydraulics (CATHARE2 V1.3E Rev. 5 and ATHLET 1.1A) and with severe accident processes (ICARE2 V2 Mod 1, ESCADRE Mod 0.1).

In conclusion computer hardware, codes and code training have been provided, along with exchange of knowledge on safety approaches, in order to get a common understanding of the phenomenology in Western Light Water Reactors (LWRs) and Russian VVER reactors during loss of coolant accidents, transients and severe accidents.
RF/RA/01 – Transfer of Western Methodology to Russia – first year

The RF/RA/01 (R5.1.2/91) project was part of the 1991 TACIS programme in the support to Technical Support Organisations domain. The beneficiary country was the Russian Federation and the plants concerned were all the VVERs.

This project was a part of a multi-annual project consisting of 4 parts spread over 6 budget years. The objective was to provide support along with the transfer, application and verification of Western Codes to Gosatomnadsor. The project was divided into several tasks. One of the tasks was incident reporting and analysis including regulatory response.

 Assistance was provided for code validation along with accident analysis and accident management, and in particular for Severe Accident Management.

R/TSO/VVER/01A – Licensing related assessment for design and operational safety of VVER - Subtask A Kola 1-2

The R/TSO/VVER/01A project was part of the 1994 TACIS programme in the support to Technical Support Organisations domain. The beneficiary country was the Russian Federation and the plants concerned were Kola 1-2 (VVER 440-230)

The objective of this project was to assist the GAN-RF and Russian TSOs in reviewing and assessing the accident analysis performed for units 1 and 2 of Kola NPP within project TACIS R1.3/91, and the transfer of know-how from the Western TSOs to the Russian TSOs.

The main tasks carried out were: the familiarisation of EU-TSOs with the relevant Russian licensing practices, the collection of information on already performed analyses in the consortium countries and Russia, the assessment of code validation and of all the VVER accident analyses of R1.3/91, the selection and execution of supplementary calculations and final assessment of the results. The main accident categories considered were: Loss of coolant accidents (LOCA), small break LOCAs, Transients without LOCA, including overcooling transient and ATWS and Beyond Design Basis accidents (BDBA). The assessment also included the analysis of confinement behaviour during LOCAs.

R/TSO/VVER/01B – Licensing related assessment for design and operational safety of VVER - Subtask B Novovoronezh 3-4

The R/TSO/VVER/01B project was part of the 1992 TACIS programme in the support to Technical Support Organisations domain. The beneficiary country was the Russian Federation and the plants concerned were Novovoronezh 1-2 (VVER 440-230)

The objective was to assist GAN in technical evaluation of the proposed safety upgrading measures, including design and operational aspects, related to Novovoronezh units 3 and 4.

The main tasks carried out were: the Safety Assessment of Confinement Systems (based on TACIS 1.10/91), the Accident Identification and collection of information from various design institutes (no Final Safety Analysis Report exists for Novovoronezh 3-4), the comparison with results originating from TACIS 1.3/91 (calculations for Kola 1-2).

The work resulted in recommendations as to the definition of design basis accident and beyond design basis accidents for these running units of the older design of V-230 units.

R/TSO/VVER/02 – Assessment of severe accidents and accident management

The R/TSO/VVER/02 project was part of the 1992 TACIS programme in the support to Technical Support Organisations domain. The beneficiary country was the Russian Federation and the plants concerned were all the VVERs.

The main objective of this project was to co-operate with and assist the Safety Authorities of Russia (GAN) and related Technical Safety Organisations to review the
Russian severe accident issues (approach, events, phenomena, existing analysis and plant specific studies on severe accidents and accident management).

The tasks in particular were: familiarization with the Russian severe accident issues (based on TACIS 3.8/91) and the regulatory approach, assistance in defining relevant events leading to beyond DBA scenarios and associated important severe accident phenomena, review of existing analysis and plant specific studies (probabilistic and deterministic) on severe accidents and accident management measures proposed by the utilities: On the basis of analysis results, the effectiveness of the measures coping with dominant beyond DBA sequences identified by the Probabilistic Safety Assessment (PSA) were reviewed and assessed.

**R2.30/94 – Severe transient analysis for RBMK reactors**

The R2.30/94 project was part of the 1994 TACIS programme in the Design Safety domain. The beneficiary country was the Russian Federation and the plants concerned were all the RBMKs.

The main objective of the project was to review the safety of the RBMK reactors by comparing Russian and Western computer codes and methodologies and by performing four transients: Control Rod Withdrawal Accident, Flow blockage in a DGH, MCP pressure header rupture ("Ultimate DBA" and BDBA) and Loss of Feedwater (ATWS) and comparing the results.

In Conclusions good code agreements were found in general and the differences between the Eastern/Western calculations were understood. Recommendations for further code improvement were also given.

**95-1295 (NUCUK 9402) – Support to Ukrainian NRA in Licensing Activity related to Completion and Safety Upgrading of Rovno 4, Khmelnitsky 2 and Zaporozhye 6**

The 95-1295 contract was part of the 1995 TACIS programme in the Design Safety domain. The beneficiary country was Ukraine and the plants concerned were all the VVER-1000s.

The main objectives of the project were to provide an independent design review of the proposed modernization programme of Rovno 4 and Khmelnisky-2.

During the Design Review, sixty four Basic Design documents (among a total of 182) were duly analysed. Among the four most important of these was one covering accident analysis. The most important identified additional needs were the analysis of the risk induced by the automated control of steam generator header failure accident; and the confirmation that pressuriser safety valves are capable of ensuring the feed-and bleed discharge flow.

**96-0518 – Development of technical documentation for engineering support for the completion and upgrading of Rovno-4 and Khmelnitsky-2.**

The 95-0518 contract was part of the 1996 TACIS programme in the Design Safety domain. The beneficiary country was Ukraine and the plants concerned were all the VVER-1000-320s.

The main objectives were to provide the basic engineering documentation and tools necessary for the financial dossier to be submitted to international financing institutions for a loan needed for the completion and safety upgrading of R4K2 (VVER 1000/320) units in accordance with the internationally accepted safety criteria.

Within the planned outputs were western computer calculation codes for design safety analyses along with hardware. The RELAP code for accident analysis was installed together with other codes and training of experts at Kiev Energo Projekt (KIEP) was provided.
U6.03/96– Development of the procedure for verification of coupled hydraulics and neutron kinetics codes for simulation of dynamic systems

The U6.03/96 project was part of the 1996 TACIS programme in the Design Safety domain. The beneficiary country was Ukraine and the plants concerned were all the VVER-1000/320s.

The objectives of the project were to increase the knowledge on VVER core behaviour through the verification of Neutron Kinetic computer codes coupled to system thermal-hydraulic codes by preparing input data calculating neutronic constants (for dynamic simulation in VVER) and collection of experimental data from Ukrainian NPP.

The main project achievements were the development of numerical algorithms for neutron transport simulation in heterogeneous systems when diffusion approximation is invalid, the development of algorithms and codes for VVER reactor accident simulations for the Safety Analysis Report.

R2.02/96 – Emergency protection signal effectiveness evaluation for VVER 440-230.

The R2.02/96 project was part of the 1996 TACIS programme in the Design Safety domain. The beneficiary country was the Russian Federation and the plants concerned were all the VVER-440/230s.

The objective of the Project was to use Western technology, software and hardware (received in TACIS 1.3/91) and to define a set of accident scenarios to act as design basis accidents for protection system of VVER-440/230, to perform necessary accident analyses and to evaluate the possible addition of new protection signals.

The six following signals were evaluated: Low SG level for Emergency Protection actuation, Low SG level for Emergency Feedwater (EFW) pump actuation, High hot leg temperature, High pressurizer level, Low primary pressure and Low pressurizer level.

The planned outputs were achieved and this know-how resulted in new ideas, which lead to the possible addition of new protection signals for VVER 440-230 substantiated by the performed analyses.

U2.01/94 (part 1) – NUCUK94U201 - Engineering for VVER1000 (part 1) - Contract n.97-0672

The U2.01/94 project was part of the 1997 TACIS programme in the Design Safety domain. The beneficiary country was Ukraine and the plants concerned were all the VVER-1000s.

The objectives of the project were to enable KIEP to perform the nuclear safety analyses using up-to-date computer codes developed in the EC countries and to implement a comprehensive QA system for engineering activities.

The CATHARE code was transferred to the beneficiary with the supply of appropriate computer hardware and training on codes to the specialists of KIEP.

R2.03/97 (Part A) “Software Development for Accident Analysis of VVER and RBMK Reactors”

The R2.03/97(Part A) project is part of the 1997 TACIS programme in the Design Safety domain. The beneficiary country was the Russian Federation and the plants concerned were all the VVERs.

The objectives of the project were the development of experimental and analytical analysis for the management of VVER-1000 accidents based on the PSB-VVER facility, a critical appraisal of the Accident Management Procedures (AMPs) of Balakovo Unit 3 and, if necessary, revision of the AMPs of this plant.
The expected results are a test matrix and pre- and post-test computational analyses of the experiments. The experiments were simulated in the reference plant and experimental results compared (scaling issue) and uncertainty analysis was applied to the calculations, but not for licensing purposes.

**BISTRO BIS/00/017/011 N "Development of experimental programme on the PSB-RBMK integral test facility for validation of thermal-hydraulic system codes aimed at safety analysis of NPPs with RBMK-type reactors**

The BISTRO BIS/00/017/011 N project was part of the 2002 TACIS BISTRO programme (The Bistro program is designed to respond quickly to requests for support to small-scale projects). The beneficiary country was the Russian Federation and the plants concerned were all the RBMKs.

The main objective of the project was to contribute to the validation of T-H codes for application to RBMK reactors. The PSB-RBMK test facility (under construction at EREC at the time of the project implementation) was considered for performance of integral experiments. After review of two already existing validation matrices for LOCA and transients, an experimental program (test matrix) for the PSB-RBMK facility was prepared and accepted. Two scenarios from the list of suggested experimental scenarios were selected for pre-test calculations performed with the T-H code ATHLET MOD1.2-Cycle D (Core cooling under natural circulation with loss of AC power and DGH partial break downstream of the check valve with loss of AC power).

The experts of the Russian subcontractor (EREC) developed the input deck for code ATHLET MOD1.2-Cycle D and performed the calculations. The pre-test calculations demonstrated that the main processes and phenomena under accident conditions can be reliably reproduced with the ATHLET code with the input deck prepared for the PSB-RBMK.

**R2.02/02 Development of safety analysis capabilities for VVER-1000 transients involving spatial variations of coolant properties (temperature or boron concentration) at core inlet**

The R2.02/02 project is part of the 2002 TACIS programme in the Design Safety domain. The beneficiary country was the Russian Federation and the plants concerned were all the VVER-1000.

The main objective is to help the Russian Authorities to address the safety issues and research needs that are related to the operation of VVER by bringing to the Russian nuclear power companies and to linked Russian research/design institutes a set of validated capabilities to address the issue of the VVER-1000 transients involving spatial variations of coolant properties (temperature or boron concentration) at core inlet.

These capabilities shall include a code system, with a thermal-hydraulic part which has been validated and which is ready for an ultimate certification by the Russian nuclear regulatory authority (GAN).

**R2.03/02 Thermal hydraulic code validation at the PSB RBMK Integral Test Facility**

The R2.03/02 project was part of the 2002 TACIS programme in the Design Safety domain. The beneficiary country was the Russian Federation and the plants concerned were all the RBMKs.

The objective of the project is to help the Russian Federation to address safety issues and research needs related to the design and operation of RBMK nuclear power plants, by significantly improving the validation status of thermal-hydraulic codes for RBMK reactor safety analysis.
The work of the project will provide a sound platform from which the preferred thermal-hydraulic codes used in the Russian Federation for RBMK safety analysis can achieve complete validation. The success of the project will be evidenced, ultimately, by the certification of the selected codes by the Russian nuclear regulatory authority (GAN) and their subsequent use in the preparation of improved safety analysis reports for the RBMKs, which must also be accepted by GAN. The certification and application of the codes is, however, outside the scope of the project.

3 CONCLUSIONS

The PHARE and TACIS programs have allowed an effective transfer Western safety culture and know-how to the accession countries of Central and Eastern Europe and the New Independent States and have improved communication and collaboration in both directions.

These programs have made available to the beneficiary countries both Western TH codes, the training, and the associated Western methodologies, experience and approach in several areas (code validation, SA phenomena, acceptance criteria, AM procedures, SAMG, etc) but once the accession countries have entered the EU, instruments like PHARE cease to apply to them.

The implementation of PHARE funds programmed before accession will continue for at least three years in the new Member States and currently only Romania and Bulgaria still have uncompleted projects.

All TACIS projects defined until the year 2006 are or will be implemented in the coming years. Started in January 2007 and running until 2013 the “Instrument for Nuclear Safety Cooperation” (INSC) has replaced the TACIS nuclear safety programme. This new instrument extends the scope of action and covers not just countries from the former Soviet Union. The approach to nuclear safety will be performed through reinforced cooperation with the Regulatory bodies, 'soft' assistance and the creation of a nuclear safety culture. Safety analysis will be considered under INSC whenever it is found necessary and useful for improving the operational safety.

REFERENCES


ABSTRACT

The effect of downcomer behaviour in blowdown scenarios has been studied in detail as part of the general strategy carried out by the authors in the field of qualifying power plant nodalizations through systematic analyses of scaled calculations. Phase separation usually occurs due to gravitational forces, which cause the liquid phase to pool at the bottom of vertical volumes or at the lower part of large horizontal pipes. However in LOCA scenarios, the downcomer geometry causes a circumferential component for coolant velocity confirmed in facilities like UPTF. This circumferential velocity can be quite big during the blowdown phase of a LBLOCA and may lead to a radial phase separation due to centrifugal forces. The paper treats the problem at two different levels. On the one hand it establishes the theoretical equations devoted to characterize the phenomenon, and on the other hand it shows the results of 3D calculations for two different geometries. The results are produced bearing in mind the general goal of helping engineering judgements devoted to explain and evaluate the suitability of scaled calculation results.

Keywords: downcomer, Large LOCA, ECCS, PWR.

1 INTRODUCTION

Helical flow appears as a result of transversal pressure gradients. Such situations are of interest in industrial applications like chemical and nuclear technology, where annular geometries are frequently used. Annular geometry is particularly interesting because some phenomena can appear that are not easily scalable. So, often related simulations show results which are inconsistent with experimental data.

In nuclear reactors an annular duct, the downcomer, drives the coolant from the inlet nozzles into the core. In some accidental PWR scenarios, the downcomer serves as a link between the Emergency Core Cooling System (ECCS) and the core itself.

In this paper, the effect of helical flow occurring in the first stages of a large break LOCA is analyzed. In this scenario the ECCS must provide core cooling, but varying amounts of coolant may bypass the core, flowing through the downcomer from the injection loops to the broken loop (see Fig. 1). In this situation, the downcomer can perform as an efficient phase-separation device, where inertial forces cause a strong stratification. The consequences of this phenomenon are not only cinematic but dynamic as well. This work can help to clarify the difficulties encountered by 1D thermal-hydraulic system codes in the simulation of large break
LOCAs, idem specially in what concerns the disagreement in the quenching time found between simulations and experiments (e.g. LOFT-L2-5).

![Flow pattern in the downcomer during a LOCA](image)

**Figure 1: Flow pattern in the downcomer during a LOCA**

## 2 THEORETICAL APPROACH

Consider a duct of annular geometry, of internal and external radius $r_1$ and $r_2$ respectively. The coordinate system used is represented in Fig. 2; the coordinate origin is placed in the external wall; the three spatial directions are represented $y$ (radial), $z$ (axial) and $\theta$ (azimuthal).

In order to simplify the stratification analysis, only azimuthal velocity ($\theta$ direction) is considered in the fluid. The main assumptions of the approach are described next.

### 2.1 Assumptions

The annular geometry can be approximated by a cylindrical pipe in the hydraulic diameter approximation. This approximation is sufficiently accurate for engineering calculations.

(a) The flow is turbulent. This assumption is clearly correct because the Reynolds number is very high in the early stages of the kind of scenarios considered.

(b) In agreement with assumption (a) the turbulent profile approximation (i.e. logarithmic) can be used for the velocity.

(c) In agreement with assumption (a), (b), a bubble diffusion coefficient $D_b$ is defined using the turbulent core approach. In this paper a good approximation (for engineering calculations) is the Prandtl law.

(d) It is assumed that at a given radial position $y$ liquid and vapor velocities, $u_l$ and $u_v$, are equal, i.e. an homogeneous model is considered. This simplification, although unusual, is not far from reality in the first stages of this kind of scenarios and, in any case, it is sufficiently accurate to provide a good picture of the degree of stratification.

(e) Vapor is in form of spherical bubbles of radius $a$, all of them having the same size. This approximation clearly induce a deviation in the estimations because the phenomenon analyzed is very sensitive to the bubble size, as it will be deduced below.
In the same way, according to assumption (e), bubble coalescence phenomena are not considered.

The lift force caused by the velocity differential around bubbles is not considered because in the scope of this work it can be neglected when to inertial forces. A complete study of the effect of lift force on bubble stratification in pipes can be found in [1].

2.2 Equations

The momentum equation in cylindrical coordinates can be write as

\[
\rho \left( \frac{\partial u_r}{\partial t} + u_r \frac{\partial u_r}{\partial r} + u_\theta \frac{\partial u_r}{\partial \theta} + u_z \frac{\partial u_r}{\partial z} - \frac{u_\theta^2}{r} \right) = - \frac{\partial p}{\partial r} + \frac{1}{r} \frac{\partial}{\partial r} \left( r \frac{\partial u_r}{\partial r} \right) + \frac{1}{r^2} \frac{\partial^2 u_r}{\partial \theta^2} + \frac{\partial^2 u_r}{\partial z^2} - \frac{u_r}{r^2} - \frac{2}{r^2} \frac{\partial u_\theta}{\partial \theta} \]

(1)

If we have an azimuthal movement (i.e. \( u_r = 0 \)), and the remaining quantities are independent of \( \theta \); i.e., \( \frac{\partial u_r}{\partial \theta} = 0 \); in steady state the Eq.(1) becomes

\[
\frac{u_\theta^2}{r} = \frac{\partial p}{\partial r} \]

(2)

This Equation represents the centripetal acceleration, which is directed towards the center of curvature of the streamline and is associated with the change in direction of the velocity of a bubble.

Consider a bubble of radius \( a \) and density \( \rho_b \), surrounded by liquid of density \( \rho_l \) and viscosity \( \mu_l \); the azimuthal velocity being \( u_\theta \). The volumetric force experienced by the bubble is given by Eq. (2). Considering a spherical bubble, using the Archimedes law, the centripetal force, \( F_c \) acting upon the bubble is:

\[
F_c = - \frac{4}{3} \pi \rho_b a^3 (1 - \beta) \frac{u_\theta^2}{r} \]

(3)

where \( r^* \) is the distance to the center of the pipe (see Fig. 1) and \( \beta = \rho_b / \rho_l \).

The centripetal force, in first approximation, is equal to drag forces. (again, the spherical approximation is used):

\[
- \frac{4}{3} \pi \rho_b a^3 (1 - \beta) \frac{u_\theta^2}{r} = 6 \pi \mu_l a v_b \]

(4)

where \( v_b \) is the velocity of the bubble in \( y \) direction (see Fig. 1).

From Eq.(4):

\[
v_b = - \frac{2}{9} \frac{a^2}{\nu_l} (1 - \beta) \frac{v_\theta^2}{r^2} \]

(5)

where \( \nu_l \) is the kinematic viscosity (\( \nu_l = \mu_l / \rho_l \)).

According to assumption (c) above, a turbulent velocity profile is considered. The azimuthal velocity profile in the radial direction is given by:

\[
u_\theta = \nu_m + \frac{1}{\kappa} \ln \frac{r}{r_b} \]

(6)
where $u_m$ is the velocity of the two-phase mixture at the centerline of the equivalent pipe (according to assumption (a)); $y$ is the distance from the outer wall

$$y = r_h - r$$  \hspace{1cm} (7)

being $r_h$ the hydraulic radius, that for a concentric annular geometry is

$$r_h = r_2 - r_1$$  \hspace{1cm} (8)

$\kappa$ is a universal constant and $\tau$ is the two-phase wall shear stress. In Eq.(6) using the Reynolds analogy, the shear stress for a turbulent fluid flow surrounded by a solid can be expressed as the product of friction factor and the mean kinetic energy per unit volume. In circular pipes:

$$\tau = \frac{\rho u_m^2 \gamma f}{2}$$  \hspace{1cm} (9)

where $\gamma$ is the kinetic energy correction factor (in circular pipes $\gamma \approx 1$) and $f$ is the friction factor. Following the Blasius relationship:

$$f = 0.079 Re^{-0.25}$$  \hspace{1cm} (10)

A more accurate expression for the shear stress in a vertical concentric annular channel that takes into account the effect of volumetric expansion and annular geometry can be found in the work by J.A.Zarate et al. [2].

The bubbles concentration, $c$, in a given sector of the annular cylinder can be represented by a simple equation:

$$\frac{d}{dy} \left( D_b \frac{dc}{dy} \right) + \frac{d}{dy} \left( cv_b + S(y) + E(y) \right) = \frac{dc}{dt}$$  \hspace{1cm} (11)

where $D_b$ is the bubbles (turbulent) diffusion coefficient and $v_b$ is the velocity of the bubbles driven by inertial forces (as defined in Eq. (5). $S$ is the source (evaporation) term and $E$ the escape term.

At this point, it is supposed that, in a first approximation and steady state, the sum $(S(y) + E(y))$ can be neglected. With this approximation, in steady state Eq. (11) can be written as:
\[
D_b \frac{dc}{dy} = -c v_b
\] (12)

which represents that the bubble flow in the inside direction (due to inertial forces) is equal to the flux in the outside direction (turbulent dispersion forces).

According to assumption (c) above, the bubbles diffusion coefficient is:
\[
D_b = \kappa^2 y^2 \frac{du}{dy}
\] (13)

Combining Eq. (6), (12) and Eq.(13), and using the Reynolds analogy Eq.(9), and taking into account that the concentration of bubbles, \(c\), is proportional to the void fraction, \(\alpha\), the following expression is obtained:
\[
\frac{1}{\alpha} \frac{d\alpha}{dy} = \Psi \frac{a^2 u_m}{r^2 y} \left(1 + \frac{1}{\kappa} \frac{f}{2} \ln \frac{y}{r_b} \right)^2
\] (14)

where
\[
\Psi = \frac{2}{9} \frac{(1 - \beta)}{\nu f} \left(\frac{2}{\nu \beta} \kappa \right)^2
\] (15)

In Eq.(14), \(r^*\) is the actual curvature radius of the fluid. This issue is quite important because the inertial effect is not considered as a cause of two-phase stratification in the present approximations of thermal-hydraulic codes.

The real curvature radius can be written as \(r^* = r_2 - y\). At this point, to simplify the equations, in an actual downcomer geometry it can be considered valid the approximation \(r_2 \gg y\), so that the curvature radius is:
\[
r^* = r_2
\] (16)

It is useful now to perform a variable change \(s = \frac{y}{r_b}\).

Integrating Eq.(14) imposing the boundary limit \(\alpha = \alpha_m\) at \(s = 1\), the following expression is obtained:
\[
\ln \frac{\alpha}{\alpha_m} = \Psi \frac{a^2 u_m}{r_2} \left[\frac{f}{6 \kappa} \ln^3 s + \frac{1}{\kappa} \frac{f}{2} \ln^2 s + \ln s \right]
\] (17)

Eq.(17) is strongly sensitive to the bubble radius \(a\).

If a dispersed vapor droplet flow regime is considered, a good approximation for the bubble size is:
\[
a = \frac{2\sigma}{\rho_i (u_i - u_t)^2}
\] (18)

within \(1.0 \times 10^{-4} \ m < 2a < 3.0 \times 10^{-3} \ m\).

Fig. 3 compares the variation in the shape of the curves predicted by Eq.(17) with the bubble radius for typical values in a steam-water system:
\(\kappa = 0.4\) ; \(\theta = 0.0158 \ m^2/\text{sec}\) ; and velocity \(u_m = 91.5 \text{ m/sec}\) ; with \(r_2 = 1.787652 \text{ m}\) and \(a/r_2 = 3 \times 10^{-3} ; a/r_2 = 8 \times 10^{-3} ; a/r_2 = 1.5 \times 10^{-2} ; a/r_2 = 1.0 \times 10^{-2} \) and a friction factor \(f = 0.0055\).
The radial stratification, and the sensitivity of this phenomenon to the bubble size, can be clearly observed in Fig. 3. This influence of the bubble radius can be found in other stratification phenomena caused by velocity, e.g. the effect of lift forces described by Bankoff [1] (effect that, as has been established in assumption (h), can be neglected when compared with centrifugal forces in the considered ranges of velocity).

2.3 Mass Flow Drop in Radial Stratification

A number of variations of the completely separated flow model have been presented. Levy[3] has given a complete analysis of the annular-flow with various combinations of turbulent and laminar flow of the liquid and gas. Besides, taking into account the Martinelli-Nelson parameter [4], $\Phi_M^2$ that relates the pressure drop in a two-phase system and in a single-phase (liquid) one for the same mass flow and geometry, it is easy show that the relation between stratified and non-stratified mass flow results as below:

$$W = W_i \cdot \Phi_M^2 \cdot \alpha_i^2$$  \hspace{1cm} (19)

where $W$ and $W_i$ are the mass flow stratified and non-stratified respectively, $\Phi_M$ is the Martinelli-Nelson parameter and $\alpha_i$ is the liquid fraction ($\alpha_i = 1 - \alpha$).

Fig.4 shows the effect of radial stratification on mass flow $W$ compared with mass flow non-stratified $W_0$ according with Eq. (19) (left side axis). The right side axis represents the Martinelli-Nelson parameter.
Figure 4: Plot of Eq.(19) showing the impact of radial stratification in the mass flow.

3. CODE APPROACH

The analytical tasks related to safety issues in nuclear power plants are usually performed using system codes like RELAP5, TRAC, TRACE, ATHLET or CATHARE. All these codes have been assessed against a large number of experiments performed in both separate effect test and integral test facilities. The outcome of code assessment is powered by implementing good practices among users, training users, evaluating uncertainties and also performing the proper engineering judgements when needed.

Two different sets of code calculations are presented below. In both cases the scenario is related to the early phase of a Large Break LOCA where cyclon effect is supposed to occur and the codes have 3D features (TRACE and Relap5-3D).

3.1 Blowdown analysis using TRACE

A TRACE full vessel model of a 1000 MWe three-loop PWR has been used to simulate a (somewhat artificial) blowdown. The vessel model is part of an actual model for a PWR Spanish plant. In order to clarify the analysis and to underline the effect of radial stratification the simulated scenario consists in an isolated full PWR vessel, initially at nominal pressure (about 16 MPa) and with liquid at full power inlet temperature (about 290ºC), that suffers a 100% break in a cold leg. The conditions at the break are maintained at 1 bar.

Two calculations have been performed. In one of them the downcomer is simulated using the outer ring of the VESSEL component in TRACE. In the second calculation, the downcomer is subdivided radially, so that three rings are used to simulate it. The axial level analyzed is the volume number 11802 (that corresponds to an elevation slightly above the middle of the core); break connection is at axial level volume number 11700 (See Fig.5).

Fig.6 shows void fraction for the selected node in both cases. In the one ring option, the code cannot show any radial stratification. In this case the code does not predict any vapour for the first second (void fraction = 0) and later on some vapour generation due to depressurization and not to radial stratification.
In the 3 ring option, the very early phase is quite different. At this time azimuthal component of velocity is important and vapour appears in the node not only due to depressurization but also to radial stratification or cyclone effect.

After this first period stratification becomes less dominant for two main reasons. On the one hand velocity decreases and on the other hand its axial component becomes more important.

This behaviour, explained through the differences observed in the selected node, has a non negligible impact on break mass flow as it can be seen in Fig.7.

![Figure 5: Nodalization of a 3D downcomer using the code TRACE®.](image-url)
3.2 Helical flow simulation using Relap5-3D

Using RELAP5/3D®, a simulation has been developed for the upper part of the downcomer of a vessel of a 1000 MWe four-loop PWR.

Fig.8 is a top view of the nodalization used. Loop 1 has been divided in six (6) azimuthal and nine (9) radial nodes. The dimensions are full scale and the water flow is introduced by the one of the cold leg nozzles and exit by another one.

Table 1 shows pressure, temperature and void fraction for the first 10 seconds.

<table>
<thead>
<tr>
<th>Time (sec)</th>
<th>Pressure (Pa ×10^7)</th>
<th>Temperature (°K)</th>
<th>Void fraction</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.0</td>
<td>1.50</td>
<td>558</td>
<td>0.00</td>
</tr>
<tr>
<td>1.0</td>
<td>1.10</td>
<td>557</td>
<td>0.05</td>
</tr>
<tr>
<td>2.0</td>
<td>1.00</td>
<td>560</td>
<td>0.15</td>
</tr>
<tr>
<td>3.0</td>
<td>0.95</td>
<td>559</td>
<td>0.07</td>
</tr>
<tr>
<td>4.0</td>
<td>0.85</td>
<td>550</td>
<td>0.12</td>
</tr>
<tr>
<td>-----</td>
<td>------</td>
<td>-------</td>
<td>------</td>
</tr>
<tr>
<td>5.0</td>
<td>0.84</td>
<td>547</td>
<td>0.35</td>
</tr>
<tr>
<td>6.0</td>
<td>0.83</td>
<td>545</td>
<td>0.34</td>
</tr>
<tr>
<td>7.0</td>
<td>0.81</td>
<td>542</td>
<td>0.42</td>
</tr>
<tr>
<td>8.0</td>
<td>0.80</td>
<td>533</td>
<td>0.52</td>
</tr>
<tr>
<td>9.0</td>
<td>0.80</td>
<td>525</td>
<td>0.61</td>
</tr>
<tr>
<td>10.0</td>
<td>0.78</td>
<td>520</td>
<td>0.67</td>
</tr>
</tbody>
</table>

Fig. 8: Nodalization of a 3D downcomer using RELAP/3D®.

Fig. 9 compares the radial stratification (variation of the void fraction with the radial position of the nodes) calculated with RELAP/3D® and the equation proposed by authors Eq.(14-17). The agreement is quite good taking into account the assumptions of the theoretical development and the simplicity of scenario modelling. The discrepancy between the values obtained in the inner positions can be due to the fact that no source term has been considered in Eq.(14).

Fig. 10 is a snapshot showing void fraction $\alpha=0.9$ at a time of 4 seconds after the break in the considered nodes. As it can be seen the radial stratification is captured and has an impact on the phenomenology of the early phase of the Large Break LOCA.
4. **CONCLUSIONS**

The main conclusions of this work are:

1. The effect of radial stratification can be not neglected: A common practice extended in the simulation of downcomer in present one-dimensional thermal-hydraulic codes is the nodalization of the downcomer using cross-junctions, with the intention of obtaining an approximation to multidimensional flow. The present work shows that this approximation has to be improved as with the use of cross-junctions the dynamics of the two-phases under a strong azimuthal velocity cannot be captured. Even using 3D codes, the radial stratification due to inertial effects is not usually predicted.

2. The cyclone effect can give a good explanation of the delay in the quenching time in simulations with system thermal-hydraulic codes. In fact, the radial stratification, as has been demonstrated, has the effect of reducing the mass flow. This can reduce the mass flow, provoking larger quenching times, as observed in experiments as LOFT-L-25.
3. Theoretical developments, even when they include assumptions leading to simplification, are an interesting tool that could help engineering judgement devoted to explain discrepancies between code results and experimental data.

NOMENCLATURE

\( a \) = bubble radius
\( c \) = local bubble concentration
\( D \) = diffusion coefficient
\( f \) = friction factor
\( g \) = gravity acceleration
\( J \) = bubble flux in direction normal to wall
\( r \) = radius
\( s \) = \( y/r \), dimensionless distance from wall
\( u \) = velocity of two-phase mixture at a distance \( y \) from wall
\( u_m \) = velocity of two-phase mixture at tube centreline
\( v_b \) = transverse bubble velocity
\( y \) = distance from wall

Griegs

\( \alpha \) = void fraction
\( \alpha_m \) = void fraction at tube centerline
\( \beta \) = \( \rho_v/\rho_l \)
\( \kappa \) = universal constant

\( \mu \) = liquid viscosity
\( \nu \) = kinematic viscosity of liquid
\( \rho \) = density of two-phase mixture
\( \tau \) = wall shear stress in two-phase system
\( \gamma \) = Kinetic energy correction factor
\( \Phi_M \) = Martinelly-Nelson parameter
\( \Psi \) = constant defined by Eq.(15)

Subscripts

\( \theta \) = circumferential component
1,2 = inside,outside
H = hydraulic
l,v = liquid,vapor
sp = single phase
tp = two-phase
sp = single-phase
m = centerline
z = axial component
r = radial component

REFERENCES


Code Validation and Scaling of the ROSA/LSTF Test 3-1 Experiment

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ABSTRACT

Integral Test Facilities (ITFs) are one of the main tools for the validation of best-estimate thermal-hydraulic system codes. Experimental data are also of great value when compared to the experiment scaled-conditions in a full Nuclear Power Plant (NPP).

The Large Scale Test Facility (LSTF) is a full-height and 1/48 volumetrically scaled test facility of the Japan Atomic Energy Research Institute (JAERI) for system integral experiments simulating the thermal-hydraulic responses at full-pressure conditions of a 1100 MWe Pressurized Water Reactor (PWR) during small break loss-of-coolant accidents (SBLOCAs) and other transients.

The paper is focused on the simulation (with RELAP5/mod 3.3) and the scaling of the ROSA/LSTF Test 3-1 experiment (an Anticipated Transient Without Scram –ATWS- during 1 % cold leg small break LOCA) to the Spanish ASCÓ-2 NPP, a 3-loop 2940.6 MWth Westinghouse PWR. One of the aims of the paper is to check if the main phenomena in the test are actually reproduced in the plant calculation.

The work presented is part of a wider going-on activity at the Technical University of Catalonia related to the qualification of NPP nodalizations based on the scaling of experiments (LOBI, BETSHY, LOFT, PKL and LSTF) to full NPP conditions.

1. INTRODUCTION

Several safety activities have been performed during the last decades under the auspices of the OECD to develop and improve computer codes. Experiments performed at the LSTF test facility for the OECD/NEA ROSA project like Test 3-1 are a part of these activities. Furthermore scaled calculations are widely used in qualification procedures of NPP models [2]. Following the study reported in reference [3] LSTF Test 3-1 has been scaled to ASCÓ-2 NPP with the goal of improving the level of qualification of its thermal-hydraulic model.

1.1. High-Power Natural Circulation events

High-power events are transients with failure of scram in which core power decrease is due to negative reactivity feedback. Depending on the transient characteristics, this situation can lead to a relatively high core power during a long time.

Natural circulation occurs in transients with gradual loss of mass inventory (SBLOCA or LOFW –losses across pressurizer relief valve due to overpressure on primary system–). While there is high core power and water in the loops, vapor and liquid with high velocity exit from the vessel to the hot legs inducing supercritical flow during natural circulation. This phenomenon affects the coolant distribution due to counter-current flow limitation (CCFL)
during condensing reflux at the inlet of the steam generators and U-tubes which cause liquid accumulation in them –see figure 1-.

Break flow rate and liquid carryover into the pressurizer are affected by this phenomenon too.

![Fig. 1 Thermal-hydraulic phenomena during SBLOCA without scram](image)

1.2. **LSTF Test Facility**

LSTF (see figure 2) is an experimental facility designed to simulate a Westinghouse-type 4-loop 3,420 MWth PWR under accidental conditions. It is a full-height and 1/48 volumetrically-scaled two-loop system with a maximum core power of 10 MW (14 % of the scaled PWR nominal core power) and pressures scaled 1:1. Loops are sized to conserve volumetric factor, to take into account the different number of loops (2 in the facility and 4 in the reference plant), and to simulate the same flow regime transitions in the horizontal legs (respecting L/√D factor).

There is one SG for each loop respecting the same scaling factors. They have 141 full-size U-tubes, inlet and outlet plena, steam separator, steam dome, steam dryer, main steam line, four downcomers and other internals.

All emergency systems are represented and have a big versatility referred to their functions and positions. Many break locations (20) are available too.

LSTF test facility has about 1,760 measurement points that allow an exhaustive analysis of the tests. There are two types of data or measurement of interest: directly measured quantities (temperature, pressure, differential pressure), and derived quantities (from the combination of two or more direct measured quantities –coolant density, mass flow rate…-).

![Fig. 2 LSTF Test Facility](image)
1.3. Ascó NPP

Ascó NPP is a 2940 MWth PWR of Westinghouse design. It has a 3-loop configuration with the pressurizer connected to the third loop. There are two HPI and two LPI pumps. Each one injects in all three loops. Accumulators are connected to each cold leg and have independent lines.

![Ascó, Pressurized Water Reactor](image)

2. ROSA/LSTF TEST 3-1

Test 3-1 simulates a SBLOCA (break size of 1%) with scram failure and loss-of-offsite-power (HPI and LPI are unavailable). Due to the high-core power, supercritical natural circulation exists in the hot legs until the loops become empty. CCFL at the inlet of the steam generators and U-tubes during the two-phase natural circulation are objectives of study in this test.

2.1. Boundary conditions

The hardware configuration of LSTF is described in references [4]. Some important points for test 3-1 are the following:

- **Break assembly**: small break in the cold leg without pressurizer (1% of the scaled cross-sectional area of the reference PWR cold leg).
- **ECCs**: HPIS (High Pressure Injection System) and LPIS (Low Pressure Injection System) are unavailable simulating loss of off-site power.
- **Core power curve**: pre-determined from a previous volumetrically scaled analysis performed with SKETCH-INS/TRAC-PF, which reproduces the transient in a commercial PWR.
- **LSTF core protection system**: Core power is modified according to the maximum fuel rod surface temperature.
2.2. Initial Conditions

Initial steady-state conditions were fixed according to the reference PWR conditions. Because of the LSTF initial core power (14% of the scaled PWR nominal core power) core flow rate was set to 14% of the scaled nominal flow rate to obtain the same PWR temperatures, and secondary pressure was raised to limit the primary-to-secondary heat transfer rate to 10 MW.

2.3. Test phase

The transient is started opening the break at t = 0 s. After 20 seconds the scram signal is generated causing the closure of the MSIV and the stop valve (turbine trip); pressurizer heater is off, main feed water is closed and auxiliary feed water is started. Three seconds later, coastdown of the primary coolant pumps is initiated. Until 300 seconds, while there is high core power, secondary pressure rises over the specified setpoint causing the continuous opening of the SG relief valves and generating two-phase natural circulation in the primary loops. Between 300 and 1,600 seconds of the transient, the primary system is coupled with the isolated secondary system, which is depressurized with the cool auxiliary feed water that condenses vapor of the steam generators.

About 1,100 seconds, core liquid level starts to decrease rising the average temperature of the system. Then, the LSTF core protection system actuates decreasing the core power until the maximum fuel rod surface temperature is achieved. As a result of the low power, the primary pressure falls down below the secondary. About 2,100 seconds after the start of the transient, the accumulator injection system initiates causing a loop seal clearing in the loop without pressurizer 100 seconds later. At 5,547 seconds the break is closed and the transient finished.

The main events are described in table 1:

<table>
<thead>
<tr>
<th>Event</th>
<th>Time [s]</th>
</tr>
</thead>
<tbody>
<tr>
<td>Break</td>
<td>0</td>
</tr>
<tr>
<td>SCRAM signal: · Turbine trip and closure MSIV</td>
<td>20</td>
</tr>
<tr>
<td>· PZR heater off</td>
<td></td>
</tr>
<tr>
<td>· End of main feedwater and begin of auxiliary feedwater</td>
<td></td>
</tr>
<tr>
<td>Start of coastdown of primary coolant pumps</td>
<td>23</td>
</tr>
<tr>
<td>Primary coolant pumps stop</td>
<td>272</td>
</tr>
<tr>
<td>End of continuous opening of SG RVs. End of two-phase natural circulation.</td>
<td>About 300</td>
</tr>
<tr>
<td>Break flow from single-phase liquid to two-phase flow</td>
<td></td>
</tr>
<tr>
<td>Core liquid level starts to decrease (core uncover)</td>
<td>About 1,100</td>
</tr>
<tr>
<td>Core power decrease by LSTF core protection system</td>
<td>1,630</td>
</tr>
<tr>
<td>Max. fuel rod surface temperature</td>
<td>1,825</td>
</tr>
<tr>
<td>Primary pressure lower than SG secondary-side pressure</td>
<td>About 1,900</td>
</tr>
<tr>
<td>Initiation of accumulator injection system</td>
<td>About 2,100</td>
</tr>
<tr>
<td>Loop seal clearing only in loop without PZR</td>
<td>About 2,200</td>
</tr>
<tr>
<td>End of the transient</td>
<td>5,547</td>
</tr>
</tbody>
</table>
3. CODE INPUT MODEL DESCRIPTION

RELAP5/mod3.3 has been used to simulate ROSA/LSTF Test 3-1 experiment taking advantage of previous LSTF and Ascó NPP nodalizations modelled with RELAP5/mod3.2.

3.1. LSTF nodalization

The improvements performed to the original supplied input (see figure 4) were:
- Adjustment of the original RELAP5/mod3.2 to the newest version.
- Re-nodalization of the cold leg in the broken loop respecting the distance between pump, accumulator nozzle, break orifice and core inlet.
- Simulation of the bypass between upperplenum (UP) and upperhead (UH) with an annulus around the upper core support plate and a multiple junction reproducing the orifices at the UH bottom.
- New break unit reproducing pipes and junctions between the orifice assembly and the storage tank.
- Adjustment of the differential pressure around the loops and into the vessel to improve the steady-state conditions.

![Fig. 4 Original supplied input nodalization](image-url)
3.2. Ascó NPP nodalization

A scaling calculation was performed over a validated UPC (Universitat Politècnica de Catalunya) Ascó NPP input model. Two criteria were used in the scaling:

- Design of a conditioning phase to adjust the Ascó NPP steady state parameters to the ROSA/LSTF Test 3-1 initial conditions (plant working at 14 % of its nominal power).
- Calculation of a scaling factor to adjust those parameters and/or geometries with large discrepancies once steady state conditions are established (main feedwater, nuclear power, break junction, accumulator volumes...).

3.2.1. Conditioning phase

The conditioning phase is divided into three intervals (see figure 5):

- -2000 to -1800 seconds: Ascó NPP model working at 100% of its nominal power. A time-dependent-volume is used to fix the primary pressure and temperature conditions.
- -1800 to -1400 seconds: Nuclear power, primary mass flow rate, main feedwater and primary-to-secondary heat transfer are linearly decreased to 14 % of its nominal values. The time-dependent-volume is not used.
- -1400 to 0 seconds: Ascó NPP model works at 14 % of its nominal values. Other parameters like the pressurizer level and the secondary steam generator level are fixed to LSTF initial conditions during this period (see figure 6).

![Fig. 5 Conditioning phase](image-url)
3.2.2. Scaling factor

LSTF test facility has a scaling factor of 48 but this value varies depending on the geometry or on the plant parameter analyzed. It happens because many aspects like flow regime or heat transfer are taking into account when a test facility is designed. LSTF scaled parameters with less discrepancies regarding “48 factor” were chosen to calculate a comparable average LSTF-Ascó scaling factor obtaining a proposal value of 40.

Initial conditions (nuclear power, primary mass flow rate, main feedwater) and geometries (accumulator volumes) similar to its theoretical scaled value were not modified to respect its original plant design (see table 2; parameters are normalized to the scaled up steady state values of LSTF simulation). Auxiliary feedwater was scaled taking into account the scaling factor.

<table>
<thead>
<tr>
<th>Parameters not scaled</th>
<th>Ascó NPP RELAP simulation [Norm]</th>
</tr>
</thead>
<tbody>
<tr>
<td>Nuclear power</td>
<td>1.026</td>
</tr>
<tr>
<td>Primary mass flow rate</td>
<td>0.987</td>
</tr>
<tr>
<td>Main feedwater flow rate</td>
<td>1.029</td>
</tr>
<tr>
<td>Steam flow rate</td>
<td>1.025</td>
</tr>
<tr>
<td>Accumulators volume</td>
<td>0.975</td>
</tr>
<tr>
<td>Accumulators liquid volume</td>
<td>0.991</td>
</tr>
</tbody>
</table>

3.2.3. Break unit

The break unit was modeled using the scaling factor and reproducing the LSTF pipes and junctions between the orifice assembly and the storage tank. Henry-Fauske coefficients used and validated for the LSTF Test 3-1 post-test were maintained. To set the orifice section the initial scaled value was modified adjusting the break mass flow rate of the Ascó NPP model to the LSTF theoretical scaled plot (see figure 7). The cross section was set to $2.57 \times 10^{-3}$ m$^2$. 

Fig. 6 Pressurizer level
3.2.4. Initial conditions

Table 3 compares initial conditions between the Ascó NPP scaled model and the LSTF Test 3-1 simulation. Values are normalized to the measured Test 3-1 steady state conditions. Secondary side collapsed liquid level was modified to adjust the Ascó NPP secondary mass with the LSTF scaled secondary mass.

Table 3 Initial conditions.

<table>
<thead>
<tr>
<th></th>
<th>Ascó NPP RELAP simulation (1/40)</th>
<th>LSTF RELAP simulation (loops w/wo PZR)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core power</td>
<td>1.016</td>
<td>0.990</td>
</tr>
<tr>
<td>Hot leg temperature</td>
<td>1.0002</td>
<td>1.0 /0.9997</td>
</tr>
<tr>
<td>Cold leg temperature</td>
<td>0.997</td>
<td>1.001 / 1.0</td>
</tr>
<tr>
<td>Mass flow rate (x loop)</td>
<td>1.026 / 1.024</td>
<td>1.04 / 1.021</td>
</tr>
<tr>
<td>Downcomer-to-hot-leg bypass</td>
<td>-</td>
<td>1.001 / 1.001</td>
</tr>
<tr>
<td>Pressurizer pressure</td>
<td>1.007</td>
<td>1.003</td>
</tr>
<tr>
<td>Pressurizer liquid level</td>
<td>0.971</td>
<td>0.971</td>
</tr>
<tr>
<td>Secondary-side pressure</td>
<td>0.998</td>
<td>0.998 / 0.998</td>
</tr>
<tr>
<td>Secondary-side liquid level</td>
<td>1.225</td>
<td>1.003 / 0.998</td>
</tr>
<tr>
<td>Main feedwater temperature</td>
<td>0.998</td>
<td>1.001 / 0.999</td>
</tr>
<tr>
<td>Auxiliar feedwater temperature</td>
<td>1.003</td>
<td>1.0</td>
</tr>
<tr>
<td>Main feedwater flow rate</td>
<td>1.036</td>
<td>1.008 / 1.031</td>
</tr>
<tr>
<td>Accumulators pressure</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>Accumulators temperature</td>
<td>1.0</td>
<td>1.0 / 1.0</td>
</tr>
<tr>
<td>Steam flow rate</td>
<td>1.029</td>
<td>1.002 / 1.029</td>
</tr>
</tbody>
</table>
4. RESULTS

4.1. Test phase

Table 4 shows the chronology of the main events occurred in Test 3-1, comparing the experimental values with the UPC LSTF model and the Ascó NPP Scaled model. The comparison between two models is described as follows.

<table>
<thead>
<tr>
<th>Event</th>
<th>Experimental [s]</th>
<th>UPC LSTF model [s]</th>
<th>Ascó NPP Scaled model [s]</th>
</tr>
</thead>
<tbody>
<tr>
<td>Break</td>
<td>0</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>SCRAM signal: · Turbine trip and closure MSIV · PZR heater off · End of main feedwater and begin of auxiliary feedwater</td>
<td>20</td>
<td>20</td>
<td>20</td>
</tr>
<tr>
<td>Start of coastdown of primary coolant pumps</td>
<td>23</td>
<td>23</td>
<td>23</td>
</tr>
<tr>
<td>Primary coolant pumps stop</td>
<td>272</td>
<td>272</td>
<td>272</td>
</tr>
<tr>
<td>End of continuous opening of SG RVs, End of two-phase natural circulation, break flow from single-phase liquid to two-phase flow</td>
<td>About 300</td>
<td>300-400</td>
<td>300--400</td>
</tr>
<tr>
<td>Core liquid level starts to decrease (core uncovery)</td>
<td>About 1100</td>
<td>About 1100</td>
<td>1125</td>
</tr>
<tr>
<td>Core power decrease by LSTF core protection system</td>
<td>1630</td>
<td>1707</td>
<td>1705</td>
</tr>
<tr>
<td>Max. fuel rod surface temperature = 903 K</td>
<td>1825</td>
<td>1890</td>
<td>2030</td>
</tr>
<tr>
<td>Primary pressure lower than SG secondary-side pressure</td>
<td>About 1900</td>
<td>1875</td>
<td>2130</td>
</tr>
<tr>
<td>Initiation of accumulator injection system (primary pressure = 4.51 MPa)</td>
<td>About 2100</td>
<td>2180</td>
<td>2240</td>
</tr>
<tr>
<td>Loop seal clearing only in loop without PZR</td>
<td>About 2200</td>
<td>2898</td>
<td>3100</td>
</tr>
<tr>
<td>End of the transient</td>
<td>5547</td>
<td>5547</td>
<td>5547</td>
</tr>
</tbody>
</table>

Primary and secondary pressure have good agreement with experimental data until 2,200 s, when the initiation of the accumulator injection system causes some discrepancies on the primary pressure and the break mass flow rate (see figures 8 and 9). It is worth mentioning that there is an asymmetrical depressurization of the broken loop steam generator after primary pressure becomes lower than the SG secondary-side pressure. This phenomenon does not occur in the LSTF simulation (probably as a result of two symmetrical loops LSTF scaling). Asymmetrical termination of natural circulation is noticed in the Scaled Ascó NPP simulation too (see figure 10).
Fig. 8 Primary and secondary pressures

Fig. 9 Break mass flow rate

Fig. 10 Cold leg flow rate
Figure 11 shows a similar secondary mass behaviour with more losses in the Ascó NPP Scaled model during natural circulation due to the safety valves opening which does not occur in UPC LSTF model (see figure 12). Auxiliary feedwater is closed at 2,450 seconds to prevent water filling the SG separators (the volume factor of the SG is smaller than the Scaling factor applied to the auxiliary feed water).

Figure 13 shows how the Ascó NPP scaled simulation reproduces emptying of the core. Figures 14 and 15 show that the total primary system mass and the rod surface temperature have a quite good agreement with the LSTF simulation until the initiation of the accumulators injection.
Local phenomena: liquid accumulation in the U-tubes due to CCFL

Supercritical flow during the two-phase flow Natural Circulation induces liquid accumulation at the U-tubes during reflux and condensation because of the counter current flow limitation in the U-tube inlet and in the bottom of the inlet plenum (see figure 1).

RELAP5/mod3.3 reproduces the supercritical flow (Froude > 1) during the two-phase flow natural circulation (see the evolution of the Froude Number until 310 seconds in figure 16).

During the period of reflux and condensation (from 350 to 750 seconds, figure 16), fluid velocity becomes nearly zero, while there is still gas circulation. This phenomenon justifies the need of modelling counter current flow limitation and the associated U-tube liquid accumulation. Figure 18 shows that this local phenomenon is quite well reproduced by RELAP5/mod3.3.
Liquid accumulation in the U-tubes affects directly to the primary-to-secondary heat transfer. During reflux and condensation, water remains stagnant at the U-tubes and heat transfer decays abruptly (see Figure 19). It causes a sharp depressurization in the broken loop steam generator and a partial increase in the primary pressure (Figure 20). This phenomenon does not occur in LSTF simulation, probably as a result of a smaller U-tube liquid accumulation (Figure 18) which does not affect to the primary-to-secondary heat transfer.
5. CONCLUSIONS

Scaling of the ROSA/LSTF Test 3-1 experiment has been performed. Model predictions have been compared with a previous ROSA/LSTF Post-Test 3-1 calculation showing a quite good agreement between them. Several conclusions have been obtained from the study of local phenomena (U-tube liquid accumulation due to CCFL) and from several preliminary calculations:

• RELAP5/mod3.3 reproduces supercritical flow and liquid accumulation in the U-tubes during High-Power Natural Circulation.
• As there are some discrepancies between the LSTF/Reference PWR scaling factor and the volume factors of different LSTF parameters, it would be interesting to keep the original values of the plant if possible. Only in case of important differences with the scaled up values, they should be changed using the scaling factor.
• Stagnant liquid at the U-tubes during reflux and condensation can affect the primary-to-secondary heat transfer, and consequently, both system pressures.
• Other differences between both simulations, like the asymmetrical depressurization of the broken loop steam generator after decoupling or asymmetrical termination of natural circulation, could be the result of having two symmetrical loops in LSTF.

The results of this study show the capabilities of Ascó RELAP5/mod3.3 model to deal with phenomena involved in natural circulation at high power like most of ATWS scenarios related to commercial Nuclear Power Plants safety analysis.

ACKNOWLEDGEMENTS

This paper contains findings that were produced within the OECD-NEA ROSA Project. The authors are grateful to the Management Board of the ROSA Project for their consent to this publication and to Spanish Nuclear Safety Council (Consejo de Seguridad Nuclear) for funding UPC’s participation in the project.

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Design and safety issues
FIRST EXPERIENCE WITH THE CONSOLIDATION OF WWER REACTOR PRESSURE VESSEL KNOWLEDGE THROUGH A NEW METHOD

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ABSTRACT

One of today’s activities of the Joint Research Centre’s (JRC) Institute for Energy (IE) concerns data management and dissemination in nuclear safety. An “Online Data & Information Network” (ODIN) is set-up, which maintains one document database and four engineering databases. These databases aim to deploy networks for energy related research & development, specifically for nuclear energy and to provide the public experimental data of European projects on mechanical and thermo-physical material properties in comparison with international standards and recommendations. Due to its long lasting experience and being in a key position as regards web based d-base (e.g. ODIN), the IAEA for example has recently transferred the reactor surveillance data-base to the IE.

Lately, many stakeholders, such as Institutes, R&D Organisations, Regulators, Utilities, Governmental Organisations, have recognised the need for collecting, preserving, consolidating (validating), and disseminating nuclear knowledge (documents, competences and data), in order to make it easily accessible to future generations through modern informatics tools and training and education measures. A broad spectrum of components and technologies should be considered, i.e. reactor pressure vessel (RPV), piping, internals, steam generator, etc. regarding knowledge, material data and practices. In the long run, it will also support future decommissioning exercises of nuclear installations as a valuable knowledge source. In addition to the knowledge in each Member State, the IE produced a long standing record of results from its own institutional activities and even more through the participation to a large number of European Network partnership projects.

It is important, besides preservation, to consolidate the enormous amount of scientific results produced since. Therefore, the IE has developed a method for consolidation of nuclear knowledge. The method relays on the mobilisation of all identified leading experts
in the EU in re-evaluating old knowledge and consolidating what is necessary to create training materials for the new generations.

This method was applied for a pilot study for consolidating and preserving WWER RPV safety related knowledge, which is scattered in many countries and in different languages, facing a serious issue in terms of getting lost. This initiative could be the start of a wider Nuclear Knowledge Preservation and Consolidation activity. Experience gained from the first exercise will be presented in this paper.

1 INTRODUCTION

Nuclear knowledge had been build up continuously since the beginning of the last century. After Chernobyl in 1986 the public opinion changed leading to a gradual phasing out process of nuclear energy in several Member States. The interest of younger generations for nuclear studies dramatically decreased and nuclear education was abandoned by many engineering faculties. In the meantime the first generation of senior nuclear experts is retiring, creating an unbalance between incoming and outgoing flows of experts. This led gradually to a shortage of professional capacity and the increased risk of loosing valuable knowledge for the nuclear community. On the other hand, due to security of supply and climate change issues (green house mitigation measures) receiving more importance lately, a renaissance of nuclear power is ongoing. In order to avoid a possible loss of capability and knowledge in the EU action should be taken now preserving and disseminating it to the new generation.

This is confirmed through various statements released by national and international organisations, such as: the European Commission, which recognises the need to efficiently disseminate research results [1,2,3]; the International Atomic Energy Agency (IAEA), which has adopted in 2002 a resolution on “Nuclear Knowledge” emphasizing the importance of nuclear knowledge management, which was reiterated in subsequent years [4,5]; the Working Party on Nuclear Safety (WPNS), which recommends the improvement of exchange of nuclear safety information [6]; the Council of the European Union, which recommends in its conclusion regarding nuclear safety the compilation and exchange of any information regarding nuclear safety research [7]; the European Atomic Energy Community, which recommends that the expected fading out of nuclear knowledge qualifies for a potential European solution [8]; the OECD Nuclear Energy Agency, which adopts a statement about qualified human resources in the nuclear field due to the tremendous decline in students for nuclear studies in the last decades [9]; the European Nuclear Energy Forum (ENEF), which raised the idea of a European Nuclear Academy directly co-ordinated with the European Nuclear Education Network (ENEN) at its inaugural meeting [10].

Therefore, nuclear knowledge preservation and consolidation activities will be carried out with a strong political support. Also from the IT industry signals are pointing into the same direction. IBM’s Nuclear Power Advisory Council recommends strongly knowledge management in nuclear technology [11].

2 THE ONLINE DATA & INFORMATION NETWORK ODIN

One of today’s activities of the Institute for Energy concerns data management and dissemination in nuclear safety. An Online Data & Information Network (ODIN) is set-up, which maintains one document database (DoMa) and four engineering databases. These databases aim to deploy networks for energy related research & development, specifically for nuclear energy and to provide the public experimental data of European projects on mechanical and thermo-physical material properties in comparison with international standards and recommendations. Moreover ODIN manages six nuclear databases, which have restricted access. They cover: (i) data on High Temperature Reactor (HTR) Fuel Elements, (ii)
HTR graphite element data, (iii) data on safety of Eastern European Type Nuclear facilities, (iv) information on current research reactor safety assessment approaches, (v) data on hydrogen incidents and accidents, (vi) information on long term radioactive waste management, to enable co-operation and technology transfer for member states with small nuclear programmes.

The web-enabled document management system DoMa is designed to enhance the dissemination of information amongst R&D community sectors. Any type of electronic record (i.e. document, spreadsheet, graphic, etc.) may be stored. The facility combines open access to general information (i.e. title, author, abstract, etc.) about a particular record with controlled access to the actual files.

3 A NEW METHODOLOGY FOR KNOWLEDGE CONSOLIDATION

The Institute for Energy has developed a methodology for consolidation of nuclear knowledge (fig.1).

![Figure 1: Nuclear Knowledge Consolidation Circle](image)

The method relies on the mobilisation of all identified leading experts in the EU or beyond, re-evaluating old knowledge and consolidating what is necessary to create training and education material for new generations of nuclear engineers.
These experts are asked to provide the papers in their possession related to a specific nuclear expert field. Furthermore, they are asked to identify still more key-experts in that area. All papers are collected centrally and stored in a protected database DoMa, which is a document database located within ODIN. The papers are stored in PDF format and additionally have information about the title, authors, keywords and abstracts stored separately in MS Word for an easy search function implementation.

After the identification of some possible reviewers amongst the expert group, the subject is subdivided in subfields, in order to reduce the heavy work of review, summary and preliminary consolidation.

When the reviewers have finished their work, they prepare a summary report for their subfield, which is sent to all experts participating to the upcoming consolidation workshop. At the workshop the reviewers present their summary and conclusion on the subfield reviewed, which is discussed between the experts afterwards. The task of the chairman is to lead the experts to an as agreed as possible consolidation of the knowledge in each particular subfield. Finally, recommendations will be made at the end of the workshop, which could lead to further consolidation efforts in certain subfields or to a final consolidation document in others.

An additionally very important item in the consolidation process is the identification of commonly agreed (consolidated) open issues in the subfields. They complement the final goal of a State-of-the-Art report in the specific expert area.

4 THE PILOT STUDY ON WWER RPV EMBRITTLEMENT

There is a huge amount of information and knowledge in WWER Reactor Pressure Vessel (RPV) embrittlement available, either published or easily available, but also publications difficult to trace. Especially those are at risk of being dispersed or lost due to a series of factors, including:

- retirement of Senior Experts who were present at the time when most WWER Nuclear Power Plants were designed and put into operation,
- generational gap (due to years of decline in new constructions, only a limited number of people started their career in that area)
- non-electronic publishing in the past
- limited dissemination possibilities
- language (many non-English publications from Eastern countries)

Therefore, the Institute for Energy had decided jointly with some key experts to perform a pilot study using the previously described methodology for consolidation of WWER RPV embrittlement knowledge. In order to manage the review process easier and to distribute the burden, several expert sub-fields in the WWER RPV Embrittlement area were proposed and the papers were allocated accordingly.

Eight reviewers received preliminary between 7 and 21 papers in their field of expertise, in order to review the content and present it for discussion and consolidation to the WWER Reactor Pressure Vessel Embrittlement experts during a dedicated workshop in December 2007. Although more than 500 papers were already collected it was decided to start with a limited number as pilot study on voluntary basis.

The reports and presentations were requested to follow the below structure:

- per paper
  - paper title, author(s), reference
The general scope of the exercise is threefold depending on the time-spans available. For the short-term it is to reach consolidated conclusions during the workshop for the individual reviews after presentation and discussion. For the medium-term it is a consolidated review in the individual expert fields. For the long-term it is to prepare a State-of-the-Art report for the complete WWER RPV Irradiation Embrittlement expert field, incl. the history and reasons of the choices made (material, composition, etc.). The last general document [12] was produced in 1981 by Alekseenko, Amaev, Gorynin and Nikolaev, which is in Russian and needs upgrading.

In the brainstorming session of the workshop the predefined fields of expertise in WWER RPV Embrittlement were discussed and redefined as described in table 1.

Table 1: Subdivision of WWER RPV Embrittlement Expert Fields

<p>| | |</p>
<table>
<thead>
<tr>
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<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>1.</td>
<td>Start-of-Life toughness</td>
</tr>
<tr>
<td>2.</td>
<td>Irradiation shift prediction</td>
</tr>
<tr>
<td>3.</td>
<td>Property-property correlation</td>
</tr>
<tr>
<td>4.</td>
<td>Annealing and re-irradiation</td>
</tr>
<tr>
<td>5.</td>
<td>Material Factors</td>
</tr>
<tr>
<td>6.</td>
<td>Environmental factors</td>
</tr>
<tr>
<td>7.</td>
<td>Mechanism &amp; micro structural evolution</td>
</tr>
<tr>
<td>8.</td>
<td>PLEX Issues</td>
</tr>
<tr>
<td>9.</td>
<td>Surveillance</td>
</tr>
<tr>
<td>10.</td>
<td>Cladding</td>
</tr>
</tbody>
</table>

The recommendations of the workshop were the following:

- A revised subdivision of the field should be carried out with multiple allocation of papers
- A 2nd round of consolidation based on the pilot study with all traceable papers in the area should take place in 2008
- Unified keywords (terminology) for the search in the ODIN database should be drafted and agreed between the participants
- Knowledge preservation should be co-ordinated where possible with the IAEA efforts already done
- A complete State-of-the-Art book on WWER RPV embrittlement should be the final goal
5 LESSONS LEARNT

It is evident that a structural shortage of nuclear experts can not be solved by initiating such a project and vice versa, that initiating such a project cannot prevent the experts to retire with their specialist knowledge. The key-problem is the effect of these developments: a shortage of human resources qualified to do the work to be done. This shortage causes difficulties everywhere in the field and it will make it even more difficult to collect the knowledge of the retiring experts in a complete and systematic way.

The above described methodology applied to the knowledge of WWER RPV embrittlement has proven to be a right step in the right direction. The experts themselves, mostly working in the field already from the beginning of the nuclear area are proud of their work. They contributed in a very idealistic and positive way to this first circle of knowledge consolidation. Some even did the reviewing work in their spare time at home. The atmosphere during the discussions of the proposed consolidated conclusions per subfield was relaxed and constructive, as were the discussions on the consolidated open issues per subfield. The outcome was preserved in a summary record, which will be the base for the second/final consolidation circle/workshop. It was interesting to notice that the experts were agreeing on their consolidated conclusions and open issues on the basis of a limited number of papers per subfield. It was clear, that the complete knowledge and experience of the experts were taken into consideration making such a judgement and not only the knowledge by reviewing the limited amount of papers. This may be a very powerful tool in order to save time in the consolidation process. After the second circle of applying the consolidation methodology on the complete set of papers reviewed under each subfield heading a thorough analysis should give more information.

As further advantage of this consolidation methodology can be seen that the summary reports of the subfields can be published openly, pointing to all reference papers, but not violating intellectual property rights (IPR). Therefore, an as wide as possible dissemination to the interested public is guaranteed free of charge and to the benefit of new nuclear engineers.

It seems more than logic to continue applying this consolidation methodology to other fields of possible nuclear knowledge loss. This could be done not only for materials, but also for technologies, components, systems, etc.

6 CONCLUSION

In consequence of the increasing shortage in nuclear experts due to a generation change and as also recognized in recently organized conferences on nuclear knowledge management and preservation by the IAEA in 2007 [13] and the European Nuclear Society in 2008 [14], the European Commission’s Directorate General Joint Research Centre has increased its efforts in nuclear knowledge management and preservation. Its Institute for Energy has developed a consolidation methodology for knowledge preservation. A first pilot study on WWER RPV Embrittlement was carried out with encouraging results.

Some key aspects learnt from the pilot study were:

- The consolidation methodology proves to be efficient
- The participation of the experts to the consolidation is excellent
- Unified keywords are essential to trace the information needed
- The consolidated summaries per expert sub-field give a good general overview on results, open issues and key references without violating IPRs
At the end of 2008 a second consolidation workshop will be organized, reviewing the full set of papers available. It will be interesting to see, whether the consolidated conclusions and/or open issues will significantly change, as the experts used already their full knowledge in the first pilot study, not restricting themselves to the limited amount of papers to review. This could then be used systematically and a lot of precious time for the retiring generation of experts could be saved.

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Computational Fluid Dynamics (CFD) as a Tool for Prediction of Thermal Fatigue in T-junctions

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ABSTRACT

In T-junctions, where hot and cold streams mix, significant temperature fluctuations can occur near the walls. The temperature fluctuations in the bulk flow induce temperature fluctuations in the solid walls, giving rise to cyclic thermal stresses and the possibility of high-cyclic fatigue cracking. Multiple phenomena are taking place in the process: fluid flow, heat transfer in fluid and solid, and fatigue stresses in the solid. These phenomena can be measured experimentally, but also simulated using CFD to determine the velocity and temperature fields and structural analysis software for estimation of the thermal stresses in the walls. The present work focuses on the simulation of the dynamic part of these phenomena, using commercial CFD software. In spite of significant progress in the development of CFD, and ever-increasing computational power, the issue of turbulence modelling is still an active research area. Therefore, an important step in setting up the CFD simulation is the proper choice of the turbulence model. We present different options for this comparing the results obtained with the Shear Stress Transport (SST) model, Scale Adaptive Simulations (SAS) and Large Eddy Simulations (LES) against using a wire-mesh sensor placed across the stream. Our findings indicate that the LES approach has the highest potential of all tested strategies to predict the mixing phenomena in the T-junction, and hence the thermal stresses in the pipe walls.

1 INTRODUCTION

In T-junctions, particularly in the regions where hot and cold streams are not completely mixed, significant temperature fluctuations can occur near the walls. Such fluctuations may induce cyclic thermal stresses in the walls and may eventually lead to fatigue cracking. These problems were first considered in the context of liquid-metal fast breeder reactors (LMFBRs) in the 1980’s. In a liquid-metal reactor the problem is particularly pronounced, due to the high thermal conductivity of the liquid-metal coolant (sodium). Thermal fatigue phenomena were also observed in the secondary loop of the French Phénix prototype LMFBR and in a T-junction of the full-scale Superphénix reactor. The International Atomic Energy Association (IAEA) organised benchmark studies to analyse these events: several papers reporting the
activities are collected together in [1]. Thermal striping is an issue in Light Water Reactors (LWRs) as well. A few incidents of high-cycle fatigue have been observed in T-junctions, such as the one at Civaux 1 [2,3].

The high cyclic nature of these phenomena makes them difficult to monitor with conventional thermocouple instrumentation, due to the limited sensor response time. Yet reliable prediction of thermal fatigue loads is an important part of managing the risk. The temperature fluctuations at frequencies up to several Hz [4] caused by the turbulent thermal mixing, present the highest risk of wall thermal fatigue. Significantly higher frequencies than these appear not to pose a risk, as they are strongly attenuated by the thermal inertia of the pipe wall. Recent research activity in this area includes the joint US-Japanese programme [5,6,7], the programme by EDF [8], the experiments and benchmarks undertaken by Vattenfall [9,10] and the comprehensive, European 5th Framework Program THERFAT [11], each addressing the different issues involved, namely thermal-hydraulic analysis, material stress/fatigue analysis, crack initiation and propagation assessment. Present research is undertaken as a part of the Plant Life Management (PLiM) project [12] in Switzerland.

The data required for estimation of thermal striping can be obtained using empirical laws extracted from mock-up experiments and/or Computational Fluid Dynamics (CFD) simulations. In the latter case, the issue of the turbulence model to be used in the CFD simulations arises. The choice of the turbulence model has a strong impact on the accuracy of the results we can expect, but also on the computational time needed to get results. Fukushima et al. [13] have attempted to avoid the turbulence model completely by performing a Direct Numerical Simulation (DNS) of a T-junction of square ducts. They compared their results with corresponding experiments based on Laser Doppler Velocimetry (LDA) and thermocouples. They report encouraging agreement between simulations and measurements, but the Reynolds number was rather limited, around 4000, not relevant for industrial installations. A more cost-effective approach is not to attempt to resolve all the turbulent structures directly, as in DNS, but to perform some sort of averaging of the governing equations to get rid of certain scales of motion. In the case of such averaging, a turbulence model must be introduced to model the effect of these neglected scales of motion. The most cost-effective class of models is represented by the RANS (Reynolds Averaged Navier Stokes) approach. These are based on the principle of time-averaging over all the turbulent scales. Depending on the formulation, these models result in additional transport equations for the turbulence quantities (such as turbulent kinetic energy $k$, or individual Reynolds stresses) and turbulent length scale (or dissipation $\varepsilon$). Industry-wise, the most popular models are the two equation $k$-$\varepsilon$ model [14] and different versions of the $k$-$\omega$ model [15]. However, no matter which RANS model is used, it provides only time-averaged results, hardly useful for fatigue analysis, which requires unsteady input.

In contrast to the RANS approach, the governing equations may be averaged over small regions of space, while preserving their unsteady nature. In such an averaging, more kinetic energy is resolved in a simulation, requiring less from the model. This is idea behind Large Eddy Simulations (LES), where averaging is conveniently referred to as filtering, while the small regions of space are conveniently taken as computational cells. Large scales of motion (large eddies) are resolved on the grid, while the scales smaller than the local grid size are modelled using a suitable Subgrid Scale (SGS) model. Away from solid boundaries, LES appears trustworthy, even with very simplistic SGS models, such as that of Smagorinsky [16]. In the wall regions, pure LES may be inaccurate, since there are no large eddies there. The LES technique has been applied to analyse the LMFBR problem (Phénix) by Gelineau et al.
[17], and by Hu and Kazimi [18, 19] to simulate T-junction mixing experiments carried out at Hitachi. Their results have demonstrated the applicability of the LES approach to this problem. However, it was still concluded that LES could still be computationally too expensive for full, industrial scales, and that simplified approaches, for example based on average temperature gradients, should be investigated. In the frame of the THERFAT project, both LES and k-ε based formulations have been used [11], and results compared against experimental measurements obtained from T-junction tests performed at Siempelkamp in Germany. Though reasonable agreement was obtained between experimental results and CFD simulations as far as the basic phenomena are concerned, it was noted that 4 s real time required 6 weeks of CPU time for LES. The numbers are typical for LES simulations. It was also concluded that, in order to provide quantitatively reliable input data for stress and fatigue calculations, further validation activities are required.

To take advantage of LES away from the boundary and to obtain trustworthy results in the near-wall regions, and hopefully reduce the computational time, a number of hybrid approaches (between RANS and LES) have been proposed. An example is known as Detached Eddy Simulation (DES), and leads to considerable savings in CPU time [20] over LES. The disadvantage of DES is its dependence on the grid size. Namely, it switches from RANS to LES region based on the local grid size, making the model inherently grid-dependent. A more sophisticated approach is the Scale-Adaptive Simulation (SAS) [21]. This is a hybrid approach similar to DES, but operates without an explicit grid dependency. As way of illustration, Fig. 1 shows how each approach to turbulence modelling is expected to capture an instantaneous velocity signal, produced experimentally or by using Direct Numerical Simulation (DNS).

As a part of our activities in the PLiM project, it was necessary to explore the relative merits of the various turbulence modelling approaches, and to chose the most appropriate one for providing reliable thermal hydraulic data for the subsequent stress analysis and fatigue studies. It was decided to compare our simulations with the measurements of the new T-junction experiment performed at PSI, using advanced instrumentation, namely Wire-Mesh (WM) sensors. The experimental set-up is described in the next subsection.

Figure 1: Illustration of velocity signals from various turbulence modelling approaches.
2 EXPERIMENTAL SET-UP

Using the analogy between turbulent mass and thermal transport and mixing, isothermal experiments have been performed using regular tap water and demineralised water. The set-up consists of a horizontal T-junction geometry of Plexiglas pipes of 50 mm inner diameter. Regular tap water flows in the longer pipe (1.5 m) and demineralised water in the shorter, branch pipe (0.5 m). A photo of the test section is given in Fig. 2. The two streams join and mix at and after the T-junction, and the mixture is drained through a flexible hose shown on the right side (green). Close to the inlets of both pipes, honeycombs are placed to straighten the flow. The honeycombs have a cell size of 3.5 mm and a length of 60 mm. Both pipes are sufficiently long to ensure a developed flow profile as the fluids arrive at the T-junction, giving well-defined boundary conditions for the CFD simulations. In the arrangement used in this work, the instrumentation consists of two wire-mesh (WM) sensors, placed one behind the other, 51 mm downstream of the junction.

Temperatures in both streams are measured by thermocouples placed downstream the honeycombs. Two electromagnetic flow meters are installed in auxiliary piping to measure the flow rates through the pipes.

The WM sensors actually measure the electrical conductivity of the fluid, and even the small conductivity difference between the tap and de-mineralized water is sufficient for the measurement. The sensor consists of two mutually orthogonal grids of parallel wires, spanning over the cross sections (Fig. 3). One set of wires transmits electrical signals, while the other receives them in order which ensures that the entire cross-section is measured. The current at a receiver wire resulting from the activation of a given transmitter wire is a measure of the conductivity of the fluid between the two wires. After scanning all transmitter and receiver wires, a matrix of measured values is stored in the computer, representing the complete two-dimensional conductivity distribution in the cross-section of the sensor.
The WM sensors used for this study have 16x16 wires, constituting a grid of 236 measurement points (from the 256 actual combinations, a few points are missing in the corners due to the circular pipe geometry). The pitch of the measurement grid, which also defines the spatial resolution of the measurements, is 3 mm. The time resolution of the measurement is 10 000 frames per second.

Although several experiments have been carried out by varying the flow rates in the run and branch pipes, for the present study we have used the case where both pipes have the same average inlet velocity of 0.5 m/s, tap water is fed into main branch and demineralised water into side branch.

3 CFD SIMULATIONS OF PSI MIXING T-JUNCTION EXPERIMENT

Following our first exploratory CFD simulations of the T-junction [22], a more systematic study was started in which several turbulence modelling approaches were compared. GRIDGEN [23] mesh-generation software was used to create the grids used in this study. We paid special attention to use only hexahedral cells for the grids in order to decrease the amount of numerical diffusion [22] and to improve the convergence rate of the solution procedure.

For RANS and SAS, a grid with approximately 800 000 cells was created, with cells clustered towards the junction, providing finer resolution in the region where most of the mixing is taking place. In contrast to RANS and SAS, the grid for LES does not feature grid refinement towards the junction, but is fine enough throughout the domain not to hinder the natural development of turbulent structures. Obviously, being non-stretched, the grid for LES contains more cells, slightly above 2 million. In the grid for the pipe cross-section (see insert to Fig. 4), which is the same for all simulations, cells are concentrated toward the pipe walls to capture the effects of wall shear more accurately. For all simulations, velocity boundary conditions are set at the inlets to the main and branch pipes, and a constant pressure at the outlet.
The passive scalar, representing the mineralized (high conductivity) water, is governed by the following transport equation:

\[
\frac{\partial \Theta}{\partial t} + \mathbf{u} \nabla \Theta = \nabla \left( D + \frac{\mu_t}{S_{c_t}} \right) \nabla \Theta, \tag{1}
\]

where \( \Theta \) is the local mass fraction of tap water, \( \mathbf{u} \) is the velocity vector, \( D \) is the diffusion coefficient, and \( \mu_t \) and \( S_{c_t} \) are the turbulent (eddy) viscosity and turbulent Schmidt number, respectively. Equation 1 is integrated together with the equations of motion and turbulence as part of the solution procedure. The scalar is defined as zero for the water entering the branch pipe and unity for that entering the main pipe for convenience.

### 3.1 RANS Calculations Using CFX-11

We have used CFX-11 to obtain steady-state solutions using the RANS approach. Following earlier experience [22], the SST \( k-\omega \) model [24, 25] was chosen as the most appropriate for the current work. The SST \( k-\omega \) model can be used as a *Low Reynolds Number* model without the use of extra damping functions [25] for the near-wall regions. However, this also means that the governing equations have to be integrated down to the solid walls, thus increasing the cost of simulations. This model switches to a \( k-\varepsilon \) formulation in the free-stream, and thereby avoids the common \( k-\omega \) problem of being too sensitive to the inlet free-stream turbulence properties. The SST \( k-\omega \) model is particularly suitable for flows with adverse pressure gradients and separated flows, the latter are featured in the present application.

The length of the inlet pipes is long enough (6 hydraulic diameters) for fully developed turbulent velocity profiles to be established before arriving at the T-junction. Therefore, at both inlets we prescribe uniform velocity profiles and set turbulence intensity levels at 10%.
Figure 5: Velocity contours in the horizontal mid-plane obtained with RANS.

Contours of velocity magnitudes in the horizontal mid-plane are shown in Fig. 5. The centre-line velocity in the inlet sections increases slightly as a result of the development of the wall boundary layers. On the upstream side of the junction where two streams meet, a small stagnant region exists. Downstream of the junction, a strong recirculation region exists and the WM sensor is well positioned to measure it. The flow accelerates markedly in this region.

Figure 6: Scalar contours in the horizontal mid-plane obtained with RANS.

Figure 6 shows the contours of the passive scalar (normalised conductivity) in the same horizontal mid-plane. One sees that the most intensive mixing occurs in the recirculation region just after the junction, and again the WM sensor is ideally placed to measure it. The zone of mixing grows only gradually after the recirculation region, and some unmixed fluid remains even at the outlet, which is 6 diameters downstream of the junction.
Conductivity contours in the WM sensor plane are displayed in Fig. 7. There is generally good qualitative agreement between calculated and measured values. Moreover, the occurrence of the kidney-shaped region (lighter blue in Fig. 7) is present in both simulation and in measurement. This region is created by the reverse flow in the recirculation region. The most notable disagreement between simulation and measurement is the width of the central mixing zone, which is significantly broader in the experiment than predicted by the code. This point is clearly illustrated in Fig. 7c, which compares the measured and calculated conductivity profiles along the mid-line of the measuring plane. One might add more diffusive mixing by decreasing the turbulent Schmidt number (see Eq. 1), but such an approach is not based on sound physical reasoning.

Figure 7: Scalar conductivities: a comparison of predictions obtained using RANS modelling and measurement: predicted (a) and measured (b) contours in the plane of the WM sensor; and (c) along the mid-line.

3.2 SAS Calculations Using CFX-11

Although the steady-state calculations reported in the previous section show good qualitative trends, they do not give us the unsteady information we need to estimate thermal cycling. Our simulations have to provide the amplitudes and frequencies of coherent turbulent motions. Simply solving the RANS turbulence equations in an unsteady mode is not the answer here, since the simulations would simply converge toward the steady results reported in the previous section. An approach which is inherently unsteady, such as DES or SAS is needed. SAS is the preferred approach, since it is not grid-dependent, as is DES. The SAS
approach allows the use of much coarser grids than those used in traditional LES computations, and is therefore computationally less expensive. Although the model is in continuous development, it is worth exploring in the context of the T-junction simulations here because of the potential savings in computational times. We performed SAS with CFX-11 software, using the same grid as for the RANS calculations. The difference is, however, that SAS simulations have to be performed in unsteady mode. In this example, we set a time step of 0.001 s, which resulted in a root-mean-square Courant-Friedrich-Lewy (CFL) number of 0.6. We estimate that this value provides sufficient time-resolution. The maximum values of the CFL (around 2.6) were observed at the pipe connection, where the two streams mix, but is confined to a very small portion of cells, and so can be tolerated. We ran SAS following the time history of the scalar value at the centre of WM sensor plane. As soon as the solution established a periodic behaviour, we stopped the simulation and performed averaging over four periods of the solution.

Performing the simulation in unsteady mode requires a considerable increase in computational time over RANS simulations: 424 CPU hours for 4.5 s of physical time, in contrast to RANS calculation, performed using the same grid, which completed in about 3.5 CPU hours, meaning that SAS increased the computational effort by a factor of 120.

![Figure 8: Snapshots of the (a) velocity magnitude and (b) conductivity contours obtained using SAS](image)

Contours of velocity magnitude and conductivity are shown in Fig. 8. The small and large separation zones upstream and downstream of the junction are also present in SAS, as they were with the RANS simulation. That is to be expected, since SAS switches to RANS mode in the regions of the flow where not much is happening. Further downstream from the
junction, where the most intensive mixing takes place, the structures obtained from the solution are also unsteady. The dynamics of these unsteady structures in the mixing region will be an important factor in assessing the thermal cycling associated fatigue.

Figure 9: Average conductivity profiles at the mid-line of the measuring plane: a comparison of results obtained using RANS and SAS.

Figure 9 shows the comparison of the time-averaged conductivity profiles in the mid-line of the WM sensor plane. The similarity of the results is quite noticeable, with both simulations suffering from too sharp mixing region. Moreover, there is too much conductivity in the region affected by the double vortex behind the junction. Both of these effects are a consequence of underestimated turbulent mixing in the simulations.

Figure 10: Conductivity trace at a typical point in the mixing region in the plane of the sensor: (a) time dependence; (b) frequency dependence.

The time history of conductivity for a typical point in the mixing region of the junction is shown in Fig. 10. At this position, the mean conductivity is about 0.7, with a superimposed oscillation of amplitude 0.12 (Fig. 10a). The Fourier transform of the signal (Fig. 10b) indicates the presence of modes of frequencies 1-4 Hz, with \( f = 4 \text{ Hz} \) being the dominant frequency. Although such frequencies, around few Hz, are potentially the most dangerous for temperature-oscillation-driven thermal fatigue, such a clearly defined dominant frequency
was not present in the measured data. Consequently, some doubts remain whether SAS, though providing information on the amplitudes and frequencies of the scalar oscillations that are needed for the associated fatigue analysis, is producing totally trustworthy results in this simulation.

### 3.2 LES calculations using FLUENT 6.3

As SAS left us with some doubts, as the next step we decided to perform LES of the mixing in the T-junction. It is important to remember, though, that LES requires more than just a finer grid and a smaller time step to be successful: advection terms must be discretized with a central differencing scheme, since upwind schemes, even if of higher order, smear out the turbulent structures at the smallest scales. Unfortunately, central differencing is notoriously unstable, and small wiggles in the flow can be amplified and ultimately destroy the solution. Special attention has to be paid in the modelling of the near-wall regions. Essentially, the same two approaches used in RANS are possible: integrating the governing equations down to the walls, taking care that the grid is fine enough to resolve the near-wall region in sufficient details, or bridging the near-wall region altogether using a wall-function. In the present simulations, the Reynolds number was low enough to permit the former approach, i.e. integration down to the wall. In this respect, the application of the simple SGS model (such as the Smagorinski model) is not adequate, and a model which damps the eddy viscosity near the walls is needed. Another requirement which needs to be fulfilled for a successful LES is the proper prescription of the inlet boundary conditions. Obviously, an unsteady velocity field at the inlet is needed, but it must mimic the true turbulent field. This means that simple random fluctuations are not sufficient since they are generally do not conserve mass and do not have the correct energy spectra.

Unfortunately, at the time of conducting the analysis presented in this paper, CFX-11 [26] was not ready for LES: it did not have the advection schemes required by LES, nor did it have a method for prescribing the inlet velocity profiles for LES. On the other hand, significant advancements in LES modelling have been implemented in the FLUENT [27] code and they are now included as standard in Versions 6.1, 6.2 and 6.3 of the code. The bounded central-differencing (BCD) scheme is much more robust than the standard central-differencing scheme, and yet it features much low numerical diffusion than the high-order upwind schemes. Further advancements in FLUENT also involve better near-wall modelling by the introduction of the WALE (Wall-Adapting Local Eddy-Viscosity) model [28]. WALE model keeps the algebraic simplicity of the standard Smagorinsky model but takes into account wall-damping effects automatically. FLUENT also has advanced options for specifying the turbulence conditions at inlets. The option used here is a stochastic vortex method in which the inlet turbulence is mimicked using the velocity field induced by many quasi-random point vortices [29].

As mentioned above, a finer mesh has been used for LES than for RANS and SAS. The cross-sectional grid is the same, but in contrast to the grid for RANS and SAS, the LES grid is uniform in stream-wise directions, not to disrupt the natural development of turbulent structures. The LES grid has 2 086 000 cells. Near-wall resolution in wall units was $y^+ = 0.25$ for the inlet branches and $y^+ = 0.50$ for the outlet branch. The WALE SGS model was used for modelling the small-scale turbulence.

Since a finer mesh was employed for LES than for SAS, the time step had to be reduced accordingly. With the aim to keep the same CFL number as used before, the time-step for the
calculation had to be reduced to $\Delta t = 0.0001$ s. The calculation was run to 5.2 s, corresponding to approximately five flow-through times, and statistics gathered over the period 3.2 s to 5.2 s, approximately two flow-through times. The CPU time for gathering the statistics was about 126 hrs on a 16-processor machine, which, if linear speed-up is assumed, equals over 2000 hrs on one CPU. That is, LES is a factor of 4 more times expensive than an equivalent SAS computation, and 480 times more expensive than RANS.

Figure 11: Time-averaged scalar conductivities: a comparison of predictions obtained using LES modelling and measurement: predicted (a) and measured (b) contours in the plane of the WM sensor; and (c) along the mid-line. The red circle shows the position of the monitoring point.

Contours of time-averaged conductivity are displayed in Fig. 11. The qualitative agreement is much closer to measured data than RANS and SAS, which is apparent from the broader mixing region. The kidney-shaped region of high conductivity liquid is also closer to measurement than seen with RANS and SAS. Figure 11c shows the conductivity profile along the centre-line of the sensor. The LES curve fits the measured data much better than with RANS or SAS. It is worth noting that even the conductivity in the recirculation region affected by the vortex behind the junction is predicted by LES in closer agreement to experiment than for RANS and SAS.
The time history of the conductivity at the central point of the WM sensor (shown by the red circle in Fig. 11a), is shown in Fig 12. This position is close to the edge of the mixing zone, and the signal exhibits large variations in conductivity. The Fourier transform of the signal (Fig. 12b) shows no dominant frequency, in contrast to the result obtained by SAS, but is in line with the findings from the experiment.

An important achievement is that cells have been small enough close to the pipe walls (within 20 µm) to allow integration of the governing equations down to the pipe wall without the need to use a wall function. Wall damping effects are accounted for automatically by the WALE model. However, we might be constrained to use the wall function approach in the future, should we want to simulate higher Re number flows. The use of wall functions for T-junctions would need validation too.

Figure 12: Conductivity trace at the centre of the sensor: (a) time dependence; (b) frequency dependence

4 CONCLUSIONS

Numerical simulations of turbulent mixing in a T-junction, and associated experiments, have been presented, with the aim to select the most suitable turbulence modelling approach for these phenomena. Using the analogy between turbulent mass and thermal transport, WM sensors have been used to measure the concentration of tap water with de-mineralized water downstream the T-junction. The high spatial and time resolution of the sensors has enabled maps of the time-averaged values and the fluctuations of the mixing scalar in the pipe cross-section to be drawn. State-of-the-art CFD simulations of one of the T-junction tests have been performed using three different turbulence modelling approaches: RANS, SAS and LES, using two commercial CFD packages: CFX-11 and FLUENT 6.3.

All three modelling approaches identified characteristic flow regions for mixing in the T-junction, in particular the secondary flow effects due to the presence of a double vortex behind the junction. However, RANS and SAS predicted a lower degree of mixing than the experiments, characterized by narrower mixing region between the streams and more demineralised water in the re-entrainment region affected by the double vortex. LES, however, predicted both the width of the mixing region and the conductivity profile in the re-entrainment region more in accord with the measurement data.
RANS-based solutions, which only give time-averaged information, do not provide the needed data for the subsequent stress and fatigue analysis. Therefore, approaches such as SAS or LES, which are based on resolving the governing equations in unsteady form, must be used. Our SAS simulations predicted a dominant frequency of 4 Hz for the case considered here, which was not seen from the measured data. The LES results, however, were in closer accord with the experimental data, and it exhibits no dominant frequency. Therefore, in spite of the higher computational cost, we envisage LES to be more appropriate turbulence modelling approach for simulating mixing in T-junctions.

Further study is required, particularly on the sensitivity on inlet turbulence level and more detailed comparison of unsteady data with WM sensor measurements (ongoing). In spite of the advancement of computer hardware and numerical algorithms, LES is remains expensive. For the considered case, one week on 16 CPUs of a state-of-the-art, Opteron-based, LINUX cluster was needed to simulate 1 s of physical time.

REFERENCES
Design Considerations for Avoiding Instability in Two-Phase Natural Circulation Systems

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ABSTRACT

Design of two-phase natural circulation systems is somewhat complicated due to the existence of two constraints, i.e. stability and critical heat flux (CHF). Depending on the system geometry and operating conditions, the design of natural circulation systems can be stability controlled or CHF controlled. Design procedures for two-phase natural circulation systems are not readily available in the open literature. This paper describes a rational design procedure based on the state of the art knowledgebase available.

1 INTRODUCTION

Natural circulation systems (NCS) are essentially fluid filled systems employed to transport heat from a source to a sink. In such systems, the fluid circulation is caused by the buoyancy force which in turn is caused by the density gradients resulting from the heat transport process itself. The heat transport is achieved noiselessly without the use of fluid moving machineries and is considered to be more reliable as its functioning is based on a natural physical law which is not expected to fail. Due to these advantages these systems find extensive applications in industry. Traditional applications include thermosyphon reboilers, steam generators in nuclear power plants and natural circulation boilers (NCB) in fossil fuelled power plants. Emerging applications include cooling of computers and electronic components which essentially use mini loops capable of transporting a few tens to a few hundred Watts and contain only a few cubic centimetres of fluid. In contrast, the NCS used in power plants has a few hundred cubic meters of fluid and is capable of transporting a few thousand MW of power. Two-phase natural circulation is the proposed coolant circulation mode in many innovative nuclear reactors such as AHWR, ESBWR and VK-300.

Two-phase natural circulation systems are susceptible to several kinds of instability. The instabilities could be static or dynamic and influenced by a large number of geometric and operating parameters. A common geometric characteristic of most industrially important two-phase NCSs is that they consist of a large number of parallel channels. Typical examples are BWRs, NCBs and thermosyphon reboilers. Parallel channel instability is a characteristic of such systems. An acceptable design must avoid all instabilities. This is done by identifying the controlling instability (instability with least stable zone) and restricting operation within its stable zone. Besides instability, avoidance of the critical heat flux (CHF) is also essential for the safe operation of two-phase natural circulation systems used in nuclear and thermal
power plants. Stability and CHF often have opposing parametric trends. For example reduction in inlet subcooling enhances stability whereas it reduces CHF. With such conflicting requirements, it is challenging to design two-phase NCSs meeting both the stability and CHF margins. By design, it is usually meant as the establishment of the geometry (of the complete system including riser height, inlet and exit loss coefficients) and operating conditions (power, pressure and inlet subcooling or sink conditions) of the NCS to achieve a specified heat transport capability.

2 STEADY STATE CHARACTERISTICS OF TWO-PHASE NC SYSTEMS

Knowledge of the steady state characteristics is essential for the design of NCSs. Since the heat transport capability depends strongly on the flow generated, the most important steady state characteristic that needs to be established is the flow rate achievable in a NCS of specified geometry and operating characteristics. For simplicity we consider a uniform diameter loop (UDL) with the length scale shown in Fig. 1 which was extensively investigated experimentally by Rao et al. [1].

Fig.1: Schematic of experimental loop

Assuming balance of the driving buoyancy force to the retarding frictional forces at steady state and using \( f = p/\text{Re}^b \), the following equation was derived to estimate the steady state flow rate in a UDL [2] and [3].

\[
W_{ss} = \left[ \frac{2}{p} \frac{D^b \rho_f^2 \bar{\rho}_h \rho \tilde{A}^{2-b} Q \Delta \varepsilon}{\mu_f N_G} \right]^{\frac{1}{3-b}}
\]

(1)

where \( \bar{\rho}_h = \frac{1}{v} \left( \frac{\partial v}{\partial h} \right)_p \) and \( N_G = \frac{L}{D} \left[ \bar{e}_{\text{eff}} \left( \bar{e}_{\text{eff}} \right)_p + \phi_{\text{LO}}^2 \left( \bar{e}_{\text{eff}} \right)_h + \phi_{\text{LO}}^2 \left( \bar{e}_{\text{eff}} \right)_c \right] \)

(2)

with \( \phi_{\text{LO}}^b = \frac{\rho_f}{\rho_c} \left[ \frac{1}{1 + x_c \left( \frac{\mu_c}{\mu_f} - 1 \right)} \right] \) and \( \phi_{\text{LO}}^b = \frac{\rho_f}{\rho_m} \left[ \frac{1}{1 + \frac{x_c}{2} \left( \frac{\mu_c}{\mu_f} - 1 \right)} \right] \)

(3)
This correlation was derived using the basic definition of $\phi_{LO}$ and McAdam’s model for two-phase viscosity [4]. It may be noted that for single-phase flows $\phi_{LO}^2 = 1$ and equation (2) reduces to that applicable to single-phase natural circulation. Also, similarity in the definitions of the thermal expansion coefficients $\beta_h = (1/\nu)(\partial \nu/\partial h)_p = (1/(\nu C_P)(\partial \nu/\partial T)_P = \beta / C_p$) and $\beta_T = (1/\nu)(\partial \nu/\partial T)_p$) valid for two-phase and single-phase flows respectively may be noted. Also, if local losses are negligible, then the effective lengths in Eq. (2) become the corresponding physical lengths. Gartia et al. [3] also checked the adequacy of this correlation by comparing it with test data from several loops and found that the data are within $\pm 40\%$ of the above correlation. The loops considered differed in diameter but had the same length scale as in Fig. 1.

2.1 Two-Phase NC Flow Regimes

The steady state flow in two-phase loops is significantly influenced by the pressure, power and loop diameter. Based on the nature of the variation of the steady state flow with power, three different natural circulation flow regimes can be observed for two-phase loops [4]. These flow regimes are designated as gravity dominant, friction dominant and the compensating regimes. In a natural circulation loop, the gravitational pressure drop (being the driving pressure differential) is always the largest component of pressure drop and the sum of all other pressure drops (friction, acceleration and local) must balance the gravity (or buoyancy) pressure differential at steady state. However, the natural circulation flow regimes are differentiated based on the change of the pressure drop components with quality (or power).

2.1.1 Gravity Dominant Regime

The gravity dominant regime is usually observed at low qualities. In this regime, for a small change in quality there is a large change in the void fraction (see Fig. 2) and hence the density and buoyancy force. The increased buoyancy driving force is to be balanced by a corresponding increase in the retarding frictional force that is possible only at a higher flow rate. As a result, the gravity dominant regime is characterized by an increase in the flow rate with power (see Figs. 3).

2.1.2 Friction Dominant Regime

Friction dominant regime is observed at low to moderate pressures when quality is high. At high qualities and low to moderate pressures, the increase in void fraction with quality is marginal (Fig. 2) leading to almost constant buoyancy force. However, the continued conversion of high density water to low density steam due to increase in power requires that the mixture velocity must increase even with the same mass flow rate resulting in an increase in the frictional force and hence a decrease in flow rate. Thus the friction dominant regime is characterized by a decrease in flow rate with increase in power (see the curve for 15 bar in Fig. 3a and b).

2.1.3 Compensating Regime

Between the gravity dominant and friction dominant regimes, there exists a compensating regime, where the flow rate remains practically unaffected with increase in power. In this regime, the increase in buoyancy force is compensated by a corresponding increase in the frictional force leaving the flow unaffected (see the curve for 7 MPa in Fig. 3a)
in spite of increase in quality. The compensating regime exists only for moderate to high pressure in small diameter loops.

2.1.4 Effect of pressure and loop diameter

The NC flow regimes depend strongly on the system pressure. In fact, at high pressures only the gravity dominant regime (as in single-phase natural circulation) may be observed if the power is low. The friction dominant regime shifts to low pressures with increase in loop diameter (see Fig. 3b, 4c). Knowledge of the flow regimes is important to understand the stability behaviour of two-phase loops.

3 STABILITY BEHAVIOUR

Two-phase natural circulation systems are susceptible to a large number of instabilities. Instability is undesirable as sustained flow oscillations may cause forced mechanical vibration of components [5]. Further, premature CHF (critical heat flux) occurrence can be induced by flow oscillations as well as other undesirable secondary effects like power oscillations in BWRs. Instability can also disturb control systems and pose operational problems in nuclear reactors. The instabilities are broadly classified into three groups; static, dynamic and compound dynamic instabilities. A more detailed description of natural circulation instabilities can be found in Nayak and Vijayan [6].

3.1 Experimental Stability Map

Generally, two unstable regions are observed for two-phase NCSs as illustrated by the stability map given in Fig. 4a for a 9.1 mm i.d. (inside diameter) UDL (the geometry of the loop is shown in Fig. 1). The first unstable zone occurs at a low power and hence at low quality and is named as type-I instability by Fukuda and Kobori [7]. Similarly, the second unstable zone in the two-phase region (Fig. 4a) occurs at high powers and hence at high qualities and is named as type-II instability [7]. Theoretical analysis by the same authors has shown that the gravitational pressure drop plays a dominant role in type-I instability where as frictional pressure drop is dominant in type-II instability. In other words, type-I instability occurs in the gravity dominant regime whereas type-II instability occurs in the friction dominant regime. Instability is not reported so far in the compensating regime.

![Fig. 3: Effect of power, pressure and loop diameter on steady state two-phase NC flow](image-url)
Both type-I and type-II instabilities are affected by a large number of parameters such as power, pressure, inlet subcooling and loop diameter.

A general characteristic of type-I instability is that it occurs right from boiling inception. Flashing and geysering induced instability also belong to this category. Broadly speaking, with increase in power the amplitude of type-I oscillations first increases, reaches a peak and then decreases eventually leading to stable flow (Fig. 4b). However, amplitude variation is a complex function of power presumably due to flow pattern transitions (Fig. 4b). For certain type-I unstable regions, the flow alternates between a stable and oscillatory regime (Fig. 4c). The dynamics of type-I instability is quite rich showing many different oscillatory patterns.

The amplitude of type-I oscillations reduce significantly with increase in pressure [8]. Further, type-I instability is not observed beyond a critical value of the system pressure [8]. While experimenting with uniform diameter loops, it was observed that the critical value of pressure beyond which the instability disappears is found to decrease with increase in the loop diameter [8].

Most practical application of natural circulation systems employs non-uniform diameter loops. A typical example is the Advanced Heavy Water Reactor (AHWR) being designed in India (Fig. 5). An integral test Loop (ITL) simulating the AHWR has been set up in BARC (Fig. 6). Extensive experimentations have been carried out to study the characteristics of Type-I instability encountered during the start-up in ITL which is a non-uniform diameter loop. From these tests also, it is found that type-I instability disappears at high pressure (Fig. 7).
3.4 General Observations on Type-II instability

Type-II instability is found to occur after the flow starts to decrease with increase in power which is characteristic of friction dominant regime (Fig. 4a). The threshold of type-II instability is found to increase with pressure as well as loop diameter. Type-II instability is observed in large diameter loops only at low pressures. A general characteristic of the type-II instability is that the oscillation amplitude keeps increasing monotonically with power (Fig. 4a). When the amplitude reaches a critical value, CHF occurs causing test section burnout which was a serious problem during the tests in the small diameter loops [8].

3.5 Analysis of Instability

Apart from dynamic instability, two-phase loops can experience static instabilities. Examples of static instability are Ledinegg and flow pattern transition instabilities. Fig. 8 shows the predicted Ledinegg stability map using the methodology described in Todreas and Kazimi [9] for different pressures for the loop shown in Fig. 1. The unstable zone is found to shift to higher powers as well as decrease with increase in pressure (Fig. 8c). Ledinegg instability is not predicted at high pressures (2 MPa). Similar observations were made in Nayak et al. [10] for the Ledinegg instability in AHWR. Likewise, flow pattern transition instability also occurs only at low pressures [11].

The dynamic stability of the uniform diameter loops were studied with a linear stability code TINFLO-S. The code uses the homogeneous and the drift flux models and a detailed write-up on the linear stability analysis is available in Nayak et al. [12] and [13].
3.5.1 Stability analysis based on Homogeneous Equilibrium Model (HEM)

Using the HEM incorporated in the code TINFLO-S, stability analysis was carried out for the loop given in Fig. 1. The results for various diameters are given in Fig. 9. The CHF profile shown is predicted using the mass flux and quality corresponding to the type-II threshold with the CHF look-up table method [14]. The CHF threshold is always found to occur significantly above the type-II stability threshold. The stable zone is found to increase with loop diameter. For loop diameters greater than 15 mm, the upper threshold does not exist (Fig. 9e and f) in the two-phase region. The reason for the disappearance of the upper threshold from the two-phase region was investigated. Fig. 10 shows the steady state performance of the 7 mm loop and 15 mm loop. From Fig. 10, it is found that the friction dominant region exists for both the loops whereas the upper threshold exists only for the 7 mm loop. The main difference is that the flow rate is almost an order of magnitude less for the 7 mm loop resulting in large change of quality for small change in flow rate. The slope of the flow rate versus power curve is more negative for the 7 mm loop than that for the 15 mm loop.
3.5.2 Stability analysis based on Drift Flux Model (DFM)

The drift flux parameters $(C_0$ and $V_{gi})$ for slug flow were used as it was the most frequently observed flow pattern during the tests. The Martinelli-Nelson two-phase friction multiplier model was used in the computations. Nayak et al. [13] compared the predicted stability map with experimental data. Fig. 11 reveals that the stability is enhanced by increasing the loop diameter which is consistent with the HEM results. For 40 mm loop diameter (not shown in Fig. 11), the type-II threshold is not found in the two-phase region (i.e. $0<\text{quality}<1$). Experiments by Mochizuki [15] indicate that this instability occurs when the quality is close to unity at high pressure and large inlet subcooling. It may be noted that the stable zone (difference in power between the upper and lower thresholds) is very low for small diameter loops whereas it is significantly large in large diameter loops (Fig. 11b).

Compared to the results of homogeneous model, it is evident that the predicted trends are similar. However, significant quantitative differences exist. For example, the DFM predicts much smaller stable region compared to HEM. With HEM, the upper threshold disappears for loop diameter $\geq 15$ mm whereas it disappears for DFM with loop diameter $\geq 40$ mm. Further, with HEM the CHF threshold was always above the type-II stability threshold meaning that the maximum power is always controlled by stability. On the other hand with DFM the CHF threshold can be above the type-II threshold or below it with the result, the maximum power can be limited by stability, CHF or partly by CHF and partly by stability (see Fig 11b).
4 DESIGN CONSIDERATIONS

Most static instabilities including Ledinegg and flow pattern transition instabilities are observed only at low pressures. Only the density wave instability is observed at high pressures which show two unstable regions designated as type-I and type-II instabilities. Experiments show that it is possible to avoid type-I instability by a pressurized start-up (Figs. 7). Both HEM and DFM analyses show that the type-II instability threshold shifts to higher power with increase in loop diameter. For large diameter two-phase loops, it is possible to entirely eliminate the type-II instability. The above results are significant to the design of two-phase NC systems like boilers and pressure tube type BWRs.

4.1 Design Types for NCSs

Since both instability and CHF (critical heat flux) needs to be avoided in the design of two-phase natural circulation systems two types of designs are possible depending on which of them is limiting the maximum power that can be extracted. These are designated as stability controlled and CHF controlled designs. Both the homogeneous and the drift flux models predict these design types albeit at different loop diameters. In view of this, only the drift flux model predictions are used for explaining this concept.

4.1.1 Stability Controlled Design

In this type of NCSs, the maximum power is limited by the stability, as the threshold of type-II instability is lower than the CHF threshold. Both HEM and DFM predictions show that this is normally the case in small diameter loops. These predictions were obtained by incorporating the CHF look-up table [14] in the TINFLO-S code. The direct substitution method (DSM) was used for the CHF prediction. In Fig 11b, stability controlled design is found on the right of the vertical dotted line.

4.1.2 CHF Controlled Design

Characteristic of this design is that the CHF limits the maximum power that can be extracted and usually this situation arises in large diameter loops. Under this category two cases are possible; one in which the CHF threshold is below the type-II instability threshold (e.g. Fig. 11c) and another in which the type-II threshold does not exist.

4.1.2.1 CHF Controlled Design with Type-II threshold

Here the CHF threshold is entirely below the type-II threshold (Fig. 11c). One could also have another case in which the system can be partly stability controlled and partly CHF controlled (e.g. Fig. 11b). By appropriate choice of the loop diameter, the design of a NCS can shift from stability controlled to CHF controlled (Figs. 9 and 11).

4.1.2.2 CHF Controlled Designs without type-II threshold

Large diameter loops do not usually have a type-II threshold in the two-phase region. Both the homogeneous and the drift flux models predict this trend although the value of the loop diameter beyond which type-II instability disappears is significantly different. The design of such loops is controlled only by CHF.
5 DESIGN OF NCS AND THE OPERATING LINE CONCEPT

Two-phase NCSs are not completely stable over the entire subcooling-power map. With the help of stability analysis techniques one could identify the stable and unstable zones as described above. Operation at the threshold of stability is not desirable as the system continues to oscillate with the same amplitude indefinitely. Besides, a small disturbance can land the system in the unstable zone. To guard against this and to provide stable operation some sort of a stability margin is desirable.

Decay ratio (defined as the ratio of amplitudes of the succeeding to the preceding oscillation) was considered to provide an indication of the stability margin. However, several instabilities are simultaneously present with each of them having decay ratio (DR) of unity at the threshold condition. Therefore there is a need to base the DR on the controlling instability i.e. the instability with the least stable zone. In addition, there is a minimum practically achievable value of DR that is different at different subcooling even for the controlling instability (Fig. 12). Hence any arbitrary value of DR cannot be chosen as a stability margin. Besides, NCSs are to be started up from cold low pressure stagnant conditions. During power and pressure raising the NCS may pass through the Type-I unstable zone. Thus specifying an operating line is a logical way to provide the required stability margin. The adopted operating line must ensure stability for all anticipated operations like start-up, power raising, and step back.

5.1 Operating line for stability controlled Designs

One would expect the decay ratio to go through a minimum while moving from the lower to the upper threshold for a fixed subcooling (Fig. 12). Ideally, the operating line shall pass through the minimum decay ratio (MDR) line (locus of all minimum decay ratio points i.e. points A, B, C and D in Fig. 12) so that all oscillations will die down in the quickest possible manner. The MDR line is plotted in Fig. 13 which shows that it is closer to the type-I threshold. In stability-controlled designs, one could choose the operating line as the MDR line if low power operation is adequate. A drawback of the MDR line is that it is closer to the lower threshold and does not utilize much of the high power stable zone. Another approach is to use the line corresponding to the mean of thresholds (MOT) which is the arithmetic mean of the type-I and type-II thresholds (Fig. 13). The MOT line is midway between the type-I and type-II thresholds and the stability margins with respect to the lower and upper thresholds are equal in terms of power. Even with the MOT line, the high power stable zone is not utilized. With the constant decay ratio (CDR) line as shown in Fig. 14 much of the low subcooling high power stable zone can be utilized. Care must be exercised, however, as constant decay ratio lines with very low DR will not allow operation with large inlet subcooling. For example, with CDR line of 0.7, operation is possible only with inlet subcooling less than 10 K (Fig. 14).

![Fig. 12: Variation of decay ratio](image1.png)

![Fig. 13: MDR and MOT lines](image2.png)
5.2 Operating line for CHF Controlled Designs

In this case, the maximum power is limited by the CHF threshold. Even for this case, it may be feasible to select the MDR, MOT or CDR line as possible operating lines depending on the location of CHF threshold. As the MDR line is again found closer to the lower threshold, it may not be able to take advantage of the larger stable zone. The MOT line is feasible if sufficient thermal margin (in terms of critical heat flux ratio or critical power ratio) is there. However, the CDR line may be the desirable operating line if higher power is required. It may be noted that the CDR line will start from the MDR line and will terminate on the CHF threshold.

![Graph showing operating lines](image)

Fig. 14: Constant decay ratio lines

5.3 Operating line for CHF controlled designs without Type-II threshold

In this case, the decay ratio is found to go through a minimum and subsequently, it is found to rise marginally and stabilise at a constant value (Fig. 15). However, the magnitudes of the minimum and the constant DR values are close to each other although, the powers at which they occur are significantly different. This constant value is found to be only marginally different for different subcooling (Fig 15a). Thus for this case, a constant DR corresponding to the highest subcooling is sufficient to provide adequate stability margin at high powers.

![Graph showing decay ratio and operating lines](image)

(a) Variation of DR with power  
(b) MDR and CDR lines

Fig. 15: Decay ratio for 40 mm loop (Type-II instability is absent)
5.4 Parallel Channel Instability

So far only single channel systems were considered. Parallel channels can significantly modify the stability behaviour. Hence a twin channel system as shown in Fig. 16a was analysed using the homogeneous model and the results are given in Fig. 16b. Since the length scales of the single channel system given in Fig. 1 is same as the twin channel system considered, the results obtained with the single channel system is also given in Fig. 16b for comparison. It is found that the twin channel system is considerably less stable than single channel system. Similar observation was also made by Nayak et al. [12] in respect of AHWR.

In this case also, the decay ratio is found to go through a minimum for a fixed subcooling. The locus of all minimum decay ratio points (MDR line in Fig. 17) show the same characteristics as was observed for single channel systems. Again the mean of thresholds (MOT) line being equidistant from both the thresholds give better stability margin. Also, constant decay ratio lines can be adopted for operation closer to the upper threshold.

6 CONCLUDING REMARKS

A rational design procedure has been proposed based on the current knowledgebase. The operating line concept and the rationale behind the minimum decay ratio, mean of thresholds and the constant decay ratio lines have been discussed along with their relative...
merits and demerits. The above design procedure is almost entirely based on the linear stability analysis. Since the decay ratio is sensitive to the way we approach a specified operating point, nonlinear analysis is required to validate the chosen operating line. It must also be mentioned that the above procedures are developed based on experiments and analysis carried out for uniform diameter loops.

**NOMENCLATURE**

<table>
<thead>
<tr>
<th>Symbol</th>
<th>Definition</th>
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</thead>
<tbody>
<tr>
<td>A</td>
<td>flow area, ( m^2 )</td>
</tr>
<tr>
<td>b</td>
<td>exponent in the friction factor equation</td>
</tr>
<tr>
<td>D</td>
<td>hydraulic diameter, m</td>
</tr>
<tr>
<td>f</td>
<td>Darcy-Weisbach friction factor</td>
</tr>
<tr>
<td>g</td>
<td>gravitational acceleration, ( m/s^2 )</td>
</tr>
<tr>
<td>h</td>
<td>enthalpy, ( J/kg )</td>
</tr>
<tr>
<td>l</td>
<td>dimensionless length, ( L_e/L_t )</td>
</tr>
<tr>
<td>L</td>
<td>length, m</td>
</tr>
<tr>
<td>p</td>
<td>constant in the friction factor equation</td>
</tr>
<tr>
<td>P</td>
<td>pressure, Pa</td>
</tr>
<tr>
<td>( Q_h )</td>
<td>total heat input rate, W</td>
</tr>
<tr>
<td>v</td>
<td>specific volume, ( m^3/kg )</td>
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<tr>
<td>W</td>
<td>mass flow rate, kg/s</td>
</tr>
<tr>
<td>x</td>
<td>quality</td>
</tr>
<tr>
<td>( \Delta z )</td>
<td>Centre line elevation difference, m</td>
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**Greek symbols**

<table>
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<tr>
<td>( \alpha )</td>
<td>void fraction</td>
</tr>
<tr>
<td>( \beta_T )</td>
<td>thermal expansion coefficient, ( K^{-1} )</td>
</tr>
<tr>
<td>( \beta_h )</td>
<td>thermal expansion coefficient, ( (J/kg)^{-1} )</td>
</tr>
<tr>
<td>( \phi_{LO}^2 )</td>
<td>two-phase friction multiplier</td>
</tr>
<tr>
<td>( \bar{\phi}_{LO}^2 )</td>
<td>mean value of ( \phi_{LO}^2 )</td>
</tr>
<tr>
<td>( \rho )</td>
<td>density, ( kg/m^3 )</td>
</tr>
<tr>
<td>( \mu )</td>
<td>dynamic viscosity, ( Ns/m^2 )</td>
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</table>

**Subscripts**

<table>
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<tr>
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<tr>
<td>B</td>
<td>boiling length</td>
</tr>
<tr>
<td>C</td>
<td>cooler/condenser</td>
</tr>
<tr>
<td>e</td>
<td>heater exit</td>
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<tr>
<td>eff</td>
<td>effective</td>
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<tr>
<td>g</td>
<td>vapour</td>
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<tr>
<td>h</td>
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<td>i</td>
<td>( i^{th} ) segment</td>
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<tr>
<td>m</td>
<td>mean</td>
</tr>
<tr>
<td>sp</td>
<td>single phase</td>
</tr>
<tr>
<td>t</td>
<td>total</td>
</tr>
<tr>
<td>tp</td>
<td>two-phase</td>
</tr>
</tbody>
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Abbreviations

AHWR: Advanced Heavy Water Reactor
BARC: Bhabha Atomic Research Centre
CDR: Constant Decay Ratio
CHF: Critical Heat Flux
DFM: Drift Flux Model
DSM: Direct Substitution Method
DR: Decay Ratio
HEM: Homogeneous Equilibrium Model
ITL: Integral Test Loop
MDR: Minimum Decay Ratio
MOT: Mean of Thresholds
NC: Natural Circulation
NCB: Natural Circulation Boiler
NCS: Natural Circulation System
UDL: Uniform Diameter Loop

REFERENCES


Licensing and Harmonisation
SIMULATOR TRAINING AND NUCLEAR SAFETY IN KOZLODUY NPP PLC

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ABSTRACT

This presentation is an overview of the Bulgarian experience in simulator training and use of simulators for enhancement of reliability and nuclear safety in the operation of nuclear units with reactors WWER-1000 in Kozloduy NPP. The presentation covers the following subjects:

1. Brief history of Kozloduy NPP and the status of these units at present.
2. Stages of the preparation of MCR operators.
3. Use of simulators in Kozloduy NPP.
4. Simulator training and human factor reliability.
5. Human factor reliability.
6. Kozloduy NPP experience in the utilization and application of the method Human Error Assessment and Reduction Technique (HEART).
7. Using simulators for events analysis in Kozloduy NPP. (Example of an analysis of an event, occurred at KNPP Unit 5 on 01.03.2006, related to loss of a reactor coolant pump and a problem subsequently discovered with Reactor Control System).
8. Testing of the system for cold over pressure Protection from cold Hydrostatic Test System (YE).
9. Use of simulators in Kozloduy NPP for testing new systems for safety enhancement. (Example: testing the model of the new Safety Parameters Display System (SPDS), prior to its installation at the referent units).
10. The future of Nuclear in Bulgaria - the Belene NPP.

Kozloduy NPP experience presented can be of interest for international cooperation in training and for NPP safety enhancement.
INTRODUCTION

Location of Kozloduy NPP

Figure 1

1 BRIEF HISTORY OF KOZLODUY NPP AND STATUS OF THE UNITS:

KNPP is situated in North Bulgaria, in the vicinity of town of Kozloduy, on the Danube River bank.

There are four WWER-440 and two WWER-1000 units at Kozloduy NPP.

Units 1 & 2 were commissioned in July, 1974 and November, 1975, respectively. They were shut down at the end of 2003.

Units 3 & 4 were commissioned in December, 1980 and May, 1982. They were shut down at the end of 2006 as a precondition for Bulgaria’s accession to the European Union.

The 1000 MW units 5 & 6 of Kozloduy NPP were commissioned in September, 1988 and December, 1993, respectively.

Large-scale modernizations have been implemented and now the units meet all international safety standards.
2 STAGES OF KOZLODUY NPP OPERATIONAL STAFF TRAINING

Preparatory
Theoretical studies
Training at the Training Centre using different technical devices
Preparation and exams at Kozloduy NPP expert commission
Simulator training
Obtaining a permit for a license, corresponding to the position of
beginner at the NPP
Exams and licensing at Nuclear Regulatory Agency (NRA)
Shadow training at the working place
Permission for unaided operation

3 USE OF SIMULATORS IN KOZLODUY NPP.

Regular training of main control room operators
Validation of Symptom-Based Emergency Instructions (SBEI);
In 2005 at the FSS protection from “Cold Hydrostatic Test System”
(YE) was tested
The new Information “OVATION” system was tested
Review of Technical Decisions
Review of Operational Rules and Procedures
Testing models of the new systems, prior to installation at the
referent units
Example: testing the model of the new Safety Parameters Display System (SPDS)

4 SIMULATOR TRAINING AND HUMAN FACTOR RELIABILITY.

4.1 Reduction of number of significant events:

Worldwide practice proves, that improvement of engineering results is an increase in the percentage of events, related to human factor. Statistically those events are about 70% of all significant events. Events assessment of what a complicated work in a team is. Team work prevents or reduces consequences from significant events.

4.2 To improve simulator training

4.2.1 Analysis of operator’s activities

- Video records and self-criticism
- Improvement of strong and correction of weak points

4.2.2 Enhancement of specific knowledge

- Theoretical
- Technological

4.2.3 Use of feedback

- Correct actions of operators
- Demonstrated operators’ attitude towards simulator training and other activities

4.2.4 Engineering, technical staff and experts assistance.

5 HUMAN FACTOR RELIABILITY

5.1 Human error probability is related to:

- Way the task had been designed, organized and supported
- Way the Unit and equipment had been designed
- Context and conditions under which the task is performed

5.2 Feedback

5.2.1 Positive feedback:

- Acknowledgement of achievements
- Enhances willingness for good performance
- Assists gaining confidence

5.2.2 Negative feedback:

- Stimulates willingness for better performance
- Shall be used carefully
- May be harmful for achievements and self assessment
- Continuous effect of negative criticism does not have better influence on training of new skills and customs

6 Kozloduy NPP Experience in Utilization and Application of the Method - Human Error Assessment and Reduction Technique (HEART).

**Human Error Assessment and Reduction Technique (HEART)**

In 2005 and 2006 in cooperation with representatives from Great Britain and Technical University in Sofia we worked on the DTI NSP B15, 16 projects.

“Development of procedures for processing and implementing a full-scale simulator data to specify human reliability, applying the HEART quantification technique”.

Developed in Great Britain. A technique, widely implemented to analyze human reliability (AHR) at NPPs in Britain and Canada. Focused on assessment and includes elements from the good practices for the AHR. Emphasizes on human errors reduction, the only technique that proposes specific corrective measures to minimize human errors. Do not include a model of dependencies.

The NARA method for Analysis of human reliability is the latest development of HEART.

In USA they use the THERP method, specially developed for PSA in USA NPPs.

The HEART method for analysis of human reliability includes data on human factor and does not require detailed dismemberment of the task. Only ergonomic and related to the task factors are considered, that could impact negatively human performance.

The failure probability of the task takes into consideration a probable restoration from an error that is deemed possible for every task. Only the contribution of cognitive errors and response errors are considered.

7 Using Simulators for Event Analysis in Kozloduy NPP.

The full scope simulator at Kozloduy NPP is used successfully to analyze events, which occurred here or in other NPPs with water-water reactors, as well as develop simulator scenarios and corrective measures to prevent similar events in future at Kozloduy NPP.

An example for using the FSS for event analysis, occurred at KNPP Unit 5 on 01.03.2006, related to failure of a reactor coolant pump and a subsequent problem with Reactor Control System.
Analysis of operators’ actions to cut out one of the reactor coolant pumps and a problem with Reactor Control System

Figure 3
8 TESTING OF THE SYSTEM FOR COLD OVER PRESSURE PROTECTION FROM COLD HYDROSTATIC TEST SYSTEM (YE)

The protection system was tested on the full-scale simulator for cold over pressure PROTECTION FROM COLD HYDROSTATIC TEST SYSTEM (YE).

This system is very important for the safety operation of NPP units, because it prevents from impermissible pressure raise in the primary circuit when the circuit is cooled down.

As a result from these tests the work algorithm of the system was changed. Two principles are used in this system.

First one - blocking and switching off equipment, which can raise the primary circuit pressure.

Second - opening of the first safety valve of the pressurizer of primary circuit.

Technological scheme of the YE system

Figure 4
9 USE OF SIMULATORS IN KOZLODUY NPP FOR TESTING NEW SYSTEMS FOR SAFETY ENHANCEMENT.

9.1 Safety Parameters Display System (SPDS) is designated to work in mode of violation of the conditions for normal operation and in emergency cases.

The main assignment of SPDS is to ensure means for interpreting the meaning from the messages of announcement signalization and, even more significant, sets or models for messages notice signalization for Control Room operators.

Main purpose of SPDS is to reduce the class of human errors, known as “wrong intentions” (opposite to “fault performance”). Those are mistakes of the thinking or knowledge, i.e. mistakes of decision taking, for example fault understanding of the present condition of the processes in NPP etc.

9.2 As for the model of decision taking from a person in real time, the function of revealing of XSPDS is designated to support the team of Control Room at the following steps or activities of this model:

Graphical display of the tree of Critical Safety Functions (CSF), performing automatic check of their status and display of the results;

An ALARM SIGNAL, i.e. to inform the operator for abnormal conditions, which may lead to threats for population’s health and safety and that’s the reason they request more detailed search.

MONITORING OF ABNORMAL, i.e. data are collected from measuring devices of the Control Room or other sources, to assist the study of the abnormal condition’s character.

CONDITION IDENTIFICATION

9.3 Main System Functions of the Safety Parameters Display System and Critical Safety Functions System:

Display of specially processed concise summaries of the main safety parameters organized in a different manner for each Unit mode, thus covering all operational situations.

Graphical display of the tree of Critical Safety Functions (CSF) is performed, and an automatic check performance of their status and display of the results also.

Presentation of computerized version of the Symptom-Oriented Emergency Instructions (SOEI) is made and CSF recovery instructions thus assisting the operators in performing diagnostics of the Unit in case of emergency as bringing the Unit to safety status.

9.4 THE CONDITION OF IDENTIFICATION may be divided into two parts:

Recognition of NPP status from point of view of known models; who usually leads directly to selection of consequent actions.

Identification of the system status and data collected in advance is summarized in clear presentation of the present status of NPP. At this time the teams will identify WHAT is not right, but not WHY and HOW they have developed abnormal conditions.
Safety Parameters Display System (SPDS) and Critical Safety Functions (CSF) System

Figure 5

9.5 The following personnel were trained:

Hardware engineers;
Software engineers;
System administrator;
Technology specialists for development and support of the Symptom-Based Emergency Instructions (SBEI);
Full scope simulator software engineers;
Full scope simulator Instructors.
10.1 Calculated parameters of the Belene NPP:

- Safe shut down earthquake – 0.24g
- Extreme meteorological conditions
- Defense from plane crush
Iberdrola New Fuel Design Licensing Process

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ABSTRACT

To improve the nuclear fuel efficiency and to maintain the security of fuel supply together with the availability of the most advanced fuel design have been the goals of Iberdrola nuclear fuel management policy during the last years. Thus, the most advanced fuel designs developed by each vendor have been loaded in the Cofrentes NPP.

With the aim of achieving the above goals, Iberdrola Generación has developed an in-house methodology (GIRALDA Methodology) to perform bundle and nuclear design and reload licensing. This methodology, together with the support of fuel vendors, is applied to the licensing process of new fuel design to be used in the Cofrentes NPP.

Iberdrola Generación is in the process of designing and licensing the GNF2 fuel design developed by GNF to be loaded in Cofrentes next outage scheduled by May 2009. This licensing process is going to be performed taken into account the feedback obtained during the licensing process of former fuel designs as the SVEA Optima2 developed by Westinghouse and the Atrium10XP developed by Areva.

1 INTRODUCTION

The Iberdrola Generación Nuclear strategy has been oriented in the last years to increase the asset value of the different nuclear power plants (NPP) included in its nuclear portfolio. In order to obtain this goal, a number of power uprates and cycle length extensions have been implemented in most of the Iberdrola Generación NPP’s.

These changes in the NPP operation strategy require the availability of advance nuclear fuel designs that allow the plant operation under these demanding conditions, together with adequate operating margins, high fuel reliability and high burnup discharge values.

Other significant goal in the Iberdrola Generación strategy is the security of supply that can assure a reliable nuclear fuel supply in case of potential disruptions in some of the fuel vendor manufacturing processes and to assure fuel market prices maintaining the competitiveness between the fuel vendors.
To achieve these strategic objectives, Iberdrola Generación launched in 1992 a project with the final goal of developing an in-house and vendor independent reload fuel design and safety analysis methodologies project. The methodology developed under this process is named “GIRALDA Methodology”.

The “Reference Safety Report for BWR Reload Fuel” document [1] integrates the IBERDROLA BWR GIRALDA reload methodology that has been in used for fuel bundle design, reload licensing analysis and plant operational modification applications during the last four Cofrentes NPP cycles, including two power uprates. Cofrentes NPP is a BWR-6 type reactor GE design, operating since 1985 that has been uprated to the 112% from the original 2894 MWt.

2 GIRALDA METHODOLOGY DESCRIPTION

The GIRALDA methodology is used by Iberdrola Generación to perform the reload fuel design and safety analyses for Cofrentes NPP. The objective of the reload fuel and core designs, consistent with the utility energy utilization plan, is to establish a Reference Core design, which will reliably satisfy the operational objectives of the plant.

The objective of the reload fuel safety analyses is to demonstrate that the plant, using the reload fuel and core design established under the Reference Core, can operates without undue risk to the health and safety of the public. To satisfy these two primary objectives, Iberdrola Generación has developed a single highly interrelated process for reload fuel applications that covers all of the required subjects for the reload fuel design and safety analysis.

Consistent with the reload fuel design process, the GIRALDA methodology is split into three key disciplines: thermal-mechanical, nuclear and thermal-hydraulic:

- The thermal-mechanical design scope includes the fuel assembly and fuel rod performance analyses, and the generation of the input data needed to the other disciplines.

- The nuclear design includes the process used to determine the number and enrichment of the reload fuel assemblies, the development of a realistic nuclear core model that can be utilized for core follow and target control rod sequences, and the development of the nuclear parameters.

- The thermal-hydraulic analyses include the methodology for establishing the minimum critical power ratio safety limit, and the development of the thermal-hydraulic input parameters to the other disciplines.

Besides the above disciplines, the GIRALDA methodology involves different codes and process according the three categories of analysis include in the reload safety evaluation: anticipated and unexpected operational occurrences or transients, accidents and special events. Anticipated operational occurrences (AOO) are those conditions of normal operation which are expected to occur one or more times during the life of the plant and include but are not limited to generator load rejection, turbine trip, isolation of the main condenser, and loss of feedwater heating. Accidents are those postulated events that potentially affect one or more of the barriers to the release of radioactive materials to the environment. These events are not expected to occur during the plant lifetime, but are used to establish the design basis for many systems. Special events are postulated occurrences that are analyzed to demonstrate plant capabilities required by regulatory requirements and guidance, industry codes and standards, and licensing commitments applicable to the plant.

A number of different codes are involved in the GIRALDA analyses scope to cover the wide range of analysis that are involved in the design and licensing process that are
Table 1: GIRALDA methodology codes

<table>
<thead>
<tr>
<th>Analysis</th>
<th>Code</th>
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<tbody>
<tr>
<td>Nuclear: Fuel Lattice</td>
<td>CASMO-4</td>
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<tr>
<td>Nuclear: Core design</td>
<td>SIMULATE-3</td>
</tr>
<tr>
<td>Thermal-Hydraulic</td>
<td>SIMULATE-3</td>
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<tr>
<td>AOO: Fast transients</td>
<td>RETRAN-3D</td>
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<td>AOO: RWE</td>
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<td>AOO: LOFWH</td>
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<td>AOO: RFRO</td>
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<td></td>
<td>FRAP-T6/APK</td>
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<td></td>
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<td>Accidents: FLE</td>
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<td>Accidents: FHA</td>
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<td>LAPUR</td>
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<tr>
<td>Special events: SBLC</td>
<td>SIMULATE-3</td>
</tr>
<tr>
<td>Special events: Overpressure Protection</td>
<td>RETRAN-3D</td>
</tr>
</tbody>
</table>

3 NEW FUEL DESIGN LICENSING PROCESS DESCRIPTION

The licensing process of a new bundle fuel to be loaded in Cofrentes NPP involves a double process: the generic fuel bundle licensing process similar to one used in most of the plants, and a specific process to qualify the GIRALDA methodology to be used with the new fuel design. These two processes are described in more detail in the next paragraphs.

3.1 Generic fuel bundle design licensing process

The fuel bundle licensing process involves all the activities and analyses that are required to obtain the permission for loading a specific design in the Cofrentes core. This fuel bundle licensing process involves two different processes. The first process includes all the generic and Cofrentes plant dependent analyses that are cycle independent. The second process involves all the cycle dependent analyses.

3.1.1 Cycle independent fuel licensing process

This process includes all the analyses and documentation required to demonstrate the new fuel design accomplish all the generic fuel licensing acceptance criteria. The fuel licensing acceptance criteria establish the basis for evaluating new fuel designs, developing the critical power correlation for these designs, and determining the applicability of generic analyses to these new designs. The generic criteria to be met by a new fuel design to be loaded in Cofrentes plant are those indicated in the Standard Review Plan SRP [2]. In addition to the generic licensing aspect, the Cofrentes specific licensing process will confirm that plant
specific cycle-independent aspects of the new fuel meet the design and licensing basis requirements of the plant.

This stage of the process is based on both generic vendor documentation and Iberdrola in-house analysis. Aspects that are covered in this process are the following:

- Rod Thermal-mechanical. These analyses that are under vendor scope include the verification of the following design criteria bases: fuel rod stresses and cladding fatigue, cladding corrosion, fuel rod hydrogen, cladding pressure loading, cladding collapse and fuel melting.
- Bundle Mechanical Design. The verification of the following design criteria are also under fuel vendor scope: stress, fatigue, fretting wear, corrosion and hydriding and compatibility including dimensional changes.
- Seismic evaluation of new fuel performance under the Cofrentes design base earthquake is analyzed by the vendor.
- Thermal hydraulic design. It is under vendor responsibility the generic compatibility analysis and to provide the critical power correlation.
- Nuclear: generic verification of Doppler, moderator void, moderator temperature reactivity coefficients is under vendor scope.
- Loss of Coolant Accidents (LOCA) analysis: It is under Iberdrola the full scope of the verification of acceptance criteria related with the peak clad temperature, maximum oxidation and maximum hydrogen generation during Loss of Coolant Accidents.
- Anticipated Transients Without Scram (ATWS). The specific verification of acceptance criteria related with the fuel integrity: peak clad temperature and clad oxidation for new fuel design is also under Iberdrola scope.
- Radiological analysis. Radiological consequences during the design bases accidents involving core damage (LOCA, fuel handling and control rod drop accidents) are analyzed with Iberdrola methods.
- Fuel storage critical analyses for both fresh and irradiated fuel are analyzed with specific Iberdrola Generación methods.

3.1.2 Cycle dependent fuel licensing process

In addition to the above generic information, the fuel licensing process includes additional activities related with the cycle specific analyses. The cycle specific analyses, which are part of the normal reload process, are documented in the specific Reload Licensing Report (RLR). These analyses are performed with the GIRALDA methodology considering each specific cycle core configuration for the limiting AOO’s and the bounding accidents. Thus, the Safety Limit Maximum Critical Power Ratio (SLMCPR) and the cycle Operating Limit Minimum Critical Power Ratio (OLMCPR) is determined.

Additional cycle specific analyses covered with the GIRALDA methodology are the thermal-hydraulic compatibility, the stability analysis and the overpressure protection analysis.

3.2 GIRALDA methodology qualification

The qualification of the GIRALDA methodology is a previous step to its application to design and licensing analyses of a new bundle design. This qualification process is required to verify that the input models developed for the new fuel design and the codes used in the methodology allow to reproduce correctly both the neutronic and thermal-hydraulic fuel expected performance. The qualification of the GIRALDA methodology is a requirement from the Spanish Regulatory Body, named Consejo de Seguridad Nuclear (CSN), previous to
the applicability of the methodology to any licensing analysis in which the new fuel design could be involved.

This process involves the qualification of the following codes and models:

- **CASMO-4 code**: two-dimensional lattice physics code that is used to calculate the nuclear data (e.g., cross sections, local peaking factors, MCPR subchannel factors, detector constants, etc.) required for the three-dimensional nodal core simulator input (SIMULATE-3). The verification is performed through comparisons of CASMO-4 results against reference calculation, usually performed with Monte Carlo codes.

- **SIMULATE-3**: three-dimensional nodal core simulator that used to analyze the core neutronic and thermal-hydraulic behaviour. The SIMULATE-3 qualification process involves the verification of the bundle friction pressure drop. The code results are compared with vendor data, either from experiment facility or vendor codes, to verify that the pressure drop is correctly reproduced. SIMULATE-3 pressure friction models can be adjusted to improve the code results in order to obtain a better reproduction of reference data.

- **RETRAN-3D**: transient analysis system of code designed to analyze operational transients. The correct reproduction of fuel model is verified against vendor data comparing axial enthalpy profile, axial void fraction, bundle pressure drop and critical power ratio (CPR). Both steady state and transient data are used in this qualification.

## 4 GNF2 FUEL LICENSING PROCESS

After the general description of the new fuel design licensing process, a more detailed explanation of the specific GNF2 licensing process, mainly focused in the qualification of the GIRLADA methodology, is presented in the following sections.

### 4.1 GNF2 fuel description

The GNF2 fuel design is going to be loaded as first time in Cofrentes cycle 18, starting in June 2009 together with the Optima2 and Atrium10XP resident fuel designs.

The GNF2 design consists of 92 fuel rods and two large central water rods contained in a 10x10 array. The two water rods encompass eight fuel rod positions. Eight fuel rods are designated as long part length fuel rods and six fuels rods designated as short part length fuel rods. This assembly is encased in an interactive fuel channel, which has been used on the GE10, GE11, GE12, GE13, and GE14 designs. The fuel rods consist of high-density ceramic UO$_2$ or (U, Gd)O$_2$ fuel pellets stacked within Zircaloy-2 cladding. GNF2 was designed for mechanical, nuclear, and thermal-hydraulic compatibility with the other GNF fuel designs.

The design includes many features of the GE10, GE11/13 and GE12/14 fuel including pellet-cladding interaction resistant barrier cladding, high performance spacers, part length rods, interactive thick corner/thin wall channel, and axial enrichment loading. New or improved features included in GNF2 are:

- Part length rod configuration that improves efficiency and reactivity margins
- Eight Alloy X-750 spacers with reduced pressure drop and improved resistance to boiling transition.
- New fuel rod design with increased uranium content
- Defender Debris Filter
4.2 Generic GNF2 bundle design licensing process

4.2.1 Cycle independent GNF2 fuel licensing process

The generic cycle independent GNF2 fuel licensing process is mainly based on the GNF2 vendor licensing documentation [3]. The scope of this report is in accordance with the fuel licensing acceptance criteria as specified in GESTAR II (NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel) and often called the Amendment 22 process. The Amendment 22 process was approved by the NRC in July 1990. The fuel licensing acceptance criteria included in GESTAR II establishes the basis for evaluating new fuel designs, developing the critical power correlation for these designs, and determining the applicability of generic analyses to these new designs. Compliance with the fuel licensing acceptance criteria constitutes USNRC acceptance of the fuel design without specific USNRC review. All of the criteria defined in GESTAR II have been met for the GNF2 fuel design. This process has been previously applied to other GNF designs (GE14, GE12 and GE11) previously loaded at Cofrentes plant. The USNRC acceptance is a significant stage in the licensing process that simplifies the licensing process with the CSN.

The fuel licensing criteria from GESTAR II are included in the report [3] covering some of the generic licensing aspects as the thermal-mechanical, nuclear, thermal-hydraulic and critical power correlation. In addition to the generic evaluations, additional analyses considering the specific Cofrentes operating conditions are performed by the vendor such as the rod thermal-mechanical, the mechanical bundle report and the seismic evaluation.

The scope of generic GNF2 licensing analyses that are performed with the GIRALDA methodology (LOCA, ATWS, storage criticality and radiological analysis) is currently in progress and it will be submitted to the CSN together with the specific cycle dependent licensing reports.

4.2.2 Cycle dependent fuel licensing process

The specific cycle 18 analyses, the first cycle including the GNF2 design, will be based on the actual core configuration and they will be submitted to the CSN within the specific cycle 18 Reload Licensing Report (RLR). This report will be submitted to the CSN three months in advance of the cycle 18 startup. Since these analyses will be performed with the GIRALDA methodology, the verification of the adequate reproduction of the GNF2 performance in the Cofrentes core with the Iberdrola Generación methodology is a significant milestone in this licensing process.

4.3 GNF2 Giralda methodology qualification

As stated above, the ultimate objective of this qualification process is to verify that the input models developed for the GNF2 fuel design, and the codes used in the methodology, accurately reproduce both the neutronic and thermal-hydraulic fuel performance. A detailed description of the qualification of the GIRALDA methods to be applied for the design and licensing analysis of the GNF2 fuel design is presented in this section.

4.3.1 CASMO-4 qualification

Since the GNF2 presents some specific features that were not included in previous GNF fuel designs as the two type partial length rods, it has been considered convenient to qualify the two-dimensional lattice physics CASMO-4 that is used to calculate the nuclear data against results obtained with a MNCP code [4]. Los Alamos MCNP is MonteCarlo transport...
code that is widely used by the nuclear industry with an extensive qualification against critical experiments.

The qualification scope cover the comparison of the results obtained with CASMO-4 against MCNP for the two more representative lattices (top and bottom lattices) of the actual GNF2 nuclear design to be loaded in Cofrentes next cycle. The comparisons cover the expected GNF2 operating conditions, considering different burnup and void fraction values and for both cold and hot conditions and two different control rod conditions. Table 2 shows the matrix cases included in the qualification of the bottom lattice.

**Tabla 2. GNF2 CASMO-4 qualification matrix**

<table>
<thead>
<tr>
<th>Void Fraction</th>
<th>Control</th>
<th>No Control</th>
<th>Control</th>
<th>No Control</th>
</tr>
</thead>
<tbody>
<tr>
<td>MCNP (zero burnup)</td>
<td>0%</td>
<td>0%</td>
<td>40%</td>
<td>80%</td>
</tr>
<tr>
<td>2D local power map</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td>K-infinite</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td>MCNP (after burnup)</td>
<td>0%</td>
<td>0%</td>
<td>80%</td>
<td>0%</td>
</tr>
<tr>
<td>2D local power map</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td>Maximum local power vs burnup</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td>K-infinite vs burnup</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
</tbody>
</table>

The parameters that have been considered for the comparisons between CASMO-4 and MCNP are the \( K_\infty \), the maximum relative power and the power distributions. The comparisons have shown an excellent agreement based on the following results:

- K\_\infty (table 3): average value differences are around 0.6\% and the maximum difference are below 2\%.

**Table 3. K\_\infty. CASMO-4 vs. MCNP differences.**

<table>
<thead>
<tr>
<th>Burnup (MWD/kgU)</th>
<th>Void 0%</th>
<th>Void 40%</th>
<th>Void 80%</th>
<th>CRD</th>
<th>CLD</th>
<th>CLD+CRD</th>
</tr>
</thead>
<tbody>
<tr>
<td>Top Lattice</td>
<td>-0.51</td>
<td>-0.29</td>
<td>0.12</td>
<td>-0.60</td>
<td>0.01</td>
<td>-0.29</td>
</tr>
<tr>
<td>0</td>
<td>-0.45</td>
<td>-0.16</td>
<td>0.37</td>
<td>-0.43</td>
<td>-0.30</td>
<td>-0.46</td>
</tr>
<tr>
<td>60</td>
<td>-1.39</td>
<td>-1.25</td>
<td>-0.73</td>
<td>-2.08</td>
<td>-0.70</td>
<td>-0.94</td>
</tr>
</tbody>
</table>

- Maximum relative power (table 4): in average, the CASMO-4 power peaking factors are 0.38\% higher than MCNP values, with a deviation between the two codes of 0.7\%.

**Table 4. Maximum relative power. CASMO-4 vs. MCNP differences.**

<table>
<thead>
<tr>
<th>Burnup (MWD/kgU)</th>
<th>Void 0%</th>
<th>Void 40%</th>
<th>Void 80%</th>
<th>CRD</th>
<th>CLD</th>
<th>CLD+CRD</th>
</tr>
</thead>
<tbody>
<tr>
<td>Top Lattice</td>
<td>0.07</td>
<td>0.44</td>
<td>1.31</td>
<td>1.34</td>
<td>-0.92</td>
<td>0.20</td>
</tr>
<tr>
<td>0</td>
<td>0.70</td>
<td>1.12</td>
<td>1.12</td>
<td>1.20</td>
<td>-0.50</td>
<td>0.78</td>
</tr>
<tr>
<td>60</td>
<td>0.33</td>
<td>0.60</td>
<td>0.45</td>
<td>0.12</td>
<td>-0.42</td>
<td>-0.97</td>
</tr>
</tbody>
</table>

- Power distributions: the quadratic average deviation (RMS) between the two codes is 1.04\% (table 5).
Table 5. Power distribution. CASMO-4 vs. MCNP differences.

<table>
<thead>
<tr>
<th>Burnup (MWD/kgU)</th>
<th>Void 0%</th>
<th>Void 40%</th>
<th>Void 80%</th>
<th>CRD</th>
<th>CLD</th>
<th>CLD+CRD</th>
</tr>
</thead>
<tbody>
<tr>
<td>Top Lattice</td>
<td>0.541</td>
<td>0.621</td>
<td>1.394</td>
<td>1.280</td>
<td>0.926</td>
<td>1.284</td>
</tr>
<tr>
<td>0</td>
<td>0.609</td>
<td>0.734</td>
<td>1.227</td>
<td>1.383</td>
<td>1.186</td>
<td>1.556</td>
</tr>
<tr>
<td>60</td>
<td>0.671</td>
<td>0.737</td>
<td>0.935</td>
<td>1.088</td>
<td>0.768</td>
<td>1.071</td>
</tr>
<tr>
<td>RMS (%) TOTAL</td>
<td>0.64</td>
<td>0.74</td>
<td>1.09</td>
<td>1.24</td>
<td>1.00</td>
<td>1.34</td>
</tr>
</tbody>
</table>

The slight differences obtained between CASMO-4 and MCNP codes, most of them below 2%, are similar to those obtained in the qualification of former fuel designs. Therefore, it is possible to conclude that the CASMO-4 code with the GNF2 input models is adequate to reproduce the neutronic behavior of the GNF2 design.

4.3.2 SIMULATE-3 qualification

The three-dimensional nodal core simulator SIMULATE-3 is used to analyze the core neutronic and thermal-hydraulic behaviour. SIMULATE-3 provides most of the thermal-hydraulic inputs that are required by other GIRLADA codes (core and bundle pressure drop, core and bundle flow distributions, initial CPR). The variable that is considered to verify the correct behaviour of the code is the bundle friction pressure drop. As result of the qualification, the SIMULATE-3 monophase friction factor can be adjusted to reproduce the friction pressure drop obtained from both experimental and code results vendor data. As result of the qualification process it has been possible to determine that the same SIMULATE-3 monophase friction factor that was obtained in the SIMULATE-3 qualification for the former GNF fuel design loaded in Cofrentes (GE11, GE12 and GE14) is also applicable for GNF2. This friction factor model has been confirmed to reproduce both the code vendor and experimental data used in the qualification.

Figure 1 shows the results of friction pressure drop obtained with both SIMULATE-3 and vendor codes. This comparison shows the correct reproduction of vendor data. One additional comparison included in the qualification is the spacer pressure drop: difference between the two codes for different Cofrentes power and flow conditions are shown in table 6. Results show that differences between the two codes are always below 2.5%.
Table 6. Spacer pressure drop. SIMULATE-3 vs. vendor code

<table>
<thead>
<tr>
<th>Power (%)</th>
<th>Core Flow (%)</th>
<th>Vendor code (psi)</th>
<th>SIMULATE-3 (psi)</th>
<th>Difference (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>112.3</td>
<td>105</td>
<td>3.87</td>
<td>3.93</td>
<td>-1.58</td>
</tr>
<tr>
<td>112.3</td>
<td>89.5</td>
<td>3.1</td>
<td>3.16</td>
<td>-2.06</td>
</tr>
<tr>
<td>100</td>
<td>105</td>
<td>3.6</td>
<td>3.69</td>
<td>-2.39</td>
</tr>
<tr>
<td>100</td>
<td>100</td>
<td>3.38</td>
<td>3.45</td>
<td>-1.92</td>
</tr>
<tr>
<td>100</td>
<td>80</td>
<td>2.49</td>
<td>2.54</td>
<td>-2.05</td>
</tr>
<tr>
<td>75</td>
<td>105</td>
<td>3.13</td>
<td>3.19</td>
<td>-2.01</td>
</tr>
<tr>
<td>75</td>
<td>80</td>
<td>2.13</td>
<td>2.17</td>
<td>-1.78</td>
</tr>
<tr>
<td>75</td>
<td>50</td>
<td>1.16</td>
<td>1.19</td>
<td>-2.16</td>
</tr>
<tr>
<td>50</td>
<td>80</td>
<td>1.74</td>
<td>1.77</td>
<td>-1.55</td>
</tr>
<tr>
<td>50</td>
<td>50</td>
<td>0.91</td>
<td>0.93</td>
<td>-1.76</td>
</tr>
</tbody>
</table>

Besides the vendor code data, specific GNF2 pressure drop data from full scale testing has been considered to qualify the SIMULATE-3 code. Thirty seven experiments considering a wide range of operating conditions (exit quality varying from 2% to 49%, exit void fraction varying from 20% to 95% and bundle flow varying from 25% to 120%) have been considered in the qualification process. Figure 2 shows the excellent agreement for the total bundle pressure drop calculated with SIMULATE-3 with the experimental data.
The good results obtained in the thermalhydraulic qualification of SIMULATE-3 with average difference between the code results and vendor data below 3% confirms that the SIMULATE-3 code with the GNF2 input model is able to reproduce correctly the GNF2 thermal-hydraulic performance.

4.3.3 RETRAN-3D qualification

An extensive qualification of the RETRAN-3D code used as transient analysis system has been considered to demonstrate the correct reproduction of GNF2 performance. This qualification has covered both the thermal-hydraulic behaviour and the capability to determine the correct critical power values.

The thermal-hydraulic qualification has been performed comparing the RETRAN-3D results against vendor code data values. The results of the two codes for the axial distribution of the most significant bundle variables (pressure drop, enthalpy and void fraction) for different power and core flow conditions have been compared. Figure 3 shows the comparison of the axial void fraction for the 100% power and 100% flow condition. The excellent agreement between the two codes can conclude that the RETRAN-3D code is able to adequately reproduce the GNF2 thermal-hydraulic performance.

![Figure 3. Void Fraction axial distribution. RETRAN-3D vs vendor code.](image)

Since the RETRAN-3D code is used to determine the diminution of the CPR evolution during the licensing transients, the correct qualification of the capability of the code to reproduce the evolution of this parameter is a significant stage of the GIRALDA qualification process. To accomplish this goal, both vendor code results and experimental data considering steady state and transient values have been considered.

The initial CPR (ICPR) results obtained with RETRAN-3D and vendor code for ten different steady state power and flow conditions are shown in figure 4. The small differences between the two codes, which are below the 0.35%, show the correct RETRAN-3D reproduction of this parameter.
The RETRAN-3D qualification has been completed with comparison against the STERN experimental facility for both steady state and transient conditions. Thirty-three experimental steady state experiments have been used to verify the modelization of GNF2 in the RETRAN-3D code. Figure 5 shows the good agreement between the CPR calculated with RETRAN-3D compared with the experimental CPR value. The maximum difference obtained in this comparison is below the standard deviation associated to the GNF2 CPR correlation.

Fifteen full scale transient tests have been used to verify the correct RETRAN-3D calculation of CPR evolution under the power and flow changes expected during licensing transients. These experiments include power increase and flow reductions that are designed to result in the onset of boiling transition onset. Also, the transient clad temperature increase is measured in the test assembly. The instant in which the temperature increase occurred in the experiment is compared with the instant in which the code determines the CPR value is below one. The RETRAN-3D code can be considered conservative when it determines the instant of CPR below one in advance the clad temperature increase is measured in the experiment.
Results included in figure 6 show that the RETRAN-3D code is reproducing with some conservatism the experimental data.

![Figure 6. Boiling transition onset time. RETRAN-3D vs. experimental data.](image)

The comparisons of RETRAN-3D results against vendor data can conclude, finally, that the GNF2 RETRAN-3D model is able to reproduce the expected GNF2 thermalhydraulic and critical power performance.

5 CONCLUSIONS

The licensing and qualification process of GIRALDA methodology for its application to a new fuel design to be loaded in Cofrentes NPP is presented in this paper.

Iberdrola Generación is applying a mature process to cover both the licensing of the new fuel design and the qualification of the GIRALDA methodology. This process is being applied for the GNF2 fuel design that is expected to be loaded in Cofrentes cycle 18 starting in June 2009.

The qualification of the main models and codes involved in design and licensing GIRALDA methodology has shown that the Iberdrola Generación methods are able to reproduce the GNF2 performance in a proper way and, therefore, this methodology can be applied for the design and licensing analyses involving the GNF2 fuel design.

REFERENCES


APROS Simulation Model for Olkiluoto-3 EPR Applications

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ABSTRACT

On the assignment of the plant owner Teollisuuden Voima Oyj, Fortum Nuclear Services Ltd has developed an APROS simulation model for Olkiluoto 3 EPR type Nuclear Power Plant. The developed model is being used for independent engineering and safety analyses of Olkiluoto-3 EPR Nuclear Power Plant Unit. The simulation model contains a detailed description of the primary and secondary circuits, emergency systems, protection systems, main and some auxiliary control systems. The simulation model has been validated in steady state, transient and accident situations. The EPR simulation model has been used in the analyses of some transients such as a spurious closure of low pressure steam admission valves and reverse flow of the low pressure turbine extraction lines. The developed simulation model has been utilized also in the safety analyses e.g. in the analysis of the cold leg large break loss-of-coolant accident (LBLOCA). For this application, a new counter-current flow limitation (CCFL) correlations for the downcomer and the upper tie plate have been implemented in APROS. In the paper, a short overview of APROS Simulation Software and its utilization in different kinds of applications are given. The paper describes the extent of the developed EPR application model and some results of the validation of the model. The description of the new CCFL correlations and the results of the validation calculations are presented. Finally, some results of the analyses of the transients and LBLOCA accident are presented and the utilization of APROS in these applications is discussed.

1 INTRODUCTION

The electricity consumption is growing in Finland. At the same time, old fossil power plants are being decommissioned. The Olkiluoto 3 Nuclear Power Plant (OL3) which is now under construction will compensate for the increasing demand and will replace obsolete production capacity. The environmental factors are becoming more important in the power generation. The OL3 will also help Finland to attain the targets defined in the Kyoto Protocol.

The OL3 project is being implemented on a turnkey basis by a French-German Consortium formed by AREVA NP and Siemens. The OL3 is a pressurized water reactor, more specifically an EPR (European Pressurized Water Reactor).
The design of the process and automation systems as well as the production of the safety analyses are some of the numerous tasks of the Consortium in the project. The interest of the plant owner is to make sure in the early phase of the project that all the plans and analyses results correspond to the requirements which are conditions for the safe and economical operation of the plant.

A simulation is an effective and useful tool to understand more deeply the behaviour of the plant. By means of the simulation, the plant owner can check the process and automation plans as well as make comparison calculations with safety analyses delivered by the Consortium. Due to this, Teollisuuden Voima Oyj asked Fortum Nuclear Services Ltd to develop a basic APROS simulation model for OL3. The developed simulation model covers the main processes of the primary and secondary circuits, safety systems, main control and protection systems as well as electrical systems. The modelling work was started in March 2005 and the validated model was ready for the applications one year later. Thereafter, the model has been applied in many engineering and safety analyses.

2 APROS SIMULATION SOFTWARE

The development of an Advanced PROcess Simulation software (APROS) was initiated 1986 in co-operation with Fortum Nuclear Services Ltd and VTT Technical Research Centre of Finland. APROS is a multifunctional simulation tool, which is suitable for various tasks during a complete project cycle of a nuclear and thermal power plant from the plant design to operator training. It can be used e.g. in preliminary design, detailed process and automation design, testing, engineering and safety analysis as well as training simulator applications.

The power plant library consists of comprehensive simulation models. The thermal hydraulic library contains three-, five- and six-equation models for the calculation of one-dimensional two-phase flow. Fast access material property tables are used for the computation of the water and steam material properties. The component library includes an extensive set of ready-made unit operation or process component models for the simulation of different kinds of processes, such as nuclear reactor and boiler plants, including automation and electrical systems.

The applied modelling and simulation tool has many useful and also unique features:
- physical, accurate and dynamical models
- graphical, interactive modelling
- fast running simulator
- extensive validation
- computer independence
- the plant including the process, automation and electrical systems can be modelled
- open and easy connection for external routines as DLL-files or communicating interfaces (APROS Communication Library ACL, OPC, OPC-XML-DA).

The validity of power plant models ranges from cold start-up to normal operation modes, normal and emergency shutdown, and load rejections and to failures of any combination of process, automation or electrical components.

3 APROS APPLICATIONS

Today, APROS has users in 22 countries. Power plants, engineering offices, safety authorities, research organisations and universities are using APROS in the process and automation design, development of emergency operating procedures, testing of I&C systems, accident analyses and training simulator applications.
During the last ten years many large projects have been performed and started at Loviisa Nuclear Power Plant to continue and improve the safety and good availability of the plant [1]. The utilization of APROS to analyse and simulate the behaviour of the power plant in transient and accident situations has shown an important role in all these projects.

A modernization and power uprating program of Loviisa VVER-440 reactors was carried out in 1995-97. Among other things, Loviisa Final Safety Analysis Report (FSAR) was extensively revised as part of the licensing process for higher reactor thermal power. The major part of FSAR safety analyses were calculated by APROS using the uprated 1500 MWth as a nominal power.

The original design life time of Loviisa nuclear power plant is 30 years. The operating license would have expired at the end of 2007 and 2010 for the units 1 and 2, respectively. Fortum, the owner and the operator of the plant, applied an operating license extension of both units for 20 years starting from the beginning of 2008. For the application, a great number of different studies and analyses were performed by APROS to show to the authorities that the plant can be operated safely for another 20 years.

It is obvious that the lifetime of the original I&C systems is not sufficient to guarantee the safety and the good availability of the plant in the future. Due to this fact, a project for the renewal of the existing I&C systems has been started. In the project, the analogue I&C systems will be renewed by digital I&C systems in four phases during 2005...2014. APROS based simulators will be used extensively during the renewal project [2]:

- an engineering simulator for design and analysis of control and process changes
- a development simulator for design, testing and qualification of the human-machine interface (HMI)
- a testing simulator for testing of the I&C software and for retuning of the controllers
- a training simulator to familiarize the operators and other technical personnel to the operation of the new monitor-based control room facilities.

4 APROS MODEL FOR OLKILUOTO-3

The basic APROS model for OL3 includes a detailed description of the reactor coolant system with all four loops as well as reactor control and shutdown equipment. The pressurizer model includes 40 calculation nodes, 20 nodes in the outer layer and 20 nodes in the inner part. Residual heat removal systems for reactor coolant and coolant treatment system are modelled including water tanks, pipe lines, valves, heat exchangers and pumps. As an example a primary circuit model is shown in Figure 1.

![Figure 1: Part of the whole model simulating the primary circuit of the EPR power plant](image-url)
The reactor core model consists of an average core divided into 20 vertical nodes having 1-D reactor kinetics calculation. The reactor by-pass flow channel is modelled with 6 nodes beside the core model. The upper plenum of the pressure vessel is divided into two parallel parts and the downcomer into four 90° sectors. The lower plenum is divided into three layers in vertical direction and each sector of these layers consists of 2 nodes, one in the center part and one on the edge.

The secondary side is modelled as a closed circuit. All four steam generators are modeled by a ready-made APROS process component specifically developed for the EPR-type steam generator. The blowdown lines are connected to the steam generators. The steam lines and valves, high and low pressure turbines, moisture separators, reheaters, turbine by-pass valves, condensers, low pressure heaters, feed water and emergency feed water pumps, high pressure heaters and tanks as well as lines between different components are included in the secondary circuit model.

The following protection systems are modelled:

- reactor trip
- partial reactor trip
- plant protection
- turbine trip
- turbine protection

The following control systems are included in the model:

- pressurizer level control
- primary pressure control
- reactor power control
- steam generator level control
- turbine control
- turbine by-pass control
- main steam release control
- feed water level control
- some other auxiliary control systems

The simulation model contains also the primary and secondary side electricity network as well as an external network, which has been modelled with one generator and one load module.

5 VALIDATION OF OLKILUOTO-3 MODEL

The heat balance tests as well as accidents and transients, e.g. SBLOCA (small break loss of coolant accident) and house load transient, were calculated with APROS. The results of the APROS calculations were compared to the results of the PSAR analyses.

The calculation results of the secondary side heat balance tests showed that the calculated steady state values of the APROS model are generally in very good agreement with the values calculated by the Consortium.

The calculation results of the transients and SBLOCA accident were essentially similar to the results of the PSAR analysis. The differences between APROS and PSAR calculations are in general easy to explain by the different initial assumptions.
6 OLKILUOTO-3 ENGINEERING APPLICATIONS

The developed OL3 simulation model has already been used to make comparison calculations with the engineering calculations delivered by the Consortium. The simulation model has also been used to understand more deeply the behaviour of the power plant in some transients. In the following, some results have been given as examples.

6.1 Spurious Closure of Low Pressure Steam Admission Valves

It is important to know the behaviour of the pressure of the moisture separator reheater (MSR) plant in case the low pressure admission valves close spuriously. The Consortium had earlier analyzed such kind of transient by own models. The plant owner wanted to make the comparison calculations and asked Fortum Nuclear Services Ltd to analyze the same transient by the developed OL3 simulation model.

Two different cases were analyzed. In the first case, the water amount stored in the moisture separator/reheaters was small (0.07 m$^3$) and in the second case about 4 m$^3$. At the beginning of the calculations, the plant was at full power. The closure of the low pressure steam valves caused the pressure increase in the MSR plant.

Figure 2: SBLOCA, primary pressure behaviour

Figure 3: Spurious closure of steam admission valves, MSR pressure behaviour
The pressure level actuated the turbine trip in 0.755 second. The turbine trip signal closed the main steam line valves and turbine extraction lines valves and the MSR plant was isolated from the rest of the turbine plant excluding the extraction steam line from the MSR plant to the feed water tank. The maximum pressure of MSR plant was reached in both cases almost at the same time (about 2.9 second).

The simulation results calculated with APROS are very close to the results delivered by the Consortium.

6.2 Reverse flow of Low Pressure Turbine Extraction Lines

A rapid reverse steam flow in the extraction lines can cause problems with the blades in the low pressure turbines. Such kinds of problems have been taken place already in some large nuclear power plant units. Due to this, Teollisuuden Voima Oy asked to study these phenomena by the developed OL3 model. In the study, 12 transients were analyzed. The basic transient cases were the house load and the turbine trip. The varied parameters in the simulations were the reactor power and the cooling water temperature of the condenser.

Eight different kinds of house load transients were analyzed. The worst case was the transient where the reactor was at full power and the cooling water temperature of the condenser was 25 °C. The total integrated mass flow from three parallel low pressure preheaters to three low pressure turbines has been presented as an example in Figure 4.

Four different kinds of turbine trip transients were analyzed. The worst case was the transient where the reactor was at 105 % power and the cooling water temperature of the condenser 0 °C. The total integrated mass flow from three parallel low pressure preheaters to three low pressure turbines has been presented in Figure 5.

Figure 4: House load, total reverse mass flow of three parallel low pressure preheaters

Figure 5: Turbine trip, total reverse mass flow of three parallel low pressure preheaters
In both transients, the reverse steam flow to the low pressure turbine took about 130 seconds. In the first transient, the integrated mass flow was 934 kg and in the second one 1139 kg. In both transients, less than 10% of the total steam came from the hot water of the drain pipe lines. The most of the steam came from the condensate of the heat exchangers.

By means of the simulation model, the plant owner has the possibility to estimate the risk to the blade failure more extensively in different kinds of transients, if needed.

7 LOSS-OF-COOLANT ACCIDENT

7.1 New CCFL Correlation

APROS has a sophisticated thermal hydraulic library including the homogeneous three-equation model as well as separated phase flow five- and six-equation models. The existing six-equation model is based on one-dimensional conservation equations of mass, momentum and energy. The six-equation model contains also models for the simulation of non-condensable gases.

Due to layout of main coolant line in- and outlets to the reactor pressure vessel, special CCFL correlations for a downcomer and an upper tie plate are needed in the analyses of loss-of-coolant accidents of EPR-type reactor. In the development, work the Glaeser CCFL correlations were implemented in APROS [3].

![Simulation model of UPTF for validation CCFL correlations](image)

The developed correlations were validated by calculating several downcomer and upper tie plate test cases of the Upper Plenum Test Facility (UPTF). The results of the downcomer and tie plate experiments show that the calculated values are close to the measurements. In the RUN 200/III, the relative difference is large but compared to the total injection mass flow the difference is not significant. In the validation of the tie plate experiment, the same water level 15 cm above the tie plate has been used.
Table 1: Data of the calculated downcomer experiments

<table>
<thead>
<tr>
<th>Run No.</th>
<th>Steam into DC (kg/s)</th>
<th>ECC flow into DC (kg/s)</th>
<th>Down flow calculated (kg/s)</th>
<th>Down flow experiment (kg/s)</th>
<th>ECC water subcooling ∆T (°C)</th>
<th>Pressure (kPa)</th>
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<td>861</td>
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Table 2: Data of the calculated tie plate experiments

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<th>ECC flow into upper plenum (kg/s)</th>
<th>Down flow calculated (kg/s)</th>
<th>Down flow experiment (kg/s)</th>
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7.2 Cold Leg Large Break Loss-of-Coolant Accident

The developed OL3 model has been used in the analysis of LBLOCA guillotine break (2ALOCA). For the analysis work some modifications were needed in the nodalizations of the pressure vessel and core model.

The following modifications were made to the downcomer, lower and upper plenum and reactor core:

- the downcomer and the first part of the lower plenum were divided into eight sectors
- in the rest of the lower plenum four sectors were used
- each downcomer sector was divided into 12 nodes in the axial direction
- all nodes of the sectors were connected to the neighbour nodes by the cross flow branches
- the upper plenum were divided into the centre flow channel and eight surrounding flow channels
- all the layers were divided into four nodes in the axial direction
- the centre and surrounding sector nodes were connected by the cross flow branches
- the reactor core was divided into eight channels based on the fuel power level
- the fuel rods were connected to the different fuel channels
- the hot fuel rod was included into the hot assembly fuel channel as one fuel rod
- the flow channels were divided into 40 thermal hydraulic volumes axially
- the nodes of the reactor core were connected by the cross flow branches
- each thermal hydraulic volume had 20 separate heat structures nodes

In the analysis of the break, the location of the leakage was in the cold leg of the loop one between the reactor coolant pump and the reactor pressure vessel.

At the initial condition of the accident case, the emergency core cooling system of the loop one injected the water directly into the break. A single failure was assumed to the check valve located in the loop three, which prevented the pump injection to that loop. Loss of offsite power was assumed and the diesel for the low and medium head safety injection pumps of the loop two was in addition assumed to be in the maintenance thus hindering water supply with these pumps. The accumulator connected to the loop two was able to supply water. The containment pressure was supplied to the model as a boundary condition value and the plant owner defined the function of the gap conductance used in the analysis. Based on the above assumptions, two accumulators connected to the loops two and four as well as one low head and one medium head safety injection pump connected to the loop four were available in the selected 2ALOCA case.

The results of the calculated accident showed that the acceptance criteria are fulfilled in the analyzed accident.

![Figure 7: LBLOCA, maximum cladding temperatures](image-url)
8 CONCLUSIONS

The developed OL3 simulation model includes the description of the primary and secondary circuits, electrical systems as well as automation and protection systems. This makes it possible to study the behaviour the whole power plant unit by one simulation model. The multifunctional simulator features of APROS e.g. to use the same software in engineering and safety analyses makes the utilization of the OL3 simulation model more effective and economical.

By means of APROS and the developed OL3 simulation model, the plant owner has an opportunity to make independent safety analyses. This increases the reliability of the simulation results of the complicated accident situations.

REFERENCES


Licensing Process For Power Uprate - Mexican Experience

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ABSTRACT

The licensing process established by a regulatory body for a power uprate in a nuclear power plant is presented. This reflects the experience for the case of the National Commission on Nuclear Safety and Safeguards of Mexico for the Laguna Verde Nuclear Power Plant. There is a description of the regulatory framework, the attributions of the regulatory body, the terms and definitions of power level for safety analysis. Furthermore, there is a discussion on the types of power uprate according to the magnitude of the change, which includes the detailed information required to the operator, in order to be assessed by the regulator. There is a summary on the experience obtained by the regulatory in this type of process, such as the power uprate permission given to the utility for a five percent increase. The technical issues are the review of the accident and transient analysis, changes to technical specifications for operation, inspection of the testing of systems, equipment and components related to safety, review of the post-modification tests and verification of fulfilling criteria. It also describes the Safety Evaluation Report and of the Final Decision and Recommendation to the Secretary of Energy, in order to issue the change to the License of Operation.

1 INTRODUCTION

The National Commission on Nuclear Safety and Safeguards (CNSNS) is the regulatory body for nuclear matters in Mexico, and is part of the Secretary of Energy. It was created in 1979 as part of a reorganization of the national nuclear administration, along with the National Institute for Nuclear Research. Before that time, both regulation and research activities were under one and only organization. Currently in Mexico, there are two power generating reactors at Laguna Verde Nuclear Power Plant (LVNPP), these are of the boiling water type (BWR) and both combined produce 1350 electrical megawatts. Both reactors are managed by the Federal Commission of Electricity, which is the state owned utility for electricity generation. The first reactor at LVNPP was authorized to operate in 1991; therefore, it has 18 years of operating experience. The second reactor went on line in 1995. The Secretary of Energy, under the technical guidance of the regulatory body, was the institution that granted both license of operation. In 1999, both reactors had their license
amended, in order to change conditions of operations for a power uprate of 5 percent of the original power. Currently, work is being done by the utility, in order to change the license of operation and obtain a power uprate change up to 15 percent more of the original rated power.

2 POWER UPRATE PROCESS

This section relates to the definition of power uprate, the types of power uprates that exists, the experience of the regulatory and the utility, the legal framework and the regulating process to obtain a licences amendment.

2.1 The importance of the maximum power level in a reactor

The maximum power level is, due to its importance, written in the operating license of a nuclear reactor. The regulator assesses and determines on this parameter, because the safety analysis must take it into account in order to demonstrate the reactor safety. The main documents, besides the license of operation, where this parameter appears are the Technical Specifications for Operation (TS), and the Final Safety Analysis Report (FSAR). Therefore, any change in the maximum power level of a nuclear reactor implies an amendment in these documents, and it is controlled by a detailed process that requires a set of analysis and tests, in a similar manner, keeping proportions, of an initial operating licensing of a nuclear reactor.

2.2 Types of power uprates

There are three types of power uprates. They depend on the complexity of the safety analysis, and the extent of the modifications of the equipment, systems and components; whether they are in nuclear steam supply system or the balance of plant. First, there is the Measurement uncertainty recapture power uprates for those less than 2 percent uprates, which are bases in the improvement of the feedwater flow measurement in order to calculate more precisely the reactor power. The second is the stretch power uprates for those changes that are less than 7 percent; in this case, small modifications, especially in the instrumentations set points, are introduced. There has to be a specific safety analysis for the operating margins. Lastly, there is the extended power uprate, this are the more complex, and require important modifications, specifically, in the balance of plant components, such as the main turbine, condensate pumps, and the generator, it also requires the licensing of advanced design fuel.

2.3 The regulatory process

The change in the maximum power level of a nuclear reactor is controlled by a regulatory process, with the exception for the case of the less than 2 percent uprate. This process is equivalent to the initial licensing of a nuclear reactor, in the sense of the detailed assessment and inspection process carried out by the regulatory body, on the safety analysis and testing done by the utility, in order to obtain the authorization.

The regulatory body regulatory responsibilities, as they are established in the nuclear law, are: to review, to assess and to authorize the basis for the siting, design, construction, operation, modification, ends of operations, closing and decommissioning of nuclear installations. In the case of an amendment of the operating license for a change in the reactor maximum power, the regulatory body gives its technical opinion to the Secretary of Energy, which is really a protocol, for the official authorization. The technical personnel of the regulatory body are qualified to carry out activities such as the evaluation and inspection of
nuclear and radiological installations. The Commission has three technical divisions: the nuclear safety division, the radiological protection division and the technical support division. For nuclear safety issues, the nuclear safety division is divided in three departments: the evaluation department, the inspection department and the enforcement and new regulation department.

The license process for a power uprate starts with an official notification from the licensee to the regulatory body that an uprate project will be developed; and a request, also to the regulatory body, for the reviewing of the available information. At this point, all generic documentation is provided to cover the general aspects and the scope of the change; the utility also sends a prospective calendar of activities related to the development of all specific safety analyses and system’s testing program. Once the generic information is reviewed by the regulatory body, the licensee starts to provide specific studies and analysis as they are applied to the plant. They contain detailed safety analysis with the new parameters, and include for example, the transient thermal hydraulics analysis, the design basis accident analysis, and the affection to other studies such as the station blackout analysis. A crucial point in the process is when the licensee sends to the regulatory body the license amendment official request. This will be done when the utility considers that most of the work of the uprate licensing process has been covered.

After the assessment of the above mentioned information, the regulatory body personnel generates a set of questions; later on, a round of questions and answer process starts, and the inclusion of the external operating experience, in the process, is a must, specially that information that comes from other power plants of similar design.

Once the questions and answer process is finished, the inspection process starts; for example the reactor’s operator training at the new plant’s conditions; the physical modifications of systems, equipments and components related to safety; and the verifications of the fulfilsments of acceptance criteria of all testing related to the process.

After the testing phase is concluded and if the results fulfil acceptance criteria, the regulatory body elaborates the safety evaluation report and writes it final determination. Both documents are sent to the Secretary of Energy; which is in charge of issuing the authorization of the amendment of the license of operation with the new value for the maximum power level.

### 2.4 The regulatory experience

The regulatory body has the experience of making safety assessments of issues related to amendments to the operation license, review of license conditions and changes to the technical specifications. Currently, the evaluation of the Improved Technical Specifications is in process. For the case of the reactor operating conditions, the following changes were reviewed: Use of operative flexibilities: increased core flow up to 7 percent of the rated core flow, extended load line which allows to be at 100 percent rated power with 87 percent core flow, maximum extended load line which allows to be at 100 percent rated power with 81 percent of core flow.

For the case of power uprate, in November 1995, the utility provided to the regulatory body the generic documentation for a power uprate of 5 percent of the original rated power for both units of Laguna Verde NPP. In July 1998, the utility sent the specific analysis and a
year later requested officially the license amendment. The regulatory body carried out the process of assessment and inspection for the safety issues and by December 1999 issued its approval for the license amendment. This power uprate was made by increasing steam flow, without changing the pressure of the reactor pressure vessel (RPV); this method avoided the modification of other equipment, systems and components. The original capacity of the balance of plant systems was utilized, in order to supply additional steam flow to the main turbine, within the design and analyzed parameters of the FSAR. There were changes to the technical specifications regarding the water level instrumentation of the RPV.

The safety assessment methodology used to license the 5 percent power uprate was based on NUREG-800 “Standard Review Plan” [1] developed by the United States Nuclear Regulatory Commission (USNRC). Other supporting documents were USNRC’s Regulatory Guides, Generic Letters, Bulletins and Information Notices. It has to be taken into account that the vendor of the nuclear reactor is from the United States, and as a policy, the Mexican regulatory body keeps in mind that, for nuclear issues, the vendor, first, has to fulfil all regulatory requirements of its country of origin. Some information of the international operating experience was also obtained from the Incident Reporting System administered by the International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency of the Organization for Economic Co-operation and Development (NEA/OECD).

The inspection process was carried out by personnel of the Inspection Department of the Nuclear Safety Division, with the support of personnel of the Evaluation Department. Special attention was given to the post-modification testing of the level instrumentation, and the nuclear engineering related activities. Once all safety requirements were fulfilled, the Safety Evaluation Report (SER) was written, along with the recommendation to the Secretary of Energy.

2.5 Future challenges

Currently the utility has informed to the regulatory body that an extended power uprate project is in course. The utility will request in the near future an amendment to its operating license, in order to change the maximum power level to 15 percent more of the original rated power. This will mean that some systems, equipment and components of the balance of plant will be modified. There will also have to request the approval of nuclear fuel with higher power density, and also they will have to validate new computer codes, based in best estimate methodologies, in order to demonstrate the safety analysis, for transient and accident conditions. For that purpose, an initial planning exercise was made, in order to calculate the resources that the regulatory body will need for the evaluation and inspection process that will be carried out, once the utility starts sending the specific information. Following the experience of other regulatory bodies, such as the NRC, it was calculated that an initial review will take around four months; the detailed evaluation will take approximately twelve to fourteen months; the total process will require at least two years; additional technical personnel will be needed, at least two specialist for each area of revision, the number of areas of revision are seventeen; therefore, the technical personnel required will be thirty four. The profile of the technical personnel assigned to the evaluation and inspection activities is to have experience in the licensing process. Currently, as is the case for the Improved Technical Specification evaluations, theses resources are being provided in part by the National Institute for Nuclear Research and by a public university.
As the methodology to be used for the safety evaluation, regulatory body personnel is currently translating to Spanish and adapting the NRC’s procedure to evaluate power uprates, named, “Review Standard for Extended Power Uprates” [2], in this document specific aspects of the evaluation process are described. Still, the most challenging issue is obtaining qualified personnel to carry out the evaluations.

3 CONCLUSIONS

The main characteristics of the licensing process for power uprates in Mexico were presented. This reflects the experience for the case of Mexican regulatory body for the Laguna Verde Nuclear Power Plant. There was a description of the regulatory framework, the attributions of the regulatory body, the terms and definitions of power level for safety analysis. There was a discussion on the types of power uprate according to the magnitude of the change, which includes the information required to the operator, in order to be assessed by the regulator. There is a summary on the experience obtained by the regulatory in this type of process, such as the power uprate permission given to the utility for a five percent, of the original rated power, increase. The technical issues are the review of the accident and transient analysis, changes to technical specifications for operation, inspection of the testing of systems, equipment and components related to safety, review of the post-modification tests and verification of fulfilling criteria. Finally, a brief discussion is presented on the challenges ahead for the regulatory body, for the case of a new request from the utility to change the maximum power level for an additional 15 percent of the original rated power.

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Operational Safety
Neutronic and Thermalhydraulics Coupled Calculations to Model a LOCA for Advanced CANDU Fuel Designs

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ABSTRACT

As Romania has chosen to increase its nuclear capacity by other 2 CANDU-6 type nuclear units (excepting the two already in operation), the Romanian nuclear community should be interested in both economic and safety enhancing of this type of nuclear reactors. The NUSIM-ACF - a national excellence research project dedicated to nuclear safety increasing at the use of advanced CANDU fuel designs - is ongoing (Institute for Nuclear Research Pitesti is the project leader) and its actual working stage is focused on nuclear safety calculations. The purpose of the paper is to simulate several Loss Of Coolant Accident scenarios (LOCAs) at a CANDU-6 Nuclear Power Plant (NPP) in the case of using advanced CANDU fuel designs along with CANDU Natural Uranium fuel. The main goal was to perform neutronics and thermalhydraulics coupled calculations in order to estimate more accurately the effects of coolant density and temperature changing on the reactor power decreasing during the chosen transients. The computer codes chosen for coupling purpose were the 3D neutronics DIREN_MG code (developed in INR Pitesti) and the AECL thermalhydraulics CATHENA code. The calculation methodology is based on the Improved Quasi-Static (IQS) method for solving the time-dependent diffusion equation. A specific routines were developed and implemented in the DIREN_MG code in order to carry out neutronics and thermalhydraulics coupled calculations and also, to allow for a cell calculation for each bundle at each time step of the transient. The flux amplitude, which directly determines the reactor power, was pursued and represented as function of time along with the maximum channel and bundle powers. General considerations regarding the fuel integrity of the most heated element from studied fuel designs are presented. The advanced CANDU fuel designs with burnable absorbers in both CANFLEX and standard 37 rods geometries showed a better behavior during the studied transients than the "classical" CANDU one, proving their claimed nuclear safety enhancements.

1 INTRODUCTION

One of the most spread nuclear safety analysis is the Loss of Coolant Accident (LOCA) analysis. We intend to simulate several LOCA at a CANDU-6 type Nuclear Power Plant (NPP) in the case of using different advanced CANDU fuel designs. The intervention of
Shutdown System no. 1 (SDS1) will be modeled until the transient is terminate by this
action. The CANDU® (CANada Deuterium Uranium) and ACR™ (Advanced CANDU Reactor)
are pressure tubes heavy water moderated reactors. CANDU is heavy water and ACR is light
water cooled. Two CANDU-6 reactors are in operation at Cernavoda, Romania, since 1996
and 2007, respectively. Nowadays, AECL promotes the ACR-1000 design [1],[2] - a
Generation III+ nuclear power reactor and also, a step towards the Generation IV inherently
safe nuclear energy systems. The point kinetics quasistatic approximation will be used to
model chosen LOCA transients and the SDS1 action in time steps. The calculations will be
carried out using DIREN_MG code [3], a 3D neutronics multigroup diffusion code developed
in the Institute for Nuclear Research (INR) and also the AECL CATHENA¹ code. The
connection between DIREN_MG neutronics code and the Thermal-Hydraulics Code (THC)
asked for a coupling program able to read the coolant densities and temperatures for each
considered passes through the reactor core at every time step of the transient. These values
will be used as input data in the WIMS² cell code [4], to perform calculations for each fuel
bundle at one time step. The connection task through the coupling routines represents our
original contribution in the paper. As the SDS1 rods are dropped into reactor core in a short
time (under 2 seconds) we used 14-15 time steps (with a total duration of about five
seconds) at which we took "snapshots" of the reactor core to calculate main physics amounts
of interest for nuclear safety (i.e. neutron flux amplitudes, maximum channel and bundle
power). To reach the secondary objective of the paper, a simplified safety analysis regarding
fuel integrity was done. The maximum bundle (thermal) power was used to estimate the
corresponding accumulated energy (heat) per fuel mass unit. This value was compared to
critical value (taken from specific literature) in order to appreciate if the fuel integrity is kept.

2 METHODOLOGY OUTLINES

The CANDU reactors are provided by two independent, fully capable shutdown
systems, SDS1 and SDS2 which can shut the reactor down in the event of a so-called
"maximum credible accident" or Design Basis Accident (DBA). A rupture in the inlet or outlet
header of the Primary Heat Transport System (PHTS) can be considered as DBA when
SDS1 is to be intervening to shut the reactor down as soon as possible. The SDS1
intervention is started by tripping logic of Platinum prompt detectors. Then, the ShutOff Rods
(SOR) are dropped into the core under the two combined forces: the gravitation and elastic
force of pre-stressed springs. Their final position schema is shown in Figure 1.

Figure 1: The final Shutoff Rod positions in CANDU-6 reactor core

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¹ CATHENA = Canadian Algorithm for Thermal-Hydraulic Network Analysis
² WIMS = Winfrith Improved Multigroup Scheme
To model the SDS1 action during such a transient when SORs rapidly change their positions it needs solving the time-dependent multigroup diffusion equation. It is difficult to solve such an equation using finite differences standard techniques. The alternative is using of the so-called improved quasistatic multigroup approximation \([5],[6],[7]\). The equations to be solved are (1) and (2), as bellow:

\[
\frac{1}{v_g} \frac{\partial}{\partial t} \Phi_g(r,t) = \nabla D_g(r,t) \nabla \Phi_g(r,t) - \sum_g^G \Sigma^{g\rightarrow}(r,t) \Phi_g(r,t) - \\
- \sum_{g'} \sum_g \Phi_g(r,t) \Phi_{g'}(r,t) + \sum_{g'} \sum_{g''} \Phi_{g''}(r,t) \Phi_{g'}(r,t) + \\
+ (1 - \beta(r)) \frac{1}{k_0} \chi^p \sum_{g'} \sum_{f} \sum_{g''} \Phi_{g''}(r,t) \Phi_{g'}(r,t) + \chi^{g,d} \sum_i \lambda_i C_i(r,t) \tag{1}
\]

\[
\frac{\partial}{\partial t} C_i(r,t) = -\lambda_i C_i(r,t) + \beta_i(r) \frac{1}{k_0} \sum_{g'} \sum_{f} \sum_{g''} \Phi_{g''}(r,t) \Phi_{g'}(r,t) \tag{2}
\]

The used variables have the following significances:

\(v_g\) = neutron speed of energy group \(g\) in cm/s, given by the equation (3):

\[
\frac{1}{v_g} = \int_{E_1}^{E_2} dE \frac{\Phi(E, r, t)}{\nu(E)} \left[ \int_{E_1}^{E_2} dE \Phi(E, r, t) \right] \tag{3}
\]

\(D_g(r, t)\) = time and space dependent diffusion constant in group \(g\),
\(\Sigma^e_g(r, t)\) = absorption cross section in group \(g\),
\(\Sigma_{g\rightarrow g'}(r, t)\) = scattering cross section from group \(g\) to \(g'\),
\(V \Sigma^e_f(r, t)\) = yield cross section,
\(\Phi_g(r, t)\) = neutron flux in group \(g\) at time \(t\),
\(\chi^{g,d}\) = delayed neutron fraction having energy in group \(g\), the same overall delayed neutron fractions,
\(\chi^p\) = prompt neutron fraction having the energy in group \(g\);
\(\beta_i(t)\) = delayed neutron fraction in precursor group \(i\), per fission act,
\(\beta_i\) = sum overall precursor groups, (it depends on position),
\(k_0\) = multiplication constant at time \(t=0\) (before the transient starting)
\(C_i(r, t)\) = delayed neutron precursor concentration in group \(i\),
\(\lambda_i\) = decay precursor constant in group \(i\).
The technique used to solve the time dependent and coupled equations (1) and (2) is to break the flux into a so-called amplitude function, $A(t)$ - the same over all energy groups and a space-dependent slowly time-dependent group spatial form function, $\Psi(r,t)$:

$$\Phi_g(r,t) = A(t) * \Psi_g(r,t) \quad (4)$$

Finally, we will find out the neutron flux behaviour based on amplitude calculations. The kinetic amplitude $A(t)$ gives the trend of maximum power pulse and also the maximum channel and bundle powers - the most important nuclear safety amounts to be pursued.

The steps to be followed in order to model SDS1 intervention are summarized below:

1) solving the static diffusion equation at $t=0$ and initial reactivity device positions (all SORs extracted) to obtain the initial flux distribution and multiplication constant $k_0$, which will be used in time dependent equation;

2) solving the adjunct static diffusion equation at $t=0$ to obtain the adjunct flux used as weighted function at kinetic parameter calculation;

3) solving the time-dependent diffusion equation at the next time step $t=T$ actuating the reactivity device positions (SORs start dropping in the core). A spatial flux form is obtained at time $T$;

4) using the spatial flux shape, the amplitude $A(t)$ can be calculated. The steps 2) and 3) have to be repeated until the $A(t)$ convergence criterion is reached.

We implemented this above procedure in the DIREN_MG code developed during the last years in our institute.

On the other side it is very well known that thermalhydraulics parameters like coolant density and temperature vary during a transient. As result, the nuclear cross section are modified with respect to the temperature. The implemented procedures into DIREN code source take into account for these variations and the first type of DIREN_MG calculation was generating the temperature-dependent cross section tables. Then, the effective kinetics calculation was started and one WIMS calculation was performed for each of the 4560 CANDU fuel bundles at each time step considered in the transient. The 3D neutron flux distribution at a time step was used only to calculate corresponding thermal power, it wasn't back into cell lattice. A WIMS calculation for each fuel bundle was done only to take into account for changing in a specific supercell configuration due to the SORs advancing and also due to the changing in nuclear cross section resulting from temperature effects. The key neutronic amount pursued was the flux amplitude along with maximum channel and bundle power which were represented as a function of time in Cap.4.

Regarding the CANDU-6 core model, we used a well established 3D equilibrium core model from Cernavoda Unit 1 NPP, [8],[9].

We performed 3D DIREN_MG point kinetic calculations for 3 LOCA levels: 15%, 25% and 35% RIH (Rupture Inlet Header).

3 CONSIDERED FUEL DESIGNS

Table 1 summarizes the fuel types characteristics while the Figure 2 shows the current ACR-1000 fuel lattice cell, taken from [1]). Three fuel bundle geometries was used in calculations, i.e.: the CANDU-37 rods bundle, the CANFLEX-43 rods bundle and a modified CANFLEX-43 rods bundle, see Figures 3 and 4. The last, was applied to ACR fuel channel, but it must be pointed that the central pin diameter along with the outer ring radii were calculated from geometric considerations. The alluded (and used in calculations) ACR fuel design doesn't represents fully the current AECL ACR-1000 fuel design, but it is based on the known ACR-1000 fuel design characteristics (available from [1]). On the other side, it is important to stress that the paper aims to have an educational purpose, since the performed studies will help the main author to accomplish his PhD thesis. We only intend to provide a
better insight and promotion of the ACR concept among the university environment. Since ACR-1000 is a registered trademark of AECL and also, to avoid undesirable confusions, the advanced CANDU fuel design proposed in this paper (and based on Ref. [1],[2]) will be simply labeled as "ACR".

Table 1: The five fuel designs used in calculations

<table>
<thead>
<tr>
<th>No.</th>
<th>Fuel type</th>
<th>Geometry</th>
<th>Composition by Inner Rings</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>ACR</td>
<td>modified CANFLEX® 43 rods</td>
<td>R1(^a): a mixture of 1%Gd-157, 5%Dy-164, 94%Zr-91 R2,R2,R4: 2.4% LEU</td>
</tr>
<tr>
<td>2</td>
<td>CANFLEX-LVRF</td>
<td>CANFLEX 43 rods</td>
<td>R1: NU+8.8%Dy R2: NU+1.9%Dy R3: 2.7% SEU R4: 2.1% SEU</td>
</tr>
<tr>
<td>3</td>
<td>CANDU-LVRF</td>
<td>CANDU-37 rods</td>
<td>R1: NU+10%Dy R2: NU+2%Dy R3: 1.92% SEU R4: 1.35% SEU</td>
</tr>
<tr>
<td>4</td>
<td>SEU-43</td>
<td>CANFLEX 43 rods</td>
<td>0.96% SEU</td>
</tr>
<tr>
<td>5</td>
<td>CANDU</td>
<td>CANDU-37 rods</td>
<td>NU</td>
</tr>
</tbody>
</table>

\(^a\)R1 to R4 denote the fuel rod rings starting from center, see Fig.2

Some additional explanations are needed concerning the studied fuel projects. The CANDU-6 cell contains 37 identical fuel rods with Natural Uranium (NU) and heavy water (both as coolant and moderator), see Figure 3. Figure 3 illustrates comparatively (at the same scale), the used CANDU-6 and ACR fuel design geometry as they appear in the mixture maps generated by DRAGON code [10], in an earlier paper, [11]. The CANFLEX fuel design consists of 43 fuel rods (the 8 inner rods are identical and thicker than the outer 35), see Figure 4. This geometry was applied to our Romanian SEU-43 fuel design [12], (Slightly Enriched Uranium, 0.96\% U235). The ACR cell contains fuel type #1 from Table 1, light water as coolant and heavy water as moderator. The central rod (R1 in Table 1) was assumed to contain 5\% Dysprosium-164, 1\% Gadolinium-157 and 94\% Zirconium-91. It

\(^3\) CANFLEX® is a registered trademark of AECL and the Korea Atomic Research Energy Institute

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contains no fissile material, as Ref. [1] stated. The other 42 rods (R2 to R4) contain LEU 2.4%. The mixture between Dysprosium, Gadolinium and Zirconium in the central rod is intended to flatten the power distribution over the cell. It also leads to lower the Coolant Void Reactivity (CVR).

The ACR pressure tube, the CO$_2$ gap and the Calandria tube are thicker than the standard CANDU ones, as Figure 3 reveals. As the current Low Void Reactivity Fuel (LVRF) design wasn't available to us, another older LVRF designs (already used in [13]) were chosen to emphasize the effect of burnable absorbers along with the using of SEU in a CANDU-6 core. The fuel #3 and #4 contain NU in the fuel rings R1 and R2, together with greater amounts of Dysprosium (1.9 to 10%) and also SEU in R3-R4, with various U235 enrichments (from 1.35 to 2.73%).
In order to find out the global influence at using of ACR fuel, a simplified ACR-1000 DIREN_MG core model has started to be developed. Some additional information is still needed. Based on the real Cernavoda Unit 1 core model [8], we elaborated a simplified ACR-1000 DIREN_MG core model with the following main assumptions:

- 24 cm Lattice Pitch and 3200 MW thermal power;
- number of channels were increased from 22 to 26;
- mesh intervals were increased from 56 to 64 in order to obtain the same core size (760 cm);
- we corrected the X axis reactivity device (RD) positions forcing to cross the nearest mesh line;
- the Y axis positions were kept unchanged (since the majority of reactivity devices are vertical and ACR-1000 core has the same size as CANDU-6, they should attain the same vertical depth);
- Z reactivity device positions were kept unchanged (ACR-1000 bundle length is the same as CANDU-6 one);
- we used WIMS code to calculate cross sections for ACR fuel and reflector and, finally, (maybe the drastic approximation), all RD incremental cross sections were kept unchanged.

4 RESULTS AND DISCUSSIONS

We performed 3D DIREN_MG kinetic quasistatic calculations for 3 LOCA levels: 15%, 25% and 35% RIH (Rupture Inlet Header). A Cernavoda Unit 1 equilibrium core model [8],[9] was used for fuel types #2 to #5. The simplified DIREN_MG ACR-1000 core model was applied to ACR fuel design (#1 in Table 1). The amplitudes supplied by DIREN code are shown in Figures 5, 6 and 7.

![Amplitude Evolution (LOCA 15% RIH)](image)

Figure 5: The Amplitude Evolution during a LOCA 15% RIH for different CANDU Fuel Designs (DIREN_MG Calculations)
All the curves corresponding to the one fuel type design have similar shape in all three LOCAs and the general characteristic is that the CANDU fuel type (i.e. NU in 37 rods geometry) shape covers the other four curves. The maximum amplitude value was obtained at the ninth time step (1.246 s, see Table 2) after the transient beginning and it is 1.914. We remark the better behavior of all advanced CANDU fuel designs considered in calculations. The best behavior pertains to the older CANFLEX-LVRF fuel design which benefits of differentiated Dysprosium and SEU through all the fuel rings. ACR fuel also made a good figure, its maximum amplitude being encountered at 15% RIH, a little bit surprisingly. Why a smaller LOCA leads to a greater flux amplitude? The explanation can be given by the using of the same time steps as for CANDU fuel type and also the same starting moment of the SDS1 action. The advanced CANDU fuel designs are less "reactive" than the CANDU one (due to the burnable poisons, of course). As result, they have not enough time that the flux
amplitudes rise to the tripping values. A fine analysis is still needed to set independent tripping moments for each fuel. Anyway, the very well visible safety margin differences are clearly in favor of the advanced CANDU fuel designs.

![Graph showing cell void reactivity](image)

**Figure 8:** Cell Void Reactivity for considered CANDU Fuel Designs, (WIMS Calculations performed in [13])

**Table 2:** The Neutron Flux Amplitude Values in the case of 35% RIH

<table>
<thead>
<tr>
<th>Step #</th>
<th>Time (s)</th>
<th>ACR</th>
<th>CANFLEX-LVRF</th>
<th>CANDU-LVRF</th>
<th>SEU-43</th>
<th>CANDU</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0.000</td>
<td>1.000</td>
<td>1.000</td>
<td>1.000</td>
<td>1.000</td>
<td>1.000</td>
</tr>
<tr>
<td>2</td>
<td>0.100</td>
<td>1.014</td>
<td>0.950</td>
<td>0.962</td>
<td>1.019</td>
<td>1.020</td>
</tr>
<tr>
<td>3</td>
<td>0.200</td>
<td>0.996</td>
<td>0.832</td>
<td>0.868</td>
<td>1.069</td>
<td>1.077</td>
</tr>
<tr>
<td>4</td>
<td>0.291</td>
<td>0.961</td>
<td>0.747</td>
<td>0.771</td>
<td>1.138</td>
<td>1.156</td>
</tr>
<tr>
<td>5</td>
<td>0.530</td>
<td>0.880</td>
<td>0.668</td>
<td>0.670</td>
<td>1.380</td>
<td>1.404</td>
</tr>
<tr>
<td>6</td>
<td>0.730</td>
<td>0.838</td>
<td>0.637</td>
<td>0.648</td>
<td>1.604</td>
<td>1.621</td>
</tr>
<tr>
<td>7</td>
<td>0.821</td>
<td>0.830</td>
<td>0.630</td>
<td>0.645</td>
<td>1.702</td>
<td>1.718</td>
</tr>
<tr>
<td>8</td>
<td>0.966</td>
<td>0.820</td>
<td>0.624</td>
<td>0.644</td>
<td>1.843</td>
<td>1.860</td>
</tr>
<tr>
<td>9</td>
<td>1.246</td>
<td>0.768</td>
<td>0.608</td>
<td>0.643</td>
<td>1.837</td>
<td>1.914</td>
</tr>
<tr>
<td>10</td>
<td>1.646</td>
<td>0.668</td>
<td>0.560</td>
<td>0.617</td>
<td>1.501</td>
<td>1.601</td>
</tr>
<tr>
<td>11</td>
<td>2.226</td>
<td>0.530</td>
<td>0.476</td>
<td>0.550</td>
<td>1.055</td>
<td>1.101</td>
</tr>
<tr>
<td>12</td>
<td>2.976</td>
<td>0.342</td>
<td>0.335</td>
<td>0.398</td>
<td>0.597</td>
<td>0.628</td>
</tr>
<tr>
<td>13</td>
<td>3.936</td>
<td>0.268</td>
<td>0.272</td>
<td>0.329</td>
<td>0.420</td>
<td>0.445</td>
</tr>
<tr>
<td>14</td>
<td>5.076</td>
<td>0.125</td>
<td>0.124</td>
<td>0.143</td>
<td>0.155</td>
<td>0.162</td>
</tr>
</tbody>
</table>

To sustain the DIREN_MG results we reproduced in Figure 8 (from [13]) the cell void reactivity calculations for the same CANDU fuel designs as in this paper. The Figure 8 illustrates the void reactivity estimated by simply reducing the coolant density through the WIMS calculations. It can be easily seen that the cell void effects are similar to those from full core calculations. Concretely, while the CANDU-6 and SEU-43 fuels show a positive void effect, the advanced CANDU fuel designs show a negative one throughout the void fraction range (from 0 to 100% voiding).
The maximum channel and bundle power peaks estimated by the DIREN_MG code during the studied LOCAs are presented in Figures 9 and 10.

![Figure 9: Maximum Channel Powers in respect of fuel type (DIREN_MG calculations)](image1)

![Figure 10: Maximum Bundle Powers in respect of fuel type (DIREN_MG calculations)](image2)

The behavior of the maximum channel and bundle powers during the three studied transients is similar to that of flux amplitudes. More precisely, once again fuel type #2 (CANFLEX-LVRF) shows the smallest values while fuel type #4 (SEU-43) supplies the highest values. Therefore, we won’t insist on this aspect.

Regarding fuel integrity, Ref. [14] showed that a heat accumulation in the fuel mass unit (without cooling i.e. in such a LOCA scenario) for 9-18 seconds at nominal power (that means more than 850 J/g (Joules per gram of fuel) can damage the fuel pellets. We simply evaluated the accumulated heat in the fuel mass units in the most dangerous transient. It corresponds to the #4 fuel type (SEU-43). Assuming that the maximum bundle power is maintained for about 5 seconds until the flux amplitude (and, consequently, bundle power) drops ten times, accumulated heat in one fuel bundle (of ≈20 kg) is about 1800 kW\(\times\) 5 s = 9000 kJ. That means a value of 9000\(\times\)10\(^3\)J/20\(\times\)10\(^3\)g= 450 J/g which is only a half of critical value. Remember that we assumed that the maximum bundle power (peak) is maintained for all the five seconds (until the flux amplitude is dropped towards zero). On the other hand, a more precisely calculation of accumulated heat in this five seconds interval may be done through integrating the power shape in due time interval (i.e. the area under the amplitude.
shape, see any of the Figures 5 to 7), which leads to a significantly lower value of accumulated heat (it easily to see that our gross estimation is based on a rectangle area (Max. bundle power multiplied by the time interval). Therefore, we can state that fuel integrity is kept for any fuel design considered in our study, of course, as results of prompt action of Shutdown System no. 1. Finally, a theoretical classification of analyzed fuel designs based on their safety performance can be done: the most secure seems to be the fuel designs which benefit for differentiated burnable absorbers and fissile material on fuel rings, the fuel types #2 and #3. They are followed by the ACR fuel with (5% Dy + 1% Gd + 94% Zr) and no fissile material in central pin but fueled by LEU 2.4% in the outer rings. The series continues with the standard CANDU fuel design and ends with the SEU-43 fuel design.

5 CONCLUSIONS

(1) An original coupling procedure between the 3D diffusion DIREN_MG code and the AECL thermalhydraulics CATHENA code has developed in order to perform neutronics and thermalhydraulics coupled calculations.

(2) Several LOCA transients were modeled through neutronics point kinetics and thermalhydraulics coupled calculations for both CANDU-6 and ACR cores. While the CANDU-6 core model corresponds to the Cernavoda Unit 1 NPP, the ACR core model is based on available data of AECL ACR-1000 design with significant simplifications.

(3) As a general rule, all advanced CANDU fuel designs proved their inherently safe features leading to a prompt decreasing of neutron flux amplitude in all studied LOCA scenarios. In fact, for these fuel types, containing burnable absorbers (#1, #2 and #3), the flux amplitude starts to drop almost immediately after the transient initiation.

(4) The obtained results are in good agreement to earlier cell calculations where the negative cell CVR was emphasized for all advanced CANDU fuel designs (containing burnable absorber).

(5) The novelty elements of the paper are the elaboration of a first ACR core model for a 3D neutronics code developed in INR and also, the coupling with a dedicated thermalhydraulics internationally recognized code as AECL CATHENA is.

6 ACKNOLEDGMENTS

The authors thank to the colleagues from Cernavoda Unit 1 Reactor Physics Group for their references and as-built data supplied and also, to Ilie Pătrulescu-INR Reactor Physics Group responsible for valuable guiding in core modelling.

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Source Term Evaluation for a Severe Accident in CANDU Type Reactor by ASTEC Code

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ABSTRACT

The most important result of Severe Accident (SA) analysis is the source term (the radioactive sources released into the environment). The different fission products (FPs) are released from the fuel bundles through the clad rupture, transported by the coolant and deposited into different regions of the Primary Heat Transport (PHT) and containment system. A part of transported amount will be released into the environment through a containment break. Different hosts (aerosols, gas, liquids) are involved in transport and deposition phenomena. The chemistry process is taken into account in order to obtain a realistic model. The paper is intended to analyze the source term formation and structure (FPs) in CANDU type reactor taking into account the transport, chemistry, and safety system intervention, by using the ASTEC code. The FPs inventory introduced into the PHT, was estimated by ORIGEN code. The data related to the nodes’ definitions, temperatures and pressure conditions were chosen as possible as real data from CANDU loss of coolant accident sequence. The FPs distribution and chemistry, in different nodes of the PHT and CANDU Containment, were obtained by a coupled calculation SOPHAEROS-CPA-IODE. All modules are integrated in SA code ASTEC. The depositions in the different parts of the reactor drastically influence the source term. In case of CANDU our analysis shows that (excepting Xe, Kr) more than 70% of the released masses are deposited into PHT. At the same time for important FPs source term the main part of the PHT transfer to containment is deposited into the sump water. The PHT and containment action as filters for many isotopes. However the failure of the containment involves a release in the environment. The paper shows the influence of the accident conditions and of the existing uncertainty in data on structure and magnitude of the source term.

1 INTRODUCTION

The most important result of Severe Accident (SA) analysis is the source term (the radioactive sources released into the environment). The different fission products (FPs) are released from the fuel bundles through the clad rupture. The FPs are transported by the coolant, a part of them is deposited into different regions of the Primary Heat Transport (PHT) System, a part is transferred through the break into the containment. The FPs are transported by different hosts (aerosols, gas, liquids) and deposited on the containment walls,
into the sump water etc. The chemistry process must be taken into account. In case of a containment wall break (by pressure increasing) a part of the FPs are released into the environment.

The paper is intended to analyse the source term formation in the case of a postulated severe accident induced by loss of coolant (LOCA) + loss of emergency cooling (LOECCS) and loss of moderator (LOM).

In the starting phase of the accidents some suppositions replace the simulation of the core degradation.

2 SPECIFICITIES OF CANDU TYPE REACTORS

The core of the CANDU power reactor is comprised of several hundred horizontal fuel channels in a large cylindrical Calandria vessel. The Calandria vessel is surrounded by a shield tank with a large volume.

To avoid the use of a large pressure vessel, the CANDU utilizes the pressure tube concept. The fuel is loaded into the horizontal zirconium alloy pressure tubes that pass through the Calandria filled with heavy water moderator in a low pressure circuit. This tank is penetrated by several hundreds fuel channels. Each fuel channel consists of:

1. an internal pressure tube, containing the fuel and the hot, pressurized heavy water primary coolant;
2. an external calandria tube separated from the pressure tube by an insulating gas-filled annulus.

High-pressure heavy water is used as the fuel coolant. The use of pressure tubes in the reactor core allows the primary coolant system to be pressurised without the need for a massive pressure vessel. Coolant does not boil.

Because the CANDU system uses neutrons more efficiently than light water reactors, it requires less uranium for a given electrical output.

The calandria vessel contains cool low-pressure heavy-water moderator that surrounds each fuel channel. The primary coolant is distributed amongst the fuel channels by common headers and individual feeder pipes. The fuel can be natural or slightly enriched UO2 fuel bundles. The fuel bundles for Cernavoda NPP contain natural uranium oxide cladded in zirconium alloy. The CANDU system is fuelled on-load: a pair of remotely operated machines, one at each end of the reactor, simultaneously inserts fresh fuel while removing used fuel bundles. On-load fuelling results in optimum reactor flux patterns and efficient utilization of fuel; in addition, it contributes to CANDU’s high capacity factor.

At each end of the Calandria there is an end shield, which contains carbon steel balls and demineralised water in order to provide both radiation shielding and cooling.

A complete coolant circuit involves two fuel tubes and two circulation loops. The heavy water enters the reactor at a temperature of about 266 °C and exits at 310 °C. It passes from the reactor to a header, that is, a junction chamber for the coolant tubes, and then to an inverted U-tube steam generator where steam is produced and carried to the turbines. The coolant then returns to the reactor, passing in the opposite direction through an adjacent fuel tube, where it is heated again before flowing to a second steam generator. The pressuriser performs the same function in the CANDU system as it does in the PWR. Although most of the heat from the fuel is carried away by the heavy water coolant, some energy is deposited in the heavy water moderator. This is removed by the moderator coolant loop.

From the point of view of severe accident the following specificities are very important:
- horizontal orientation of fuel and cooling;
- fuel channels instead of pressure vessel;
- water near core (moderator and light water from reactor vault);
3 ASTEC CODE

The ASTEC code is dedicated for SA analysis of PWR reactors and involves a lot of models and methods. Some of them are presented in [1, 2]. The use of ASTEC at CANDU type reactors introduces many difficulties especially for the core degradation phenomena [3]. However, the code may be successfully used for simulating the main severe accident phenomena in CANDU such as FPs transport in the PHT and containment, thermalhydraulics, FPs chemistry, etc. [3]. The main important effort of the investigation of the ASTEC adaptability to CANDU NPP is performed under FP6 SARNET project.

In order to evaluate the source term for CANDU postulated severe accident, we have used the following modules of ASTEC:

- SOPHAEROS – for FPs transport, chemistry and deposition phenomena in the PHT;
- CPA – for thermal hydraulics of the containment and FPs transport, deposition phenomena;
- IODE - for the chemistry of iodine in the containment.

Initial and boundary conditions were used in order to supply the initial inventory of the releasing from fuel bundles, pressure and temperature distributions in the PHT. The SOPHAEROS-CPA-IODE calculation was performed in the coupled mode and the used version was ASTEC V1.3 R1.

4 DESCRIPTION OF THE PROBLEM

Taking into account the impossibility for the simulation of the core degradation process by the ASTEC code, the geometry of the problem consists only of the PHT and containment. The CANDU containment is similar with the containment of PWR type. The CANDU PHT has some specificities that are described bellow.

The PHT system for CANDU consists of two closed loops and as a consequence in the event of a loss of coolant accident the rate of the reactor coolant blow down is reduced. After a LOCA the loops are isolated and the intact loop inventory can be maintained. Generally, the PHT of CANDU is similar with PWR primary heat transport, but some important differences exist:

- the circulation is bi-directional (each particle of fluid goes through the core twice before it gets back to where it started);
- the presence of the fuel-channels instead of pressure vessel (each channel is individually connected to collectors (headers) above the core);
- D₂O as coolant instead of H₂O.

That means a more complicated thermalhydraulics model.

For PHT System only half of the circuit (fig.1) was simulated: 190 horizontal fuel channels connected to 190 horizontal out-feeders, then through vertical feeders to the outlet-header; the circuit continues from the outlet-header with a riser and then with the steam generator and a pump. After this pump, the circuit was broken; in this point the FPs are transferred to the containment. The containment model consists of 12 rooms connected by 14 links (fig.2). The steam and FPs are injected from PHT (through broken pipe) in the room CV760.

The data related to the nodes’ definitions, temperatures and pressure conditions were chosen as possible as actual data from CANDU NPP loss of coolant accident sequence. Temperature and pressure conditions in the time of the accident were calculated by
CATHENA code [4] and the source term of FPs introduced into the PHT was estimated by ORIGEN code [5].

Fig.1 Geometrical model for CANDU PHT (one complete loop)

Fig.2 Geometrical model for CANDU Containment
5 RESULTS AND DISCUSSIONS

The source term formation is strongly determined by the following factors:
(1) existing inventory of FPs stored in the fuel matrix and gap (dependent on the fuel type, burnup, etc.);
(2) releasing fractions from fuel to the PHT (dependent on the fuel temperature, mechanism of failure, etc.);
(3) deposition fractions in the PHT system structure (dependent on the geometry, chemistry, temperature and pressure conditions, involved phenomena, injection of additional coolant, etc.);
(4) transport and deposition in the containment;
(5) fraction of radioactive isotopes and the importance of them from environmental point of view;
(6) amount of releasing by containment failure.

In Table 1 the FPs inventory, calculated by ORIGEN code, the releasing fractions (literature data [6] and our hypothesis - the releasing is achieved in a short time after the cladding failure, but with releasing fractions of melted fuel) and released masses into PHT are presented. The releasing fractions are in accordance with the volatility class. Therefore, the total amount of FPs released into CANDU half PHT is supposed at 34.72 kg.

Table 1 FPs Release into CANDU PHT

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Core inventory [kg]</th>
<th>Releasing fractions</th>
<th>Released mass [kg]</th>
</tr>
</thead>
<tbody>
<tr>
<td>Kr</td>
<td>7.33</td>
<td>0.50</td>
<td>3.665</td>
</tr>
<tr>
<td>Rb</td>
<td>6.54</td>
<td>0.20</td>
<td>1.308</td>
</tr>
<tr>
<td>Sr</td>
<td>28.00</td>
<td>0.05</td>
<td>1.400</td>
</tr>
<tr>
<td>Y</td>
<td>11.90</td>
<td>0.05</td>
<td>0.595</td>
</tr>
<tr>
<td>Zr</td>
<td>61.60</td>
<td>0.01</td>
<td>0.616</td>
</tr>
<tr>
<td>Mo</td>
<td>43.40</td>
<td>0.05</td>
<td>2.170</td>
</tr>
<tr>
<td>Tc</td>
<td>8.64</td>
<td>0.01</td>
<td>0.086</td>
</tr>
<tr>
<td>Ru</td>
<td>35.80</td>
<td>0.05</td>
<td>1.790</td>
</tr>
<tr>
<td>Pd</td>
<td>3.96</td>
<td>0.01</td>
<td>0.040</td>
</tr>
<tr>
<td>Te</td>
<td>13.00</td>
<td>0.20</td>
<td>2.600</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Core inventory [kg]</th>
<th>Releasing fractions</th>
<th>Released mass [kg]</th>
</tr>
</thead>
<tbody>
<tr>
<td>I</td>
<td>11.90</td>
<td>0.20</td>
<td>2.380</td>
</tr>
<tr>
<td>Xe</td>
<td>81.40</td>
<td>0.50</td>
<td>40.700</td>
</tr>
<tr>
<td>Cs</td>
<td>28.00</td>
<td>0.20</td>
<td>5.600</td>
</tr>
<tr>
<td>Ba</td>
<td>36.60</td>
<td>0.05</td>
<td>1.830</td>
</tr>
<tr>
<td>La</td>
<td>22.00</td>
<td>0.05</td>
<td>1.100</td>
</tr>
<tr>
<td>Ce</td>
<td>58.70</td>
<td>0.05</td>
<td>2.935</td>
</tr>
<tr>
<td>Pr</td>
<td>14.30</td>
<td>0.01</td>
<td>0.143</td>
</tr>
<tr>
<td>Nd</td>
<td>38.70</td>
<td>0.01</td>
<td>0.387</td>
</tr>
<tr>
<td>Pm</td>
<td>3.27</td>
<td>0.01</td>
<td>0.033</td>
</tr>
<tr>
<td>Sm</td>
<td>5.25</td>
<td>0.01</td>
<td>0.053</td>
</tr>
</tbody>
</table>

Related to the interval time and releasing profile, we introduce a very simple assumption: the releasing are active between 150 and 155 s after the accident initiation, in accordance with the fuel cladding failure by temperature increasing [7].

Based on this initial conditions and assumptions the FPs deposition in the PHT and consequently the transfer to the containment through the pipe break was calculated by SOPHAEROS module.

In Table 2 the transfer fractions (transferred mass to released mass ratio) to the containment for different FPs are presented.

These fractions are calculated for each FPs including all isotopes taken into account by ORIGEN. For present analysis we have selected the most important FPs from the point of view of source term: Cesium, Iodine and Strontium.
Table 2 FPs condensation, deposition and injection in containment

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Condensed on walls [%]</th>
<th>Condensed on deposited aerosols [%]</th>
<th>Deposited aerosols [%]</th>
<th>Total deposited [%]</th>
<th>Out_1 (Vapour) [%]</th>
<th>Out_2 (Aerosols) [%]</th>
<th>Total out [%]</th>
</tr>
</thead>
<tbody>
<tr>
<td>I</td>
<td>68.49%</td>
<td>27.31%</td>
<td>0.00%</td>
<td>95.79%</td>
<td>0.00%</td>
<td>4.10%</td>
<td>4.10%</td>
</tr>
<tr>
<td>Cs</td>
<td>30.21%</td>
<td>39.93%</td>
<td>0.00%</td>
<td>70.14%</td>
<td>6.30%</td>
<td>23.56%</td>
<td>29.86%</td>
</tr>
<tr>
<td>Te</td>
<td>66.44%</td>
<td>29.05%</td>
<td>0.00%</td>
<td>95.49%</td>
<td>0.18%</td>
<td>4.31%</td>
<td>4.49%</td>
</tr>
<tr>
<td>Kr</td>
<td>0.00%</td>
<td>0.00%</td>
<td>0.00%</td>
<td>100.00%</td>
<td>0.00%</td>
<td>100.00%</td>
<td></td>
</tr>
<tr>
<td>Xe</td>
<td>0.00%</td>
<td>0.00%</td>
<td>0.00%</td>
<td>0.00%</td>
<td>100.00%</td>
<td>0.00%</td>
<td>100.00%</td>
</tr>
<tr>
<td>Rb</td>
<td>11.98%</td>
<td>0.38%</td>
<td>0.00%</td>
<td>12.36%</td>
<td>39.27%</td>
<td>48.36%</td>
<td>87.64%</td>
</tr>
<tr>
<td>Sr</td>
<td>59.96%</td>
<td>34.78%</td>
<td>0.00%</td>
<td>94.74%</td>
<td>0.05%</td>
<td>5.21%</td>
<td>5.26%</td>
</tr>
<tr>
<td>Mo</td>
<td>48.53%</td>
<td>48.19%</td>
<td>0.00%</td>
<td>96.72%</td>
<td>0.00%</td>
<td>3.28%</td>
<td>3.28%</td>
</tr>
<tr>
<td>Ru</td>
<td>53.11%</td>
<td>43.55%</td>
<td>0.00%</td>
<td>96.66%</td>
<td>0.00%</td>
<td>3.34%</td>
<td>3.34%</td>
</tr>
<tr>
<td>Ba</td>
<td>57.21%</td>
<td>37.22%</td>
<td>0.00%</td>
<td>94.43%</td>
<td>0.00%</td>
<td>5.57%</td>
<td>5.57%</td>
</tr>
<tr>
<td>Nd</td>
<td>0.00%</td>
<td>0.00%</td>
<td>99.66%</td>
<td>0.00%</td>
<td>0.00%</td>
<td>0.00%</td>
<td>0.00%</td>
</tr>
<tr>
<td>Y</td>
<td>0.00%</td>
<td>0.00%</td>
<td>99.67%</td>
<td>0.00%</td>
<td>0.33%</td>
<td>0.00%</td>
<td>0.33%</td>
</tr>
<tr>
<td>Zr</td>
<td>0.00%</td>
<td>0.00%</td>
<td>99.66%</td>
<td>0.00%</td>
<td>0.34%</td>
<td>0.00%</td>
<td>0.34%</td>
</tr>
<tr>
<td>Tc</td>
<td>0.00%</td>
<td>0.00%</td>
<td>99.68%</td>
<td>0.00%</td>
<td>0.32%</td>
<td>0.00%</td>
<td>0.32%</td>
</tr>
<tr>
<td>Pd</td>
<td>0.00%</td>
<td>0.00%</td>
<td>99.60%</td>
<td>0.00%</td>
<td>0.30%</td>
<td>0.00%</td>
<td>0.30%</td>
</tr>
<tr>
<td>La</td>
<td>0.00%</td>
<td>0.00%</td>
<td>99.67%</td>
<td>0.00%</td>
<td>0.33%</td>
<td>0.00%</td>
<td>0.33%</td>
</tr>
<tr>
<td>Ce</td>
<td>0.00%</td>
<td>0.00%</td>
<td>99.67%</td>
<td>0.00%</td>
<td>0.33%</td>
<td>0.00%</td>
<td>0.33%</td>
</tr>
<tr>
<td>Pr</td>
<td>0.00%</td>
<td>0.00%</td>
<td>99.67%</td>
<td>0.00%</td>
<td>0.33%</td>
<td>0.00%</td>
<td>0.33%</td>
</tr>
<tr>
<td>Pm</td>
<td>0.00%</td>
<td>0.00%</td>
<td>99.63%</td>
<td>0.00%</td>
<td>0.37%</td>
<td>0.00%</td>
<td>0.37%</td>
</tr>
<tr>
<td>Sm</td>
<td>0.00%</td>
<td>0.00%</td>
<td>99.62%</td>
<td>0.00%</td>
<td>0.30%</td>
<td>0.00%</td>
<td>0.30%</td>
</tr>
</tbody>
</table>

There are 11 major radioactive isotopes of Cesium, but only three of them are very important from the aim of our analysis:
- Cs-134 (T1/2=2.1 yr, specific activity 1300 Ci/g, beta decay energy=0.16 MeV, gamma decay energy=1.6 MeV);
- Cs-135 (T1/2=2.3 million yr, specific activity 0.0012 Ci/g, beta decay energy=0.067 MeV);
- Cs-137 (T1/2=30 yr, specific activity 88 Ci/g, beta decay energy=0.19 MeV) (the importance of Cs-137 is additionally connected with its decay product, barium-137m [8]);

Of the 40 important radioactive isotopes of Iodine, we selected for the analysis:
- I-129 (T1/2=16 million yr, specific activity 0.00018 Ci/g, beta decay energy=0.064 MeV, gamma decay energy=0.025 MeV);
- I-131 (T1/2=8.0 d, specific activity 130000 Ci/g, beta decay energy=0.19 MeV, gamma decay energy=0.38 MeV);

For Strontium we take into account two isotopes:
- Sr-89 (T1/2=50.5 d, specific activity 28 200 Ci/g, beta decay energy=1.463 MeV);
- Sr-90 (T1/2=29 y, specific activity 140 Ci/g, beta decay energy=0.20 MeV);

The amount of important radioactive isotopes from the total transferred quantities in the containment system is presented in Table 3.
Table 3 Transferred masses for important radioactive isotopes of Cs, Sr and I to the containment

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Fraction of the isotope in the total mass for each element</th>
<th>Total fraction of important radioactive isotopes</th>
<th>Transferred mass for important radioactive isotopes to the containment [g]</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cs134</td>
<td>1.4%</td>
<td>53.1%</td>
<td>222.95</td>
</tr>
<tr>
<td>Cs135</td>
<td>3.4%</td>
<td>66.4%</td>
<td>12.27</td>
</tr>
<tr>
<td>Cs137</td>
<td>48.40%</td>
<td>77.1%</td>
<td>18.88</td>
</tr>
<tr>
<td>Sr89</td>
<td>11.9%</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Sr90</td>
<td>54.5%</td>
<td></td>
<td></td>
</tr>
<tr>
<td>I129</td>
<td>63.7%</td>
<td></td>
<td></td>
</tr>
<tr>
<td>I131</td>
<td>13.4%</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

We can observe that only a reduced amount of radioactive Sr and I is transferred to the containment, due to the complexity of CANDU circuit (the total length exceeds 110 m for a complete loop, large number of feeders/pipes, small diameters of pipes, etc.).

The second part of investigation consists of the calculation of the distribution of the radioactive isotopes in the rooms of the containment and for different hosts: sump water, wet surfaces, aerosols, painted walls, etc.

The distribution is strongly dependent on the geometry configuration and thermal-hydraulics conditions in the containment. Also the influence of the safety systems is determinant for the retained amount of each isotope. The main important effect is connected with the dousing system functioning. For CANDU type reactor the dousing is powerful pressure suppression type and the water source is the elevated tank around building dome with a capacity of 1560 m$^3$ and a flow rate of 4500 kg/s. The dousing operation is performed by 6 spray head each with 2 valves in series, that turn on when building pressure reaches 14 kPa and turns off if pressure falls below 7 kPa.

From the point of view of the FPs releasing into the atmosphere, the threshold pressure for through-wall cracking of the containment is about 330 kPa (at 2.7 times design pressure negligible leakage through wall cracks) and the structural failure is considered at 530 kPa (at 4.3 times design pressure).

![Fig. 3 Containment pressure in region R1 (CV 740) and R2 (CV780) (left graph) and water mass in the dome tank in case of no recirculation mode](image)

In figure 3 the evolution of the pressure for the first 1000 s after the severe accident starting is presented and also the water mass decreasing in the water tank (spray system is working in direct mode, no recirculation). A maximum pressure of about 167 kPa is reached...
rapidly, but the spray system is able to reduce the pressure at relative normal values. After the consuming of the water from the dome tank the recirculation mode is started.

Therefore, for the postulated conditions of the accident the pressure does not reach the threshold pressure for through-wall cracking of the containment. Thus, the source term formation should be analysed at the level of the distribution for each species in the different regions of the containment and different hosts.

Cesium and Strontium have a similar behavior: the transfer from the injection region (CV760) to the other region of the containment occurred relatively fast (after 200-400 s the majority of Cesium is transferred into CV770); in the first stage Cesium is deposited mainly in water-surface (water condensed on walls) and gas-surface, but at about 440 s as an effect of the washing, the content of water-surface is transferred in the sump water. The distributions were calculated for all isotopes, but for simplicity only the distributions for Cesium-137 and Strontium-89 at t= 2000 s after the starting of the accident is presented in Table 5.

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Host</th>
<th>Water-surface</th>
<th>Water</th>
<th>Gas-surface</th>
<th>Aerosols</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cs-137</td>
<td></td>
<td>1%</td>
<td>36%</td>
<td>43%</td>
<td>20%</td>
</tr>
<tr>
<td>Sr-89</td>
<td></td>
<td>1%</td>
<td>36%</td>
<td>43%</td>
<td>20%</td>
</tr>
</tbody>
</table>

The Iodine species distribution is detailed calculated by the IODE module and it is presented in Table 6. It should be noticed that practically all the mass of iodine is deposited in the sump water as I⁻.

<table>
<thead>
<tr>
<th>Species &amp; Hosts</th>
<th>Mass [Kg]</th>
</tr>
</thead>
<tbody>
<tr>
<td>I₂ Gas</td>
<td>4.0374E-09</td>
</tr>
<tr>
<td>I₂ Dry surface</td>
<td>1.1848E-42</td>
</tr>
<tr>
<td>I₂ Sump water</td>
<td>1.7948E-09</td>
</tr>
<tr>
<td>I₂ Wet surface</td>
<td>1.1848E-42</td>
</tr>
<tr>
<td>I₂ Dry paint</td>
<td>1.1848E-42</td>
</tr>
<tr>
<td>I₂ Wet paint</td>
<td>1.1848E-42</td>
</tr>
<tr>
<td>CH₃I Gas</td>
<td>8.5018E-29</td>
</tr>
<tr>
<td>CH₃I Sump water</td>
<td>3.5237E-32</td>
</tr>
<tr>
<td>HIO Sump water</td>
<td>1.3734E-11</td>
</tr>
<tr>
<td>I⁻ Sump water</td>
<td>2.4477E-02</td>
</tr>
<tr>
<td>IO₃⁻ Sump water</td>
<td>3.6128E-10</td>
</tr>
<tr>
<td>CH₃ Sump water</td>
<td>3.7573E-33</td>
</tr>
<tr>
<td>CH₃OH Sump water</td>
<td>1.3980E-31</td>
</tr>
<tr>
<td>R Sump water</td>
<td>7.2470E-33</td>
</tr>
</tbody>
</table>

For the point of view of potential risk for the environment the aerosol distribution should be discussed. The distribution of the aerosols on the containment regions and for different size classes was calculated. In the condition of the postulated severe accident the main part of the aerosols is located in the rooms CV730 (46.7%) and CV760 (36.4%).
6 CONCLUSIONS

(1) During the evolution of the postulated severe accident initiated by LOCA+LOECC+LOM an important part of the released FPs is retained in the primary system. In CANDU type reactors PHT circuit action as a filter for the majority of species, except Kr, Xe and Rb for that the transfer factors are 100%, 100% and 87.64%, respectively. For other species the main part is deposited in the different volumes of the circuit. For the main important source term species, the transfer fractions are: cesium-29.86%, strontium-5.26%, iodine-4.10%.

(2) The large volume of the CANDU containment (approx 50000 m³) leads to small concentration of species in the different rooms. The main amount of Cesium and Strontium are washed by spray system and transferred in sump water, but an important part remains on the aerosols and in the water-surface. Iodine is practically totally transferred into the sump-water as I⁻.

(3) In the postulated condition of a severe accident for CANDU type plant, the failure of the containment didn’t occur, therefore there is no transfer of FPs to the environment. However, at some hours after the accident’s start, about 134 g of Cesium is distributed on aerosols and gas-surface inside the rooms of the containment and also 7.3 g of Strontium and are not washed.

(4) The distribution of the FPs in the containment is strongly dependent on the thermal-hydraulics conditions, operating of the spray system, and, of course, on the uncertainties (such as the releasing factors, the profile of the releasing, etc.).

ACKNOWLEDGMENTS

The work presented here was prepared in the frame of the SARNET NoE, which is part of the EC Sixth Framework Programme. The ASTEC code and users training were received by INR in the frame of the PHEBEN2 project (EC Fifth Framework Programme), also dedicated to Severe Accidents Management.

We thereby express our thanks to the ASTEC Maintenance Team from IRSN Cadarache, France, whose hints on parameters values were very helpful. Special thanks to the ASTEC Workgroup coordinator in SARNET, Jean-Pierre Van Dorsselaere.

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[8] Human Health Fact Sheet, Argonne National Laboratory, EVS, August 2005
International Efforts Towards Grounding of Nuclear Concepts Safety (Survey of ISTC Programs Results)

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ABSTRACT

As for today - a set of demonstration and basic-type experiments, which fit closely with IAEA/INPRO and GIF program, with EU Frame-Work programs, - had been done or under development now in the frame of ISTC projects and programs with active international collaboration. The ISTC - as a unique international tool – is ready to take part and manage further this activity.

The following information will be included in the review, with special attention on details of corresponding experimental programs:

- Novel reactor concepts:
  - Fast reactors (Sodium-, Lead-and Lead-Bismuth-cooled);
  - Supercritical Pressure Water aspects;
  - HTGR – critical modeling, engineering.
  - Molten salts.

- Nuclear Fuel Cycle options (including Partitioning and Transmutation).

- Reactor data benchmarking and verification, critical experiments.

- Nuclear Power Plant life management.

- Nuclear data measurements.

- Severe accident study.

- Sky-shine experiments.

- RAW storage.


The presentation addresses some consequences of the ISTC projects and programs, related to nuclear science and technologies, as well as methods and approaches employed by the ISTC to foster close international collaboration and joint manage projects towards fruitful results.

1 INTRODUCTION

The ISTC is a unique international organization created in Moscow in 1994 by Russia, USA, EU and Japan. Later Korea and Canada, and several CIS countries as well acceded to ISTC. The basic idea behind establishing the ISTC was to support non-proliferation of the mass destruction weapons technologies by re-directing former Soviet weapons scientists to peaceful research thus preventing the drain of dangerous knowledge and expertise from Russia and other CIS countries.
Presently, the ISTC now has about 40 member countries (27 from EU), representing the CIS, Europe, Asia, and North America. The Partner list includes over 200 organizations and leading industrial companies from all ISTC parties.

The presentation addresses some consequences of the ISTC projects and programs, related to nuclear science and technologies, as well as methods and approaches employed by the ISTC to foster close international collaboration and joint manage projects towards fruitful results.

2. CONCEPT

Challenge of the World Nuclear Community is to prove to Public over the World, that newly proposed nuclear concepts are safe and effective.

The only acceptable method, which is trusted and accepted by Public both now and always, is basic-type and demonstration-type Experiment, in advance of computer or paper-type arguing.

Important that results of these experiments are to be available for international analysis and validation.

Problems are that nuclear experiments are very complex, its require special licensing, long time preparation, appealing to high-skilled personnel, purchasing by nuclear and special materials and tools, as a result - raised budgeting.

In this sense the ISTC clients (first of all – “nuclear-related” institutes in Russia and CIS) have all set, ready, licensed, and equipped unique nuclear installations, high-skilled personnel, good cooperation. Essential, that the ISTC projects:

- Are managing internationally;
- Have plans and results, available for international collaborators;
- Results may be passed to international centers (OECD/NEA and/or others) for further international benchmarking.

As for today - a set of demonstration and basic-type experiments, which fit closely with IAEA/INPRO and GIF program, with EU Frame-Work programs, - had been done or under development now in the frame of ISTC projects and programs with active international collaboration. The ISTC - as a unique international tool – is ready to take part and manage further this activity.

Among five thousand project proposals submitted to ISTC, there are about five hundred related to different aspects of nuclear technologies and Nuclear Fuel Cycle (NFC), first of all – to safety issues.

Survey of the ISTC project results. The following information will be included in the review, with special attention on details of corresponding experimental programs:

- Novel reactor concepts.
- Nuclear Fuel Cycle options (including Partitioning and Transmutation).
- Reactor data benchmarking and verification, critical experiments.
- Nuclear Power Plant life management.
- Nuclear data measurements.
- Severe accident study (Corium modelling, Quench-effect, Chernobyl).
- Sky-shine experiments.
- Accelerator Driven Systems (experimental modeling).
3. NOVEL REACTOR CONCEPTS

3.1. Advanced Fast Reactors, Including Heavy-Metal Technology (Lead, Lead-Bismuth)

Concepts and critical experiments
The project reports include results of conceptual study of advanced fast reactors with sodium and heavy-metal (lead, lead-bismuth) coolants. Predictable neutron-physics characteristics of reactor core concepts, based on both novel oxide, inert, nitride fuels and engineering approaches, were confirmed by relevant experiments, carried out on BFS critical assembly (#0220, #0650, #0815, and others, IPPE, Obninsk). The benchmark-models of the experiments were created for its verification and validation, for estimation of uncertainties connected with both distinction of the model from the real experiment and of material composition, geometry, etc., and for further corrections.

Technical and physical review of BREST-300 project with lead coolant was summarized in the monograph (project #1418, NIKIET).

Basic of Heavy-Metal technology
This set of the projects includes both experimental study and summarising of relevant data (IPPE, Obninsk), for instance:
Hydrodynamics and heat/mass transfer processes in liquid metals (reference manual #1611);
Heavy liquid metals interaction with structural materials, water, air (#1652);
Improvement of corrosion resistance of constructional steels in liquid Pb and Pb-Bi alloys (#2048);
Laser separation of Lead isotopes (#2573);
Control of Oxygen content in Lead coolants (#3020); and
Monograph "Natural Safety Fast Neutron Lead Cooled Reactor for Large Scale Nuclear Power" (#1418).

3.2. Gas-Cooled Fast Breeder Reactor

Design of 1000 MWe power facility with a fast helium-cooled reactor BGR-1000 is based on the concept of the core fuelled by coated micro-fuel and directly cooled by the cross flow of the helium coolant of moderate temperature.
Basic requirements to a reactor design include such as:
Thermal efficiency is not lower than 48-55%;
Exclusion of essential radiation consequences of any severe accident or diversions due to the application of multilayer protective coatings and preserving integrity at temperatures up to 1600°C;
Non-positive void and other reactivity effects (#2973, RRC Kurchatov, Moscow).

3.3. HTGR – Critical Modeling and Engineering

Critical experiments
Critical experiments at the modified ASTRA critical facility, RRC Kurchatov Institute, will be created for validation safety and inherent self-protection of HTGR-M with uranium fuel, including measurement of temperature reactivity effects and its constituents for core, which may be heated from 20°C up to 500-600°C, and control rods worth of as a function of temperature (#0685.2).

Components of gas (helium) technology
Experimental study of technologies for high-temperature components, recuperator design, creation of the stand for turbo-compressor materials testing, development of uranium and U-Pu microsphere-type fuel were done by OKBM, N-Novgorod - #0352, #0769, #1313, #1410.

Experimental study of the cooler model (#2379), Creation of gas seal for shaft (#2395), Experimental study of the seal mock-up (#2399), and Design of powerful diaphragm coupling (#2400).

### 3.4. Heavy-Metal Technology (Lead, Lead-Bismuth)

**Critical experiments**

These experiments, carried out on BFS critical assembly (IPPE, Obninsk), are focused on confirmation of predictable neutron-physical characteristics of reactor core with heavy coolant. The benchmark-model of the experiments were created for its verification and validation, for estimation of uncertainties connected with both distinction of the model from the real experiment and of material composition, geometry, etc., and for further corrections.

### 3.5. Molten Salts

*Molten salts (fluorides with actinides) – measurement of principal parameters*

The experimental stand Thermal Convection Loop had been designed by the inter-institute team leaded by Kurchatov institute and VNIITF (Cheljabinsk-70) and assembled in the hot-chambers of VNIITF. Physical and chemical properties of new solvent systems with TRUs, PuF$_3$ additions and / or Th dissolved in molten 58NaF - 15LiF- 27BeF$_2$ (mole%) systems are studied. Corrosion tests with load / no-load of Ni based materials at natural convection loop with molten salt (more than 1500 hours, up to 100°C, flow 5 cm/s) as well as and after-test examinations were done (#1606).

*Curium in molten chlorides*

Thermodynamics of Curium in molten chlorides and formation of oxygen/ oxygen-free curium compounds are studied (#3261, NIIAR, Dimitrovgrad). These fundamental data can be subsequently used for feasibility assessment of the processes of curium recovery in molten chlorides. The application of the potentiometric titration method, using an oxygen pump made of zirconium-yttrium or zirconium-scandium ceramics, makes it possible to reduce by a factor of tens the amount of the studied element, which is involved in the experiment. This fact is of great importance because of a high curium cost.

*Molten Salts for RAW Treatment*

Extraction and evaporation of alfa-nuclides, concentrating of the liquid RAWs, production of solid mineral-like matrix (#1608, Khlopin RI, St-Petersburg).

### 3.6. NPP with Supercritical Pressure Water

*Aspects of application of supercritical pressure water (# 2689 - IPPE, Obninsk).*

An overview of available information will be carried out on the problems of SCP power plants including physical and thermal hydraulic processes, water regimes under different operating conditions, solubility of structural materials, peculiarities of equipment operation under high temperature and pressure conditions, mechanical properties of structural materials. On this background, physical, thermal and strength evaluations and safety assessments will be made.
The control experimental investigations, based on the thermal simulation methodology, will be carried out on a special Thermal test facility. Using of modelling fluid with low critical parameters (pressure, temperature, and heat capacity or heat evaporation) permits simplify and reduce cost of the experiments.

**Computer modeling for Channel-Type Reactor with Coolant of Supercritical Parameters**

(#3213 – NIKIET, Moscow).

The Project objective is to create a package of computer models and codes for design features and specific characteristics of channel-type reactor cooled by light water of supercritical parameters and moderated by heavy water.

The codes will allow studying stationary and transient 3D neutronics and thermo-hydraulic processes.

## 4. NUCLEAR AND REACTOR DATA MEASUREMENTS AND BENCHMARKING

Set of the ISTC projects had been fulfilled in the frames of ISTC program “MOX and Plutonium disposition”. These benchmarks were focused on modeling of VVER-type reactors (#371/ #371.2, #116, #1836) and modified fast reactors (#220, #650, #1483, #2423, #0731) with MOX and other plutonium fuel (particularly – weapon-grade Plutonium) and fit with the international program “Plutonium disposition in Russia”.

International collaboration (ORNL, INEEL, IRSN and others) is supporting the experimental program with unique in the World family of critical stands COBRA/ BFS in IPPE, Obninsk, for ICSBEP benchmark Handbook of OECD/NEA. The program includes evaluation of critical safety uncertainty (#815), thorium critical experiments (#2432), fast reactor with lead coolant (#2661), justification of MA transmutation (#2884). The project results are available for international verification.

## 5. RAW TRANSMUTATION

### 5.1. Nuclear data (in-Reactor Experiments)

Radiochemical study and activation measurements of the isotopic composition changes of the minor actinide samples (U-234, Np-237, Pu-238, Pu-240, Am-241, Cm-243+Cm-244) after long-time irradiation in BN-350 reactor (#1372).

Based on the review of results of more than thirty related ISTC projects with respect to accuracy revealed in obtaining their main neutronics parameters, the project #2578 aims at definition and selection of vital parameters needed in further clarification through experimenting.

Thus, as a result, one could expect the priority list for future experimental and theoretical studies in the area of waste transmutation, proved by “supply-and-demand” evaluation and expert judgments.

The recommendation for transferring of measured data into proper data files and/or cross-section libraries will be made also.

### 5.2. Partitioning
Projects #2068/ #3405 (Khlopin Radium institute, St-Petersburg): Systematic study of physical chemical properties of new calixarene extractants with functional groups of different types was done.

Target radionuclides were removed:
Pu > 99,999%; Am > 99,999%; Np > 99,0%;
U > 99,98%; Cm > 99,999%;
Nd, Pr, Ce, La and U were combined into rare-earth elements (REE) strip product.
Am, Cm, Pu, Np, Eu, Sm, Y and Gd were combined into trans-plutonium elements (TPE) strip product.
Authors can purify the Am – Cm fraction from heavy or from light REE in one extraction cycle.

6. ACCELERATOR DRIVEN SYSTEMS

6.1. Integral-Type Experiments with Sub-Critical Blankets

**YALINA (14 MeV neutron generator with uranium blanket - #B-070, Minsk)** - a sub-critical uranium-polyethylene assembly driven with a neutron generator of high intensity ($10^{12}$ n/s for 14.1 MeV neutrons and $3 \times 10^{10}$ n/s for 2.5 MeV neutrons) and Cf-252 source was being put into operation in RPCP Institute in Minsk, Belarus. The neutron generator operates in two modes: continuous and pulse ones.

The methods of measurement of sub-critical level, neutron source importance, reactivity effects in sub-critical systems, spectral indexes, transmutation rates were elaborated and tested in the frame of the current project activity.

Measurements of neutron flux distributions over the uranium blanket with different configurations, neutron source importance, reactivity effects, sub-critical levels, dynamic characteristics, spectral indexes, transmutation rates and etc. were performed at the sub-critical assembly (thermal spectrum) driven with the neutron generator. Transformation of blanket into fast booster-type zone with three experimental channels is planned. The YALINA program was accepted by international collaborators as the benchmark.

**SAD – Design of the cyclotron-based ADS with plutonium (MOX) blanket – #2267, JINR, Dubna**

The purpose of the project is to develop and create an experimental installation (SAD) on the basis of existing accelerator and a sub-critical blanket with MOX uranium-plutonium fuel.

The experimental electronuclear installation will include:
- 660MeV proton accelerator;
- Beam transport channel;
- Heavy replaceable target (Pb, W, Pb-Bi);
- Sub-critical blanket with BN-600 type FE with $K_{eff} = 0.95$;
- Protective and supervision systems;
- Control and measuring complex.

The proton beam interacts with the target and produces neutrons, which enter the blanket. The uranium-plutonium blanket and the lead reflector surround the neutron source of the target. A small beryllium insert equipped with experimental channels will be placed behind the lead reflector.
The experimental program and technical project is managing by international SAD-YALINA Steering committee (projects #2267 and #B-070).

1 MW Pb-Bi target system (IPPE, Obninsk)

In the frames of #0559 project the pilot 1 MW molten lead-bismuth target complex TC-1 has been developed, fabricated and tested. Beam-off thermal hydraulics tests were made under isothermal conditions. Unique Russian experience with lead-bismuth alloy as coolant in nuclear submarines was implemented.

Later the TC-1 has been actually delivered at the Harry Reid Center for Environmental Studies, University of Nevada Las Vegas (UNLV) - #2083. Now the set of joint experiments and tests are developing in UNLV.

6.2. ADS - Target (Spallation reactions)

Different measurements of characteristics of spallation reactions cascade initiated by protons (with up to 3 GeV energy) in thin and thick targets were made in frames of more than dozen ISTC projects. Number and energy spectrum of neutrons, heat release, reaction products were measured for different target materials.

6.3. Blanket problems (MA, FPs)

Transmutation efficiency of minor actinides (MA) and fission products (FP) were measured under neutron irradiation. Fast and thermal reactors and accelerators were used. About twenty fulfilled projects correspond to different neutron spectrum, exposure and other conditions. Review of the results of this program and recommendations was made specifically by #2578 project and evaluated by he CEG.

7. STRUCTURE MATERIALS STUDY

The purpose of the project #2048 is to develop and substantiate an effective way to protect constructive steels from corrosion in liquid Pb and Pb-Bi melts at temperatures higher than 500°C via modification of their surface properties with the help of pulsed intense electron beams. An increase in corrosion firmness is reached by formation of oxide covers or by surface doping of steels.

8. SEVERE ACCIDENT STUDY

About fifteen projects related to different aspects of severe nuclear accidents with degradation of core are being managed by international group of collaborators – CEG SAM. This group through its regular meetings (two times a year) and topical workshops coordinates projects with European FW programs and programs of other countries.

8.1. Corium study and modelling

The ultimate goal of the proposed project is the nuclear reactor safety enhancement in case of a severe accident involving the core degradation.
The subject of project series #0833, #0833.2 (METCORE), #1950, #1950.2 (CORPHAD), #K-1265 (INVECOR), #3345 (EVAN), #3592 (METCORE-2) is in-depth theoretical and experimental study of physical-chemical processes taking place at core melt interaction with reactor vessel steel, for instance:
- Corium melt (have different – vessel steel specimen interaction;
- Degree of melts superheating;
- Composition of above-melt atmosphere, inert and steam options are proposed for the Second phase;
- Fission product release to the PWR containment atmosphere.

The results can be used for:
- Elaboration of numeric models, codes and data for corium melt - vessel steel interaction processes;
- Verification of calculation codes modelling free convection processes in the melt pool in terms of physic chemistry;
- Calculation and safety upgrade of operated and designed reactors VVER, PWR and BWR.

8.2. Quench effect

#1648/#1648.2 (QUENCH): Development of data base to describe the VVER and PWR core behavior under severe accident conditions (NIIAR, Dimitrovgrad).

The primary task of the Project is to obtain data on VVER reactors core behavior under severe accident conditions in order to develop physical models and codes applicable to VVER reactors. It is assumed, that this task will be solved as complementary to the QUENCH project in FZK, Karlsruhe. Realization of the same methodical approach will allow comparison of the behavior of VVER and PWR materials and design elements as well as to provide the possibility of application of the same approach to develop the numerical codes and determine safety criteria of these reactors.

Small-scale tests and an integral experiment under quench conditions are carrying out with VVER material in order to build up a database for modeling and verification of codes. The integral fuel bundle experiment will be carried out using non-irradiated materials. The small-scale experiments with irradiated fuel are assumed in order to develop basis for database for irradiated core materials.

The project includes three stages:
- Study of the spent fuel rod segments (RIAR, Dimitrovgrad);
- Integral experiment with model bundle with 31 fuel rod simulators under “quench” conditions (RIAR, FZK Karlsruhe);
- FA Quench Model: Development of models and codes to describe VVER core behavior under “quench” stage conditions (RIAR, IBRAE, FZK).

#3194 and #3690 (PARAMETER): Experimental investigations of behaviour of VVER-1000-type fuel rods and fuel assemblies under simulated conditions of a severe accident including the stage of low rate flooding from top or high rate flooding from top and bottom (LUCH, Podolsk).

The experimental program include study of:
- Thermal-mechanical and corrosion behaviour of VVER fuel rod assemblies in simulated conditions of a severe accident development stages and determining their damage parameters.
Thermal-mechanical behaviour of structural components of VVER fuel rod assemblies (fuel rod cladding, fuel pellets, guiding tube, spacing grids) under flooding from top/top and bottom of the lead assembly superheated up to 2000°C.

VVER fuel rod assemblies in condition of high rate flooding form top and simultaneous flooding from top and bottom.

Determining an oxidation degree of the VVER fuel rod assembly structural components.

Interaction and structural-phase changes in the VVER fuel rod assembly materials (fuel cladding, fuel pellets).

Hydrogen release rates under severe accident conditions including stage of bundle flooding.

#2936 (IBRAE, Moscow): Modelling of reactor core molten materials behaviour at consecutive stages of an accident development: from the early stage, when the core is mostly intact and the first Zr cladding melting occurs, to the late stage, when the core is completely degraded and a molten pool is formed in lower head of the PRV.

The following processes are studied:
Melt formation, onset of melt relocation:
Simultaneous dissolution of ZrO₂ crust and UO₂ fuel (fresh and irradiated) by molten Zircaloy,
Cladding oxide shell failure,
Release of U-Zr-O mixture from the cladding breach.
Candling process: flowing down in the form of drops and rivulets during the first stage of melt relocation;
Formation of massive coolant channel blockage (slug), its oxidation and downward relocation in the course of the second stage of melt relocation process;
Thermal hydraulic behaviour of molten pool in the lower head of the RPV.

8.3. Chernobyl lava data-base

#2916 (CHESS): The "Chernobyl lessons" are very costly for the world society. That is why it is very important to understand them and make their most use. First and foremost, this concerns the results of the giant and practically unrepeatable "experiment" made on nuclear fuel of the Chernobyl’s reactor.

Understanding of the processes in this nuclear fuel during the active accident phase is of much importance for nuclear power safety-related issues. Such understanding is equally important for specific practical tasks to be solved in the future during removal of Fuel-Containing Materials (FCM) from the "Shelter".

The following five main tasks were solved:
Task 1. Acquisition and analysis of the data on:
initial condition of nuclear fuel inside Unit 4;
post-explosion status of Unit 4; and
amounts and composition of released radioactive materials.

Task 2. Determining the key parameters (and their ranges) to be described by the future model based on analysis of verified data on:
composition and amount of radionuclides released from the core during the active accident phase;
nuclear, chemical and mineralogical composition of lave-like FCM;
geometry of intra-reactor compartments, etc.
Task 3. Collection of the data needed to determine concentrations of uranium and zirconium in metal corium generated during the accident for subsequent comparisons with the results of model calculations.

Task 4. Selection of the main FCM parameters and their simulation using the available calculated models. Principal result: development of a model describing most accurately the processes of the active accident phase (corium behaviour).

Task 5. Application of the results achieved in simulation of the selected FCM parameters for identification of both lacking input data and shortcomings of the available calculated models. Based on analysis and estimates of the results, development of proposals for the second phase of the Project and elaboration of approaches to solution of applied problems related to safety of the “Shelter”.

A distinctive feature of the Project consists in maximum possible use of huge and unique experimental data achieved during investigations at the "Shelter" - about 7000 measurements done during twenty years after accident.

9. Sky-shine experiments

# 0517 (NIKIET, Kaz NNC): Series of unique experiments had been done together by Kazakh, Russian and Japanese experts with especial experimental reactors located in desert area (Kazakhstan) for measuring of neutron and gamma, scattered from clouds, over more than 1 km radius. About dozen runs were done under different weather conditions. The results were transferred to OECD/NEA for further international benchmarking.

10. Experimental modelling with steam-generator and other power systems

Goal of the project # 0502 (Bauman TU, Moscow) – to fulfil theoretical and experimental analysis of hydraulically induced vibrations in compact curling tube steam-generators and to verify the ViCAN code via set of experiments and modelling at the family of three experimental stands, especially erected at the laboratories of:

Bauman MGSU (TU), Moscow,
Dollezhal NIKIET, Moscow, and
OKBM, N-Novgorod.

Physical and mathematical models of thermo-hydraulic processes were designed and checked.

11. Novel protection materials for transport and storage casks

Goal of the set of VNIIEF projects (#2691, #2693 and #2694, as well as earlier #0963) is to study specific radiation-protection features of depleted uranium as related to possible applications for nuclear waste geological burial. Particularly, plans included production, experimental study and modelling of the radiation-shielded concretes and ceramics, and depleted uranium dioxide as an aggregate, proposed as constructional and radiation-shielded materials for the casks to be applied for transportation and storage of spent nuclear fuel and radioactive wastes.

The investigations were done under conditions similar Yucca Mountain area, USA.

The three projects above are been managing jointly by the International Steering Committee, which consists of US collaborators, representatives of Russian recipient institutes and ISTC.
12. Nuclear Power Plant (NPP) plant-life management

Set of proposals / projects, having as prime objective the improvement of understanding of features of nuclear reactor structure materials under irradiation, particularly phenomena affecting the residual plant life and the development of methodologies to increase the plant safety and efficiency, to predict reactor life-time and to design plant management strategies, is managing by international CEG (Contact Expert Group) PLIM (Plant Life Management). It includes, for instance:

- Residual Resource Evaluation for Power Equipment (MGTU, Moscow),
- Analytical Methods of Hydrogen and Helium Embrittlement (KIAE, Moscow),
- Neutron Diffraction Study of Nuclear Reactor Materials (Institute of Metal Physics, Ekaterinburg),
- VVER-1000 Reactor Pressure Vessel (KIAE, Moscow),
- “Methodical Development for WWER-1000 Reactor Pressure Vessel Safe Operation Lifetime Evaluation Allowing for Anticorrosive Cladding” (NIKIET, Moscow),
- “Modelling of Brittle and Ductile Fracture and Prediction of Irradiation Damage Effect on Fracture Toughness Properties of Steels for Reactor Pressure Vessels” (PROMETEY, St-Petersburg).

CEG collaborates with RosEnergoAtom (Russian NPP utility), Ukraine USTC and institutes, with EU Frame Work programs NULIFE, PERFECT, COVERS

13. Conclusions

For specific technical information on the projects you may address to project Recipient institutes (addresses are available on the ISTC web-site).

Goals of this presentation are:
- To introduce some of the ISTC programs to international nuclear community,
- To give examples of international cooperation, created in the frames of ISTC,
- To illustrate the statement of importance of international nuclear experiment as a tool for evidence of new nuclear concepts acceptance, and
- To make a call for further joint collaboration.

REFERENCES


2. Web-site: www.istcinfo.ru
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ABSTRACT

During a severe nuclear reactor accident with loss of coolant, the reactor core can melt and form a particle bed (debris) in the lower plenum of the reactor pressure vessel (RPV). Due to the decay heat this particle bed must be cooled to avoid melting of the RPV-wall. Following the experiments on the coolability of particle beds of single-sized and polydispersed spheres with the IKE test facility "DEBRIS" (Fig. 1) [1, 2], this paper presents the experimental results of boiling and dryout tests on a particle bed with irregularly shaped particles. The measured single- and two-phase pressure drop as well as the dryout heat flux of the irregular-particle bed are very similar to the bed with single-sized 3 mm spheres, despite the fact that the current bed is composed of particles with equivalent diameters ranging from 2 to 10 mm. Various transient dryout phenomena have been observed. Under top-flooding conditions, the pressure gradients are all smaller than the hydrostatic pressure gradient of water, indicating an important role of the counter-current interfacial shear stress. For bottom-flooding with a relatively high liquid inflow velocity, the pressure gradient increases consistently with the vapour velocity and the fluid-particle drags become important. With additional forced liquid inflow from the bottom, the dryout heat flux increases dramatically. Several models were used to predict the pressure drop characteristics, however none of them can provide accurate predictions for both top- and bottom-flooding conditions. More sophisticated models should include the effects of non-uniform particle sizes and the multi-dimensional nature of the bed.

1 INTRODUCTION

During a severe accident in a light water reactor, the core can melt and be relocated to the lower plenum of the reactor pressure vessel. There it can form a particulate debris bed due to the possible presence of water. This can lead to the failure of pressure vessel due to the insufficient heat removal of decay heat in the debris bed. Therefore, addressing the issue of coolability behaviour of heat generating particulate debris bed is of prime importance in the...
Due to the large surface area of porous media, the coolability of a porous bed is normally not limited by the heat transfer from the particle to the coolant, viz. the boiling critical heat flux. A special feature of boiling beds with volumetric heat sources is a rise of the vapour velocity with increasing bed height. At a certain vapour velocity the uprising vapour will block the penetrating water from an overlying waterpool. Not enough water can then enter the porous bed and it will dry out. Therefore, the coolability of slender high beds is generally worse compared to wide and shallow beds. Many experiments have been carried out on this counter-current flooding limitation and led to the dryout model first presented by Lipinski [4]. In a modified experiment performed by Hofmann [5], in which water was supplied by a lateral water column to the bottom of the bed, a drastically increased coolability was observed. This increase could not be explained by models without interfacial drag [6-8]. Better agreement was found with models including interfacial drag [9, 10]. But larger number of experimental data is needed in order to validate these models.

In order to gain a deeper insight into the boiling phenomena of a debris bed with volumetric heat sources, a non-nuclear separate effects experimental facility called DEBRIS (Fig. 1) was built up at IKE, which focuses on the general understanding of two-phase flows in porous media. The major tasks of the experimental investigations are the determination of local pressure drops for steady state boiling for checking friction laws, the determination of the DHF (dryout heat flux) under various conditions for comparison with existing results, and the analysis of the quenching process of dry hot debris beds. Following the experiments carried out at IKE on particle beds composed of single-sized spheres [1] and polydispersed spheres [2], this paper presents the boiling and dryout experimental results on a particle bed with irregularly shaped particles (fragmentation particles from PREMIX-experiment, Research Centre, Karlsruhe [3]) mixed with stainless steel balls to improve the inductive heating efficiency.

2 EXPERIMENTAL SET-UP

The experimental set-up consists of a pressure vessel designed for pressures up to 40 bar in which the crucible filled with particles is mounted. The pressure vessel is connected to a storage tank filled with demineralized water and a pumping system, which allows performing boiling experiments with feeding water to the crucible at the bottom (bottom-flooding) or at the top (top-flooding). Figure 1 shows the complete set-up including piping and heat removal system. The debris bed is heated via an oil-cooled 2-winding induction coil by a radio frequency (RF) generator. The RF-generator operates at a frequency of 200 kHz and has a nominal output power of 140 kW.

For boiling and dryout experiments, a crucible made of PTFE (polytetrafluoroethylene) is used. It has a total height of 870 mm and an inner diameter of 125 mm. It is equipped with 60 thermocouples (1 mm, Type N), of which 51 are located in the debris bed on 25 levels. The thermocouples measure the temperature in the voids between the particles, which are filled by liquid, vapour or a mixture of both. For pressure measurements, 8 differential pressure transducers are used (100 mbar, class 0.1). The pressure taps are uniformly distributed in 100 mm intervals along the bed height (pressure transducer dp8 is used for measuring the pressure difference between level PL0 and PL7). The exact position of the thermocouples and pressure taps can be seen in Fig. 2.

The emphasis of this study is getting an insight on the coolability of realistic debris beds. The fragmentation particles from the PREMIX-experiment carried out at the Research Centre Karlsruhe [3] are used, with equivalent diameters ranging from 2 mm to 10 mm. These
particles are mainly composed of Al₂O₃, thus the inductive heating efficiency is very low. In order to pride enough heating power to the debris beds, additional 6 mm and 3 mm steel spheres are mixed to the irregularly shaped particles. The composition of the bed is given in Table 1. The spheres comprise about 44.1% volume of the bed and 68.5% weight. The bed height is 640 mm. The measured porosity of the bed is 0.38.

Boiling and dryout experiments with top- and bottom-flooding were carried out at ambient pressure (1 bar). The top-flooding experiments are started with a water saturated bed and a waterpool of 300 mm height. The vapour is condensed directly above the waterpool, so the water level remains constant. For bottom-flooding experiments, additional water is pumped into the bottom of the bed at a prescribed flow rate.

Table 1: Bed composition.

<table>
<thead>
<tr>
<th>particles</th>
<th>Wt, g</th>
<th>Wt, %</th>
</tr>
</thead>
<tbody>
<tr>
<td>6 mm spheres</td>
<td>11371.6</td>
<td>43.74</td>
</tr>
<tr>
<td>3 mm spheres</td>
<td>6442.5</td>
<td>24.78</td>
</tr>
<tr>
<td>5~10 mm Al₂O₃</td>
<td>5410.9</td>
<td>20.81</td>
</tr>
<tr>
<td>2~5 mm Al₂O₃</td>
<td>2775.5</td>
<td>10.67</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td><strong>26000.5</strong></td>
<td></td>
</tr>
</tbody>
</table>
3 EXPERIMENTAL RESULTS

3.1 Single-Phase Pressure Drop

In order to determine the pressure drop characteristics of the bed, single-phase pressure drop experiments have been carried out. The pressure drops were measured at different liquid superficial velocities as shown in Fig. 3 (blue triangular symbols). For a given flow rate, the 6
data points correspond to 6 measurement positions, i.e. dp1 to dp6 (see Fig. 2). The earlier experimental data on the beds with single-sized spheres (3 mm and 6 mm) and with polydispersed spheres (6/3/2 mm, 50% 6 mm spheres, 30% 3 mm spheres and 20% 2 mm spheres, in weight) are also shown in the figure. The pressure gradients for the irregular-particle bed are smaller than those for the polydispersed-sphere bed but higher than those for the 6mm-sphere bed. The pressure drop behaviour of the irregular-particle bed and the 3mm-sphere bed are quite similar. In fact, if an average particle diameter $D_p = 3$ mm is used in Ergun’s equation [11], i.e.

$$\frac{dp}{dz} = \rho_g u^2 \left( \frac{1 - \varepsilon}{\varepsilon} \right)^2 \frac{\mu}{d_p^2} + 1.75 \frac{(1 - \varepsilon)^2 \rho u^2}{\varepsilon d_p^2}$$

The predicted pressure gradients agree well with the experimental data (the line in Fig. 3). Hence, the average particle diameter is considered as 3 mm in all further calculations.

Fig. 3: Single-phase pressure gradient.

3.2 Two-Phase Pressure Drop

The major aim of the boiling experiments is the verification of the friction laws which are included in the dryout models. This is done by measuring the pressure gradient under steady-state boiling conditions and defined flow conditions. In top-flooding experiments no additional water is pumped into the bed (superficial liquid velocity $J_{l0} = 0$ mm/s). The water, only driven by gravity, has to flow from the waterpool down into the porous bed, so a counter-current flow of water and vapour is established inside the bed. In bottom-flooding experiments additional water is pumped into the bottom of the bed (forced inflow condition). Depending on the water inflow rate and the applied heating power, the liquid-vapour flows can be co-current or counter-current (e.g. very low water inflow rates).

Figure 4 shows the measured pressure gradients against vapour superficial velocity $J_g$ for top-flooding conditions ($J_{l0} = 0$ mm/s) and several forced inflow conditions with $J_{l0}$ ranging from 0.37 mm/s to 7.01 mm/s. For each condition, the experiments were repeated for 2 to 5 times.

Under top-flooding conditions ($J_{l0} = 0$ mm/s), the measured pressure gradients are smaller than the hydrostatic pressure gradient of water. For small vapour velocities in the range of 0 to 0.05 m/s, there is a steep decrease in pressure gradient. For higher vapour
velocities an increase in pressure gradient can be seen. At a velocity of around 0.4 m/s, the measured pressure gradient is close to that of the hydrostatic one. For vapour velocities greater than 0.4 m/s, the pressure gradient decreases again.

For the bottom-flooding with very small water injection rate, the behaviours of pressure gradient are similar to those for the top-flooding, e.g. for $J_{l0} = 0.37$ mm/s. However, in this case the pressure gradient turns positive for $J_g$ between 0.4 and 0.55 m/s. With increasing water injection rate, the negative pressure gradient region (in terms of $J_g$) becomes smaller and smaller. For $J_{l0} = 7.01$ mm/s, the pressure gradient is positive for all vapour velocities and it increases with increasing vapour velocity, indicating that the flow is co-current throughout the bed height.

![Fig. 4: Two-phase pressure gradient.](image)

### 3.3 Dryout Phenomena

Due to the use of the PTFE-crucible the dryout experiments had to be stopped when a maximum temperature of 160 °C in the bed was reached. A rise above saturation temperature indicates that the supply of coolant is insufficient and that a dry spot has formed inside the bed.

Figure 5 shows the heat input, measured temperatures and pressure differences of a dryout experiment with top-flooding. The initial dryout position and the dryout area shortly before the stop of heating are also shown in the figure. The heating power is increased in small steps. When the heat input increases from 9.14 kW to 9.72 kW, the pressure difference $dp5$ decreases to a negative value within a few seconds, and it decreases continuously, indicating that the counter-current flow limit at the position “dp5” (Fig. 2) has been reached. With the heating going on, the counter-current limit is reached at lower bed positions (“dp4”, “dp3” and “dp2”). Dryout does not occur immediately after the drop of $dp5$. The initial dryout
occurs at about 1250 seconds after applying the dryout heat flux (DHF) and at the position of “T13” (lower bed position), when dp2 almost decreases to its lowest value. As the heating continues the dryout area extends. Immediately after the stopping of heating, the temperatures of the dryout area decrease quickly.

Fig. 5: Transient dryout behaviour - top-flooding, $Q_{\text{dryout}} = 9.72$ kW.

The dryout position depends on the applied heating power and also on the stepwise increment of heating power. Figure 6 shows the dryout behaviours when the heating power is suddenly increased from 0.94 kW to 10.32 kW. There is a sudden jump of pressure differences after the increase of the power input. Then the pressure differences decrease quickly to negative values. The dryout occurs in about 150 seconds, which is much shorter than the case shown in Fig. 5. The initial dryout position also shifts to a higher bed position “T29”. However, as the heating continues the dryout area extends mainly downwards. After the stop of heating, the temperatures of the dryout area decrease quickly. The pressure differences do not recover until all temperatures return to the saturation temperature, since the superheated particles in the dryout area provide heat source for evaporation.

For an even higher heating power, as shown in Fig. 7 for $Q = 15.4$ kW, the dryout area shifts further to the upper bed region, and the time required for the dryout to occur decreases to just about 60 seconds. Theoretically, under top-flooding conditions the dryout should occur at the bottom of the bed. However, when the heat input is increased in a big step to a value higher than the minimum dryout heat input, the water supply from the top pool can be throttled completely and dryout can occur at upper bed area, as shown in Figs. 6 & 7. Also note that, for top-flooding, the quick decrease in pressure gradient is always followed by the dryout of the bed.
Figure 6 shows the transient behaviours of dryout under bottom inflow condition with a water flow rate $J_1^0 = 0.7$ mm/s. The inlet temperature of the water (T54) has an impact on the DHF. Due to the heat loss from the pumping line, the inlet temperature is always lower than

![Graph showing transient dryout behaviour - top-flooding, $Q_{dryout} = 10.32$ kW.](image)

**Fig. 6:** Transient dryout behaviour - top-flooding, $Q_{dryout} = 10.32$ kW.

Figure 7 shows the transient behaviours of dryout under bottom inflow condition with a water flow rate $J_1^0 = 0.7$ mm/s. The inlet temperature of the water (T54) has an impact on the DHF. Due to the heat loss from the pumping line, the inlet temperature is always lower than

![Graph showing transient dryout behaviour - top-flooding, $Q_{dryout} = 15.4$ kW.](image)

**Fig. 7:** Transient dryout behaviour - top-flooding, $Q_{dryout} = 15.4$ kW.

Figure 8 shows the transient behaviours of dryout under bottom inflow condition with a water flow rate $J_1^0 = 0.7$ mm/s. The inlet temperature of the water (T54) has an impact on the DHF. Due to the heat loss from the pumping line, the inlet temperature is always lower than
the saturation temperature, and it also depends on the flow rate. In this case, the heat input is kept constant at 19.4 kW. For a water inlet temperature of 84°C, no dryout occurs. However, dryout occurs when the initial inlet temperature is increased to 89°C. Similar to the top-flooding cases, there is a quick drop in pressure differences before the appearance of dryout, however only at the upper half of the bed (dp4, dp5 and dp6). At such low water injection rate, a large portion of the injected water can be evaporated at the lower part of the bed. Thus for the rest of the bed area, the flow patterns and dryout behaviours are similar to those for the top-flooding. However, compared to the top-flooding the dryout heat input increases by 9.7 kW (see Tab. 2) and it is also seen that the dryout position for the bottom-flooding shifts to the upper part of the bed (compare Figs. 5, 6&7).

Figure 9 summarises the time needed for a physical dryout (lack of liquid resulting in a rise of bed temperature) to occur after applying the dryout heating power. It is clear that this time period depends strongly on the heat input which determines the evaporation rate. For the top-flooding, as the heat input approaches the minimum dryout heat input (about 9.7 kW), the time required for the appearance of the dryout increases dramatically. Similar tendency is seen for the bottom-flooding, which however, is also influenced by the inlet temperature.

The (minimum) dryout heat inputs under various conditions are summarised in Table 2. For the top-flooding, the dryout heat input for the bed with irregular particles is similar to that for the bed with 3 mm spheres, but much lower than that for the bed with 6 mm spheres. With additional forced liquid inflow from the bottom, the dryout heat input increases strongly even with a very small flow rate (with J_{l0} up to 0.70 mm/s).

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**Fig. 8: Transient dryout behaviour - bottom-flooding.** $Q_{\text{dryout}} = 19.4$ kW, $J_{l0} = 0.7$ mm/s.
Table 2: Dryout heat inputs & DHF under top- and bottom-flooding conditions.

<table>
<thead>
<tr>
<th>Bed Conditions</th>
<th>Q\text{\textsubscript{dryout}} (kW)</th>
<th>DHF (kW/m\textsuperscript{2})</th>
</tr>
</thead>
<tbody>
<tr>
<td>3 mm Spheres Top-Flooding</td>
<td>10</td>
<td>813</td>
</tr>
<tr>
<td>6 mm Spheres Top-Flooding</td>
<td>16.3</td>
<td>1330</td>
</tr>
<tr>
<td>6/3/2 mm Spheres Top-Flooding</td>
<td>8.3</td>
<td>680</td>
</tr>
<tr>
<td>Natural Circulation Loop*</td>
<td>18.8</td>
<td>1530</td>
</tr>
<tr>
<td>Irregular Particles Top Flooding</td>
<td>9.7</td>
<td>792</td>
</tr>
<tr>
<td>Bottom-Flooding, J\textsubscript{in}=0.37 mm/s, T\textsubscript{in}=64 °C</td>
<td>14.3</td>
<td>1165</td>
</tr>
<tr>
<td>Bottom-Flooding, J\textsubscript{in}=0.70 mm/s, T\textsubscript{in}=86 °C</td>
<td>19.4</td>
<td>1581</td>
</tr>
</tbody>
</table>

\* Under top-flooding conditions, the valves on the right hand side in Fig. 1 are opened, thus part of the coolant on the top water pool can circulate through the interconnection to the bottom of the bed. The flow rate will establish itself according to the pressure field.

3.4 Comparison with Experiments and Models

The pressure gradients of the bed with irregular particles are compared with the experimental data for beds with single-sized and polydispersed spheres for top- and bottom-flooding, as shown in Figs. 10a & b. It can be seen that, although the irregularly shaped particles consist of particles ranging from 2 to 10 mm size, their behaviour is very close to 3 mm diameter spheres, for both top- and bottom-flooding. The bed with 6 mm spheres has much lower pressure gradients than other beds, while the bed with polydispersed spheres has the highest pressure gradients (Fig. 10b).

Various models are found in literature describing the two-phase flow in porous media as well as dryout heat flux. Most models are based on the extension of Ergun’s equation (Eq. 1). It can be expressed in following form

$$-\nabla p = \frac{\mu}{K} j + \frac{\rho}{\eta} \left| \frac{\partial}{\partial t} j + \rho g \right.$$  

$$\text{(1)}$$
where, $K$ and $\eta$ are the permeability and the passability of the porous media and are given as

$$K = \frac{\varepsilon^3 d_p^2}{150(1-\varepsilon)} \quad \text{and} \quad \eta = \frac{\varepsilon^3 d_p}{1.75(1-\varepsilon)}$$

(Fig. 10: Comparison of experimental pressure gradients: (a) top-flooding, (b) bottom-flooding.

Equation (2) is valid for single-phase flow. For two-phase flows the Ergun-equation is set up for each phase separately. The mutual influence of the two phases (e.g. reduction of the flow path) is taken into account by the relative permeabilities $K_{rel,l}$, $K_{rel,g}$ and the relative passabilities $\eta_{rel,l}$, $\eta_{rel,g}$. The momentum equations for vapor and liquid phase are given as
The permeability and passability are expressed in terms of the void fraction and have been given by various models. A summary is given in Table 3.

**Table 3: Relative permeabilities and relative passabilities.**

<table>
<thead>
<tr>
<th>Model</th>
<th>$K_{rel,v}$</th>
<th>$\eta_{rel,v}$</th>
<th>$K_{rel,l}$</th>
<th>$\eta_{rel,l}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reed [6]</td>
<td>( \alpha^3 )</td>
<td>( \alpha^5 )</td>
<td>(1-( \alpha ))^3</td>
<td>(1-( \alpha ))^5</td>
</tr>
<tr>
<td>Lipinski [8]</td>
<td>( \alpha^3 )</td>
<td>( \alpha^3 )</td>
<td>(1-( \alpha ))^3</td>
<td>(1-( \alpha ))^3</td>
</tr>
<tr>
<td>Hu &amp; Theofanous [7]</td>
<td>( \alpha^3 )</td>
<td>( \alpha^6 )</td>
<td>(1-( \alpha ))^3</td>
<td>(1-( \alpha ))^6</td>
</tr>
</tbody>
</table>

Most models like Reed, Lipinski and Hu & Theophanous exclude the interfacial drag term between liquid and vapor. However, it is necessary to include the vapor-liquid interfacial drag terms at high relative velocities between vapor and liquid. Schulenberg & Müller [10] used the correlation obtained from air-water experiments to formulate the interfacial drag term. Tung & Dhir [9] also used the interfacial drag term besides formulating the vapor permeability and passability depending upon the flow regime.

To compare the experimental data of the bed with irregular particles with the simulation models, an effective particle diameter must be determined which is obviously very difficult. From the results already shown above, the pressure drop and DHF of the irregular particle bed are very similar to those for the single-sized 3mm-sphere bed. Therefore, in the following comparisons, an effective particle diameter of 3 mm is chosen, and the measured bed porosity of 0.38 is used.

A comparison is made between the above mentioned models and the experimental results for dryout heat flux (Table 4) and pressure drop characteristics (Figs. 11a-d). For the top-flooding, the dryout heat flux is well predicted by the models of Reed and Schulenberg & Müller (Tab. 4). However the Reed model, like all the models without explicit consideration of interfacial drag, fails to predict the two-phase-flow pressure drop below the critical heat flux (Fig. 11a). The better prediction is given by the Schulenberg & Müller model, which is able to predict the pressure drop characteristic with typical lying-S shaped curve. The reason can be that, in case of top-flooding, the vapor velocities are low and thus the interfacial drag plays an important role, which is taken into account in the Schulenberg & Müller model. The Lipinski model largely overpredicts the DHF, while the Tung & Dhir model and Hu & Theophanous model underpredict the DHF value.

For bottom-flooding, the Lipinski model gives better predictions of the DHF than the other models for \( J_0 = 0.70 \text{ mm/s} \) (Tab. 4). Although the variations in the predicted dryout heat fluxes among all the models are small, there is a significant difference in pressure drop characteristics (Figs. 11b-d). At a low inflow velocity of 0.70 mm/s (Fig. 11b), no model seems to be able to predict the pressure gradients accurately. In this case, the Schulenberg & Müller model gives the best tendency of the experimental data. For the inflow velocity of 2.83 mm/s (Fig. 11c), the Reed model gives the best prediction. The models of Hu & Theofanous and Lipinski give reasonably good predictions for the low vapour velocities, for a vapour velocity higher than 0.4 m/s the deviations become bigger. The predictions from the models of Schulenberg & Müller and Tung & Dhir are much smaller than the experimental values. When the flow velocity increases to 7.01 mm/s (Fig. 11d), the particle-liquid drag becomes
more dominant, the pressure gradients increase consistently with vapour velocity. All the models predict this tendency correctly, however the deviations are generally very high except for the model of Tung & Dhir.

Table 4: Comparison of top-flooding dryout heat flux results - kW/m².

<table>
<thead>
<tr>
<th></th>
<th>Reed</th>
<th>Lipinski</th>
<th>Hu &amp; Theophanous</th>
<th>Schulenberg &amp; Müller</th>
<th>Tung &amp; Dhir</th>
<th>Exp.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Top-flooding</td>
<td>800</td>
<td>1075</td>
<td>640</td>
<td>729</td>
<td>518</td>
<td>792</td>
</tr>
<tr>
<td>Bottom-flooding</td>
<td>1504</td>
<td>1523</td>
<td>1497</td>
<td>1497</td>
<td>1491</td>
<td>1581</td>
</tr>
</tbody>
</table>

Fig. 11: Comparison of experimental pressure gradients with models: (a) top-flooding; (b) bottom-flooding \( J_{l0} = 0.70 \) mm/s; (c) bottom-flooding \( J_{l0} = 2.83 \) mm/s; (d) bottom-flooding \( J_{l0} = 7.01 \) mm/s.

4 SUMMARY AND CONCLUSIONS

Boiling and dryout experiments with top- and bottom-flooding were carried out at ambient pressure using volumetric heating. To study the coolability of realistic debris beds, the irregular particles with equivalent diameters ranging from 2 mm to 10 mm were used.

Various transient dryout phenomena have been observed. For the top-flooding, when the heat flux approaches the minimum dryout heat flux, the dryout tends to initiate at the bottom of the bed and the time needed for the dryout to occur is very long. With the increase of the applied heat flux, the dryout area shifts to a higher bed position and the dryout time
becomes shorter. With additional forced liquid inflow from the bottom, the dryout heat flux increases dramatically even with very small flow rates, and the dryout initiates normally on the upper half of the bed. For the top-flooding, the quick decrease in pressure gradient is always observed before the dryout occurs inside the bed.

The measured single and two-phase pressure drop characteristics as well as the dryout heat flux of the irregular-particle bed are very similar to the bed with 3 mm spheres. For the top-flooding and bottom-flooding with very low inflow rates, the pressure gradient shows a typical lying-S shaped curve, indicating a counter-current flow characteristics, where the vapour-liquid interfacial drag is important. For the bottom-flooding with relatively high flow rates, the particle-liquid drag becomes more dominant, the pressure gradient increases consistently with vapour velocity. Several models were applied to predict the pressure drop characteristics, however none of them can provide accurate predictions for both top- and bottom-flooding conditions. The model of Schulenberg & Müller is more suitable for the top-flooding and low flow rate bottom-flooding, whereas the models without interfacial drag term can better predict the bottom-flooding with relatively high flow rates. Therefore, more sophisticated models should include the effects of non-uniform particle sizes and the multi-dimensional nature of the bed.

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REFERENCES

A Study of the Influence of Plant Condition on the Heat Removal Feasibility After a Failure of the RHRS

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ABSTRACT

Current low power and shutdown Probabilistic Safety Assessment (LPSA) studies show that the loss of the Residual Heat Removal System (RHRS) transient is one of the most risk-significant events under low power conditions. This accident type is supposed to occur with the plant under several operational modes and configurations. Among them mid-loop operation represents one of the main contributors. Under these conditions, the primary inventory is reduced to perform maintenance tasks when a reload is undertaken. This paper focuses on analyzing the loss of the RHRS of a PWR plant operating at mid-loop conditions and using the risk information provided by the plant LPSA, which addresses the Limiting Conditions for Operation imposed by the current Technical Specification 3/4.4.1.5 of a PWR nuclear facility. Such Technical Specification establishes the requirement of having two trains of the RHRS available, and one of them operating to evacuate the residual heat generated by the core. However, there is a possibility of having alternative or complementary sources for heat removal others than the RHRS available, e.g. Steam Generators, what influences the LPSA conclusions. In addition, current LPSA normally assumes the reactor coolant system is open; however, this assumption is revised herein to address the possibility of having it closed since during the process of shutting down or starting up the plant one may face any of both situations. To explore the ability of the different heat sinks depending on the plant situation a best estimate code was used. Various best estimate thermal-hydraulic codes have been used to analyze the loss of the RHRS during low power and shutdown conditions. In particular, RELAP code can give good results as derived after a number of benchmark exercises using results from experiments at research facilities.

1 INTRODUCTION (12 PT, BOLD)

Results from current Probabilistic Safety Assessment (PSA) studies of Pressurized Water Reactor (PWR) Nuclear Power Plants (NPP) show the importance of some risky scenarios with the plant at low power and shutdown conditions as compared to the accident scenarios with the plant operating at full power. In particular, current low power and
shutdown PSA (LPSA) studies show that the loss of the Residual Heat Removal System (RHRS) transient is one of the most risk-significant events under low power conditions [1]. This accident type is supposed to occur for various plant operating states (POS), of which plant operation at mid-loop and with reduced water inventory represent two of the main contributors.

This paper focuses on analyzing the loss of the RHRS of a PWR plant in Cold Shutdown, in particular operating:

1) at mid-loop (POS 9 to 11) and
2) with reduced water inventory (POS 5 and 6) respectively,

using the risk information provided by the plant LPSA, which must address the Limiting Conditions for Operation (LCO) imposed by the current Technical Specification (TS) of a PWR nuclear facility in Cold Shutdown. Current TS for the plant in Cold Shutdown distinguishes two situations:

a) Main Reactor Cooling System (RCS) fully filled with water (TS 3/4.4.1.4) and
b) RCS partially filled (TS 3/4.4.1.5).

Situation b) is the only one of concern in this paper, since this is the case of the plant at midloop or with reduced water inventory, which establishes the requirement of having two trains of the RHRS (2 of 2) available, and one of them operating. However, it is also relevant for this paper to take into account what situation a) establishes about requiring only one train of the RHRS (1 of 2) available or, alternatively, requiring two Steam Generators (SG) available and filled with water.

In addition, while the plant is at midloop or with reduced water inventory in Cold Shutdown, two different plant configurations can be distinguished, which depend on the particular POS:

1) RCS open and
2) RCS closed.

Thus, in the process of shutdown, the NPP enters Cold Shutdown operational mode with the RCS fully filled with water and closed (POS 5), while it ends this operational mode with the RCS with reduced water inventory and open (POS 6). RCS with reduced inventory and closed is just an intermediate state (end of POS 5). On the contrary, in the process of startup, the NPP enters the Cold Shutdown operational mode with the RCS open and at mid-loop (POS 9) while it ends with the RCS full and closed (POS 11). RCS at midloop and closed is just another intermediate state (end of POS 9). However, TS does not make difference between RCS open or closed, or between midloop or reduced inventory, which is of concern only to situation b) above.

This paper explores the possibility of having alternative or complementary sources for heat removal others than the RHRS trains available, which is the one required in the TS under option b). Being aware of the alternative sources proposed under option a), especial attention is paid to the role of Steam Generators as an effective heat sink. Such alternatives will influence LPSA implementation results. Moreover, current LPSA normally assumes the RCS is open while the plant is at midloop or with reduced water inventory; however, this assumption is revised herein to address the possibility of having the primary system closed since during the shutdown or startup processes one may face any of both situations as indicated above.

This study has been performed within the framework of the risk informed decision making for analyzing changes to current TS using LPSA, which applies for the NPP operating under other modes than full power. The results of this study could be used to justify the need of a change to the above LCO within current TS based on the risk information provided by the LPSA, which in addition would have to be updated following the deterministic analysis performed with the use of the RELAP thermal-hydraulic code in this work.
Different best estimate thermal-hydraulic codes have been used to analyze the loss of the RHRS during low power and shutdown conditions [2, 3]. In particular, RELAP-5 code gives good results in the simulation of accidental sequences under these conditions, as derived after a number of benchmark exercises using results from experiments at research facilities (e.g. ROSA-IV, BETHSY, PKL) [4].

2 TRANSIENT PHENOMENOLOGY

In a Nuclear Power Plant shutdown process, the loss of the RHR system causes an increase of the core temperature and the subsequent steam formation in the core. The later thermo-hydraulic evolution will depend of the POS where the plant is in that moment, since during shutdown process, the plant goes through a variety of operational states with significant differences in time spent and characteristics of operation, as open or closed primary, amount of mass inventory, availability of support systems, etc.

In general, in the case of closed primary, the residual heat generated in the core, causes an increase in the coolant temperature and a subsequent steam formation that causes a rise of reactor pressure, while when the primary is open, the primary system pressure is maintained stabilized around the atmospheric pressure (except for determinate local effects). In both cases it is necessary to remove the residual heat to guarantee the plant safety.

With the primary closed and with the rise in pressure it is possible to achieve a heat transfer through the steam generators from primary to secondary sides. The heat transferred to the secondary sides of available steam generators or in wet lay up, modifies the water until saturation conditions causing an increase in the secondary pressure of such steam generators. If the steam generator is considered to be operable (that is, calibrated to open at one fixed pressure) when the coolant pressure reaches the calibration pressure, the relieve valves and feedwater to control maintain the pressure and level conditions in the steam generator secondary side, and the residual heat is removed from the reactor core. Otherwise, the steam generator can only be on wet layout, what means that no control on the secondary side pressure is maintained and there is no possibility of feedwater injection. So if the secondary side valves remain closed during the transient, there is an increase in the secondary side pressure and a decrease in the secondary side level due to the heat transferred from the primary side rises the secondary side water temperature and when saturation conditions are reached, part of this water is evaporated. In other case, if the secondary side valves remain open, the pressure is maintained and the level in the secondary side decreases, and the final heat sink is the opening of the secondary side valves.

On the other hand, if the primary system is open, the removal of the residual heat generated in the reactor core is carried out through primary opening with a loss in the primary system inventory. In that case, it is not possible to use the steam generators as heat sink, as the primary circuit pressure is not high enough to start a coolant circulation towards the inlet side of the steam generator U-tubes and consequently, the cannot be a heat transfer to the secondary side.

In all cases, the low power and shutdown Probabilistic Safety Assessment considers that it is possible to restart the RHR system as long as the plant conditions at the time that this system is required to operate, especially of RHR pump cavitation, make its operation possible.
3 LPSA DRESSING THE LOSS OF THE RHR INITIATING EVENTS

The LPSA provides risk information about the accidental sequences departing from the loss of the RHRS. In particular, two types of event trees are developed to address the loss of the RHRS initiating event with the plant in Cold Shutdown, which eventually should represent RCS fully and partially filled with water respectively in order to address what NPP Technical Specifications establish, see Figure 1.

![Event Tree Diagram](image)

a) RCS closed

b) RCS open

Figure 1: Accidental sequence for the loss of RHR

In addition, each type of event tree is associated with a particular situation of the primary system representing RCS closed or open respectively. Thus, the former is used to address the case of the loss of the RHRS with the RCS fully filled with water, while the latter is used to address the loss of the RHRS with the plant at midloop or with reduced water inventory.

Consequently, in the current LPSA it is normally assumed implicitly that the RCS is open while the plant is at midloop or with reduced inventory, which is also a consequence of what is assumed in LCO in current TS as discussed in section 1. However, this is not always the case as discussed in section 1 also. Then, both options in Figure 1, RCS open or closed, could be used to address the loss of the RHRS under midloop and reduced water inventory.
conditions and consequently current LCO could be adapted to make such a distinction if applicable and relevant from a risk viewpoint.

Thus, although the actuation of the steam generators is not considered in the technical specification with the RCS partially filled with water, SG could act as alternative way to remove the residual heat generated by the reactor core based on the transient phenomenology exposed in section 2 with the plant at mid-loop conditions or with reduced inventory when the RCS is closed.

This paper is focused on the study of the plant behavior after the loss of the RHRS with the plant in Cold Shutdown and with the RCS not fully filled with water, taking into account the different plant configurations, RCS closed or open, depending on the plant operational state (POS 5-6 for reduced water inventory and POS 9-11 for midloop). The main objective of this study is to analyze different alternatives to remove the residual heat not established in the technical specification that could be successful to maintain the plant in safe conditions.

In this way, the plant evolution with the reactor coolant circuit open or closed is analyzed combined with the number of steam generators in wet layout that could be necessary to transfer the amount of residual heat generated in the core.

As well as the number of steam generators available, it is also important to take into account their operational state. That is, one steam generator with the secondary side pressure and level control activated has more capability to extract the heat than one just in wet layup.

4 CASE OF APPLICATION

Due to fact that Technical Specification 3/4.4.1.5 applies to different plant operational states, with a diversity of pressures, temperatures, levels, etc, its implementation allows a high variety of results in the plant evolution.

This study focuses on the evolution of the plant performance when the RHRS system is lost under different plant operational states. This variety of final states can conduce to the possibility of having alternative or complementary sources for heat removal others than the RHRS available, which are not contemplated in current TS and LPSA.

A loss of the RHR system of a three loop pressurized water reactor has been simulated using RELAP-5 thermal-hydraulic code, and the time to core boiling point, to core uncover, the peak cladding temperature (PCT) has been monitored. The capabilities of the alternative ways proposed to evacuate the residual heat, which are not established in the Technical Specification, are also analyzed based on the evolutions of the variables above exposed.

In particular, Table 1 shows the plant configurations of all the transients simulated. A total of fourteen different plant configurations were studied in which the loss of the RHR system is assumed, with the reactor coolant circuit open/closed, the level of water inside this system, the number of steam generators in wet layout and their mode of operation (available or not).
Table 1. Cases of study

<table>
<thead>
<tr>
<th>Primary System Configuration</th>
<th>Primary System Level</th>
<th>Number of SG</th>
<th>SG Operational State</th>
</tr>
</thead>
<tbody>
<tr>
<td>Closed</td>
<td>reduced</td>
<td>0</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td></td>
<td>1</td>
<td>Wet lay up</td>
</tr>
<tr>
<td></td>
<td></td>
<td>1</td>
<td>2 bars</td>
</tr>
<tr>
<td></td>
<td></td>
<td>2</td>
<td>Wet lay up</td>
</tr>
<tr>
<td></td>
<td>midloop</td>
<td>0</td>
<td>-</td>
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4.1 Initial Conditions

All the plant configurations considered belong to Cold Shutdown Mode, and take part in at least two plant operational states of this Mode.

The experiments are based on a scenario where the nuclear power plant is in Cold Shutdown with primary pressure of 1 bar, and temperature at the core outlet of 60°C. The reactor coolant circuit is filled with nitrogen above the level of water inventory. It is postulated a complete failure of RHRS.

4.2 Results

In the nuclear power plant shutdown process, once the plant is in Cold Shutdown, the reduction of the reactor coolant system level is produced once the system has been depressurized and opened. In these conditions (reduced water inventory and primary open), if the RHR fails the only way to evacuate the residual heat is by means of the loss of steam through the primary circuit opening without a significant influence of steam generators actuation, and without dependence on the initial primary level.

Thus, fig. 2 shows the evolution of the core level along the transient assuming the initial reactor coolant circuit level is set to reduced inventory and primary open. In this case, the figure shows the results obtained supposing there is no steam generators in wet layout and with one steam generator in wet layout. For both cases the time to core uncover is around 9000 sec.
If the same analysis is performed supposing the initial primary level is set at mid-loop, see fig. 3, the results are quite similar to the previous case. Now, the time to core uncover is earlier than that for the previous one (see fig. 2 and fig.3), i.e. around 7000 sec. when the level reaches the lower limit. What concerns the effect of the number of steam generators in wet layout, figure 3 shows that it does not affect the evolution of the transient in a similar way as in the previous case.

So, as shown in fig. 2 and fig. 3, there is no influence of the steam generators when the primary circuit is open, as the conditions of pressure are not adequate to initiate a coolant circulation towards the steam generators U-tubes and there is no heat transfer from the primary to the secondary side of the steam generators.

On the other hand, also during Cold Shutdown process, while the primary level of the nuclear power plant is reduced, the primary circuit can be found depressurized but closed. In such situation, independently of level, the effect of the number of steam generators in wet layout, and their operation mode significantly influences the later transient evolution, as shown in the following figures.

Figure 4 shows the core level evolution when the RHR system is lost and the primary is closed and with reduced inventory. In this case the effect of the number of steam generators is clearly evidenced. Thus, as many steam generators in wet layout the time to core uncover increases.
However, the operation mode implies great differences in terms of time available until core uncover. In figures 5 and 6 it can be appreciated that for both simulations in which the secondary side pressure and level controls are activated, 1 SG (fig. 5) and 2 SG (fig. 6), the core uncover is significantly delayed, especially when two steam generators are considered, see figure 6, where the delay is around 20000 sec.

Figure 5: Core Level – Reduced Inventory (primary closed and 1 SG in wet layout versus operating)
In Figure 7 all the results for the core level evolution obtained considering reduced inventory and system closed as initial conditions are shown. The maximum time to core uncover is obtained using as heat sink two steam generators with secondary side pressure and level controls.

For lower levels in the primary, as mid-loop, and with the primary closed, the presence of steam generators seems to be very important too, as shown in Figure 8. In this case, the increase in the number of steam generators in wet layout also delays the time to core uncover. In fact, this case the delay to time uncover is larger than the delay obtained with the same plant configuration for reduced inventory, see Figure 4.
In a similar way as it happens with reduced inventory, the steam generators operation mode influences the transient evolutions. Thus, as can be observed in figure 9, considering only one steam generator, if it is available the time to core uncover is significantly delayed, around 10000 sec.

The same behavior is observed considering two steam generators as heat sink. Thus, figure 10 shows the evolution of the core level considering two steam generators and mid-loop conditions. Once again, the steam generators act as heat sink and the time to core uncover is considerably delayed. Also in this case the control on the secondary side pressure and level delays core uncover. And the most significant time available is obtained using two steam generators.
Finally when comparing the maximum time available depending on the primary circuit initial mass inventory, see figures 7 and 10, it can be observed that the time available is similar

In particular in the simulations performed, for reduced inventory the time is 40460 sec. and for mid-loop is 43400 sec. So in this case the time available is larger if the primary circuit is in mid-loop conditions when the RHR system is lost.

5 CONCLUSIONS

When the plant is at low power and shutdown conditions some accidental sequences are found to be important from the risk point of view. Important sequences are those whose initiating event is the loss of RHR. In these conditions the Technical Specification imposes the limiting conditions for operation. The Technical Specification analyzes in this work establishes the actions to be taken to remove the residual heat with the plant in Cold Shutdown depending on the primary circuit is full of water or not. But, in Cold Shutdown conditions, the same Technical Specification covers a wide range of plant operational states and, in some POS, measures established in the technical specification to evacuate the residual heat could be extended; so in this paper alternative paths to remove the residual heat with different plant configurations have been studied. In particular, the possibility of removing the residual heat through the steam generators is analyzed.

Thus, considering the primary circuit open when the RHR is lost, the only way of removing the residual heat is through the restart of one of the RHR trains, and is not possible to use the steam generators as alternative heat sink, as the primary pressure conditions does not allow a circulation of coolant through the U-tubes.

However, if the primary circuit is closed, steam generators can be a good system to evacuate the residual heat, as the results exposed in section 5 reveals that the time to core uncover is extended when one or more steam generators, in wet layout or available, are considered, independently of primary circuit level to be mid-loop or reduced inventory.

As well as the number of steam generators, their operational state (wet layout or operating) also influences the plant response. So, if the steam generators are available, what means that secondary side pressure and level are controlled, the time to core uncover is extended as compared with the time calculated considering the steam generators are in wet layout.

So, in general, the criterion established in the technical Specification analyzed of using steam generators available as heat sink if the primary circuit is full, could be extended to
others levels if the primary circuit is closed. Otherwise, if the primary circuit is opened, it is necessary to restart the residual heat removal system.

Consequently, the current LPSA could be also adapted to take into account the results found. One way could be to distinguish two different event trees to address the loss of the RHRS with the plant at mid-loop or with reduced inventory depending on the primary system being closed or open respectively, i.e. using both types of event trees in Figure 1 instead of just the second one considered in current LPSA for mid-loop or reduced inventory conditions.

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Influence of NPP Krsko Core Nuclear Characteristics on RCCA Ejection Accident

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ABSTRACT

The mechanical failure of control rod mechanism pressure housing may result in the ejection of a Rod Cluster Control Assembly (RCCA) due to the pressure difference. The consequences include a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. Worth of the ejected control rod and power distribution in reactor core obtained with remaining control rods will determine quantity of damaged fuel rods. The reactivity transient is terminated by the Doppler reactivity effect due to increased fuel temperature, and by subsequent reactor trip activated by high neutron flux signals, before conditions are reached that can result in significant fuel melt.

Single control rod ejection calculation has been performed for NPP Krsko HFP and HZP core conditions using best-estimate calculation tools and realistic assumptions. Control rod ejection accident was analyzed with 3D neutron kinetics computer code PARCS v2.5 in order to assess behaviour of the reactor core during and after the accident. Special attention was paid to thermal-hydraulic model of the core which was realised using modified COBRA code. More positive reactivity inserted into the core means higher fuel enthalpy and fuel temperature in local rods, so additional sensitivity runs were performed to explore influence of uncertainty in RCCAs reactivity worth and in delayed neutron fractions on limiting core conditions.

After performing core-wide calculation, for most adverse conditions, analysis was made for hot fuel rods with FRAPTRAN computer code, to determine mechanical and thermal performance and possible damage of the rods.

1 INTRODUCTION

The mechanical failure of control rod mechanism pressure housing may result in the ejection of a Rod Cluster Control Assembly (RCCA) due to the pressure difference. The consequences include a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. Worth of the ejected control rod and power distribution in reactor core obtained with remaining control rods will determine quantity of damaged fuel rods. The reactivity transient is terminated by the Doppler reactivity effect.
effect due to increased fuel temperature, and by subsequent reactor trip activated by high neutron flux signals, before conditions are reached that can result in significant fuel melt.

The RCCA ejection accident is classified as ANS IV condition event (limiting fault). Any damage to reactor core or reactor coolant system pressure boundary must not prevent long-term core cooling and any off-site radiological consequences must be within requirements set in 10CFR100. The quantitative acceptance criteria used in analysis of the accident are usually related to: peak radial average fuel enthalpy (<230 cal/g), peak fuel centreline temperature (below UO$_2$ melting temperature) or size of the radius of melting zone at the hot spot (<10% of the radius of the pellet), maximum cladding temperature (<1482.2 C, rarely used) or cladding integrity.

The calculation of the accident, as found in Chapter 15 of Safety Analysis Report, is usually divided into two parts, a neutron kinetic analysis to obtain average core nuclear transient results and a hot spot fuel heat transfer transient analysis to obtain local limiting thermal parameters to be compared against acceptance criteria. The neutron kinetic analysis is traditionally 1D, but it can be done in 3D geometry too. During the calculation different conservative assumptions related to input data or to exchange of the results between two parts of the calculation were used. The example is prescribed design hot channel factor (needed to relate average core conditions and hot spot location) and its time variation, assumption that steady state hot spot and transient hot spot coincide, conservatively high ejected reactivity worth and addition rate at conservatively low delayed neutron fraction, and so on.

This paper shows one possible way to perform best-estimate realistic calculation of RCCA ejection for NPP Krsko, still using some conservative assumptions to see their influence on limiting core parameters. In order to do that 3D nodal code PARCS was used together with detailed thermal-hydraulics core model based on COBRA code. As the result of that first part of the calculation, location and heat input to the limiting hot pin were obtained and used in second part which is based on FRAPCON/FRAPTRAN code. The final results of the second part are hot pin local conditions.

2 CALCULATIONAL TOOLS, METHODOLOGY, AND DATA

The main computer tools used in the RCCA ejection calculation, or any other similar type of the accident, are:

- 2D transport collision probability code FA2D v1.01c for calculation of cross section data at fuel assembly level,
- nodal diffusion code PARCS v.2.5 for 3D depletion calculation and transient calculation of reactivity initiated transient (RIA),
- COBRA-VIP code for steady state and transient calculation of thermal-hydraulics feedbacks and DNBR within PARCS,
- FRAPCON v3.3 for steady state calculation of fuel rod thermal and mechanical properties (evaluates dependency of fuel parameters on burnup) and FRAPTRAN v1.3 for transient response of limiting fuel rod.

Using the mentioned calculational tools it is possible to integrate fuel management and steady-state and transient safety analyses [1]. In the preparatory phase required cross section data were produced and the depletion calculation of the fuel management type for existing loading scheme was performed. The obtained burnup dependent data can be used to start further steady-state or transient calculations at any point within the fuel cycle.

Important part of any spatial neutronic calculation is preparation of cross section libraries. In this case homogenized, burnup dependent two-group cross section data at fuel assembly level were calculated using FA2D, 2D collision probability transport code
developed at FER. A full NPP Krsko fuel assembly is modeled due to asymmetry of 16x16 configuration, especially for some IFBA (Integral Fuel Burnable Absorber) combinations. The performed calculations have similar characteristics as usual core design calculations, but can use less detailed discretization. The depletion calculations at fuel assembly (FA) level are at the nominal power density, average boron concentration and average thermal-hydraulic conditions (fuel average temperature, clad temperature, moderator density, without control rod insertion). The separate library is prepared for each material composition used in the core. The material composition is determined by fuel enrichment, burnable absorber enrichment and number of IFBAs. For NEK cycle 21, which was used as a testing platform for this RIA calculation, 13 fuel compositions and 3 reflector compositions are used. In order to prepare information needed for thermal-hydraulics feedback modeling branch point calculations were performed at selected burnup points using isotopic compositions calculated during depletion under average conditions. Only 3-4 branch points in each relevant variable (moderator temperature/density, fuel temperature, boron concentration) were used to keep the number of calculations at a reasonable level. It is possible to perform additional calculations in order to include burnup history (spectral) effects in produced cross section library. The post-processing program saves cross section (XS) 2-group data for each material composition in a format similar to the cross section library format used in the OECD MSLB benchmark. In addition to usual macroscopic cross sections, discontinuity factors were saved, microscopic cross sections, yield fractions and decay constants for Xenon (Iodine) and Samarium (Promethium), average neutron fluxes, power factors to be used in pin power reconstruction (in separate library), delayed neutron data (needed only for transient calculation) and multiplication factors for each of the branch points. A trilinear interpolation procedure (similar to the one used in 3D linear Finite Element Method (FEM)) is part of the library implementation. A separate cross section library of the same structure is used to describe rodded fuel assemblies.

A modified version of the NRC’s 3D nodal code PARCS v2.5 [2] was used for depletion at core level and for transient calculation. The modification of the code was done in order to provide internal depletion capability (not originally available in that version) and to make possible usage of cross section tables prepared by FA2D code. In addition code is able to perform multi-cycle fuel management calculation. All required data, including burnup from previous cycles, and their axial variations are provided within new fuel assembly related FAS files (idea is similar to the one presently used by Institute Jozef Stefan in CORD-2 package [3]). The same cross section library is used for depletion and for subsequent steady state or transient calculations. Local cross section data were calculated within PARCS for given thermal-hydraulics conditions using trilinear interpolation for two burnup points of the interval containing the wanted local burnup (final values are result of linear interpolation between two burnup points). As one of the outputs in this fuel management calculation phase (that calculation should be performed first) 3D burnup distribution and 3D distributions of history variables for each cycle burnup point are provided. That information and corresponding interpolation procedure makes possible later calculation at any cycle burnup. When used in fuel management or transient role PARCS model assumes one node per assembly or 2×2 nodes per fuel assembly. The thermal-hydraulic (TH) model assumes one TH channel per neutronic calculational cell and can be realized using available internal model or more accurate subchannel COBRA-VIP code. A pin power reconstruction can be included or excluded from the calculational loop.

Our version of PARCS code includes simplified internal thermal model (water and steam properties based on IF-97 formulation, and fuel and cladding thermal properties from MATPRO library, [4]), and as an additional option it is possible to use integrated core thermal-hydraulics model based on COBRA-VIP code to perform more accurate thermal-
hydraulics core calculation. COBRA-VIP is based on COBRA-EN [5] and COBRA IV-I [6] computer codes with some capabilities found in EPRI (Electric Power Research Institute) VIPRE code (mostly related to input processing). The code has water and steam properties based on IF-97 formulation and fuel and cladding thermal properties from MATPRO library. It is consistent with related implementation found in updated PARCS internal TH model. The COBRA-VIP can be used in three roles: to perform improved thermal-hydraulic feedback calculation for open core, to provide pin-by-pin thermal calculation of selected fuel assembly and for core-wide DNBR calculation. The separate part of the code is developed to facilitate automatic preparation of channel and fuel input data using minimum information on core and FA layout. When coupled to PARCS it can be called as part of usual steady state and/or transient runs or as part of depletion run. COBRA heat structures have heat source calculated in PARCS code, which is responsible for core neutron kinetics. Thermal-hydraulic variables needed in PARCS TH feedback calculation are taken from COBRA-VIP channels and fuel rods. When using subchannel core model each fuel assembly can be modeled as one channel (or as for channels using 2x2 discretization) in reasonable time taking into account radial mixing (open channels) and homogenous two-phase flow model. In axial direction both codes usually use the same spatial discretization, but that is not a requirement and each code can have its own axial mesh sizes as long as they share common boundaries. In present situation thermal-hydraulic calculation, including DNBR calculation, is automatically performed for fuel assembly average channel. For hot fuel assembly it is possible to perform pin by pin subchannel calculation in steady state using pin power reconstruction data from PARCS. The classical subchannel approach was used due to its capability for whole core calculation in reasonable period of time and due to natural extension to fuel assembly pin by pin DNBR calculation. As side effect of COBRA-VIP integration option for additional output of different 3D distribution exists. As a result of coupled PARCS/COBRA run hot assembly can be identified using following criteria: maximum rod linear power density or maximum energy input (with or without pin power reconstruction data) or minimum DNBR for average TH channel per assembly. A limiting fuel pin can be analyzed in subsequent FRAPTRAN run with data retrieved from PARCS output (time dependent average liner power density based on pin power reconstruction data and axial shape for average channel per FA).

FRAPCON [7] and FRAPTRAN [8] are US NRC codes developed for licensing calculation of fuel rod during burnup and transient, respectively. The FRAPCON v 3.3 is modified to allow axial variation of enrichment and pellet geometrical properties as well as to include model for additional He production in IFBA rods. In this work FRAPCON is used to prepare initial data for FRAPTRAN transient runs. A separate file is prepared for average pin for each fuel assembly using linear power densities and axial shapes obtained during PARCS depletion run. FRAPTRAN v 1.3 run can be applied after completed PARCS transient run to calculate limiting rod for hot fuel assembly (identified during PARCS run), or for any rod for which corresponding FRAPCON burnup dependent data were prepared.

Using the described tools and calculation flow it is possible to prepare realistic 3D analysis of control rod ejection accident (or any other RIA accident), identify hot spot and calculate local fuel and cladding properties for limiting fuel rod in systematic way. When rest of the primary loop, outside of the reactor core, has important influence on accident progression it is possible to use PARCS/COBRA code system coupled to RELAP5 mod 3.3 code, [9].
3 CONTROL ROD EJECTION CALCULATION

A control rod ejection calculation performed for NPP Krsko Cycle 21 was used to illustrate the described calculational procedure. The Reference loading pattern for Cycle 21 (Figure 1), was taken from NPP Krsko Nuclear Design Report (NDR), [10].

The depletion calculation up to the end of the cycle was performed first, for existing cross section libraries with 13 material compositions. The full core model with 2x2 nodes per FA was used in PARCS calculation. 24 equidistant layers exist in axial direction. The reflector is taken into account explicitly using fictive reflector cells of the same size as fuel assemblies. The COBRA-VIP model has one TH channel and one equivalent fuel rod per PARCS calculation node and the same axial subdivision. 8 nodes in fuel pellet and 2 nodes in cladding were used in COBRA-VIP fuel rod. The fuel rod gap size is fixed and heat transfer coefficient is constant at 10000 W/m²K. The lateral communication between channels is enabled, so an open core TH model was used. All inlet coolant conditions are nominal during depletion. The obtained results are compared against NDR data. The resulting 3D burnup and history variables distributions are saved for subsequent use. The obtained results are typically within 40 ppm of the reference NDR results for critical boron concentration, and within 5% of the relative fuel assembly powers. An accurate prediction of axial power shape is important for rod ejection accident. The differences between End of Cycle (EOC) axial power distributions found in plant’s NDR and obtained by PARCS are taken as satisfactory for this type of the verification.

When depletion calculation is completed, in order to perform control rod ejection calculation it is enough to select cycle burnup, initial power level and position of control rod banks. For those conditions new critical boron concentration is determined. Even though some Xenon transient axial power shapes can be more limiting, for the purpose of this simulation equilibrium conditions are assumed at the beginning of the accident. During normal operation rod bank positions are determined by control rod insertion limits, [10].
position of the control rods are continuously indicated in control room and any deviation for more than 5% of the span will issue alarm. So, it is realistic assumption that initial control rod position will be determined by rod insertion limits.

In NPP Krsko case that means that D bank can’t be more inserted than position of 174 steps withdrawn at Hot Full Power (HFP) conditions. For Hot Zero Power (HZP) D bank is fully inserted and C and B banks have position 46 and 174 steps withdrawn, respectively. In HFP case that means that rod cluster from D bank is the only candidate for ejection and for HZP case, even though both bank C and B positions can be checked, the rod cluster from fully inserted D bank is again an obvious candidate for ejection. The control rod from bank D for which calculation was performed is located at position 7-K, [10]. Other three rods from D bank are on the same position in different quadrants of the core with quadrant symmetry. It is usual to perform control rod ejection calculations at least for Beginning of Cycle (BOC) and for EOC conditions. Due to smaller delayed neutron fraction in EOC case, due to axial power shape shifted to the peripheral part of the core (important for partially inserted D bank at HFP) and due to feedback effects, the performed calculations showed that EOC results (20.5 GWd/tU) are limiting in all cases. So, only EOC results for HFP and HZP will be shown in this paper. Two parameters of the core have main influence on severity of the control rod ejection accident (for given rod position, insertion depth, and cycle burnup): core delayed neutron fraction and ejected control rod worth. The calculated EOC core delayed neutron fraction was 519 pcm (effective NDR value is 518.6 pcm). In the used cross section model each fuel material composition has its own delayed neutron data that are burnup and TH variables dependent. NPP Krsko SAR used delayed neutron fraction of 430 pcm for performing RCCA ejection accident calculation. In modified PARCS it is possible to use uniform multiplier to modify overall core delayed neutron fraction and it was used to decrease core beta to 430 pcm. When determining what reactivity worth is of ejected control rod it is possible to use steady state calculation for inserted and withdrawn position of the rod to assess the value. The obtained values are close to the values obtained in real transient calculation. After performed calculation (with usual 10% uncertainty included in control rod reactivity worth) obtained reactivity worths are 40 pcm and 745 pcm for cluster from D bank ejected from insertion limit at HFP and HZP, respectively. The ejected reactivity worths assumed in SAR calculation were 300 pcm at HFP and 950 pcm for HZP conditions. In modified PARCS cross section model it is possible to apply correction factor to rodded composition macroscopic cross sections. The multiplier was used to increase ejected control rod reactivity fraction from 40 pcm to 57 pcm for HFP case (approx 40% increase) and from 745 pcm to 825 pcm for HZP case (approx 10% increase). Preliminary calculations were performed for three HFP cases and three HZP cases. First cases used decreased delayed neutrons fraction and increased ejected reactivity worth, second cases used nominal delayed neutrons fraction and increased ejected reactivity worth, and last one used nominal values for both delayed neutrons fraction and ejected reactivity worth. The nuclear power obtained in preliminary calculations for three HFP and three HZP cases are shown in Figure 2 and Figure 3 respectively. The peak power reached in HFP cases and the time of the peak are: 0.100 s 1.198 \( P_n \), 0.098 s 1.164 \( P_n \), and 0.098 s 1.119 \( P_n \). Corresponding values obtained in HZP cases are: 0.106 s 122.86 \( P_n \), 0.112 s 67.19 \( P_n \), and 0.118 s 32.38 \( P_n \). As expected, higher nuclear power peaks are obtained for smaller betas and larger ejected reactivity worth. The differences are larger in prompt critical HZP cases where shift of time of power peak toward smaller values can be seen too. In HFP cases only first case can produce power above high flux protection system set point (1.18 \( P_n \)). In order to address behaviour of the system using the same reasonable conservative assumptions for both HFP and HZP conditions only first case will be presented in more detail in this paper.
Figure 2: Nuclear power during HFP ejection with different betas and ejected worth

Figure 3: Nuclear power during HZP ejection with different betas and ejected worth
3.1 HFP EOC Control Rod Ejection

Control rod ejection was performed for NPP Krsko Cycle 21 EOC from HFP conditions. The cluster from D bank on position 7-K, inserted to the insertion limit (174 steps withdrawn), was ejected in 0.1 s. Initial conditions are nominal except that power was assumed to be 1.02 $P_n$. All inlet conditions are kept constant during calculation. Conservative DNBR analysis can ask for some lower than nominal mass flow rate, pressure lower than nominal and some increase in inlet coolant temperature, but it was not expected that min DNBR will reach limiting value for assumed ejected reactivity of 57 pcm (core delayed neutron fraction was decreased to 430 pcm). High neutron flux protection set point was assumed to be 1.18 $P_n$, with usual delay between trip and start of RCCA movement of 0.5 s. It was assumed that all rods except ejected one are available for scram insertion, and that travelling time is 2.8 s. Even decrease of the available scram reactivity to the value used in SAR calculation (1600 pcm) has no influence to the limiting values reached during the accident. The calculation was performed using 2x2 neutronic nodes per FA in PARCS model and corresponding 484 TH channels and fuel rods in COBRA-VIP model. The duration of the calculation is 10 s.

The behavior of the nuclear power, calculated by PARCS, and the power transferred to the coolant, calculated by COBRA-VIP, can be seen in Figure 4. Peak nuclear power was reached at 0.1 s (time when ejected rod left the core) and was 1.209 $P_n$. High neutron flux protection set point trip was reached little bit before that. Maximum of the thermal power was 2091 MW at 0.58 s. Used COBRA-VIP model assumes that 2.6% of the produced power will be directly in the coolant. Decay heat was not taken into account in the present calculation. Nuclear power peaking factor and radial peaking factors calculated without and with pin power reconstruction (PPR) are shown in Figure 5. In initial power distribution the depression due to 4 inserted D bank control rods exists. The depression near location 7-K is little bit deeper due to artificially increased reactivity worth of control rod to be ejected. With that configuration $F_q$ peaking factor is about 2.12. When ejected rod starts to move $F_q$ is first little bit decreased due to shift of the axial shape even though power of the corresponding assembly will start to increase, as can be seen from radial peaking factors. In any case the changes in peaking factors and power distribution are rather small and are increased only during scram movement (between 0.5 s and 3.3 s, $F_q$ reached in that period value of about 6.3, but at that time power level is already decreased below 10% of nominal). The reactivity change during the accident can be seen in Figure 6. The initial reactivity increase due to control rod ejection is almost linear. Close to the end of the movement it is partially suppressed due to negative fuel and moderator feedbacks. Due to rather high negative EOC moderator temperature coefficient, negative reactivities caused by fuel temperature and moderator density change are almost equal at 0.2 s. Due to EOC axial power shapes inserted control rods are rather effective and their influence can be seen shortly before 0.6 s. The core is again subcritical at about 0.63. As can be seen scram initiation and inserted reactivity are not related to reaching peak power values but can influence total energy deposited in the fuel. The assumptions used in this calculation are in that sense best estimate (10% reduction is taken into account in control rod reactivity worths).

The maximum fuel center temperature and the core average fuel temperature are predicted by COBRA-VIP (2x2 per FA model). The maximum temperature means current global core maximum and it is not necessary all the time at the same point of the core. The increase of the fuel temperature is generally small, slow and smooth. The maximum is around 2018 K and it is reached at 0.62 s. 2D distribution of maximum of the fuel centre temperature at t=0.8 s is shown in Figure 7. The temperatures at the position of ejected rod are little bit above other temperatures in corresponding positions of other quadrants.
Figure 4: Nuclear power and power to coolant for HFP ejection

Figure 5: Power peaking factors during HFP ejection
Figure 6: Reactivity change during HFP ejection

Figure 7: 2D distribution of maximum centreline temperature at 0.8 s

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The COBRA-VIP TH core model predicted core-wide min DNBR values based on average channel per FA calculation. The minimum DNBR, reached in fuel assembly where ejected rod was located, is 3.14 at 0.63 s (Figure 8). That is result of small decrease compared to steady state value. Axial distribution of limiting min core-wide DNBR values (minimum DNBR value for that axial position taking into account all core channels) as predicted by Westinghouse WRB-1 and W-3 correlations at t=0.7 s is shown in Figure 9. Both correlations predicted similar behaviour of min DNBR. Hot channel calculation can’t reduce min DNBR below safety analysis limit and it was not performed.

The hot spot of the core predicted in PARCS calculation using any of the available criteria (peak linear power density, maximum energy input and minimum DNBR) was located in fuel assembly Y-35 at the core position 7-K. Using 3D rod linear power density from PARCS calculation the data needed for FRAPTRAN calculation are extracted. The time dependent spatial distribution of linear power density for location 7-K was split in time dependent average rod linear power density, Figure 10, and number of axial power shapes. The number of power shapes is reduced using change in related axial offset as criterion for reduction. The FRAPTRAN hot pin input data for the same assembly are defined using average pin axial power shapes and time dependent average linear power density determined taking into account 2D pin power reconstruction data. Once when the assembly of the hot power pin is determined FRAPTRAN time dependent linear power density values are constructed using one additional conservative assumption. The hottest pin from the selected assembly at the current time (so it is not necessary the same fuel pin all the time) is used to produce FRAPTRAN input vector.

The axial distribution of radially averaged fuel enthalpy for average and hot pin of FA Y-35 is shown in Figure 11 for times t= 0.1 s, 0.3 s and 0.8 s. It can be seen that the peak fuel enthalpy practically stays constant, both in value and position (the peak value is around 70 cal/g for hot pin). The increase of the power produced in analysed fuel pin is mostly due to increase in upper part that was under influence of inserted control rod (enthalpy increase compared to steady state value is only about 6 cal/g and it is again located in upper part of the fuel rod). The axial distribution of the centreline temperature has almost the same behaviour. Peak temperature for hot pin is around 1950 K. The peak value in average pin is for about 70 K lower. The peak fuel temperature as calculated by COBRA-VIP 2x2 model was about 2017 K and it is obtained with lower linear power density. The FRAPTRAN model is more accurate than corresponding COBRA-VIP model and takes into account real behaviour in fuel rod gap. There are 24 mesh points in fuel pellet and 5 additional in gap and cladding in FRAPTRAN rod model. The radial depression in power produced within fuel rod is taken into account. Total number of mesh points in COBRA-VIP fuel rod was 8. In axial direction 24 equidistant subdivision were used. The radial temperature distribution in hot pin at t=0.3 s for axial position 1 and 4 (location of peak temperature) is shown in Figure 12. It can be seen that fuel rod gap for first axial layer still exists, but it is closed for higher elevations what decreases temperature drop in gap.

2D radial distribution of rod average linear power density with pin power reconstruction taken into account is shown for t=0.128 s (little bit after power peak) in Figure 13. The increased power production is present near position where the control rod was ejected. For the spatial distributions near time of the power peak it is due to reactivity insertion in that region and for spatial distribution toward end of the calculation it is because ejected control rod is missing and flux is shifted in that region by other inserted control rods. As shown before the power peaking factor is higher toward end of calculation than at the time of power peak, but produced power is much lower.
Figure 8: Change in core-wide min DNBR value during HFP ejection

Figure 9: Axial dependence of min DNBR in hot FA at t=0.7 s

Figure 10: Rod average linear power density for average and hot pin in hot FA Y-35
Figure 11: Axial distribution of radially averaged fuel enthalpy

Figure 12: Fuel temperature radial distribution in hot pin at 0.3 s, axial position 1 and 4

Figure 13: 2D rod linear power density with PPR at 0.128 s (4th quadrant)
3.2 HZP EOC Control Rod Ejection

The control rod ejection modeling was performed for NPP Krsko Cycle 21 EOC from HZP conditions. The fully inserted cluster from D bank on position 7-K was ejected in 0.1 s. Initial conditions are nominal except that power was assumed to be 0.1% of nominal power. All inlet conditions are kept constant during calculation. The assumed ejected reactivity was 825 pcm at core delayed neutron fraction of 430 pcm. The high neutron flux protection set point was assumed to be 0.35 Pn, with usual delay between trip and start of RCCA movement of 0.5 s. It was assumed that all rods except ejected one are available for scram insertion, and that travelling time for fully withdrawn control rod is 2.8 s. The decrease of the available scram reactivity was performed down to the value used in SAR calculation (1600 pcm), but had no influence to the limiting values reached during the accident. The calculation was performed using 2x2 neutronic nodes per FA in PARCS model and corresponding 484 TH channels and fuel rods in COBRA-VIP model. The duration of the calculation is 10 s.

The behavior of the nuclear power, calculated by PARCS, and the power transferred to the coolant, calculated by COBRA-VIP, can be seen in Figure 14. The peak nuclear power was reached at 0.106 s and was about 159 Pn, but was down to the nominal value at 0.116 s. The width of the peak at nominal power was about 0.02 s. The high neutron flux protection set point trip was reached at 0.095 s. The maximum of the thermal power was 2176 MW at 0.11 s, but power decreased almost immediately after that and increased again when heat energy was accumulated again. The decay heat was not taken into account in the present calculation. The nuclear power peaking factor and radial peaking factors calculated without and with pin power reconstruction (PPR) are shown in Figure 15. The initial power distribution (rod linear power density) at axial level 20 can be seen in Figure 17. The depression due to 4 inserted D bank control rods can be seen. The depression in front of the figure is deeper due to artificially increased reactivity worth of control rod to be ejected and the asymmetry effect is more significant than in HFP case. Another factor influencing the distribution is HZP axial power profile which is shifted toward top of the core. All peaking factors are larger than in HFP case, as expected. The maximum of the F_q peaking factor was 8.4 at 0.1 s. When ejected rod starts to move most of the power increase is in the region close to that location. Immediately after ejection F_q decreased, then increased during scram movement (between 0.5 s and 3.3 s), and then additionally increased toward end of the calculation due to power shift from rodded locations to missing control rod location at position 7-K. The reactivity change during the accident can be seen in Figure 16. The inserted reactivity was about 800 pcm. The core was prompt critical. Close to the end of the movement the core was made again subcritical due to Doppler feedback and sudden increase of the fuel temperature. Some small recriticality was experienced little bit later, but with increase of coolant temperature core was made again subcritical after 0.17 s. Due to rather high negative EOC moderator temperature coefficient, negative reactivities caused by fuel temperature and moderator density change are almost equal at 0.35 s. Due to EOC HZP axial power shape inserted control rods are rather effective and their influence can be seen at 0.6 s. As before, scram initiation and inserted reactivity are not related to reaching peak power values but can influence total energy deposited in the fuel. The maximum fuel center temperature and core average fuel temperature as predicted by COBRA-VIP 2x2 per FA model are shown in Figure 18. The maximum temperature means current global core maximum and it is not necessary all the time at the same point of the core. The increase of the fuel temperature is initially very fast, but then maximum fuel temperature stays below 1217 K. The maximum is reached at about 1. s. 2D distribution of maximum of the fuel centre temperature at t=1.0 s is shown in Figure 19. The temperatures around position of the ejected control rod are much higher than in other parts of the core what is indication of localized power increase.
Figure 14: Nuclear power and power to coolant for HZP ejection

Figure 15: Power peaking factors during HZP ejection

Figure 16: Reactivity change during HZP ejection
Figure 17: Average rod linear power density [kW/m], HZP, t=0.0 s, ax level 20

Figure 18: Change in maximum and average fuel temperature during HZP ejection

Figure 19: 2D distribution of maximum centreline temperature [K] at 1.0 s
The COBRA-VIP TH core model was again used to predict core-wide min DNBR values based on average channel per FA calculation. The min DNBR value of 1.57 reached in fuel assembly Y-52 on location L-7 adjacent to the ejected rod location at 0.17 s, Figure 20. Taking into account axial power shape and amount of localized power increase decrease of min DNBR value was expected. The axial distribution of limiting min core-wide DNBR values (minimum DNBR value for given axial position taking into account all core channels) as predicted by Westinghouse WRB-1 and W-3 correlations at t=0.17 s is shown in Figure 21. WRB-1 correlation predicted little bit smaller min DNBR values toward exit of the channel. The min DNBR values reached in neighbouring channels, including ejected rod position 7-K, were very close to global DNBR minimum. The hot channel min DNBR calculation was not performed. This time further reduction of min DNBR is possible, close to the safety analysis limit. The duration of DNBR promoting conditions is in any case very short and limited to locations around ejected rod position.

The hot spot of the core predicted in PARCS calculation using any of the available criteria (peak linear power density, maximum energy input and minimum DNBR) was located in fuel assembly Y-52 at the core position 7-L. That is just below ejected control rod position and the difference between hot rod in Y-35 and Y-52 is marginal. Using 3D rod linear power density from PARCS calculation the data needed for FRAPTRAN calculation are again extracted. Time dependent spatial distribution of linear power density for location 7-K and 7-L were split in time dependent average rod linear power density, and number of axial power shapes. The axial power shapes on position 7-K, fuel assembly Y-35, during ejected control rod movement are shown in Figure 22. The initial HZP axial power profile is changed during control rod ejection and in that period of time power shape follows tip of the moving rod. When control rod is ejected, profile is again similar to the initial one, but with higher peak in upper part. After control rod ejection the axial power shape is mostly determined by scram rods movement and after that it is close to normal HZP axial power profile. The axial power profile at location 7-L is almost the same as described one. The global axial power profile is again similar, but with less significant changes related to ejected control rod influence. The number of power shapes is again reduced using change in related axial offset as criterion for reduction, but due to more intensive change in power shapes more profiles were used than for HFP. The FRAPTRAN hot pin input data for assemblies Y-35 and Y-52 are defined using average pin axial power shapes and time dependent average linear power density determined, taking into account 2D pin power reconstruction data.

The average fuel temperature for average pin from Y-35 and limiting pins from Y-35 and Y-52 are shown in Figure 23. The hot pins in both fuel assemblies have almost the same fuel temperatures. The axial distribution of fuel enthalpy increase after steady state for average and hot pin from Y-35 and for hot pin from Y-52 at 0.12 s are shown in Figure 24. Both hot pins have very similar peak enthalpy increase values around 50 cal/g. As expected, the time of peak of fuel enthalpy increase is close to the time of power peak. The fuel centreline temperatures have very similar axial distribution, but their peak is reached around 1.3 s. In case of hot pin from Y-52 maximum value is 1297 K, and in the case of hot pin from Y-35 maximum value is for 27 K lower. In the same time average pin temperature maximum is around 900 K. The peak fuel temperature as calculated by COBRA-VIP 2x2 model was about 1217 K. This time COBRA predicted the temperature which is for about 80 K lower than FRAPTRAN maximum temperature. This time simplified COBRA model was not able to compensate for higher peaking values obtained in hot pins based on PPR results.

The 2D radial distribution of rod average linear power density with pin power reconstruction taken into account is shown in Figure 25 for t=0.1 s. Compared to HFP case less uniform power distribution can be seen with more power produced close to ejected control rod location. The similar situation is present for all later power distributions.
Figure 20: Change in core-wide min DNBR value during HZP ejection

Figure 21: Axial dependence of min DNBR in hot FA at t=0.17 s

Figure 22: Axial power shapes at hot assembly position during control rod ejection
Figure 23: Average fuel temperature for average and limiting pins

Figure 24: Fuel enthalpy increase after steady state for average and limiting pins

Figure 25: 2D rod linear power density with PPR at 0.1 s (4th quadrant)
4 CONCLUSION

Capabilities of calculation procedure which includes 3D nodal diffusion code PARCS, and core model based on COBRA-VIP code for global core calculations and FRAPCON/FRAPTRAN code for hot pin calculation were demonstrated for realistic simulation of NPP Krsko control rod ejection accident. EOC HFP and HZP results were presented. The results are to some extent different than results usually found in SAR Chapter 15 of the plant. That is partially due to the assumptions that are more close to realistic situation and partially due to different calculational tools. The important influence of core delayed fraction and ejected control rod reactivity was illustrated, as well as differences between HFP and HZP control rod ejection accidents. In first case milder transient was experienced with lower power peak and peaking factors, but with higher maximum fuel temperatures. In second case sudden change in power level and localized power production was experienced in prompt critical power increase. In both cases the final values obtained are well within acceptance criteria and are less serious than corresponding SAR cases. The predicted HZP behavior is closer to the classic SAR behavior due to initially fully inserted control rod. The HFP control rod ejection modeling was performed for control rods at rod insertion limit what puts limit on maximum possible ejected reactivity. The calculational procedure can give more insight to thermal and mechanical cladding behaviour during control rod ejection accident what can be benefit in case of new Standard Review Plan 4.2 rev. 3 interim criteria application.

REFERENCES


Application of the System Thermal-Hydraulics Codes for the Evaluation of the Operating PWR Plant Safety in the Shutdown Conditions

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ABSTRACT

Deterministic safety analyses have been an important tool for confirming the adequacy and efficiency of provisions within the defense in depth concept for the safety of nuclear power plants. Traditionally, nominal power conditions have been evaluated in this frame. However, in-depth studies revealed that un-negligible possibility to severely damage the core also exists in the shutdown conditions.

Thermal-hydraulic system codes are essential tool for the deterministic safety analyses and are suitable to calculate complex accident scenarios expected in the shutdown conditions of the water cooled nuclear reactors. The outputs of those analyses are essential for the estimation of the safety margins of the nuclear power plants in the specific situation of the shutdown, maintenance conditions.

Low-power, low inventory, presence of non-condensable gases in the primary systems and unavailability of safety systems are just some of the peculiarities that need to be investigated in order to estimate safety margins during the plant outages.

The present paper covers the applications of the complex, system thermal-calculation codes in the application for the pressurized water plant for the estimation of the safety margins during the plant outages. It also discusses methodologies applied as well the outcomes of these analyses and implications on the plant operational safety during outage activities.

1 INTRODUCTION

In the past decade, operational experience and performance of the probabilistic safety analyses for the shutdown modes have indicated the importance of the risk contribution from those, previously considered safe operating modes. Most part of the risk comes from the unavailability of equipment due to maintenance activities undertaken during an outage. Adequate planning and preparation of activities during outages reduce both the probability and the consequences of possible events. Safety studies performed to-date on Nuclear Power Plants (NPP) clearly indicated that the reduced inventory situations are the most critical
periods. Special attention should be paid to loss of Reactor Coolant System (RCS) inventory and loss of residual heat removal events during shutdown operation of Pressurized Water Reactors (PWR). The shutdown operation represents a higher risk condition because of many "single" failure potentials due to various maintenance and operational configurations. Also, in modes 5 and 6 (low temperature and pressure) plant operates at the reduced RCS inventory and loss of Residual Heat Removal (RHR) system poses significant potential for severe core damage. The consequences of the loss of RHR system depend on a variety of factors, such as the configuration of the RCS (geometry of reactor, number of loops, position and status of the openings in the system), the mass of liquid in the system and the time after shutdown at which the transient occurred. Loss of RHR system causes the loss of forced flow through the core and coolant heat-up followed by the boiling in the RCS. If the RCS boundary is open the boiling will occur at atmospheric pressure. If the RCS boundary is closed boiling will occur at the RHR relief valve setpoint. The short time to start to boiling is associated with high decay power level and low RCS pressure. The presence of the openings in the system has two major consequences on transient outcome. First, due to low pressure (atmospheric conditions), a quick boiling in the core will occur. Secondly, the RCS inventory may be lost through the openings by either dislocating and spilling the liquid or by direct boil-off steam from the core.

2 MATHEMATICAL MODEL OF NPP KRŠKO

2.1 Description of the Nodalization

For the analysis of the shutdown conditions transients the RELAP5/mod3.3 model for NPP Krško developed at FER Zagreb has been used. The model is based on the RELAP5/mod3.3 model that has been developed in compliance with [1], described in [2] and qualified on the steady state level, [3]. Previously, the RELAP5/MOD2 model for NPP Krško that includes a detailed model of the RHR system has been developed and used for the analysis of the transients in the shutdown conditions, [4]. For the purpose of the evaluation of the safety margins during shutdown for NPP Krško, the model of the RHR system from [4] has been upgraded and included in the base NPP Krško model described in [2]. Parts of the Nuclear Steam Supply System (NSSS) important for power operation that are not used at shutdown have been excluded from the model. The following are the differences of the NPP Krško model used in the analysis when compared with the base model: 1) Models that are omitted in the model include Feedwater system and Auxiliary Feedwater (AFW) system. Control systems that are not used during shutdown were omitted in the model (e.g., pressurizer pressure and level control system, rod control system, Steam Generator (SG) level control system, etc.). The RELAP5/mod3.3 model (including RHR system) used in this analysis consists of 765 volumes and 793 junctions. The model has 273 heat structures with total number of mesh points equal to 1785. Both RHR trains were modelled. The RHR inlet and outlet are connected with the RCS hot and cold legs by downward oriented valve junctions (from the hot legs) and with the valves to the Emergency Core Cooling System (ECCS) accumulator injection point (to the cold legs), respectively. The main components of the RHR system are the RHR pump, RHR heat exchanger and the accompanying isolation valves. Each train to the inlet of the RHR system is equipped with a pressure relief valve (RHR valve 1 - upflow of the RHR pump) aimed to protect the system from inadvertent overpressurization during plant cooldown and startup. On the RHR discharge side each RHR train is equipped with a pressure relief valve (RHR valve 2 - downflow of the RHR pump) aimed to relieve the back-leakage flow through the valves separating the RHR system from the RCS.
2.2 Qualification of the Nodalization

For the purpose of the on-transient qualification for shutdown conditions calculations, BETHSY test 6.9c has been scaled, [5]. Some differences from the NPP Krško model described in [2] were introduced. In the experiment the SG secondary sides were full of air that was maintained at constant temperature. The heat transfer from the U tubes to the secondary side was found to be of no significance. According to the recommendation given in [6], SG secondary sides were not modelled. The ECCS model was omitted in the input data set and a model of the gravity feed injection in the cold leg 2 was introduced. Only control variables for the calculation of parameters measured in the experiment and of gravity feed flow were modelled. Three sensitivity study analyses for NPP Krško were performed in order to investigate influence of different scaling assumptions. Position and orientation of the pressurizer and SG 1 manway are sketched in Figure 1. Contrary to the NPP Krško, in the BETHSY facility, the pressurizer manway is situated at the ending of the vertical pipe, which is positioned on the top of the pressurizer, Figure 1. For the SG manway, there are differences both in the position and orientation. In the BETHSY, the SG manway is situated on the vertical part of the intermediate leg 1. Its axis, which is horizontal is located 720 mm above the hot leg elevation. In the NPP Krško, the SG manway is situated at the bottom of the SG outlet plenum, Figure 1. Both manway openings are modelled by valve component. The parameters scaled up in all three NPP Krško sensitivity study cases were: core power, manway flow areas and gravity feed injection flow. These parameters were obtained by multiplying the related BETHSY parameters by scaling factor \( K_v \). Scaling factor \( K_v \) was obtained by dividing the NPP Krško (NEK) primary system volume (195.3 m\(^3\), [2]) by the BETHSY volume of the primary system (2.88 m\(^3\), [7]) giving ratio of 67.81. Fuel rod material as well as core power axial distribution for the NPP Krško were the same as in the experiment.

Initial conditions for the experiment and NPP Krško sensitivity study calculations are summarized in Table 1. Initial conditions were obtained after 2000 seconds of steady state calculation. Before the steady state calculation the primary system was filled with saturated steam at atmospheric pressure. The reactor coolant system was then filled with water until the requested conditions for the sensitivity study analysis were achieved. In the NEK CASE 1 and NEK CASE 3, the initial primary mass was adjusted to achieve midloop conditions. In the NEK CASE 2, the initial primary mass was \( K_v \) scaled. In the NEK CASE 2, the hot as well as cold legs were thus liquid filled.

Table 1: Initial and boundary conditions for the experiment and NPP Krško \( K_v \) scaled calculations

<table>
<thead>
<tr>
<th>QUANTITY</th>
<th>UNIT</th>
<th>Experiment scaled</th>
<th>NEK CASE 1</th>
<th>NEK CASE 2</th>
<th>NEK CASE 3</th>
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<td>61109</td>
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<td>Reactor vessel mass</td>
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<td>46835</td>
<td>42320</td>
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<td>Hot leg void fraction</td>
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<tr>
<td>Cold leg void fraction</td>
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<td>0.0/0.0/0.0/0.0/0.0</td>
<td>0.49/0.57</td>
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<tr>
<td>Hot leg temperature</td>
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<td>373.7/373.7</td>
<td>374.1/374.1</td>
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<tr>
<td>Cold leg temperature</td>
<td>K</td>
<td>366/366/366</td>
<td>373.8/373.8</td>
<td>373.3/373.3</td>
<td>373.7/373.7</td>
</tr>
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</table>
3 NPP KRŠKO ACCIDENT ANALYSIS IN SHUTDOWN CONDITIONS

Specific configuration of the plant in the shutdown conditions is characterised with the number of systems in manual mode, permissive blocks on protection signals activated and reduced capability of the safety systems. In these conditions timely action of the plant operators is essential for successful coping with the potential accident conditions. Therefore, in order to estimate time available for operator action, a series of the analyses were conducted with aim to estimate time for operator to react.

3.1 Analysis of Small Break LOCA During Mode 3 and Mode 4

Small Break Loss of Coolant Accident (SB LOCA) analysis using RELAP5/Mod3.3 computer code during NPP Krško operation in shutdown modes (Mode 3 and Mode 4) was done in order to test the performance of the ECCS to mitigate such an event. ECCS is normally tested against a large break LOCA occurring at full power conditions because much higher core heat generation rates exist during full power operation. Nevertheless, ECCS system should provide protection in all modes of operation.

Figure 1: Nodalization of the pressurizer and SG manway for the BETHSY test 6.9 c and NPP Krško sensitivity study calculation
Mode 3 (hot standby) is defined as a condition where $K_{\text{eff}}$ is less than 0.99 and the RCS temperature is greater than 450 K. Mode 4 (hot shutdown) is defined as a condition where $K_{\text{eff}}$ is less than 0.99 and the RCS temperature is between 366 K and 450 K.

Probability of a large break LOCA occurrence during Mode 3 and Mode 4 operation is very low, therefore only the analysis of the SB LOCA in the cold leg was performed.

During Mode 3 or Mode 4 operation, the normal alignment of ECCS equipment is different than during the power operation. Accumulators are isolated and the automatic safety injection signal on low pressurizer pressure is blocked. Therefore, in these modes of operation, the operator action is required to mitigate the consequences of a LOCA event.

The objective of the analyses was to demonstrate that the consequences of LOCA would be mitigated and the limits of 10CFR50.46 would not be exceeded if there was a timely operator action to initiate safety injection flow.

The break was located in the cold leg, which is the limiting break location, ref. [9], due to the fact that the ECCS piping is attached to the cold leg. The break located at the cold leg could result in the spilling of the safety injection (SI) flow through the break prior to entering the reactor vessel. Initiation of the SI flow, which has to be manually established, was assumed at 10 minutes after the LOCA symptoms were recognized (the loss of the pressurizer level and the loss of the hot leg subcooling), ref. [8]. It was considered that 10 minutes would be enough time for the operator to recognize LOCA symptoms and to take the proper action, ref. [8].

3.1.1 Steady-State Calculation for Modes 3 and 4

Prior to transient analyses steady state calculations for shutdown modes were performed for parameters to stabilize. The most important input parameters are listed in Table 2.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Mode 3</th>
<th>Mode 4</th>
</tr>
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<tbody>
<tr>
<td>RCS pressure</td>
<td>6.97 MPa</td>
<td>2.85 MPa</td>
</tr>
<tr>
<td>RCS temperature</td>
<td>493 K</td>
<td>450 K</td>
</tr>
<tr>
<td>SG pressure</td>
<td>2.25 MPa</td>
<td>0.93 MPa</td>
</tr>
<tr>
<td>Pressurizer level</td>
<td>20 %</td>
<td>20 %</td>
</tr>
<tr>
<td>SG NR level</td>
<td>69 %</td>
<td>69 %</td>
</tr>
<tr>
<td>Core power</td>
<td>19.73 MW</td>
<td>17.07 MW</td>
</tr>
</tbody>
</table>

RCS pressure and temperature for Mode 3 operation were taken from ref. [10]. Those are the conditions at which accumulator isolation valves are closed, since the assumption in the calculation was that no accumulators would be available during the transient. For Mode 4 operation RCS pressure and temperature correspond to the conditions at which RHR system can be placed in operation, ref. [11]. SG pressures of 2.25 MPa for Mode 3 and 0.93 MPa for Mode 4 operation are the saturation pressures at temperatures 491.5 K and 450 K, respectively, which were the SG steam temperatures assumed in the analysis, ref. [10].

The pressurizer level was conservatively assumed to be at 20 % level, ref. [10]. That assumption was based on the minimum no-load pressurizer level for Westinghouse plants. SG narrow range (NR) level was maintained at approximately 69 % according to ref. [11].

The decay heat level was determined based on a 28 K per hour cooldown rate from a temperature of 561 K, ref. [10]. A plant needs to be shut down for 2.5 hours to get to the desired temperature 491.5 K for Mode 3 operation, ref. [10], and 4 hours to cool down to the temperature 450 K for Mode 4 operation, ref. [10]. Core decay heat values corresponding to the shutdown times of 2.5 h and 4 h were 19.73 MW and 17.07 MW, respectively.
Conservatively, constant decay heat values were assumed in the analyses during the steady-state and the transient calculations.

3.1.2 Transient Analysis

The analyzed transient was 6-inch cold leg pipe break in the loop that contains pressurizer, during Mode 3 and Mode 4 operation. It was assumed that the offsite power was lost at the start of the accident. That lead to the Reactor Coolant Pumps (RCPs) trip and the isolation of the steam lines. AFW, which provided water injection to the SGs, was shut off at the start of the accident due to the assumption of the loss of the offsite power. However, for 6-inch breaks the role of AFW is not significant since the SGs act as a heat sink for only a short time unless the operator takes actions to depressurize them.

The initiation of the break lead to a rapid depressurization of the RCS (Figure 2). The pressurizer drained quickly and emptied at approximately 5 s. The core uncovering process was slower in Mode 4 than in Mode 3 operation because the RCS pressure was lower so the RCS coolant was discharged through the break at a slower rate. The depressurization of the primary system was stopped when the liquid in the hot legs started to evaporate. The pressure remained constant until the loop seals have cleared. There was a sudden increase in the break mass flow rate following the loop seal clearing because vapour from the hot legs pushed water from the loop seals to the break. In addition, the system pressure started again to decrease (Figure 2).

The accident was mitigated by manual initiation of safety injection flow at 10 minutes after both loss of the pressurizer level and the loss of the hot leg subcooling occurred. During Mode 3 or Mode 4 operation the normal alignment of ECCS equipment is changed from that which is available during the power operation. The cold leg accumulators are isolated when the system pressure is below 6.97 MPa, ref. [10], to avoid the injection when the pressure is reduced. Also, the automatic safety injection signal on low pressurizer pressure is blocked at 12.3 MPa, ref. [10]. Therefore, in these modes of operation, operator action is required to mitigate the consequences of a LOCA event.

The difference in Mode 3 and Mode 4 transient entry conditions was the availability of the Safety Injection (SI) pumps, ref. [8]. One high head SI pump and one low head (RHR) SI pump were available during Mode 3 operation. During Mode 4 operation the assumption was that only one high head SI pump would be available for the safety injection. In Mode 4 RHR system is prepared for the operation and there is a potential concern related to the operability of the RHR pumps following their realignment for the safety injection. Successful short term mitigation with the high head SI pumps will provide enough time for the RHR system to be vented and cooled, if needed, so that an RHR pump can be later used for recirculation or cooldown.

Additional set of calculations without the safety injection flow was performed. The time when the core exit temperature reached 923 K was an indicator of the core damage. In these cases the core damage occurred at 2300 s for Mode 3 and at 3160 s for Mode 4 operation.

In case for Mode 3 with SI flow, core water level started to increase because the break flow rate was smaller than the SI flow rate. Once when the break flow rate and the SI flow rate equalized the core water level was stabilized at around 80 %. In Mode 4 operation SI flow was initiated at 810 s. There was no significant increase in the core water level as in the case of Mode 3 SB LOCA because the break flow rate was more or less the same as the SI flow rate.

The Refuelling Water Storage Tank (RWST) that provides SI flow capacity to deliver was also evaluated and it has been that in Mode 3 operation the RWST will be emptied at 8925 s and in Mode 4 at 36500 s. Therefore, prior to depletion of the RWST, the operator has
sufficient time to establish recirculation to maintain the RCS inventory to remove the decay heat. There was no core cladding temperature increase following the LOCA and considerable margin to 1477 K limit of 10CFR50.46 was obtained in cases where SI was actuated.

Time sequence of the main events is shown in Table 3.

Table 3: Time sequence of the main events

<table>
<thead>
<tr>
<th>Event</th>
<th>Mode 3</th>
<th>Mode 4</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pressurizer empty</td>
<td>5 s</td>
<td>5 s</td>
</tr>
<tr>
<td>Loop seal clearing</td>
<td>300 s</td>
<td>450 s</td>
</tr>
<tr>
<td>Start of the SI injection</td>
<td>670 s</td>
<td>810 s</td>
</tr>
<tr>
<td>RWST empty</td>
<td>8925 s</td>
<td>36500 s</td>
</tr>
<tr>
<td>Core damage (with no SI injection)</td>
<td>2300 s</td>
<td>3160 s</td>
</tr>
</tbody>
</table>

Figure 2: RCS pressure

Figure 3: Core water level
3.2 Analysis of the Loss of the RHR System During Low Temperature Shutdown Conditions

The analyses aimed to identify time available to act based on the estimation of the time of the start of the boiling in the core, dynamics of the loss of the inventory through the openings, time of the core uncovery and the begin of the fuel cladding temperature rise. These times are used in plant procedures for the establishment of needed capacity and allowable configurations of the safety systems.

3.2.1 Steady-State Calculation for Modes 5 and 6

Five different initial conditions have been considered:
- Case 1: RCS closed and water solid
- Case 2: RCS closed and partially drained, the rest filled with noncondensables
- Case 3: SG U tubes partially drained, pressurizer manway open
- Case 4: SG U tubes completely drained, pressurizer and SG manways open
- Case 5: SGs drained, SG nozzle dams installed, pressurizer manway open

The boundary and initial conditions for the analyzed cases are summarized in Table 4. A conservatively high decay power value that corresponds to one day after shutdown was assumed constant throughout the transient simulation.

In order to achieve steady state for the cases 1 through 5 the controlled draining of the RCS in accordance with the procedure described in [12] has been performed. The sequential procedure to achieve the steady state for the analyzed cases is described below.

Case 1 - Prior to the start of the controlled draining the RCS conditions correspond to plant shutdown state 1 with RCS water solid and both trains of RHR in operation. Hot leg pressure equals to 5.93E5 Pa.

Case 2 - The RCS is drained starting from the Case 1 using letdown flow until the level equal to 1.7 m above the center leg elevation is attained. Simultaneously, the air at pressure 4.71E5 Pa is admitted to the pressurizer and reactor vessel head through pressurizer relief valve and reactor vessel head vent system, respectively. The SG U tubes are filled with liquid.

Case 3 - The RCS is depressurized from the Case 1 to atmospheric conditions. The RCS level equal to 1.7 m above center leg elevation is maintained. The SG U-tubes are partially drained due to saturation conditions at the top. After steady state conditions had been attained, the pressurizer manway is opened.

Case 4 - Starting from the Case 3 the letdown flow is used to completely drain the SGs. The RCS level is controled at level: center leg (C.L.) elevation + 0.7 m. Finally, the inlet and outlet manways at both SGs are opened.

Case 5 - Starting from the Case 4, the SGs are isolated from the rest of the RCS.
Table 4: Initial and boundary conditions for the low temperature shutdown conditions

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Case 1</th>
<th>Case 2</th>
<th>Case 3</th>
<th>Case 4</th>
<th>Case 5</th>
<th>Case 6</th>
</tr>
</thead>
<tbody>
<tr>
<td>Hot leg pressure (Pa)</td>
<td>5.93E5</td>
<td>4.91E5</td>
<td>1.177E5</td>
<td>1.081E5</td>
<td>1.081E5</td>
<td>1.081E5</td>
</tr>
<tr>
<td>Average temperature (K)</td>
<td>333.15</td>
<td>333.18</td>
<td>333.17</td>
<td>333.19</td>
<td>333.19</td>
<td>333.19</td>
</tr>
<tr>
<td>Core inlet m. flow (kg/s)</td>
<td>269.8</td>
<td>269.8</td>
<td>271.1</td>
<td>269.2</td>
<td>269.2</td>
<td>269.2</td>
</tr>
<tr>
<td>Core power (MW)</td>
<td>12.41</td>
<td>12.41</td>
<td>12.41</td>
<td>12.41</td>
<td>12.41</td>
<td>12.41</td>
</tr>
<tr>
<td>Initial RCS mass (kg)</td>
<td>190188.</td>
<td>153604.</td>
<td>136836.</td>
<td>73409.</td>
<td>73409.</td>
<td>73409.</td>
</tr>
<tr>
<td>RCS level above center leg elevation (m)</td>
<td>RCS solid</td>
<td>1.7</td>
<td>1.7</td>
<td>0.7</td>
<td>0.7</td>
<td>0.7</td>
</tr>
<tr>
<td>Manways open</td>
<td>-</td>
<td>-</td>
<td>pressurizer</td>
<td>pressurizer</td>
<td>pressurizer</td>
<td>pressurizer</td>
</tr>
</tbody>
</table>

3.2.2 Transient Analysis

Complete Loss of RHR has been considered from all the cases with the additional consideration in Case 6 where 1 inch break at RHR pump discharge from case 5 conditions was considered. For this case, two different transient scenarios for this case have been analyzed: a) trip of both RHR pumps and b) closure of the RHR isolation valves at the accumulator discharge line.

RCS closed: Case 1, Case 2

Case 1 - For the water solid case (Case 1), a rapid pressure rise occurred and the RHR relief valve 1 in both RHR trains opened at the very beginning of the transient, Figure 4. The pressure in the system was maintained between the opening and closing setpoint pressure of the RHR relief valve 1. The boiling in the core has started at time = 12060 sec, followed by a pressure rise and an increased flow through the relief valves.

Case 2 - In the Case 2 much slower pressure increase than in the Case 1 was obtained, Figure 4. Fluid expansion was accommodated by a large quantity of air present in the system. Before the boiling the liquid expanded in the upper head, thus compressing the air blanket. As soon as boiling in the core started the vapour flowed upwards into the upper head and mixed with the air. Following the start of the boiling (at time = 7053 sec) the more rapid pressure rise resulted. At time = 12340 sec the discharge through the RHR relief valve 1 started. The air was expelled from the reactor pressure vessel, while it remained in the pressurizer till the end of simulation.

RCS open: Case 3, Case 4, Case 5, Case 6

In these cases, following the loss of the RHR capability the coolant heated-up and the boiling occurred very fast after start of the transient, Table 5, because of low system pressure.

Case 3 - Following the start of boiling, liquid entrainment into pressurizer surge line and pressurizer took place. The vapor at the top of the SG U tubes condensed and the SG U-tubes were entirely filled with liquid. Discharge through the pressurizer manway began first after the primary pressure reached 2.1E5 Pa because of high initial RCS liquid mass and the great amount of entrained liquid into pressurizer and the pressurizer surge line. In particular, for NPP Krško, the connection junction of the pressurizer surge line with the hot leg (centrally and side oriented) and the orientation of the first part of the surge line (almost horizontal) contribute to clogging of the surge line with the liquid and hamper the discharge of the vapor.
through the pressurizer manway, [13] and [14]. Along with the emptying of the hot legs, liquid entrainment into the pressurizer surge line was stopped. Consequently, the RCS pressure decreased (at app. 4000 sec) and the pressurizer manway flow remained pure vapor.

Case 4 - Because of the fact that prior to the begin of the transient the liquid level was close to the SG manway openings, the increase of the liquid specific volume led to the discharge of the liquid through the SG manways even before the boiling started, Table 5. Unlike the pressurizer surge line which is prone to the clogging because of liquid entrainment, the position and orientation of the SG inlet manways provided a free path for discharge to the environment. The RCS pressure, Figure 5, remained low throughout the simulation and the negligible amount of the RCS inventory was discharged through the pressurizer manway.

Case 5 - Following the start of the boiling a rapid expansion into pressurizer and pressurizer surge line resulted. As soon as enough pressure was built-up, a discharge through the pressurizer manway started, Figure 5. When compared with the Case 3 much less pressure had to be built up to begin the discharge through the pressurizer manway. In the Case 5 the initial RCS mass was much less than in the Case 3 where the SG U tubes were filled with liquid before transient begin. Consequently, in the Case 5, the emptying of the hot legs and the decrease of the liquid entrainment into the pressurizer surge line occurred much earlier than in the Case 3. Thus, in the Case 5 the free path for the vapor relief through the pressurizer manway was obtained soon in the transient.

Case 6 - Following the break occurrence, RCS pressure initially dropped, Figure 6. Similarly to the Case 3 and Case 5, the discharge through the pressurizer manway started first after sufficient pressure has been built-up (at app. time = 1000 sec), Figure 6. In the case 6b a much greater loss through the break than for the case 6a was obtained because RHR pumps were running. Consequently, an earlier emptying of the hot legs as well as the earlier RCS pressure drop in the case 6b than in the case 6a was obtained, Figure 6. At time=1285 sec in the case b the RHR pump in the loop with the break was stopped on a signal: void fraction greater than 0.3. After that point, the break flow for the case 6b decreased. In the case 6a, the high RCS pressure was maintained for about 800 sec longer than for the case 6b. After emptying the pressurizer as well as pressurizer surge line and the subsequent pressure drop at approximately 1700 sec, the flow through the pressurizer manway for the case 6a was equal to that of the case 6b, Figure 6. Finally, almost the same amounts of the discharged masses through the openings as well as the time of the core dry-out for both the case 6a and case 6b were obtained (3570 sec and 3575 sec, respectively), Figure 7, Table 5.

Table 5: Critical parameters for the loss of RHR system for NPP Krško

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Case 1</th>
<th>Case 2</th>
<th>Case 3</th>
<th>Case 4</th>
<th>Case 5</th>
<th>Case 6</th>
</tr>
</thead>
<tbody>
<tr>
<td>Time to boiling (sec)</td>
<td>12060</td>
<td>7053</td>
<td>249</td>
<td>211</td>
<td>499</td>
<td>a:272</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>b:291</td>
</tr>
<tr>
<td>Time to core dry-out (sec)</td>
<td>15300</td>
<td>14820</td>
<td>5800</td>
<td>3615</td>
<td>4000</td>
<td>a:3570</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>b:3575</td>
</tr>
</tbody>
</table>
Figure 4: Fluid temperature at the top of the core and saturation temperature (TSAT)

Figure 5: Hot leg (loop with pressurizer) pressure (Case 3, Case 4, Case 5)

Figure 6: Hot leg (loop with pressurizer) pressure (Case 6 a, Case 6 b)
4 CONCLUSION

The evaluation of the safety margins during shutdown conditions for NPP Krško has been performed using RELAP5/mod3.3 code. The code proved to be sufficiently robust for the calculations.

Main goal of the analysis was to predict time frames for operator actions in case of loss of RHR and to consider capability of the ECCS system to cope with the SBLOCA in cold shutdown conditions. In the calculations where mid-loop operation was considered it was shown that liquid carryover to the RCS openings plays major influence on the depletion of the liquid. Critical parameters for operator actions (time to boiling, time to core dry-out) were determined. Use of the detailed, qualified best-estimate model has provided detailed insight with reasonable effort into the possible plant states during shutdown. Obtained information has been used to optimise plant procedures for shutdown operations.

ACKNOWLEDGMENTS

The authors gratefully acknowledge the contribution and support of NPP Krško for the work described in this paper.

NOMENCLATURE

<table>
<thead>
<tr>
<th>Acronym</th>
<th>Definition</th>
</tr>
</thead>
<tbody>
<tr>
<td>AFW</td>
<td>Auxiliary Feedwater</td>
</tr>
<tr>
<td>ECCS</td>
<td>Emergency Core Cooling System</td>
</tr>
<tr>
<td>FER</td>
<td>Faculty of Electrical Engineering and Computing</td>
</tr>
<tr>
<td>NEK</td>
<td>NPP Krško</td>
</tr>
<tr>
<td>NPP</td>
<td>Nuclear Power Plants</td>
</tr>
<tr>
<td>NSSS</td>
<td>Nuclear Steam Supply System</td>
</tr>
<tr>
<td>PWR</td>
<td>Pressurized Water Reactors</td>
</tr>
<tr>
<td>RCPs</td>
<td>Reactor Coolant Pumps</td>
</tr>
<tr>
<td>RCS</td>
<td>Reactor Coolant System</td>
</tr>
<tr>
<td>RHR</td>
<td>Residual Heat Removal</td>
</tr>
<tr>
<td>RWST</td>
<td>Refuelling Water Storage Tank</td>
</tr>
<tr>
<td>SB LOCA</td>
<td>Small Break Loss of Coolant</td>
</tr>
<tr>
<td>SI</td>
<td>Safety Injection</td>
</tr>
<tr>
<td>SG</td>
<td>Steam Generator</td>
</tr>
</tbody>
</table>
REFERENCES


PROBABILISTIC EVALUATION OF POTENTIAL PTS SCENARIOS IN PWR

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ABSTRACT

In order to determine the frequency of occurrence per reactor year of potential challenging events for vessel integrity (in the case of possible pressure thermal shock), a systematic approach is required that will identify all relevant sequences of multiple failures from all pertinent initial events in a probabilistic safety assessment (PSA) plant model. The event tree (ET) approach is an orderly method for performing this quantification and has been used in the present effort as discussed in the paper. This effort has taken advantage of existing PSA model that were developed for other purposes (Individual Plant Examination). In general, the approach taken is to first identify the set of all initiating transients or events that either by them or along with succeeding failures could lead to potential challenges to vessel integrity. The sequence of accompanying branching chains of events, including component failures and associated probabilities, is logically traced out in the vent trees. The output of the event tree is a set of end states and frequencies. These end states can than be evaluated for potential challenges to the vessel from pressure thermal shock. The sum of the frequencies of the end states that are potential challengers becomes the total frequency per reactor year of vessel integrity challenges summed over all types of initiating events. All frequencies given in full paper are mean values.

1 INTRODUCTION

A PTS category is a group of cool down events caused by a common mechanism such as a Primary System Boundary Rupture. Table 1 shows a list of potential PTS categories that was suggested by the NRC in Reference [5]. Combining this NRC suggested list and the events identified by the following discussion of potential stagnation, a number of potential PTS categories are identified.

Table 1 US NRC Suggested Potential PTS Categories

<table>
<thead>
<tr>
<th>1. Stagnated Loop Flow with extended HPSI operation</th>
</tr>
</thead>
<tbody>
<tr>
<td>a) LOCA's greater than makeup capacities</td>
</tr>
<tr>
<td>b) S/R valves fail open with reduced ECCS</td>
</tr>
<tr>
<td>c) Small LOCA's with no cooling from one SG</td>
</tr>
</tbody>
</table>
d) LOFW with feed and bleed
e) Failure of all RCP seals
f) Steam line break with subsequent SGTR
g) ATWS
h) Steamline break with stuck open PORV/unisolated letdown line/RCP seal failure

2. Steamline Break or Equivalent
   a) Pipe breaks
   b) Steam generator relief valves stuck open
c) Turbine bypass valves stuck open
d) Feedwater line break
e) Steam flow - power mismatch

3. Steam Generator Tube Rupture
   a) Extended HPSI operation after faulted SG is isolated
   b) SGTR with stuck open secondary valve

4. Excess Feedwater Flow
   a) Failure to control MFW following plant trip
   b) Failure to control AFW following plant trip

In their previous PTS evaluation, the NRC has identified that events, including single or multiple equipment failure situations where loop flow is lost, may warrant consideration in a detailed PTS evaluation as potential additional transients contributing to the expected frequency of flaw extension. This concern exists because once loop flow is lost; the RCS cool down will be accelerated due to the lack of mixing between safety injection flow and the warm loop flow. In the case of an accident where all reactor coolant pumps are tripped, natural circulation flow will normally develop in the RCS if at least one steam generator is available to transfer decay heat out of the core. Natural circulation flow through each loop is generated by the thermal head resulting from the density difference between the hot and cold sides of the RCS. Mechanisms which significantly alter the necessary temperature distribution or pressure drops through a loop have the potential to impede flow in one or more loops. Table 2 summarizes the identified PTS transient categories in previous Westinghouse studies.

### Table 2 PTS Transient Categories – WOG Stagnant Loop Evaluation

<table>
<thead>
<tr>
<th>Category (Abbreviations)</th>
<th>Description</th>
<th>Examples</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Secondary Depressurization</td>
<td>Sustained Excessive Steaming Of One Or More Steam Generators (Except Feedline Break)</td>
<td>Steamline Breaks Stuck Open Secondary Valves Reactor Trip Without Turbine Trip Steam Dump Control System Failures</td>
</tr>
<tr>
<td>2. Loss Of Coolant Accident (LOCA)</td>
<td>Sustained Loss of Primary Coolant (Except SGTR and LOHS)</td>
<td>Primary Piping Breaks Stuck Open Primary Valves Control System Failures RCP Seal Failures</td>
</tr>
<tr>
<td>3. Steam generator Tube Rupture (SGTR)</td>
<td>Rupture Of One Or More Steam Generator Tubes</td>
<td>Double Ended Tube Break Steam Generator Tube Leak</td>
</tr>
<tr>
<td>4. Loss of Secondary Heat Sink (LOHS)</td>
<td>Sustained Inability To Cool The Primary System With The Steam Generators</td>
<td>Loss of Offsite Power Loss of All Feedwater</td>
</tr>
</tbody>
</table>
### 5. Excessive Feedwater (EXFW)

<table>
<thead>
<tr>
<th>Description</th>
<th>Examples</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sustained Excessive Feedwater Addition To One or More Steam Generators</td>
<td>Feedwater Control System Failure</td>
</tr>
</tbody>
</table>

### 6. Anticipated Transients Without SCRAM (ATWS)

<table>
<thead>
<tr>
<th>Description</th>
<th>Examples</th>
</tr>
</thead>
<tbody>
<tr>
<td>Failure of The Reactor To Trip Upon Demand</td>
<td>Uncontrolled Rod Withdrawal at Power Uncontrolled Boron Dilution Loss of Normal Feedwater Loss of External Load</td>
</tr>
</tbody>
</table>

### 7. Main Feedline Break (MFB)

<table>
<thead>
<tr>
<th>Description</th>
<th>Examples</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sustained LOSS of Secondary Coolant Through A Feedline Rupture</td>
<td>Feedline Rupture</td>
</tr>
</tbody>
</table>

### 1.1 Important mechanisms

An evaluation [3] for was performed to identify mechanisms or events that may potentially affect flow of the primary coolant during natural circulation. A summary of the potential ways by which natural circulation flow may be stopped, leading to the possibility of relatively stagnant loop conditions as discussed is provided in the following discussion.

#### 1.1.1 Inadequate Core Heat Generation (Low Decay Heat)

At very low decay heat fractions (less than 0.5 percent of full power) the capability to maintain natural circulation flow becomes more difficult. Since decay heat generation is the heat source which provides the necessary thermal head that drives flow to the steam generators, very low decay heat levels will decrease the magnitude of the flow rates through the loops. In addition, the susceptibility of the system to some of the mechanisms described in the following sections will be increased under low decay heat conditions.

#### 1.1.2 Loss of RCS Inventory

A loss-of-coolant accident of a large enough size (greater than approximately 1 to 2 inches in equivalent diameter) may result in the draining of the RCS such that subcooled or two phase natural circulation may be lost. The decay heat generated will then be removed by reflux cooling or pool boiling in the core. In this case continued safety injection operation is necessary to replace RCS mass losses and prevent or minimize the possibility of core uncovery. This sustained injection of SI into a stagnant RCS loop can potentially lead to rapid cooldown of the vessel wall. For approximately breaks smaller than approximately 1-2 inches in diameter natural circulation flow will continue as long as secondary inventory is maintained.

#### 1.1.3 Inadequate Heat Removal (Symmetric)

The failure of all auxiliary feed pumps after a plant trip and the inability to establish an alternate feed source using either main feed or condensate pumps will lead eventually to steam generator dry-out (loss of secondary heat sink accident). At that time the RCS will undergo a heat-up and the inability to transfer a significant amount of heat to the steam left in the secondary will lead to a reduction in natural circulation flow. It should be noted that these conditions by themselves will not create a PTS concern. However, under this situation, the operators are instructed to establish an alternate heat removal method in order to prevent core uncovery. The recommended mode is to use "bleed and feed" where at least two pressurizer...
PORVs are opened and maximum safety injection is initiated. This mode, if initiated in a timely manner, can prevent core uncovery and maintain core cooling for a long period. However, this will result in a fairly severe RCS cooldown in the cold legs where the loop flows are relatively stagnant. The "bleed and feed" action is necessary to prevent core uncovery and possible fuel damage while continuing steps to re-establish a secondary heat sink to complete plant recovery.

1.1.4 Non-Symmetric Heat Removal

Steaming Imbalance

A large steaming imbalance that results in an overcooling of one steam generator in relation to the other steam generators may cause a cessation of natural circulation in the loops that are not overcooled. As the secondary steam temperatures become equal to or greater than the core exit temperature, the flows in the unaffected loops will decrease significantly or stop due to the loss of the thermal driving heads that normally exist due to temperature difference through the loops. There are three basic scenarios that could lead to this situation.

1. A steam break will result in an uncontrolled steam release from one (or more) steam generator and can result in an RCS cooldown large enough that the other steam generators would not be able to remove heat from the primary system. In that case, flow might stop in the intact loops until the faulted steam generator is isolated and heat removal capability is re-established in the intact steam generators. Previous analyses in small steam breaks have shown that they become a PTS concern only for low decay heat conditions.

2. During a steam generator tube rupture event recovery, the ruptured steam generator will be isolated by the closing of its MSIV and the termination of auxiliary feedwater to it. A rapid cooldown using the intact generators is then performed. This will be followed by a depressurization of the RCS, with subsequent termination of safety injection and the equilibration of RCS and ruptured steam generator pressure to stop primary to secondary leakage. The action to cool the RCS down, with the ruptured generator isolated, may result in a loss of natural circulation flow through that loop. However, in order to terminate primary to secondary leakage in an expeditious manner it is necessary that this action be taken. It is recognized that a relatively stagnant loop may result and that a subsequent SI termination under the proper conditions is important not only to minimize leakage, but also to minimize the possibility of thermal shock.

3. Any situation, which results in a significant cooldown in one steam generator versus the others, can lead to loss of natural circulation flow in one loop. For example, excessive feedwater to one steam generator; unavailability of equipment or manual operator action that results in unsymmetrical cooldown of the steam generators may result in one-loop becoming stagnant if the difference in primary to secondary heat transfer is significant. In particular, if the steam temperature in one (or more) steam generator is equal to or greater than the average core outlet temperature, the flow will approach stagnation in that loop.

Feed Imbalance

A feed imbalance itself should not lead to a loss of natural circulation flow through a loop. However, an imbalance which results in the uncovery of the steam generator tubes may cause a loss of flow through that loop. There are three situations in which this may occur.
1. A feedline break will remove the ability to provide feedwater flow to the faulted steam generator and also may result in a significant loss of inventory if the break is not isolable.

2. During recovery from a steam break the auxiliary feedwater is isolated from the faulted steam generator to prevent RCS overcooling. However, if the steam is unisolable, it will eventually lead to dry out of the faulted steam generator. This will result in the loss of adequate heat removal required to generate the necessary thermal head to drive the natural circulation flow through the faulted loop.

3. A significant feed imbalance due to operator action or equipment unavailability that leads to a significant uncovering of the steam generator tubes can lead to loss of natural circulation in one or more loops. The emergency response guidelines specify a minimum non-faulted steam generator level to be just in the narrow range span including uncertainties. Therefore, under normal conditions, this level will assure the tubes are covered and that natural circulation flow capability will be maintained.

1.2 Mechanism To Draw Additional Cold water into Vessel

In addition to the RCS cooldown due to SI injection into a stagnant loop, there are additional mechanisms that can draw more cold water into the vessel downcomer and, hence, will create a more severe temperature cooldown leading to potential PTS concern. The cold water can result either from the buildup of relatively cold water in the cold leg and crossover leg after a period of stagnation, or it may result from a surge of cold SI, or accumulator injection flow. There are basically three mechanisms that will draw the cold water into the vessel downcomer.

1. Starting an RCP in a loop that has been relatively stagnant will result in the moving of the contents of that loop into the downcomer. Starting an RCP in another loop will cause reverse flow to occur in the inactive loop and, therefore, will not draw the water into the vessel. In the former case the impact of the cold water flow into the downcomer will be minimal due to the short duration of this cold flow. The water temperature will rise after the cold fluid in the cold leg and crossover leg volume has been purged and warm fluid from the SG begins to mix with the SI flow.

2. The opening of a pressurizer PORV to depressurize the RCS can result in a surge of cold water into the downcomer, if the PORV is on the loop other than the stagnant loop. If the SI is off, any cold water flow into the downcomer would be of short duration and have minimal impact. If the SI is on, the cold flow would be continuous until the PORVs are closed. However, the opening of PORVs directed by the guidelines is normally necessary only for a short period until depressurization to the necessary level is completed (except in the case of a "bleed and feed" action).

3. With a stagnant loop a subsequent break in the system either in the hot leg or upper head region could result in the drawing of cold SI water into the vessel downcomer. Since under this situation the highest priority from a safety standpoint is to maintain RCS inventory to assure adequate core cooling, a severe RCS cooldown will result.
2 EVENT TREE ANALYSIS (ETA)

The first step in the PTS risk assessment is to identify, using existing event tree analysis, the broad categories of events that could potentially result in a pressurized thermal shock of the reactor vessel. The criteria used to identify the PTS transient categories documented in this paper are:

(1) Any one of the following conditions exist in one or more PTS scenarios within the category:
   − Sustained loss of primary coolant
   − Sustained loss of secondary coolant
   − Sustained injection of SI into a stagnant RCS loop
   − Sustained excessive main or auxiliary feed water flow to one or more steam generators.

AND

(2) The PTS risk associated with the category is greater than 10^{-6}/year. This number was selected based on prior experience documented in [1].

The transient categories that are identified using these criteria for the WOG stagnant loop study [3] are listed in Table 2. The seven categories identified are believed to include all potential stagnant loop transients that can contribute to the overall PTS risk. Combinations of these categories are also considered if they meet the above criteria.

2.1 Frequency of Challenging Transients

The basic approach covers the developing the transient sequences frequencies, which are subsequently evaluated for vessel integrity. To establish a basis for saying that PTS risk is not dominated by scenarios other than small steamline breaks, small LOCAs, and SGTRs, refinements have been added to the original methodology to assess other transient scenarios including that involving loop stagnation.

In order to determine the frequency of occurrence per reactor year (/Rx year) of potential challenging events for vessel integrity, a systematic approach is required that will identify all relevant sequences of multiple failures from all pertinent initiating events. The event tree approach is an orderly method for performing this quantification and has been used in the present effort as discussed below. This effort has taken advantage of existing PRA data bases that were developed for other purposes (Krsko Level 1 PSA, [5]) but also includes some original effort on tree structures.

In general, the approach taken is to first identify the set of all the initiating transients or events that either by them or along with succeeding failures could lead to potential challenges to vessel integrity. The sequence of accompanying branching chains of events, including component failures and associated probabilities, is logical traced out in the event trees.

Output of the event tree is a set of end states and frequencies. These states can then be evaluated for potential challenges to the vessel pressurized thermal shock. The sum of the frequencies of the end states are potential challengers becomes the total frequency per reactor vessel integrity challenges summed over all types of initiating events. The frequencies given in this paper are mean values. The probabilistic methods used in this paper are generically applicable to operating Westinghouse PWRs Probabilities are developed uniformly from generic operating plant history without weighting the more recent years of experience.
The transients chosen as initiating events for the vessel integrity evaluation include those that either directly or through consequential failures may lead to a pressurized thermal shock to the reactor vessel. The events that could lead directly to a challenging cooldown include small LOCAs, excessive feedwater, and steamline rupture. These events could also propagate into compound challenges through consequential failures. An example of this is a consequential stuck open secondary steam relief valve during a small LOCA initiator. The result would be a more severe cooldown than the initiating event due to the secondary depressurization.

The initiating events, which by themselves would not necessarily pose a threat to vessel integrity, are the following: turbine trip, reactor trip, or loss of main feedwater for example, would typically result in a plant trip accompanied by plant stabilization or a normal cooldown assuming that safety and --on--ro' systems function as designed. Should a steam dump valve actuate and fail to re-close then a primary overcooling would result due to the secondary depressurization. The frequency of this event would add to all other sources of secondary depressurizations including the initiating steamline ruptures. The consequential events postulated as the result of the initiators are secondary depressurization, pressurizer PORV LOCAs, and excessive feedwater. The system failure questions asked to determine the probability of these consequential events is described in the event tree discussions [5].

Two types of event trees have been used to evaluate the sequences of events important for pressurized thermal shock. The plant event tree, which asks consequential failure questions for plant systems, yields a pinch point or set of cooldown states. The mitigation event tree, which models automatic or manual termination or alleviation of the cooldown state conditions, yields the next pinch point of end states. The example plant event tree shown in Figure 1 is used to ask specific plant system success or plant response questions. Each initial event has a separate, specific event tree which should be assessed to identify possible PTS sequence and make possible its frequency quantification. At each node the up direction denotes the success or yes response to the question asked at that node and the down direction denotes the failure or no response. The following paragraphs describe the implementation of methodology on small break LOCA event tree. The first node is simply the entry point for the initiating event.

2.2 Small Break LOCA Event Tree Example

The small LOCA event tree (illustrated by Figure 1) applies to breaches in the RCS which are large enough that the break flow exceeds the capacity of the normal reactor makeup system. The break size, however, is not large enough to provide core decay heat removal. This covers breaks from 3/8 to 3/4 inches in equivalent diameter. Smaller size breaks would generally not cause a reactor trip and are not included in this category.

The following symbols are used in the small LOCA event tree to define top events.
- SLO initiating event, small LOCA
- RT1 reactor trip
- S12 high pressure safety injection
- LC2 high pressure recirculation
- AF1 auxiliary feedwater and secondary cooling
- MF1 operator action - establish main feedwater
- ES1 operator action to cooldown and depressurize the RCS per ES-1.2
- OP2 operator action, depressurize the RCS for ACC and S11
OFB  operator bleed and feed
S13  low-head safety injection
LC3  low-head recirculation

2.2.1 Initiators

The small LOCA event tree models all reactor coolant system ruptures inside containment with blowdown rates beyond the capacity of the normal makeup system (3/8 inch equivalent diameter) and less than 3/4 inches in equivalent pipe diameter. Reactor coolant pump seal LOCA and control rod ejection events are also included in this initiating event category. Small breaks may also result from consequential failures following other initiating events. The consequential small LOCA resulting from other events are generally transferred to the small LOCA event tree to account for all core damage due to small LOCAs.

2.2.2 Accident Progression

For the break areas modeled by the small LOCA event tree, the CVCS cannot maintain reactor coolant system (RCS) inventory control. The RCS will depressurize causing an automatic reactor trip and an SI signal to be generated when the low pressurizer pressure setpoints are reached. The accident is mitigated by high pressure safety injection and the subsequent removal of decay heat via the steam generators and auxiliary or main feedwater. Initially, more subcooled liquid volume would be leaking through the break than would be added. The pressurizer would continue to empty and pressure would steadily drop. Saturation conditions could eventually be reached in the hotter regions of the reactor coolant system. Since the break is small and unable by itself to remove all the decay heat, another heat sink must be available for heat removal. Secondary cooling is provided by the steam generators and auxiliary feedwater. If auxiliary feedwater is not available, main feedwater could be recovered for secondary cooling. If this method fails, it is possible to mitigate the event by removing the decay heat through bleed and feed cooling. As the core decay heat rate decreases, reactor coolant pressure and temperature will stabilize. As long as secondary heat removal and safety injection are not terminated, adequate core cooling will be maintained. Eventually, switchover to recirculation will be required if the RWST level reaches the low level alarm. For long term cooling, the operator can then establish high pressure recirculation. If still required, secondary cooling with the steam generators would also be used. Per the emergency procedures, the operators would be directed to follow ES-1.2 "Post-LOCA Cooldown and Depressurization", for a more optimal recovery from a small LOCA. In this procedure, the operator starts a cooldown using the steam generators, regains control of pressurizer level by performing an RCS depressurization (using pressurizer spray or opening a PORV), and continues to depressurize the RCS by sequentially stopping the high pressure SI pumps when specified RCS subcooling criteria are met. When RHR cut-in conditions are achieved, the RHR System can be aligned for service to continue the cooldown to cold shutdown. For small LOCAs it is possible to stop all high pressure safety injection, realign to normal charging, and depressurize to near atmospheric pressure to stop or substantially reduce the break flow. By reducing safety injection flow (by reducing break flow), it is possible to avoid switchover to recirculation during the 24 hours following the accident.

2.2.3 Top Event Descriptions

The top events of the small LOCA event tree are described below to provide an understanding of the system functions and operator actions involved.
SLO - Initiating Event, Small LOCA

A small LOCA could be initiated by a random or consequential failure of the RCS piping and break sizes ranging from 3/8 inch to 3/4 inch equivalent diameter. A RCP Seal LOCA and Control Rod Ejection event would also be a small LOCA.

RT1 - Reactor Trip

Reactor trip signal would be automatically initiated on low pressurizer pressure reactor trip signal; manual actuation would be also available. Failure of the reactor trip function is assumed to lead to an AWS event.

S12 - High Pressure Safety Injection

High pressure safety injection is automatically actuated on a SI signal upon receipt of a low pressurizer pressure signal. The high pressure SI pumps take suction from the RWST and inject borated water into the cold legs. The success criterion is 1 of 2 high pressure pumps injecting through the intact injection line (the injection lines could be the location at which SLO occurs) or 1 of 2 injection lines into the reactor vessel. The actuation of the SI pumps is modeled directly in the ECCS fault tree.

AF1 - Auxiliary Feedwater and Secondary Cooling

Secondary cooling is needed to remove the energy stored in the primary system at the onset of the small break. Since the main feedwater system is isolated by the safety injection signal, secondary cooling is assumed to be provided first using the auxiliary feedwater (AFW) and steam dump systems. Auxiliary feedwater is started automatically by the SI signal or by the steam generator low-low water level signals. Manual actuation is also possible. Auxiliary feedwater flow control is assumed necessary for the success of the system. The decay heat removal using a single steam generator is adequate, so it is assumed that feed flow to one steam generator is required. The success criterion for AFW for the various scenarios of interest is 1 of 3 pumps delivering water to one steam generator. Secondary steam relief via the condenser steam dump valves (if available), the safety valves (five per SG), or the SG PORV (one per SG) is also assumed. The condensate storage tanks or the essential service water would be required to provide water flow for the entire event.

MF1 - Operator Action - Establish Main Feedwater

This top event models the availability of the main feedwater to remove decay heat via the steam generator if auxiliary feedwater is unavailable. As explained above, the success criterion used is the main feedwater delivering sufficient feedwater flow to one SG through the AFW injection line (MFW on bypass). Steam relief paths as described above for AF1 are also necessary for MF1 success. If normal feedwater is needed after feedwater isolation and because AFW fails, it would most likely be restored using FR-H.1 procedure, "Response to Loss of Secondary Heat Sink". MFW availability also includes availability of off-site AC power and operation of a condensate pump. Alternatively the feedwater flow can be delivered using the condensate pump only. In this case the intact SG depressurization to less than about 30 kp/cm² is required to allow the injection solely from the condensate pump without operation of the main feedwater pump. This operation requires a complex set of actions some of which have to be performed locally. For this reason and considering the short time allowed to initiate and complete these actions before SG dry-out, this alternative means of recovery is conservatively not considered in the top event. The condensate pump takes suction from the condenser hotwell which can be fed directly by the Condensate Storage Tanks (CSTs - 2 Tanks). The operator has to control the feedwater flow to SG to:
1. prevent excessive RCS cooldown and avoid SG overfilling, and
2. delay the time at which condenser hotwell and CSTs are emptied (if operator controls feed flow the total condensate inventory would assure 24 hours operation) in case of unavailability of the steam dump to the condenser.

OP2 - Operator Action Depressurize the RCS for Low Pressure Safety Injection

If high pressure injection fails, the plant can still be successfully recovered if the operator performs a depressurization to allow injection from the low pressure SI system (accumulators and RHR pumps). This cooldown and depressurization would be performed per the functional restoration procedure for degraded and inadequate core cooling, FR-C.2 and FR-C.1 respectively. The secondary depressurization can be accomplished by dumping steam to the condenser, if available, or by dumping steam through the SG PORVs to the atmosphere. The conservative limiting success criterion is operation of one of the two SG PORVs on steam generators with auxiliary feedwater. The plant specific analysis shows operator action to initiate the depressurization at 3.6 hours will result in success.

S13 - Low-Head Safety Injection

The low-head safety injection function is provided by the RHR pumps. After the accumulators have injected most of their contents, the RHR pumps begin to inject borated water from the RWST to the RPV. This system is activated by the SI signal (or manually, per the emergency procedures). One of the two RHR pumps with its associated valves and piping is sufficient to meet the requirements for core cooling when provided with sufficient water supply from the RWST. The Class 1E emergency busses are required to provide electrical power to the ECCS pumps and motor-operated valves. The CCW system is required to provide cooling for the RHR pumps. The CCW is also required for RHR pump room coolers to maintain the ambient environment within the design specifications of the pump motors. Essential service water is necessary as supporting system for CCWS. Consistent with the Medium LOCA, the success criterion is one of two low pressure safety injection (RHR) pumps delivering flow to the intact injection line to the RPV (the injection line could be the location at which SLO occurs). As far as the running time is concerned, the exact time is again not important since RHR pump operation would be necessary in the longer term for cold leg recirculation.

OFB - Operator Bleed and Feed

Upon failure of auxiliary feedwater, the operator will establish bleed and feed if high pressure safety injection is successful. This mode of decay heat removal is established per emergency procedure FR-H.1 "Response to Loss of Secondary Heat Sink" if the operator is not able to restore the auxiliary or main feedwater systems to service. With success of S12, one out of two pressurizer PORVs (and their associated block valves) opened, is taken to be adequate for decay heat removal based on the plant specific MAAP analysis [4]. The analysis of also shows that for the transient initiating event, SG dryout is expected to occur 51 minutes after all feedwater to the SG is lost. The analysis additionally shows that bleed and feed initiated by 32 minutes after SG dryout will result in successful recovery (i.e. prevent core damage). The transient case should be bounding for the small LOCA as a primary relief path already exists. Pressurizer PORV operation requires the operator to reset SI and instrument air to containment to supply the Pressurizer PORVs.

The opening of a pressurizer PORV to depressurize the RCS can result in a surge of cold water into the downcomer, potential for PTS, if the PORV is on the loop other than the stagnant loop. If the SI is off, any cold water flow into the downcomer would be of short
duration and have minimal impact. If the SI is on, the cold flow would be continuous until the PORVs are closed. However, the opening of PORVs directed by the guidelines is normally necessary only for a short period until depressurization to the necessary level is completed (except in the case of a "bleed and feed" action).

**LC2 - High Pressure Recirculation**

If the primary system pressure remains high, long term cooling and RCS makeup can be established by pumping sump water via the RHR pumps to the suction of the high pressure SI pumps. This requires operator action to reposition valves to align the high pressure SI pumps to the RHR discharge lines. The success criterion for small LOCA is one of two RHR pumps supplying flow to the suction of one of two high pressure SI pumps, injecting through 1 of 3 available injection lines. In fact, one of the 4 potential high pressure injection lines (2 injection lines into cold legs and 2 injection lines into RPV) could be the location at which the break occurs. Thereby this top event is the same as for MLO. Since high pressure recirculation for small LOCA would not be needed for several hours (about 8 hours as given by MAAP analysis [4] for the most limiting case), it is possible that the operator would have stopped the RHR pumps during the injection phase per the emergency procedures E-1 "Loss of Reactor or Secondary Coolant" Step 13. The operator would do this if RCS pressure is stable and greater than the shutoff head pressure of the RHR pumps. Therefore, the fault tree should consider the possibility that the RHR pump has to be restarted for high pressure recirculation. Component cooling and essential service water support conditions are also necessary for recirculation. In addition to pump and pump room cooling, component cooling water and essential service water are needed to cool the RHR heat exchangers. The essential service water is a support system of the component cooling water system. The valves which must operate during the switchover to high pressure recirculation are the same as those needed for low-head recirculation (LC3) plus those needed to align the suction of the high pressure SI pumps to the discharge of the RHR pumps and to close the suction from RWST.

**LC3 - Low-Head Recirculation**

If the operator has depressurized the system and SI8 has been successful, switchover from the injection mode to the low-head recirculation mode must be accomplished. Low-head recirculation provides the necessary makeup to the RCS and removes decay heat from the core and sensible heat from the containment sump water. The low head recirculation function is provided by the RHR system. An adequate volume of water must be delivered during the injection phase to assure that sufficient water is available within the containment to meet the NPSH requirements of the RHR pumps during the recirculation mode. This required amount of water determines the low level alarm for the RWST.

**ES1 - Operator Action to Cooldown and Depressurize the RCS per ES-1.2**

In the emergency procedure ES-1.2 “Post LOCA Cooldown and Depressurization ” the operator initiates a cooldown at a rate no greater than 55°C/hr using the available intact SGs. Once RHR entry conditions are established, the RHR system can be placed in service to continue the cooldown to cold shutdown. As the cooldown progresses, the high pressure SI pumps would sequentially be stopped (based on specified subcooling and RCS inventory criteria) until the last charging SI pump is realigned to normal charging. The RCS would also be depressurized (using pressurizer spray or a PORV) to increase inventory and to minimize the break flow. Using ES-1.2, it may be possible for small break cases to depressurize the RCS to near atmospheric pressure and thereby terminate or substantially reduce the break flow. By so doing, the charging flow can be reduced and switchover can be avoided. Although the ES-1.2 actions appear complex, the operator would have a long period of time...
to perform these actions. With the ES-1.2 actions, for break sizes 0.75 inch diameter or smaller, the operator will be able to avoid switchover for at least the 24 hour time frame assumed for the event tree and fault tree modeling. This is confirmed by the plant specific MAAP analysis. Success would also require steam dump from at least one steam generator (i.e., steam dump to condenser, if available, or operation of a SG PORV). The active SG(s) used for the cooldown would also need a supply of auxiliary feedwater (AF1 success) until the RHR system can be aligned for service. Another function recommended to ensure ES-1.2 success would be a means for RCS depressurization - the operation of pressurizer spray or one PORV (or auxiliary spray) could accomplish this. Normal spray requires operation of an RCP in one of the two loops with a spray line connection. In order to achieve cold shutdown conditions, it should be assumed that operation of at least one train of RHR is required. Success for one train of RHR would be similar to LC1 except that the valves to the hot leg (instead of the sump) would need to be opened. Finally the operation of 1 out of 2 CVCS charging pumps should be assumed since this allows the SI pumps to be stopped at acceptable subcooled values and allows subsequent makeup control. Charging pumps are also required for auxiliary spray success.

2.2.4 Example PTS sequence identification

For subjected illustration, just the potential sequence for PTS was assessed. This sequence considers a small LOCA with no secondary system cooling from steamgenerators because of a lack of secondary system makeup (failure of AFW and unsuccessful attempt to recovery MFW).

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<th>RT1</th>
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<th>AF1</th>
<th>MF1</th>
<th>OP2</th>
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Figure 1 Small LOCA Event Tree Assessment
Identified PTS sequence driven by small break LOCA initiator is illustrated by red line on Figure 1.

Sequence is described as (SLO)(Success-RT1)(Sucess-SI2)(Failure_AF1)(Failure-MF1). Total frequency of such sequence is $1 \times 10^{-08}$ /yr taking into account assumptions given in Table 3

<table>
<thead>
<tr>
<th>Event</th>
<th>Frequency / Probability</th>
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<tr>
<td>Small LOCA initiator (IEV-SLO)</td>
<td>$1 \times 10^{-03}$ /yr</td>
</tr>
</tbody>
</table>
| Auxiliary Feedwater (AFW) failure (AF1) | $1 \times 10^{-04}$  
  (Both AFW MDP failure prob. = $1 \times 10^{-03}$  
  AFW TDP failure prob. = $1 \times 10^{-01}$) |
| Main Feedwater (MFW) not established following AFW failure | $1 \times 10^{-01}$  
  Note: dominated by operator error probability, related to establishing MFW |

**Sequence Frequency**

$$f = 1 \times 10^{-03} \times 1 \times 10^{-04} \times 1 \times 10^{-01} = 1 \times 10^{-08} \text{ /yr}$$

3 CONCLUSION

Paper discuss the methodology for identification of potentially PTS sequences, development of their plant specific catalog and frequency quantification through the assessment of existing event trees what is a part of more complex study shown on Figure 2 [6]. The further work will be performed taking into account described methodology to review all existing event trees and identify all possible PTS potential accident sequences with associated frequencies.

![Figure 2 Proposed Methodology Flow Chart for PTS Evaluation](image-url)
REFERENCES


[4] Krško Level 2 PSA, Section L2-2.0, „Level 1/Level 2 Integration“, 1993


ABSTRACT

Increased usage of Probabilistic Safety Assessment (PSA) models for various practical applications in the nuclear industry makes quantification precision even more important. Common approach to the PSA model quantification is performed by means of using only most important minimal cutsets. This is done because unbearable resources related difficulties to find complete minimal cutset solution and to properly perform various other results (i.e., important measures, configuration variations for on-line risk monitoring) in acceptable time.

PSA model presented with event trees and fault trees can be easily represented with equivalent fault tree model. The fault trees can be very economically encoded by means of Binary Decision Diagrams (BDDs). It is also known that the probabilities of the top event can be computed from the BDD encoding the initial fault tree by very simple and efficient recursive algorithm once that BDD is constructed.

This paper is discussing various quantifications for the PSA model represented by BDD in comparison with conventional approach. BDD quantification produces exact solution comparing to the approximate conventional approach. First, it will be discussed about quantification of PSA model with BDD. Second, it will be investigated what is the difference in the quantified top event results and basic event importance measures between conventional partial quantification approach and exact BDD approach.

Results from practical examples of complete PSA model solution quantification are presented in order to illustrate potential importance. These results demonstrate error regarding conventional partial PSA results quantification. Current results are showing difference which might be regarded not crucial, but that is something dependent on the application. Further investigation is needed to illustrate more completely, and define criteria by which result error should be considered important.

1 INTRODUCTION

With advances of various risk informed applications of the Probabilistic Safety (PSA) results (i.e., risk monitoring [9]), there is a increased need to use PSA results much more frequently and for lot of different applications and model configurations. Today most of the PSA tools are capable only to partially evaluate minimal cut sets (MCS) during the initial regular or special model solving and quantification. This means that only most important set of basic events groups are selected for all quantifications. These MCS are certainly sufficient for the qualitative assessments. Their sufficiency for the quantitative results depends on the
number of factors which are not known. These cut sets provide a good approximation of the system under the study only with initial probability assignment of basic events. Additional care must be taken if these MCS were used for various additional model configuration quantifications or even for basic events quantification. Different approach to the PSA model analysis is possible if initial fault tree logic is transformed into the binary decision diagram since they allow exact solution of the model for any parameter changes and variation. Binary decision diagrams (BDD) than can be a valuable in case of repeated evaluation of different configurations and to accommodate task like risk monitoring application. BDD can be used to efficiently represent ALL minimal cut sets of the system under the study.

Since the major focus of this paper is to compare two different approaches of the PSA model quantification there is no space for further details about these two approaches. BDD’s are based on Shannon’s decomposition of logical function on selected variable ordering scheme. More details about BDD could be found in the [1 and 2]. Most important issue of the variable ordering is addressed for example in the [3, 4, 5, 6, and 7].

1.1 PSA model quantification

Conventional approach to the PSA model quantification in all practical situations is burdened with approximate result even without good error estimation. There is effort to estimate the importance of this approximation and produce better error estimate. In [10] it was illustrated how important might be truncation value on the meaning of the PSA results for the risk-informed applications, and it was also suggested how to better decide about truncation level. Further effort to advance approximate PSA quantification is presented in the [12] with Monte-Carlo error determination. Even if these are real advances it still remain difficult with this approach to address error made during the importance measures quantification. Significant effort to compare the importance of approximate quantification on the top results and importance measures is provided in [11] where was pointed that there is huge importance of complete model quantification. Since model used in that exercise is not available it was hard to speculate about the major reasons for such results.

This paper is presenting similar comparison with much smaller scope. Analysis was focused on one significant event tree for the Level 1 PSA of the nuclear power plant.

2 APPROACH TO COMPARISON

In order to compare results obtained by BDD and conventional quantification we need two different tools applied to the same models. Our approach was to compare results obtained with BDD tool we developed to the conventional results obtained with Risk Spectrum tool. The selected test cases mainly belong to two categories, the first being representative of fault tree models for at the system level, and the second for whole PSA sequence and event tree. Second type of model is consisted of several system level fault trees and therefore always has more complex structure with higher number of basic events. Selected models are coming from the real PSA model for the nuclear power plant. For this
phase of the investigation original basic events probabilities are used in order to test relevance for the practical situations.

Tests were performed as follows; first we built the BDD representation for the set of ALL minimal cut sets of tested fault tree model representation. After the BDD was build we performed basic event importance measures. The generated test cases were also analysed by means of Risk Spectrum tool where all quantifications are built in functionality. We checked the results from the Risk Spectrum tool with different cut off values, but finally cut-off value of 1E-12 was selected for all cases as satisfactory for the influence on final results. Obtained results with these two different approaches were then compared first at the top level and then at the most important basic event importance measures level. Fussell-Vesely (FV) and Risk increase factor (RIF) were selected as importance measures for comparison. More about these and other importance measures could be found in the [8]. Shortly we emphasise that FV is related to other importance measures (i.e., Risk decrease factor) and that for each basic event represents the ratio between the MCS which contains that basic event and complete results from all MCS. RIF importance measure is defined for each basic event as the ratio of all MCS with basic event probability set to 1 and all MCS. It is not further elaborated that importance measures could be calculated for group of basic events, and that emphasises even more potential importance of more precise quantification.

3 TEST RESULTS

From the numerous test results here we present top level results for the three test cases, and part of the results for the basic events for two test cases. Allowed space is not permitting presentation of more results. However, even limited selected results are representative for the work performed and good enough to illustrate most important conclusions.

Top level results for one system fault tree model, one event tree sequence and total result for all event tree sequences are presented in the Table 1. Table is listing models number of gates and basic events. These are not small models but they are not as big as could be found in large nuclear power plant PSA models. Besides final top level probability Table is also showing some details which could emphasise differences between BDD and conventional approach related to the number of evaluated minimal cut-sets.

Table 1: Top level results comparison for one FT, sequence and ET (all CD sequences)

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<th>SEQ5</th>
<th>ET_PSL</th>
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<td>BDD</td>
<td>RS</td>
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Selected detail results for the basic events importance measures are presented for two models in two separate tables. Table 2 is presenting Fussell-Vesely (FV) and Risk increase factor (RIF) importance measures for the most important basic events in the fault tree model FT9FA. In order to minimize presentation only most important basic events are presented based on the FV importance measure. RIF importance measures are also listed for completeness. Table is listing basic event importance measures results and rank for the both Risk Spectrum and BDD quantification. Table 3 is presenting similar results for whole event tree (ET_PSL). In order to better use presented data here listed most important basic events are ranked based on the RIF importance measure.

Table 2: Importance measures results comparison for the fault tree FT9FA

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### Table 3: Importance measures results comparison for the event tree ET_PSL

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</table>

### 3.1 Results Analysis

Results are analysed first at the top level and then related to the basic events importance measures quantified with two different approaches: conventional approximate, and BDD complete.

From the top level result it is visible that there is huge difference in the number of minimal cut-sets used for the quantification. This is driven mainly by the model complexity.
But it is visible that bigger models are having even bigger difference. For the presented FT there is almost two orders of magnitude difference. Analysed sequence model and complete event tree model are showing tremendous difference of about six orders of magnitude in the number of complete and approximate number of minimal cut-sets. Here we compare MCS used for the quantification since number of saved MCS in the conventional approach is much smaller.

Fortunately this extreme difference in the number of evaluated MCS is not reflecting that much to the top level results. This is not to say that observed difference is insignificant. Exact result obtained by the BDD is always smaller than one obtained with approximate approach. The difference is for the presented FT less than 10%, for event tree about 30% and for sequence about 100%. This is certainly significant difference.

Analysis of the basic events importance measures quantitative results difference is somehow more complicated and less conclusive. It is evident that difference exists. However, this is in general much smaller then at the top level, and it does not reflect greatly on the order of the basic events importance. In the Table 2, for the FT, only two basic events (10th and 11th) have significant difference in FV importance measure value. Even that is not resulting in the huge difference on these basic events rank. Difference for the RIF values are even smaller.

Similar results are obtained at the sequence and event tree level. Table 3 is illustrating this more focused on the most important basic events based on the RIF importance measure. Somehow this is more reflected on the ranking difference, and this might need further investigation related to the potential importance for some practical application.

4 CONCLUSION

With the increasing application of the PSA models for the various practical applications of risk based prioritization it is clear that importance of confidence in the model and results becomes more important. Most widely used approximate fault analysis techniques have proved themselves as a valuable tool for such evaluations. Qualitative results provided with the list of the most important minimal cut-sets are certainly satisfactory for all practical purposes. Numerical results suffer limitation that they are not providing reliable measure of the error. The proven potential of BDD representation and complete exact solution of such functions gives us strong evidence that it can be used in evaluation procedures. Ideally one would always prefer complete solution if it can be created. This is major limitation when BDD approach is applied to the huge real size PSA models. There is work in progress to overcome this problem.

Here presented comparison of the results obtained with the conventional approximate and complete BDD approach are showing significant difference at the top level, and somehow less significant difference at the basic events importance level. These are only preliminary and not complete results. It will be important to investigate what is making this difference bigger, and see for example event trees for the various parts of the PSA mode (i.e., seismic PSA, etc.).
REFERENCES


Slovenian Power System Reliability Analysis
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ABSTRACT

A new method for power system reliability analysis using fault tree analysis approach is presented. The method is developed combining the linear network flow method with the fault tree analysis features. The fault trees are constructed for all load points of the power system. The unreliability of the power delivery to the each load in the power system and the weighted system unreliability are obtained based on the quantitative evaluation of the fault trees. The obtained importance measures enable identification of the most important elements of the power system. The algorithm of the computer code, which facilitates the application of the method, has been applied to the Slovenian power system. The implication of the changes of the power system configuration on the system reliability was tested and the obtained results are presented. The obtained results show that introduction of the new line between substation NPP Krško and substation Beričevo significantly improves the reliability of the Slovenian power system and the reliability of the power delivery to the house load of the NPP Krško. The results confirm the applicability of the developed method for the estimation and improvement of the power system reliability and safety of the nuclear power plants.

1 INTRODUCTION

The need for analysis of power system reliability emerges as a result of the recently reported blackout events [1], [2]. Most of the approaches for determination of the power system reliability use approximation or simplification of the problem in order to degrade the problem on the solvable level [3], [4], [5], [6], [7]. The obtained results and corresponding measures don’t provide assessment of the power system reliability and identification of the main contributors.

The nuclear power plant (NPP) safety and reliability depends on the network reliability and vice versa. The failure of the power system results in loss of offsite power (LOOP) initiating event, which is important contributor to the overall core damage frequency (CDF) of the corresponding NPP. The disconnection of the NPP from the power system results in the deficit of generation directly affecting the reliability and stability of the power system.

The objective of the developed method is to estimate the unreliability of power delivery to the loads in the system including the house loads of the NPP using the fault tree analysis approach [8]. The configuration and the state of the power system in the substation, where the load is connected to, are considered. The main contributors to the unreliability of the power
delivery to the specific loads are identified with the introduction of importance measures and the overall system reliability is estimated.

2 METHOD DESCRIPTION

The unreliability of the power delivery to the i-th load ($Q_{Gi}$) is calculated as the top event probability of the respective fault tree, and the values of weighted unreliabilities of power delivery to loads are considered to get the weighted system unreliability:

\[ Q_{PS} = \sum_{i=1}^{NL} \frac{Q_{Gi} K_i}{K} \]

Where:
- $Q_{PS}$ - Weighted system unreliability.
- $Q_{Gi}$ - Unreliability of the power delivery to i-th load (top event probability of the respective fault tree).
- NL - Number of loads in system.
- $K_i$ - Size of i-th load (MW).
- K - Total load in the system (MW).

The fault tree analysis is performed for all loads in the power system.

2.1 Fault Tree Analysis

The classic fault tree is mathematically represented by a set of Boolean equations. The qualitative analysis (in the process of Boolean reduction of a set of equations) identifies the minimal cut sets, which are combinations of the smallest number of basic events which, if occur simultaneously, lead to the top event.

The quantitative analysis represents a calculation of the top event probability, equal to the unreliability of the power delivery to the corresponding load. The calculation of the top event probability:

\[ Q_{GD} = \sum_{\text{cut}} Q_{\text{MCSi}} - \sum Q_{\text{MCSi} \cap \text{MCSj}} + \sum Q_{\text{MCSi} \cap \text{MCSj} \cap \text{MCSk}} - \ldots + (-1)^{n-1} Q_{\bigcap_{i=1}^{n} \text{MCSi}} \]

Can be simplified and approximated (using rare event approximation) as:
\[ n \]
\[ Q_{GD} = \sum_{i=1}^{n} Q_{MCSI} \]  
(4)

Where:
- \( Q_{GD} \) - Top event probability of the fault tree.

Probability of each minimal cut set is calculated using the relation of simultaneous occurrence of independent events:

\[ Q_{MCSI} = \prod_{j=1}^{m} Q_{Bj} \]  
(5)

Where:
- \( Q_{MCSI} \) - Probability of minimal cut set i.
- \( m \) - Number of basic events in minimal cut set i.
- \( Q_{Bj} \) - Probability of the basic event \( B_j \) corresponding to the unreliability of the component \( B_j \).

The fault tree analysis results include importance measures Risk Achievement Worth (RAW) and Risk Reduction Worth (RRW) in addition to the top event probability. Risk achievement worth identifies components, which should be maintained well in order that the reliability of the system is not reduced significantly. Risk reduction worth identifies components, which are candidates for redundancy, because their reliability is worth to increase in order that the system reliability is significantly increased (i.e. risk is reduced).

\[ RAW_k = \frac{Q_{GD}(Q_k = 1)}{Q_{GD}} \]  
(6)

\[ RRW_k = \frac{Q_{GD}}{Q_{GD}(Q_k = 0)} \]  
(7)

Where:
- \( RAW_k \) – Risk Achievement Worth for component k.
- \( RRW_k \) – Risk Reduction Worth for component k.
- \( Q_{GD}(Q_k = 1) \) – Top event prob. when failure probability of the component \( k \) is set to 1.
- \( Q_{GD}(Q_k = 0) \) – Top event prob. when failure probability of the component \( k \) is set to 0.
- \( Q_{GD} \) – Top event probability.

### 2.2 New Importance Measures

The network importance risk measures: Network Risk Achievement Worth (NRAW) and Network Risk Reduction Worth (NRRW) are defined using the definition of the importance measures from single fault tree and the power system unreliability expression given in Equation 1. As the term network is a descriptive term for the power system in this paper, NRAW and NRRW can be expressed as power system risk achievement worth and power system risk reduction worth.
\[ NRAW^k = \frac{Q_{PS}(Q_k = 1)}{Q_{PS}} = \frac{\sum_{i=1}^{NL} Q_{GD_i}(Q_k = 1) K_i}{\sum_{i=1}^{NL} Q_{GD_i} K_i} = \frac{\sum_{i=1}^{NL} Q_{GD_i}(Q_k) K_i \cdot RAW^k_{GD_i}}{\sum_{i=1}^{NL} Q_{GD_i} K_i} \]

\[ NRRW^k = \frac{Q_{PS}(Q_k = 0)}{Q_{PS}} = \frac{\sum_{i=1}^{NL} Q_{GD_i} K_i}{\sum_{i=1}^{NL} Q_{GD_i} K_i} = \frac{\sum_{i=1}^{NL} Q_{GD_i}(Q_k) K_i \cdot RRW^k_{GD_i}}{\sum_{i=1}^{NL} Q_{GD_i} K_i} \]

Where:

- \( NRAW^k \) – Network risk achievement worth of element \( k \).
- \( NRRW^k \) – Network risk reduction worth of element \( k \).
- \( Q_{PS} \) – Weighted system unreliability.
- \( Q_{PS}(Q_k = 1) \) – Weighted system unreliability when unreliability of element \( k \) is set to 1.
- \( Q_{PS}(Q_k = 0) \) – Weighted system unreliability when unreliability of element \( k \) is set to 0.
- \( Q_{GD_i}(Q_k = 1) \) – Unreliability of the power delivery to the \( i \)-th load when unreliability of element \( k \) is set to 1.
- \( Q_{GD_i}(Q_k = 0) \) – Unreliability of the power delivery to the \( i \)-th load when unreliability of element \( k \) is set to 0.
- \( NL \) – Number of loads in the system.
- \( K_i \) – Size of the \( i \)-th load (MW).
- \( RAW^k_{GD_i} \) – Value of RAW for element \( k \) corresponding to the fault tree built for load \( i \).
- \( RRW^k_{GD_i} \) – Value of RRW for element \( k \) corresponding to the fault tree built for load \( i \).
- \( Q_{GD_i} \) – Unreliability of the power delivery to the \( i \)-th load.

Element groups may contain elements (components) of same type, elements corresponding to specific substation or any other combination.

### 2.3 Fault tree construction procedure

The first step in developing the corresponding fault trees for power delivery to the loads is the identification of all the possible energy delivery flow paths from adjacency matrix of the corresponding power system.

The identified flow paths of energy delivery to the loads are tested for consistency, namely:

1. Only a part of the flow path ending with substation, which is directly connected to generators with total installed capacity equal or larger than load, is taken for overload test.
2. The flow path is discarded if there is overloaded line in it.
3. If there is a substation with disrupted voltages in the flow path obtained from previous test, then that flow path is discarded.

Test of overloaded lines in a flow path and voltages in the substations is performed using the approximate direct current (DC) model.

Flow paths, which were accepted in previous test of consistency, are used in next step for fault tree construction. The fault tree for each substation, which is connected to a load, is created using the modular fault tree, shown on Figure 1 with the structure and the failure probabilities inserted depending on the elements modelled. Basic events (BE) marked in red squares are optional, depending if there are common cause failures (CCF) between lines or there are multiple generators in the substation. The lines for which CCF are considered include: CCF of lines due to the common tower and CCF for lines, which are on a common right-of-way for part of their length.

Figure 1: Modular fault tree used for fault tree construction

Fault trees are analyzed by computer code for fault tree analysis with a bottom up algorithm. Minimal cut sets (MCS), which satisfy predefined cut-off, are identified. Quantitative analysis is performed according to the equations given in section 2.1 and section 2.2.

Detail description of the procedure is given in the corresponding references [1], [10], [11], [12] and [13]. The developed method was tested and verified on the IEEE 1996 Reliability Test System [14].

3 SLOVENIAN POWER SYSTEM RELIABILITY ANALYSIS

The configuration of the Slovenian system used in the analysis is given on Figure 2 and consists of: 19 substations, 13 of them directly connected to loads, 8 substations directly connected to generators and 25 interconnections (10 transformers and 15 lines). The common cause failures (CCF) are considered for 12 interconnections. The simplified configuration of the Slovenian power system was constructed on the basis of the corresponding reference [15].
Only the 220 kV and 400 kV lines of the Slovenian power system were considered, including the two 110 kV lines connected to power plants in substations Šoštanj and Brestanica. The 110 kV line to substation Šoštanj was included in the analysis due to the thermal power plant Šoštanj blocks 1-3 connected into it. The 110 kV connection to substation Brestanica was included in the model because the power plant Brestanica can be connected directly to the nuclear power plant Krško (island mode of operation) as alternative offsite power source. The hydro power plants were considered as generators added in the substations Maribor and Podlog. Interconnections with the neighboring power systems of Austria, Italy and Croatia weren’t included in the analysis. The power flows through interconnections with neighboring systems were considered in the loads of corresponding substations, where those lines are connected. The inclusion of the neighboring power systems will improve the overall model and it will require detailed input data for all parameters of those power systems. The size of the loads and generators are given in Table 1.

Table 1: The size of the loads and generators of the Slovenian power system

<table>
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<tr>
<th>Substation number</th>
<th>Substation name</th>
<th>Load MW</th>
<th>Load MVar</th>
<th>Generator MW</th>
<th>Generator MVar</th>
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<td>1</td>
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<td>30</td>
<td>0</td>
<td>600</td>
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<td>RTP Krško</td>
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Figure 2: Basic configuration of Slovenian System
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<th>Load MVar</th>
<th>Generator MW</th>
<th>Generator MVar</th>
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<td>53</td>
<td>0</td>
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<td>Brestanica</td>
<td>70</td>
<td>0</td>
<td>100</td>
<td>197</td>
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</table>

The unreliabilities of the interconnections in the Slovenian power system are given in Table 2.

**Table 2: Slovenian power system lines reliability parameters**

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<tr>
<th>Inter. No.</th>
<th>From bus</th>
<th>To bus</th>
<th>Unreliability</th>
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<td>NPP Krško</td>
<td>RTP Krško</td>
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</tr>
<tr>
<td>2</td>
<td>NPP Krško</td>
<td>Maribor</td>
<td>1.05E-02</td>
</tr>
<tr>
<td>3</td>
<td>Maribor</td>
<td>Podlog</td>
<td>5.28E-03</td>
</tr>
<tr>
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<td>Podlog</td>
<td>Šoštanj G5</td>
<td>4.23E-03</td>
</tr>
<tr>
<td>5</td>
<td>Podlog</td>
<td>Beričevo</td>
<td>4.23E-03</td>
</tr>
<tr>
<td>6</td>
<td>Okroglo</td>
<td>Beričevo</td>
<td>8.43E-03</td>
</tr>
<tr>
<td>7</td>
<td>Okroglo</td>
<td>Beričevo</td>
<td>8.43E-03</td>
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<tr>
<td>8</td>
<td>Beričevo</td>
<td>Divača</td>
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<td>Beričevo</td>
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</tr>
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<td>Beričevo</td>
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<td>Podlog 3</td>
<td>Soštanj</td>
<td>2.38E-03</td>
</tr>
<tr>
<td>18</td>
<td>RTP Krško</td>
<td>Brestanica</td>
<td>2.38E-03</td>
</tr>
<tr>
<td>19</td>
<td>RTP Krško</td>
<td>Brestanica</td>
<td>2.38E-03</td>
</tr>
<tr>
<td>20</td>
<td>Beričevo2</td>
<td>Beričevo3</td>
<td>1.06E-03</td>
</tr>
<tr>
<td>21</td>
<td>Beričevo2</td>
<td>Beričevo3</td>
<td>1.06E-03</td>
</tr>
<tr>
<td>22</td>
<td>Podlog 2</td>
<td>Podlog 3</td>
<td>1.06E-03</td>
</tr>
<tr>
<td>23</td>
<td>Podlog 2</td>
<td>Podlog 3</td>
<td>1.06E-03</td>
</tr>
<tr>
<td>24</td>
<td>Kleče2</td>
<td>Kleče</td>
<td>1.06E-03</td>
</tr>
<tr>
<td>25</td>
<td>Kleče2</td>
<td>Kleče</td>
<td>1.06E-03</td>
</tr>
</tbody>
</table>
The substations unreliabilities are calculated from the actual configuration of the substations in the Slovenian power system. Figure 3 shows the configuration of the substation NPP Krško. The elements (disconnect switches and circuit breakers), which are open during normal mode of operation, are marked in blue colour.

![Diagram of substation NPP Krško](image)

Figure 3: Configuration of the substation NPP Krško

The obtained results for the unreliability of the power delivery to the loads in the power system and weighted system unreliability are given in Table 3.

<table>
<thead>
<tr>
<th>Load substation</th>
<th>Substation name</th>
<th>The unreliability of the power delivery to the load</th>
<th>Weighting factor</th>
<th>Weighted unreliability of the power delivery</th>
<th>Load capacity [MW]</th>
</tr>
</thead>
<tbody>
<tr>
<td>101</td>
<td>NPP Krško</td>
<td>1.55E-04</td>
<td>2.22E-02</td>
<td>3.44E-06</td>
<td>30</td>
</tr>
<tr>
<td>102</td>
<td>RTP Krško</td>
<td>5.79E-02</td>
<td>1.88E-01</td>
<td>1.09E-02</td>
<td>254</td>
</tr>
<tr>
<td>103</td>
<td>Maribor</td>
<td>1.48E-03</td>
<td>1.03E-01</td>
<td>1.52E-04</td>
<td>139</td>
</tr>
<tr>
<td>108</td>
<td>Podlog3</td>
<td>1.34E-03</td>
<td>7.39E-02</td>
<td>9.90E-05</td>
<td>100</td>
</tr>
<tr>
<td>110</td>
<td>Cirkovce</td>
<td>3.16E-03</td>
<td>6.95E-02</td>
<td>2.20E-04</td>
<td>94</td>
</tr>
<tr>
<td>111</td>
<td>Beričevo</td>
<td>1.24E-03</td>
<td>8.50E-02</td>
<td>1.05E-04</td>
<td>115</td>
</tr>
<tr>
<td>112</td>
<td>Beričevo2</td>
<td>3.83E-05</td>
<td>5.47E-02</td>
<td>2.10E-06</td>
<td>74</td>
</tr>
<tr>
<td>113</td>
<td>Beričevo3</td>
<td>1.35E-03</td>
<td>5.91E-02</td>
<td>7.98E-05</td>
<td>80</td>
</tr>
<tr>
<td>114</td>
<td>Kleče 2</td>
<td>3.49E-03</td>
<td>8.35E-02</td>
<td>2.91E-04</td>
<td>113</td>
</tr>
<tr>
<td>116</td>
<td>Divača</td>
<td>8.62E-03</td>
<td>5.69E-02</td>
<td>4.90E-04</td>
<td>77</td>
</tr>
<tr>
<td>117</td>
<td>Divača 2</td>
<td>4.89E-03</td>
<td>3.55E-02</td>
<td>1.74E-04</td>
<td>48</td>
</tr>
<tr>
<td>118</td>
<td>Okroglo</td>
<td>9.74E-03</td>
<td>1.18E-01</td>
<td>1.15E-03</td>
<td>159</td>
</tr>
<tr>
<td>119</td>
<td>Brestanica</td>
<td>6.51E-03</td>
<td>5.17E-02</td>
<td>3.37E-04</td>
<td>70</td>
</tr>
</tbody>
</table>

Weighted system unreliability: 1.40E-02

The results in Table 3 show that the largest value of the unreliability of the power delivery is obtained for the loads in the substations RTP Krško, Okroglo and Divača. This result is due to the size of the loads (RTP Krško, Okroglo) or weak interconnection with the
power system (load in the substation Divača). The same ordering is obtained for the weighted unreliability of the power delivery to the loads.

The obtained results for the network risk reduction importance measure (NRRW) are given in Table 4.

Table 4: Basic events with highest NRRW

<table>
<thead>
<tr>
<th>BE ID</th>
<th>BE description</th>
<th>NRRW</th>
</tr>
</thead>
<tbody>
<tr>
<td>G2 101-1</td>
<td>Generator 101-1 failure</td>
<td>4.64E+00</td>
</tr>
<tr>
<td>L2-111 118</td>
<td>CCF line 111-118</td>
<td>1.08E+00</td>
</tr>
<tr>
<td>L1-101 102</td>
<td>Trans. failure 101-102</td>
<td>1.06E+00</td>
</tr>
<tr>
<td>G2 106-1</td>
<td>Generator 106-1 failure</td>
<td>1.06E+00</td>
</tr>
<tr>
<td>G2 107-1</td>
<td>Generator 107-1 failure</td>
<td>1.06E+00</td>
</tr>
<tr>
<td>L1-111 116</td>
<td>Line failure 111-116</td>
<td>1.03E+00</td>
</tr>
<tr>
<td>L1-112 115</td>
<td>Line failure 112-115</td>
<td>1.02E+00</td>
</tr>
<tr>
<td>L1-105 110</td>
<td>Line failure 105-110</td>
<td>1.01E+00</td>
</tr>
<tr>
<td>L2-102 119</td>
<td>CCF line 102-119</td>
<td>1.01E+00</td>
</tr>
<tr>
<td>L1-115 117</td>
<td>Line failure 115-117</td>
<td>1.01E+00</td>
</tr>
</tbody>
</table>

The identified elements of the Slovenian power system with the largest NRRW are: generator in NPP Krško (BE “G2 101-1”), the CCF of lines between substations Beričevo and Okroglo (BE “L2-111 118”), transformer between substations NPP Krško and RTP Krško (BE “L1-101 102”) and generators in substations Šoštanj4 and Šoštanj5 (BE “G2 106-1”, “G2 107-1”). The obtained results are expected considering the size of the generating units (NPP Krško), importance of the interconnection (lines between substations Beričevo and Okroglo) and power flows between substations (transformer between substations NPP Krško and RTP Krško). The identified elements are candidates for redundancy, because their reliability is worth to increase in order that the power system reliability is significantly increased.

The obtained results for the network risk achievement importance measure (NRAW) are given in Table 5.

Table 5: Basic events with highest NRAW

<table>
<thead>
<tr>
<th>BE identification</th>
<th>BE description</th>
<th>NRAW</th>
</tr>
</thead>
<tbody>
<tr>
<td>B1-101</td>
<td>Substation 101 failure</td>
<td>2.06E+01</td>
</tr>
<tr>
<td>B1-111</td>
<td>Substation 111 failure</td>
<td>1.96E+01</td>
</tr>
<tr>
<td>B1-102</td>
<td>Substation 102 failure</td>
<td>1.81E+01</td>
</tr>
<tr>
<td>L1-101 102</td>
<td>Trans. failure 101-102</td>
<td>1.81E+01</td>
</tr>
<tr>
<td>B1-112</td>
<td>Substation 112 failure</td>
<td>1.78E+01</td>
</tr>
<tr>
<td>G2 101-1</td>
<td>Generator 101-1 failure</td>
<td>1.46E+01</td>
</tr>
<tr>
<td>B1-105</td>
<td>Substation 105 failure</td>
<td>1.18E+01</td>
</tr>
<tr>
<td>B1-115</td>
<td>Substation 115 failure</td>
<td>9.54E+00</td>
</tr>
<tr>
<td>L1-112 115</td>
<td>Line failure 112-115</td>
<td>9.52E+00</td>
</tr>
<tr>
<td>B1-118</td>
<td>Substation 118 failure</td>
<td>9.43E+00</td>
</tr>
</tbody>
</table>

The NRAW identifies elements, which should be maintained well in order that the reliability of the power system is not reduced significantly. The identified elements in Table 5 with largest value of NRAW are: substations NPP Krško (BE “B1-101”), substation Beričevo (BE “B1-111”), substation RTP Krško (BE “B1-102”) and transformer between substations...
NPP Krško and substation RTP Krško (BE “L1-101 102”). The obtained results are expected because:

- the failure of substations NPP Krško will disrupt power delivery from largest unit (NPP Krško) to the system,
- the failure of substation Beričevo will disrupt power flows to the central part of the Slovenian power system and to adjacent substations and
- the failure of the substation RTP Krško will disrupt power delivery to the largest load in the power system.

The reliability of the substation NPP Krško and the corresponding generator are the most important elements for the overall reliability of the Slovenian power system. This conclusion is supported by the NRRW and NRAW importance measures values obtained for these elements. The obtained result is expected considering the installed power of the NPP Krško as a largest unit in the Slovenian power system and circulated power through substation NPP Krško.

The implication of the changes in the power system related to the addition of new lines and change of the load/generation was tested. The obtained results are given in Table 6. The new interconnection, single (model number 2) and double (model number 3), was added to the basic configuration of the Slovenian power system (power system model number 1) between substations NPP Krško and substation Beričevo. The new nuclear power plant, identical to NPP Krško was added in the corresponding substation and two scenarios of the load increase were tested:

- proportional increase of the load when all loads were proportionally increased for the sum increase equal to the new NPP size (models 4 and 5) and
- increase of the load only in the substation Divača for a size of the newly added unit (models 6 and 7).

The obtained value for the weighted system unreliability together with the value of the unreliability of the power delivery to the house load of the NPP Krško is given in Table 6.

Table 6: The weighted system unreliability and unreliability of the power delivery to the NPP Krško for different power system configurations

<table>
<thead>
<tr>
<th>No.</th>
<th>Power system model</th>
<th>Weighted system unreliability</th>
<th>Unreliability of power delivery to NPP Krško</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.</td>
<td>The basic configuration of the Slovenian power system</td>
<td>1.40E-02</td>
<td>1.55E-04</td>
</tr>
<tr>
<td>2.</td>
<td>Single line NPP Krško – Beričevo is added</td>
<td>1.37E-02</td>
<td>1.80E-06</td>
</tr>
<tr>
<td>3.</td>
<td>Double line NPP Krško - Beričevo is added</td>
<td>1.37E-02</td>
<td>3.23E-07</td>
</tr>
<tr>
<td>4.</td>
<td>NPP Krško 2 is added, proportional load increase is assumed, single line NPP Krško – Beričevo is added</td>
<td>1.40E-02</td>
<td>3.24E-05</td>
</tr>
<tr>
<td>5.</td>
<td>NPP Krško 2 is added, proportional load increase is assumed, double line NPP Krško - Beričevo is added</td>
<td>1.40E-02</td>
<td>2.40E-05</td>
</tr>
<tr>
<td>6.</td>
<td>NPP Krško 2 is added, load increase in Divača is assumed, single line NPP Krško - Beričevo is added</td>
<td>3.18E-02</td>
<td>1.80E-06</td>
</tr>
<tr>
<td>7.</td>
<td>NPP Krško 2 is added, load increase in Divača is assumed, double line NPP Krško - Beričevo is added</td>
<td>3.17E-02</td>
<td>3.24E-07</td>
</tr>
</tbody>
</table>
The obtained value for the unreliability of the power delivery to the house load of the NPP Krško is correlated to the change of the LOOP initiating event frequency and consequent CDF of the NPP.

The change of the weighted system unreliability given in Table 6 is small because introduction of the new line NPP Krško-Beričevce doesn’t affect the reliability of the power delivery to the biggest load in the system situated in the substation RTP Krško, which is the largest contributor to the weighted system unreliability.

The obtained results in Table 6 show that:
- The introduction of the power line between substation NPP Krško and substation Beričevce improves the overall system reliability and even more the reliability of the power delivery to the house load of the NPP Krško. The obtained result is important from the aspect of the nuclear safety because, improvement of the reliability of the power delivery to the house load (self consumption) of the NPP Krško implies the more reliable offsite power and the increase of the NPP safety. In addition, the introduction of the line NPP Krško – Beričevce improves the reliability of the power delivery to other loads in the system resulting with the decrease of the weighted system unreliability.
- The obtained results show that substitution of the interconnection between substation NPP Krško and substation Beričevce from single to double lines doesn’t decrease noticeably the weighted system unreliability. The CCF of the interconnection is identified as an important contributor to the reliability of the power delivery. The change of the line between substation NPP Krško and substation Beričevce from single to double will additionally improve reliability of power delivery to the house load of the NPP Krško.
- The unreliability of the power delivery to the house load (sum of the self consumption of the both nuclear power plants) in NPP Krško is decreased compared to the unreliability obtained for the single nuclear power plant in basic configuration of the Slovenian power system, if the proportional increase of the loads is assumed and if the new nuclear power plant in the substation NPP Krško is added. The necessity of the interconnection between substation NPP Krško and substation Beričevce (single or double) is confirmed.
- With the increase of the load only in the substation Divača, the largest value of the weighted system unreliability is obtained, indicating the lowest level of the power system reliability compared to the values from other scenarios. Contrary to this, the lowest value of the unreliability of the power delivery to the house load, considering self consumption of both nuclear power plants in NPP Krško, was obtained. The obtained result indicates that from the aspect of the nuclear safety, configuration with increased load in the substation Divača corresponding to the export to the Italy, is the safest one.

4 CONCLUSIONS

The new method for the estimation of the power system reliability is presented. The method is developed integrating the fault tree analysis approach with the power flow calculations of the power system. The weighted system unreliability and importance measures for the elements of the power system are introduced. The developed method is applied on the simplified Slovenian power system. The obtained results consider the implication of the different changes in the power system on the unreliability of the power delivery. The introduced importance measures identify the elements or groups of elements, which are candidates for the improved redundancy and/or improved maintenance. The obtained results include unreliability of the power delivery to the particular load of the power system and the corresponding importance measures. This allows analysis for specific load, for example house load of the nuclear power plant, confirming the applicability of the developed method for the improvement of the nuclear safety. The obtained results show that substation NPP Krško and
the corresponding generator are the most important elements for the overall reliability of the Slovenian power system. The importance of the introduction of the new line between substation NPP Krško and substation Beričevo for Slovenian power system reliability and improved power delivery to the house load of the NPP Krško is confirmed. The implication of the change of the load size and location on the power system reliability was tested. The power system model with the load increase only in the substation Divača, corresponding to the export to the Italy, was identified with a lowest reliability of the power system. This power system model was identified as the most reliable from the aspect of the power delivery to the house load of the nuclear power plant and consequently nuclear safety.

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REFERENCES


Influence of Continuous Corium Field on Steam Explosion Simulation Results

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ABSTRACT

A steam explosion can induce dynamic loadings on nuclear power plant systems, structures and components that may potentially lead to the release of radioactive material into the environment. Although steam explosions are recognized as an important nuclear safety issue, the level of steam explosion process and consequences understanding is still not adequate. A comprehensive study was performed to highlight uncertainties and to capture the most challenging ex-vessel steam explosions in a typical pressurized water reactor cavity. The results revealed that the predicted pressure loads may be significantly higher than obtained in the OECD programme SERENA. The subsequently performed detailed analysis of the most explosive scenario revealed that the so called jet sourced droplets, originating from the continuous corium field, were an important source of melt droplets, which were actively involved in the steam explosion process. The jet sourced droplets are artificially created from the continuous corium field in the transition between the premixing and the explosion application of the steam explosion simulation to enable the treatment of stratified steam explosions. In the paper the contribution of the jet sourced droplets to the calculated high pressure loads is highlighted and discussed.

1 INTRODUCTION

Steam explosions are an important nuclear safety issue, since they can jeopardize the nuclear power plant primary system and the containment integrity. Nevertheless, the steam explosion process and consequences understanding is still not adequate. For that reason the OECD (Organisation for Economic Co-operation and Development) established the programme SERENA (Steam Explosion REsolution for Nuclear Applications) to increase the level of the steam explosion event knowledge and understanding [1]. One of the SERENA programme task was to perform reactor calculations and to compare the simulation results of in-vessel and ex-vessel steam explosion cases. The calculated pressure loads for in-vessel steam explosions were below the reactor vessel’s capacity, but for ex-vessel steam explosions the calculated pressure loads were above the capacity of the reactor cavity’s walls [1]. Since it was not possible to establish reliable safety margins in the ex-vessel cases, a need for reducing the uncertainties to acceptable levels for the steam explosion risk assessment was arisen.

A comprehensive study was performed to highlight the uncertainties sources and to capture the most challenging ex-vessel steam explosions in a typical pressurized water reactor cavity. Central and side melt pours were considered and parametric analyses were done.
varying the primary system pressure and the cavity water temperature. Namely, based on the comprehensive experimental programs (e.g. KROTOS, FARO, TROI) it can be concluded that the explosivity of the premixture and the strength of the steam explosion depends on a number of conditions, the most important being the melt material properties (the energy conversion ratio in steam explosion experiments with prototypic materials was one order of magnitude lower than with simulant materials), melt pouring mode (multiple pours form a more extended premixture than single pours), system confinement (confined systems allow more time for heat transfer between the melt and coolant), water subcooling (with higher water subcooling the premixture void fraction is lower, resulting in a stronger steam explosion), non-condensable gases (non-condensable gases hinder the direct melt water contact, reducing the explosivity of the premixture), system pressure (with a higher system pressure the vapour film around the melt droplets becomes more stable, reducing the explosivity of the premixture) [2][3].

The results of our comprehensive study revealed that the predicted pressures (up to 300 MPa) are significantly higher than obtained in the programme SERENA (up to 40 MPa). Also the obtained maximum pressure impulses (up to 0.7 MPa·s) were significantly higher than the pressure impulses obtained in the programme SERENA (up to 0.1 MPa·s). The obtained pressure impulses significantly exceed the pressure impulses, which could be important for the reactor cavity integrity and are estimated to be of the order of some tens of kPa·s. Among the performed analyses of the central and side melt pours, the maximum pressure was gained for the central melt pour scenario at 2 bar primary system over-pressure and water temperature of 60°C. The detailed analysis of the most explosive central melt pour scenario revealed that important contributions to the obtained high pressure loads were: the pressure focusing in the centre of the axial symmetrical geometry and the probable active melt droplets over-prediction. The amount of active melt droplets was probably over-predicted due to the unconsidered melt droplets crust formation. In addition, the active melt droplets were potentially over-predicted also due to the formation of the so called jet sourced droplets, originating from the continuous corium field, in the transition from the premixing to the explosion phase simulation with the MC3D code. [4][5]

In the paper, the estimation of the jet sourced droplets influence on the steam explosion results is presented and discussed in detail.

2 REACTOR CAVITY MODEL

The typical pressurized water reactor cavity was modelled in a simplified axial symmetrical cylindrical 2D geometry (Figure 1). The use of the axial symmetric representation is considered suitable for the central melt pour scenario case, where the nature of the steam explosion phenomena is essentially 2D. Also the applied FCI (fuel-coolant interaction) models are adjusted to such a cylindrical geometry. The radius of the cavity cylindrical part was 2.46 m and its height was 13.16 m. The mesh size was 25x35 cells. The numerical mesh was adequately refined in central regions, which were more important for the FCI phenomenon modelling. [2]
Figure 1: Geometry and mesh of the axial symmetric reactor cavity model.

3 MC3D COMPUTER CODE

The steam explosion simulations were done with the computer code MC3D, which is being developed by IRSN, France [6][7]. MC3D is composed by a set of two FCI applications, which are specifically built for the evaluation of the complex FCI phenomenon.

In general, the steam explosion simulation is performed in two steps. First, the premixing application is used to determine the melt, water and vapour phases distribution at the steam explosion triggering time. In the premixing application it is particularly important to appropriately determine the melt distribution. For that reason the premixing application describes the corium jet break up from continuous corium into melt droplets (order of cm in diameter), the melt droplets coalescence to continuous corium, the coarse melt droplets break up (order of mm in diameter) and the melt droplets fine fragmentation into fine fragments (less than 100 µm in diameter). The simulation of the premixing phase with the premixing application determines the initial conditions for the second step of the calculation, when the explosion application is used for the steam explosion escalation and propagation simulation through the premixture. The explosion application deals with the melt droplets fine fragmentation and the heat exchange between the produced fragments and the coolant. [6]

4 ESTIMATION OF JET SOURCED DROPLETS INFLUENCE

In the steam explosion simulation with MC3D, version 3.5, patch 1, the reactor cavity geometry given on Figure 1 was used. In both, the premixing and the explosion application MC3D default or recommended numerical and model parameters values were used. The premixing simulation of the central melt pour case was performed with initial conditions, which were set reasonable according to the expected conditions at the vessel failure during a severe accident in a typical pressurized water reactor (Table 1). Similar initial conditions were used also for the ex-vessel steam explosion simulations in the programme SERENA. The mass of corium was set conservative. [1][2][6]

The simulation results post-processing was performed with the Microsoft Excel and Lawrence Livermore National Laboratory VisIt software [8].
Table 1: Initial conditions for the steam explosion simulation. [2]

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>corium mass</td>
<td>50 t</td>
</tr>
<tr>
<td>corium density</td>
<td>8000 kg/m³</td>
</tr>
<tr>
<td>corium level</td>
<td>1.25 m</td>
</tr>
<tr>
<td>corium temperature</td>
<td>3000 K</td>
</tr>
<tr>
<td>corium liquidus temperature</td>
<td>2800 K</td>
</tr>
<tr>
<td>corium solidus temperature</td>
<td>2700 K</td>
</tr>
<tr>
<td>water level</td>
<td>3 m</td>
</tr>
<tr>
<td>water temperature</td>
<td>60°C</td>
</tr>
<tr>
<td>water sub-cooling</td>
<td>51.4°C</td>
</tr>
<tr>
<td>reactor vessel opening</td>
<td>position central</td>
</tr>
<tr>
<td>reactor vessel opening</td>
<td>radius 0.2 m</td>
</tr>
<tr>
<td>primary system pressure</td>
<td>3.5 bar</td>
</tr>
<tr>
<td>containment pressure</td>
<td>1.5 bar</td>
</tr>
</tbody>
</table>

4.1 Treatment of jet sourced droplets

In the transition from the premixing to the explosion calculation, MC3D transforms the continuous corium field into melt droplets with a diameter of 3 cm [6]. These melt droplets are called “jet sourced droplets” in this paper and are an additional source of melt droplets, which can be actively involved in the steam explosion process (active melt droplets). The jet sourced droplets are for that reason a potential source of the melt droplets amount over-prediction if the given simple modelling approach is used. Consequently the pressure loads can be over-predicted. On the other hand the jet sourced droplets, if appropriately determined, enable the treatment of jet surface fragmentation (stratified steam explosion). This indicates the importance of the appropriate transition between the premixing and the explosion phase simulation on the correct active melt droplets amount prediction.

Two separate simulations with the explosion application were performed to evaluate the influence of the jet sourced droplets. The first simulation was performed with the original unchanged MC3D source code (designator USC – unchanged source code), where the fragmentation process takes place if the melt droplets volume fraction is lower than 0.7 and if the melt droplets are not solidified [6]. The second simulation was performed with the aim to establish the influence of the jet sourced droplets on the pressure loads. To accomplish this task the MC3D source code was appropriately modified (designator MSC – modified source code). In this modification the melt droplets temperature was set 1 K below the solidus temperature in cells, where the fragmentation process could be possible and the jet sourced droplets represented more than half of the active (total) melt droplets volume fraction. By setting the melt droplets temperature below the solidus temperature, in these cells fine fragmentation was disabled. So with this code modification the contribution of jet source droplets was in average cancelled.

4.2 Steam explosion simulation results

The initial conditions for the steam explosion simulations are given in Table 1. First the premixing phase was simulated with the premixing application. The time to trigger the steam explosion (i.e. to begin the explosion application) was chosen based on the calculated explosivity criterion, which represents the liquid melt droplets volume in contact with water. The triggering time 1.3 second was used to set the initial conditions for the steam explosion simulation with the explosion application. The steam explosion phase was simulated for 0.1
seconds after triggering. The steam explosion was triggered in the cell, where the local explosivity criterion was highest.

The detailed fragmentation, pressure development, melt droplet volume fraction and temperature analysis of both USC and MSC simulations cases is presented on Figure 2. In the USC case an intense fragmentation of the jet sourced droplets and consequently an intense pressure field development was present on the cavity’s bottom. On the other hand, the fragmentation of the melt droplets on the cavity’s bottom was significantly less expressive in the MSC case, where the jet sourced droplets were in average not taken into account (see Section 4.1). Consequently the pressure development was less intense since in the MSC case the amount of the active melt droplets on the cavity’s bottom was significantly smaller.

Integral results of pressure loads for both USC and MSC simulated cases are given on Figure 3 and 4. The obtained pressure profile on Figure 3 was due to the fragmentation of the melt droplets along the jet’s stem and the melt droplets on the cavity’s bottom, which are partially jet sourced. The maximal pressure of 249.1 MPa was obtained on the cavity’s bottom at the central axis with USC and of 128.1 MPa at the central axis with MSC (Figure 3). Both maximal pressure peaks were achieved in the reactor cavity centre due to pressure focusing in the axial symmetrical geometry (Figure 2). The maximal pressure impulses were achieved on the cavity’s bottom near the central axis. The maximal impulse was 484.2 kPa·s with USC and 184.2 kPa·s with MSC (Figure 4). On the cavity’s lateral wall the maximal pressure impulse was 272.8 kPa·s with USC and 102.0 kPa·s with MSC (Figure 4).

4.3 Discussion of the steam explosion simulation results

The jet sourced droplets on the cavity’s bottom were recognized to be an important contributor to the calculated high pressure loads. Namely, the formation of the jet sourced droplets strongly increases the amount of active melt droplets on the cavity’s bottom. If we assume that in reality at most only a small part of the continuous corium would fragmentise during the steam explosion (stratified steam explosion if corium crust not too thick), then due to the artificial creation of the jet sourced droplets the calculated pressure loads are probably over-predicted (case USC). But we see that also if we in average cancel the contribution of the jet sourced droplets (case MSC), the pressure loads are still above the loads obtained in the programme SERENA [1].
Figure 2a: The USC (left) and MSC (right) simulation results are shown for the initial stage of the simulation. Melt droplet fragmentation (GAFRAG) together with the melt droplet temperature (TEMPGOU), melt droplet volume fraction (TXGOU) and pressure (PRESSION) are given.
Figure 2b: The USC (left) and MSC (right) simulation results are shown for the later stage of the simulation. Melt droplet fragmentation (GAFRAG) together with the melt droplet temperature (TEMPGOU), melt droplet volume fraction (TXGOU) and pressure (PRESSION) are given.
Figure 3: The maximal pressure $p$ time development in the reactor cavity for the USC and MSC simulations. Different time scales are shown.

Figure 4: The maximal pressure impulse $I$ time development on the cavity’s bottom (left) and on the cavity’s wall (right) for the USC and MSC simulation.

5 CONCLUSION

The performed comprehensive analysis of ex-vessel steam explosions in a typical pressure water reactor cavity revealed that significant higher pressure loads were calculated than obtained in the OECD programme SERENA [1],[2]. Among the performed simulations for central melt pour scenarios the maximal pressure was gained at 2 bar primary system over-pressure and water temperature of 60°C. A detailed analysis of this most explosive central melt pour scenario was performed to explore the reasons for the predicted so high pressure loads [5]. The detailed analysis revealed two important reasons for the obtained high pressure loads. First, the pressure focusing in the centre due to the axial symmetrical geometry of the reactor cavity and second the probably over-predicted active melt droplets amount due to the unconsidered melt droplets crust formation [5].

The amount of active melt droplets was also potentially over-predicted due to the jet sourced droplets formation from the continuous corium in the transition between the premixing and the explosion application [5]. For that reason we tried to estimate the influence of the jet sourced droplets on the calculated pressure loads. The results revealed that the pressure development was importantly influenced by the fragmentation of the jet sourced
droplets on the cavity’s bottom. The maximal pressure reduction by a factor of ~2 and the pressure impulses reduction by a factor of ~2.6 were obtained if the contribution of the jet sourced droplets was not taken into account. The results strongly indicate that FCI codes should be able to establish the proper mass of the melt droplets involved in the steam explosion process. Consequently also the transition of the continuous corium field between the premixing and the explosion phase has to be appropriately considered.

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Safety of Future Reactor Designs
Experimental Characterization of a Two-Phase Natural Circulation Loop Simulating a Passive Emergency Heat Removal System

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ABSTRACT

Among the several types of passive safety systems proposed and adopted in new generation reactor designs, the experimental investigation of a closed loop, two-phase flow, natural circulation system is depicted. Emergency Heat Removal Systems (EHRS) based on this kind of solution are envisaged as safety engineered features for advanced and innovative nuclear reactors, as in the IRIS reactor.

An experimental facility simulating one EHRS-like loop has been built and operated at SIET labs in Piacenza (Italy). The facility is a natural circulation, sliding pressure, electrically heated loop, with a helical coil steam generator as a heat source and a horizontal tube pool condenser with boiling water at atmospheric pressure as a heat sink. The loop is full scale on height (20 m) and pressure-temperature (up to 80 bar), scaled on power and heat transfer surfaces. The combination of the two-phase natural circulation with the self-pressurizing behaviour (i.e. the loop establishes its working pressure depending on the initial and boundary conditions), renders this facility a quite unique test case in the outline of past two-phase flow natural circulation experimental investigations.

The Filling Ratio (FR), i.e. the ratio between the water mass stored into the closed loop at initial conditions and the total mass that could be stored in the whole loop in cold conditions (i.e. “solid” loop), has been selected as a key parameter for the investigation of the performance of the system, in terms of heat rejection capability and working pressure. The evaluation of the suitable conditions for the loop working has been carried out as well, achieving a proper representation of its dynamic behaviour. The dynamics shows an oscillating but stable behaviour for a wide range of FRs. For the largest explored one (FR=0.79), i.e. with the loop operating almost full of water, the dynamic behaviour is characterized by large amplitude-long period oscillations of the main parameters, i.e. temperature, flowrate and pressure. The strong link between loop flowrate, heat source inlet subcooling and heat sink outlet subcooling has been highlighted. At low thermodynamic equilibrium qualities at steam generator outlet, the role of hot-leg (i.e. heat source and riser) pressure drops looks to be critical in sustaining the flowrate oscillations in the loop.

The experimental results collected on the facility appear very useful for future validations of analytical models on the natural circulation thermosyphon loops, as well as of analyses carried out by means of the best-estimate thermalhydraulic computer codes, as RELAP.

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INTRODUCTION

The innovative passive safety systems of advanced LWRs are based on natural forces, such as gravity and natural circulation. They can be considered more reliable than active systems, since the lack of mechanical moving parts or other active components should reduce the probability of hardware failure.

The natural circulation (NC) principle has been widely adopted in the past in several power conversion systems. Subcritical fossil fuelled power stations use NC as boiler flow driving mechanism, with the great advantage of a higher simplicity and of the reduction in operation costs related to the absence of pumps. In the nuclear field, the pressurized water reactors use the same principle in U-tube steam generators. In recent years, NC has become very attractive even for emergency core cooling applications, relying on elements which always exist in a nuclear reactor: the heat source, the heat sink, the piping and the gravity law [1] [2] [3].
The apparent simplicity of this physical principle in reality covers the complexity of the phenomena that mutually interact in a natural circulation loop: operating pressure, flowrate, flow quality, heat transfer coefficients are all linked together. The behavior of the NC passive systems usually entails oscillations in the thermalhydraulic parameters, e.g. the flowrate, hence leading to possible fluid dynamic instabilities. In particular, the problem of thermalhydraulic instabilities is one of the most crucial drawbacks on natural circulation systems. Instabilities could cause oscillations of the main loop parameters, inducing mechanical and thermal fatigue problems, as well as making the system unable to perform its duty due to excessive deviations from the expected behaviour (according to the so called “functional failure”, e.g. a cancellation of driving force because of too high frictional pressure drops) [4] [5].

IRIS reactor [6] belongs to the innovative NPPs making an extensive use of natural circulation principle for safety purposes. IRIS is a low/medium power (335 MW_e) pressurized water reactor for electricity production developed by an international consortium led by Westinghouse. IRIS development started in late 1999 as part of the NERI program and has rapidly progressed to a nuclear reactor design with market entry targeted for deployment in the 2012-2015 time frame. The plant conceptual design was completed in 2001 and the preliminary design is currently underway. The pre-application licensing process with NRC started in October, 2002 and IRIS is one of the designs considered by US utilities as part of the ESP (Early Site Permit) process.

The main safety system of IRIS reactor is the Emergency Heat Removal System (EHRS), aimed to transfer core decay heat and sensible heat from the reactor coolant to the environment during transients, accidents or whenever the normal heat removal paths are lost. It consists of a pair of steam generators (heat source), a hot leg (loop riser), a heat sink composed by a heat exchanger bundle submerged in the Refuelling Water Storage Tank (RWST) and a cold leg (downcomer) which closes the loop by bringing cold condensed water to steam generators (Fig.1).

IRIS shall be equipped with four EHRS subsystems, each connected to one of the four steam and feedwater lines in the penetration area outside the containment, and each one designed to provide sufficient heat removal to match core decay heat. During normal plant operations V1 and V2 valves are open and V3 valve is closed; pre-heated water coming from the regeneration line is pushed into the steam generator where is evaporated, slightly superheated and sent to the turbine. If a reactor trip occurs, the core decay heat will be normally removed by the SGs thanks to the start-up feedwater system and the steam will be directed to the condenser via the steam dump valves. In case of malfunctioning of the start-up feedwater system, the EHRS is available to remove the decay heat by means of the closure of V1 and V2 valves and the opening of V3 valve [7].

The challenge in analyzing the complex physics involved in the behavior of a natural circulation loop, requires to rely not only on sophisticated modelling tools, e.g. best estimate codes, but also on suitable experimental facilities, fundamental tools to provide data for validation of computer codes and the safety system design. In this

Fig.1 – Sketch of IRIS Emergency Heat Removal System (EHRS).
paper the results of an experimental campaign on a facility simulating a two-phase, natural circulation, closed loop, sliding pressure safety system, similar to the type adopted for IRIS EHRS, are presented. As a preliminary step, a steady state analysis has been performed to characterize the system behavior: no accident transients have been investigated, the main goal being the dynamics of the loop at given primary thermal power to be rejected to the heat sink, at different system configurations (e.g. water mass inventories into the loop). A specific feature of the system and the facility is the closed loop, sliding pressure behavior. Usually, in innovative BWR and PWR designs the two-phase NC loop for passive safety systems are connected to large primary volumes, i.e. the primary circuit, acting as a large expansion volume similar to a pressuriser. In the IRIS EHRS the loop is limited in volume since the in-pool condenser is directly connected to the helical coil steam generator tube bundle, hence a small volume when compared with the above mentioned configurations. It is expected that a dynamics with large pressure variations should occur, with possible feedbacks on the general behavior. Moreover, the water mass inventory entrapped into the loop when the isolation valves activate, should affect the system behavior in a more sensible way with respect to the above mentioned BWRs and PWRs safety systems, where larger mass inventories are expected.

The main simplification adopted in the facility, in comparison with the real safety system, is the imposed electrical power to obtain the heat source simulation, instead of the imposed temperature given by primary fluid. Although not strictly representative of the real loop conditions, it allows to investigate a preliminary stability level for the system, apart from the thermal coupling with the primary loop which could introduce further complexity. Anyway, a useful experimental database for model validation is one of the main results.

2 EXPERIMENTAL FACILITY

The IES facility (Iris Ehrs Simulator) was built and operated at SIET labs in Piacenza, as an extension of an electrically heated test section used for investigation on the full scale helical-coil steam generator tube [8]. The experimental loop is composed by a heat source, a riser, a heat sink and a downcomer (Fig.2). It is scaled 1:1300 on power with respect to the real one, whereas it is full scale on height (~20 m) and on thermalhydraulic conditions (pressure and temperature). The full height allows to reproduce the driving force.

The heat source is the electrically heated steam generator, previously built for the study of two-phase pressure drops and critical flux into helical-coil. The tube, as well as the piping, is thermally insulated by means of rock wool. The thermal losses were measured via evaporated mass in the pool and estimated as a function of the temperature difference between external tube wall and the environment. The riser is a 21.3 m long AISI 316 stainless steel tube with an inner diameter of 20.93 mm and an outer diameter of 26.27 mm. Downcomer tube is a 31.8 m long AISI 316 stainless steel tube with the same diameters of the riser. Riser and downcomer diameters have not been scaled with respect to IRIS EHRS riser and downcomer expected pressure drops. The facility operates with one pool condenser tube, 1 m long with 59 and 73 mm of inner and outer diameters. The tube simulates a condenser with horizontal tubes, slightly inclined (3°). A different solution with vertical tubes arrangement will be investigated in the next future. The condenser tube is submerged into a 250 liters pool. A metallic slab is placed few centimetres before the vapour release duct in order to reduce the presence of liquid droplets in the exiting steam. The evaporating water is continuously replaced, according to a pool water level control.

The quantities measured in the loop (more than 200 measuring points) are flowrates, pressures (absolute and differential), temperatures and powers.
The loop flowrate has been measured by a calibrated orifice (5 mm diameter) placed at steam generator inlet and instrumented with a differential pressure transducer calibrated at SIET labs (all the measurement devices are calibrated at SIET certified lab) with an estimated maximum uncertainty of 2%. The loop absolute pressure is measured at steam generator inlet via an absolute pressure transducer with a maximum uncertainty of 0.1%. Differential pressure transmitters are placed across the throttling valves and along the downcomer, also with the aim of evaluating the possible presence of two-phase mixture at condenser tube outlet. Fluid temperature measurements are obtained with K-class thermocouples, with a maximum error of 0.4°C at 100°C. They are located at steam generator inlet and outlet headers, at condenser tube inlet and outlet inside the pool. The electrical power is measured via a volt-amperometric digital instrument with a relative uncertainty of 2.5%. The main geometrical data of the facility are summarized in Tab.1.

2.1 The filling procedure

The constant volume of the loop makes its performance dependent on the water mass actually stored in it. The loop Filling Ratio (FR) is defined as the ratio between the total mass in the closed loop and the total mass of cold water that could be stored into the loop. The loop absolute pressure is measured at steam generator inlet via an absolute pressure transducer with a maximum uncertainty of 0.1%. Differential pressure transmitters are placed across the throttling valves and along the downcomer, also with the aim of evaluating the possible presence of two-phase mixture at condenser tube outlet. Fluid temperature measurements are obtained with K-class thermocouples, with a maximum error of 0.4°C at 100°C. They are located at steam generator inlet and outlet headers, at condenser tube inlet and outlet inside the pool. The electrical power is measured via a volt-amperometric digital instrument with a relative uncertainty of 2.5%. The main geometrical data of the facility are summarized in Tab.1.
test section, a specific FR was obtained according to the following procedure, starting from an empty loop (with reference to Fig. 2):
- V1 and V3 valves open: the loop is completely filled with cold water via an external feed pump (the total measured mass of cold water storable in the system was nearly 25 kg);
- V1 valve closed;
- warm up of the steam generator;
- extraction, condensation and weighting of the extracted water-steam;
- V3 valve closed when the desired FR is obtained;
- operation of the loop at the desired power level conditions.

2.2 The heat losses compensation

The larger surface over volume ratio of the facility with respect to the real system, makes the thermal losses of riser and downcomer not negligible, notwithstanding the thermal insulation. These heat losses have been compensated by heating the riser and downcomer piping with an electrical wire and tuneable power. Since the heat losses are proportional to the tube wall temperature, hence the two-phase mixture temperature, hence the system pressure, the suitable power needed to compensate the heat losses is a function of the system operating pressure. Therefore, a dynamic compensation was required, until the steady state was reached.

The inherent dynamic nature of the procedure and the time periods needed to run out the transient, make the procedure quite time consuming. On the other hand, a precise evaluation of the heat losses is achievable, by measuring the offset electrical power.

3 STEADY STATE RESULTS

3.1 Effects of test section power and Filling Ratio on system pressure

The most important effect of the thermal power on system behavior, at constant FR, is the set up of the working pressure for the loop. A physical explanation is related to the working principle of the pool condenser. The general formula describing the heat sink performance is:

\[ Q = U \cdot S \cdot LMTD \]  

The only way for transferring a higher thermal power to the pool is to increase \( U \) or \( LMTD \) (or both). The possibility of increasing \( U \) is very small due to the strong thermal resistance of tube wall and to the weak dependence of two-phase condensation and boiling heat transfer coefficients on flowrate. Thus, to increase the mean temperature difference implies an increase in loop pressure, being the pool water boiling temperature fixed by atmospheric pressure.

FR effect is physically more difficult to explain, and it seems to be strictly linked to condenser outlet subcooling. Pool condenser outlet subcooling plays a key role in terms of the loop capacity in storing water mass. High FRs turn into long subcooled zones both in the steam generator and in the condenser: usually these zones, together with the downcomer piping, accommodate the largest amount of charged water, being small the mass stored in the two-phase zones due to the rapid grow of void fraction with quality. An increase of the FR implies wider single-phase zones and higher subcooling at condenser outlet, which would result in principle in a reduction of the exchanging power capability. The thermal power is anyway fixed by the electrical heating, thus the LMTD hence the loop pressure must increase.
Fig. 3 resumes all the runs carried out on the loop for different values of power and FR, with and without riser and downcomer heat losses compensation [10]: it is apparent the system pressure increases as a function of FR and thermal power to be rejected.

3.2 Effects of Filling Ratio and power on pool condenser outlet subcooling

The effects of FR and electrical power on pool condenser outlet subcooling are shown in Fig. 4. Both the FR and the power increase the pool condenser outlet subcooling. The data at FR equal to 0.18 and 0.31 have been omitted, due to the presence of a two-phase mixture in the downcomer piping.

As discussed in the previous section, an increase in FR, i.e. loop mass content, induces the circuit to increase pool condenser and steam generator tubes subcooling lengths, in order to allocate the increased mass inventory, which could not be contained in the two-phase legs, since the void fraction shows typically high values (and furthermore, the downcomer is already filled with water). The increased subcooled zone lengths have a direct impact on pool condenser overall heat transfer coefficient; according to Eq.(1), \( U \) must be smaller due to the increased importance of the liquid zone (which gives lower HTCs), thus an increase of mean logarithmic delta temperature between loop and pool water, the latest being always fixed at 100°C, is needed. This demonstrates the necessary increase in system pressure from lower to higher FRs, which is consistent with the outcomes in Fig. 3.

In the previous reasoning the flowrate impact on pool condenser exchanging capabilities has been neglected. As it will be discussed in the next paragraph, flowrate changes due to a change in power, but not monotonically, due to the complex interactions between downcomer gravitational driving force, riser gravitational counter-driving force and loop friction.

By neglecting any change in loop flowrate with respect to the thermal power, which is indeed small, the loop allocates a larger water mass content by increasing the pool condenser outlet subcooling.

3.3 Effects of Filling Ratio and power on loop flowrate

Fig. 5 shows that the effect of FR and electrical power on system flowrate is not monotonic.

At low-medium FRs, an increase in power (considering pressure, FR and subcooling unchanged) increases the flow quality in the riser piping, reducing the gravitational counter-driving force.
Thus, the flowrate would increase until a higher steam generator outlet quality re-establishes the balance between the friction pressure drops and the overall gravitational driving force (i.e. the density difference between riser and downcomer piping). This explanation is valid when the increase in power slightly changes the steam generator inlet subcooling, that is the case of the value of FR (0.49) reported in Fig.5.

At higher FRs, the increased SG single phase zone length, as a consequence of the effect discussed in par.3.2, makes the counter-driving force higher despite the increased power, causing a reduction in flowrate.

3.4 Non-condensable gas effect on system pressure

In order to investigate the effect of the presence of non-condensable gases (e.g. air) on system behavior, some tests were performed after the introduction of a certain amount of air into the system. The presence of non-condensable gas may come from some potential sources:

- the air naturally dissolved in the secondary circuit, due to a not perfect behavior of the de-aerator of secondary cycle;
- the introduction of helium or hydrogen into the secondary circuit, coming from primary circuit, due to a steam generator tube rupture or leakage;
- the possibility of maintenance errors.

Condensation heat transfer coefficients are reduced if a non-condensable gas is added in the condensing fluid [11]. In fact, the gas collects at the interface between liquid film and condensing steam, reducing vapour partial pressure and thus its temperature (which drives the thermal exchange). This degradation in convective coefficients is more pronounced if system pressure is reduced and in stagnant mixtures rather than in forced convective condensation. A typical parameter introduced in order to quantify the amount of non-condensable gas is the ratio between the mass of air and the mass of steam ($\frac{M_a}{M_g}$). The total quantity of steam, dependent on FR and system pressure, is obtained with the simplifying hypothesis that air content does not influence steam presence.

Therefore, the loop is schematized as a volume with a defined portion occupied by liquid and the remaining portion by steam in saturated conditions, resulting in:

$$M_g = \frac{V_t - V_i \cdot FR \cdot M_0}{v_g - v_i}$$

being $M_0$ the maximum mass of liquid storable in the loop in cold conditions ($\approx 25$ kg).

The effect of non-condensable gas presence on system pressure is summarized in Fig.6. Air mass over steam mass ratio covered a range from 0% to 20%, though the highest values
must be considered not realistic in real system operation and were achieved only with the aim of emphasizing non-condensable content effect on system performance. The degradation of heat transfer coefficients at pool condenser tube results in system pressure increase; nevertheless the effect has to be considered small, being a 3% of pressure increase every 1% of air content increase the rate observed.

4 LIMITING BOUNDARIES FOR LOOP FILLING RATIO

Simple considerations allowed to identify the effects of the main thermalhydraulic parameters, especially the water mass content, on the loop steady state conditions. As far as the dynamic oscillating behavior of the loop is concerned, typical for a natural circulation system, a performance map has been obtained, where the limiting working conditions of the closed loop system are referred to the FRs and are function of system pressure, hence of thermal power to be rejected. Once an estimation of the loop pressure is obtained, the map allows to identify the maximum and the minimum FRs ensuring a suitable dynamics for the system. It is important to point out that the map does not refer to a strict and thorough stability analysis, while the range of FRs leading to a suitable dynamics for the loop is related to engineering evaluations: working conditions where flowrate inversion in the loop or two-phase mixture at Steam Generator inlet or loop completely filled with liquid occur, are evaluated as unfitted for a two-phase flow, closed loop safety system.

Large amplitude oscillations have been observed at the lowest FRs investigated. Taking into account the SG tube, it can be assumed that a necessary but not sufficient condition for the stability is to have a negative thermodynamic quality at inlet: an increase in inlet quality, at fixed power, causes a wider two-phase zone length, leading to higher frictional pressure drops which are destabilizing. As previously stated, a boundary case to be avoided as prone to unstable behaviour, is the operating condition with two-phase mixture entering the heated channel. This limit is in practice related to the volume of the downcomer with respect to the total volume of the loop. If the water initially stored into the loop is not enough to completely fill in the downcomer (i.e. small FR), the configuration previously described will occur. Obviously this minimum value must depend on system pressure because of the dependence of liquid density on pressure in saturated conditions. In order to quantify this limiting boundary by computing the minimum water inventory to be stored in the loop, the following simple conservative assumptions are made:

- homogeneous mixture flowing in the loop;
- downcomer completely filled with liquid water at saturation temperature;
- Steam Generator with complete evaporation of the mixture and linear increase of quality (from \( x=0 \) to \( x=1 \));
- riser filled with saturated steam;
- condenser tube with complete condensation and linear decrease of quality (from \( x=1 \) to \( x=0 \)).

According to the previous assumptions, the masses stored in the four components of the loop (steam generator, riser, condenser and downcomer) are respectively:

\[
M_{SG} = V_{SG} \ln\left(\frac{v_{g}}{v_{l}}\right)
\]

\[
M_{riser} = \frac{V_{riser}}{v_{g}}
\]
\[ M_{\text{cond}} = V_{\text{cond}} \ln\left(\frac{v_g}{v_l}\right) \]

\[ M_{\text{down}} = \frac{V_{\text{down}}}{v_l} \]

The minimum value of the mass stored in the system in order to ensure a downcomer full of liquid water is thus:

\[ M_{\text{min}}(p) = M_{\text{SG}} + M_{\text{riser}} + M_{\text{cond}} + M_{\text{down}} \]

and the corresponding minimum filling ratio \((FR_{\text{min}})\) for the loop is:

\[ FR_{\text{min}}(p) = \frac{M_{\text{min}}(p)}{M_0} \]

At very high FRs, i.e. very high liquid contents, there must be a temperature hence a pressure in correspondence of which liquid specific volume is high enough to fill the entire loop volume. In this case, the degree of subcooling of the liquid leaving the pool condenser allows the loop to store a higher quantity of mass, thus rising the maximum allowable FR. For a simple evaluation of the maximum FR, all the loop is assumed filled with saturated liquid, except for the downcomer which is filled with subcooled water. With these assumptions the total mass stored is:

\[ M_{\text{max}}(p, \text{subcooling}) = v_g(p)(V_r - V_{\text{down}}) + v_{l,s}(\text{subcooling})V_{\text{down}} \]

resulting in a maximum allowable FR equal to:

\[ FR_{\text{max}}(p, \text{subcooling}) = \frac{M_{\text{max}}(p, \text{subcooling})}{M_0} \]

The map with loop maximum and minimum FR as a function of system pressure, has been drawn both for IES experimental facility and for IRIS EHRS loop design, as shown in Fig.7 and Fig.8. It has to be intended as a quick tool to define the range of the mass to be charged into the system, in order to ensure a suitable functioning. The different shapes of lower curves in the two diagrams between IES facility and IRIS EHRS are due to their different volumes distributions. In particular, in IES loop the Steam Generator volume is just.
16% of total loop volume, whereas it is about 40% in IRIS EHRS loop. The increased importance of SG volume in EHRS makes possible to store more mass in the component, thus the system minimum mass could increase passing from low to high pressures, due to the decrease of steam specific volume with pressure in Eq.(3). On the contrary, IES loop has a smaller SG volume and an increase in system pressure has the main effect of reducing liquid inventory into the downcomer.

The map in Fig. 7, related to IES facility, reports the experimental data (FR vs. system pressure) of the test matrix, already used to set up the graph in Fig. 3. The cases with heat losses compensation and those without compensation are taken into account; it is apparent how the compensation contributes to enhance system pressure. The dynamics and behaviour of the runs with FR equal to 0.49, 0.61 and 0.79 always fall inside the loop working region, whereas the smallest FRs explored (0.18 and 0.31) were beyond the minimum acceptable FR. The measurements for these runs show the presence of a two-phase mixture in the downcomer piping, leading also to large uncertainties in the measurement of the system flowrate.

5 STABILITY ANALYSIS

Many Authors in the past studied the stability features of a natural circulation loop, both with analytical tools and with experimental works. Instability phenomena can be produced by appropriate combinations of geometrical and operating parameters, namely loop height and length, as well as heating power and friction. A pioneering work in this field is due to Welander [12], who showed that these instabilities turn in the amplification of perturbed temperature slugs generated in the heated and cooled sections. These analyses have been recently resumed by Ambrosini et al. [13] [14] [15], facing both analytically and with numerical codes the linear and non-linear stability of a single-phase natural circulation loop. As far as the analytical approach is concerned, the main technique consists in linearising the momentum and the energy equations as well as the boundary conditions, and in the adoption of a first-order perturbation method. Suitable physical and geometrical parameters, evaluated in accordance with specific heat transfer laws and friction factor correlations, are used to completely identify the dynamic behaviour of the system. On this basis, stability maps can be drawn, discriminating between stable and unstable regions according to the sign of the real part of the dynamic matrix eigenvalues (negative for stability and positive for instability conditions).

Other Authors referred the stability analysis to the interpretation of the dynamic flow oscillations which can appear in the loop [16] [17] [18]. In particular, Jiang et al. [19] experimentally studied a particular configuration which is named flow excursion, a new kind of static flow instability occurring at very low steam quality conditions and determined by the loop flow resistance and the internal driven head inside the natural circulation loop. Characteristic curves, operational curves and bifurcation curves are the available methods to analyze these instabilities, as proposed by Yang [20]. The same works show moreover how this flow excursion is prior to the low steam quality density wave oscillation, characterized by a dynamic interrelation between void fraction, pressure drops and flowrate and by a strong coupling between the heated section and the heat sink condenser outlet.

In this paper, a particular kind of low frequency oscillations, detected during the IES experimental campaign at the highest explored FRs, is discussed and interpreted in accordance with literature results.

At the very low qualities which derive from a high FR, small flowrate variations can in fact cause large variations in riser void fraction, changing its gravitational pressure drops. The perturbation of loop pressure drops has a direct impact on the flowrate, influencing also pool condenser outlet subcooling.
The coupling and the time delays between flowrate, pressure drops and downcomer subcooling become the main responsible for the observed low frequency oscillations. These oscillations refer to the steady state behaviour of the system, since they occur at equilibrium conditions and not during transients. They are mainly induced by the specific boundary conditions, i.e. the imposed power at the heat source, the imposed temperature at the heat sink and the unavoidable phase displacement due to the fact that the fluid is moving in the loop.

More in detail, the mentioned low frequency oscillations, with periods of about 300 s, were detected only in the runs with the highest FR (0.79) and with the lowest power, i.e. 23 kW and 33 kW. In Fig. 9 mass flowrate, pool condenser outlet temperature and steam generator inlet temperature are plotted for a run in which such high amplitude-long period phenomena were detected. The period of the oscillations is 347 s. The transit time for a particle flowing into the loop, i.e. the time period needed to travel the entire closed loop, is:

\[ T_{\text{loop}} = \frac{M}{\Gamma} \]  

(11)

being \( M \) the stored mass and \( \Gamma \) the flowrate. The run reported in Fig. 9 refers to a FR of 0.79, i.e. a stored mass of 19.75 kg, while the measured mean flowrate was 0.086 kg/s, resulting in a particle transit time equal to 230 s, i.e. a period of oscillation nearly 1.5 time the fluid transit time during that run.

The similarity between the oscillation period and the particle residence time in the loop suggests that this oscillatory mode is somehow related to very slow enthalpy waves which travel in the loop with the same mean velocity of the mixture. These waves create a strong coupling between heated test section flowrate, riser pressure drops and condenser outlet subcooling. This type of oscillations, discussed in the above mentioned works e.g. [19], is probably related to the specific shape of the hot leg (steam generator and riser) characteristic (pressure drops dependence on loop flowrate, at different subcooling). The calculated Steam Generator pressure drops are shown in Fig. 10. All the terms of pressure drops are included, except for the negligible accelerative term. The main result is that, everything else being the same, an increase in Steam Generator inlet subcooling reduces its total pressure drops.

Observing the curves reported in Fig. 9, the similarity between SG inlet subcooling and loop flowrate supports the existence of a link between the two. In particular, SG inlet subcooling increase (point A) is always anticipating loop flowrate increase (point B), suggesting that the former is causing the latter. In order to understand the low frequency
oscillations propagation phenomenon, it is useful to consider a perturbation in the operative conditions, for instance an enthalpy wave characterized by a decreasing temperature ramp exiting the pool condenser. The enthalpy decreasing ramp entering the SG is equivalent to an increasing subcooling. According to the results of Fig.10, this subcooling wave will cause a reduction of SG total pressure drops which will increase loop flowrate (point B in Fig.9). The increased flowrate has the main effect of causing an ascending ramp of temperature at pool condenser outlet (point C). This fact can be explained by a transient behaviour of the heat exchanger where, due to the increased flowrate, tube wall and liquid bulk are not able to reach a steady state condition and have not enough time to exchange the power, resulting in a reduced extracted power and thus an increased outlet temperature. The increase in flowrate brings a decreasing SG outlet quality, which is then responsible for a first reduction in loop flowrate (from point D to point E), due to the growing of riser counter-driving force caused by the increased gravitational pressure drops. The loop flowrate will be further reduced when the temperature ascending ramp, travelling along the downcomer, will reach steam generator inlet. A reduction in SG inlet subcooling has in fact the effect of increasing its total pressure drops, thus reducing the flowrate. This reduction in loop flowrate will cause finally an enthalpy wave in the form of a temperature descending ramp starting from the pool condenser, which will create the conditions for a new oscillating cycle of the system.

6  CONCLUSIONS

The experimental campaign carried out in SIET labs on the IES facility simulating an IRIS EHRS-like loop has been described in the paper, the main goal being to investigate the influence of the mass inventory charged in the fixed volume system. The most important effect of FR on loop performance is strictly linked to the condenser outlet subcooling. High FRs turn into long subcooled zones both in the Steam Generator and in the condenser. The increased subcooled zones lengths have a direct impact on pool condenser overall heat transfer coefficient: the HTC must be smaller due to the increased importance of single-phase heat transfer, resulting in an increase of the mean logarithmic temperature drop hence of the pressure. The effects of FR and electrical power on system flowrate have been evaluated, as well as the influence of non-condensables. The latter causes a reduction in condensation heat transfer, bringing to a slight loop pressure increase.

A simplified characterization of the facility has been proposed, to identify the FRs suitable for an effective behavior of the system. The evaluation of boundary values for FR has been summarized into a map, where the possible working conditions of the natural circulation loop are represented as a function of system pressure, both for IES experimental facility and for IRIS EHRS. The map is useful to quickly identify how much water has to be stored, according to the pressure level hence to the power to be rejected. The experimental runs on the IES facility and corresponding measurements and behaviour confirmed the validity of the map.

A particular type of low frequency oscillations, detected during the runs with the highest FR explored, has been investigated. These high amplitude-long period oscillations are related to the particular boundary conditions of the system, i.e. the imposed power at the heat source, the imposed temperature at the heat sink and the unavoidable phase displacement due to the fact that the fluid is moving in the loop. At low qualities, a small SG inlet subcooling variation leads to sensible variations in SG and riser void fractions. The pressure drops influence loop flowrate which impacts also on pool condenser outlet subcooling. The coupling and the time delays between flowrate, pressure drops and downcomer subcooling cause the observed phenomenon of low frequency oscillations. The period of these
oscillations (about 300 s) is interpreted according to the concept of transit time for a particle flowing into the loop.

The explanation of the low frequency oscillations proposed in the paper appears physically reasonable, being in accordance with all the observed phenomena and without contradiction with the experimental results. Nevertheless, a confirmation of the mentioned cause-effect relations could come from a quantitative model of the system. The experimental campaign has indeed to be intended also as an important database useful to validate the accuracy of analytical models devoted to the dynamics of natural circulation thermosyphon loops. Both a simplified analytical model and a best estimate (e.g. RELAP5) numerical model of the IES facility are under development and will be validated on the experimental data.

REFERENCES


Research Reactors
Safety Issues of Cold Neutron Sources in Research Reactors

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ABSTRACT

There are 17 cold neutron sources (CNS) operating in research reactors (RR), 2 under construction, about 5 planned and 11 decommissioned. Some CNS have been built at the same time as the reactor itself, others have been back-fitted, either in the frame of a major power upgrade, or without touching general lay-out of the reactor. The CNS are secondary moderator volumes placed close to the core of the RR, and cooled to cryogenic temperatures. The moderation lowers the energy of the neutrons to below 20 meV, an energy range extremely useful for studying a variety of topics like molecular structures, stress in material, magnetism, and even the properties of the neutron itself. The safety risks are of three kinds: chemical, nuclear, and cryogenic. These risks are considered in detail, and examples are given about how to cope with them.

TOPICS

1. Definition and use of a cold neutron source
2. Different design types
3. Safety aspects
   3.1 Safety philosophy
   3.2 Chemical risks
   3.3 Nuclear risks
   3.4 Cryogenic risks
4. References
5. Table I : Table of CNS occurrence
1 DEFINITION AND USE OF A COLD NEUTRON SOURCE

A cold neutron source (CNS) is a secondary moderator volume placed close to the core of the research reactor, and cooled to cryogenic temperatures (Fig. 1).

All CNS are built according the following scheme:

- Close to the reactor core: the moderator chamber with its vacuum jacket
- Pumping line, filling line, cryogenic lines leading to out-of-pile components
- Moderator filling and storage system
- Vacuum system, refrigerator with compressor
- Control system

The moderation lowers the energy of the neutrons to below 20 meV, an energy range extremely useful for studying a variety of topics like molecular structures, stress in material, magnetics, chemical composition, and even the nuclear properties of the neutron itself.

Efficient moderator substances should have a high inelastic scattering cross section and low absorption. Liquid hydrogen or hydrogen-rich chemical compounds, as well as liquid deuterium, are the best moderators. The cold neutrons escape from the moderator chamber with an energy distribution corresponding to the moderator temperature, usually well below 100 K (Fig. 2). To get a beam of cold neutrons out of the reactor one collects them in one or several beam tubes crossing the biological shield. Cold neutrons can be guided over several 10 meters to be used on neutron scattering instruments and other experiments with neutrons.
Fig. 2: Theoretical spectral distribution of neutron flux in cold and warm moderators.

2 DIFFERENT DESIGN TYPES

The world-wide 30 CNS (see Fig.3) which are operating, under construction, or decommissioned, can be classified by the following features (see also Table I further down):

- Composition and state (solid or liquid or gaseous) of the moderator
- Process of heat removal

Fig. 3: Geographical occurrence of CNS (not only the reactor bound CNS)
2.1 Composition and state of the moderator substance

As mentioned before, hydrogen is – together with deuterium – the best moderator. It is kept cool at about 20 K, in order to slow down as many neutrons as possible to velocities below 1 km/s, corresponding in energy to about 5 meV. It is used either as a liquid, or, only for hydrogen, also in the supercritical state at a high pressure such that the proton density is close to the one in the liquid. Isotopic mixtures are sometimes used too. The liquid or supercritical fluid can easily be cooled by contact with the heat exchanger of a refrigerator.

Hydrogenous compounds like methane, water, mesitylene, or clathrates are used because of their high hydrogen densities, even at higher than cryogenic temperatures. They suffer from decomposition or polymerisation under nuclear radiation (radiolysis, see below).

All moderator substances (besides water) react more or less violently with oxygen, a fact which has to be considered a serious safety hazard.

2.2 Process of heat removal

The great advantage of hydrogen and its isotopes is that they stay fluid at very low temperatures. The heat generated in the moderator by nuclear radiation can therefore easily be removed by the moderator fluid itself. The fluid circulates in a closed cycle between the moderator chamber and a heat exchanger, which can be located relatively far away from the reactor core, and which is cooled by an industrial refrigerator (Fig. 4).

We distinguish between natural convection cooling (gravity driven) and forced convection (driven by a pump or a blower). In the first case (also called thermal siphon) the heat exchanger is placed at a level above the moderator chamber, from which the warm (low density) fluid lifts up, is cooled in the heat exchanger, and, being more dense, flows back.

Solid moderators are directly cooled by the refrigerant (e.g. He gas) circulating from and to the refrigerator. The direct cooling limits the heat removal to a few W while the closed loop cooling can handle up to several kW.

Fig. 4: Methods of heat removal: natural convection (left), and forced convection (right).
3 SAFETY ASPECTS

3.1 Safety philosophy

Practically all research reactors with a CNS adopt the following safety philosophy:

- Any anomalous behaviour of the CNS shall not affect the safety of the reactor itself.
- No explosive mixture with air shall develop in the reactor or on the RR site.
- No radioactive contamination above authorized levels shall affect the environment.

If the reactor is "emergency" shut down because of a dysfunction of the CNS, then only in order to avoid damaging the CNS.

The installation and operation of a CNS in a research reactor is always subject to a licensing procedure, no matter whether it is installed in parallel with a new reactor construction, or back-fitted into an existing RR. The Safety Analysis Report (SAR) is best elaborated according to the guidelines of the IAEA Safety Guides [1] “Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report”, and "Safety in the Utilisation and Modification of Research Reactors".

The most recent Safety Analysis Reports have been edited in Oak Ridge [2], at NIST [3], in Garching at FRM2 [4], and in Sydney by ANSTO [5] (see also Table I).

The TÜV Rheinland in Germany (A. Scheuer, co-author of this paper) is specializes as independent expert in the field of CNS licensing.

3.2 Chemical risks

The risk of explosion of the moderator substance in contact with air can be minimized by creating at least 2 solid barriers between the moderator and the atmosphere. This double containment concerns the components close to the reactor core and inside the reactor building. Two barriers usually enclose an inert gas liner (He or nitrogen), the pressure of which is above atmospheric pressure. The integrity of the barriers is assured by continuous leak testing. In spite the fact that the occurrence of an explosive mixture with air is extremely improbable, some regulators require that at least one barrier can withstand the pressure peak of an atmospheric explosion in the moderator chamber or in the cryogenic vacuum jacket.

For liquid moderators it is important that the gas has the required purity (absence of oxygen) from the beginning, and that eventual refills are made with high purity gas.

In solid moderators there are two additional risks, namely the development of radiolysis products and of lattice defects due to the strong nuclear radiation field. In both cases the recombination can be sudden and strongly exothermal, leading to explosion-like pressure peaks. The best countermeasure is to anneal the moderator substance from time to time at high enough temperature.
3.3 Nuclear risks

These are of two kinds:
- effects on reactivity
- activation of the moderator substance and structure.

A sudden evaporation or leakage of the moderator creates a reactivity surge which has to be compensated for by the reactor control system. In practice this effect is usually negative and never fast enough to provoke a power excursion. Nevertheless the reactivity change is calculated and the worst case assumption considered in the layout of the reactor control system.

The moderator activates in the continuous flux of thermal and cold neutrons. Especially in a deuterium moderator a significant amount of tritium builds up with time (at ILL Grenoble 0.5 ppm/cycle, equivalent to about a GBq). But, since the system is gas-tight and surrounded by a containment for the reasons enumerated above, the environmental contamination risk is very low. The activated moderator and the activation of in-pile CNS structure material (in particular zircaloy) represents a risk, however, if the CNS is decommissioned.

3.4 Cryogenic risks

In case of failure of the refrigerator the moderator does no longer cool the moderator chamber, it warms up, and there is a risk of destruction the chamber if the reactor is not shut down in time. Some reactors (BER2, FRG2, RRR, WWR Budapest) continue operation with an emergency cooling of the moderator chamber, others can only restart when the refrigerator is operating again (ILL HFR, ORPHEE). There is a strong demand for availability of the refrigerator, MTBF of >1000 hours typically. Some installations have doubled sensitive components for redundancy.

All fluid moderators solidify if the temperature becomes low enough. If this happens, there is a risk of blockage of the moderator flow followed by a warm-up in the moderator chamber because of lack of cooling. The moderator then evaporates and eventually destroys the moderator chamber. A good temperature and/or pressure control (usually with 2-of-3 logic) in the moderator circuit is required in order to minimize this risk.

4 REFERENCES

[2] The SAR is not public, information can be obtained from D. L. Selby (yb2@ornl.gov)
[3] The SAR is not public, information from R. E. Williams (robert.williams@nist.gov)
[4] Private communication from FRM2 Garching (klaus.schreckenbach@frm2.tum.de)
[5] The SAR is not public, information can be obtained from Ross Miller (rmx@ansto.gov.au)

More references with details of each existing or projected CNS are given in the OTTOSIX home page (www.ottosix.com).
### TABLE I

Cold Neutron Sources: operating, under construction, or planned

<table>
<thead>
<tr>
<th>Place, Country</th>
<th>Reactor</th>
<th>Power</th>
<th>CNS Type</th>
<th>Temperature</th>
<th>Power</th>
</tr>
</thead>
<tbody>
<tr>
<td>Austin, TX, USA</td>
<td>TRIGA Mark II</td>
<td>1 MW</td>
<td>mesitylene</td>
<td>40 K</td>
<td>16 W</td>
</tr>
<tr>
<td>Beijing, CN</td>
<td>CARR *)</td>
<td>60 MW</td>
<td>liq. hydrogen</td>
<td>23 K</td>
<td>~800 W</td>
</tr>
<tr>
<td>Berlin, DE</td>
<td>BER-2</td>
<td>10 MW</td>
<td>hydrogen gas</td>
<td>28 K</td>
<td>1800 W</td>
</tr>
<tr>
<td>Budapest, HU</td>
<td>WWR</td>
<td>10 MW</td>
<td>liq. hydrogen</td>
<td>20 K</td>
<td>250 W</td>
</tr>
<tr>
<td>Cairo, EG</td>
<td>ETRR2</td>
<td>22 MW</td>
<td>project</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Daejeon, KR</td>
<td>KAERI Hanaro*)</td>
<td>30 MW</td>
<td>(liq. hydrogen)</td>
<td>20 K</td>
<td>~1000 W</td>
</tr>
<tr>
<td>Delft, NL</td>
<td>HOR</td>
<td>2 MW</td>
<td>liq. hydrogen</td>
<td>project</td>
<td></td>
</tr>
<tr>
<td>Dubna RU</td>
<td>IBR-2 pulsed ¹)</td>
<td>2 MW</td>
<td>solid methane</td>
<td>30-70 K</td>
<td>150 W</td>
</tr>
<tr>
<td>Gaithersburg, USA</td>
<td>NIST NBSR ¹)</td>
<td>20 MW</td>
<td>liq. hydrogen</td>
<td>20 K</td>
<td>850 W</td>
</tr>
<tr>
<td>Garching, DE</td>
<td>FRM2 ¹)</td>
<td>20 MW</td>
<td>liq. deuterium</td>
<td>25 K</td>
<td>5000 W</td>
</tr>
<tr>
<td>Gatchina, RU</td>
<td>WWM-R</td>
<td>15 MW</td>
<td>LD2 + LH2</td>
<td>20 K</td>
<td>4000 W</td>
</tr>
<tr>
<td>Gatchina, RU</td>
<td>PIK</td>
<td>100 MW</td>
<td>project</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Geesthacht, DE</td>
<td>FRG2 MTR</td>
<td>5 MW</td>
<td>hydrogen gas</td>
<td>28 K</td>
<td>1150 W</td>
</tr>
<tr>
<td>Grenoble, FR</td>
<td>ILL HFR</td>
<td>57 MW</td>
<td>liq. deuterium</td>
<td>25 K</td>
<td>6500 W</td>
</tr>
<tr>
<td>Grenoble, FR</td>
<td>ILL HFR ¹)</td>
<td>57 MW</td>
<td>liq. deuterium</td>
<td>25 K</td>
<td>3000 W</td>
</tr>
<tr>
<td>Juelich, DE</td>
<td>DIDO ²)</td>
<td>23 MW</td>
<td>liq. hydrogen</td>
<td>19 K</td>
<td>700 W</td>
</tr>
<tr>
<td>Kjeller, NO</td>
<td>Jeep-2</td>
<td>2 MW</td>
<td>liq. hydrogen</td>
<td>21 K</td>
<td>60 W</td>
</tr>
<tr>
<td>Kyoto, JP</td>
<td>KURR ²)</td>
<td>5 MW</td>
<td>liq. deuterium</td>
<td>25 K</td>
<td>250 W</td>
</tr>
<tr>
<td>Lucas Heights, AU</td>
<td>RRR</td>
<td>20 MW</td>
<td>liq. deuterium</td>
<td>18 K</td>
<td>4000 W</td>
</tr>
<tr>
<td>Mianyang, CN</td>
<td>CMRR</td>
<td>20 MW</td>
<td>liq. hydrogen</td>
<td>20 K</td>
<td>1500 W</td>
</tr>
<tr>
<td>Oak Ridge, USA</td>
<td>HFIR</td>
<td>85 MW</td>
<td>hydrogen gas</td>
<td>20 K</td>
<td>3000 W</td>
</tr>
<tr>
<td>Saclay, FR</td>
<td>ORPHEE</td>
<td>14 MW</td>
<td>liq. hydrogen</td>
<td>20 K</td>
<td>500 W</td>
</tr>
<tr>
<td>Saclay, FR</td>
<td>ORPHEE</td>
<td>14 MW</td>
<td>liq. hydrogen</td>
<td>20 K</td>
<td>650 W</td>
</tr>
<tr>
<td>Serpong, ID</td>
<td>Siwabessy</td>
<td>30 MW</td>
<td>project</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Tokai-mura, JP</td>
<td>JRR-3M, JAERI</td>
<td>20 MW</td>
<td>liq. hydrogen</td>
<td>20 K</td>
<td>350 W</td>
</tr>
</tbody>
</table>
Neural Network application to on-line monitoring of CRDM and thermo-hydraulic condition

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Summary
The work is carried out on ENEA RC-1, a Triga Mark II 1 MW - demineralized water / natural convection cooled - research reactor, located at ENEA Casaccia Centre. Activity is aimed at intelligent processing of the data obtained by reactor measurements through soft-computing models based on neural networks (NN). A first application is made for CRDM rod position validation and a second one is devoted to the fuel temperature prediction. Both use the NN-based algorithms working on real data coming from sensors. The bias between real and calculated values is used to train the NNs for improving the performance in the further experiments.

1. Introduction
Continuous remote monitoring is more and more needed as the new generation reactors tend to dramatically decrease the outage time for maintenance and refuelling operations.
Plant sensor on-line monitoring, data validation through soft-computing process and plant condition monitoring techniques would help identify plant sensors drift or malfunction and operator actions in addressing nuclear reactor control. On-line recalibration can often avoid intervene with manual calibration or physical replacement of the drifting component.

Further, neuro-fuzzy logic allows to link field measurement and physical system behaviour, so interpreting the data coming from the reactor instrumentation and revealing eventual mismatch due to miscalibration or sensing faults.

The activity described in the paper is aimed at validating data obtained by reactor measurements through soft-computing models based on neural networks (NN).
Application is made on the CRDM rod position and the fuel temperature control which are both measured and re-calculated by NN-based algorithms and compared with the real data coming from sensors. The bias is used to train the NNs for improving the performance in the next experiments.

2. Description of the reactor

The reactor is a typical TRIGA light-water (research) reactor with an annular graphite reflector cooled by natural convection, with a power of 1MW. The reactor core is placed at the bottom of the 6.25-m-high open tank with 2-m diameter. The core has a cylindrical configuration.

In total there are 91 locations in the core, which can be filled either by fuel elements or other components like control rods, a neutron source, irradiation channels, etc. The core lattice has an annular but not periodic structure.

The core temperature is measured by 20 thermocouple situated above and under the core (see Fig.1), while the fuel temperature is measured in two fuel elements instrumented with thermocouples.

Fig.1 – TRIGA Core: thermocouples position
3. Neural Network application to validation of control rod position

In this first application the position of the control rod is being validated. In order to, a neural network elaborates the data measured on the thermo-fluid-dynamic system of the reactor, in parallel with the control system. The net is feed-forward 3-layers network trained through second-order algorithms from Levenberg-Marquart. It consists of two nets, see Fig.2, working in parallel: a first one is used for data validation and initially trained through the use of reactor data set; a second one is used for training and updates its parameters every time a given error occurs due to the difference between the exact value generated by the first net and the value measured from the instruments and validated by the operator. The new training parameters will then be transferred to the first net.

The geometry of the two nets is common and consists of a 35 neurons input layer, a 22 neurons hidden layer and a 2 neurons output layer, see Fig.3.
3.1 Levenberg-Marquardt Algorithm

The Levenberg-Marquardt (LM) algorithm is an iterative technique that locates the minimum of a multivariate function that is expressed as the sum of squares of non-linear real-valued functions.

LM can be thought of as a combination of steepest descent and the Gauss-Newton method. When the current solution is far from the correct one, the algorithm behaves like a steepest descent method: slow, but guaranteed to converge. When the current solution is close to the correct solution, it becomes a Gauss-Newton method.

Let $f$ be an assumed functional relation which maps a parameter vector $p \in \mathbb{R}^m$ to an estimated measurement vector $x = f(p), \ x \in \mathbb{R}^n$. An initial parameter estimate $p_0$ and a measured vector $x$ are provided and it is desired to find the vector $p$ that best satisfies the functional relation $f$, i.e. minimizes the squared distance $\epsilon^T \epsilon$ with $\epsilon = x - x$. The basis of the LM algorithm is a linear approximation to $f$ in the neighborhood of $p$. For a small $\| \delta p \|$, a Taylor series expansion leads to the approximation

$$f(p + \delta p) \approx f(p) + J \delta p$$

(1)

where $J$ is the Jacobian matrix.

Like all non-linear optimization methods, LM is iterative: Initiated at the starting point $p_0$, the method produces a series of vectors $p_1, p_2, p_3, \ldots$ that converge towards a local minimizer for $p^*$. Hence, at each step, it is required to find the $\delta p$ that minimizes the quantity

$$\| x - f(p + \delta p) \| \approx \| x - f(p) + J \delta p \| = \| \epsilon - J \delta p \|.$$

The sought $\delta p$ is thus the solution to a linear least-squares problem: the minimum is attained when $J \delta p - \epsilon$ is orthogonal to the column space of $J$. This leads to

$$J^T (J \delta p - \epsilon) = 0$$

which yields $\delta p$ as the solution of the so-called normal equations:

$$J^T J \delta p = J^T \epsilon$$

(2)

The matrix $J^T J$ in the left hand side of Eq. (2) is the approximate Hessian, i.e. an approximation to the matrix of second order derivatives. The LM method actually solves a slight variation of Eq. (2), known as the augmented normal equations

$$N \delta p = J^T \epsilon$$

(3)

where the off-diagonal elements of $N$ are identical to the corresponding elements of $J^T J$ and the diagonal elements are given by
\[ N_{ii} = \mu + [J^T J]_{ii} \]

for some \( \mu > 0 \). The strategy of altering the diagonal elements of \( J^T J \) is called \textit{damping} and \( \mu \) is referred to as the \textit{damping term}. If the updated parameter vector \( p + \delta p \) with \( \delta p \) computed from Eq. (3) leads to a reduction in the error \( \varepsilon \), the update is accepted and the process repeats with a decreased damping term. Otherwise, the damping term is increased, the augmented normal equations are solved again and the process iterates until a value of \( \delta p \) that decreases error is found. The process of repeatedly solving Eq. (3) for different values of the damping term until an acceptable update to the parameter vector is found corresponds to one iteration of the LM algorithm.

In LM, the damping term is adjusted at each iteration to assure a reduction in the error \( \varepsilon \). If the damping is set to a large value, matrix \( N \) in Eq. (3) is nearly diagonal and the LM update step \( \delta p \) is near the steepest descent direction. Moreover, the magnitude of \( \delta p \) is reduced in this case. Damping also handles situations where the Jacobian is rank deficient and \( J^T J \) is therefore singular [3]. In this way, LM can defensively navigate a region of the parameter space in which the model is highly nonlinear. If the damping is small, the LM step approximates the exact quadratic step appropriate for a fully linear problem. LM is adaptive because it controls its own damping: it raises the damping if a step fails to reduce \( \varepsilon \); otherwise it reduces the damping. In this way LM is capable to alternate between a slow descent approach when being far from the minimum and a fast convergence when being at the minimum’s neighborhood [3].

The LM algorithm terminates when at least one of the following conditions is met:
- The magnitude of the gradient of \( \varepsilon^T \varepsilon \), i.e. \( J^T \) in the right hand side of Eq. (2), drops below a threshold \( \varepsilon_1 \);
- The relative change in the magnitude of \( \delta p \) drops below a threshold \( \varepsilon_2 \);
- The error \( \varepsilon^T \varepsilon \) drops below a threshold \( \varepsilon_3 \);
- A maximum number of iterations \( k_{\text{max}} \) is completed.

3.2. NN Training

3.2.1 Training data set

Thermo-hydraulic values of the system are varied with the position of the control rod. The aim of this application is the validation of control rod position. In order to do this, core temperature, fuel temperature and water temperature are selected as meaningful variables.

The training data set for neural network training is retrieved by the reactor data capture software which records all values every 4 seconds. It is composed by:

Input data: core temperature (20 thermocouple); fuel temperature; water temperature.

Output data: rod position (2 channels).
Figures 4 and 5 display, the core temperature trends from thermocouples N3T N3B, S2T S2B, as a significant example of the whole 20 thermocouples set.

In Figure 6 the trend of fuel and water temperatures is shown, according to the control rod withdrawal and the power raise.
In Figure 7 the channels SH1 and SH2 rod position is shown.

3.3 Training

In order to implement the previously described L.M algorithm, \texttt{TRAINLM} function of Matlab is used.

The chosen parameters for training the net have been:
- The magnitude of the gradient: $10^{-10}$
- The relative change in the magnitude: 0.01
- The error: 0
- A maximum number of iterations: 140 epochs

After the training process, the value of the achieved train performance has been 9.8e-8. It has been noticed, either, that increasing the number of the epochs, the output data remain almost unchanged. The error’s trend is shown, below, in Fig.8.
As the error resulted to be sufficiently low, the choice of used data demonstrated to be appropriate for neural network training.
The outcome of training is displayed in figs. 9 and 10, below.
In Fig.9 it’s shown that the experimental data match with neural network data.
In fig. 10, it’s shown how the rod position error remains low, ranging between $10^{-2}$ and $10^{-3}$ steps.
4. Improvement and further research

The outcome obtained in the previous application have been satisfactory as the error in steady state resulted less than the expected one and the training method quite effective. Next step is the employ of the neural net during the normal operation of the reactor in order to verify the real performance.

To do this a new campaign of data acquisition has been carried out.

From this campaign emerged that some signals, as originated by the thermocouples, are very close when inverting the control rods position.

In Fig.11 the signal trend from thermocouples n°4 and n°7 is shown, where bleu and green signals are obtained varying the only rod n°1, while red and yellow signals have been obtained varying the only rod n°2.

In Fig.12 the signal trend from thermocouples n°3 and n°5 is shown, where bleu and green signals are obtained varying the only rod n°1, while red and yellow signals have been obtained varying the only rod n°2.
This unexpected difficulty could cause a mistake in the training phase. In order to remedy to this, since the rods can be moved one for time, and since the position of a control rod depends both on thermo-hydraulic conditions and the position of the other control rod, it is chosen to divide the net in two different nets, see Fig.13.

4.1 NN Training

Data validation SHIM1 net
Input data: core temperature (8 thermocouple); fuel temperature; SHIM2 rod position.
Output data: SHIM1 rod position.

Data validation SHIM1 net
Input data: core temperature (8 thermocouple); fuel temperature; SHIM1 rod position.
Output data: SHIM2 rod position

After the training process (42 epochs), the value of the achieved train performance has been 3.6e-28.
The error’s trend is shown in Fig.14.
The outcome of training is displayed in Fig.15 and Fig.17. Fig.15 and Fig.17 shown that the train data match with neural network output. Fig. 16 and Fig.18 shown the rod position percentage error remains low (± 1.5e-4% and ±1e-4%).

Fig.15 – SHIM1 data validation

Fig.16 – SHIM1 percentage error

Fig.17 – SHIM2 data validation

Fig.18 – SHIM2 percentage error
4.2 NN simulation

In order to make the case study concrete several simulations, with different data, from those used for training, have been carried out.
Below two simulations are shown.

Fig.19 and Fig.20 shown the SHIM1 rod data validation and the SHIM1 rod percentage error (± 4%).
Fig.21 and Fig.22 shown the SHIM2 rod data validation and the SHIM2 rod percentage error (± 8%).

The increase of the percentage error is due to the signal noise, see Fig.12
5. Correlated application

In this application the value of the fuel temperature for assigned value of power is predicted. To do this we use a neural network trained with the measured power and fuel temperature.

The used net is feed-forward network trained using supervised learning algorithm (back-propagation with steepest descent).

Training

The training set is formed of couple (Power, Temp.fuel). The structure of neural network is two internal layer and three neuron for level.

The training set is displayed below

Prediction

In the simulation phase the temperature values have been demanded from the net correspondently to the power values which were introduced.

The predicted set is external to the training data range. The percentage error between the simulated data and experimental data depends from the noise of the training set.

The outcome of simulation is displayed below
6. Conclusion

The potential for the application for neural network technology in the process industries is vast. The ability of neural network to capture and model process dynamics and severe non-linear process make them powerful tools in model based control and monitoring. The outcome obtained in this applications have been satisfactory as the error in steady state resulted less than the expected one and the training method quite effective. Testing will continue with increasing of data scanning rate and signal filtering, to improve the answer during status transitions and investigate how to decrease oscillations during the steady-states.

Further activity will concern the application of the method to validation and thermohydraulic prediction in a III+ generation light water nuclear power plant featured with an integral pressurised primary system, where access to CRDM system is physically hampered and rod positioning can be accurately and safely controlled from exterior only.

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ANALYSIS OF THE EFFECT OF INTERNAL FIRE ON THE SAFETY OPERATION OF TEHRAN RESEARCH REACTOR

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ABSTRACT

Internal fires with potential hazards for loss of Tehran research reactor safety functions, which are capable of reactor safe-shutdown or prevent radiation releases above the acceptable limits, are analyzed. The investigations show that fire hazards associated with electrical cable insulation, lubricating oils used for reactor coolant pumps, diesel-driven generators, electrical equipment and carbon as filter of emergency ventilation system may lead to unsafe situations called core damage states which vary with respect to safety systems that these fires have influence on. The occurrence frequency of core damage states after the occurrence of each possible fire scenario in critical fire compartments is evaluated in the present study. As the major step, the overall reactor site is partitioned into distinct closed enclosure surrounded by non-combustible barriers, called fire compartments, and then the location of safety-related components and their associated cables are ascertained. Thereafter, all fire scenarios which may affect those components and cables are identified for each fire compartment. A qualitative screening process is conducted to eliminate compartments in which fire does not cause a safety-related initiating event and consequently, they cannot be a contributor to the risk of fire for the reactor of study. Using internal event PSA logic models, system-specific event trees which model the behaviour of safety systems in the event of internal fires are developed. In order to estimate the frequency of presumed fire scenarios as initiating events, probability that fire ignited in the given fire compartment will burn long enough to cause the extent of damage defined by the fire scenario in that compartment is calculated using detection-suppression event trees. Presumed fires are simulated by CFAST fire modelling software to obtain the activation time of automatic fire detection systems and also time to damage of safety-related components. Applying SAPHIRE 7.0 software, total core damage frequency is calculated $1.697 \times 10^{-3}$ (1/yr) which emphasizes the necessity of improving fire detection and suppression systems of this reactor to reduce the non-suppression probability of each fire scenario and consequently to reduce the occurrence frequency of the initiating events caused by fire.

1 INTRODUCTION

Results of already performed assessments of the consequences of internal fires in many nuclear reactors have shown that fires can be a significant core damage contributor [1]. Although fires can cause consequences for nuclear reactors which range from loss of function
of an equipment and radiation releases to core damage states, within the scope of this study only risk contributions resulting from fires which can initiate core damage are investigated.

Tehran research reactor is a pool-type reactor of TRIGA type. It is designed for use by scientific institutions and universities for purposes such as graduate education, private commercial research, non-destructive testing and isotope production. Fire in the most important compartments of this reactor, i.e., mechanical room, control room, diesel-generator room, pump room, electrical room and fan room are considered in the evaluation.

Both Deterministic and probabilistic approaches can be used for fire risk assessment. However, the probabilistic analysis of fire events and their potential impact on the nuclear safety of a reactor will provide more detailed information on the problems inside a given zone, based on the in-depth analysis of every possible fire origin and the associated fire frequency (Fernandez, 1996).

Using probabilistic models, the possibility of a fire at specific locations; detection and suppression of the fire; the effect of fire on safety-related cables and equipment; the possibility of damage to these cables and equipment; and assessment of the impact of this damage on reactor safety is investigated in this study. This paper represents the results of evaluating the quality of fire protection measures and also safety operation of Tehran research reactor when internal fires occur.

2 METHODOLOGY

For evaluating the frequency of core damage states caused by fire using event tree technique some data such as non-suppression (NS) probability of fire in critical compartments are necessary. These data are produced using the output of CFAST fire modeling software. In this process data needed for determination of critical fire compartments, fire protection systems and fire scenarios of these compartments and also data needed for developing logic models which model the consequences of fire scenarios must be collected. For more details additional discussion below presents a general overview of steps required for evaluating the core damage frequency caused by fire. A schematic diagram of methodology chart and its steps is presented in the figure 1.
2.1 Data collection and assessment

Such analysis relies on the availability of reactor information which is required for fire risk modelling. In the scope of this step the following data are collected:

- A list of fixed and transient combustibles present in each location of the reactor site, which is useful in determination of locations where major fire loads are located. This list contains oil, cable insulation (even those within cabinets or electrical motors), filters, diesel, lubricating oil, etc.
- Fire detection and suppression systems installed in each location of the site.
- A list of electrical equipment, such as pumps, motors, drives, electrical cabinets, etc.
- Cable routings for safety systems based on the location of each safety-related component.
- A list of initiating events, mitigating systems and internal events logic models used in the reactor level 1 PSA, in order to develop a logic model suitable for calculating the conditional unavailability of the required safety systems in the event of an internal fire.
- Familiarization with the components of safety systems, where they are located and how they are operated.

Safety systems are those systems which have mitigating effects on the wide variety of accidents that may occur in the reactor and will maintain the reactor in a safe mode which prevents core damage. Safety systems are generally used to control reactivity, remove heat generated in the core during operation and also decay heat, maintain primary reactor coolant inventory, and protect containment integrity (from the isolation and overpressure point of...
view). In this reactor safety systems are: scram system, emergency electrical power supply system, reactor core cooling system, emergency ventilation system and containment sealing system.

2.2 Definition of critical fire compartments

Once the safety-related components and cables have been identified, it is necessary to ascertain their location within the reactor building and associated buildings which are partitioned into distinct fire compartments. A fire compartment is defined as a closed enclosure surrounded by non-combustible barriers [7]. Based on data collected in the previous step, the overall reactor site is systematically examined to determine critical fire compartments which are defined as compartments at which the safety-related components and their associated cables are located. Critical compartments of the reactor of study are: mechanical room, control room, diesel-generator room, pump room, electrical room and fan room which contain components and cables related to emergency ventilation system, scram system, emergency electrical power supply system, reactor core cooling system, electrical power supply system and containment sealing system, respectively. All fire scenarios which affect those components and cables are identified for each critical fire compartment. Thereafter, a qualitative screening process is conducted to eliminate those locations that do not contain any safety-related component, from further consideration. Fire in such compartments does not cause a safety-related initiating event and consequently cannot be a contributor to the risk of fire for the reactor of study.

2.3 Developing logic models

After defining all fire scenarios which have potential hazards for the reactor safety systems, using internal events logic models, system-specific event trees for calculating the frequency of core damage states (CDS) after the occurrence of each fire scenario are developed. The nodes of such event trees are safety systems necessary to mitigate the effects of the fire caused initiating event. Failure probability of these safety systems was calculated by fault trees which trace back from an undesirable event to any of the possible cause or roots. Figure 2 shows the event tree developed for modelling the consequences of fire occurred in pump room and affect the cables which supply electric power for primary coolant pump.
Figure 2: Pump room fire event tree

Figure 3 shows fault tree used for calculating the failure probability of electrical power supply system as one of the nodes of the above event tree.

Figure 3: Electrical power supply fault tree
In order to calculate the frequency of core damage as a consequence of fire scenario, the frequency of that fire scenario and failure probability of each safety system credited in the event tree must be quantified. The later is calculated by fault trees using data presented in [4, 5, 6], however the former can be estimated by multiplication of the ignition frequency of ignition sources with the probability that fire ignited in a given fire compartment will burn long enough to cause the extent of damage defined by the fire scenario of that compartment. This probability is calculated by detection-suppression analysis which is described in the next section.

### 2.4 Detection and suppression analysis

Evaluating the capability of fire detection and suppression systems installed in each fire compartment, to control the assumed fire in that compartment before safety-related components damage is represented in the risk analysis as the non-suppression probability before target damage occurs. This probability is calculated using an event tree with nodes for fire initiation, fire detection and fire suppression which is called detection-suppression event tree. Based on the fire detection and suppression features (manual or automatic) of each fire compartment, the detection-suppression event tree is developed for each critical compartment. Figure 4 represents fire detection-suppression event tree of pump room.

![Fire detection-suppression event tree of pump room](image)

As shown in this figure some of the branches of these event trees result in non-suppression (NS) end-state, which represents the conditions under which the assumed fire is not suppressed. In order to estimate the probability of fire brigade inability in suppressing the fire assumed in the compartment of study, the assumed fire was simulated by CFAST fire...
modelling software to obtain smoke detectors activation time and the time at which safety-related components defined as targets are damaged. Using Eq. (1), the available time for manual suppression before target damage \(t_{ms}\) is calculated.

\[
t_{ms} = t_{dam} - t_{fr} - t_{det}
\]  

Where

- \(t_{ms}\) is available time for manual suppression
- \(t_{dam}\) is time to target damage
- \(t_{fr}\) is fire brigade response time
- \(t_{det}\) is time to fire detection

This time is an input to the curves which represent the non-suppression probability versus \(t_{ms}\) [7]. The output will be the probability of fire brigade inability in fire suppression.

Experimental information or available documents are used to generate the input data for simulating each fire scenario by CFAST. In this process heat release rate (HRR) is the most challenging parameter to find (Hostikka et al., 2003).

In the case of fire caused by cables and electrical cabinets, different forms of analytical curve are reported for defining the time behaviour of heat release rate. In this work, according to [7], the HRR of cables and electrical cabinets in the growth stage of fire is defined by the following \(t^2\)-curve:

\[
\hat{Q}(t) = \min\left\{ \hat{Q}_{max} \cdot \hat{Q}_{max} \left( \frac{t}{t_g} \right)^2 \right\}
\]  

Where \(t\) is time, \(t_g\) is the HRR growth time and \(\hat{Q}_{max}\) is the maximum HRR. After getting to the peak, HRR will be constant in the peak value for a period of time and then it is assumed to decay in a pattern similar to the growth stage.

In the case of combustible liquids used as fuel or lubricants of the fixed fire sources, such as pumps, diesel generators and transformers, HRR values are assumed constant in the maximum value of heat released in fire they caused. Here, the assumption of a constant HRR is reasonable, since flammable liquid fires are characterized by a rapid growth to its peak intensity.

Quantifying the detection-suppression event tree, results in calculating the non-suppression probability of fire which causes a given initiating event in the compartment of study.

It must be emphasized that fire detection in this reactor is just by two means: the first, smoke detectors which are installed just in some special locations in the reactor building and the second, manual detection where automatic fire detection systems are not installed.
3 RESULTS AND DISCUSSION

3.1 Fire simulation and system modelling results

According to the flammable safety-related components and also fire ignition sources in each critical fire compartment, more than 20 fire scenarios are determined in this reactor. The result of simulating just six possible fire scenarios of them and also non-suppression probability of those scenarios are presented in table 1. The results show the fire vulnerability of critical compartments and provide a quantitative measure of the effectiveness of fire protection systems of this reactor.

SAPHIRE 7.0 software is used to quantify the probability of top events of fault trees developed for calculating the failure probability of safety systems. [4, 5, 6] are used to generate probability of basic events. Consequently, the failure probabilities of safety systems are applied to the event tree headings to calculate the occurrence frequency of each sequence of event trees which evaluate the consequences of fire scenarios.
Table 1: Result of simulating six possible fire scenarios in critical fire compartments of Tehran research reactor (Thermal damage of targets is considered).

<table>
<thead>
<tr>
<th>Critical fire compartment</th>
<th>First ignition</th>
<th>Time to Detect (sec)</th>
<th>Target</th>
<th>Available time for manual suppression (sec)</th>
<th>Probability of fire brigade inability</th>
<th>Non-suppression probability of presumed fire</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mechanical room</td>
<td>Oil</td>
<td>50</td>
<td>1.Cable</td>
<td>450</td>
<td>0.53</td>
<td>0.08</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>2.Electrical cabinet</td>
<td>450</td>
<td>0.53</td>
<td>0.08</td>
</tr>
<tr>
<td>Control room</td>
<td>Cable</td>
<td>300</td>
<td>1.Electrical cabinet1</td>
<td>350</td>
<td>0.13</td>
<td>0.03</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>2.Electrical cabinet2</td>
<td>400</td>
<td>0.10</td>
<td>0.02</td>
</tr>
<tr>
<td>Diesel generator room</td>
<td>Diesel</td>
<td>900</td>
<td>Cable</td>
<td>3850</td>
<td>0.01</td>
<td>0.10</td>
</tr>
<tr>
<td>Pump room</td>
<td>Oil</td>
<td>100</td>
<td>Electric motor</td>
<td>-300(^{1})</td>
<td>1</td>
<td>0.15</td>
</tr>
<tr>
<td>Electrical room</td>
<td>Cable</td>
<td>900</td>
<td>Electrical cabinets which supplies electric power for different parts of reactor</td>
<td>-300(^{1})</td>
<td>1</td>
<td>0.23</td>
</tr>
<tr>
<td>Fan room</td>
<td>Oil</td>
<td>900</td>
<td>Cable</td>
<td>2400</td>
<td>0.03</td>
<td>0.10</td>
</tr>
</tbody>
</table>

\(^{1}\) Minus sign means that target is damaged before the fire is detected ($t_{\text{dam}} < t_{\text{fr}} + t_{\text{det}}$)
With respect to the damage occurred in abnormal situation, four core damage states (CDS) is defined for Tehran research reactor in this fire analysis study. The definitions of each core damage state and also total frequency of these states are presented in Table 2.

### Table 2: Definitions and frequencies of core damage states

<table>
<thead>
<tr>
<th>End States</th>
<th>Definition</th>
<th>Frequency (1/yr)</th>
</tr>
</thead>
<tbody>
<tr>
<td>CDS1</td>
<td>When the reactor can shutdown successfully but there is no primary heat removal</td>
<td>$8.275 \times 10^{-7}$</td>
</tr>
<tr>
<td>CDS2</td>
<td>When the reactor fails to shut down and there is no primary heat removal.</td>
<td>$1.697 \times 10^{-3}$</td>
</tr>
<tr>
<td>CDS3</td>
<td>When the reactor does not shut down in case of reactivity accident, but the primary heat removal system works normally.</td>
<td>$2.098 \times 10^{-8}$</td>
</tr>
<tr>
<td>CDS4</td>
<td>When the reactor does not shut down in case of reactivity accident, and the primary heat removal system also fails.</td>
<td>$2.399 \times 10^{-10}$</td>
</tr>
</tbody>
</table>

**TOTAL** = $1.697 \times 10^{-3}$

### 3.2 Importance analysis

The purpose of the importance analysis is to identify major contributors to core damage frequency that may include accident initiators, system failures, component failures and human errors. Fussell-Vesely measure of importance is used to calculate the contribution of the events in core damage states and to address major weaknesses of the reactor safety system. The Fussell-Vesely measure of importance is an indication of the fraction of the minimal cut set upper bound (or sequence frequency) that involves the cut sets containing the basic event of concern. It is calculated by finding the minimal cut set upper bound of those cut sets containing the basic event of concern and dividing it by the minimal cut set upper bound of the top event (or of the sequence).

Based on the results presented in Table 2, it is evident that the occurrence frequency of core damage state 2 is more than other states. Therefore, it is the most important contributor to total core damage states. A brief result of the importance analysis for CDS2 is shown in Table 3. According to this table, major contributor to the frequency of core damage state 2 is the fire ignites in pump room and causes an electrical damage of the primary coolant pump. This fire scenario is represented as "PUMP ROOM_FIRE1" in the table. Other major contributors to CDS 2 are: short circuit in the electrical panel of control room, fire ignites in pump room and causes mechanical damage of the primary coolant pump (PUMP ROOM_FIRE2) and fire in transformer post, respectively.
4 Conclusion

As a result of this fire analysis in the reactor of study, some conclusions are drawn:

1. Fire protection programs that have been used in this reactor are all based on deterministic approach but this project provides a probabilistic framework for evaluating the effectiveness of fire protection programs in preventing core damage states which are caused by internal fires. Comparing with deterministic approach the probabilistic analysis of fire events and their potential impact on the nuclear safety of a reactor will provide more detailed information on the problems inside a given zone, based on the analysis of each possible fire scenario and the associated frequency of occurrence and consequence.

2. Generally, the non-suppression probability of presumed fires in critical fire zones is relatively high. This fact indicates the importance of fire analysis within the scope of reactor safety analysis.

3. As evidenced by the value calculated for non-suppression probability of presumed fires, fire vulnerability of compartments in which automatic fire detection systems are not installed, such as electrical room or fan room, with 0.23 and 0.10 non-suppression probability of fire, is relatively high comparing with rooms equipped with automatic fire detection systems such as control room.

4. The large value obtained for non-suppression probability of fire in pump room in spite of the fact that this zone is equipped with smoke detectors, can be justified by the special arrangement of large amount of flammable materials in that compartment which affect reactor core cooling system operation.

5. The fact that Tehran research reactor is not equipped with automatic fire suppression systems can be considered as one of the factors which cause the high value of non-suppression probability of fire in some critical compartments. Availability of automatic fire suppression systems will reduce the non-suppression probability of presumed fires and consequently, reduces the fire vulnerability of the reactor.

6. The overall fire-induced core damage frequency for this research reactor was calculated to be $1.697 \times 10^{-3}$ per reactor year. This large quantity emphasizes the necessity of improving fire detection and suppression systems of this reactor to reduce the non-

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Table 3: Importance analysis results for CDS2

<table>
<thead>
<tr>
<th>Event Name</th>
<th>Probability of failure</th>
<th>Fussell-Vesely Importance</th>
<th>Risk Reduction Ratio</th>
<th>Risk Increase Ratio</th>
</tr>
</thead>
<tbody>
<tr>
<td>PUMP ROOM FIRE1</td>
<td>4.400E-003</td>
<td>3.197E-001</td>
<td>1.470E+000</td>
<td>7.164E+001</td>
</tr>
<tr>
<td>MAIN PANEL OF CONTROL ROOM FIRE</td>
<td>2.500E-003</td>
<td>2.162E-001</td>
<td>1.276E+000</td>
<td>8.385E+001</td>
</tr>
<tr>
<td>MAIN PANEL OF ELECTRICAL ROOM FIRE</td>
<td>1.026E-003</td>
<td>1.291E-001</td>
<td>1.148E+000</td>
<td>1.186E+002</td>
</tr>
<tr>
<td>PUMP ROOM FIRE2</td>
<td>7.922E-004</td>
<td>5.627E-002</td>
<td>1.060E+000</td>
<td>7.047E+001</td>
</tr>
<tr>
<td>TRANSFORMER FIRE</td>
<td>1.200E-002</td>
<td>3.481E-002</td>
<td>1.036E+000</td>
<td>3.859E+000</td>
</tr>
</tbody>
</table>
suppression probability of each fire scenario and consequently to reduce the occurrence frequency of fire-induced initiating events.

REFERENCES


Fuel Cycle Safety
ABSTRACT

Nuclear Fuel Plant (FCN) is a facility that produces fuel bundles CANDU-6 type for CANDU nuclear power plant. Like nuclear material in this facility only natural uranium is presented in bulk and itemized form.

Uranium and wastes from the plant are handled, processed, treated, and stored throughout the entire facility. The quantity of uranium presented in the plant in different forms and the activities with zircaloy, beryllium and other hazardous substances, wastes, high temperatures, explosive materials, etc. conducted to special organization for nuclear safety by Nuclear Safety Department (DNS). The industrial safety and security, health of workers, radiological safety, personal dosimetry, decontamination, hygienization, environmental control, nuclear safeguards control, fire extinguishing, emergency and physical protection belong to this department.

In the same time DNS is dealing with the application of national and international legislation, standards and guides in this field, control, improvement, and development of nuclear safety activity in the plant. DNS obtained all the authorizations from the competent authority and perform the instruction and training of staff. The issuing of IAEA safety guide for fuel cycle facility and special for uranium fuel fabrication facility (UFFF) means a new approach and FCN improvement in the future.

1. INTRODUCTION

Nuclear Fuel Plant (FCN) is a subsidiary of National Society NUCLEARELECTRICA SA, a company that has in subordinance another nuclear facility Cernavoda Nuclear Power Plant. FCN is a facility for manufacturing of the nuclear fuel bundles CANDU type with 37 elements, based on natural uranium (0.711% U-235) and depleted uranium (a small quantity with 0.25% U-235 and 0.52% U-235). The annual production is about 10,000 fuel bundles CANDU type that means about 200 tons of natural uranium in UO$_2$. The depleted uranium is processed in campaigns only at the starting of a new unit from Cernavoda NPP.

Different types of nuclear material in many stages and operations of the technological flow are presented in cap. 4.

The personnel working in FCN is about 420 people, and the activity is continuous.

FCN is located at 130 km from Bucharest in Mioveni town on the nuclear site belonging to FCN, Institute for Nuclear Researches (INR) and National Agency for Radioactive Wastes (ANDRAD).

The distances to Cernavoda Nuclear Power Plant for delivery the nuclear fuel bundles is about 340 km and to Feldioara Conversion Plant for supplying the UO$_2$ sinterable powder is 140 Km.
2. NUCLEAR NATIONAL FRAMEWORK

The regulatory body for control of nuclear activities is National Commission for Control of Nuclear Activities (CNCAN) subordinate to Cancellation of Prime Minister. The legislation in this field contains law, norms, regulations, orders, etc.

Law no. 111/1996 for deployment, regulation, licensing and control of nuclear activities, republished in 2006. The law is following in each nuclear area, upon the specific activity, by orders, norms, guides, etc.

For FCN the group of norms with direct application to the specific activity are: norms for radiological security, norms for radioactive mines, norms for quality management, norms for physical protection, norms for nuclear safeguards, norms for management of the radioactive wastes. Romania has ratified the Convention of Nuclear Safety (CNS) through the Law no. 94/24 May 1995, and its reports [2] elaborated by CNCAN in collaboration with the facilities give a comprehensive presentation of the nuclear safety status in Romania.

3. NUCLEAR PLANT LICENSES

The regulatory body issued several licensees for FCN during the period 2008-2010 that cover the activities in nuclear field using radioactive material. The documentations sending by FCN are based on Norms for Radiological Safety and Norms for Radioactive Mining requirements:

a) Possesion, Using, Handling, Processing of the nuclear raw material, Producing of the nuclear fuel bundles, Storage and Supplying

b) For Personnel Dosimetry Laboratory CNCAN has issued an accreditation for 2005-2008

c) For Transportation of Radioactive Material FCN is authorized by CNCAN for the period 2004-2009

d) For Quality Management System CNCAN has issued an authorization during the period 2006-2008

In July 2007 FCN was certified by TUV CERT Turingen for Environmental Management System under ISO 14001

Starting with 01.12.2007 FCN is under IAEA Integrated Safeguards (IS) for its Material Balance Area (RO-D).

Non-nuclear plant licenses are for environmental protection, labour safety, sanitary, working with chemical and hazardous substances.
4. OPERATIONS AND PRODUCTION

The starting point of the nuclear fuel production is **UO₂ sinterable powder** that is transformed in **green pellets** and then in **sintered pellets**. Finally, the sintered pellets are loaded into zircaloy tubes forming the fuel elements and assembled in the last stage in nuclear fuel bundles CANDU type with 37 elements.

The FCN has an area for recycling scraps, non-conforming nuclear material generated during the various operations from technological flow.

In the **Recycling Area** the nuclear material is processed for uranium recovering in wet path using chemical processes. The obtained UO₂ sinterable powder is then input into the flow.

For obtaining the green pellets the UO₂ powder is processed in **Pelleting area** using special equipment and operations: blending conditioning, pre-pressing and pressing.

To obtain the sintered pellets the UO₂ green pellets are sintered at 1700°C in furnaces in **Sintering area**. The obtained UO₂ sintered pellets are grind, washed, dried, sorted and formed the stacks.

The stacks with UO₂ sintered pellets are transferred to the **Assembling area** for loading into zircaloy tubes, and end-caps welding for forming the nuclear fuel elements. 37 fuel elements are disposed on 5 circles for welding with two end-plates, forming the nuclear fuel bundles CANDU-6 type.

The main characteristics of nuclear fuel bundles CANDU-6 type are:
- **total weight**: aprox. 24 Kg,
- **mass of UO₂**: 22 kg
- **mass of natural or depleted uranium**: 19.5 Kg;
- **nominal length**: 495.30 mm;
- **maximum diameter**: 102.49 mm;
- **number of fuel elements**: 37

The maintenance and quality control are assured by plant departments in charge for these tasks.

The production of the plant was about 5,600 nuclear fuel bundles. In 2007 the Unit 2 of Cernavoda Power Plant was commissioned and for this reason the capacity of the plant was doubled at about 10,000 fuel bundles per year. This means that the quantity of uranium throughout the plant, stored and processed, was doubled.

**Table 1**: Table with the main areas that working with uranium, the operations and the hazards

<table>
<thead>
<tr>
<th>AREA</th>
<th>OPERATION</th>
<th>Activity/Products</th>
<th>ASSOCIATED RISKS</th>
</tr>
</thead>
<tbody>
<tr>
<td>RECYCLING</td>
<td>Dissolution</td>
<td>Uranyl nitrate impure</td>
<td>Radiations, Aerosols</td>
</tr>
<tr>
<td></td>
<td>Purification</td>
<td>Uranyl nitrate pure</td>
<td>Radiations, Aerosols</td>
</tr>
<tr>
<td></td>
<td>Precipitation</td>
<td>Ammonium Diuranate – wet form</td>
<td>Radiations, Aerosols</td>
</tr>
<tr>
<td></td>
<td>Drying Granulation</td>
<td>Ammonium Diuranate dry</td>
<td>Radiations, Aerosols</td>
</tr>
<tr>
<td></td>
<td>Reduction</td>
<td>UO₂ powder</td>
<td>Radiations, Aerosols</td>
</tr>
<tr>
<td></td>
<td>Homogenization</td>
<td>UO₂ sinterable powder</td>
<td>Radiations, Aerosols</td>
</tr>
</tbody>
</table>

Radiological Hazards

Non-Radiological Hazards

Radiations

Aerosols

Nitric acid, temperature

Kerosene, Tributyl-phosphate, temperature

Ammonia, temperature

Temperature, water vapors

Hydrogen, nitrogen, temperature
### 5. ORGANIZATION of NUCLEAR SAFETY in FCN

The top management of FCN focused the major priorities of plant to quality control, nuclear safety and environmental protection. The system applied by management is **Integrated Management System (IMS)** including the three components: **Quality Management System**, **Environmental Management System**, **Labour Health and Safety System** all based on international standards: ISO 9001:2005; ISO 14001:2005; ISO 18001:2006. In order to secure a special place for nuclear safety, the management has proposed for a new organizational scheme of FCN.

Starting with 01 October 2007 FCN has a new organization scheme with a new department dedicated to nuclear safety, Department of Nuclear Safety (DNS) reporting to FCN Director.

The main DNS tasks are to apply the FCN policy, the SNN-SA nuclear safety policy [2] and the Romanian nuclear safety policy [1]. These are:

- To protect employees
- To protect visitors and contractors
- To protect population
- To protect environment
- To assure security and safety for nuclear material, radioactive sources, installation and goods

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<table>
<thead>
<tr>
<th>Process</th>
<th>Description</th>
<th>Radiations</th>
<th>Aerosols</th>
<th>Chemicals</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>PELLETIZING</strong></td>
<td>Blending and compaction</td>
<td>Presable UO2 powder</td>
<td>Radiations</td>
<td>Zinc stearate</td>
</tr>
<tr>
<td></td>
<td>Pressing</td>
<td>Pressing UO2 powder in green pellets</td>
<td>Radiations</td>
<td>Aerosols</td>
</tr>
<tr>
<td></td>
<td>Sintering</td>
<td>Transforming the green pellets in sintered pellets</td>
<td>Radiations</td>
<td>Aerosols</td>
</tr>
<tr>
<td></td>
<td>Grinding</td>
<td>Shaping of the UO2 sintered pellets</td>
<td>Radiations</td>
<td>Aerosols</td>
</tr>
<tr>
<td></td>
<td>Washing and drying pellets</td>
<td>Sintered Pellets</td>
<td>Radiations</td>
<td>Aerosols</td>
</tr>
<tr>
<td></td>
<td>Column forming</td>
<td>Sintered Pellets</td>
<td>Radiations</td>
<td>Aerosols</td>
</tr>
<tr>
<td><strong>ASSEMBLING</strong></td>
<td>Zircaloy components</td>
<td>Zircaloy elements covered with beryllium</td>
<td>Radiations</td>
<td>Aerosols</td>
</tr>
<tr>
<td></td>
<td>Washing cleaning</td>
<td></td>
<td>Radiations</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Coating with graphite</td>
<td>Zircaloy tubes coated inside with graphite</td>
<td>Radiations</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Loading pellets in sheaths</td>
<td>UO2 Sintered Pellets and Nuclear Fuel Elements</td>
<td>Radiations</td>
<td>Aerosols</td>
</tr>
<tr>
<td></td>
<td>End-cap welding</td>
<td>Elements and Nuclear Fuel Bundles</td>
<td>Radiations</td>
<td>Aerosols</td>
</tr>
<tr>
<td></td>
<td>End-plate welding</td>
<td>Elements and Nuclear Fuel Bundles</td>
<td>Radiations</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Wrapping Packing</td>
<td>Nuclear Fuel Bundles</td>
<td>Radiations</td>
<td></td>
</tr>
</tbody>
</table>
- To assure security and safety for proper pertinent information
- To promote the policy of health, safety and security
- To train employees, visitors and contractors for nuclear safety issues
- To inform the public

5.1. DNS ORGANIZATIONAL SCHEME

DNS is divided into three services for fulfilling the tasks regarding nuclear safety:
- labour safety service (LSS) to apply the safety of labour (industrial safety, chemical noxious, noise, light, etc.) and to survey the health of workers
- radioprotection and nuclear safeguards service (RNSS) for dosimetry of personnel, monitoring of work places, nuclear safeguards control, environmental protection and radioactive wastes management
- information and physical protection service (IPPS) for emergency control and preparedness, physical protection, security, fire prevention and informational protection

5.2. RADIOLOGICAL ASPECTS

5.2.1. Exposure of workers
In FCN nuclear material is presented in bulk form (ammonium diuranate, powder, pellets) or itemized form (trays, grids, elements, bundles). In the same manner the liquid and solid presentation of nuclear material can contribute to the exposure of workers.

1. External Exposure
The doses arising from the presence of nuclear material based on natural uranium are not high.

The monitoring is performed with TLD (Termoluminiscent Dosimeter) PANASONIC type. Changing of TLDs, measurement, interpretation of doses, assessment, irradiation of TLDs, and reports to sanitary authority and CNCAN are done by DNS-RNSS.

FCN Laboratory for Personal Dosimetry has the accreditation from CNCAN for this type of activity. The TLDs are posted on the racks in different areas of the plant with TLDs quality control and TLDs for local background.

Figure 2: One panel with TLDs
2. Internal Exposure. 

Being a facility where nuclear material is processed and handled in bulk form, taking into account the contamination hazard of air and surfaces, the **internal exposure** is very important and must be treated carefully.

The employees working in **controlled area** with bulk material like: pure and impure uranyl nitrate, uranyl phosphate, ADU powder, UO2 powder, slurries, uranium ashes, etc. are designated to submit periodically the urine for uranium analysis. Similar analysis is performed for beryllium in urine for that part of workers.

3. Total doses. 

By summing the external dose and internal dose the total dose is obtained which is compared to dose effective limit of 20 mSv/year. 

The average total dose of FCN exposed workers is 2.7 mSv/year, and the collective dose is about 1 man Sv.

The individual dose and collective doses for the last 5 years in NFP are shown in next table:

<table>
<thead>
<tr>
<th>Year</th>
<th>Individual Dose [mSv]</th>
<th>Collective dose [man Sv]</th>
</tr>
</thead>
<tbody>
<tr>
<td>2003</td>
<td>2.461</td>
<td>1.038</td>
</tr>
<tr>
<td>2004</td>
<td>2.611</td>
<td>1.099</td>
</tr>
<tr>
<td>2005</td>
<td>2.745</td>
<td>1.161</td>
</tr>
<tr>
<td>2006</td>
<td>2.172</td>
<td>0.921</td>
</tr>
<tr>
<td>2007</td>
<td>2.442</td>
<td>1.042</td>
</tr>
</tbody>
</table>

4. Contamination Monitoring. Direct measurements (total contamination) and indirect measurements (loose contamination) on working surfaces and objects leaving the controlled area are performed daily according on Radioprotection Plans for Surfaces Contamination Control. 

The control of two areas is performing to assure the good functioning of the areas as well as the barriers and for preventing the spread of contamination.

5.3. FIRE PROTECTION and PREVENTION

The organization for Fire Prevention and Protection in FCN is very important because of the activities with hydrogen, kerosene, equipment that works at high-temperatures, and storages where the fuel bundles are kept. All these areas are provided with smoke detectors. The measures that were taken for keeping under control the zircaloy components are very important for FCN because zircaloy in different forms is very inflammable. In the same category there are some substances from recycling area: kerosene, nitric acid, ammonia. 

The activities are procedured. The procedures are designated for each work place involving activities with fire hazard.

The interventions are assured by Fire Extinguishing Team of INR, local formation and national authority.

5.4. RISKS of EXPLOSIONS

The Hydrogen Station that produces hydrogen for reduction and sintering furnaces and the tanks where the hydrogen is kept are carefully surveyed, because of the possibility for explosions. The furnaces are monitored for hydrogen leakage that should be a cause of explosion by gas accumulation.
5.5. ENVIRONMENTAL PROTECTION and MONITORING

FCN compartment in charge of environmental protection is named Collective of Environmental Control (CEC) and it is performing:
1. Control of Radioactive Liquid Effluents (RLE) discharging to the Purification Station of Institute for Nuclear Researches (INR)
2. Control of Radioactive Gaseous Effluents (GRE) emissions in air
3. Control of generated, collected, stored, and final disposed of solid radioactive wastes
4. Control of transferring of liquid radioactive wastes category scrap from FCN to Station of Treatment of Radioactive Wastes from INR.
5. Control of soil and vegetation contamination
6. Control of the outdoors contamination
7. Control of outdoors doses
8. Control of the non-radiological environmental factors

5.6. HEALTH SURVEILLANCE

The health surveillance is the task of DNS, specially Labour Safety Service. All the employees have proper Medical File from the hiring until the retirement. A medical office for first aid is in main buildings. The medical emergencies are treated at the INR medical office or in the Mioveni Town Hospital.

The file is connected with Radiological Exposing Historical File that is kept in FCN Radioprotection Archive for 30 years.

5.7. EMERGENCY PREPAREDNESS and PLANNING

5.7.1. Internal emergencies

Radiological Anomalies and Emergency

The postulated events are in small scale of INES categorized like anomalies the spreading being in local dimension:

a) Cracking and spreading the uranium solution from the tanks
b) Turning of a container with UO2 powder
c) Turning of a carriage with UO2 sintered pellets

The postulated events involving a fire which can affect outside perimeter of FCN
a) Fire at the sintered and reduction furnaces
b) Fire at the final storage for nuclear fuel bundles

Non-radiological emergency

a) power supply interrupted – a second supplier for power is in function and it is tested periodically
b) supply with water and other utilities.

This activities is provided with instruction for intervention when appears one of more of this situation

For natural events: calamities like tornadoes, torrent rains, flooding, snow falls, extreme temperature earthquake, the possibility is very reduced due to the site stability. Events involving human factors like: airplane crashing, terrorism acts, etc. are taken into account with the entire site emergency plan

These events are in direct relationship with the INR and ANDRAD and the local, regional and national organisation for emergency situation
5.7.2. External emergencies
a) On-site emergency due the FCN activity
   - A large fire or an explosion
b) On-site emergency due the Institute activity
   - Radiological emergency
   - A large fire on the site or in vicinity of site

5.8. QUALITY ASSURANCE

5.8.1. Procedure of elaborating and control of nuclear safety documents
All the nuclear safety activities were included into the quality system of FCN. Measures were taken so that procedures and documents for production, maintenance, quality to contain nuclear safety requirements. The internal and external audits were performed each year with the aim to help the improvement of nuclear safety and to implement the elements of nuclear safety in documents with direct application in FCN activity.

The initiative to have a FCN Radioprotection Archive integrated into the General Archive of FCN is a very important step to have a history of:
- external exposures of FCN personnel in the radiation fields
- internal exposures of workers that working with uranium in bulk
- airborne contamination at the work places
- dosimetric measurements of the contaminated surfaces,
- gamma doses at the work places,
- radioactive wastes measurements, and
- environmental factors

5.8.2. Audit
Until this moment some of the internal audits performed in FCN by qualified auditors with the CNCAN assistance are focused on the nuclear safety process. In 2008 Department of Quality Management and Audit (DQMA) performed three audits on DNS activities:
- audit for environmental protection and management
- audit for physical protection and plant security
- audit for health of occupationally workers, monitorizing the workers and training the personnel

The conclusions of nuclear safety audits are directly involved in the nuclear safety activity improvements.

In the future we are expecting to be assisted by IAEA SEDO Mission (Safety Evaluation During Operation) and to use the conclusions of the mission to increase the level of nuclear safety in FCN. Thus the application of [3] and [4,5] after officially issued would be very useful for us.

5.9. TRANSPORTATION of RADIOACTIVE MATERIAL

FCN is licensed by CNCAN and Romanian Roads Authority for transportation the nuclear material like fuel bundles, UO2 powder, non-conforming material, radioactive wastes (category of very low activity). The procedures are prepared for safety and security of all transportation activities.

Additionally, FCN crew is escorted for every shipment by guard team, special trained for this type of transportation.
From 1996, over 100 transportations of fuel bundles (1.3 millions tons of natural and depleted uranium) to Cernavoda Nuclear Power Plant were performed. Thus, about 100 transportations of UO₂ powder and other kinds of transportations were performed by FCN without incidents.

Figure 3: The truck for radioactive material transportation

5.10. NON-PROLIFERATION and SAFEGUARDS

Romania has signed in 1972 the Treaty and Non-Proliferation and in 1999 the Additional Protocol. FCN Material Balance Area in IAEA is coded RO-D and WRMD in EURATOM evidence. Starting with 01 December 2007 RO-D passed from Classic Safeguards (CS) to Integrated Safeguards (IS) having a direct mail box for daily transmitting to the IAEA Safeguards Departments. Each year a physical inventory of uranium is checked and a variable number of random inspections by IAEA and EURATOM are performed.

Figure 4: The storage of UO₂ sinterable powder
Due the working with natural and depleted uranium the probability and possibility of criticality situation doesn’t exist.

Assessment of the uranium inventory from the last years, mass of natural and depleted uranium existed in FCN vary from 45-70 tons with perspective to decrease because the transformation in fuel bundles and shipments to Cernavoda Nuclear Power Plant, Unit 1 and Unit 2.
The perspective for doubling the actual capacity and passing to the production of 20,000 fuel bundles per year (forecasted for 2013) doesn’t mean the increase of the uranium mass due to the permanent transfer to Units 1, 2, 3, and 4.

5.12. PHYSICAL PROTECTION and PLANT SECURITY

FCN perimeter is with the Pitesti nuclear platform including the Institute of Nuclear Researches (INR) and the National Agency for Radioactive Wastes (ANDRAD). FCN has several barriers to assuring the security of the nuclear equipment, installation and nuclear material: platform’s fence with security guard, security of proper perimeter, buildings and rooms.

Along of FCN fence, 24 video cameras are placed to monitor the plant perimeter 24 h per day.

Inside the plant buildings there are 14 video cameras for surveillance the entrances from surveyed zones to controlled zones and the working areas.

![Figure 8: The fence of NFP with video camera](image)

The plant entrance where a room for physical protection is placed is provided with detecting portal for nuclear material, radioactive substances and radioactive sources. It is shown in the next picture:

![Figure 9: The main entrance in NFP with detecting portal](image)
5.13. PHYSICAL PROTECTION and SECURITY of the STORAGES for NUCLEAR MATERIAL

Each storage with nuclear material (fuel bundles, UO2 powder, radioactive wastes), is under surveillance of physical protection personnel and it’s provided with alarm, locks and proximity card. The similar measures are taken with the storages for zircaloy, beryllium and chemical substances. The Fig. 4 and 7 show the UO2 powder storage and Fresh Nuclear Fuel Bundles storage.

5.14. SAFETY CULTURE, TRAINING of EMPLOYEES

FCN has issued at start of each year a document titled: *Programme for instruction and training*. Instruction and training for nuclear safety are divided into more fields, each of them having proper plans and trainers:

1. Radiological Safety
2. Labour Safety
3. Emergency Situations
4. Environmental Protection

After each annually or bi-annually instruction a test with multiple choices questions (20-40 questions) is applied for verifying the knowledge of the Romanian legislation in these fields, principles and practices aspects.

A number of 18 FCN employees is certified by CNCAN for possessing the permit level 2 for working in the nuclear field for domain Nuclear Raw Material – Fuel Elements Fabrication and Radioactive Material Transportation.

5.15. DECOMMISSIONING PLANNING

The FCN has elaborated a preliminary decommissioning plan dealing with the main activities: removing the nuclear material and equipment cleaning, disassembling of installation, dismantling, segregation of radioactive wastes from non-radioactive wastes, demolition, dosimetric measurement, radioactive wastes characterisation, disposal of radioactive wastes, restauration after dosimetric measurements.

6. PUBLIC INFORMATION

For public information FCN has issued a leaf sheet that presents the activity of the plant including the forecasting and developing for more years.

REFERENCES


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