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Fuel cycle
LIFE CYCLE ASSESSMENT OF THE EPR: THE IMPACT OF DIFFERENT FUEL CYCLE STRATEGIES

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ABSTRACT
Using Life Cycle Assessment (LCA) methodology this paper analyses and compares the complete fuel cycles of current (Gen II) and future (Gen III) nuclear reactor operations in order to determine the contributions to greenhouse gas (GHG) emissions from the individual stages of the nuclear energy chains. The research focused on electricity generated by European Pressurised Reactors (EPR), here assumed representative of a Gen III technology, located in both Switzerland and France in the year 2030. These are compared to electricity from current Pressurised Water Reactor (PWR) technology in Switzerland as well as a range of future fossil and renewable generating systems. The three nuclear plants operate under different fuel cycle strategies which provide diverse opportunities for comparison. The LCA shows a reduction of GHG emissions by 25% for the EPR in Switzerland in 2030 compared to the current PWR cycle which emits just below 6 g(CO2-eq)/kWh. Due in part to differences in cooling technologies which affect output capacities the French EPR is expected to achieve marginally better reductions of almost 30%. Over half of the reductions from 2005 to 2030 are attributable to the EPR’s higher generation capacity over a longer lifetime which implies a lower material intensity. This is further assisted by a higher burn-up of the fuel as well as marginally higher thermal conversion efficiency. The different strategies result in only small overall variations in GHG emissions between the assessed EPR cycles. In general, the nuclear scenarios show very good GHG emission performance compared to the non-nuclear technologies used in the comparison, which represent advanced future concepts for both centralised and dispersed electricity generation.

1. Introduction
Nuclear energy is among the main topics in today’s energy policy debate and considered by several European countries to be a central element of their climate change mitigation and energy security strategies. Such strategies include the replacement of existing nuclear reactors approaching the end of their service life-time as well as the expansion of nuclear capacities. Within Europe current new nuclear construction projects are employing the Generation III (Gen III) European Pressurised Reactor (EPR) design. This is larger than currently operating Gen II reactors, has a designed operational lifetime of 60 years and incorporates significant advancements in safety and reliability. Further improvements include increased efficiency and ability to use a higher percentage of recycled fuel [1]. Results are set in context by the comparison with electricity from a current Gen II pressurised water reactor (PWR) and electricity from a range of other non-renewable and renewable technologies.

2. Goal, scope and methodology
Life Cycle Assessment (LCA) forms the basis for the analysis of complete energy chains, covering extraction of resources, processing and transport of materials and energy carriers,

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1 Developed LCA of nuclear cycle when formerly researcher at PSI; this work served as background for the paper; current affiliation: Resun AG, Rain 41, CH-5001 Aarau, Switzerland.
conversion of fuels, as well as waste disposal. These are modelled and analysed for all single systems used in the technology comparison. The development of technologies envisaged to be operational by 2030 is focused on the power plants themselves rather than on advances in other parts of the fuel chains. An exception to this is the assumption of enrichment purely via centrifuges instead of a quota from diffusion technology which still currently contributes to the worldwide enrichment services. An average European electricity mix in 2030, based on [2], was defined and used in the LCA calculations in order to reflect a business as usual development of the economy. Except for this electricity mix, ecoinvent data v2.0 [3] was used as background life cycle inventory data and reflecting conditions in the year 2005. As a means of analysing the nuclear fuel cycles and with a focus on climate change mitigation, this paper analyses the differences in greenhouse gas (GHG) emissions, measured in terms of their equivalent global warming potential to CO₂ over a period of 100 years [4], caused by 1kWh of electricity generated in future EPR’s. In reporting only the GHG emissions it is recognised that this short paper does not consider the wide range of impact indicators available to an LCA study which, in turn, can form a vital aspect for a more holistic sustainability assessment and multi criteria decision analysis (see: [5-8]).

3. Technology characterisation

The nuclear reactor technologies used in the assessment reflect current (operating in 2005) and future (expected to be operating in 2030) technologies which employ different fuel cycle strategies as modelled by [7, 8]. Main features of each reactor technology and fuel cycle are given in Table 1. The current technology is based on the average 1000 MW class Swiss (CH) PWR (Gen II) [9]. In this scenario, with 8% MOX fuel elements, 60% of fuel elements are assumed to be enriched to 4.2% via diffusion in a French facility and 40% in centrifuges at a European facility (URENCO) throughout the whole lifetime (using 40 years). The adjacent river water is used in an open cycle direct cooling system. Changes in the policies of reprocessing facilities and a memorandum in Switzerland in 2006 rejecting further reprocessing led to the adoption of an open fuel cycle without reprocessing meaning that the life cycle of the currently operating PWR combines periods of both “open” (60%) and “closed” (40%) fuel cycles (share calculated on a lifetime basis). The analysis therefore assumes that the further operation of the PWR will be within an open fuel cycle and that this policy will be continued for the complete lifetime of a Swiss EPR. For the EPR’s (Gen III) in Switzerland and France all fuel is assumed to be enriched in modern, European located centrifuges which are operated on an average European electricity mix. An EPR in Switzerland will most likely have to operate with a tower-based closed hybrid cooling system due to river temperature restrictions and landscape/aesthetical motivations which leads to a slightly lower overall net capacity in comparison with the French EPR with wet cooling tower. All spent fuel from the French EPR is assumed to be reprocessed. For both EPR scenarios fuel enrichment and burn-up are conservatively estimated with 4.8% and 60 MWd/kg, respectively. The research also assumed constant natural uranium concentrations and accessibility of the mined ore bodies through to 2100. The same repository concepts for disposal of radioactive waste are used for both the Swiss and the French nuclear chains, based on the Swiss disposal strategies [9]. However, these have been demonstrated to have a relatively small impact of the overall LCA burdens of the fuel cycle (see Fig.1).

<table>
<thead>
<tr>
<th>Location</th>
<th>Reactor type</th>
<th>Capacity (MWₑ)</th>
<th>Electric eff. (%)</th>
<th>Lifetime (yrs)</th>
<th>Burn-up (MWd/kg)</th>
<th>MOX (%)</th>
<th>Cooling strategy</th>
<th>Fuel cycle</th>
</tr>
</thead>
<tbody>
<tr>
<td>CH</td>
<td>PWR (Gen II)</td>
<td>1000</td>
<td>32</td>
<td>40</td>
<td>42.8</td>
<td>8</td>
<td>River</td>
<td><em>Mixed</em></td>
</tr>
<tr>
<td>CH</td>
<td>EPR (Gen III)</td>
<td>1500</td>
<td>33.8</td>
<td>60</td>
<td>60</td>
<td>0</td>
<td>Hybrid tower</td>
<td><em>Open</em></td>
</tr>
<tr>
<td>FR</td>
<td>EPR (Gen III)</td>
<td>1530</td>
<td>33.8</td>
<td>60</td>
<td>60</td>
<td>13</td>
<td>Wet tower</td>
<td><em>Closed</em></td>
</tr>
</tbody>
</table>

Table 1 Main relevant characteristics of the nuclear technologies and fuel cycles.

4. Intra nuclear comparison

In Figure 1 the contributions to GHG emissions per kWh from each stage of the nuclear fuel cycles as well as construction, disposal and operation of the power plants (PP) are shown.
With regard to the total GHG emissions per kWh it can be seen that the Swiss PWR operating today causes 32% and 41% higher emissions than the Swiss and French EPR’s assumed to operate in 2030, respectively. Reduced contributions from construction and decommissioning of the power plant itself have a large part to play in this due to the much higher lifetime generation capacities of the EPR’s. This difference between the Swiss PWR and the Swiss EPR accounts for 56% of the overall reduction from 5.8 to 4.4 g(CO₂-eq.)/kWh.

Figure 1 GHG emissions at the different fuel and life cycle stages of the nuclear chains.

The interaction between power plant attributes and the percentage of MOX fuel elements used causes the GHG emissions from uranium mining and processing for the Swiss EPR fuel to be slightly lower than for the Swiss PWR. Using 13% MOX fuel in the case of the French EPR provide further reductions at this fuel cycle stage; the same degree of reductions are reflected through to the conversion stage.

For the Swiss PWR 60% of the fuel is enriched via diffusion using French nuclear power and 40% via centrifuge using the UCTE mix. The almost CO₂-free nuclear electricity and the heavily fossil UCTE mix explain why even with 60% diffusion enrichment – being much less energy efficient than centrifuge enrichment – the GHG emissions for this stage of the PWR fuel cycle are only slightly higher than those of the Swiss EPR. All the fuel for this plant is enriched by centrifuge technology using the 2030 UCTE electricity mix which is still expected to be based largely on fossil fuels. In order to appreciate the sensitivity of this aspect of the fuel chain on overall GHG emissions a comparison of using a specific electricity supply for enrichment of the Swiss PWR fuel in either a centrifuge or a diffusion plant was conducted. The electricity was assumed to be from an average of currently operating hard coal power plants from nine European countries (meaning higher GHG emissions per kWh than the electricity mixes used in Figure 1, as an input for enrichment). This showed that enriching at a modern centrifuge plant the quantity of fuel required for 1 kWh electricity generated by the Swiss PWR would contribute 0.61 g(CO₂-eq) to the overall emissions from the entire fuel cycle. On the other hand, the same fuel enriched in a diffusion plant and with the same coal based electricity supply would cause 38.6 g(CO₂-eq), in both cases as a result of electricity supply to the enrichment process alone.

The so-called “downstream” stages of the spent fuel (conditioning, reprocessing, and waste storage) very strongly reflect the differing regimes governing its further processing. The French EPR reprocesses all spent fuel and this stage therefore shows higher emissions than the Swiss EPR without reprocessing and the Swiss PWR for which only 40% of total spent fuel is reprocessed (all pre-2006). Conditioning the remaining 60% and less completely burnt-up fuel from the PWR shows only marginally lower GHG emissions than the conditioning of all spent fuel from the Swiss EPR.
The analysis revealed that disposing of spent fuel and high level waste generates nearly 17 tonnes of GHG emissions per m³ whereas low and intermediary level waste is in the order of 2 tonnes per m³. Although the French fuel cycle reprocesses all spent fuel, Figure 1 shows a higher amount of GHG emissions due to radioactive waste treatment and disposal than the Swiss EPR fuel cycle – which sends all spent fuel directly to final repository after conditioning – and reflects the larger volumes of high level waste generated by a closed fuel cycle. Actual operation of the power plants is then one of the smallest contributing factors to total emissions and seen to be very similar across all three scenarios.

5. Comparison with other technologies

Having looked specifically at the nuclear fuel cycle it is important to place the GHG emissions within the context of alternatives. In addition to the three nuclear scenarios already described, the following comparison includes a range of technologies determined by [7, 8] as likely to be operational in 2030.

<table>
<thead>
<tr>
<th>Technology</th>
<th>Description</th>
<th>Capacity</th>
</tr>
</thead>
<tbody>
<tr>
<td>Biogas combined heat and power (CHP)</td>
<td>Agricultural slurry in anaerobic fermentation; 3% diesel fuel additive; allocation: exergy. Location CH, lifetime 15 yrs</td>
<td>100kW (electricity)</td>
</tr>
<tr>
<td>Photovoltaic (PV) multi-silicon crystaline</td>
<td>Roof-top installed system with cell efficiency of 19.8%. Location CH, lifetime 30yrs</td>
<td>3kW peak 850kWh/(kWpyr)</td>
</tr>
<tr>
<td>Wind (Offshore)</td>
<td>30km offshore in a park of 50 turbines. Average wind speed at nacelle 10m/s. Location DK, lifetime 20 yrs.</td>
<td>20MW; 4000 full load hrs/year</td>
</tr>
<tr>
<td>Geothermal</td>
<td>Enhanced geothermal system: heat extraction at 5500m and 230°C. Organic Rankine Cycle turbine. Location CH, lifetime 30 yrs.</td>
<td>36MW (electricity)</td>
</tr>
<tr>
<td>Hydro (River)</td>
<td>Run-of-river using 14.6m fall. Kaplan turbines. Location CH, lifetime 80 yrs.</td>
<td>51.3 MW 306 GWh/yr</td>
</tr>
<tr>
<td>Integrated Gasification Combined Cycle (IGCC) plant</td>
<td>Hard coal fuel (19.4 MJ/kg). 51.5% electric efficiency. Location Germany, lifetime 30 yrs.</td>
<td>450 MW</td>
</tr>
<tr>
<td>Natural gas Combined Cycle (CC) plant</td>
<td>Base load operation at 63% electric efficiency. Location CH, lifetime 25 yrs.</td>
<td>500 MW</td>
</tr>
</tbody>
</table>

Table 2 Technologies used in the comparison

![Figure 2](image)

Figure 2 gives the GHG emissions for all technologies selected and shows that, relative to electricity from the fossil fuels of coal and gas, all other alternatives have a relatively similar performance. Looking more closely however, the effect on emissions of higher infrastructure demand relative to plant capacity in comparison to the centralised nuclear plants becomes evident. The production of electricity from biogas causes fermentation emissions of N₂O which has a global warming potential 298 times that of CO₂ [4]. GHG emissions occur in the
life cycle of PV from the manufacture, processing and transport of components. With increasing unit capacity and placement in areas of suitably high average wind speed the emissions from wind generated electricity are expected to become increasingly similar to that of nuclear power. Although functioning as a base load power plant, drilling to the depths required and the use of steel to line the wells causes electricity from geothermal to be relatively high in comparison to other base load plants, except for coal or natural gas. With a much longer operational lifetime, it's ability to operate at full load for 68% of that time and the natural availability of the primary energy source the hydro power plant shows the lowest GHG emissions of all alternatives and which is around 35% lower than for the French EPR.

6. Conclusions
Life Cycle Analysis (LCA) of the fuel cycles and power plant facilities of European pressurised reactors (EPR) hypothesized, as representative Gen III technology, to operate in Switzerland and France in 2030 showed a relatively important reduction of greenhouse gas (GHG) emissions compared to current technology and allowed a deeper understanding of the influence of both open and closed fuel cycles on GHG emissions. For reference these EPR’s were compared to a currently operating Swiss pressurised water reactor (PWR) which has operated for approximately 40% of its lifetime (in terms of overall electricity generation) within a closed fuel cycle with reprocessing of spent fuel and is expected to operate for the remainder of its lifetime within an open one without reprocessing. In comparison with alternative renewable and fossil electricity technologies the GHG emissions from all three nuclear scenarios with emission factors between 4.1 and 5.8 g(CO2-eq)/kWh are very low, bettered only by a run-of-river hydro plant. Although large reductions in GHG are therefore difficult to achieve, the increase in capacity and operating lifetime of the EPR compared to the current PWR accounted for more than half of the emissions reductions between the current and future Swiss plants. The GHG emissions associated with the fuel chains are then more influenced by the electricity sources assumed for processing new and spent fuels than the differing fuel cycle strategies themselves. It is appreciated that this short paper has only been able to analyse the nuclear energy chains using one, albeit critical, environmental impact indicator and the results are therefore not representative of a broader and balanced assessment of electricity generation technologies. Results from a wider range of indicators and showing how the rankings of technologies can change accordingly are available in [5-8].

References
ABSTRACT

The Joint Research Centre (JRC) is a Directorate of the European Commission, distributed over sites in Belgium, Germany, Italy, the Netherlands and Spain. Its mission is to provide customer-driven scientific and technical support for the conception, development, implementation and monitoring of EU policies. Peripheral to its fundamental mission are the Decommissioning and radioactive Waste Management (D&WM) activities associated with the JRC nuclear installations constructed and operated during the early years of its research sites. Since the 1980s, the JRC evolving mission has progressively reduced the need for nuclear installations, that must now be decommissioned.

The removal of nuclear material from an installation is a prerequisite to start the decommissioning. This paper reports the successful experience at the JRC Italian site (Ispra) on the preparation, packaging and transfer of Plutonium (PuO$_2$) and Mixed Oxides (MOX) reference materials used for over twenty years, from the Ispra PERformance LABoratory (PERLA) to the original owner abroad. In particular the need for design, licensing and construction of new glove box lines and auxiliary tools as well as the safety provisions for material containers handling and transport are described.

1. Introduction

The PERformance LABoratory (PERLA) of the Joint Research Centre (JRC) in Ispra, Italy, houses an extensive collection of well-characterised nuclear reference material and instrumentation. The activities are specifically oriented to the development, validation and implementation of nuclear NDA techniques and to the training of the Euratom Safeguards Office (ESO) and International Atomic Energy Agency (IAEA) nuclear inspectors.

The laboratory has made use for about 20 years of Plutonium (PuO$_2$) and Mixed Oxides (MOX) reference material, named in the following text as "models" and "pellets", loaned by a company based outside Italy. In the frame of an agreement on the return to the owner company, in 2009 the JRC has concluded a project for the preparation, packaging and transfer of the mentioned reference material to a reprocessing plant prior its return to the owner’s country. The evacuated material corresponded to the major part of the Plutonium hosted on the Italian territory and has caused a dramatic reduction of the JRC Ispra radioactive inventory as well as considerable savings in terms of safety and security costs. The entire activity has been driven by tight deadlines imposed by organizational, technical and legal frameworks of the involved stakeholders and countries.

2. The PERLA laboratory

The PERLA laboratory is part of a JRC nuclear complex named INE (Essor Nuclear Installation). INE houses various labs, a shut-down research reactor, as well as fresh and spent fuel storage areas. PERLA is made of three main areas: the reference material storage, the benches areas for NDA techniques, the two-glove box room designed for preparation, decontamination and emergency interventions on the material. The PERLA
nuclear license provisions are strictly linked to those of INE, therefore any structural improvement required in PERLA implies a complex and time consuming change in the INE licensing status.

3. Characteristics of the reference material
3.1 Models
The models are 59 double tight-proof stainless steel containers, and contain PuO₂ or MOX powder at various burn-ups, for a total of a few tens of kg of Pu. The models are cylindrical and made of an inner container and an outer container. The inner container is made of stainless steel (for the PuO₂ models only, an additional glass vessel is present inside the inner container), contains the material powder, is closed by a non-leak proof screw tap and is enclosed in a sealed PVC bag. The outer container is made of stainless steel, has a leak proof closure system with a metallic o-ring and 8 screws or a single screw tap (Model 5 only). Fig. 1 shows the model types. The containers leak tightness has been successfully tested for several years. The outer containers have been opened only once, at their arrival from the owner's premises to have the original polyurethane (PU) bags wrapping the inner containers changed in polyvinylchloride (PVC) bags, as a result of a provision by the Italian nuclear safety authorities. The models gamma and neutron contact dose rates and the outer surface temperatures in the ventilated storage can reach, respectively up to 6 mSv/h and ca. 40°C, thus requesting special protective means when handled.

![Fig. 1: Models for PuO₂ and MOX powder (left to right, Model 5, Model 200, Model 1000, Model 2500)](image1)

3.2 Pellets
MOX pellets are contained in 16 stainless steel or Zircalloy short rods (total max length 0.5 m) and 17 long pins (total max length 2.5 m), equivalent to a total of a few kg of U and less than one kg of Pu. The contact dose rates are up to 4.8 mSv/h (Fig. 2).

![Fig. 2: Examples of long pin (left) and short rod (right) containing MOX pellets.](image2)
4. The project

The reference material preparation, transport and delivery to the reprocessing plant require a very strict operational time schedule that takes into account technical and legal constraints. The main project drivers are the provisions for the transport by the safety authorities (venting of all models in He atmosphere), by the receiving reprocessing plant (strict time schedule to satisfy worldwide customers requests) and by the legislative framework of the stakeholders countries (safety and security issues).

The project has gone through the following main steps implemented in the period comprised between January 2007 and April 2009:

- application for new or upgraded nuclear licenses for all operations inside and outside PERLA
- design, procurement, testing and installation of a new shielded glove box with He atmosphere for models repackaging in the PERLA two-glove box room
- procurement, testing and installation of a gas-based leak tightness test for models in lieu of the bubble-based test previously used
- design, procurement, testing and validation of spare outer containers for the models
- identification and setup of paths and loading bays for transport casks in INE
- implementation of safeguards protocols for containment and surveillance measures
- validation of models venting operations in He atmosphere and of radiological protocols through cold tests (dummy models) and hot tests (real models)
- trans-boundary transport to the receiving plant with escorted armoured trucks.

In the text paragraphs we will provide more details on the PERLA refurbished two-glove box room and the venting operations on the models.

4.1 The refurbishment of the two-glove box room

The PERLA glove box room housed two plexiglas® glove boxes, working in depressurised air atmosphere, connected with an exhaust duct linked to the INE ventilation system. During the venting campaign the models needed to be handled for several minutes and therefore, due to the high dose rates involved, the glove boxes needed to be appropriately shielded. It was decided to replace just one glove box with a brand new one (named GB2 in the following text) designed to work in He atmosphere with special lead glass shielding and other internal features to facilitate the operations. The other plexiglas® glove box (named GB1 in the following text) was equipped with additional shielding. The two glove-boxes were then connected each other through a transfer airlock in order to avoid bag-in and bag-out operations, and thus reducing exposure to operators and operations time. Fig. 3 and Fig. 4 show, respectively, the new glove-box and the new glove-box system.

![Fig. 3: the new shielded glove box](image1)

![Fig. 4: the two-glove box system (on the left the existing one used for decontamination, on the right the new shielded one used for the venting operations)](image2)
4.2 The venting operations

As a result of radiolysis and heat induced degradation of the inner containers PVC wrapping, due to irradiation from the growing Am content and the U daughters, in nearly 20 years hydrogen has been produced in the models and trapped by the leak-proof outer containers cap screw. In order to avoid any possible explosive mixture with air humidity in case the hydrogen is liberated during particular transport conditions (road accident, etc), all models have been subject to a venting campaign in an inert atmosphere (He gas). In addition, also all those transport casks hosting highly active vented models (liable to produce further hydrogen during the transport time) have been vented with He after being loaded and before being sealed. After venting, both models and transport casks have been leak tested.

The actions performed on each model during the venting can be summarised as follows: extraction from the storage and transfer to the glove-box room inside He purged GB2 through the GB1, opening, o-ring removal, venting (for bigger models a He-gun was used to optimize the venting), o-ring slot cleaning, new o-ring positioning, closure, model and GB2 decontamination down to GB2 allowable levels, transfer to GB1 for decontamination down to environmental levels, extraction from GB1, leak-tightness test and return to the storage awaiting final loading in transport casks.

Spare parts (model containers, o-rings, screws, etc) were made available in suitable quantities. In order to cope with scenarios in which the model could not be opened in any other way than cutting it, prior the commencement of all operations a set of spare model containers were made constructed and validated according to the original qualified design. Similarly, back-up staff has been scheduled for any crucial services (maintenance, emergency, decontamination, etc).

Due to the campaign time constraints (Fig. 5) and the hazardous nature of the operations, a risk analysis (Tab. 2 contains a simplified extract) has suggested countermeasures to mitigate possible spreads of contamination, delays and technical/contractual consequences for all the stakeholders.

![Fig 5. Operations time constraints per model (in brackets the number of models). T₀ is the time at which the reference material must be delivered at the receiving plant.](image-url)
Tab. 2 Main venting operations and risk analysis on the models

<table>
<thead>
<tr>
<th>Operation</th>
<th>Risk</th>
<th>Consequence</th>
<th>Countermeasure</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 Extract model from storage to benches area</td>
<td>fall of model</td>
<td>NO: models are qualified to fall in the benches area and not to be damaged</td>
<td>more attentive handling</td>
</tr>
<tr>
<td>2 Check surface contamination and dose rate levels</td>
<td>model is contaminated</td>
<td>YES: model cannot be moved within benches area</td>
<td>model (wrapped) sent back to storage for later decontamination</td>
</tr>
<tr>
<td>3 Take model to GB room</td>
<td>fall of model</td>
<td>NO: models are qualified to fall in the GB room and not to be damaged</td>
<td>more attentive handling</td>
</tr>
<tr>
<td>4 Transfer model to GB still of model</td>
<td>undue exposure and prolonged operations</td>
<td>YES: operators must shift</td>
<td>make good plan for waste cleaning, model transit and operators presence</td>
</tr>
<tr>
<td>5 Open (unscrew) outer container cap</td>
<td>screws break</td>
<td>YES: model remain closed</td>
<td>outer container cut, content put in spare model</td>
</tr>
<tr>
<td>6 Visually check PVC and inner container status</td>
<td>re-suspension of ashed PVC or material powder</td>
<td>YES: GB2 contaminated</td>
<td>decontamination</td>
</tr>
<tr>
<td>7 Venting speed up with He gun</td>
<td>re-suspension of PVC broken by the gun</td>
<td>YES: GB2 contaminated</td>
<td>decontamination</td>
</tr>
<tr>
<td>8 Replace metallic o-ring</td>
<td>new o-ring breaks or unfit</td>
<td>YES: o-ring and model seat can be damaged</td>
<td>change o-ring or model (up to two sets of o-rings, screws and one set of model are available per model)</td>
</tr>
<tr>
<td>9 Close (screw) outer container cap with new screws</td>
<td>screws break or unfit</td>
<td>YES: model not closable</td>
<td>change model and o-ring (up to two sets of o-rings, screws and one set of model are available per model)</td>
</tr>
<tr>
<td>10 Decontaminate outer container and GB2 down to allowed GB2 levels</td>
<td>model is still contaminated</td>
<td>YES: model not movable outside GB2</td>
<td>additional massive decontamination</td>
</tr>
<tr>
<td>11 Evacuate waste from GB2</td>
<td>waste containers break and release contaminants</td>
<td>YES: GB2 and/or GB room contaminated</td>
<td>additional massive decontamination</td>
</tr>
<tr>
<td>12 Transfer model to GB1</td>
<td>undue exposure and prolonged operations</td>
<td>YES: operators must shift</td>
<td>make good plan for waste cleaning, model transit and operators presence</td>
</tr>
<tr>
<td>13 Decontaminate outer container and GB1 down to allowed GB1 levels (environmental)</td>
<td>model is still contaminated</td>
<td>YES: model not movable outside GB1</td>
<td>additional massive decontamination</td>
</tr>
<tr>
<td>14 Evacuate waste from GB1</td>
<td>waste containers break and release contaminants</td>
<td>YES: GB1 and/or GB room contaminated</td>
<td>additional massive decontamination</td>
</tr>
<tr>
<td>15 Extract the model from GB1</td>
<td>fall of model</td>
<td>NO: models are qualified to fall in the GB room and not to be damaged</td>
<td>more attentive handling</td>
</tr>
<tr>
<td>16 Model subject to gas leak-tightness test</td>
<td>test fails</td>
<td>YES: model cannot be sent back to storage nor shipped</td>
<td>repeat venting operations</td>
</tr>
<tr>
<td>17 Take model back to storage</td>
<td>fall of model</td>
<td>NO: models are qualified to fall in the benches area and not to be damaged</td>
<td>more attentive handling</td>
</tr>
</tbody>
</table>

5. Conclusions
All JRC operations have been designed and approved in a 2 years time, and performed in 6 months between 2008 and 2009 (from installation of the new glove-box system to the transport casks convoy departure from the site).
The 59 models venting has been carried out in 28 working days. The venting of each model required up to 4 hours handling from exit to return to the storage, with the venting operation itself taking ca. 2 hours. A total number of ca. 50 staff has been involved in this project, with a strict radiological protocol over all operations, two 8-hour shifts per day schedule.
It has been a challenging project, as it has been performed in a facility mostly shut down, with a very tight schedule, a considerable turnover of key personnel, and massive technical and legal implications for the stakeholders.

6. Acknowledgements
The authors want to acknowledge all their JRC colleagues and contracted experts who have contributed in successfully concluding this project.
LAGUNA VERDE: A 120% EXTENDED POWER UPRATE PROJECT DEVELOPED BY IBERDROLA

1. INTRODUCTION

The experience which this document wants to present, describes the work being developed by IBERDROLA Ingeniería y Construcción, for the Laguna Verde plant in Mexico, owned by the Federal Electricity Commission (CFE). This generation plant consists of two light boiling water type units (BWR) design by General Electric in the 80s.

The objective of this project is to perform the modifications on the thermal cycle of the plant required by an Extended Power Uprate, to achieve a safe and reliable operation of the plant at 120% of its original thermal power, whilst upgrading and renovating plant equipment and installations to achieve a license renewal from 40 to 60 years of operation.

The consortium formed in 97% by IBERDROLA Ingeniería y Construcción SAU and in 3% by ALSTOM Mexicana, S.A. de CV, was awarded the contract in an international bid, competing against General Electric and Siemens. The project began in March 2007 and is scheduled to finalize in December 2010. At this point the work carried out include modifications of the main condenser replacement, moisture separator reheaters (MSR's) and feedwater heaters no. 5 and 6 in the two units, therefore having executed two out of four scheduled outages.

Fig. 1. Panoramic photo of the Laguna Verde Nuclear Power Plant (Mexico)

The scope, development and organization of this project, whose basic elements include the design, engineering, training, supply of equipment, dismantling, installation, testing, commissioning, treatment and delivery of radioactive waste generated during the project implementation to CFE, is aimed to ensure a safe and reliable operation of the plant under the new conditions of increased thermal power of the reactor, with a thermal cycle optimized so that the gross power of the generator increase from the current 686.7 MWe to a value of 817.1 MWe in both units.
2. IMPLEMENTATION OF THE EQUIPMENT MODIFICATIONS

A maximum period of 90 days per Unit, during both outages scheduled to take place during years 2008 and 2010 for Unit 1 and years 2009 and 2010 for Unit 2, is available for the implementation of the described below modifications.

Additionally, other outages were available, in March of 2007 for Unit 1 and in October of 2007 for Unit 2, which have been used to conduct inspections at both plants in order to properly plan the activities of modifications and replacement of equipment described in the previous section.

The basic program for the replacement and modification of equipment is as follows:

- Prior to 2008 unit 1 outage and 2009 unit 2 outage, a new turbine building crane has been installed, as it is necessary to carry out the installation of the equipment.
- The work carried out during the unit 1 outage, which took place in the autumn of 2008, and the Unit 2 outage of spring of 2009, has been related to the change of the condenser, moisture separator and main steam reheater as well as changing the feedwater heaters and the distributed control system of the same.
- During the second outage, which is going to take place in the spring and autumn of 2010, the turbogenerator unit will be installed: main turbine (high and low pressure), full electric generator and exciter, electrohydraulic turbine control system and the supervisory system, main steam bypass valves, transformers, isolated phase bus and main circuit breaker, off-gas condenser, installation of two new pre-filters and demineralizers and rehabilitation of the turbine building “Heating, Ventilating and Air Conditioning”(HVAC) system.

The most significant modifications, referring to major components, are detailed below.

2.1 Replacement of the turbogenerator

The increase of power to 120% leads to increased steam flow exceeding the original conditions by 20%. The current turbine, manufactured by Mitsubishi, does not offer sufficient capacity, and therefore needs to be changed for a larger capacity unity. After an analysis of potential suppliers, the consortium led by IBERDROLA, decided to implement a turbine offered by ALSTOM, with a model manufactured expressly to suit the characteristics of the plant, characterizing with a very efficient thermodynamic design and integrated rotors, both in high pressure and low pressure in order to minimize the need for maintenance of low pressure rotor and ensure its operation during the potentially expanded lifetime, avoiding the cracks that usually appear on the discs and couplings in the nonintegrated nuclear rotor blades, typical of the 70s.

In order also to meet the expected increase in electrical power, it has been as well decided, to replace the generator by one of a higher-capacity, also supplied by ALSTOM and with a nominal capacity of 950 MWA.

Replacement of the turbogenerator includes changing the rotors of high-pressure turbine, the two frames of the low pressure turbine, including inside frames, complete electrical generator and exciter. Also providing a new electro-hydraulic control system of the turbine and a new supervisory system.

The design of these components and execution of the implementation and installation work will be performed by ALSTOM. The assembly and installation of the turbogenerator unit at the CNLV, will begin in March 2010 for Unit 1 and August 2010 for Unit 2.
2.2 Installation of a new bridge type crane in the Turbine Building

With a capacity of 165 tons, this new crane, manufactured by Sistemas Hormiga (Mexico)-Demag has significantly increased the flexibility of the plant during outage periods.

The new turbine crane is capable of handling the removal of the current generator and placement of the new one, which weighs approximately 300 metric tons. This capacity required the reassessment on the porches beams and structure of the turbine building to verify that they are capable of withstanding such a maneuver.

Evaluations prepared by the IBERDROLA consortium confirms the feasibility of the maneuver, which will save considerable effort by not making it necessary to install a crane outside of the building in order to lift the turbine generator.

Currently, each unit is equipped with this new crane, which has been installed during the pre-outage (plant operation). These cranes have been successfully utilized in the extraction and insertion maneuvers of the MSRs.

2.3 Replacement of the moisture separator reheater (MSRs)

These equipments reheat the steam leaving the high pressure turbine, which is directed to the low-pressure turbines. This way, the entry of moisture into the lower corpuses turbines is avoided, which in turn improves their performance and guarantees a long-term operation.

During the first two outages (September 2008 and March 2009) high efficiency MSRs were installed, consisting of a moisture separator or high performance chevron with a moisture removal efficiency of around 99.5%, and two reheating stages designed of finned tubes with 4 passes per stage, able to obtain TTD’s (Terminal temperature difference) lower than 9°F. These devices are designed and manufactured by TEI / Swecomex (Mexico), and are fully compatible to operate with the current MHI turbine and the future ALSTOM turbine.

To improve the efficiency of new MSRs, the scavenging lines of drains have been replaced for others, made of alloy steel. In addition, new instrumentation was installed for MSRs, improving temperature and pressure measurements, necessary for calculating their performance.
2.4 Replacement of the main condenser

In order to improve the performance of the cycle, ensure full power operation of the plant at warmer operation conditions (the condenser is cooled directly by water from the Gulf of Mexico), and improve the chemistry of the reactor, avoiding copper input to the reactor from the current copper-nickel tubes or admiralty, the current condenser tube bundles have been replaced for other bundles of an optimized thermal design and Titanium tubes.

The new condenser design allows adequate removal of incondensable gases from the condenser, guaranteeing their proper subcooling. The replacement of existing ejectors and associated systems will be also performed, in order to ensure the proper pressurization of the condenser.

This modification raises some significant mounting difficulties, as the condenser is not directly accessible from the outside of the turbine building. This has required the transport of the bundles across the turbine plant and lowering them into the location of the condenser through an equipment opening, and has therefore been necessary to divide each tube bundle into four smaller bundles, assembled right before their introduction into their final location inside the condenser.

The difficulty of the operation has required a very detailed planning, considering also the 43 days timeframe available for the replacement of tube bundles.

Additionally, in order to ensure adequate thermal performance of the condenser, a tube cleaning system utilizing balls has been installed for the new Titanium bundles, which is an inert material susceptible to biological growth.

A cathodic protection system has also been installed, to improve the corrosion behavior of the condenser.
2.5 Condensate, feed water and heaters’ drains system

The water supplied to the reactor needs to be heated in order to obtain the necessary conditions of pressure and temperature prior to entering the reactor. The power increase with its consequent increase in flow, requires a modification to the feed water conditioning system, in order to maintain required variables. This new situation has been achieved with a modification made in the last stage of low pressure heaters (heaters 5A/B) and modification of the high pressure stage (heaters 6A/B).

Additionally, an evaluation of the condensate and heaters drains system has been performed to meet the new flow expected after EPU 120%. In general, the flow of condensate will increase by around 20%, which requires increasing the capacity of existing condensate pumps, changing the current impellers with new ones. In order to maintain an adequate redundancy, to ensure system operation during possible transient operation, an additional booster and condensate pump will be installed, which in normal operation will remain in reserve.

Fig. 8 New feed-water heater (5B)

Studies and evaluation of the heaters’ drains system concluded that it only requires the replacement of the drain pumps’ motors for more powerful ones.

At the same time the drain system requires control valves replacement with higher CV because of the new system flow.

From the instrumentation point of view, the whole system has been revamped in order to implement a complete field bus system.

2.6 The condensate Polishers and prefilters system

The capacity of the system will be expanded, by installing two additional pre-filter vessels with backflush elements and two additional mixed bed ion exchangers (Polishers), in order to comply with the removal of soluble and insoluble material going into the reactor, ensuring its structural integrity and its operation under the new EPU conditions. This installation includes not only equipment installations, but also all instrumentations associated with the local and remote system, control, alarm indicators and wiring.

2.7 Main steam bypass valves replacement

To upgrade the existing bypass valves, new bypass valves with an independent body will be installed, in order to improve their reliability. The design of the new Bypass Valve system will be fully integrated with the electrohydraulic control system.
2.8 Main transformers replacement

To meet the new electrical power generated by the plant, it is necessary to replace the seven main transformers of which three transformers serve each unit and one serves as a backup. The new single phase transformers have a transformation ratio of 22/440KV and a power of 105/300MVA (ONAN / OFAC) per unit, with an impedance of 12 ± 7.5%, almost 5% more impedance of the current transformers.

The transformers are being manufactured by IEM, in the State of Mexico and will be the biggest single-phase transformers installed in Mexico. These transformers will be equipped with the most modern on-line monitoring equipment (oil, terminals, etc.). It will be also installed in a new anti-explosion system based on nitrogen injection.

The first four transformers, already manufactured, will be transported to plant at the end of this year, to be installed during the next outage.

The scope also includes the dismantling and removal of existing transformers to CFE warehouses.

2.9 Isolated phase bus replacement

To meet the nearly 20% increase in the electrical power produced by the generator, which is transported from the generator to the transformers by isolated phase bars, the replacement of these bars for others of greater capacity is necessary.

The new bus is equipped with a forced cooling system (as the current one) and its design is influenced by many dimensional constraints:

- Dimensions of the current installation: the new bus, considering it’s higher capacity, is larger than the present one, but the design must be compatible with available space, because it is not possible to carry out any civil works.

- New equipment dimensions: All equipments that interface with the bus will be also replaced, therefore the bus size shall be designed to be connected to these equipments:
  - Main transformer
  - Generator circuit breaker
  - Main generator
  - Auxiliary system Transformer
2.10 Main generator breaker replacement

The current pneumatically controlled breaker (BBC), with three independent poles and cooled with water, will be replaced by a new one, HEC7C ABB type (this type of breaker has the highest interrupting capacity currently available on the market). It is a three-pole breaker with SF6 and electrohydraulic drive. The main features are:

<table>
<thead>
<tr>
<th>Feature</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Nominal voltage</td>
<td>25.2 kV</td>
</tr>
<tr>
<td>Rate short circuit withstand current (1 seg)</td>
<td>250 kA</td>
</tr>
<tr>
<td>Rated interrupting time</td>
<td>61 ms</td>
</tr>
</tbody>
</table>

Fig. 11 Generator circuit breaker

2.11 Turbine building HVAC System

Because of the new system conditions and foreseeing a significant increase regarding the heat loads released into the turbine building environment after the EPU implementation, an evaluation analysis of the system has been prepared. The main equipments to be replaced are the exhaust and supply fans. Likewise, new chillers and coils will be installed to ensure a proper air temperature and the cooling of certain plant rooms, which currently characterize with high temperature. These units will be installed at the beginning of 2010.

Fig. 3 Supply fans in the shop

3. OTHER OUTAGES DATA

In the implementation of the modifications at the CNLV, the number of people involved during the first two outages reached around 200 people of Iberdrola Ingeniería y Construcción, from both Spain and Mexico, and more than 1,500 people working directly on site, for more than 50 different sub-contractors. During the second two outages, more than 1,900 people is working directly on site.
It has been necessary to establish different shifts and schedules, in order to optimize the use of time and develop the work according to the program (43 days for the first outage and 47 for the second).

It is worth mentioning that the loads carried during the work exceeded 2700 tons (output/ input) per outage, during the first outages:

- MSR: Marine Transport with a hydraulic platform.
- Tube bundles introduction using hydraulic systems.
- Tube bundles extraction and Introduction using air pallet.
- Advanced cutting methods: Cutting using diamond wire.
- Platform Design to Remove and introduce the tube bundles.

Since the outage has a very limited duration, the maximization of work during the pre-outage is a crucial factor. That’s why it is very important to have a comprehensive planning for all the work that can be done during operation.

After the completion of the outage 13 work performed at unit 1, interdisciplinary meetings have been held in order to analyze the potential improvement areas and to establish actions to achieve these improvements. As a result of these meetings a number of guidelines have been established to help us improve in future nuclear EPU projects as well as other modifications to be carried out as part of the CNLV EPU.
4. CONCLUSIONS

An Extended Power Up-rate means an opportunity to modernize equipments, to improve maintenance, to get a better plant knowledge and to motivate the employees facing a challenging project.

This project, being the biggest foreign nuclear contract awarded to a Spanish company, demonstrates our ability to handle challenging projects which have impact on countries´ growth, both in the field of productivity and technical development. It also demonstrates client´s commitment to the long term development of nuclear energy, a clean and non contaminating energy, free of CO2 emissions.

Other advantages of the EPU:

• Generates significant economic value in the short term, with a very competitive marginal price per kWh in comparison with other technological alternatives
• Involves a comprehensive review of plant design, which allows to evaluate and confirm margins in front of the regulatory organism

The following are some of the key success factors of an outage:

• Implementation of lessons learned during the previous outage
• Detailed and realistic planning of the modification packages activities
• Establishment of contingency plans
• Coordination between the customer and IBERDROLA teams
• Maximization of pre outage work during operation: Planning all work that can be done during operation
• Reengineering of equipment installation procedures, based on lessons learned
FEASIBILITY STUDIES ON MINOR ACTINIDES TRANSMUTATION IN SODIUM FAST REACTORS

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ABSTRACT

Within a heterogeneous fuel distribution model of a sodium-cooled fast reactor we study the minor actinides transmutation feasibility. We have used the EVOLCODE2 advanced burn-up code in order to obtain the Doppler coefficient of reactivity, void worth, the fraction of delayed neutrons and the isotopic mass evolutions in both driver and blanket regions. We have obtained an estimation of the MA transmutation factor in a blanket containing 10% weight MA. Results are presented and compared to other case with 20% content, as well as to a slightly different blanket configuration.

1. Introduction

The management of minor actinides in fast reactors can be face up with different strategies. Aspects such as fuel manufacture or safety reactivity coefficients play an important role in the choice between homogeneous, heterogeneous or a mixed multirecycling model. In the homogeneous strategy MA are diluted in the UO₂ fuel assembly, in the heterogeneous model they are distributed only in the periphery of the core and in the mixed configuration some (U+MA)O₂ fuel assemblies are spread out over the MOX driver.

The preference for the radial blankets is based on the small impact of the MA in the sodium void coefficient and also on the fact that it has almost no impact on the core management. This last point is important because it enables longer irradiation periods for MA giving rise to high transmutation rates.

We have studied transmutation performances of radial blankets loaded with a mixture of depleted uranium and minor actinides oxide (U+MA)O₂. We focused on a realistic scenario (from the point of view of MA content) where the MA mass fraction is 10%, and we show also some preliminary results with 20%. Additionally, as a sensitivity study, a limited comparison with a 10% case with slightly different assembly position distribution (Ref. 1) is done.

2. Reactor core description

A representative sodium-cooled fast reactor design is chosen based on open bibliography (Ref. 2). The main characteristics are a driver core with MOX fuel and a blanket composed by 90% depleted U and 10% MA. Some nominal properties are shown in Table 1.

<table>
<thead>
<tr>
<th>Core nominal power</th>
<th>3600 MWth</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel composition</td>
<td>Driver: MOX</td>
</tr>
<tr>
<td></td>
<td>Blanket: U + MA</td>
</tr>
<tr>
<td>Coolant</td>
<td>Sodium</td>
</tr>
<tr>
<td>Driver: Pu/U ratio</td>
<td>14.7/85.3</td>
</tr>
<tr>
<td>Blanket: MA/U ratio</td>
<td>10/90</td>
</tr>
<tr>
<td>Fuel burn up (GWh/dHM)</td>
<td>91.74</td>
</tr>
<tr>
<td>Irradiation period (d)</td>
<td>2000</td>
</tr>
<tr>
<td>Assembly configuration</td>
<td>Hexagonal</td>
</tr>
</tbody>
</table>

Table 1. Main geometrical characteristics of the core reactor
The geometrical and additional material information as well as the MOX and MA fuel vectors concerning the core model was taken from Ref 2. The MA vector corresponds to a 45000 MWd/THM irradiated LWR fuel, 30 years cooled down. There are 90 fuel assembly positions around the driver which configure the blanket. Some results will be compared with another configuration (Ref. 1) with 78 positions for the blanket (slightly closer towards the core centre, in average).

3. Computational tool

EVOLCODE2 (Ref. 3 and 4) is the computational tool used in our group to estimate cores burnup. It is an in-house development to automatically couple MCNPX and ORIGEN/ACAB codes for static and burnup calculations.

The first stage solved by the computational tool is the MCNPX code, which allows an important degree of the heterogeneity description in the reactor core model. Fuel material homogeneously evolves along several burnup steps for cells formed by groups of fuel rods. In a second computational stage, EVOLCODE2 obtains a single group cross section in every core cell under MCNPX predicted spectra conditions and feeds ORIGEN code (or ACAB) to provide the inventory evolution under irradiation for a user-input burnup step. After this stage, EVOLCODE2 automatically generates a second MCNPX input to estimate new spectra and relative fluxes for the next burnup step.

Particularly in this study the irradiation was performed by means of 10 EVOLCODE2 steps with 200 days per step. A simplified single batch cycle of 2000 equivalent full power days (EFPD) has been assumed instead of several real batches. Representative results of the full first cycle are then obtained, and also those representative of an equilibrium cycle when looking to the segment comprised between 820 and 1230 EFPD (separated analysis demonstrates this argument).

The neutronic cross section library and radioactive decay data used in this work is the JEFF3.1 database.

4. Results

We present now the results regarding neutronic spectrum, isotopic composition evolution, \( k_{\text{eff}} \) and reactivity coefficients, as well as the transmutation factor.

![Figure 1. Neutronic spectrum in the driver and blanket](image)

Figure 1 provides the energy spectrum of the neutron flux at beginning and end of irradiation for the driver (right-hand side) and the comparison between the spectrum in the blanket and the driver at BOI (left-hand side). A significant difference in the spectrum can be seen in driver and blanket regions due to the composition of the fuel and the geometrical
position. Also, the impact on the epithelial region of the spectrum of fission products accumulation may be studied.

In Table 2 the initial and final masses of the actinides are shown. In average, it may be observed that there is a net Uranium consumption, Plutonium and Curium production and Neptunium and Americium stabilization (in fact, a small Am production). Looking at Table 3 we see that each nuclide mass evolution is different in the driver than in the blanket.

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>Mass (kg) at BOL</th>
<th>Mass (kg) at EOL</th>
</tr>
</thead>
<tbody>
<tr>
<td>U</td>
<td>58000</td>
<td>49520</td>
</tr>
<tr>
<td>Pu</td>
<td>7923</td>
<td>8890</td>
</tr>
<tr>
<td>Am</td>
<td>1023</td>
<td>1047</td>
</tr>
<tr>
<td>Cm</td>
<td>87.90</td>
<td>186.6</td>
</tr>
<tr>
<td>Np</td>
<td>225.2</td>
<td>225.1</td>
</tr>
</tbody>
</table>

**Table 2 Evolution of the actinides element masses evolution**

<table>
<thead>
<tr>
<th></th>
<th>Blanket</th>
<th>Driver</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mass (Kg) BOI</td>
<td>Mass (Kg) EOI</td>
<td>Mass (Kg) BOI</td>
</tr>
<tr>
<td>235U</td>
<td>36.07</td>
<td>28.09</td>
</tr>
<tr>
<td>238U</td>
<td>11990</td>
<td>11624</td>
</tr>
<tr>
<td>238Pu</td>
<td>0</td>
<td>110.9</td>
</tr>
<tr>
<td>239Pu</td>
<td>0</td>
<td>308.1</td>
</tr>
<tr>
<td>242Cm</td>
<td>0.2728</td>
<td>12.82</td>
</tr>
<tr>
<td>244Cm</td>
<td>68.66</td>
<td>79.26</td>
</tr>
<tr>
<td>245Cm</td>
<td>16.83</td>
<td>18.29</td>
</tr>
<tr>
<td>241Am</td>
<td>809.8</td>
<td>648.5</td>
</tr>
<tr>
<td>242mAm</td>
<td>0</td>
<td>23.72</td>
</tr>
<tr>
<td>242Am</td>
<td>3.213</td>
<td>0.06577</td>
</tr>
<tr>
<td>243Am</td>
<td>209.7</td>
<td>174.2</td>
</tr>
<tr>
<td>237Np</td>
<td>225.2</td>
<td>190.7</td>
</tr>
<tr>
<td>237Np</td>
<td>225.2</td>
<td>190.7</td>
</tr>
<tr>
<td></td>
<td>243Am</td>
<td>0</td>
</tr>
<tr>
<td></td>
<td>239Np</td>
<td>0</td>
</tr>
</tbody>
</table>

**Table 3 Mass of main actinides isotopes at BOL and EOL**

Table 3 shows that there is a net Am-241 consumption along the cycle (production in the driver is well balanced by destruction in the blanket) while the occurrence of Am -243 in the driver is large. Therefore, in this design, there is not a net transmutation efficiency concerning Americium. Furthermore, the same can be said concerning Neptunium and much more clearly with Curium isotopes. In the case with 78 assemblies in the blanket, Ref. 1, we also obtained that for Am -241 there was no net transmutation between the driver and the blanket along the 2050 EFPD cycle.

The transmutation factor is plot in right-hand side of Figure 2 where it is shown the proportion of minor actinides that have been consumed, as it is expressed by the following formula: $\frac{M_i - M_i(t)}{M_i}$ *100. Results with 10% and 20% MA mass content are shown at the bottom panel.

The upper parts of Figure 2 shows results for a core with no MA recycling, but including a small amount of MA, cumulated during storage. The absolute value of the transmutation
factor is high (i.e., low effectiveness as transmuter) though the involved MA mass is small at BOL.
There is a big difference in the generation of MA between this three cases (no MA, 10 and 20%) since in the 0% MA case there is always a generation of MA but for 10 and 20% at a certain time the consumption is larger than the generation of MA.

![Comparison of the transmutation factor (left-hand side) and transmutation rate per year (right-hand side) in terms of the burnup time](image)

**Figure 2.** Comparison of the transmutation factor (left-hand side) and transmutation rate per year (right-hand side) in terms of the burnup time

As it was expected the larger the amount of MA loaded in the blanket is, the larger the transmutation factor becomes. While -9.2% is obtained at the end of the 2050 EFPD irradiation period is obtained in the 10% case, a small positive value is predicted for the 20% MA case. For comparison, the Ref. 1 case predicts -19% for the same period.

In left-hand side of Figure 2 we present the annual variation or transmutation rate of minor actinides, \( \frac{M_i - M_f(t)}{M_i} \frac{365}{t(days)} \times 100 \) (where \( M \) is the total mass of MA)

![Calculated k\textsubscript{eff} values along the cycle](image)

**Figure 3.** Calculated k\textsubscript{eff} values along the cycle

Values of k\textsubscript{eff} along the cycle are presented in Figure 3. A comparison between the core without MA (the whole core loaded with MOX) and a 10% MA fraction in the blankets is shown. The k\textsubscript{eff} is higher when no MA are present because in this case blankets positions
are filled with 15% Pu MOX assemblies. The 10% MA core starts with a keff equals to 0.97, which means that Pu content should be tuned to obtain enough margin for the burnup. This exercise has not been pursued. At around 600 burn-up days the amount of fissile material is maximum and then slightly decreasing because the effect of the MA and fission product generation is larger than the Pu breeding.

Concerning the blanket, our insight is that the residence period of time should be in the order of 4000 days for an optimization of the transmutation, just twice the analysed period, before fast neutron fluence in the cladding will lead to damage limit. Therefore, a more complete analysis could be done replacing the 2000 days burnt driver by a fresh one before obtaining rigorous conclusions concerning the blanket performance. This exercise has not been however analyzed here.

Some results regarding escape, capture and fission reaction are shown in Table 4 for the 10% MA case. We can see that the fraction of leakage remains the same at BOC and EOC, the capture reactions rise just a little and the fission reactions fall almost in the same rate.

<table>
<thead>
<tr>
<th>Reaction</th>
<th>Ratio</th>
<th>Energy (MeV)</th>
<th>Ratio</th>
<th>Energy (MeV)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>BOC</td>
<td></td>
<td>EOC</td>
<td></td>
</tr>
<tr>
<td>Escape</td>
<td>0.055</td>
<td>0.012</td>
<td>0.055</td>
<td>0.012</td>
</tr>
<tr>
<td>Capture</td>
<td>0.610</td>
<td>0.102</td>
<td>0.614</td>
<td>0.104</td>
</tr>
<tr>
<td>Loss to fission</td>
<td>0.337</td>
<td>0.282</td>
<td>0.332</td>
<td>0.272</td>
</tr>
</tbody>
</table>

Table 4. Escape, capture and loss to fission reaction ratios at BOC and EOC

In order to complete the results, we present selected reactivity coefficients in Table 5. Doppler coefficient is calculated as the difference in reactivity between cases where the fuel is at 900K and 600K (representative of Hot Zero Power). In both cases coolant and structural material are kept at 600K. Therefore,

\[ C_{\text{Doppler}} = \frac{\Delta \rho}{300} \text{ where } \Delta \rho = \frac{k_{600K} - k_{900K}}{k_{600K}k_{900K}} \]

<table>
<thead>
<tr>
<th>CASE</th>
<th>Keff (600 K)</th>
<th>Keff (900 K)</th>
<th>( C_{\text{Doppler}} ) (pcm/K)</th>
<th>Sodium Void worth (pcm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>10% MA – 90 S/A Blanket (this study)</td>
<td>BOL 0.97940</td>
<td>0.97374</td>
<td>-1.9783</td>
<td>+657</td>
</tr>
<tr>
<td></td>
<td>EOL 0.95554</td>
<td>0.95298</td>
<td>-0.9371</td>
<td>+1983</td>
</tr>
<tr>
<td>10% MA – 78 S/A Blanket (Ref. 1 study)</td>
<td>BOL 0.99345</td>
<td>0.98786</td>
<td>-1.899</td>
<td>+1601</td>
</tr>
<tr>
<td></td>
<td>EOL 0.98135</td>
<td>0.97836</td>
<td>-1.038</td>
<td>+2315</td>
</tr>
</tbody>
</table>

Table 5. Estimated reactivity coefficients

MCNPX code allows taking or removing the delayed neutrons in reactivity calculations. Using that option, we have computed the fraction of delayed neutrons as:

\[ \bar{\nu}^{\text{prompt}} = \left( 1 - \beta_{\text{eff}} \right) \bar{\nu}^{\text{total}} \]

According to the results the prompt neutron multiplicity is \( \bar{\nu}^{\text{prompt}} = 2.90525 \) and the total neutron multiplicity is \( \bar{\nu}^{\text{total}} = 2.91927 \). That means that the fraction of delayed neutrons is \( \beta_{\text{eff}} = 0.00480 \) at BOL (0.00458 in Ref. 1 case).

These three reactivity coefficients have reasonable values concerning safety parameters.

5. Conclusions
We have presented our methodology to compute with EVOLCODE2 the burn-up of sodium fast reactors with a heterogeneous strategy of MA management. The results presented in this work demonstrate that such strategy for MA helps the design to get acceptable values of reactivity coefficients. Doppler and Void reactivity coefficients are in good agreement with other studies of similar sodium fast reactors (Ref.2), showing reasonable values from the licensing point of view for this SFR, which is a necessary request before starting further detailed analysis.

Concerning transmutation, results have shown that the introduction of blankets with 10% MA ratio significantly reduces the MA generation along the irradiation when compared with a reactor with no management of MA. This MA generation is even smaller if blankets with 20% MA ratio are fuelled. Additionally, results suggest that the reactor would be a net transmuter if we increase the cycle length with 20% MA ratio in the blankets. The analysis of this possibility including the estimation of the reactivity coefficients is planned as future work.

6. References


RADIONUCLIDE RELEASE FROM IRRADIATED TH-PU MOX FUEL

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ABSTRACT

Plutonium and minor actinides produced as by-products of the UO$_2$ nuclear cycle could be considered as waste or energy source depending on the strategy selected in the nuclear energy programme. Considering Pu and Minor Actinides as a source, they can be burned in existing water reactor for diminishing the radiotoxicity of the spent fuel, it is necessary to use “inactive” materials as matrix like ThO$_2$. ThO$_2$ matrix has demonstrated its Pu burning efficiency and higher corrosion resistance than UO2. Uranium-plutonium mixed oxide (MOX) fuel efficiency is low because the presence of U in MOX results in the creation of some new Pu under irradiation.

The dissolution behaviour of irradiated (Th,Pu)O$_2$ pellets with burn-up of 38.8 MWd/kg Th has been studied in carbonated (20mM HCO$_3^-$), deionised and granite ground water solution in a hot cell. The dissolution behaviour of Th, Pu, U and Np was studied in order to find out whether radionuclides release is depending on the matrix dissolution (solubility control). After irradiating the samples, K-ORIGEN and ORIGEN ARP codes were used to find out the theoretical inventory. Afterwards, fuel samples were dissolved completely and analyzed, in order to determine the experimental radionuclide inventory of the irradiated fuel. Th matrix alteration appears to reach an steady state and radionuclides dissolution shows dependence on the matrix behaviour as can be observed through the FIAP results.

1. Introduction

Various programmes are running internationally on the use of Pu in either inert matrix fuels (IMF) or Th based fuels [1], from a waste minimization point of view in order to get a long-term objective of sustainability of energy resources [2-5], because it reduce Pu stocks and Minor Actinides (MA), then decreasing of the long-term radiotoxicity of nuclear waste.

Based on the Nuclear Energy Program [6-8] applied, thoria based fuel could be taken into account as advanced fuel. Interesting Th-based fuels international studies are performed by IAEA [9, 10]. Making use of thoria involves addition of fissile materials like $^{235}$U, $^{233}$U or Pu [1, 2]. The aim of using thoria is, by one side, burning Pu and the MA as (Th/Pu) mixed oxide fuel and immobilizing and conditioning tetravalent long-lived actinides [11] and Pu in a geological repository.

Thoria properties are lower radiotoxicity production than UO$_2$ because it produces lower long-lived MA and Pu [8]; it has a fcc isomorphic structure with tetravalent actinides like UO$_2$, NpO$_2$ or PuO$_2$ [4]. Burning Pu as Th,Pu-MOX decreases the Pu inventories from the existing nuclear fuel cycle and finally ThO$_2$ has higher corrosion resistance than UO$_2$ [4, 5]. ThO$_2$ is chemically stable because the low aqueous solubility (as reported in the literature, solubility ThO$_2$(hyd, am) is about $10^{-19}$M in natural water conditions and room temperature[7, 12-17]. Th has only one stable oxidation state. $^{232}$Th is a fertile nuclear material which can replace $^{238}$U [4]. The similarity of properties of thoria with those of UO$_2$ makes its rapid introduction in
existing LWRs feasible [2]. Focused on burning Pu in thermal, Light Water Reactors and fissile actinides in nuclear reactors requires ThO$_2$ matrix has also demonstrated its Pu burning efficiency. U,Pu-MOX fueling of LWR’s does not allow a rapid reduction of the Pu stock-piles: the efficiency is low because the presence of U in MOX results in the creation of some new Pu under irradiation.

The aim of this work is to obtain an experimental data on the matrix and MA (generated as by-product) dissolution behaviour of the advanced irradiated fuels, in order to asses with thermodynamic constants in this new matrix from a long-term safety of a geological point of view.

2. Experimental procedure
2.1 Fuel Specification and leaching experiments

ThMOX with 3% of Pu fabrication was performed via sol-gel technique and sintered in the ITU facilities in order to ensure an homogeneous distribution of Pu in the ThO$_2$ matrix. The ratio (Pu/(Pu+Th)) in pellets were 0.0323 and 0.0071, before and after irradiation respectively , this makes a valuable process for burning Pu. Pellets were irradiated four times in a commercial PWR in Obrigheim (Germany) until reaching burnup of 38.8 MWd/ kg Th.

After irradiation, pellets came back to ITU for developing some experiments inside Hot Cells in glass shielded reactors under static conditions without solution renewal.

Th,Pu-MOX pellets were previously cleaned applying two wash cycles, first with acetone and the second with the same aqueous solution than the own experiments. Dissolution tests were performed in aqueous phases, representative for groundwater repositories, under deareated atmosphere. Three similar experiments using 40 cm$^3$ of approximately starting volume of water without controlled pH were carried out in different leachants with different ionic strength, increasing order demineralised water, carbonated water and granite groundwater (GW). The composition of leachants is given in Table 1.

At the end of each experiment the leaching vessel was opened, and the sample was removed from the sample holder. 1% HNO$_3$ was used to rinse the vessel for 20 h to 40 ml to solubilise any remaining precipitates or colloids that could have been deposited on the vessel walls or on the sampling rod. After one day of rinse the vessel, a sample was taken and the solution was discarded.

Different radionuclides concentration in solution was determined. A typical set of data is going to be shown. Sample concentrations were analyzed by elements using ICP-MS, where 1ppb of Ho was added to the samples as internal standards.

<table>
<thead>
<tr>
<th>Aqueous phase</th>
<th>Sample mass / g</th>
<th>Duration of static test / d</th>
<th>pH</th>
</tr>
</thead>
<tbody>
<tr>
<td>Carbonate water</td>
<td>1.1881</td>
<td>239</td>
<td>8.91</td>
</tr>
<tr>
<td>GW</td>
<td>2.4284</td>
<td>330</td>
<td>7.99</td>
</tr>
<tr>
<td>Demineralized water</td>
<td>1.3185</td>
<td>330</td>
<td>5.69</td>
</tr>
</tbody>
</table>
2.2 Solution Analysis

The Fraction of Inventory in Aqueous Phase (FIAP) were calculated in order to compare the results of the three experiments developed for different nuclides and is given by Eq 1

\[
FIAP = \frac{m_{i,aq}}{m_{i,fuel}} = \frac{[i] V_t}{m_S f_i}
\]

Where \(m_{i,aq}\) and \(m_{i,fuel}\) is the mass of radionuclide \(i\) in aqueous phase and in the initial fuel respectively, \([i]\) corresponds to \(i\) concentration in \(\mu g \cdot dm^{-3}\), \(V_t\) in ml is the solution volume at each sampling time and \(f_i\) is the inventory fraction for the radionuclide (mg/ (g fuel)\(^{-1}\)). This value will be calculated from the experimental inventory.

3. Results and discussion

3.1 Pre-leaching sample characterization

Pre-leaching characterization was done before irradiate the pellets. X-Ray diffraction (Figure 1) shows a very similar diffraction pattern compared with pure thoria pellets fabricated in the same technique than \((\text{Th-Pu}_{0.03})\text{O}_2\) pellets. There is only a small deviation that corresponds to the solid solution of Pu replacing Th positions in thoria lattice [1]. The SEM micrograph is shown in Figure 2 and Figure 3. It indicates an homogeneous distribution of the powders without segregation \((\text{Th-Pu}_{0.03})\text{O}_2\).
3.2 Inventory composition

<table>
<thead>
<tr>
<th>Minor Actinides</th>
<th>Measured inventory / %</th>
<th>Theoretical inventory (ORIGEN-ARP) / %</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{232}$Th</td>
<td>97.2468</td>
<td>97.3404</td>
</tr>
<tr>
<td>$^{233}$U</td>
<td>1.6690</td>
<td>1.5096</td>
</tr>
<tr>
<td>$^{234}$U</td>
<td>0.2216</td>
<td>0.2353</td>
</tr>
<tr>
<td>$^{235}$U</td>
<td>0.0498</td>
<td>0.0560</td>
</tr>
<tr>
<td>$^{236}$U</td>
<td>0.0056</td>
<td>0.0053</td>
</tr>
<tr>
<td>$^{237}$U</td>
<td>0.0347</td>
<td>0.0000</td>
</tr>
<tr>
<td>$^{237}$Np</td>
<td>0.0012</td>
<td>0.0005</td>
</tr>
<tr>
<td>$^{238}$Pu</td>
<td>0.0197</td>
<td>0.0244</td>
</tr>
<tr>
<td>$^{239}$Pu</td>
<td>0.1089</td>
<td>0.1228</td>
</tr>
<tr>
<td>$^{240}$Pu</td>
<td>0.1794</td>
<td>0.2634</td>
</tr>
<tr>
<td>$^{241}$Pu</td>
<td>0.2515</td>
<td>0.2197</td>
</tr>
<tr>
<td>$^{242}$Pu</td>
<td>0.1368</td>
<td>0.1364</td>
</tr>
<tr>
<td>$^{241}$Am</td>
<td>0.0139</td>
<td>0.0174</td>
</tr>
<tr>
<td>$^{243}$Am</td>
<td>0.0379</td>
<td>0.0392</td>
</tr>
</tbody>
</table>

Table 3 ORIGEN-ARP and experimental inventory obtained for (ThPu$_{0.03}$)O$_2$ with 38.8 MWd/kg

ORIGEN-ARP results of the most representative isotopes are presented were taken into account for performing chemical evaluation in order to know the elements ratio against Th. The experimental inventory of irradiated (Th-Pu$_{0.03}$)O$_2$ sample was determined by acid dissolution of the fuel and ICP-MS and Gamma-spectrometry analysis.

3.3 Radionuclides release

The results will be shown by elements, illustrated like both concentration evolution and FIAP. The concentration - time profile was measured by ICP-MS as leachate concentrations at each sampling point, and due to there was no replenishment the results correspond to cumulative concentration or FIAP. Because of isobaric interference of ICP-MS, it is not possible to distinguish between $^{238}$U and $^{238}$Pu or $^{241}$Pu and $^{241}$Am.

The radionuclide releases observed during experiment is illustrated like concentration in solution with time. Figure 4 shows experimental Th concentration evolution as a function of leaching time for three different aqueous solution, as the major component in the fuel. The experiments in carbonate media are illustrated as squares. Spheres represent low carbonate media and triangles are showing deionised water. The continuous line represents dissolved concentration in solution.
Figure 4 Th concentration evolution in different groundwater conditions. Th concentration slightly increased with time as a consequence of the hydrolysis of Th giving high soluble amorphous hydrous Th oxide formed in the surface. After 50 days, Th reached a steady state value that maintain when we dissolve the species sorbed on the walls (dots line), taking into consideration rinse concentration values. Then, sorption/precipitation on the vessel walls takes place. Amorphous hydrated Th dioxide solubility is $10^{-10}$M (represented as red dash dot line) and the dissolution of crystallized ThO$_2$ is lower than amorphous $[18]$. In all the cases, Th concentration rose above Th(OH)$_4$(am) solubility limit, this is the reason for explaining the difference between concentration in solution and rinse, probably forming an amorphous hydrated phase. Figure 5 shows data for Th(IV) measured in this work compared with other measured by Felmy et al. $[13]$ including the model curves as obtained using data from the NEA TDB.

Figure 5 ThO$_2$ solubility data at different pH and ionic strengths. Lines represent Th(OH)$_4$ calculated solubility with NEA database and MEDUSA program.

Th and Medusa. Our data are just higher than the thorium solubility for pH values considered. The next pictures correspond to MA representative of the fuel. The question is if they have the same behaviour than matrix.

Figure 6 Pu total concentration released in solution depending on the media.

Figure 7 Pu behaviour of ThMOX vs. references under granite environment.

Figure 8 Pu behaviour of ThMOX vs. references under deionised water environment.

Figure 9 U total concentration released in solution depending on the media.

Figure 10 Np total concentration released in solution depending on the media.
Comparatively almost all the elements concentration increases with time, until a steady state is reached (between 50-100 days). Taking into account Pu concentration evolution, is closed to the matrix as can be observed in Figure 6 with the total amount of Pu vs time. The mayor solubility control of Pu in this medium should be Pu(OH)$_4$ (am), but moreover in carbonate media Pu(CO$_3$)$_2^-$ could be in precipitated form. The experimental values correspond to amorphous hydroxide of Pu in groundwater as it is shown in Figure 7 and the same in the MilliQ water leachant (Figure 8). In the case of U is relatively the same than before. Take notice of only at low bicarbonate concentration we can distinguish sorption phenomena. Total U concentration remains under solubility limit and with the same behaviour than before. With regard to Np, we can observe that there is not any sorption process on the vessel walls, and from 45 days, Np reach a steady state (Figure 10).

![Figure 11 FIAP released in solution on MilliQ as leachant vs time](image1)

![Figure 12 FIAP released in solution on groundwater vs time](image2)

![Figure 13 Total radionuclides FIAP released in solution on 20 mM HCO$_3$ media](image3)

Regarding to cumulative FIAP results (plotted versus time in Figure 11, Figure 12 and Figure 13), in general, the fractional release of minor actinides show similar profile than Th, depending on the media considered, especially it can be observed stabilization for deionized and carbonated media. At high carbonate concentration media, the aqueous fraction is higher than deionized and groundwater, but as a summary, all the elements illustrated are controlled by solubility.

4. Conclusions

The leaching behaviour of Th-MOX irradiated fuel was investigated experimentally under three different aqueous leachants. In summary, the dissolution behaviour in almost all the analyzed minor actinides (used as indicators of matrix alteration) achieved the steady state. The leached (Th-Pu$_{0.03}$)$_2$O$_2$ pellet were characterized by Optical Microscopy and XRD previously, showing a solid solution with Pu uniform distribution on the ThO$_2$ matrix. Leaching tests performed on the ThO$_2$ pellets doped with 3 %w Pu showed that due to the oxidizing conditions, the concentration of Th in solution is in the expected range. Pu dissolution behaviour it is related to matrix dissolution. Pu concentration in solution is increasing with time until reaching a steady state and no sorption phenomenon on the reactor walls is detected.

U concentration evolution achieved a steady state. However sorption phenomenon in the vessel was detected. Np release behaviour shows dependence with the matrix alteration rate. No sorption phenomenon was measured.

A sorption phenomenon was detected for matrix element and U with independence of the media considered.

In summary, minor actinides release is highly dependent on Th release, reaching rapidly the steady-state in the bulk solution; it proves the applicability of thoria as matrix for immobilizing minor actinides in a deep geological repository.

Future surface post-characterization allow us to identify precipitated secondary phases.

5. References


02.06.2010

Wed 08:30 – 10:10

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Fuel cycle
TOWARDS INTEGRATED SAFEGUARDS IN SPAIN: IMPLEMENTATION ISSUES AT THE JUZBADO PLANT

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ABSTRACT

Nuclear safeguards are implemented on the global international scale by the International Atomic Energy Agency (IAEA), which aims to verify that a State uses nuclear energy for only peaceful purposes, whereas in the European Union (EU) the European Commission applies nuclear safeguards to make certain that nuclear materials are not diverted to purposes other than those for which are intended. Thus, nuclear operators in the EU are subjected to the scrutiny of two different Inspectorates with different goals. To cope with this issue, safeguards in the EU are undertaken on a joint team partnership approach (Euratom/IAEA), designed to minimise the burden on nuclear operators whilst ensuring that both Inspectorates are able to meet the goals derived from their respective safeguards regime. Notwithstanding, nuclear safeguards implementation has largely evolved since 1950s to the present day, which implies a continuous effort to adapt the Euratom/IAEA’s partnership agreements to state-of-the art approaches without excessive burden to nuclear operators. In this context, Integrated Safeguards (IS) can be seen as the latest evolution of the IAEA’s safeguards regime. IS introduces new key factors to provide credible assurance of both the non-diversion of nuclear material from declared nuclear activities and the absence of undeclared nuclear material and activities in the State as a whole. The adaptation of the EU States to this new regime has just been finished. Nevertheless, the way to IS has not always been straight and easy, mainly because several parties were involved in the discussions and the common minimum denominator was sometimes hard to achieve. In particular, discussions in Spain started by 2004 and finished at the end of 2009. An example is the Facility Specific Partnership Approach agreement for the Juzbado Plant, which entered into force recently in September 2009 and governs IS implementation at the Juzbado Plant. This paper briefly describes the evolution of safeguards during the last decades and the outstanding issues identified during the implementation of IS in Spain, as well as the principal features of the approach applied to the Juzbado Plant.

1. Introduction

The proliferation of nuclear weapons has been an issue of paramount concern for the international community since they were first used in 1945. Even then, in the beginning of the nuclear age, it was crystal clear that some form of international oversight on the nuclear materials had to be put forward in order to restrict their use to weapon’s manufacture. These initial efforts and the subsequent discussions at the United Nations in the early 1950s led to the establishment of the International Atomic Agency (IAEA or the Agency).

According to its Statute (1), which entered into force in 1957, the IAEA has the authority to ‘establish and administer safeguards designed to ensure that special fissionable and other materials, services, equipment, facilities and information made available by the Agency ... are not used in such a way as to further any military purpose’ (Art. III.A.5). Therefore, IAEA’s safeguards are to be intended as a set of measures designed and implemented to detect proliferation of nuclear weapons.
In the European Union (EU), safeguards are also applied under the 1957 Euratom Treaty (2). The overall objective of these safeguards is to allow the European Commission (EC) to satisfy itself that in the territories of Member States 'ores, source materials and special fissile materials are not diverted from their intended use as declared by the users' (Art. 77.a). According to this, EC's safeguards are to be intended as system of measures set up to detect diversion of nuclear materials by operators.

Thus, nuclear operators in the EU are subjected to the scrutiny of two different Inspectorates with different goals. To cope with this issue, safeguards in the EU are undertaken on a joint team partnership approach (Euratom/IAEA), as it will be seen in the following paragraphs.

2. Historical background of IAEA's safeguards system

Practical implementation of IAEA's safeguards is made by means of safeguards agreements involving individual States and the Agency. During the 1960’s a number of facility or material specific agreements were concluded and set out in a document titled 'The Agency’s Safeguards System', usually know by its publication number as (Information Circular) INFCIRC/66 (3). This ‘model’ originally dates from 1965 and outlines the basic system requirements, namely, to provide the Agency with reports showing receipts, transfers and inventories of nuclear material, and to allow the Agency inspectors to audit records and verify the nuclear material concerned. As it was said before, the main feature of these INFCIRC/66 type agreements is that they are facility or material specific and then, not comprehensive in their scope, which means that they do not oblige the State concerned to place under safeguards other nuclear materials or activities that are not specified in the agreement.

The issue of comprehensiveness is central to non-proliferation development. It moved to establish a worldwide regime to prevent the spread of the nuclear weapons developed during the 1960s and led, eventually, to agreement on the text of a Treaty on the Non-Proliferation of Nuclear Weapons in 1968 (4). This Treaty, known as the Non-Proliferation Treaty or NPT, entered into force in March 1970. Each Non-Nuclear Weapons State (NNWS) party to the NPT undertakes ‘not to acquire nuclear weapons or other nuclear explosive devices’ (Art. II), and to conclude a safeguards agreement with the Agency which covers all its nuclear material in all its peaceful nuclear activities (Art. III). Safeguards provisions of the NPT therefore establish comprehensive (or Full-Scope) safeguards coverage as the norm for NNWS. The situation for Nuclear Weapons States (NWS) party to the NPT (a NWS is defined in the Treaty as a State ‘which has manufactured and exploded a nuclear weapon or other nuclear explosive device prior to 1 January 1967’) is significantly different and is out of the scope of this paper.

The model for the comprehensive or full scope safeguards agreement required for NNWS parties to the NPT is described in a document titled ‘The Structure and Content of Agreements Between the Agency and States Required in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons’ (INFCIRC/153) (5). The provisions of INFCIRC/153, and those of the INFCIRC/66 model, are supplemented by so-called Subsidiary Arrangements to translate them into State and Facility specific practicalities. Detailed arrangements for each facility are contained in documents known as Facility Attachments.

3. Strengthened and Integrated Safeguards

During the last two decades several undeclared nuclear programmes have been unveiled, posing concerns on the effectiveness of the global non proliferation regime. These discoveries drew attention to the fact that the effectiveness of the international nuclear safeguards regime was, at least in part, dependent upon States acting in good faith in declaring their holdings of nuclear material and related activities. These challenges led the international community to drive forward a reinforcement of the IAEA’s safeguards system by
promoting the signature and entry into force of Additional Protocols (AP) to the safeguards agreements, based on a Model Protocol (INFCIRC/540) agreed by the IAEA in 1997 (6), which give further information and access rights to the IAEA inspectors to nuclear and fuel cycle-related installations and activities, aiming to enable the IAEA to depict a more precise picture of NNWS nuclear-related activities. A key feature of the strengthened system is that, for the first time, it gives the IAEA access to information on nuclear fuel cycle-related activities even where nuclear material is not involved.

The rationale behind the agreement of INFCIRC/540 was the understanding by States that the new measures would not simply be added to the ‘traditional’ measures implemented under comprehensive safeguards agreements, but that the two would be combined in an optimal manner to achieve the maximum effectiveness and efficiency to produce a so-called ‘integrated safeguards system’. Integrated safeguards involves, inter alia, a redefinition of safeguards implementation parameters, particularly for less sensitive nuclear material, with corresponding reductions in the level of inspection effort on such declared material. For NNWS with both a comprehensive safeguards agreement and an additional protocol in force, the IAEA has the ability under the strengthened safeguards system to draw conclusions, and hence provide credible assurance, of both the non-diversion of nuclear material from declared nuclear activities (the focus of comprehensive safeguards agreements) and the absence of undeclared nuclear material and activities in the State as a whole (the focus of additional protocols). Once such credible assurance has been obtained (and reaffirmed on an annual basis) for a State, an ‘integrated safeguards’ approach can be implemented in that State.

4. Safeguards in the European Union

As indicated above, European Commission Safeguards are applied under the 1957 Euratom Treaty, which establishes the requirements for the implementation of safeguards in the EU territories and outlines the means by which the safeguards objectives of the Treaty are to be achieved, namely, to provide the EC with Basic Technical Characteristics (BTCs) describing the location and intended activities of the installations, to keep and report nuclear material accountancy records, to entitle the Commission to inspect installations, audit records and impose sanctions in the event of infringement of the Treaty safeguards obligations. The various reporting requirements are amplified in Regulation (Euratom) No. 302/05, which entered into force on 20 March 2005 (7).

EU Member States part of the NPT are then subjected to the scrutiny of both the IAEA and EC safeguards systems adding a surplus of complexity to the whole matter: INFCIRC/153 model agreements are to be applied between the Agency and a State, whereas in the EU there are 27 Member States (2 of which are NWS) and the EC, and safeguards provisions are to be applied in the whole territory of the EU in an homogeneous manner. Aiming to take into account these particularities, the safeguards agreement INFCIRC/193 (8) was concluded in by the NNWS members of the EU, Euratom and the Agency. This agreement, which entered into force in 1977, includes provisions to apply safeguards on a joint team partnership approach (Euratom/IAEA) which it is designed to minimise the burden on nuclear operators whilst ensuring that both Inspectorates are able to meet the goals derived from their respective safeguards regime.

Furthermore, in 2004 the European Commission (EC) started a process to implement a broad reform of the approaches under which Euratom Treaty safeguards are implemented, aimed also to improve the efficiency of the system, while maintaining, or even improving, its effectiveness. The reform was first announced nearly at the same time that the AP to the safeguards agreement of EU Member States entered into force and in coincidence with the 2004 enlargement of the EU. The impact of the reform was largely discussed within the EU, particularly in connection with the implementation of the IAEA safeguards agreements. In
parallel, negotiations were started with the IAEA to pave the way towards the application of IS EU-wide.

5. The Spanish approach to Integrated Safeguards: the Juzbado case

Spain joined the EU as Member State in 1986, is a NNWS party of the NPT since 1987 and adopted an Additional Protocol in 2003. Therefore, the Spanish nuclear facilities are subjected to the joint IAEA/Euratom inspection scheme provided by the Safeguards Agreement INFCIRC/193 and to the provisions of INFCIRC/540. This means that Spain has both a comprehensive Safeguards Agreement and an Additional Protocol in force since 2004, which paved the way to Integrated Safeguards once a Broader Conclusion was drawn and the implementation procedure for the different installations was approved by the Agency. The Partnership Approach Paper for the ENUSA’s Juzbado LEU Fuel Fabrication Plant was the first and toughest issue to address.

In this context, the MITYC, the Spanish competent authority for safeguards, and ENUSA, in line with their traditional cooperative and transparent policy in relation to safeguards implementation, worked jointly for two years with the IAEA and the DG-TREN of the EC to design and later implement a six months field trial of a Short Notice Random Inspections (SNRI) scheme, tailor-made to suit the ENUSA’s Juzbado LEU Fuel Fabrication Plant operational constrains. The implementation arrangements of this first of a kind field trial within the EU was agreed upon by the parties in January 2007 and took place from March to September 2007. Lessons learned from the trial were taken on board to further refine the Juzbado plant’s SNRI scheme, with the aim of agreeing on its permanent implementation.

The basic objective of SNRI is to provide full coverage of the nuclear material involved in domestic transfers through random selection of the timing of the inspections. It was in 2004 when the IAEA first announced to the State authorities and ENUSA its wish to introduce the SNRI’s concept in the Low Enriched Uranium plants of the EU, in the context of the redefinition of its safeguards system. Developments around Euratom Treaty safeguards reform slowed down the negotiations to conclude the agreement for introducing the SNRI concept in Juzbado plant. At first, the EC was not much enthusiastic with SNRI schemes, since one of the main objectives of the reform was to save resources by focusing only on Euratom Treaty obligations, paving the way to an effective separation of Euratom and IAEA safeguards activities. However, from the onset of the negotiations, the Spanish side made very clear that there would not be possible to reach an agreement unless joint implementation of all routine Euratom and IAEA safeguards activities was envisaged within a single SNRI scheme, as required by the Protocol to the INFCIRC/193 safeguards agreement.

On the other hand, the IAEA did not facilitate much a meaningful participation of the EC in the SNRIs, rejecting to plan in common the inspections and even to give the Commission earlier notice than to the operator and the State. In spite of these difficulties, after more than one year of hard negotiations an agreement was reached, in which the IAEA would give 2-hour advance notice to the operator, the EC and the State authorities, and would be allowed to access the plant alone during the first day of the inspection to carry out very specific activities, on the condition that EC inspectors would arrive at the Juzbado Plant 24 hours later to conclude jointly the rest of the inspection activities agreed upon. Even though letters ratifying the agreement were exchanged, this scheme was never implemented as such because the EC claimed that 24 hours was not enough time to reach the plant with a well-prepared set of inspection activities. Moreover, the position of the EC matched better the Spanish side position, which from the very beginning had insisted that the Commission ought to have a genuine participation and not to join the inspection simply as an ‘invited’ party.

A second round of negotiations was started in mid-2006. In view of the immovable position of the IAEA and the EC, after the first contacts the Spanish side tabled a new proposal, in which any of the two Inspectorates could arrive at Juzbado 48 hour after the inspection was
triggered by the other. After completing the activities of the first inspection day, the inspectorate that triggered the inspection would remain inactive until the other inspectorate reached the plant. Of course this proposal was not the best solution in terms of efficiency, but the best is enemy of the better. This was the only possible solution to match all the parties’ wishes. The IAEA and the EC finally agreed to put to test the scheme in a six-month field trial.

The field trial took place from the 1st of March to the 1st of September 2007. Although there never was any official assessment of the trial neither from the IAEA nor from the EC, by the end of July 2007 a meeting between the IAEA, the EC, the State and ENUSA took place in Salamanca on the margins of the annual PIV to have a preliminary discussion on the results of the trial and to agree a way ahead for the rest of the year. All the parties concurred that the three SNRI foreseen in the field trial had gone off without noticeable difficulties and with satisfactory results. The two Inspectorates had been able to reach the plant in time and ready to carry out the inspection activities agreed upon. The organisation of ENUSA, heavily improved to handle the new inspection scheme, worked to satisfaction. Only minor issues regarding the exchange of information and some improvement in the interface between IAEA and EC for triggering the inspections were identified as matter for further developments.

ENUSA was particularly satisfied with the outcome of the trial and expressed its readiness to go ahead with a permanent implementation of the agreed SNRI scheme, which foresaw 4+1 short notice inspections plus 1 annual PIV. The IAEA and the EC could not agree on the operator’s request since the development and the results of the trial were to be subject of further assessment, with the aim of shaping the procedure taking into account the experience gained. However, giving the fact that for the rest of the year no routine inspection had been planned, it was agreed to extent the trial, under the conditions agreed upon, until the next PIV or until a revised procedure was available for permanent implementation of the SNRI scheme at Juzbado.

Negotiations were resumed in early 2008 and endured until January 2009. At the same time discussions between the EC and the Agency on the practical arrangements for IS implementation in EU Member States took place, aiming to produce generic Facility-Type Partnership Approach papers as a basis to issue more detailed Facility Specific papers on which the particularities of each installation were taken into account. In the case of Spain, the decision was taken to issue a specific paper covering all the NPP and another for the Juzbado FFP, being the rest of nuclear installations covered by the generic papers agreed upon as indicated above.

During the last round of negotiations for the Juzbado scheme, the mains topics of discussion were the information to be reported in the mail-box daily notification system and the retention time, the later being the hottest issue to be addressed. Retention time is to be understood as the time between the receipt and processing of items or their production and packing for shipment. The standpoint of the Agency was to ask for 48 hours of retention time, position unacceptable for ENUSA as it may seriously affect to the production schedule. The common denominator found for this issue was to accept 48 hours as normal retention time, but ENUSA would keep the right to decide whether to apply shorter retention times depending on the status of the production, with the requirement to report the fact 48 hours in advance via the mail-box system and, in case of inspection, include in the sampling plan the assemblies packed 48 hours prior to the arrival of inspectors. The detailed information to be included in the mailbox system was also finally agreed upon the basis of minimum interference with the existing operator’s information management system. The main features of the agreed scheme are:

- The maximum number of inspections will be 4+1 per year plus 1 PIV. The EC will be always present in the inspections triggered by the Agency.
- The inspectorate triggering the SNRI will notify the rest of the parties with 24 hours advanced notice. The communications will be delivered only on Juzbado’s working days an between 9:00 to 9:30h.
- ENUSA will provide the EC and IAEA with detailed daily information on production, receipts and shipments, which will be delivered via an encrypted email-box system.
- ENUSA has the right to apply retention times shorter than 48 hours. If this is the case, information on the items/batches affected will be included in the daily report. In case of inspection, the assemblies packed 48 hours prior to the arrival of the inspectors will be included in the sampling plan and thus, could be subjected to verification.

The Juzbado Specific Partnership paper was finally countersigned by the IAEA and the EC and officially distributed among all the involved parties by mid July 2009. On September 1st, 2009 the new system entered into force at the Juzbado Plant.

During the last term of 2009, discussions among the EC, IAEA, MITYC and Nuclear Power Plants’ (NPP) operators led to the Spanish NPP Specific Approach, which was countersigned and distributed among the involved parties in December 2009. This closed all the open issues to finalize the implementation of Integrated Safeguards in Spain.

6. Conclusions

Provided that no indication had been found that would give rise to a possible proliferation concern, the Agency drew a Broader Conclusion for Spain in 2008, stating that all nuclear material in the State has remained in peaceful activities as thus, Spain would be in a position to join the Integrated Safeguards regime once all the discussions referred to above were closed, as it happened to be at the end of 2009. Finally, Spain joined Integrated Safeguards on January 1st, 2010.

As it has been explained above, the way to Integrated Safeguards in Spain has not been straight and easy, mainly because several parties were involved in the discussions with different objectives in mind, so that the common minimum denominator was sometimes hard to achieve. Nevertheless, the outcome of multilateral discussions always give broader horizons to the actors involved and in this case, lessons learned during the pathway have given ENUSA the opportunity to improve its organization in terms of efficiency and capability of response to inspector’s requirements. On the other hand, the experience gained by the EC and the IAEA from the 2007 SNRI field trial at Juzbado put the basis to the design of the rest of SNRI schemes for other LEU plants in the EU.

7. Acknowledgements

The first part of this paper has been written with material available from the Integrated Safeguards Working Group of the European Safeguards Research and Development Association (ESARDA).

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NUCLEAR MATERIAL ATTRACTIVENESS: AN ASSESSMENT OF MATERIAL ASSOCIATED WITH A CLOSED FUEL CYCLE

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ABSTRACT

This paper examines the attractiveness of materials mixtures containing special nuclear materials (SNM) associated with the various processing steps required for a closed fuel cycle. This paper combines the results from earlier studies that examined the attractiveness of SNM associated with the processing of spent light water reactor (LWR) fuel by various reprocessing schemes and the recycle of plutonium as a mixed oxide (MOX) fuel in LWR with new results for the final, repeated burning of SNM in fast-spectrum reactors: fast reactors and accelerator driven systems (ADS). The results of this paper suggest that all reprocessing products evaluated so far need to be rigorously safeguarded and provided moderate to high levels of physical protection. These studies were performed at the request of the United States Department of Energy (DOE), and are based on the calculation of "attractiveness levels" that has been couched in terms chosen for consistency with those normally used for nuclear materials in DOE nuclear facilities. The methodology and key findings will be presented. Additionally, how these attractiveness levels relate to proliferation resistance (e.g. by increasing impediments to the diversion, theft, or undeclared production of SNM for the purpose of acquiring a nuclear weapon), and how they could be used to help inform policy makers, will be discussed.

1. Introduction

The United States Department of Energy (DOE) requested an assessment of the attractiveness, from an international safeguards and domestic physical protection perspective, of the special nuclear materials (SNM) (i.e., Pu, 233U, and 235U), alternate nuclear materials (ANM) (i.e., 237Np and Am), and other actinides that have a critical mass (e.g., Cm) that are associated with reprocessing and are handled in forms largely decontaminated of fission products. Each potential malefactor is unique in the material to which he has access and in the degree of sophistication he could utilize in weaponizing the material. Collectively, proliferant States and sub-national groups could consider a broad spectrum of SNM and ANM to be attractive for use in a nuclear explosive device. Although earlier studies [1,2] developed figures of merit (FOM) that were intended to explain the attractiveness or preferences for a range of nuclear materials across a span of credible nuclear adversaries, this study exclusively uses FOM1, which indicates the broadest range of nuclear materials that can credibly be used in a nuclear explosive device.

A credible nuclear threat from a sub-national group is different than that from a proliferant State. On the one hand, the perceived threat from a sub-national group is more dependent upon the fact that a device may produce any nuclear yield than it is upon the actual amount of yield. Even in a low technology, low quality device, any nuclear yield will, in most cases, exceed the conventional explosive yield. Thus, any device capable of generating a nuclear yield in the hands of a sub-national group would meet their requirements. On the other hand, a proliferant State is likely to have a preference for materials that are more easily and efficiently fabricated into higher yield nuclear weapons than those materials of interest to a
sub-national group. All SNM and ANM should be protected and safeguarded according to the highest level of attractiveness derived from both of these threats.

The point at which the nuclear explosive energy exceeds the conventional explosive energy is the point at which there is a potential nuclear threat. This point is also known as a threshold nuclear yield. The primary factors of material attractiveness are the bare critical mass, the internal heat generation, and the radiation dose rate [3].

2. Methodology

The metric used herein is given in Eq. (1) and is applicable for evaluating the attractiveness of SNM or ANM for a sub-national group, for most of the less advanced proliferant nations, or for a technically advanced proliferant State. The latter would be capable of building nuclear devices that assemble very rapidly to limit the impact of pre-initiation [4]. For a sub-national group pursuing nuclear terrorism any nuclear yield is acceptable so pre-initiation is not a significant issue. These generic cases are evaluated using the following formula:

\[
FOM_1 = 1 - \log_{10}\left(\frac{M}{800} + \frac{Mh}{4500} + \frac{M}{50}\left[\frac{D}{500}\right]^{\frac{1}{2}}\right)
\]  

(1)

where \(M\) is the bare critical mass of the metal in kg, \(h\) is the heat content in W/kg, and \(D\) is the dose rate of 0.2·\(M\) evaluated at 1 m from the surface in rad/h.

The basic concept of FOM1 is to relate candidate nuclear material to accepted standards. The well-established standards are: 1) the threshold for low enriched uranium (i.e., \(^{235}\)U enrichment less than 20%), 2) radioisotope thermoelectric generator plutonium (i.e., \(^{238}\)Pu enrichment greater than 80%), and 3) a self-protecting dose rate (i.e., 500 rad/h at 1 m). Historically, the self-protecting dose rate was assumed to be 100 rem/h at 1 m [5]. Upon recent technical review [6,7], it has been increased to 500 rad/h at 1 m.

FOM1 was reviewed by nuclear weapons experts at both LANL and LLNL. While it was determined that there are a number of smaller factors that are not captured, it was agreed that FOM1 captures the dominant factors quite nicely in an unclassified format.

Table 1 gives the meaning of FOM1. To make a material unattractive for use in a nuclear device, FOM1 must be less than 1. Table 1 does not distinguish between degrees of proliferation resistance that might characterize a nuclear material or mixture. Table 1 reflects the fact that while a particular nuclear material might be preferable for use in a nuclear weapon or explosive device, the design and construction of effective nuclear weapons from any of the materials with FOM1 >1 is theoretically possible. For example, plutonium from typical civilian used fuel could be used in a nuclear device [5]. As this paper will show, plutonium with \(^{239}\)Pu content ranging above 90% (often characterized as low proliferation resistance) or between 50% and 20% (often characterized as high proliferation resistance) both have a FOM > 1. The fact that potential proliferant States or sub-national groups might “prefer” one material over another should not imply that either material in question is “proliferation-proof,” or that any reduction in international safeguards and national physical protection requirements can be justified.

<table>
<thead>
<tr>
<th>FOM1</th>
<th>Weapons Utility</th>
<th>Attractiveness</th>
<th>Attractiveness Level [8]</th>
</tr>
</thead>
<tbody>
<tr>
<td>&gt; 2</td>
<td>Preferred</td>
<td>High</td>
<td>~B</td>
</tr>
<tr>
<td>1-2</td>
<td>Attractive</td>
<td>Medium</td>
<td>~C</td>
</tr>
<tr>
<td>0-1</td>
<td>Unattractive</td>
<td>Low</td>
<td>~D</td>
</tr>
<tr>
<td>&lt; 0</td>
<td>Unattractive</td>
<td>Very Low</td>
<td>~E</td>
</tr>
</tbody>
</table>

Table 1. The Meaning of FOM1 When Applied to Metals or Alloys
The term fast spectrum reactor (FSR) encompasses a large class of reactors with different physical characteristics (e.g., coolant or burn-up). To capture the effect of this variability, this study analyzed actinide compositions from Ref. 9, 10, and 11. Discharge compositions were subsequently aged (i.e., allowed to decay) using ORIGEN2.2 [12] calculations.

3. Results

FSR (i.e., fast reactors and accelerator driven systems (ADS)) recycle all of their discharged transuranic elements, TRU. When this recycled TRU is fabricated into fresh fuel, any lost reactivity is compensated by topping the recycled TRU with higher-reactivity TRU from used uranium oxide (UOX) or mixed oxide (MOX) fuel discharged from light water reactors (LWR). Used fuel is characterized by its burn-up, expressed in MW·d/kg of initial heavy metal, and its age at the time of reprocessing. FOM₁ results for candidate topping materials are given in Figure 1 [1,2]. For comparison the FOM₁ of weapons-grade (WG-Pu) and reactor-grade (RG-Pu) plutonium, high (HEU) and low (LEU) enriched uranium, ²³⁷Np, ²³₃U contaminated with 10 ppm ²³²U, and a 80:20 mixture of ²³⁸Pu and ²³⁹Pu are shown on the left side of each figure. The FOM₁ of Pu and TRU decreases with increasing burn-up, because the concentration of ²³⁹Pu decreases and the concentration of ²³⁸Pu, which is an intense heat source, increases with increasing burn-up. The FOM₁ for TRU is strongly dependent on used fuel age (i.e., time since discharged); because ²⁴²Cm and ²⁴⁴Cm are intense heating sources and quickly decay away (their half lives are 163 days and 18 years, respectively).

Figure 1. FOM₁ Versus Burn-up of Used-UOX Source of Topping Candidates Cooled 1 (left) and 10 (right) Years: RG-Pu (red curve), Pu from Used MOX (blue curve labelled MOX-Pu) Burned to 60 MW·d/kg, and TRU (green curve). RG-Pu is the MOX plutonium at charge. The letters H, M, and L denote high, medium, and low attractiveness, respectively (see Table 1). Included for reference are the following data points: a – LEU (20%), b – HEU (93%), c – ²³⁷Np, d – ²³³U (10 ppm ²³²U), e – WG-Pu, f – RG-Pu (45 MW·d/kg and cooled 10 years), and g – ²³⁸Pu/²³⁹Pu (80:20).

The first set of FSR results is presented in Table 2 and is based on the isotopic compositions given in Ref. 9 for three transmutation approaches: 1) 1XT – an advanced liquid metal reactor (ALMR) that is topped with TRU from used MOX fuel (2nd MOX pass); 2) 3M – an ADS topped with TRU from used advanced LWR (ALWR) fuel; and 2) 3T – an ALMR topped with TRU from used ALWR fuel. These FSR use metal fuel, are sodium cooled, and recycle the TRU in their discharged fuel after cooling for 2 year. The fuel at charge and discharge for the ALMR is unattractive because of a high uranium concentration (see Ref. 1 and 2). However, the TRU from used FSR fuel that has been cooled for 2 years is moderately attractive in all three cases, as is the topping material. The 3M charge is also moderately attractive because the ADS is driven and doesn’t need uranium to maintain criticality.

Because the fuel in Table 2 is from an equilibrium cycle, the effect of cycle number was explored and is shown in Figure 2 for an ADS that burns fuel to 260 MWt·d/kg with conversion ratio (CR) ~ 0.25; tops with LWR discharges burned to 60 MWt·d/kg and cooled for 10 years; and immediately recycles discharged TRU. The FOM₁ of the discharged fuel reaches an equilibrium that is consistent with the 3M value in Table 2 within four cycles. The TRU from 2-yr old used ADS fuel is moderately attractive. The first charge is highly attractive.
The effect of varying CR on material attractiveness at equilibrium is shown in Figure 3. To decrease CR in the sodium-cooled, metal-fuelled fast burner (CR < 1) design used in Ref. 10 with collocated reprocessing, pin radii were reduced to reduce the fuel volume fraction, resulting in reduced uranium concentration and increased burn-up, all to maintain criticality. To breed (i.e., CR > 1), the reactor core was divided into TRU-bearing driver and uranium-only blanket regions (the average over the entire core, including any blankets, is shown as "<core>") in Figure 3). To increase breeding, more blankets and larger driver pin radii were used. No major reactor modifications were required to change CR. The fuel at charge and discharge is unattractive with the exception of the charge at CR = 0.01, because Cm provides a heat deterrent for CR < 1 and uranium dilutes the fissile isotopes for CR > 1. TRU from used fuel is moderately attractive for CR < 1 and highly attractive for CR > 1, because the Cm concentration decreases and the $^{239}$Pu concentration increases as CR and $^{238}$U concentration increase and the burn-up decreases. The $^{239}$Pu concentration in the blanket is of particular concern, because it reaches a maximum concentration of 97.4% of the TRU at CR = 1.7. Consequently, the SNM associated with all FSR must be safeguarded.

![Table 2. The FOM₁ of the Fuel Actinides (Charge and Discharge) and TRU from Aged Used Fuel Associated with Three Transmutation Approaches [9] and the Corresponding Topping Material.](image)

<table>
<thead>
<tr>
<th>Case</th>
<th>Reactor Type</th>
<th>Burn-up (MWt·d/kg)</th>
<th>CR</th>
<th>TRU Consumed (%)</th>
<th>State (– Age)</th>
<th>U (%)</th>
<th>FOM₁</th>
</tr>
</thead>
<tbody>
<tr>
<td>1XT</td>
<td>ALMR</td>
<td>119</td>
<td>~0.5</td>
<td>18</td>
<td>Charge</td>
<td>62</td>
<td>0.54</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Discharge</td>
<td>64</td>
<td>0.09</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>TRU – 2 yr</td>
<td>0</td>
<td>1.06</td>
</tr>
<tr>
<td>3M</td>
<td>ADS</td>
<td>273</td>
<td>~0.25</td>
<td>29</td>
<td>Charge</td>
<td>2</td>
<td>1.17</td>
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<tr>
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<td></td>
<td></td>
<td></td>
<td></td>
<td>Discharge</td>
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<td>0.46</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>TRU – 2 yr</td>
<td>0</td>
<td>1.01</td>
</tr>
<tr>
<td>3T</td>
<td>ALMR</td>
<td>118</td>
<td>~0.5</td>
<td>19</td>
<td>Charge</td>
<td>68</td>
<td>0.82</td>
</tr>
<tr>
<td></td>
<td></td>
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<td></td>
<td></td>
<td>Discharge</td>
<td>70</td>
<td>0.30</td>
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<tr>
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<td></td>
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<td></td>
<td></td>
<td>TRU – 2 yr</td>
<td>0</td>
<td>1.40</td>
</tr>
<tr>
<td>1XT Topping</td>
<td>ALWR</td>
<td>51</td>
<td>—</td>
<td>—</td>
<td>TRU – 2 yr</td>
<td>0</td>
<td>1.17</td>
</tr>
<tr>
<td>3M or 3T Topping</td>
<td>ALWR</td>
<td>51</td>
<td>—</td>
<td>—</td>
<td>TRU – 10 yr</td>
<td>0</td>
<td>1.80</td>
</tr>
</tbody>
</table>

4. Conclusions

Lowing CR affords some security benefit for reprocessed TRU, but no safeguards benefit because the operator, design permitting, may be able to change the CR. However, we have
not identified a "silver bullet" technology that would eliminate safeguards and security issues. None of the proposed flowsheets examined to date justify reducing international safeguards or physical security protection levels. All of the reprocessing or recycling technologies evaluated to date still need rigorous safeguards and high levels of physical protection.

Figure 3. The Burn-up (left) and FOM, at Charge and Discharge (middle) and for TRU from Used Fast Reactor Fuel at Various Ages at Time of Reprocessing (right) Versus Conversion Ratio for the Blanket, Driver, and Averaged Core (CR > 1) and for the Entire Averaged Core (CR < 1). See Figure 1 for explanation of H, M, and L and symbols a – g.

5. Acknowledgements

The authors would like to thank Ed Hoffman of Argonne National Laboratory, Sam Bays and Steve Piet of Idaho National Laboratory, and Holly Trellue of Los Alamos National Laboratory for sharing their isotopic composition data.

6. References


8 “Nuclear Material Control and Accountability,” U. S. Department of Energy manual DOE M 470.4-6 Chg 1 (August 14, 2006).


Safeguarding the nuclear fuel cycle is a key aspect of proliferation resistance. The application of extrinsic measures to achieve the detection and timeliness goals has a strong relationship with the intrinsic design features of facilities.

By taking into account design features that facilitate the implementation of international safeguards very early in the design phase, a concept known as “safeguards by design” (SBD), the overall process can be made more effective and efficient with benefits to all the involved stakeholders.

At the end of 2008, the IAEA launched a new task on “Guidance for Designers and Operators and Measures to facilitate the implementation of Safeguards at Future Nuclear Cycle Facilities”, with contributions by EURATOM and Member States Support Programmes, with the aim to formulate SBD Guidelines to designers and operators. The main driving force for this new activity is the foreseen growth in the number of nuclear power generating facilities, and the corresponding increase in other fuel cycle activities such as fuel fabrication and enrichment, all of which require the application of safeguards. This will pose an additional workload to the IAEA, whose resources however, won’t increase at a comparable rate, thus demanding for more efficiency.

This paper will develop on the achievements of the IAEA task in 2009, and the contents of the first document of the IAEA Safeguards by Design series, as well as on methodological developments.

1. Introduction

Safeguarding the nuclear fuel cycle is a cornerstone of proliferation resistance. Design choices are the result of an optimal compromise among economic, operational, safety and security factors, which should also take into account the requirements for application of safeguards at an early stage. However, international safeguards are often introduced too late into a consolidated facility design, resulting in costly re-design and project delays. Instead, by taking into account design features that facilitate the implementation of safeguards very early in the design phase, a concept known as “safeguards by design” (SBD), the overall process could be made more effective and efficient with benefits to all the involved stakeholders.
To achieve this, the IAEA started developing SBD as an approach whereby safeguards are fully integrated into the design process of a nuclear facility, from the initial planning through design, construction, operation, and decommissioning.

With this aim, on October 28-31, 2008, the IAEA convened the “Facility Design and Plant Operation Features that Facilitate the Implementation of IAEA Safeguards” workshop, with participants from Member States, the European Commission, nuclear industry, and the IAEA, whose results are reported in [1]. Based on its conclusions, the IAEA started to review the overall framework and timeline of interaction among the various stake-holders, including the update of the existing safeguards documentation, and to develop a series of new SBD guidelines, which should be concluded in 2011 covering all main facility types, i.e. reactors, enrichment, fuel fabrication, reprocessing plants and final repositories. This would apply to innovative facility designs, but also to the so-called evolutionary ones, that is still based on existing facilities’ design, but with a consistent degree of improvements.

The EURATOM Support Programme undertook the task of drafting the first high level guidance document of the SBD Series, which is the object of this paper.

2. SBD process and IAEA Safeguards approaches

The SBD process is a complex multi-disciplinary interactive process, optimizing the intrinsic features of the nuclear facility (scope, process, materials, planning) with safeguards, safety, security, economics and sustainability requirements.

Before developing about the SBD process, it is worthwhile recalling that the IAEA’s safeguards system for verification of declarations is implemented by:

- design information verification (DIV);
- nuclear material accountancy and control (NMAC);
- containment and surveillance (C/S) measures to monitor access or movements of the nuclear material.

2.1 Design information

Much of the information required by the IAEA to plan the safeguards approach is provided with the design information questionnaire (DIQ) when a new facility is planned. The DIQ is followed by the design information verification (DIV), performed during a visit and not to be confused with the inspections for verification of nuclear materials.

2.2 Nuclear material accountancy and control

The purpose of the nuclear material accountancy and control (NMAC) system is to establish the quantities and locations of the nuclear material present in a nuclear facility and the changes that take place in the inventories of nuclear material. The essential elements of such an accounting system and the associated inspection procedure are that:

- The operator identifies, counts or measures the nuclear material in the facility, maintains inventory records, and submits these reports to the IAEA through the Regional and/or State’s Systems of Accounting and Control (R/SSAC)
- IAEA inspectors visit the facility to verify the inventories and inventory changes, to determine the validity of the operator’s accountancy declarations and the correctness of the reports made to the IAEA.

Under the Integrated Safeguards foreseen by the Additional Protocol, inspections may be unannounced, upon short-notice or for complementary access.
2.3 Containment and surveillance

Containment and surveillance (C/S) is based on a system of video-surveillance installed by the safeguards inspectorates, or belonging to the operator in shared use. The frequency of surveillance can range from continuous to intermittent sampling, based on the estimated operating time required to accomplish a diversion. Seals of various types are installed by inspectors to maintain continuity of knowledge and insure against unverified opening of possible diversion routes not monitored by cameras.

3. Safeguards by Design process

The Safeguards by Design process is outlined next.

3.1 SBD stake-holders

Six groups of stake-holders are identified for the Safeguards by Design (SBD) process, interacting at various times and in various ways.

The first is of course the IAEA, whose main expected benefits from SBD would be cost savings, improvement of inspection conditions compared to present standards and finally implementation of more effective and efficient safeguards, which is also the goal of the Regional and/or State System for Accounting and Control (R/SSAC).

The main motivations for the operator would be the minimisation of the impact of safeguards on the overall cost of the facility and the reduction in safeguards intrusiveness during the operation of the plant. The integration of the SBD process with existing processes would avoid construction delays and reduce construction costs caused by the addition, or back-fitting, of safeguards equipment to existing or new but frozen-design plants.

For the same reason, designers should be made aware already at the inception and design phase of the future safeguards requirements such as penetrations, camera views, needs for sealing, potential locations of unattended measurement stations, cabling pre-installation needs and more.

Not less important, among the stake-holders are equipment suppliers and the scientific community, all benefiting from an early-starting and interactive process.

3.2 SBD process timeline

The different SBD stake-holders should interact continuously through the design, construction, operation, and even decommissioning phases of a facility. A comprehensive and interactive SBD process should already begin during the R&D phase, with an exchange of information between the IAEA, the R/SSAC, the operator, and the designers:

- The IAEA should provide designers with a statement of safeguards requirements as early as possible when plans for a new facility are communicated;
- The R/SSAC, in turn, should provide the IAEA with preliminary design information;
- High level safeguards guidelines specific for the facility type in question, should be available to the designers, for a first concept to be presented to the operator, the R/SSAC and to IAEA;
- Low-level details and requirements should then be addressed at the beginning of the facility design and construction process, specifying the safeguards system principles and test acceptance criteria.
As a result, the development of the safeguards approaches and their elements should ideally match the new facility’s features. This would apply to innovative facilities, but also to so-called evolutionary facilities, still based on existing facilities’ design, but with a consistent degree of improvements. The experience of operating facilities must be used by all parties for the purpose of identifying and compiling the best state-of-the-art design features and safeguards practices.

### Facility Phases

<table>
<thead>
<tr>
<th>Facility Phases</th>
<th>R/SSAC to IAEA</th>
<th>IAEA</th>
<th>Designer and operator</th>
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<td><strong>R&amp;D phase</strong></td>
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<td>Information under AP</td>
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<td>Best safeguards practices compilation</td>
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<td>Preliminary design information</td>
<td>Safeguards requirements; High level SG guidelines</td>
<td>Preliminary concept</td>
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<tr>
<td><strong>Preliminary design</strong></td>
<td></td>
<td>Medium level SG guidelines</td>
<td>Preliminary design</td>
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<td>Safeguards approach</td>
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<td></td>
<td>Detailed SG guidelines</td>
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<tr>
<td><strong>Final design</strong></td>
<td>Draft facility attachment</td>
<td></td>
<td>Final design</td>
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<tr>
<td><strong>Construction</strong></td>
<td></td>
<td>DIV Safeguards systems installation</td>
<td></td>
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<tr>
<td><strong>Commissioning</strong></td>
<td>Final facility attachment</td>
<td>DIV</td>
<td></td>
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<tr>
<td><strong>Operation</strong></td>
<td></td>
<td>DIV and Inspections</td>
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</table>

Tab 1: Safeguards by Design process and interaction timeline

### 4. SBD implementation

Safeguards by Design can be addressed by improving the facility’s design and/or improving the techniques by means of which safeguards are implemented. Design “intrinsic” features, including of course the layout of the facility, can facilitate the implementation of “extrinsic measures”, i.e. the safeguards approaches we saw above, in ways that will be further developed.

Improving the facility design is a complex multi-attribute exercise, because each design feature has a primary goal that can conflict with those of other features. The overall design is thus the result of the complex interaction among all the features, coping with safety, security, safeguards, as well as economics and sustainability requirements.

### 5. Methodologies for SBD

The Safeguards by Design process requires an understanding by designers and operators of proliferation resistance and its underlying principles. Relevant studies and methodologies developed by the Proliferation Resistance and Physical Protection (PR&PP) Working Group of GEN-IV [2], and IAEA INPRO [4], can be instrumental to the SBD process. Both methodologies include in particular the concept of safeguardability, defined as “a concept that reflects the degree of ease with which a facility can be put under safeguards” [3] which could be used in the conceptual and preliminary design phases, as it deals with the system’s features and does not require a detailed design yet.

### 6. Safeguardability

The issue of how to tackle the problem of the Safeguardability of nuclear facilities existed since the foundation of the International Atomic Energy Agency, and during the last thirty years at least two different approaches emerged:
• developing new safeguarding techniques and equipment to enhance safeguards effectiveness and efficiency;
• providing guidelines for designers of new systems in order to enhance systems’ Safeguardability during early design stages.

Its goal is to provide system designers with a list of attributes to be taken into account at very early design stages in order to facilitate the implementation of International Safeguards. As first iteration, the single features can be looked at and grouped for their relevance to the safeguards components implemented (i.e. NMAC, DIV, C/S) and for their direct relevance to nuclear material management and plant operation.

The safeguardability relevance of the features (e.g. extent of access to the nuclear material, or type and form of the nuclear material) can be estimated by means of utility functions. The overall contribution of the various features is then aggregated by weighed multiplication of the results. Uncertainties should be taken into account, estimating how they propagate through the model. A list of design attributes and their safeguardability relevance are proposed in [5] and a more extensively developed model in [6].

7. Conclusions

The SBD process can certainly have a positive impact on the implementation of international safeguards and benefit its various stake-holders.

Various international fora are carrying out activities which can support the IAEA for SBD implementation.

The first draft of the SBD Guidelines has been provided by EURATOM to the IAEA, and it is expected that it will be improved also by the contribution of other support programmes.

Further steps will encompass the definition of low level guidelines specific for the various facility types.

8. References

A SELECTION OF RECENT ACHIEVEMENTS AND FUTURE CHALLENGES IN SAFEGUARDS R&D AS IDENTIFIED BY THE EUROPEAN SAFEGUARDS RESEARCH AND DEVELOPMENT ASSOCIATION

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ABSTRACT

In the year of its 40th anniversary, ESARDA can look back on a wealth of achievements in the European safeguards area, to which its members have substantially contributed. Also today, ESARDA is more active than ever, both due to an extended partnership (with many new members joining in recent years) and a pro-active attitude to tackle new and upcoming issues through its many Working Groups.

ESARDA constitutes today of 9 working groups: Non Destructive Analysis (NDA), Destructive Analysis (DA), Containment and Surveillance (C/S), Verification Technologies and Methodologies (VTM), Implementation of Safeguards (IS), Nuclear Material Accounting and Control Audit Group (NMACAG), Training and Knowledge Management (TKM), Editorial Committee (EDC) and the newly established Novel technologies & approaches (NT/NA). The Fuel Fabrications Plants WG has recently been stopped and part of the activities possibly moved into IS WG. Most of these working groups organise regular meetings in which they discuss the progress, achievements, problems, challenges and future needs for R&D. They jointly work out solutions to problems and stay abreast of the latest safeguards R&D aspects. This paper will provide an overview of some most relevant recent achievements of the ESARDA Working Groups and will provide an insight in the challenges identified for safeguards R&D in the near and medium term future.

1 International Nuclear Safeguards and ESARDA

ESARDA, the European Safeguards Research and Development Association, was created in 1969 when there was a need to ensure that the safeguards requirements arising from the Non Proliferation (NPT) and the European Atomic Energy (EURATOM) treaties could be technically fulfilled for a peaceful use of nuclear energy.

The 1969 agreement formed the framework for the activities of the safeguards research association, which was given its present name “ESARDA” in 1973. ESARDA’s main objective is to assist the European safeguards community with the advancement of safeguards, enhancing the efficiency of systems and measures, as well as investigating how new techniques can be developed and implemented. ESARDA also strives to fulfil an educational role and to reach the general public and, in particular, those interested in safeguards. ESARDA aims to bring together all those involved in safeguards, so that progress and continuous improvement in international safeguards can be achieved efficiently and to a professional standard. The principal issues are co-ordination of research, exchange of information and joint execution of R&D programmes.

ESARDA is participated by European organisations actively involved in the Research and Development of Nuclear Safeguards. Close contacts are held with international partners and organisations.
2 Membership and contributions

The current official Parties to ESARDA are 25 from 17 European countries:

AREVA (France), ATI (Austria), CEA (France), CNCAN (Romania), EDF (France), ENEA (Italy),
European Commission, FZJ (Germany), HAEA (Hungary), IKI (Hungary), IRSN (France), MITYC
(Spain), NNL (UK), NRI (Netherlands), NRPA (Norway), PAA (Poland), SCK/CEN (Belgium),
Sellafield (UK), SFOE (Switzerland), Springfields Fuels (UK), SSM (Sweden), STUK (Finland),
UKAEA (UK), VATESI (Lithuania), WKK (Germany). In addition, ESARDA counts four individual
members. The European Commission represents the European Atomic Energy Community
(EURATOM).

The ESARDA’s network includes national regulatory authorities (carrying out the controls),
operators of nuclear facilities (those being controlled), and research centres and universities
(carrying out the safeguards-related R&D).

Representatives from other world-wide organisations regularly take part in ESARDA’s activities,
amongst them international ones like the IAEA, and regional ones like ABACC, or national ones
like various US labs. About 230 contributors from Parties and external organisations are today
active in the Working Groups.

3 ESARDA Communication

Within the objectives of ESARDA, communication is very important, for it deals with the Bulletin
publication, the definition of Symposia programmes, and the web-site.
The ESARDA Editorial Committee has an important role in making all these activities a success.

3.1 Annual Meetings and Symposia

Open symposia provide an opportunity for collaboration and the exchange of scientific
information. The latest developments and different perspectives on tackling the challenges in the
field are presented and debated at the meetings, which traditionally take place in May.
Alternating with open symposia, a biennial ESARDA internal meeting takes place where all
ESARDA working groups convene and have the opportunity to discuss issues of common
interest in joint meetings. The proceeding are distributed to participants, and later made available
on the web-site.

3.2 ESARDA Bulletin and web site

The ESARDA Bulletin contains scientific and technical articles relating to safeguards and
verification are published. It includes a dedicated section for peer reviewed papers, in addition to
more general information and news on ESARDA and its members. On average two issues plus a
special issue are published every year. The web site features the latest issues of the Bulletin, as
well as scientific and practical information about the working groups and symposia. It also
contains also general information and reference materials relating to safeguards. It can be
reached at www.esarda.eu.

4 Working Groups

As mentioned and listed above, WGs are established to promote and undertake collaborative
R&D and information exchange activities in particular fields. The R&D activities are performed by
about 230 experts (members or observers of ESARDA).

4.1 Non Destructive Analysis Working Group (WG NDA)

The goal of the WG NDA is to provide the Safeguards Community with expert advice on Non
Destructive Analysis methods, procedures on standards and reference materials and on the
performance of NDA methods. This is achieved through a wide range of activities such as:

• Maintain a list of NDA methods and instruments currently used or under development for
accountancy and verification purposes.
• Determine the reliability of NDA methods where possible with inter-comparison exercises
and benchmarks of safeguards measurements.
• Advise EURATOM and IAEA on the implementation of new and improved methods and advise on areas where R&D is needed, consequently promote and coordinate R&D programs and assist in the development of new NDA methods in support of new safeguards requirements.
• Promote the production and correct use of Reference Materials.
• Assess and disseminate Performance Values for Uncertainties in NDA methods of nuclear material measurement.
• Promote the use of internationally agreed definitions and terminology in the reporting of measurement results.
• Promote cooperation with other working groups and the inspection authorities.

The major achievement of the NDA working group in 2009 is the special issue of the ESARDA Bulletin, number 42, November 2009, addressing the ESARDA Multiplicity Benchmark and the Good Practice Guide for the use of modelling codes in Non Destructive Assay.

Fig 1. Recent ESARDA Bulletins

4.2 Techniques and Standards for Destructive Analysis WG (WG DA)
In traditional nuclear safeguards applications, DA methods essentially serve for highly accurate element and isotope assay of nuclear material. In strengthened safeguards, highly sensitive analytical techniques have been developed, aiming at the detection of undeclared nuclear material or undeclared activities (e.g. through particle analysis). In a nuclear security context, a comprehensive and thorough analysis of samples of nuclear material can reveal important information on the history of the material. Such investigative radio-analytical techniques (often called nuclear forensics) are increasingly applied, not only to nuclear material intercepted from illicit trafficking, but also to samples taken during safeguards inspections (e.g. UF6 samples taken in enrichment plants and conversion plants, where impurities measurements provide information on consistency of declarations). Information inherent to the material may then be used to check consistency with declared operations and processes.

The activities of the WG DA reflect all of these requirements. One of the main achievements of the group had been the promotion of the concept of target values for measurement uncertainties, initially conceived in the late 1970’s. The concept matured over the past 30 years and several issues of the Target Values were jointly elaborated between the WG DA and special INMM committees.

4.3 Containment and Surveillance Working Group (C/S WG)
The WG C/S aims at providing the Safeguards Community with expert advice on Containment and Surveillance (C/S) instruments and methods and on their performance; and acts as a forum for the exchange of information on such instruments and methods, including unattended and
remote monitoring systems. The working group currently has 18 members and observers from R&D establishments, safeguards equipment manufacturers, safeguards inspectorates, plant operators, regulatory agencies, and ministries. Recent discussion topics and studies include:

- Performance and assurance of C/S instrumentation
- Guidelines for sealing, identification, and containment verification systems
- Enhanced Data Authentication
- XCam: IAEA's New Generation Surveillance System
- Interface between the EOSS electronic seal and video surveillance
- Trial of 3D laser scanning equipment in France
- Remote Data Transmission system (jointly operated TREN and IAEA)
- Technical Sheet on Laser Based Design Information Verification

4.4 Verification Technologies and Methodologies Working Group (VTM WG)

The WG VTM was founded in 2003. Its mission stems from the acknowledgment by ESARDA members of the need for a forum where to discuss on verification issues of multinational instruments as IAEA safeguards, CTBT, exports controls, chemical and biological weapons convention, arms control treaty, fissile material cut-off treaty, environmental treaty (Kyoto protocol), trying to provide the safeguards community elements with expert advice on modern verification technologies and methodologies and to act as a forum for the exchange of relevant information in this area. Individual working group members volunteer to prepare discussion and working papers, subgroups have been established, conferences and meetings with special topics are performed. VTM has an important explorative role for ESARDA. For example, in 2006 organised a first workshop on Export Control issues that since then became increasingly known to ESARDA members.

4.5 Implementation of Safeguards Working Group (IS WG)

The WG IS was created in 2000 with the objective to provide the Safeguards Community with expert advice on methodologies and approaches to integrate INFCIRC/193 and INFCIRC/540 measures and to present a forum for the exchange of information, views and experiences in that regard. Its members represent inspectorates, national authorities, operators and research centres active in the field of safeguards. The work has been mainly oriented towards practical issues of the implementation of the Additional Protocol (AP) and IS approaches taking into account the point of view of all parties involved in the implementation process, such as operators, as well as national, regional, and international inspectorates. IAEA's broader conclusion was recently drawn for nearly all of the EU member states and IS are fully implemented in EU. The future work will hence focus on issues of practical IS implementation in the EU member states, reflected in the new name “Implementation of Safeguards” WG, maintaining the same IS acronym witnessing for a success story.

4.6 Nuclear Materials Accounting and Control Audit Group (NMAC AG)

In order to provide DG TREN with advice how to practically implement the audit principle, the ESARDA Executive decided in 2008 to resume the WG formerly dealing with NMA activities with revised terms of reference. The new Group was formed in October 2008 to tackle the harmonization and clear interpretation of the audit criteria contained in the Commission Recommendation (2009/120/Euratom) and the identification and appropriate information exchange in support of NMAC audits. A document that specifically discusses the applicability of audits in item (e.g. reactors, storage facilities) and bulk facilities (e.g. reprocessing -, MOX-, enrichment facilities) as well as in small and research installations is under preparation was completed by end of 2009. NMACAG will hence stop its activities, and the implementation phase will be considered within the IS WG.

4.7 Training and Knowledge Management Working Group (TKM WG)

The WG TKM was created based on the needs of a “generation change” in the field of nuclear experts. The average nuclear scientist is well over 50 years old. More in particular the Nuclear Safeguards and Non-Proliferation issues are not dealt with in the standard academic curriculum for nuclear engineering. Education in the nuclear field still seems strongly influenced by national
history and dislikes to tackle the more international and interdisciplinary (even interfaculty) issues of Non-Proliferation. ESARDA courses, the latest of which was held in Ispra, Italy in March 2010, aim to complement nuclear engineering studies by including nuclear safeguards in the academic curriculum. For the last six years, as part of the international effort to promote nuclear knowledge, the WG TKM has been educating more than 250 young professionals and students about nuclear safeguards and non-proliferation. The TKM WG recently strengthened its international network by actively participating and promoting the international initiative for Nuclear Safeguards and Security Education and Training (NuSaSET).

4.8 Novel Technologies/Novel Approaches Working Group (NT/NA WG)
Stemming from VTM WG and recently approved by the Executive Board, NT/NA starts its activities as an horizontal WG.

5 Reflection Group 2010 and new directions for ESARDA
The new Reflection Group 2010 has started its work in January 2010 under the chairmanship of Michel Richard. As in the previous reviews of 1988, 1993 and 2000, the new group will assess the changes implemented following the recommendations of the Reflection Group 2000, and provide an opportunity to think ahead to how ESARDA might stimulate R&D in areas of concern that have emerged during the last decade. The work should be concluded by the end of 2010 and provided to the Steering Committee.

Many issues arose since the last reflections took place, and ESARDA should look at opportunities and needs. A European Union “Strategy against the proliferation of weapons of Mass Destruction” was formulated in 2003. DPRK’s nuclear tests have been a demonstration of how nuclear proliferation continues in the face of international treaty, and how technology and knowledge gaps can be circumvented by illicit import practices and trafficking, calling for improved export control and nuclear security.

Nuclear renaissance is announced to extend to new regions, bringing a requirement for the development of new legal frameworks and infrastructure. This may also generate an increased potential for nuclear proliferation as e.g. over 1700 subcontractors in 27 States are needed for a NPP presently under construction. Technology transfer increases the spread of common working standards for safety, security and safeguards, but also opens the possibility of clandestine replication of technology. Export control becomes then an even grater necessity, which in particular in EU-27 generates the need for improvements towards a more homogeneous licensing and enforcement system.

2010 is also the year of the NPT Review Conference, which offers the chance to rethink this key legal instrument in the light of new challenges. In May 2009 Mr. Mohamed ElBaradei, former Director General of the IAEA, stated that, in the near future, the number of nuclear weapons states could increase by 10 to 20 states. In line with this comes the renewed emphasis on nuclear disarmament called for by USA president in 2009 and the Fissile Material Cut-off. ESARDA must take a look ahead and see how it can best prepare for defining answers to a number of the issue raised above- i.e. at the threats and dangers of today and tomorrow. ESARDA’s vision from the year 1969 “to harmonize the R&D activities in the area of international safeguards and ensure a mutual exchange of information and technical assistance” can still be functional also today.

6 References
All the information presented can be retrieved from ESARDA on-line database on www.esarda.eu.
NUCLEAR MATERIAL ATTRACTIVENESS: AN ASSESSMENT OF MATERIAL FROM PHWR’S IN A CLOSED THORIUM FUEL CYCLE


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ABSTRACT

This paper examines the attractiveness of material mixtures containing special nuclear materials (SNM) associated with reprocessing and the thorium-based LWR fuel cycle. This paper expands upon the results from earlier studies [1,2] that examined the attractiveness of SNM associated with the reprocessing of spent light water reactor (LWR) fuel by various reprocessing schemes and the recycle of plutonium as a mixed oxide (MOX) fuel in LWR. This study shows that 233U that is produced in thorium-based fuel cycles is very attractive for weapons use. Consistent with other studies, these results also show that all fuel cycles examined to date need to be rigorously safeguarded and provided moderate to high levels of physical protection. These studies were performed at the request of the United States Department of Energy (DOE), and are based on the calculation of "attractiveness levels" that has been couched in terms chosen for consistency with those normally used for nuclear materials in DOE nuclear facilities [3]. The methodology and key findings will be presented.

1. Introduction

The FOM is an empirically derived formula that has been reported in previous publications [1, 2] as a metric to describe the weapons utility of nuclear material. In the context of safeguards and security requirements, the FOM can be equated to the concept of nuclear material attractiveness. The bounding case is referred to as FOM1, which is applied here to Th-based fuel cycles that are fuelled by 233U and/or Pu. FOM1 uses two physical parameters associated with the product material that is to be weaponized (e.g. bare critical mass and heat content) and one physical parameter that is normally associated with the source material from which the weapons usable nuclear material is derived (e.g. dose rate). There is a fourth physical parameter that is relevant to the desirability or preferrability of the material for use in weapons (e.g. intrinsic neutron rate), but it is generally not relevant to preventing the nuclear material from being used to make an effective nuclear weapon.

The metric used herein is given in Eq. (1) and is the bounding case for evaluating the weapons utility of special nuclear material (SNM) or alternate nuclear material (ANM) to various potential adversary groups.

\[
FOM_1 = 1 - \log_{10} \left( \frac{M}{800} + \frac{M h}{4500} + \frac{M D}{500} \right)^{\frac{1}{\log_{10} 2}}
\]

In this equation, \(M\) is the bare critical mass in kg, \(h\) is the heat content in W/kg, and \(D\) is the dose rate of 0.2·M evaluated at 1 m from the surface in rad/h.

In the context of safeguards, the bare critical mass and the heat content are of the purified element after it has been removed from the used fuel. In the context of security, the bare
critical mass and heat content of an impure alloy that is derived from the spent fuel but not chemically purified would normally be the reference case.

In this study, the dose rate is calculated after the material has been processed for potential weapons use; it is not of the starting item. This is a very conservative approach in accounting for the effect of the dose rate. This basically means that the adversary has access to shielded hot cells or equivalent handling facilities. If the adversary does not have access to these capabilities, then credit can be taken for the size and mass of the fuel assembly. In this case, the dose rate would be taken from the used fuel assembly and the M/50 term would be replaced with an N/10 term, where N is the net weight of the fuel assembly in kg.

The figure of merit formula is derived by comparing the properties of the material in question to accepted standards. The established standards are: 1) the threshold for low enriched uranium (i.e., $^{235}$U enrichment less than 20%), 2) radioisotope thermoelectric generator plutonium (i.e., $^{238}$Pu enrichment greater than 80%), and 3) a self-protecting dose rate (i.e., 500 rad/h at 1 m). Historically, the self-protecting dose rate was taken to be 100 rem/h at 1 m [4]. Upon recent technical review [5,6], an increase to 500 rad/h at 1 m is proposed.

Table 1 shows the relationship between FOM$_1$, weapons utility, and materials attractiveness in a safeguards and security context. Materials that have a FOM$_1$ greater than 1 are attractive for weapons use and materials that have a FOM$_1$ less than 1 are not attractive for weapons use. Beyond this simple binary distinction, it should be noted that the lower the FOM$_1$ the better. Even though a material may still need to be safeguarded and secured, a process that produces a material with a FOM$_1$ of 1.1 should be encouraged over a process that produces a material with a FOM$_1$ of 2.5.

Table 1. The relationship between FOM$_1$, weapons utility, and materials attractiveness

<table>
<thead>
<tr>
<th>FOM$_1$</th>
<th>Weapons Utility</th>
<th>Materials Attractiveness</th>
<th>Attractiveness Level$^*$ [3]</th>
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<tbody>
<tr>
<td>&gt; 2</td>
<td>Preferred</td>
<td>High</td>
<td>~B</td>
</tr>
<tr>
<td>1-2</td>
<td>Attractive</td>
<td>Medium</td>
<td>~C</td>
</tr>
<tr>
<td>0-1</td>
<td>Unattractive</td>
<td>Low</td>
<td>~D</td>
</tr>
<tr>
<td>&lt; 0</td>
<td>Unattractive</td>
<td>Very Low</td>
<td>~E</td>
</tr>
</tbody>
</table>

*The DOE attractiveness levels are in reasonably good alignment with FOM$_1$ for metals and alloys.

FOM$_1$ was reviewed by nuclear weapons experts at both LANL and LLNL. While it was determined that there are a number of smaller factors that are not captures, it was agreed that FOM$_1$ captures the dominant factors quite nicely in an unclassified format.

The FOM represents a small but important part of the overall proliferation and security risks that are posed by various materials and processes in the nuclear fuel cycle. To contextualize the FOM$_1$, it overlaps strongly with one of the six proliferation resistance measures (Fissile Material Type) that is identified in the PR&PP methodology [7], and it overlaps strongly with the material attractiveness criteria which is a key part of the DOE graded safeguards table [3]. So in the case of proliferation resistance there are five other factors that need to be considered, e.g. proliferation technical difficulty, proliferation cost, proliferation time, detection probability, and detection resource efficiency. In the case of physical protection,
there are two other factors that need to be considered, e.g. material quantity and security category.

2. Background and Approach

For many years India has promoted the long term goal of a sustainable fuel cycle based on $^{233}$U and Th. With the discovery of its own large thorium reserves, the United States has taken a renewed interest in this fuel cycle. The current worldwide fleet of light and heavy water reactors breed reactor-grade plutonium which is weapons usable largely because the bare critical mass is small. The impurities in the plutonium, $^{238}$Pu & $^{240}$Pu, increase the heat content and the intrinsic neutron rate, respectively, thus making the material slightly less attractive. A new generation of light or heavy water reactors based on Th will produce $^{233}$U, which is also weapons usable because of its small critical mass. The primarily impurity in the uranium, $^{232}$U, substantially increases the dose rate of the material. The high dose rate arises from an intense high energy gamma-ray decay from $^{208}$Tl, a daughter product of $^{232}$U. This study evaluates the attractiveness of the $^{233}$U that is produced in Th-based fuel cycles and compares it to other nuclear materials of interest.

For the purpose of this study, the reactor design is assumed to be that of the advanced heavy water reactor (AHWR) that has been proposed by India [8]. The AHWR is a vertical pressure tube type reactor cooled by boiling light water and moderated by heavy water. The reactor is designed for a sustainable Th-based fuel cycle. The used fuel is reprocessed to extract and recycle the $^{233}$U. The reactor uses $^{233}$U-Th rods in the inner fuel blanket and Pu-U rods in the outer fuel blanket.

In this analysis, three different material cases have been considered: 1. Pu-Th Fuel Rods, the 20 Rod Outer Fuel Blanket, 2. $^{233}$U-Th Rods, the 16 Rod Inner Fuel Blanket, and 3. Total $^{233}$U-Pu-Th in the assembly. The three cases are used to determine the attractiveness of the individual Pu-Th rods, the $^{233}$U-Th rods, and the complete assembly. Although a specific reactor design is considered, it is clear from the results that follow that the design and type of fuel cycle chosen will not have much effect on the overall attractiveness of these materials. The isotopic compositions of the used fuel were determined using SCALE [9]. The physical properties of the materials for the FOM calculations were determined using MCNPX [10].

3. Results and Discussion

For each of the three cases, the FOM$_1$ is calculated as a function of age, measured from insertion into the reactor. Then the material is in the reactor for the first two and a half years. This corresponds to a burn-up of 20,000 MWD/THM. The remaining age is cooling time out of the reactor. The FOM$_1$ is calculated for $^{233}$U and Pu when separated from the Th fuel matrix. In all cases, the FOM$_1$ of the unseparated fuel has a value of $-\infty$. The Th-based fuel at charge even in metal form does not have a critical mass unless it is heavily moderated.

For the outer Pu-Th rods, the inventory of Pu per the outer 20 rods (~50 kg of heavy metal at start-up) is 2.58 kg and the inventory of $^{233}$U and Pu at discharge is 0.35 and 1.82 kg, respectively. For the inner $^{233}$U-Th rods, the inventory per the inner 16 rods (~30 kg of heavy metal) of $^{233}$U at start-up is 2.20 kg and the inventory of $^{233}$U and Pu at discharge is 2.04 kg and 0.3 mg, respectively. For the combined $^{233}$U-Pu-Th rods, the inventory of $^{233}$U and Pu (~80 kg of heavy metal) per the entire assembly at start-up is 2.20 and 2.58 kg, respectively. The inventory of $^{235}$U and Pu at discharge is 2.38 and 1.82 kg, respectively. Total mass of a typical assembly is 181 kg. The equilibrium concentration of $^{232}$U in the $^{233}$U in a closed, sustainable fuel cycle will be between 700 and 1,000 ppm.
Figure 1 shows the FOM$_{1}$ for the SNM component in the used nuclear fuel. All of these materials are very attractive for nuclear weapons use, except for the small amount of Pu in the $^{233}$U-Th rods which is over 90% $^{238}$Pu. Only the Pu in the Pu-Th rods displays any significant change for the time period shown. For both cases, the Pu FOM$_{1}$ drops while in the reactor because $^{238}$Pu is consumed. After discharge, the Pu FOM$_{1}$ increases because initially $^{241}$Pu is decaying away and then $^{238}$Pu.

The intact used fuel before reprocessing is not substantially different in attractiveness than ordinary used LWR or HWR fuel. In other words, the Th-based spent fuel is not attractive as long as the dose rate is on the order of 500 rad/h at one meter or higher. A more detailed analysis of the attractiveness of the intact used fuel assemblies as a function of age is still needed to show how long the intact used fuel will remain self-protecting.

One can also examine the attractiveness of $^{233}$U as a function of $^{232}$U content and age. In Figure 2, the heat term (bare critical mass times heat content) and the dose term are plotted as a function of age. The bare critical mass is for practical purposes constant as a function of age, i.e. 15.5 kg. The FOM$_{1}$ is also plotted as a function of age. In summary $^{233}$U is very attractive at any practical concentration of $^{232}$U and age. The material is the least attractive at about 10 years of aging. There is little difference in attractiveness for freshly purified materials and for very old materials. The curve with 800 ppm of $^{232}$U most closely matches what is expected in these Th-based fuelled reactors.
4. Conclusions

Thorium-based reactors produce very attractive materials. The $^{233}$U that is produced has a substantial amount of $^{232}$U. The presence of $^{232}$U increases the dose of the material particularly at ages of about 10 years after irradiation. This is due to the in growth of $^{208}\text{Tl}$ which has an intense high energy gamma-ray emission. In terms of weapons utility or material attractiveness this dose rate is only a nuisance to the adversary. It is not anywhere near sufficient to incapacitate a dedicated adversary. So if long term health and safety is not a concern to the adversary, $^{233}$U is one of the most attractive of all nuclear materials.

Even though $^{233}$U is very attractive, like reactor-grade Pu, it is not normally attractive when it is contained within used nuclear fuel. The high dose rate of the used fuel in combination with the large mass of the used fuel assembly and the low concentration of SNM makes the material self-protecting for many years. Like used LWR and HWR fuels, however, the material eventually becomes attractive as the dose rate decays with age.

Consistent with other studies of fuel cycles, the Th-based materials and processes need high levels of safeguards and moderate to high levels of security. Full safeguards would be needed on all facilities handing greater than 8 kg of $^{233}$U and Pu. However, security can be reduced for the used fuel while the dose rate is high enough for it to be self-protecting (e.g. Cat III), but security needs to be high in the recycling and fuel fabrication facilities (e.g. Cat I) and moderate to high in any fresh fuel handing facilities (e.g. Cat II or I).

5. References

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Life science applications
Voxel-geometry head phantoms can be used for assessment of dose distribution in substantially inhomogeneous medium like regions in close vicinity of the skull. In this study, dose distribution calculations for bidirectional interlaced microbeam radiation therapy and detailed voxel-geometry head phantom model were performed with the Monte Carlo code MCNPX. Dose distributions in dimensions which are smaller than those of voxels were achieved; and were calculated in the vicinity of intracranial targets of 20 x 6.8 x 20 mm³ and 20 x 20 x 20 mm³. Depth-dose profiles and dose falloffs at the lateral side of the targets were calculated. The parallel pattern of the microbeam arrays was preserved through the head phantom which makes it possible to interlace microbeam arrays even at deep seated targets. As the dimensions of the target volume were increased, the valley dose values increased with the number of microbeams and this sets limits on the size and position of the target.

1. Introduction

Microbeam Radiation Therapy (MRT) [1, 2] is a preclinical technique which uses the principle that the normal tissue can tolerate high radiation doses in small volumes [3, 4, 5]. MRT is carried out using arrays of highly intense synchrotron-wiggler-generated X-rays, typically with 25 µm beam width, 200 µm beam interval [5, 6] and 120 keV mean energy. It has the potential to be used for treating infantile brain tumours when other kinds of radiotherapy would be highly toxic to the developing normal brain [5, 6, 7]. The main attribute of the method is its remarkable sparing effect in healthy tissue, including the central nervous system [8]. The reason for the tolerance of healthy tissue to high radiation doses in MRT has not been fully understood, however it is assumed that the microvasculature in the paths of the beams is regenerated from the angiogenic cells surviving between the beams’ paths [9, 10]. However, this is not observed in tumour tissue and microvessels of the tumour are damaged which can lead to the loss of blood perfusion and tissue necrosis [11]. The normal tissue tolerance to MRT depends on the ratio of the peak dose to the dose delivered to tissues adjacent to the peaks, i.e., the valley dose [3].

Potential applications of MRT have been studied experimentally at National Synchrotron Light Source at Upton, New York, USA and at the European Synchrotron Radiation Facility (ESRF) in Grenoble, France. In recent studies it was observed that the rat central nervous system tolerated microbeams as thick as 0.68 mm and an unsegmented radiation field could be produced by interlacing two orthogonal “thick” microbeam arrays aimed at the target [9, 12]. Dilmanian et al [9, 12] stated that for “thick” interlaced microbeam arrays, Monte Carlo calculations had indicated a 30 µm dose falloff (80 – 20%) at the edge of the target, which is much less than the 2 to 5 mm value for conventional radiotherapy.

In this study, dose distribution calculations for bidirectional interlaced irradiating geometry and detailed voxel-geometry Zubal head phantom model [13, 14] were performed with the Monte Carlo code MCNPX [15]. Dose distributions were calculated in the vicinity of targets of 20 x 6.8 x 20 mm³ and 20 x 20 x 20 mm³ dimensions, location of which is similar to that of the target selected by Dilmanian et al [9]. Depth-dose profiles and dose falloffs at the side of the targets were calculated.
2. Methods

2.1. Sampled radiation source

The measured spectrum [3] (Fig 1) at the ID17 beamline of ESRF was selected as the X-ray energies to be sampled. As seen in Fig 2, for a target of 20 x 20 x 20 mm³ dimensions, 0.68 mm beam thickness, 20 mm beam height and 0.68 mm beam interval, 15 planar beams were sampled for irradiation from side and top of the head phantom. The collimators and slits were not incorporated and unidirectional photon sources were defined close to the phantom’s surface.

![Fig 1. X-ray spectrum measured at ID17 beamline normalized to maximum value](image)

![Fig 2. Irradiation geometry](image)

2.2. Modified Zubal head phantom

Modified voxel-geometry Zubal head phantom [14] was used in this study. This phantom is based on the Zubal phantom [13] which was created from 124 transverse MRI images of a healthy human male head. The head and outer air space in the close vicinity were formed with 85 x 109 x 120 lattice of voxels with dimensions of 2.2 x 2.2 x 1.4 mm³. 29 critical structures of the head were identified by their individual universes. The voxel phantom was attached to a simplistic mathematical model of the neck and torso, under the upper jaw. In Fig 3 and Fig 4 the three dimensional plot of the voxel-geometry head phantom and two dimensional slice view, which were obtained by plotting the densities of voxels with respect to the locations, are given.

![Fig 3. 3-D view of the modified Zubal head Phantom](image)

![Fig 4. Slice view of modified Zubal head Phantom](image)
2.4. The Monte Carlo simulation

An irradiating geometry similar to that of Dilmanian et al's [9] experimental study was selected, thus a target of 20 x 20 x 20 mm³ dimensions, lateral to Cz in Fig 4, and 6 cm superior to the interauricular line was chosen. For the target of 20 x 20 x 20 mm³, 15 beams in each interlacing arrays with 0.68 mm beam thickness and 0.68 mm beam interval values were sampled and photon–electron coupled simulations were performed with 1keV energy cutoff.

The voxels were only used to describe the medium in which the radiation transport was simulated, rather than for tallying purpose. Dose distributions in dimensions which are smaller than those of voxels were achieved with the “mesh tally” feature of MCNPX. For this type of tally, a grid (meshes), independent of the materials, can be defined inside the medium and used for tallying energy deposition, however, since the tally meshes are independent of the medium, the tally results are given in the unit of “MeV / cm³ per initial particle”. Thus dose distributions were obtained by dividing the tally results for each mesh by the densities of the corresponding voxels.

For observing the 2-dimensional dose distribution throughout the voxel-phantom model and decision for the dimensions of the meshes for calculation of peak and valley dose values, simulations were performed with meshes of 200 µm x 200 µm x 0.5 cm dimensions with 2E8 initial photon histories.

Mesh tallies of 1 mm x 0.68 mm x 0.5 cm dimensions were defined in the region between the side of the phantom where the incident microbeams impinge and the opposite side for depth-dose distribution calculations. And mesh tallies of 2 cm x 10 µm x 2 cm were defined beside the target region to calculate the dose falloff at the target side.

The simulations were performed for the cases with 5-beam (20 x 6.8 x 20 mm³-target) (case 1) and 15-beam (20 x 20 x 20 mm³-target) (case 2) interlacing arrays. The number of histories is 2E8 in Monte Carlo calculations for all cases.

3. Results

3.1. Two-dimensional demonstration of dose distribution inside voxel-geometry-phantom

In Fig 5, 2-dimensional dose distribution in the voxel head phantom is given as a contour plot. The plot plane is indicated in Fig 2. The dose values are given in units of MeV/g per sampled source photon. Relative error at the distal side of the target is below 15% which is adequate for qualitative assessment for 2-dimensional radiation transport. X-ray diffraction was not observed and the parallel pattern of peak and valley regions is preserved. Thus it was decided that for peak and valley depth-dose distribution calculations, meshes with thickness of 0.68 mm could be used.
3.2. Effect of the number of microbeams: Comparison between 5-beam and 15-beam arrays

Peak dose distributions with respect to depth for interlaced 5- (case 1) and 15-beam-arrays (case 2) are given in Fig 6. Skin entrance, target entrance and target exit (dose at distal end of the target) peak dose ratios between case 1 and case 2 are 0.702 ± 0.010, 0.635 ± 0.014 and 0.628 ± 0.016 respectively. Skin entrance to target exit peak dose ratios for case 1 and case 2 are 2.45 ± 0.04 and 2.19 ± 0.05 respectively. Valley dose distributions with respect to depth for case 1 and case 2 are given in Fig 7. The dose values between -4 and -2 cm are peak doses for irradiation from top of the phantom.

Minimum peak to valley dose ratio (PVDR) values between the entrance and target entrance region (except in skull) for case 1 and case 2 are 8.9 ± 0.3 and 5.3 ± 0.2 respectively. 80-20% dose falloff distance of ~11 µm and ~28 µm were obtained for case 1 and case 2 respectively. 28-µm dose falloff distance is consistent with that given by Dilmanian et al (2006, 2008).

4. Discussion

Both qualitative and quantitative information on transport of X-ray microbeams and dose distribution inside detailed voxel-geometry head phantom were obtained in this study. It was observed that the microbeams were not diffracted by the skull and parallel pattern of the microbeam arrays was preserved through the head phantom, and this makes it possible to interlace microbeam arrays even at deep seated targets.

In comparison of 5-beam (case 1) and 15-beam array (case 2) cases, it was observed that, for the selected target location and irradiation geometry, the minimum PVDR values between the skin entrance and target entrance region (except in skull) for the case 1 and case 2 are 8.9 ± 0.3 and 5.3 ± 0.2 respectively. And they are above the value 5 which was stated as reasonable for microbeam therapy [16]. As the dimensions of the target volume increase, the valley dose values increase with the number of microbeams and the PVDR values decrease; and this sets limits on the size and position of the target.

For the same dose value at the distal side of the tumour, the skin entrance dose for case 1 was determined to be 1.12 ± 0.04 of the entrance dose calculated for case 2; and this is due to dose build up in the second case caused by more number of beams.

5. Conclusion

This was the first study that a detailed voxel-geometry head phantom has ever been used in dosimetric Monte Carlo calculations for the X-ray microbeam therapy technique. The selected phantom includes 17 materials and depicts the medium in a more realistic way.
compared to the cylindrical or spherical simplified mathematical water phantoms used in previous studies. Although similar dose falloff results were obtained with the mathematical models for a deep seated target [9, 12], the voxel-geometry phantoms have the advantage that they can be used for assessment of dose distribution in substantially inhomogeneous medium like regions in close vicinity of the skull.

Calculated dose falloff distance is consistent with that given by Dilmanian et al [9, 12]. Furthermore, the peak and valley doses (hence PVDR) were consistent with the semi-qualitative results given in Dilmanian et al [9] experimental study with anthropomorphic head phantom. As a future work, different X-ray spectrums and other concepts such as stereotactic microbeam therapy can be studied to make contribution to development of the technique with the help of experience gained in this study.

References

The aim of this study was to make prognosis of thyroid doses in case of an accident at a nuclear power plant. In order to achieve this, the following was done:

- assessment of thyroid doses formed due to maximum design basis accident (MDBA) at NPP with pressurized water reactor (at different distances from the station);
- estimation of the length of the territories at which dose exposure may exceed the current international generic criteria for protective and other actions in case of a nuclear emergency.

Various scenarios of possible weather conditions were considered in order to simulate the transfer of radionuclides in the atmosphere.

This study allowed estimating the thyroid doses, as well as the necessity of protective and other actions in case of the MDBA.

Results of prognosis using the international models have demonstrated that:

- thyroid doses due to inhalation of iodine-131 don’t exceed 2 mGy, due to consumption of contaminated milk – 40 mGy, vegetables – 5 mGy;
- there is no need in thyroid blocking for population;

The results of modeling performed in this study form a basis for developing national arrangements for response to a MDBA at considered NPP.

1. Introduction

Despite all the precautions that are taken in the design and operation of nuclear facilities and the conduct of nuclear activities, there remains a possibility that a failure (intentional act) or an accident may give rise to a nuclear emergency. In some cases, this may lead to the release of radioactive materials within facilities and/or into the public domain, which may necessitate emergency response actions [1].

The following categories of initiating events are considered within the safety justification process: design and beyond design basis accidents. Design basis accident are accident conditions against which a facility is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits. The design basis accident with the most serious consequences is called the maximum design basis accident (MDBA).

Even in case of MDBA radioactive material emitted from the nuclear power plant can reach areas of the population residence and contaminate the environment in the region. Severity of effects and consequences for inhabitants and environment of contaminated areas depends on the amount of radioactive materials released and radiation doses to certain organs or whole body created by the emitted radionuclides.

Consequence projection due to a possible nuclear power plant accident is one of the stages prior to construction of a nuclear power plant. In this regard, the following was carried out:

- assessment of thyroid doses formed due to MDBA at NPP at different distances from the station;
- estimation of the length of the territories at which radiation doses may exceed the current international generic criteria for protective and other actions in case of a nuclear emergency.

As a result of the Chernobyl accident currently there is a significant increase of thyroid cancer incidences among inhabitants of Russia, Ukraine and Belarus [2].

In connection with this fact the study paid particular attention to the prognosis of thyroid doses due to the maximum design basis accident at the NPP.
2. Material and methods

For modeling of the nuclear emergency scenario geographic location and technical characteristics of the plant (type, power of the reactor, presence of sprinkler and filtration systems), weather conditions (wind speed and direction, temperature, air pressure, precipitations, specific values of the atmospheric turbulence) were taken into account.

Simulation and estimation of consequences of potential nuclear emergency is possible using the software which takes into consideration all the above mentioned significant parameters. InterRAS (The International Radiological Assessment System) is an example of such program. It is based on the U.S. NRC's software RASCAL (Radiological Assessment System for Consequence Analysis) and conforms to the IAEA Basic Safety Standards (1996) as well as others international documents. InterRAS was modified to allow assessment a greater range of accidents and enable to calculate doses to public [3].

The model «ST-DOSE» (Source Term to Dose) was used for dose estimation. It allows calculating activities of released radionuclides and making assessment of integrated radiation doses which form due to the accident release of radioactive material into the atmosphere.

A large break (break of the primary system pipe of diameter exceeding 100 mm) is considered as MDBA as it is the most dangerous project failure in terms of the core damage degree and, consequently, leads to more severe radiological consequences. Conservatively, assumption that total (100 %) depressurization of fuel elements (cartridges) in reactor core has occurred was made. Primary system pipe rupture causes the coolant of primary system discharge, which as a consequence leads to increase of pressure in containment.

During the calculations certain conservative assumptions were used. They allowed obtaining scientifically substantiated upper levels of radiation doses to public due to the possible MDBA. The release was taken as ground level, because in this case higher dose levels are expected to form at the considerable distance from the NPP.

Table 1 shows model parameters used for calculation, parameters which represent plant conditions and correspond to the maximum design basis accident.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Parameter value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor type</td>
<td>Water-water energy reactor (WWER)</td>
</tr>
<tr>
<td>Reactor power at the beginning of the accident</td>
<td>3200 MWt</td>
</tr>
<tr>
<td>Core damage</td>
<td>100 % gap release</td>
</tr>
<tr>
<td>Leak rate</td>
<td>0,1 % per day (project leak rate)</td>
</tr>
<tr>
<td>Hold-up time</td>
<td>—</td>
</tr>
<tr>
<td>Containment sprays</td>
<td>on</td>
</tr>
<tr>
<td>Filters</td>
<td>off</td>
</tr>
<tr>
<td>Duration of the radionuclides release</td>
<td>2 hours 30 minutes</td>
</tr>
<tr>
<td>Release height</td>
<td>0 м (ground level)</td>
</tr>
</tbody>
</table>

Various real possible weather scenarios were considered to simulate transfer of radionuclides in atmosphere. The worst meteorological conditions, which lead to the highest doses to population, are presented in Table 2. The data correspond to “winter” and “summer” seasons.

Consideration of "summer" and "winter" weather conditions allows taking into account or not contribution of internal dose caused by the consumption of contaminated foodstuff into total thyroid dose.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Parameter value («winter» scenario of MDBA)</th>
<th>Parameter value («summer» scenario of MDBA)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Wind Direction</td>
<td>western turning into south-western</td>
<td>south-western</td>
</tr>
<tr>
<td>Wind Speed</td>
<td>5.5 – 11 m/s</td>
<td>6.4 – 6.7 m/s</td>
</tr>
</tbody>
</table>
Thyroid dose consists of dose due to the inhalation of iodine-131 and dose due to the ingestion of contaminated products, generally milk and leafy vegetables. The first constituent that is thyroid dose to an adult person while he/she is performing a light physical activity was calculated using the InterRAS model. Dose due to the alimentary intake of iodine-131 was calculated using the following formula [3]:

\[
E_{\text{ing}} = \sum (C_f \times U_f \times DI \times CF) \times \prod RF_j = E_{\text{ing}}
\]

(1)

where:
- \(E_{\text{ing}}\) – thyroid dose from ingestion of I-131, mGy;
- \(C_f\) – activity concentration of I-131 in food (milk, leafy vegetables), kBq/kg;
- \(U_f\) – the amount of a food \(f\) consumed by the population of interest per day, kg/day or l/day (Table 3);
- \(CF\) – ingestion dose conversion coefficient per unit intake of I-131, mGy/kBq (Table 4);
- \(DI\) – days of intake. It is the period food is assumed to be consumed. If \(T_{1/2} > 21\) days use 30 days. If \(T_{1/2} < 21\) days use the mean life \((T_m)\) of the isotope.

\[
T_m = T_{1/2} \times 1.44
\]

(2)

where:
- \(T_{1/2}\) – radiological half-life;
- \(RF_j\) – Reduction Factor is the fraction of the contamination remaining after decay or some process \(j\) used to reduce the contamination before food is released for consumption. The reduction factors for milk and leafy vegetables were set equal to 1 (food processing isn’t used) to carry out conservative assessment of dose.

### Tab 3: Milk and leafy vegetables consumption by rural population

<table>
<thead>
<tr>
<th>Age, years</th>
<th>0 - 1</th>
<th>1 - 2</th>
<th>2 - 7</th>
<th>7 - 12</th>
<th>12 - 17</th>
<th>older 17</th>
</tr>
</thead>
<tbody>
<tr>
<td>Milk consumption (l/day)</td>
<td>0.24</td>
<td>0.3</td>
<td>0.3</td>
<td>0.5</td>
<td>0.51</td>
<td>0.5</td>
</tr>
<tr>
<td>Vegetables consumption (kg/day)</td>
<td>0</td>
<td>0.003</td>
<td>0.006</td>
<td>0.02</td>
<td>0.028</td>
<td>0.03</td>
</tr>
</tbody>
</table>

### Tab 4: Ingestion dose conversion coefficient per unit intake of I-131, mGy/kBq

<table>
<thead>
<tr>
<th>Age, years</th>
<th>0 - 1</th>
<th>1 - 2</th>
<th>2 - 7</th>
<th>7 - 12</th>
<th>12 - 17</th>
<th>older 17</th>
</tr>
</thead>
<tbody>
<tr>
<td>Dose conversion coefficient, mGy/kBq</td>
<td>1.8E-01</td>
<td>1.8E-01</td>
<td>1.0E-01</td>
<td>5.2E-02</td>
<td>3.4E-02</td>
<td>2.2E-02</td>
</tr>
</tbody>
</table>

### Results

Total release of radionuclides into the environment in case of the above described accident scenario regardless of weather conditions amounted to \(1.1 \cdot 10^{14}\) Bq. Activities of emitted iodine isotopes are represented in Table 5.

### Tab 5: Activity of iodine isotopes released into the environment due to MDBA, Bq

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>I-131</th>
<th>I-132</th>
<th>I-133</th>
<th>I-134</th>
<th>I-135</th>
</tr>
</thead>
<tbody>
<tr>
<td>Activity, Bq</td>
<td>4.70E+11</td>
<td>6.70E+11</td>
<td>9.50E+11</td>
<td>1.00E+12</td>
<td>8.30E+11</td>
</tr>
<tr>
<td>TOTAL, Bq</td>
<td>3.92E+12</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
Iodine emission is 4% of the total radioactive material released into the environment as a result of nuclear accident. This value is in a good agreement with mean estimate of the iodine fraction usually released during such accidents [4].

Calculated thyroid doses caused by inhalation of iodine for “winter” and “summer” scenarios of MDBA are presented on Figure 1.

Fig 1. Thyroid dose at different distances from the source of release, mGy

The differences in thyroid doses for the different scenarios which are observed in the chart can be explained by differences in weather conditions. In the "winter" version of the accident the weather corresponds to the atmospheric stability class F (moderately stable conditions), which is most unfavorable for the dispersion of the radioactive release, and, consequently, leads to higher doses.

Internal exposure of thyroid caused by the consumption of contaminated food will be maximum if the accident occurs in spring - summer season and especially during the harvest season. The isotopes of iodine, as well as other radionuclides, deposit from the atmosphere and then retain on the surface of plants after which go into agricultural production during harvesting. Particularly important migration way of iodine isotopes during the early stage of the accident is food-chain "pasture – animal – milk”.

Figures 2 – 3 show the calculated thyroid doses to six age groups caused by consumption of milk and leafy vegetables contaminated by iodine in the case of MDBA.

Fig 2. Thyroid dose to six age groups in 30 days after MDBA caused by their consumption of milk contaminated by iodine-131, mGy
The graphs show that the consumption of contaminated milk leads to higher thyroid doses than the consumption of leafy vegetables does. The main reason of this is differences in the consumption of these foods by age groups (Table 3). The highest thyroid doses due to the intake of contaminated food are observed at a distance of 3 km away from the station, and it can be explained by the transport of radioactive cloud by air stream. The magnitude and rate of absorption, accumulation of iodine by the thyroid gland and the rate of its excretion depend on the age, sex. Since these parameters are higher for children than for adults, the highest doses caused by the consumption of contaminated milk by iodine-131 were observed in children who are not older than two years.

4. Conclusion

Thyroid dose estimation in case of MDBA showed that maximum doses due to the iodine inhalation are not more than 1.7 and 0.88 mGy under “winter” or “summer” accident conditions, respectively; radiation dose due to the consumption of milk contaminated by iodine-131 is 40 mGy, leafy vegetables – 5 mGy. Total exposure doses didn’t exceed generic criteria for protective actions and other response actions in emergency exposure situations – 50 mGy in the first 7 days after accident [4] – in both considered MDBA scenarios and even in the worst one. As the generic criteria is not exceeded so there is no need in thyroid blocking for population.

The results of modeling performed in this study form a basis for developing national arrangements for response to a MDBA at considered NPP.

5. References

PROTECTION OF MAN: THE EXPOSED INDIVIDUAL

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ABSTRACT

Present methods for quantifying radiation exposure rely on a standardized reference man (75 kg) with defined average anatomical and physiological data. But individual person actually exposed differs from this idealized standard man. Therefore the focus of investigations at the Institute for Radiation Research (Institut für Strahlenforschung, ISF) which was founded at Karlsruhe Institute of Technology (Karlsruher Institut für Technologie, KIT) in 2009 is based on the vision to place the exposed individual with its anatomical and physiological particularities, under consideration of age, gender, body height, body shape and environment, in the centre of an individual-related quantification of the external and internal radiation exposure.

Research work at the ISF is aiming at quantifying radiation exposure by improved determination of doses essentially caused by external radiation fields and the intake of radionuclides into the body. The three main topics of the institute are

- external dosimetry (e.g. using a (voxel) model of the hand to simulate skin dose distribution)
- internal dosimetry (e.g. body size related efficiency calibration of in-vivo counting equipment)
- numerical methods/modeling (e.g. development of a mathematical/voxel-hybrid model of the human body)

1. Introduction – The Institute for Radiation Research at Karlsruhe Institute of Technology

The ‘new’ Institute for Radiation Research (ISF) was founded January 1st 2009 at the former Karlsruhe Research Centre (Forschungszentrum Karlsruhe, FZK) which has merged with the University of Karlsruhe to the Karlsruhe Institute of Technology (KIT) at October 1st 2009. Since KIT still being part of the German Helmholtz Association of Research Centres (Helmholtz Gemeinschaft deutscher Forschungszentren, HGF) the institute belongs to the Programme Nuclear Safety Research with its topics - reactor safety, transmutation and partitioning, safety of nuclear waste disposal research, emergency management and radiation protection research. Therefore the new institute is well inserted in the nuclear research field and according to the long tradition of radiation protection RP at the former research centre FZK a wide range of knowledge and applied experiences are available.

The vision of the research at the new Institute for Radiation Research is to place the exposed individual with its anatomical and physiological particularities - under consideration of age, gender, body height/shape and its environment - in the centre of an individual-related quantification of radiation exposure.

Apart from the different research projects, the Institute of Radiation Research (ISF) participates in academic education in the nuclear field [1]. Lectures on Radiation protection topics are held by scientists of ISF, who also supervise practical studies and theses (bachelor, master and PhD) performed in the institute. The ISF is strongly linked to the Department of Safety Management of the KIT, which runs several radiation protection laboratories, including an in-vivo monitoring lab, radioanalytical and dosimetric services for the KIT site and for external customers. Besides their routine works all radiation protection
labs are used for the experimental work of ISF scientists. The mix between routine and experimental work ensures an application oriented research. The problems observed in routine applications give rise to scientific projects and the results of research can be introduced in the daily work at the lab.

2. Research topics for individual-related dosimetry

Currently research work at ISF is concentrated in three main fields:

- external dosimetry (i.e. doses resulting from irradiation by external fields)
- numerical methods/modelling (i.e. modelling and simulation of radiation protection scenarios)
- internal dosimetry (i.e. doses resulting from incorporation of radionuclides)

More details about these fields and examples for several research projects will be presented in the following chapters.

2.1 External Dosimetry

External dosimetry deals with the determination of doses caused by radiation fields. Therefore these fields need to be characterized either by measurements or simulations. At ISF research projects in this domain investigate the influence of external ionising radiation on the human body, but also on irradiated objects and measurement instruments (e.g. dosimeters). The use of hybrid-models, a combination of particle-transport-calculation with physical models of the materials response, offers the possibility to describe the variation of properties of solid state bodies in radiation fields. These results can be used to optimize existing detectors or to develop new dosimeters. Currently track etch detectors are studied with this method [2].

The simulation of the complex radiation fields generated by the irradiation facilities installations at KIT or from complex distributions of (e.g. beta emitting) sources is another task carried out in order to characterize these fields and to calculate the radiation doses for persons or probes irradiated [3]. For electron fields, large inhomogeneities are observed, therefore special attention is paid to the simulation of electron transport in thin layers where large deviations in electron flux or dose can be observed for different codes and their parametrization [4].

The determination of physical properties of different (inhomogeneous) radiation fields is also of interest for a dosimetry under consideration of individual movements of a person and work-place-specific aspects. Here the doses resulting from different handling procedures of radioactive material (e.g. container with radioactive liquid) can be evaluated and compared, thus an optimization of processes in terms of minimizing dose of the worker can be achieved. In addition to measurements in these fields particularly modelling and simulation with Monte-Carlo programmes of the according handling scenarios is an important tool. Geometrical and voxel models allow to simulate the skin dose distribution on a moving human hand during handling of beta-gamma sources (e.g. for procedures in nuclear medicine applications) [5]. Figure 1 shows a dose to the hand calculated for the handling of a Y-90 source in a syringe. Individual radiation protection measures can be based on these dose calculations taking into account the individual handling of the syringe with the beta sources. A prerequisite for these calculations is the generation of the virtual scenarios from video-recorded real handling procedures, which is also performed at ISF.
2.2 Numerical Methods/Modelling

An important issue for modelling radiation protection scenarios is the development of individual body models. These are generated from tomographical or photo-optical pictures. For an individual body model the organs and tissues are identified in the image data and segmented according to their physical properties, but also with respect to the need in radiation protection tasks. Software which is used to generate, handle and modify large voxel models and to implement them in the simulation codes (e.g. MCNP) is developed at the institute. A big task is the development of procedures to adapt existing models to an individual human by suitable transformations of the models. At ISF the focus of this work is the application of such individualized models for the calibration of the in-vivo counters. Several sensitive parameters which influence the counting efficiency have been identified systematically. Recently one of these parameters (Chest Wall Thickness, CWT) was studied in more detail. In preparation for a new germanium detector system, a method was developed to measure CWT in real humans with ultrasound. A corresponding technique for calculating CWT for virtual models has been developed and implemented [6]. Using this method, chest wall thickness profiles of several existing human models as shown in figure 2 can be calculated. These CWT-profiles can be used for optimizing detector positioning and to check the plausibility of the model, which is useful especially for customized models.

Figure 2: CWT-profiles: colour-coded skin-pleura distances for MEETMan, ICRP AM and the LLNL-Torso voxel model.
Besides the models of the human body also the models of detectors used in radiation protection are studied and optimized. Models of the equipment at the in-vivo monitoring laboratory of ISF have been constructed from technical sketches and were optimized and validated by further measurements. Systematic studies of different available simulation codes are performed with these models [7]. A model of the lab’s LLNL-Torso phantom has been constructed from CT scans of the phantom. Models of the new HPGe-Detectors of the lab, which were optimized by a series of point source measurements, have been used to compare measurement and simulation of different scenarios in order to validate the models [8]. Models of the other phantoms available at ISF are currently segmented in order to extend these validation studies.

2.3 Internal Dosimetry

Internal dosimetry deals with the detection and quantification of radiation exposure after the intake of radioactive substances into the human body. The two main tasks for the assessments of the resulting doses are:

- the measurement of incorporated activities, either by direct methods (in-vivo counting) or by indirect methods like monitoring of excretion (in vitro bioassay).
- a description of the distribution of the radionuclides in a human body (biokinetic modelling)

At ISF in-vivo counting methods and biokinetic modelling are performed. A laboratory for the in-vitro bioassay is also available at KIT.

A main project in internal dosimetry is a new assembly of a partial body counter with additional whole body counting capabilities. Four electrically cooled HPGe-detectors are used for this. Simulated iso-flux of full energy photons and standard deviations of these fluxes calculated from different input scenarios were used to optimize the position of the detectors for whole body counting with the new machine [9].

Biokinetic behaviour of radionuclides is described by compartmental models, which divide the human body in such different parts (compartments) that are important in the metabolism of the radionuclide. The compartments are linked by flows of the incorporated material. The whole system is described by a set of differential equations. One of the research projects at ISF is aiming at the description of decorporation therapy with DTPA, which is applied to reduce doses of significant intakes of actinides like plutonium. One approach is to apply (semi-)empirical corrections to the solutions of the existing models [10]. An example for the application of these corrections to urinary excretion of plutonium after DTPA therapy is shown in figure 3.

![Figure 3: Model description of urinary excretion data from a real plutonium incorporation case from KIT. The blue curve shows a fit of the (semi-)empirical model to the data. The green line shows the predicted excretion for the same intake but without DTPA therapy.](image-url)
Another approach is to couple the models for the undisturbed biokinetics to new structures describing the biokinetic behaviour of the drug [11]. Studies on the physiological basis of the biokinetic models complement this work [12].

One more issue of biokinetic models, which is studied at ISF, is the uncertainties of the models predictions. The models and their parameters describe an idealized averaged person, which may not be representative for an individual. At ISF sensitivity studies of the biokinetic parameters are performed and the results of inter- and intra-individual variations are studied by simulations [13].

3. Conclusions

The described research fields and examples from different projects show that there is a wide range of interesting research topics according individual-related dosimetry.

4. References

SIMULATION OF THE DOSE GIVEN BY AN ELEKTA PRECISE LINEAR ACCELERATOR ON ANTHROPOMORPHIC HUMAN PHANTOMS

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ABSTRACT
This work focuses on simulating the irradiation of a reconstructed voxelized human phantom with a linear accelerator and studies the introduction of techniques based in Monte Carlo for this purpose.

The voxelized phantom is reconstructed from a series of segmented tomography slices of the head of the Zubal numerical phantom by means of a MATLAB algorithm, and the simulation of the irradiation of the phantom with the photon beam generated in an Elekta Precise linear accelerator is performed with the Monte Carlo transport code MCNP (Monte Carlo N-Particle), version 5.

The MATLAB program developed by the authors is designed to compute the voxelization of the Computed Tomography (CT) slices. This methodology is validated throughout the Snyder Head Phantom, which is voxelized and irradiated, and the results compared with those of an already voxelized version of the same phantom. The work results in dose mapping calculations within the voxelized anthropomorphic phantom.

1. Introduction
The work developed here studies the influence of heterogeneities in the irradiation of a voxelized human phantom with a linear accelerator Elekta Precise and the introduction of techniques based in Monte Carlo for such study. Monte Carlo algorithms are considered within the most precise and fast methods for dose calculations, with the only limitation of the large computing time cost.

The Monte Carlo computer code MCNP5 was employed to calculate the absorbed dose inside the phantom, given by a photon beam generated in an Elekta Precise linear accelerator. Prior works have validated the linear accelerator unit MCNP model by using a heterogeneous water cube-shaped phantom [1].

The reference phantom model utilized in this work is the Zubal Phantom [2], provided by Dr. George Zubal from Yale University, which consists of a series of presegmented CT slices of a human torso and head, presented in tiff files format.

The MATLAB algorithm developed here is first verified by simulating the irradiation of a voxelized phantom resulting from applying the voxelizing method to the Snyder Head phantom provided by Goorley et al [3], from the Massachusetts Institute of Technology, and comparing the results with the simulation of the irradiation of the 8 mm x 8 mm x 8 mm Voxelized Surface Snyder Head Phantom, a voxelized model generated by Goorley [4].

In the end, MCNP provides flux distribution calculations inside the voxelized phantom, via the FMESH tally tool, and the results are converted to dose by another algorithm built by the authors.
2. Materials and methods

1.1 Voxelization method and validation

The MATLAB algorithm developed in this work requires segmented images to start the calculations. The main routine of the algorithm first calls a subroutine, which inputs the CT slices and asks for the x, y and z resolution, interpolating slice-to-slice, so that the phantom is prepared with an equal pixel resolution in the three axis directions, and resulting in a 3-dimensional matrix, whose entries designate an index number for each pixel, which indicates the organ or structure to which this pixel belongs. Once the images are interpolated, a next subroutine divides the slices into squares. The proportion of different pixel intensities is computed in each square, storing this information in cells. Another subroutine identifies the intensities of the pixels with the organ materials, and applies the proportion of each pixel intensity to the corresponding material composition in the corresponding cell or square. Two matrices result defining the mixture material composition and the material density in each square. Afterwards, an interpolation between every square in the z axis that belongs to the same voxel is performed, resulting in two 3-Dimensional cell matrices where every cell depicts the material mixture of a voxel of the phantom.

A final subroutine transforms these two matrices in an MCNP input deck format text file, making use of the lattice geometry tool for defining geometry, which reduces the computing time around 6 times, as shown by Abella et al. [5]. This input includes the surface source, the MCNP5 tally tool FMESH superimposed mesh tally (a tally defined over the problem geometry), the desired physics options, and the voxelized phantom. In order to validate the MATLAB voxelizing algorithm, a simplified numerical head phantom has been voxelized and compared with an already voxelized version of the same phantom. A simulation of the irradiation of both models has been performed for the comparison. The Snyder Head Phantom provided by Goorley et al. simulates a human head throughout three ellipsoids with three main materials defining the head structure: skin, skull and brain, and one more for the air around it. This set of images are voxelized by our Matlab program and the resulting phantom is input in MCNP for the simulation of its irradiation. On the other hand, the 8 mm Voxelized Surface Snyder Head Phantom is an MCNP5 input deck, provided by the Nuclear Engineering Department at the Massachusetts Institute of Technology. This input is the reference model used for the validation of the model created by the MATLAB code developed in this work, and its irradiation is also simulated via MCNP5 under the same conditions. The comparison between the 8 mm x 8 mm x 8 mm Voxelized Surface Snyder Head Phantom and the voxelization of the Snyder Head Phantom has been performed via particle flux maps. The following graphs show the relative error for both XY and YZ central planes.

![Relative error (%) in each voxel for the XY central plane.](image-url)
Results show that, in the worst case, a relative error of 4% has occurred, while in most of the cases the relative error is lower than 1%, offering a satisfactory verification.

1.2 Zubal phantom and application

The MATLAB algorithm creates an MCNP voxelized human head phantom input from the Zubal Phantom, created by Zubal et al [2]. This phantom consists of 243 pre-segmented slices, 42 of which correspond to the head. The Zubal Phantom is a reconstruction from X-ray CT images which result in a 128 x 128 x 243 bytes volume with the same resolution of 4 mm in the three directions, and the portion used for this paper is a 128 x 128 x 42 bytes multiple-image file that corresponds to the head of the phantom.

The 42 slices corresponding to the head of the Zubal Phantom are input into the MATLAB program, which identifies each index number with a material composition, and in the end transforms it into an MCNP input. The 31 initial index numbers are limited to the compositions defined in the ICRU REPORT 46 [6] and 44 [7], resulting in 13 different biological materials. The ones not defined in the ICRU REPORT are defined as average tissue material. The MATLAB algorithm includes a subroutine that joins the voxels around the phantom which have an AIR material assigned, in order to reduce the computing time for the MCNP simulation. The resulting head MCNP model is an input of 177828 voxels or cells, shown in figure 5.

![Figure 5. Sketch of the Zubal Phantom MCNP5 model: full head and inside view.](image)

3. Simulation approach

The MATLAB program builds up a voxelized anthropomorphic phantom which is output together with the Elekta Precise linear accelerator model in an MCNP format input. Miró et al [1] proved the validation of the Elekta Precise linear accelerator MCNP model at the Hospital Provincial de Castelló by comparing the calculations of the energy deposition in a homogeneous and in a heterogeneous phantom with experimental data. The phantom used in the irradiation comprises a water tank with a polystyrene heterogeneity inside. The complete model consists of an Elekta Precise irradiator positioned to give a photon beam focused on the axis origin of the phantom. The linear accelerator has a leaf collimator to provide rectangular fields. A 10 cm x 10 cm has been used in this work. This model was used in a previous simulation in order to obtain a surface source file, which is the file written as result of the MCNP5 simulation when the SSW/SSR card is used. This option of the MCNP code allows registering the particles that reach a defined surface and stores them in a file that can be used as a new source for further simulations, saving up computing time. In this case, the rssa surface source file was the starting point of 940700 particles registered and resampled to $10^{11}$ (from the original $10^9$ with which the validation was initially simulated).
The model is designed with a Surface Source Distance (SSD) of 100 cm from the original source to the first voxel of the phantom in the x axis.

The FMESH tally is utilized with the purpose of defining a mesh tally superimposed over the problem voxel geometry. The FMESH tally calculates by default the track length estimate of the particle flux, averaged over a mesh cell, which in our case corresponds to a voxel, in units of particles/cm²·s. The process of conversion from particle flux to equivalent dose rate requires that the particle flux in each voxel was divided in 28 bins of energy from 0 to 7 MeV (there are no photons or electrons above this energy cut-off), using the EMESH option. The tally results are presented with an associated percentage error value. The conversion to dose units is carried out separately with a MATLAB program. The NIST Physical Reference Data web page [10] was used in order to calculate both photon cross sections and electron stopping power for the different tissues. The information obtained is stored in 3D matrices which are called by the dose calculation algorithm developed by the authors. Dose given by photons is added to that given by electrons and converted to Sievert/s. Results are presented in the form of relative dose rate profile curves and relative depth dose rate curves, for each desired plane.

The MCNP code has been parallelized in an SGI Altix 3700, using the MPI parallel protocol, in this case with 6 processors. Furthermore, the MCNP code has been adapted to the problem geometry by increasing the maximum number of entries allowed in one lattice cell [11] to 3000000 with the Intel Fortran Compiler, on the Linux parallel computing machine. The final total computing time for the tallied photons was of 719.4 minutes (11.99 hours), and for the tallied electrons was 651 minutes (10.85 hours).

4. Results and conclusions

The results for the simulation of the phantom irradiation are presented first with a relative dose rate graph for the central transverse plane (from 0 cm to 0.4 cm of the origin of axis), in the form of relative dose rate isodose curves. Results are given in units of Sievert/s.

Figure 9. Relative dose rate isodose curves for an XY plane situated at the center of the phantom.

Figure 9 shows the evolution of the dose given by the photon beam, which is collimated to give a field of 10 cm x 10 cm at the surface (in the x direction) of the phantom, coming from the left side (positive x axis). The dose is highly absorbed in materials with high material density, therefore the highest dose peaks appear in zones where a big change in density occurs, like from air to skin or from skeletal muscle to skull. The angle of the collimated beam is evidenced in the graphs.
The next MCNP dispersion graph presented in figure 10 for the transverse plane, shows error values of between 10% and 5%, which is a bit above the MCNP confidence interval, within the photon beam. Outside the beam, we can observe higher relative error values in zones where not enough particles reach in order to reduce the statistical error.

![Figure 10. Relative error (dispersion) in each voxel of the XY plane situated in the center of the phantom.](image)

The depth dose curve shows peaks of dose absorption where a big change in material density appears, as from air to skin or from skeletal muscle to skull. There the photon collisions produce secondary photons still with enough energy to be ionizing. Thus, the effect in the last part of the curve is smoother.

![Figure 18. Relative depth dose rate with $\pm 2\sigma$ error for the central transverse plane and Y=0.](image)

A dispersion (standard deviation) of between 0.1 and 0.5 is estimated for both photons and electrons at the transverse plane and Y=36. A lower dispersion would allow smoother graphs, which are more common in radiotherapy plans.

5. References


[8] International Commission on Radiation Units and Measurements 2003 Basic Anatomical Data and Physiological Data for use in Radiological Protection: Reference Values ICRU REPORT 89


ABSTRACT

The use of ionizing radiation has led to major improvements in the diagnosis and treatment of patients. However, new developments in medical technology and the increased complexity of medical radiation techniques can produce high doses to medical personnel. In particular, interventional radiology and cardiology and nuclear medicine have been identified as fields where medical staff can receive potentially high doses.

Within this framework, the ORAMED project, a collaborative project funded by the EU FP7, proposes new methodologies to improve standards of protection for medical staff. The main studied topics presented in this paper include extremity and eye-lens protection in interventional radiology and cardiology, extremity dosimetry in nuclear medicine, especially in therapy and positron emission tomography, and the evaluation and the improvement of active personal dosemeters used in pulsed medical radiation fields.

An extensive dose measurement campaign of doses received in extremities and eye-lenses in more than 60 hospitals from 9 European countries has been undertaken. The analysis of the results highlights the most critical procedures and the effectiveness of the protection measures.

1. Introduction

Medical practices are evolving fast, and new techniques with ionizing radiation are being increasingly used. Advances in imaging techniques such as interventional radiology, computed tomography, single-photon emission computed tomography and positron emission tomography enable early diagnosis and have simplified many medical procedures. Monitoring of workers constitutes an integral part of any radiological protection program to ensure that the new medical applications of radiation do not involve unacceptable doses for the personnel exposed to radiation sources or generators.

The ORAMED project (Optimization of RAdiation Protection for MEDical Staff) is a collaborative project funded by the European Atomic Energy Community's Seventh Framework Programme [1]. It started in February 2008 and will be finished in January 2011. 10 research institutes and 2 instrumentation companies from 9 European countries are involved in it. The main studied
topics include extremity and eye-lens protection in interventional radiology and cardiology, extremity dosimetry in nuclear medicine, especially in therapy and positron emission tomography, evaluation and improvement of active personal dosemeters used in pulsed radiation fields, as well as the development of eye-lens dosemeters to be worn by interventionists.

The main objective is to improve the standards for protection of medical staff for procedures that may result in high exposures. This paper describes the proposed method in order to obtain a complete overview of the doses received in potentially highly exposed medical personnel. The preliminary results and conclusions from the measurement campaigns should provide guidelines to reduce the occupational dose in medical practice.

2. Method

An extensive measurement program has been performed in various hospitals in Europe. Two measurement protocols, for interventional radiology and cardiology and for nuclear medicine respectively, were established in order to obtain a set of comparable standardized data on staff doses in the selected procedures. Thermoluminescence detectors were used for the dose evaluation.

In interventional radiology, the measuring points were the two ring fingers, the wrists, the legs and the eyes. The protocol includes detailed instructions on the detector positions and useful information for the data analysis, such as, the type and complexity of the procedure, the position of the operator and the protective equipment, the experience of the operator, the field parameters (kV values, filtration, projections, etc.) and finally the kerma area product (KAP) values. The monitored interventional procedures consisted of cardiac angiographies (CA) and angioplasties (PTCA), radiofrequency ablations (RFA), pacemaker and cardiac defibrillator implantations (PM, ICD), angiographies (DSA) and angioplasties (PTA) of the lower limbs (LL), the carotids (C) and the reins (R), embolisations and endoscopic retrograde cholangiopancreatographies (ERCP).

In the field of nuclear medicine, diagnostics and therapy, including two different phases: preparation and administration were considered in the protocol. Special gloves were designed to measure the hand dose on 11 different points of the hand. Series of 5 measurements for two different technicians for each one of the selected procedures were programmed. Complementary information about the dominant hand of the worker, its experience and the radiation protection equipment used was collected for each monitored worker in order to correlate it with the measured extremity doses.

3. Results

800 interventional procedures in 33 European hospitals and 635 nuclear medicine procedures from 31 European nuclear medicine departments have been monitored.

Figure 1 shows the median doses measured during some selected interventional procedures. It indicates that the highest doses are registered for PM, DSA PTA R
and embolisations. Furthermore, depending on the procedure, the parts of the body that receive the highest doses are left finger (PM/ICD, DSA PTA LL, DSA PTA R) and left wrist (CA PTCA, RFA, DSA PTA C&C, embolisations, ERCP). The highest doses to the eyes were measured during embolisations (0.085 mSv for the left eye and 0.059 mSv for the region between eyes).

Figure 1: Median values of $H_p(0.07)$ for various interventional procedures in the different anatomic regions (LR-left ring, RR-right ring, LW-left wrist, RW-right wrist, LL-left leg, RL-right leg, L/R E-lateral eye, ME-region between eyes).

The most important parameters influencing the operator’s dose were found to be his position with respect to the X-ray tube, availability and use of additional protective equipment, tube configuration, and individual behavior [2]. It is shown that using, both, table and ceiling shielding, the doses to legs and eyes are reduced significantly. When the X-ray tube is above the table the doses to the hands and the eyes are in general much higher than when the tube is below the table. However, the doses to legs are higher when the tube is below the table. Nevertheless, the doses to the legs are low, for both tube configurations, provided that a table shielding is used. Leaving the operating room during cine is found to be very effective in reducing the operator dose. Moreover, it is essential to avoid exposing the hands in the primary beam to maintain acceptable extremity doses.

In the field of nuclear medicine, in order to compare measurements from different departments, the measured doses were normalized to the manipulated activity. Figures 2 and 3 show the average doses found for each measuring position for diagnostic and therapy procedures, respectively.
Figures 2 and 3 show large variations of skin doses across the hands, with mean normalised skin doses from 0.07 to more than 7 mSv/GBq, depending on the radionuclide and the procedure, and individual worker values up to 35 mSv/GBq [3]. The maximum skin dose is usually measured at the index tip/nail and thumb of the non dominant hand, thus the positioning of the dosemeter used in routine monitoring strongly affects the estimates of the extremity doses. From the available results it is shown that a ring dosemeter should be preferable to a wrist one for extremity monitoring in nuclear medicine. In general, mean normalised skin doses are much higher in therapy than in diagnostic procedures and more specifically, they are higher during the radiopharmaceutical preparation than during administration. In addition, a large variation between Nuclear Medicine departments and, even, between individuals at the same department is found. This highlights the need to improve the present handling procedures in some of the monitored departments, in order to reduce personnel doses.

4. Conclusions

In most of the studied situations, the organ at risk is the skin. Depending on the workload, some of the monitored workers could surpass the legal skin dose equivalent limit of 500 mSv per year, averaged over any skin area of 1 cm² regardless of the area exposed [4]. Doses to the eye lens are generally at low levels, except for embolisations, however, if the current eye lens dose limit [5] is lowered they could become of concern.
The preliminary results reported in this paper have identified the most critical parameters to reduce personnel dose and have already helped to establish some first recommendations for good practice in the field of occupational radiation protection in interventional radiology and cardiology and nuclear medicine. Monte Carlo calculations are under progress and should provide a better understanding of the measurements and the effectiveness of the available protection means. In January 2011, an international workshop to present the final results of the ORAMED project will be organized in Barcelona.

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5. References

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Plant operations
ELECTRIC AND MECHANICAL BASIC PARAMETERS TO
ELABORATE A PROCESS FOR A TECHNICAL VERIFICATION OF
SAFETY RELATED DESIGN MODIFICATIONS

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ABSTRACT

This paper presents a systematic process to check a design in order to achieve all the requirements that regulations demand. Nuclear engineers must verify that a design is done according to the safety requirements, and this paper presents how we have elaborated a process to improve the technical project verification.

For a faster, better and easier verification process, here we summarize how to select the electric and mechanical basic parameters, which ensure the correct project verification of safety related design modifications. This process considers different aspects, which guarantee that the design preserves the availability, reliability and functional capability of the Structures, Systems and Components needed to operate the Nuclear Power Station with security. Electric and mechanical reference parameters are identified and discussed as well as others related ones, which are critical to safety.

The implementation procedure to develop tasks performed in any company that has a quality plan is a requirement. On the engineering business, it is important not to use the personal criteria to do a technical analysis of a project; although, many times it is the checker’s criteria and knowledge responsibility to ensure the correct development of a design modification. Then, the checker capabilities are the basis of the modification verification.

This kind of procedure’s development is not easy, because in an engineering project with important technical contents, there are multiple scenarios, but lots of them have a common basis. If we can identify the technical common basis of these projects, we will make good project verification but there are many difficulties we can encounter along this process.

1. Introduction

The Design verification is the confirmation of various applying aspects of a design modification in order to guarantee the availability, reliability, and the functional capability of the structures, systems and components (in the future SSC) necessary to the safe operations of the plant.

The Technical Procedures Handbook published by the Consejo de Seguridad Nuclear (in the future CSN) dedicated to the design modifications in a Nuclear Power Plant, is a good guide to know the important points to inspect in a design modification of a new or existing SSC. The important points to be considered when performing an SSC design verification related to security are defined in this procedure. In the Attachment 71111.17 for a permanent Plant Modifications and the 71111.18 for Plant Modifications published by the NRC for the plant
modifications inspections requirements, show a general inspection guidance to assure that the modifications to risk-significant structures will not affect safety functions of important safety systems.

In this paper we will not explain the complete technical procedure. It is a summary of the complete verification procedure, and is based on the technical procedures published by the CSN and the NRC described before. Only basic electrical and mechanical parameters will be described in this paper, focused in permanent plant modifications.

2. Design review

2.1 Energy needs, electricity, steam, fuel and air

Important things to considerer in a design related with energy needs are the criteria 2, 4, 5, 17, 18 and 50 of the appendix A to Part 50 (General Design Criteria for Nuclear Power Plants, on the future GDC). Criterion 17 refers to the electric power systems, and is the basis to design the offsite and outside electric power supplies; Section 8 of the Standard Review Plant for the Review of Safety Analysis Reports for Nuclear Power Plants (NUREG-0800 SRP) is related to the electric power. Section 8.1 establishes the primary responsibility of the electric power to the organization responsible for electrical engineering. This section has a matrix on table 8-1 to identify the acceptance criteria, the guidelines and their applicability referred to the electric power systems. The appendix 8-A of the SRP referred to the Branch Technical Positions represent the guidelines to supplement the acceptance criteria established in Commission Regulations, guidelines presented in Regulatory Guides, and recommendations presented in applicable IEEE standards.

When doing design modification, it is necessary to verify whether the energy requirements generated by the new or the existent SSC can be supplied by the supporting systems when they are required under an accident conditions. To assure that the energy supplied is enough, it is necessary to identify clearly which are the supporting systems from the SSC, and verify that the total power installed in that supporting system is enough and appropriate to the service that it will supply. However, it is important that the existing energetic requirements are not modified beyond the design basis, when the rating power of the new SSC is added.

To verify the electric energy needs, it is only necessary to check that the total power of the load list of all elements connected to the supporting system is less than the total power available, including the simultaneity coefficient. The simultaneity coefficient must be acceptable and demonstrable. Sometimes this procedure is not enough because a lot of electric equipment has energy storage elements that have a dynamic response in its connecting and disconnecting operation. This factor must be taken into consideration to increase the electric energy needs, multiplying by a coefficient the rating power of the electric equipment. The Spanish Law Voltage regulatory establishes in its 47th Complementary Technical Instruction, good guidelines to establish the power need for the power source, and the wire sizing used to supply the energy to the SSC, when the load is a machine. When the load is a transformer, the power source for a LV-LV transformer must have the capability to supply the energy requirements that the transformer characteristic has (ITC-BT-48). Protections and power sources for a transformers with one side connected to a HV power line, must permit certain level of overload depending of the environmental temperature, foresight load and the duration of the overload (this characteristics are exposed in ANSI/IEEE C.57.92 and ANSI C.57.96). In all cases, a HV system must comply with the MIE-RAT-09 included in the Regulatory about Technical Conditions and Security Warranties in Electric Power Plants, Substations and Transformers Stations.

If the energy requirements could not be accomplished with total security using the static methods described before, a dynamic analysis must be done to justify the new dynamic situation as described in section 2.3.

In case the energy source is fuel, steam or air we should also make sure energy requirements are enough for the SSC to perform its safety feature under accident/event conditions. Any flow or increase of energy or physical conditions change should be verified as they could lead to a function variation.
If it’s required, redundancy, diversification and physical and electrical separation should be met.

2.2 Materials / Replacement Components. Material Compatibility / Functional Properties / Seismic – Environmental Qualification / Classification

Verify that the Code and Safety classification of replacement SSCs is consistent with design bases.

If we replace mechanical equipment, it should be verified if it is seismically qualified according to RG 1.29 “Seismic Design Classification” and environmentally qualified according to RG 1.26 “Quality Group Classifications and Standards for Water, Steam and Radioactive-Waste Containing Components of Nuclear Power Plants”. If it is an electrical replacement it should comply with IEEE-308 “Criteria for Class 1E Power Systems for Nuclear Power Generation Stations”.

All security related SSC should be seismically and environmentally qualified if they are located in severe conditions and require to be working in case of accident.

Every seismically qualified SSC should comply with RG 1.100 which endorses IEEE-344 “Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations”. If it is also environmentally qualified it should comply with RG 1.89 which endorses IEEE-323 “Qualifying Class 1 Electric Equipment for Nuclear Power Generating Stations” too.

Apart from the generic regulation for equipment qualification, equipment should comply with its specific regulation such as ASME Boiler Pressure Vessel Code for mechanical equipment, IEEE-383 “Type Test of Class 1E Electrical Cables, field slices and connections for Nuclear Power Generating Stations” for cables, IEEE-382 "IEEE Standard for Qualification of Safety-related Valve Actuators" for valve actuators.

Materials or replacement components properties should serve functional requirements under accident or event conditions. Replacement schedule should be consistent with in-service/equipment qualification life.

2.3 Timing: Sequence / Response Time / Duration

In a design modification we should verify that any sequence changes of a design basis event are bounded by accident analyses and loading on support systems are acceptable.

Any modified or new SSC should be able to function for the duration required under accident/event conditions.

Response time for each element should be sufficient to serve accident/event functional requirements assumed by design analyses. In case there is a response time modification, its impact during a transient should be studied.

Modified SSC response time should not cause an unintended interaction with other SSCs. If there is a valve modification we should guarantee that the time response to perform its security function is not affected.

When the static analysis is not enough to assure that the electric state variables are in the range of the plant licensing, it is necessary to do a dynamic analysis. This is necessary, for example, to assure that the acceptance criteria of the power system is according to the Appendix 8-A. Branch Technical Positions (PSB) of the NUREG-0800 §8.2 “Offsite Power System”.

To ensure that the dynamic requirements of the dc power system have been met, it is necessary to assure that the capacity of the dc supply is adequate to the power prescribed loads. IEEE Std 485 provides an acceptable method for sizing stationary lead acid batteries, and the Regulatory Guide 1.128 “Installation Design and Installation of vented lead-acid storage batteries for nuclear power plants”, that endorses the IEEE Std-484 “IEEE recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations”, provides the basis of the review design. Special care must be taken with the time gap between the actuation of different contacts of hand switches and relays. Little differences between them could affect the complete system.
2.4 Heat Removal

In any change of design we should verify that any function of SSC used to remove heat in case of Design Base Accident is not affected, and heat removal requirements can be addressed by support systems under accident/event conditions. If it’s required, redundancy, diversity and physical and electrical separation should be met. Any coolant condition modification in a heat exchanger could lead to an efficiency variation to transfer heat. An example could be an increase of water hardness which could lead to mineral salts incrustation. Ductwork, valves and equipment capacity variations should be able to transfer the necessary heat in normal operation as well as in accident conditions with its operational safety margins. All containment Heat Removal Systems should comply with the requirements of GDCs 38, 39, 40. Reactor Auxiliary Cooling water Systems should comply with GDCs 44, 45, 46. GDC 34 requires the capability to transfer decay heat and other residual heat from the reactor such that fuel and pressure boundary design limits are not exceeded. Emergency Core Cooling System should comply with GDCs 35, 36, 37.

2.5 Equipment Protection: Fire / Flood / Missile / High Energy Line Break / Freeze

In any design modification, it is mandatory to justify that any new risks have been generated apart from the ones analyzed before, like: piping failures (break / crack), missile emission, earthquakes, explosions/fire and heavy load handling. It is necessary to justify that the effects generated by the previous list of risks will not be modified, and also all protective elements installed to protect in front of that risks. To sum up, assure that the modification maintains the protection level of the SSC important for the security and the dynamic risks postulated (GDCs 2, 3, 4 and R.G. 1.59: “Design Basis Floods for Nuclear Power Plants).

2.6 Flowpaths

Revised flowpaths should serve system functional requirements under accident/event conditions. Ductwork, electrical trays, new geometries or interfaces with other systems as well as incorporation of new elements could compromise SSC functional requirements. New seals or weldings should be carried out according to specifications so as to avoid leakage. The requirements for physically separated has been described by the NRC Regulatory Guide 1.75 “Physical Independence of Electric Systems”. This RG provides acceptable criteria of complying with the IEEE Std 279. IEEE Std 384 “IEEE standard Criteria for Independence of Class 1E Equipment and Circuits” is generally acceptable and provides an adequate basis for complying with IEEE Std 279. Some subjects of the IEEE Std 384 have been considered in the regulatory position of the RG 1.75, and it is a mandatory to accomplish them.

2.7 Pressure Boundary

Pressure boundary integrity (cladding, RCS and containment) should not be compromised after the change of design. GDCs 14, 30, 31, 32, 51, 55 should be taken into consideration. Any change of material, diameter, and schedule should comply with ASME II, III and its applicable subsection. Welding should comply with ASME IX and quality assurance program. Any process condition changes shouldn’t exceed the acceptable limits, and adverse effects such as fatigue, stress, vibration, corrosion or transients different from the studied ones should be verified. If it’s necessary, a support calculation should be done in order to verify that the obtained values are below the acceptable limits and always guarantee the integrity of the pressure boundary. As required by GDC 14, safety-related valves that are part of the Reactor Coolant Pressure Boundary should be designed and tested such that these valves will not experience any abnormal leakage, or increase in leakage, from their loading.
Meeting GDC 51 will ensure that the containment pressure boundary remains intact during the harshest expected conditions, thereby precluding the release of radioactivity to the environment.


2.8 Ventilation Boundary

We should verify the ventilation barrier integrity after any change of design in order to prevent radioactive emissions and/or contamination propagation and prevent infiltrations to clean areas.

NUREG 0800 Chapter 9.4 should be taken into account for ventilation and air filtration. R.G. 1.52 “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants” provides guidance and criteria for the design, inspection, and testing of air filtration and iodine adsorption units of engineered-safety-feature (ESF) atmosphere cleanup systems in light-water-cooled nuclear power plants.


Filters should be included whenever necessary so as to prevent uncontrolled release of radioactivity to the environment as well as occupied indoor areas.

NUREG 0696 “Functional Criteria for Emergency Response Facilities” and GDC 19 should be considered to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess.

GDCs 41, 42, and 43 require that containment atmosphere cleanup systems be provided as necessary to reduce the amount of radioactive material released to the environment following a postulated Design Basis Accident.

The power supply and electrical distribution system for the ESF atmosphere cleanup system should be designed in accordance with RG 1.32.

Doors, penetrations, leak-tight dampers, materials, joining compounds, filters, weldings, and ductwork should be checked in every modification in case they cause a ventilation boundary infiltration change or a release to the environment.

We should guarantee the ventilation system functionality under accident or event conditions checking that the modified SSCs don’t change process or operation conditions such as new pressure losses, change of ventilation boundary.

If it’s necessary calculations of radiological consequence in case of accident should be modified or calculated. Likewise, in case of fatigue or vibration modification, new calculations should be performed verifying supports.

2.9 Process Medium: Fluid Pressures / Fluid Flow rates / Voltages / Currents

Regulatory Guide 1.32 (Criteria for Power Systems for Nuclear Power Plants) endorses IEEE Std 308 (Criteria for Class 1E Power Systems for Nuclear Power Generation Stations), and gives references to comply with 1E safety related class systems, for both alternating current and direct current power systems, except for multi-unit nuclear power plants, as described in Regulatory Guide 1.81.

To verify that voltage and current variables are adequate to SSC requirements it is necessary to verify that voltage and current values are between the maximum and minimum values specified in each case, taking into consideration the degraded voltage established in the NPP. Technical guidance application Annex 2, published by Ministerio de Ciencia y Tecnologia, gives methods to calculate voltage drop in cable systems, and IEEE red book recommends that steady-state voltage drop in power, heating or lighting feeders should be no more than 3%, and the total drop including feeders and branch circuits should be no more
than 5% overall. To select cables it is necessary to verify voltage drop to assure that voltage is always higher than degraded voltage and assure that cable ampacity is higher than the maximum load current. A good guidance to make ampacity calculations is the Instrucción Técnica Complementaria ITC-BT-19 (part of the Reglamento Electrotécnico para Baja tensión). This regulatory endorses UNE 20.460-5-523. It contains the requirements that ensure the satisfactory life of conductors and insulators subjected to thermal effects of admissible ampacity in different forms of installation. To assure that the new protective elements are in accordance with the new requirements, it is necessary to assure that cable ampacity is always higher than overcurrent protective device trip. The coordination of System protective devices must be calculated to assure that the energy needs of other SSC are not suspended due to a fault in the SSC affected by the modification. UNE UNE 20-460-1 (Instalaciones Eléctricas en Edificios. Parte 1: Campo de aplicación, objeto y principios fundamentales) gives valid rules to do a safety electric network. UNE 20-460-90 parte 4-473 (Instalaciones eléctricas en edificios. Protección para Garantizar la Seguridad. Aplicación de las medidas de protección. Protección contra las sobreintensidades) gives guidelines for implementing measures to ensure security and electric protection against overcurrent for facilities to which is applied UNE 20-460-1. Interrupting rating of circuit breakers must be higher than the maximum short circuit calculated in the protected line.

Another consideration to take into account in a design modification is IEEE STD 279-1971 “Criteria for Protection Systems for Nuclear Power Generating Stations”. This IEEE establishes criteria for protection systems in a Nuclear Power Plants. Also, for motor operated valves it is mandatory to comply with the R.G. 1.106 “Thermal Overload Protection for Electric Motors on Motor-Operated Valves” to prevent from non desirable actuations.

Diameter change, volumetric coefficient, or any other pressure loss change that implies an increase or decrease of flowrate should be revised. Transients and its important parameters (pressure, temperature, flowrate…) should be revised as well as possible cavitation or water hammer effects.

**2.10 Failure Modes**

We should check that those failure modes introduced by the modification are bounded by existing analyses. Verify that the proposed design modification neither leads to any SSC new failure mode or malfunction nor modifies the effects or consequences to postulates failures. With this aim, we verify that the nuclear safety criteria requirements are met (simple failure, redundancy, physical and electrical separation and independency). These requirements are established in ANSI 51.1 “American National Standard Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants”.

After the modification, the Plant Secure Operation should be kept; not leading to any considered and analysed failure mode, as well as the security function reliability performed by directly or indirectly modified SSC.

**3. References**

1. 10 Code Federal Regulation Part 50. Domestic licensing of production and utilization facilities.
4. R.G. 1.29 “Seismic Design Classification”
7. R.G. 1.89 “Environmental Qualification of Certain Electric Equipment important to safety for Nuclear Power Plants”
8. R.G 1.100 “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants”
10. Regulatory Guide 1.128 “Installation Design and Installation of vented lead-acid storage batteries for nuclear power plants”.
15. ASME Boiler Pressure Vessel Code
16. ASME N509-1989, "Nuclear Power Plant Air-Cleaning Units and Components"
17. ASME N510-1989, "Testing of Nuclear Air-Treatment Systems"
21. IEEE 323 “Qualifying Class 1 Electric Equipment for Nuclear Power Generating Stations”
22. IEEE-344 “Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations”
27. UNE 20-460-1 Instalaciones Eléctricas en Edificios. Parte 1: Campo de aplicación, objeto y principios fundamentales.
30. UNE-EN 60439-1. Conjuntos de Aparamenta de Baja Tensión. Parte 1: Conjuntos de serie y conjuntos derivados de serie.
MULTI-SPHERE UNIT CELL MODEL TO CALCULATE THE EFFECTIVE THERMAL CONDUCTIVITY IN PEBBLE BED REACTORS

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ABSTRACT

A proper understanding of the mechanisms of heat transfer, fluid flow and pressure drop through a packed bed of spheres is of utmost importance in the design of a high temperature Pebble Bed Reactor (PBR). While the gas flows predominantly in the axial direction through the bed, the total effective thermal conductivity is a lumped parameter that characterises the total heat transfer in the radial direction through the packed bed. The study of the effective thermal conductivity is important because it forms an intricate part of the self-acting decay heat removal chain, which is directly related to the PBR safety case.

The effective thermal conductivity is the summation of various heat transport phenomena. These are the enhanced thermal conductivity due to turbulent mixing as the fluid passes through the voids between pebbles, heat transfer due to the movement of the solid spheres and thermal conduction and thermal radiation between the spheres in a stagnant fluid environment. In this study, the conduction and radiation between the spheres are investigated.

Firstly, existing correlations for the effective thermal conductivity are investigated, with particular attention given to its applicability in the near-wall region. Several phenomena in particular are examined namely: conduction through the spheres, conduction through the contact area between the spheres, conduction through the gas phase and radiation between solid surfaces.

A new approach to simulate the effective thermal conductivity for randomly packed beds is then presented, namely the so-called Multi-sphere Unit Cell Model. The model is validated by comparing the results with that obtained in experiments.

Nomenclature

<table>
<thead>
<tr>
<th>BEM</th>
<th>Boundary Element Method</th>
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<tbody>
<tr>
<td>CFD</td>
<td>Computational Fluid Dynamics</td>
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<tr>
<td>ENEN</td>
<td>European Nuclear Education Network</td>
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<tr>
<td>HTTU</td>
<td>Heat Transfer Test Unit</td>
<td></td>
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<tr>
<td>PBR</td>
<td>Pebble Bed Reactor</td>
<td></td>
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<tr>
<td>PBMR</td>
<td>Pebble Bed Modular Reactor</td>
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<td></td>
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<tr>
<td>As</td>
<td>Surface area of sphere, m²</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Ar</td>
<td>Conduction area, m²</td>
<td></td>
<td></td>
</tr>
<tr>
<td>a</td>
<td>Empirical constants</td>
<td></td>
<td></td>
</tr>
<tr>
<td>at</td>
<td>Thermal accommodation coefficient</td>
<td></td>
<td></td>
</tr>
<tr>
<td>dp</td>
<td>Sphere diameter, m</td>
<td></td>
<td></td>
</tr>
<tr>
<td>F1-2</td>
<td>Diffuse view factor between surface 1 and 2</td>
<td></td>
<td></td>
</tr>
<tr>
<td>k_e</td>
<td>Radiation conductivity, W/mK</td>
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<td></td>
</tr>
<tr>
<td>k_o</td>
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<td>K</td>
<td>Temperature, K</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Lr</td>
<td>Geometrical radiation length, m</td>
<td></td>
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<tr>
<td>n</td>
<td>Coordination flux number</td>
<td></td>
<td></td>
</tr>
<tr>
<td>r_o</td>
<td>Contact Radius, m</td>
<td></td>
<td></td>
</tr>
<tr>
<td>r_L</td>
<td>Long-range radiation conductivity, W/mK</td>
<td></td>
<td></td>
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<tr>
<td>r_S</td>
<td>Short-range radiation conductivity, W/mK</td>
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<tr>
<td>Q</td>
<td>Heat flux, W</td>
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<td></td>
</tr>
<tr>
<td>r</td>
<td>Conduction area, m²</td>
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<tr>
<td>r</td>
<td>Coordination flux number</td>
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<tr>
<td>r</td>
<td>Contact Radius, m</td>
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1. Introduction

Inherent safety is claimed for gas-cooled Pebble Bed Reactors (PBRs) as a result of its design characteristics, materials used, fuel type and physics involved. Therefore, a proper understanding of the mechanisms of heat transfer, fluid flow and pressure drop through a packed bed of spheres is of utmost importance in the design of a high temperature PBR. In this study, some correlations describing the effective thermal conductivity through packed pebble beds are examined. The effective thermal conductivity is a lumped parameter which is a representative of the overall radial heat transfer through such a packed bed of spheres, and is a summation of various components of the overall heat transfer.

This phenomenon is of importance because it forms an intricate part of the self-acting decay heat removal chain, which is directly related to the PBR safety case. In this study standard correlations generally employed by the thermal fluid design community for PBRs are investigated, giving particular attention to the applicability of the correlations when simulating the effective thermal conductivity in the near-wall region.

The bed effective thermal conductivity ($k_{bed}$) in high temperature PBRs is derived by lumping all of the relevant heat transfer mechanisms into a single value representing the heat transfer through a packed bed of spheres. The concept of the effective thermal conductivity in a bed saturated with a stagnant gas can be split up into three components [2].
The first component is that of the effective thermal conductivity \( (k_{\text{eff}}) \) consisting of four distinct heat transfer mechanisms namely: (1) conduction through the solid; (2) conduction through the contact area between adjacent spheres with a smooth surface; (3) conduction through the stagnant gas phase; and (4) thermal radiation between the solid surfaces. The second component is the enhanced fluid effective conductivity \( (k_{\text{f,eff}}) \) due to the turbulent mixing of the highly irregular flow paths while the solid phase is motionless, also referred to as the braiding effect or dispersion. Lastly, the third component is when the gas phase as well as the solid phase is in motion \( (k_{s,\text{eff}}) \) because of stirring or vibrations in the packing. The bed effective thermal conductivity is therefore given by:

\[
k_{\text{bed}} = k_{\text{eff}} + k_{\text{f,eff}} + k_{s,\text{eff}}
\]

(1)

The focus of this study is to analyze the porous structure in a randomly packed annular pebble bed and its impact on the first component of the bed effective thermal conductivity, i.e. \( k_{\text{eff}} \).

2. Analysis of Effective Thermal Conductivity

Most of the difficulties encountered in predicting the effective thermal conductivity are associated with the fact that it is a phenomenological characterization of a solid-fluid medium rather than a thermo-physical property [1]. Therefore, before any heat transfer analysis is attempted, one should have a thorough understanding of the structural arrangement of the packed bed under consideration.

At higher temperatures the thermal radiation component of the effective thermal conductivity dominates the other components [26]. This is also mentioned by Zou et al. [28] who developed a two dimensional Boundary Element Method (BEM) for the evaluation of the effective thermal conductivity in pebble beds. Not only did they note that radiation becomes very important at high temperatures, but also that a variation in sphere diameter and solid conductivity has an influence on the radiative exchange. Therefore, the porous structure and thermal radiation will be the main focus of this paper. To address these components a new approach denoted as the Multi-sphere Unit Cell Model was developed. It addresses the porous structure in a more explicit manner and sub-divides the heat transfer in two primary components as follows:

\[
k_{\text{eff}} = k_{g,c} + k_{r}
\]

(2)

where the thermal conductivity \( (k_{g,c}) \) incorporates conduction through the solid, conduction through the contact area between spheres and conduction through the gas phase, while the radiative thermal conductivity \( (k_{r}) \) addresses the thermal radiation between the solid surfaces.

2.1 Porous Structure

The porous structure of a packed bed is commonly characterized in terms of the bulk porosity (void fraction) or variation in the porosity in the axial or radial directions. However, Du Toit et al. [8] showed that the porous structure can in addition be characterized successfully with the coordination number and the contact angles between adjacent spheres. The coordination number \( (N_{c}) \) is defined as the number of spheres in contact with the sphere under consideration, whilst the contact angle \( (\phi_{c}) \) is defined as the angle between the line connecting the centre points of two spheres in contact and the line perpendicular to the direction of the heat flux.
Employing only porosity to quantify the porous structure in effective thermal conductivity calculations is usually valid as long as the model is used within specified porosity bounds [22]. Van Antwerpen [22] stated that the most common simplification associated with the effective thermal conductivity models found in the open literature is to assume some form of structured packing arrangement for a specific porosity range, such as Simple Cubic (SC), Body Centre Cubic (BCC) and Face Centre Cubic (FCC), to quantify the porous structure in the bulk and near-wall regions in a randomly packed bed. Table 1 shows that the densest packing one can obtain in a structured packed bed is that of FCC. In addition, [8] showed that in a randomly packed bed at 0.5d_p away from any wall a radial porosity of approximately 0.2595 is also achieved. However, in contrast they showed that the average coordination number for a randomly packed bed with a porosity of approximately 0.2595 is in the vicinity of 4.5 not 12.

<table>
<thead>
<tr>
<th>Type of Arrangement of Spheres</th>
<th>Porosity (ε)</th>
<th>Coordination Number (N_c)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Simple Cubic</td>
<td>0.35 – 0.5</td>
<td>6</td>
</tr>
<tr>
<td>Body Centered Cubic</td>
<td>0.2595 – 0.35</td>
<td>8</td>
</tr>
<tr>
<td>Face Centered Cubic</td>
<td>≤ 0.2595</td>
<td>12</td>
</tr>
</tbody>
</table>

This simplification of the porosity can also be traced to the well known Schlünder and co-workers [3,26] unit cell model. The model uses an empirical function \( k_g = (1 - \sqrt{1 - \varepsilon}) k_g \) to quantify the effective thermal conductivity (point contact/gas conduction) as a function of porosity (\( \varepsilon \)). This empirical function was developed by fitting a correlation through data obtained from various structured packings, which was specifically packed in an ordered manner to obtain certain porosities. This simplification is therefore only valid if used in the bulk region of a randomly packed bed where it has a better correspondence to a structured packing. For that reason Du Toit et al. [8] stated that defining the porous structure using porosity alone is not sufficient.

The porous structure varies sharply near any wall, as the geometry of the packing is disrupted in this region. This wall effect is composed of two separate components namely the effect of the sidewall (radial direction) and the effect of the top-bottom wall (axial direction), referred to by Zou et al. [29] as the thickness effect. In this study only the effect of the porous structure in the radial direction is investigated.

One fundamental distinction between pebble bed types must be highlighted i.e. a stationary and a slowly moving pebble bed. A stationary pebble bed consists of a packed bed with no moving pebbles, whilst a slowly moving pebble bed is a bed where the pebbles are removed at the bottom and replaced at the top. Various reactors have been envisaged with a slowly moving pebble bed for continuous fuel loading. The core structures of some of these reactors have at least in part been designed with dimples/grooves at the surface of the inner and outer reflector to avoid the order-effect [24] and the associated uncertainties.

This study will focus on methods to analyse the porous structure in a stationary annular packed bed with smooth reflector surfaces, which is randomly packed. Therefore, the effects of dimples or grooves at the surface of the inner and outer reflectors are not taken into consideration, and aims to address the uncertainties associated with the wall effect more directly.
Van Antwerpen [22] subdivided the area next to the wall into two regions, namely the wall region defined as $0 \leq z \leq 0.5$ and the near-wall region defined as $0.5 < z \leq 5$ as displayed in Figure 1, where $z$ is the number of sphere diameters away from any wall. This is important due to the fact that a separate set of equations (developed in [22]) should be used in the wall region as opposed to the near-wall and bulk regions when calculating the effective thermal conductivity. For this study only the near-wall and bulk regions are considered.

![Figure 1: Packing regions](image)

Van Antwerpen et al. [23] reviewed the different radial porosity models and showed that the model developed by De Klerk [6] compared best with the experimental and numerical radial porosity data presented by Du Toit [7]. The oscillatory porosity model of De Klerk [6] is rewritten for an annular packed bed and is given by:

$$
\varepsilon(z) = \begin{cases} 
2.14z^2 - 2.53z + 1 & z \leq 0.637 \\
\varepsilon_b + 0.29\exp(-0.6z) & z > 0.637 \\
\times \left[ \cos(2.3\pi(z - 0.16)) \right] + 0.15\exp(-0.9z) & 
\end{cases}
$$

(3)

$$
z = \frac{(r - R_i)}{d_p} \quad \text{for} \quad R_i \leq r \leq \frac{(R_o + R_i)}{2}
$$

(4)

$$
z = \frac{(R_o - r)}{d_p} \quad \text{for} \quad \frac{(R_o + R_i)}{2} \leq r \leq R_o
$$

(5)

(6)

where $\varepsilon(z)$ is the variation in the porosity in the radial direction as a function of sphere diameters from the inner wall, $\varepsilon_b$ the porosity in the bulk region, $R_i$ the radius of the inner reflector, $R_o$ the radius of the outer reflector, $r$ the radial distance from the center and $d_p$ the sphere diameter.

Van Antwerpen et al. [23] also investigated the variation in the average coordination number ($\bar{N}_c$) in the radial direction as a function of porosity in a randomly packed bed and developed an empirical correlation which accounts for the bulk and near-wall regions given by:

$$
\bar{N}_c = 25.952\varepsilon^3 - 62.364\varepsilon^2 + 39.724\varepsilon - 2.0233 \quad (0.2398 \leq \varepsilon \leq 0.54)
$$

(7)

Furthermore, Van Antwerpen et al. [23], have developed an expression for the relationship between the average contact angle and average coordination number. A contact angle of
$\phi_\text{c} = 0^\circ$ implies that the touching sphere does not contribute to the overall heat transfer between the two spheres in the radial coordinate direction, whilst a contact angle of $\phi_\text{c} = 90^\circ$ implies maximum contribution to the heat transfer. The following empirical correlation was developed:

$$\overline{\phi}_\text{c} = -6.1248\overline{N}_\text{c}^2 + 73.419\overline{N}_\text{c} - 186.68$$  \hspace{1cm} (8)

It must be noted that both coordination number and contact angle could have been derived as a function of sphere diameters away from any wall. However, it is done in this manner due to the porosity being the parameter mostly used to characterize porous structure.

Another aspect to consider in the near-wall region is the length scales or discretisation lengths of the porous structure. Porous structure parameters such as porosity, coordination number and contact angle can be obtained by employing a fine discretisation of the empirical porosity equations. However, in reality only one contact angle and one coordination number value can exist between two adjacent spheres and a fine discretisation may be misleading. This has lead to a further simplification of the description of porous structure in the near-wall region denoted as a porosity correction factor ($\varepsilon^*$), valid for $0.5 < z \leq 3.8$.

![Figure 2: Two-dimensional radial distribution function for the High Temperature Test Unit ($\Delta r = 1.5 \text{mm}$)](image)

The porosity correction factor ($\varepsilon^*$) is developed as an empirical correlation through porosity points where the probability of finding sphere centres in the near-wall region is the highest using a two dimensional radial distribution function given by Tanemura [19]. Results are displayed in Figure 2 based on a numerically packed bed developed for the High Temperature Test Unit (HTTU), a test facility constructed as part of the Pebble Bed Modular Reactor (PBMR) program [14]. The porosity correction factor was developed to be used in collaboration with Eqns. (3) and (4) to simulate a smoother reduction in average coordination number and contact angle in the near-wall region, implying the calculation of the porous structure parameters at the highest probabilities of finding sphere centres in the radial direction. This porosity correction factor is given by:

$$\varepsilon^* = -0.0127z^2 + 0.0967z - 0.2011$$  \hspace{1cm} (9)

$$z = (r - R_i)/d_p \quad \text{for} \quad R_i \leq r \leq (R_0 + R_i)/2$$  \hspace{1cm} (10)

$$z = (R_0 - r)/d_p \quad \text{for} \quad (R_0 + R_i)/2 \leq r \leq R_0$$  \hspace{1cm} (11)
2.2 Multi-sphere Unit Cell Model (Conduction)

The conduction component of the Multi-sphere Unit Cell Model is based on calculating a joint thermal resistance in a unique manner from an arrangement of parallel and series thermal resistances in different regions. It is unique in the sense that it divides the effective thermal conductivity between two spheres into three regions i.e. inner, middle and outer. Each of these regions addresses specific heat transfer mechanisms which are summed to give the joint thermal resistance ($R_j$). It is further assumed that sphere one and sphere two consist of the same material, which entails that the solid material thermal resistances in a specific region can be summed together. An illustration of this thermal resistance network denoted as the Hertzian contact network is given in Figure 3.

![Figure 3: Multi-sphere Unit Cell Model (conduction)](image)

The Hertzian contact network consists of several thermal resistance components: (1) the inner solid material resistance ($R_{in,1,2}$) (summation of $R_{in,1}$ and $R_{in,2}$); (2) the macrocontact constriction/spreading Hertzian resistance between two hemispheres ($R_{HERTZ,1,2}$) developed by Chen and Tien [5]; (3) the middle solid material resistance ($R_{mid,1,2}$) (summation of $R_{mid,1}$ and $R_{mid,2}$); (4) the resistance of the interstitial gas in the Knudsen regime (Smoluchowski effect) of the macrogap ($R_{G1}$); (5) the outer solid material resistance ($R_{out,1,2}$) (summation of $R_{out,1}$ and $R_{out,2}$); and (6) the resistance of the interstitial gas in the macrogap ($R_{C}$). The Hertzian contact network in essence discards surface roughness and treats solid surfaces as smooth. The thermal resistances are given in Table 2.

The thermal conductivity of dilute gases is, according to the molecular theory of gases, independent of pressure. This is valid only if the molecular mean free path ($\lambda$), of the gas molecules is small compared to the geometrical dimension ($d$) of the corresponding voids [20]. The molecular mean free path is defined as the average distance a gas molecule travels before colliding with another gas molecule [10].
Table 2: Various thermal resistances of Multi-sphere Unit Cell Model

<table>
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<th>FORMULA</th>
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<tbody>
<tr>
<td>$R_j = \left( \frac{1}{R_{\text{HERTZ},1,2}} + \frac{1}{R_3} + \frac{1}{R_G} \right)^{-1} \left( \frac{1}{R_{\text{in},1,2}} + \frac{1}{R_{\text{mid},1,2}} + \frac{1}{R_{\text{out},1,2}} \right)^{-1}$</td>
<td>(12)</td>
<td>[22]</td>
<td>Calculation of joint thermal resistance with the Hertzian contact network configuration.</td>
</tr>
<tr>
<td>$R_{\text{HERTZ},1,2} = \frac{0.64}{k_s r_c}$</td>
<td>(13)</td>
<td>[5]</td>
<td>Hertzian contact radius $r_c$ calculated by $r_c = \left[ 3\left(1 - \frac{\mu^2}{4E} \right)Fr_p \right]^{1/3}$</td>
</tr>
<tr>
<td>$R_{\text{in},1,2} = \frac{(d_p - \epsilon_0)}{k_s \pi r_c^2}$</td>
<td>(14)</td>
<td>[22]</td>
<td>Calculation of the bulk solid material in the inner region, $\epsilon_0 = \frac{r_c^2}{2r_p}$</td>
</tr>
<tr>
<td>$R_{\lambda} = \frac{2}{\pi k_g \left( A_1 \frac{A_1 - 2B_1}{A_1 - 2C_1} + 2B_2 - 2C_2 \right)}$</td>
<td>(15)</td>
<td>[22]</td>
<td>Calculation of the thermal resistance of the interstitial gas in the middle region.</td>
</tr>
<tr>
<td>$A_1 = 2r_p + j - \epsilon_0$, $B_2 = \sqrt{r_p - r_c^2}$, $C_2 = \sqrt{r_p - r_c^2}$ and $r_c = \frac{\sqrt{r_p - r_c^2}}{(r_p - 0.5\epsilon_0 - 5\lambda)}$</td>
<td></td>
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</tr>
<tr>
<td>$R_{\text{mid},1,2} = \frac{(d_p - \epsilon_0)}{k_s \pi (r_p^2 - r_c^2)}$</td>
<td>(16)</td>
<td>[22]</td>
<td>Calculation of the thermal resistance for the middle solid region.</td>
</tr>
<tr>
<td>$R_G = \frac{2}{\pi k_g \left( A_2 \frac{A_G}{A_G - 2B_G} - 2B_G \right)}$</td>
<td>(17)</td>
<td>[22]</td>
<td>Calculation of thermal resistance of the interstitial gas in the micro-gap.</td>
</tr>
<tr>
<td>$A_2 = 2r_p - \epsilon_0$, and $B_G = \sqrt{r_p - r_c^2}$</td>
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<tr>
<td>$R_{\text{out},1,2} = \frac{\ln \left( \frac{A_{\text{out}} + B_{\text{out}}}{A_{\text{out}} - B_{\text{out}}} \right)}{k_s \pi B_{\text{out}}}$</td>
<td>(18)</td>
<td>[22]</td>
<td>Calculation of the thermal resistance of the bulk solid material in the outer region.</td>
</tr>
<tr>
<td>$A_{\text{out}} = r_p - 2(0.5\epsilon_0 + 5\lambda)$, $B_{\text{out}} = \sqrt{r_p - r_c^2}$ Assume isothermal temperature boundary.</td>
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The ratio of the molecular mean free path and the corresponding void size is called the Knudsen number ($Kn = \lambda/d$). The thermal conductivity of gas changes when the Knudsen number lies within the transition region. Springer [17] stated that conduction heat transfer in the gas between two isothermal plates is divided into four regions: continuum, temperature jump or slip, transition and free molecular, where the regions are quantified by the Knudsen number. Smoluchowski [14] discovered this effect as early as 1898 and therefore this phenomenon is denoted as the Smoluchowski effect in this study.

For all four aforementioned heat transfer regions one can calculate heat transfer using $q^* = \left( k_g / (d + j) \right)(T_1 - T_2)$, with $j$ defined as the temperature jump distance at the surface under consideration, [10]. The temperature jump distance is given by:

$$j = \left( \frac{1 - \alpha_{r_1}}{\alpha_{r_1}} + \frac{1 - \alpha_{r_2}}{\alpha_{r_2}} \right) \left( \frac{2\gamma_g}{1 + \gamma_g} \right) \frac{\lambda}{Pr}$$

where $\alpha_{r_1}, \alpha_{r_2}, \gamma_g$ and $Pr$ are the thermal accommodation coefficients for surface 1 and 2, the ratio of the gas specific heats ($\gamma_g = C_p/C_v$) and the gas Prandtl number, respectively [13].
The thermal accommodation coefficient \( (a_r) \) represents the degree to which the kinetic energy of a gas molecule is exchanged when it collides with a solid surface. Song and Yovanovich [16] proposed a correlation to estimate \( (a_r) \) for engineering surfaces which is given by:

\[
(a_r) = \exp \left[ -0.57 \left( \frac{T_0 - T_s}{T_0} \right) \left( \frac{M^*}{6.8 + M^*} \right) + \frac{2.4 \mu}{1 + \mu} \left( 1 - \exp \left[ -0.57 \left( \frac{T_0 - T_s}{T_0} \right) \right] \right) \right]
\]

where \( M^* = M_g \) for monoatomic gases and \( M^* = 1.4M_g \) for diatomic/polyatomic gases, with \( T_0 = 273K \), the solid surface temperature, \( T_s \), and the molecular mass ratio between the gas and the solid surface, \( \mu = M_g / M_s \).

Furthermore, Van Antwerpen [22] developed a thermal resistance to address the Smoluchowski effect in the middle region and defined a mean free path radius, \( r_\lambda \) which increases in magnitude with a decrease in gas pressure. The Smoluchowski effect is therefore relevant for \( r_\lambda \leq r \leq r_p \). By assuming a profile of a circle with centre point \( (0, r_p - \omega_b/2) \) one obtains a total distance between two spheres as a function of radial position \( D_{tot}(r) = 2r_p - \omega_b - 2\sqrt{r_p^2 - r^2} \). Van Antwerpen [22] further defined the mean free path radius boundary when \( D_{tot}(r) = 10\lambda \) since at this point the thermal conductivity in the gas starts to decline at a higher rate with decreasing pressure. The mean free path radius is given in Table 2, where \( r_e \leq r_\lambda \leq r_p \).

Combining the different aforementioned parallel and series thermal resistances leads to an overall joint resistance \( (R_j) \). Calculating the effective thermal conductivity vector \( (k_{e}^{g,c}) \) through two spheres using \( (R_j) \) yields:

\[
\frac{R_j}{k_{e}^{g,c}} = \frac{L_j}{R_j A_j} = \frac{(d_p - \omega_b)}{d_p^2 R_j}
\]

where \( L_j = (d_p - \omega_b) \) and \( A_j = d_p^2 \) is the length between the two sphere centres and the joint conduction area for any two half-spheres in contact with each other within in a packed bed respectively. Although the Multi-sphere Unit Cell Model is essentially a cylindrical control volume with cross-sectional area \( (\pi d_p^2) \), the heat transfer must be normalised by the effective square area \( (d_p^2) \) of the control volume into which the cylinder fits because the integration over all the control volumes must add up to the entire cross-sectional area of the packed bed.

In order to obtain only the radial heat transfer and to account for the porous structure in a randomly packed bed, the radial component of the effective conductivity vector with the appropriate local averaged contact angle \( (\bar{\phi}_c) \) needs to be calculated, and multiplied by the average coordination flux number \( N_c = \bar{N}_c / 2 \) to account for the actual number of spheres in contact in a specific region. This leads to:

\[
k_{e}^{g,c} = \frac{\bar{N}_c (d_p - \omega_b)}{2d_p^2 R_j} \sin \bar{\phi}_c
\]

where \( \bar{\phi}_c \) is in degrees and can be obtained from Eq. (8). The coordination number \( \bar{N}_c \) can be obtained from Eq. (7).
2.3 Multi-sphere Unit Cell Model (Radiation)

The effective thermal conductivity due to thermal radiation (denoted as radiation conductivity) is generally defined in packed beds as:

\[ k'_e = 4F_E \sigma d_p \overline{T}^3 \]  \hspace{1cm} (23)

where \( F_E \) is defined as a radiation exchange factor, \( \sigma \) the Stefan-Boltzmann constant, \( d_p \) the sphere diameter and \( \overline{T} \) the average temperature between the two corresponding surfaces. Early attempts to define the radiation exchange factor were done by Kasparek and Vortmeyer [9] who defined the radiation exchange factor as a function \( F_E = f(\varepsilon_1, \varepsilon_r) \). However, Singh and Kaviany [12] demonstrated that the radiative conductivity is strongly influenced by the solid conductivity \( k_s \) and sphere emissivity \( \varepsilon_r \), resulting in the radiation exchange factor to be a function \( F_E^* = f(\Lambda_l, \varepsilon_1, \varepsilon_r) \), where \( \Lambda_l = k_s/4d_p \sigma T^3 \) is a dimensionless solid conductivity which is closely related to the so-called Planck number. The reason for this is that the isothermal surface temperature assumption is no longer valid and large temperature gradients arise on the surfaces of the spheres as the solid conductivity \( (k_s) \) decreases with increasing temperature. This results in a lower surface temperature at the tip of the sphere, implying a decrease in thermal radiative exchange at that point [22].

The single most important limitation of the radiation exchange factor is that the value of \( F_E \) or \( F_E^* \) cannot be calculated easily. For each packing set a new radiation exchange factor should be developed. This limitation is overcome with the newly derived Multi-sphere Unit Cell Model as it treats the thermal radiation \( (k'_e) \) between spheres in contact (short-range radiation) and spheres not in contact (long-range radiation) separately as well as incorporating the dependence of solid conductivity with a non-isothermal correction factor, so that:

\[ k'_e = k'_{e,S} + k'_{e,L} \]  \hspace{1cm} (24)

where \( k'_{e,S} \) is the short-range radiation and \( k'_{e,L} \) the long-range radiation. The dominant area of thermal radiation heat transfer occurs in the region \( r_c \leq r \leq r_p \).

A parameter for short-range thermal radiation is developed by considering standard radiative heat transfer between two gray diffuse parallel plates with \( \varepsilon_r,1 = \varepsilon_r,2 = \varepsilon_r \) and \( A_{j,1} = A_{j,2} = A = 4\pi r_p^2 \) so that:

\[ Q_i = A\varepsilon_r \sigma \left( T_1^4 - T_2^4 \right) \left( \frac{2 - 2\varepsilon_r}{\varepsilon_r} + \frac{1}{F_{i,2}} \right) \]  \hspace{1cm} (25)

The diffuse view factor between two spheres is \( F_{i,2} = 0.0756 \) [22]. The radiative conductivity is considered to be \( k'_{e,S} = Q_i L_r / A_i (T_1 - T_2) \) where \( L_r = d_p \) is a geometrical length characterising the radiative conductivity. The radiative conductivity is also normalised with the same area as that for conduction in Eq. (21), i.e. \( A_i = A = d_p^2 \) and accounted for by the average coordination number and contact angle. For many interface problems where \( \Delta T/\overline{T} \ll 1 \), the approximation \( (T_1^4 - T_2^4)/(T_1 - T_2) \approx 4\overline{T}^3 \) is valid. This leads to the derivation of the short-range parameter shown in Table 3.
The long-range thermal radiation is a complex phenomenon, due to the difficulty in characterising the porous structure. Vortmeyer [25] argued that unit cell models have never taken into account the long-range effects that must exist in packed beds. However, in the Multi-sphere Unit Cell Model particular attention is given to long-range thermal radiation in the form of an average long-range diffuse view factor ($F_{1-2,\text{avg}}^L$).

Now consider the long-range diffuse view factor ($F_{1-2}^L$) which decreases when the geometrical length $L_r$ increases, where $F_{1-2} = 0.0756$ when $L_r = d_p$ or $z=1$ (two touching spheres). It is expected that the function will look like that displayed in Figure 4. Therefore, it can be stated that $\lim_{L_r \to \infty} F_{1-2}^L (L_r) = 0$ where $L_r$ is the distance between the centre of the sphere under consideration and the centre of the long-range sphere.

![Figure 4: Decreasing long-range diffuse view factor in the bulk region based on the surface area of a full sphere](image)

Using the coordinates in the bulk region of a numerically generated packed bed and calculating the view factors via a CFD package a graph like that displayed in Figure 4 was generated [22]. The average long-range view factor is further obtained by fitting a curve through the simulation results and integrating to obtain the area under the curve. Further it was found that $F_{1-2}^L (2.25) = 0$. Therefore, by dividing the area by $(2.25 - 1) = 1.25$ one obtains $F_{1-2,\text{avg}}^L = 0.0199$ and the average geometrical length $L_{r,\text{avg}} = 1.33d_p$. The same methodology was followed as for short-range radiation to develop the long-range radiation, $r_{ek}$ parameter and is given in Table 3. Lastly, it was empirically found by comparing the Multi-sphere Unit Cell Model with experimental results that the number of spheres contributing to the long-range radiation is $n_{\text{long}} = 4.7$ [22].

A non-isothermal correction factor ($f_k$) was developed by Van Antwerpen [22] via CFD simulations using a surface-to-surface thermal radiation model to address the influence of solid conductivity on the radiation conductivity. Results were obtained by simulating the thermal radiation between two hemispheres with a 0.5 mm gap. This eliminated any chance of thermal conduction between the two surfaces. Based on the numerical simulation results, a non-isothermal correction factor was proposed and is given by:

$$f_k = a_1 \tan^{-1} \left( \frac{a_2}{\Lambda_s} \right) + a_3 \quad \text{(26)}$$

where $\Lambda_s = k_s/4d_p \sigma T^3$ and $f_k = 1$ when $1/\Lambda_s < 0.01$. The empirical constants are valid for $0.2 \leq \varepsilon_r \leq 1$ and for $0.01 \leq 1/\Lambda_s \leq 10$ and can be obtained by:


\[ a_1 = 0.0841 \varepsilon_r^2 - 0.307 \varepsilon_r - 0.1737 \]  
\[ a_2 = 0.6094 \varepsilon_r + 0.1401 \]  
\[ a_3 = 0.5738 \varepsilon_r^{-0.2755} \]  
\[ a_4 = 0.0835 \varepsilon_r^2 - 0.0368 \varepsilon_r + 1.0017 \]  

It must be emphasised that the function presented above is only valid for the given dimensionless solid conductivity range and should be re-evaluated if used outside the limits specified for \( \lambda_s \).

It must further be emphasized at this stage that the long-range radiation conductivity presented here is only valid up to a temperature of 1200°C. In the derivation of the Multi-sphere Unit Cell Model certain assumptions were made that affects long-range radiation. The major limiting assumptions are that the average temperature (\( \bar{T} \)) across the effective long-range pebble is the same as the average temperature over two spheres in contact, and that the non-isothermal correction factor \( f_k \) is the same in long-range radiation as for short-range radiation. In reality the average temperature (\( \bar{T} \)) will typically be higher between the long-range sphere and the sphere under consideration than the assumption previously made, resulting in \( f_k \) to decline more sharply at higher temperatures. Improvements can be made to the Multi-sphere Unit Cell Model by deriving a non-isothermal correction factor specifically tailored for long-range radiation using CFD simulations. Therefore, further investigation should be done into long-range radiation to accurately calculate this damping effect at temperatures above 1200°C.

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| \[ k_{\text{eff},S}^L = \frac{2\bar{N}_3 d_p \sigma A_0 \bar{T}^3}{A_1 \left( \frac{2 - 2 \varepsilon_r}{\varepsilon_r} + \frac{1}{F_{1-2}} \right)} f_k \sin \phi \] | (31) | [22] | \( \Delta T / \bar{T} < 1 \), \( 0.01 \leq \lambda_s \leq 10 \), \( 0.2 \leq \varepsilon_r \leq 1 \)  
Calculation of short-range thermal radiation in the bulk and near-wall region. \( F_{1-2} = 0.0756 \), \( A_0 = 4 \pi d_p^2 \), \( A_1 = d_p^2 \) |
| \[ k_{\text{eff},L}^L = \frac{\bar{n}_{\text{long}} 5.32 d_p \sigma A_0 \bar{T}^3}{A_1 \left( \frac{2 - 2 \varepsilon_r}{\varepsilon_r} + \frac{1}{F_{1-2, \text{avg}}} \right)} f_k \] | (32) | [22] | \( \Delta T / \bar{T} < 1 \) for up to 2.25 \( d_p \), \( 0.01 \leq \lambda_s \leq 10 \)  
0.2 \( \leq \varepsilon_r \leq 1 \), \( f_k \) \( \bar{T} \) stays the same for thermal radiation between surfaces further apart than \( d_p \).  
Calculation of long-range thermal radiation in the bulk and near-wall region.  
\( F_{1-2, \text{avg}} = 0.0199 \), \( A_0 = 4 \pi d_p^2 \), \( A_1 = d_p^2 \) and \( \bar{n}_{\text{long}} = 4.7 \) |

The effective thermal conductivity for the bulk and near-wall regions of a randomly packed bed can then be calculated as

\[ k_{\text{eff}} = k_{\text{eff},S}^L + k_{\text{eff}}^L \]  

3. **Comparison between Multi-sphere Unit Cell Model, existing models and SANA-I experimental results**

The Multi-sphere Unit Cell Model in this paper is compared with the results obtained in the
SANA-I experimental test facility constructed and operated in Jülich nuclear research centre, Germany to investigate the effective thermal conductivity for high temperature PBRs [18].

The SANA-I experimental test facility had a central heater element with a diameter of 0.13 m and an outer cylinder diameter of 1.5 m. Graphite pebbles with two different diameters were used namely $d_p = 60\text{mm}$ and $d_p = 30\text{mm}$, in a randomly packed arrangement with a bed height of $L_{\text{bed}} = 1\text{m}$. The bed was heated by a graphite electrical resistance heater for which temperature profiles were generated at three different heights.

For this study, the 10 kW and 35 kW tests with a long heater element and pebble diameter of $d_p = 60\text{mm}$, saturated with helium to atmospheric pressure are considered. Test results were obtained from [18].

The effective thermal conductivity for the SANA-I experimental test facility was previously extracted as a function of temperature only as shown in [11]. This approach, however, is not suitable to observe any effects that the porous structure may have on the effective thermal conductivity in the radial direction. Therefore, it was decided to divide the bed into radial increments and that the following approach would be used to extract the effective thermal conductivities for the two aforementioned steady-states at a pebble bed height of 0.91 m:

$$k_{\text{eff}} = \frac{Q \ln \left( \frac{r_{i+1}}{r_i} \right)}{2\pi L_{\text{bed}} \left( T_i - T_{i+1} \right)}$$

(34)

A comparison between the Multi-sphere Unit Cell Model, other correlations and the SANA-I experimental results obtained from [11] as a function of temperature, is shown in Figure 5. The contact force distribution was calculated by calculating the average contact force with increasing bed height for a bed with similar geometrical dimensions and pebble diameter as the SANA-I experiment [22].

When comparing the effective thermal conductivity simulation results with the experimental data given in Figure 5, it must be remembered that heat is transported not only by thermal conduction and radiation, but also by natural convection in the pebble bed. Natural convection in the SANA-I experimental test facility was not suppressed as the developers of the test facility wanted to determine how natural convection assists in the decay heat removal process. This explains why the majority of the results based on correlations presented in Figure 5 are lower than the measured values [11]. Nonetheless, Figure 5 also indicates that the Multi-sphere Unit Cell Model is in close agreement with the IAEA ZS Total correlation, which is indicated by Niessen and Ball [11] as the most accurate correlation to simulate the effective thermal conductivity in the bulk region of PBRs.

In Figure 6 (a), the effect of natural convection can be seen more clearly with the peak values at $3.5d_p$ and $9.5d_p$ sphere diameters from the inner wall for the 10 kW and 35 kW helium steady-state cases.

Figure 6 (b) shows results obtained by Van Antwerpen [21] for the natural convection process using a systems CFD simulation for another SANA-I test case namely the 4.34 kW nitrogen filled steady-state. Although Figure 6 (a) represents the 10 kW and 35 kW helium cases, it is insightful that the position of the stagnant region in the natural convection case in Figure 6 (b) at $7.6d_p$ corresponds closely with the position in Figure 6 (a) where the experimental results are in agreement with the calculated values from the models. This enforces the presumption that the differences seen between the experimental results and the models for the helium cases are due to natural convection, even though the natural convection in the nitrogen cases will be much more pronounced.
That said, one can see that the Multi-sphere Unit Cell Model predicts with reasonable accuracy the effective thermal conductivity at $7.6d_p$. It is also demonstrated that the IAEA ZS Total correlation fails in the near-wall regions whereas the Multi-sphere Unit Cell Model captures the expected decrease in effective thermal conductivity.

\[
k_g = 0.0557251 + 0.000357143T - 4.87013E-08T^2 \quad [K], \quad (\text{Helium})
\]

\[
d_p = 0.06m \quad F = 72.307 \cdot Z_{\text{depth}} + 7.8716 \quad [N], \quad Z_{\text{depth}} = 0.5m, \quad \epsilon = 0.085, \quad \epsilon_r = 0.8, \quad 
\]

\[
\rho = 0.01, \quad P_g = 9atm = 85kPa
\]

\[
\delta_p, \quad \text{Eq} (26), \quad \rho = 0.01
\]

---

**Figure 5:** Comparison between various correlations and SANA-I experimental data versus temperature saturated with helium [11]

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**Figure 6:** Comparison correlations and experimental results of the SANA-I experimental test facility for the 10 kW and 35 kW steady-state [22]

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### 4. Conclusion

In this paper a new approach to simulate the effective thermal conductivity was developed for randomly packed pebble beds. Special attention was given to the packing structure based on the coordination number and contact angles. The Multi-sphere Unit Cell Model accounts for solid conduction, gas conduction, contact area as well as radiation. An attempt was
made to validate the *Multi-sphere Unit Cell Model* with the results of the SANA-I experimental test facility and good agreement was achieved in the stagnant gas region. It was also shown that the *Multi-sphere Unit Cell Model* captures the expected decrease in effective thermal conductivity in the near-wall regions whereas the IAEA ZS Total correlation fails in this regard.

However, two shortcomings in the experimental data must be highlighted; the natural convection process and the fact that the tests were only done for temperatures up to 1200 °C. This highlights the need for experimental results where the effects of natural convection are minimized, as is the case in the near-vacuum tests in the PBMR HTTU facility [14].

Lastly, it must be added that the *Multi-sphere Unit Cell Model* discussed in this paper has two shortcomings that must be addressed, i.e. the non-isothermal correction factor should be extended to \(1/\Lambda_s > 10\) for simulations done with spheres with lower solid conductivities, as well as the long-range radiation assumptions should be reconsidered for temperatures higher than 1200 °C.

5. Acknowledgements

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6. References


1. BACKGROUND

In the site evaluation study for licensing a new nuclear power facility, the criteria involved could be grouped in health and safety, environment, socioeconomics, engineering and cost-related. These encompass different aspects such as geology, seismology, cooling system requirements, weather conditions, flooding, population, and so on.

The selection of the cooling system is function of different parameters as the gross electrical output, energy consumption, available area for cooling system components, environmental conditions, water consumption, and others.

Moreover, in recent years, extreme environmental conditions have been experienced and stringent water availability limits have affected water use permits. Therefore, modifications or alternatives of current cooling system designs and operation are required as well as analyses of the different possibilities of cooling systems to optimize energy production taking into account water consumption among other important variables.

There are two basic cooling system configurations:

- Once-through or Open-cycle
- Recirculating or Closed-cycle

In a *once-through cooling system (or open-cycle)*, water from an external water sources passes through the steam cycle condenser and is then returned to the source at a higher temperature with some level of contaminants. To minimize the thermal impact to the water source, a cooling tower may be added in a once-through system to allow air cooling of the water (with associated losses on site due to evaporation) prior to returning the water to its source.

This system has a high thermal efficiency, and its operating and capital costs are very low. So, from an economical point of view, the open-cycle is preferred to closed-cycle system, especially if there are no water limitations or environmental restrictions.

In a *recirculating system (or closed-cycle)*, cooling water exits the condenser, goes through a fixed heat sink, and is then returned to the condenser. This configuration results in relatively low water withdrawal. Typical heat sink options for closed-cycle systems are wet cooling system (mechanical or natural draft cooling towers, and cooling ponds).

When water availability is low, a dry cooling system may be utilized. Dry cooling can be either direct or indirect and in each case uses convective heat transfer to provide cooling, eliminating evaporation losses.

### 2. Wet Cooling Towers

Wet cooling towers are commonly used by the power industry. In these towers, water from the condenser is distributed in fine droplets over an internal fill to increase contact time with the air and maximize heat transfer. Circulating air is brought into direct contact with the water to promote heat transfer, primarily by *evaporation*. The cooled water is then returned to the condenser.
Heat transfer in the wet cooling tower occurs when the free energy content available for exchange (or enthalpy) of the water is greater than that of the air. It is this differential in enthalpies which determines the tower’s capacity to remove waste heat. Therefore, the rate of heat transfer is governed by the difference between the temperature of the hot water passing through the tower, and the wet-bulb temperature of the air entering the tower.

There are two types of wet cooling towers: natural and mechanical drafts. The mechanical draft tower uses fans to circulate air through the tower, and can either be forced draft (fan located at the bottom of the tower section) or induced draft (fan located at the top of the tower). These two types are physically very different, although they both use the same heat transfer process.

Wet cooling towers consume considerably huge quantity of water and are thermodynamically less efficient than once-through systems. This is due mostly to evaporation which constitutes about 75% of the heat transfer from the wet tower. Although evaporative towers respond more quickly to changes in the ambient meteorology than do ponds or once-through, these may exhibit poorer performance during extreme environmental conditions.

Usually, wet cooling towers are used as a closed cycle cooling system though they may also be used to supplement once-through cooling systems.

One type of evaporative tower is Hybrid Cooling Tower. This one is a wet cooling tower in which the plume is mixed a dry and hot air stream prior to leaving the cooling tower. This air stream is generated in heat exchangers where the water to be cooled serving as the heating medium. Therefore, no additional energy is required to heat the air.

The quantity of hot air added is such that the plume leaving the cooling tower is undersaturated, and remains undersaturated even when it is mixed with the ambient air. Consequently, it remains invisible.
Hybrid cooling towers are primarily used in plants where disturbing effect on the surroundings must be avoided. They are normally equipped with sound prevention components such as low-noise fans and sound attenuators. Also, they provides for control of environmental factors related to fogging, plume abatement and water conservation usually associated with wet cooling towers.

3. DRY COOLING TOWERS

Dry cooling towers uses air instead of water in order to evacuate the heat at the cold end. The condenser water passes through heat exchanger (such as finned tubes bundles) where heat is transferred to the ambient air by conduction.

Two options also exist for achieving the circulation of ambient air:

- Mechanical Draft: Fans
- Natural Draft

The performance of dry cooling systems is primarily dependent on the ambient dry bulb temperature of the air. And as the ambient dry bulb temperature of the air is higher than the wet bulb temperature, dry cooling are less efficient than wet cooling.

Nowadays, there are basically two main categories of dry cooling systems:

**Indirect Dry Cooling.**- Heat is transferred from the turbine exhaust to the circulating water in the condenser and dispersed to the atmosphere through a fin tube array in a tower, much like the operation of a wet cooling tower.

**Direct Dry Cooling.**- Steam is directly condensed inside finned tubes, which are externally cooled by ambient air. The condensate is re-circulated to the power plant boiler to reuse in the turbine. At no point during the cycle there is any contact between the outside air and the steam or condensate.

Due to circulating water in the dry tower does not come into contact with the atmosphere; there is no evaporative water loss. This is important in terms of overall
water conservation and added flexibility for sitting in arid regions. Also, problems of icing, fogging and visible vapour plumes (of particular concern in urban areas) are eliminated. There is no blowdown associated with dry cooling and thus, no water treatment is required. Except for relatively large land requirements (compared to wet towers), and possible aesthetic considerations, the environmental behaviour of dry towers improves their overall usefulness and also allows more flexibility for power plant sitting which makes possible savings in electrical transmission and fuel transportation costs.

In spite of this one, these ones have not seen much use due to very high capital costs and poor performance during extreme meteorological conditions, resulting in high operating and replacement expenses.

An innovative indirect dry cooling system is the Heller® System. The operation of this cooling system is the following: the cold end of the steam cycle is air-cooled by an intermediate heat transfer circuit filled with condensate quality water and circulated by pumps. Water films formed by the fill of this circuit condense the exhaust steam in the Direct Contact Jet Condenser attached to the steam turbine. Then, a fraction of the warm circuit water (equal to the condensate stream) is pumped forward to the thermal cycle while most of the warm circuit water is pumped via pipeline to a dry cooling tower where it cools down in water-to-air heat exchangers arranged vertically around the tower circumference. Excess pressure head of water leaving the water-to-air heat exchangers is recovered by hydro-turbine installed in common shaft with the pump.

The Heller® System air moving equipment can be either a natural draft or a mechanical draft.

The Heller® System design concepts and equipment provides the maximum possible availability and minimum maintenance. Also, it is totally environmental-friend as saves water equivalent to the consumption of a town of 50,000 inhabitants for each 100 MWe facilitating the licensing of power projects.
THE CHALLENGE OF THE GLOBAL MANAGEMENT OF PLANT DESIGN MODIFICATIONS. EXAMPLE OF THE NEW EJ SYSTEM AT VANDELLOS NPP

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ABSTRACT

One of the most challenging areas in the operation of nuclear power plants (NPP) is related to the management of plant design modifications. Plant modifications can be made to improve reliability, facilitate operation, improve safety or get better results. In any of these situations, plant modifications imply many different activities that have to be done in a coordinated manner.

NUREG-0711 (Human Factors Engineering Program Review Model) shows a global approach to manage most of these activities. Although this approach is mainly focused on the design and construction of new plants, it can also be applied to plant modification management.

Successful global management will require performing every activity in a specific order, taking advantage of the output coming from some tasks as input for others and finalizing every task when necessary. This will provide the best results in terms of quality, time required for implementation, safe and reliable operation and maintenance, and cost.

Tecnatom is involved in most of the activities related to the operational areas and has applied a global approach to get advantages in terms of quality and cost, which is outlined in this paper.

As an example of this approach, the Vandellos NPP experience is shown in this presentation. Vandellos NPP carried out an important design modification that consists of replacing an old essential service water system with a new one. This was a three-year project that implied the construction of new reservoirs, new buildings, the implementation of new equipment, and new panels in the main control room. This paper shows the way in which all of these activities were performed.

1. General Considerations about Plant Modifications Global Management
1.1 Plant Modifications Process

The safe exploitation of a nuclear power plant (NPP) requires all activities to be performed in accordance with the design bases that were established in the initial licence. The licensing process finalizes with the approval of the operation and to reach this point, the regulatory bodies review many different aspects of the NPP.

Engineering design (Mechanical, electrical, I&C and fuel), according to the classic meaning, seems to be considered as the most relevant, but there are others that have to be considered also part of the initial basic design, and have a relevant impact on the safe operation of the plant: Operating and Maintenance Procedures, Human Machine Interface, Technical Specifications, Organization and Staffing, Training and Qualification, etc.
Plant Modifications (PM) imply changes in the design, usually from the engineering point of view, but the other aspects also have to be considered to assure that the safety standards are kept at least at the same level as in the initial design. The PM Global Management has to cover all of these activities in an integrated manner.

Plant Modifications are designed and implemented to meet different goals, for example:
- Safety Improvement
- Reliability Enhancement
- Efficiency Improvement
- To Facilitate Maintenance and Operation
- Equipment Renovation/Modernization
- New Regulations Fulfilment

The number of the activities that are carried out to cover the complete scope of a PM can be large, but for the purpose of this presentation, some of the most common and relevant are pointed out:
- Mechanical Design
- Electrical Design
- I&C Design
- Human Machine Interface (HMI) Design
- Operating & Maintenance Procedures Review & Development
- Training: Development & Review of Training Materials and Delivery of the training
- Operating Experience Review & Implementation
- Verification & Validation of the Design from an Engineering and Operational point of view (mechanical, Electrical and I&C design, HMI design, and Procedures). In most cases of relevant PM, this V&V process requires the use of simulators or Simulation Assisted Engineering (SAE) tools.

NUREG 711 provides a general approach that covers many of these activities. Although this document is oriented toward new plant construction, the approach that is shown can be successfully applied to the PM, while in many cases this approach has to be simplified to be accommodated to the magnitude of the PM. Tecnatom has developed and validated a specific methodology for implementing NUREG 711 in the PM process in a systematic and efficient manner.

Although each NPP has his own procedures, typically, the PM life includes several phases:
- Change proposal (Statement of the problem or reason for the change)
- Assessment of solutions and alternatives
- PM Technical specification development
- Design
- Review of the design
- Purchase and Development
- Implementation
- Operation

### 1.2 Management of Plant Modifications (phases vs activities)

One of the most critical aspects is the assignment of the activities to the phases. It’s clear that in the end all the activities have to be performed by the time the PM is in operation, but the order and the organization and links between the different activities have an important impact on the quality of the results and the cost of the PM.
An analysis of the different activities mentioned above show that there are many relationships and synergies among them. In fact, for example, the HMI design, and especially the task analysis, can provide very useful output to review and develop the operating procedures and the training material. Furthermore, the updating of the training simulator could be used, not only for providing the training, but also for adjusting and validating the operational response of the plant in different operational conditions and performing the integrated HMI validation.

The following Figure shows, in an illustrative way, potential alternatives that will reach the objective in the end, but get different results in terms of cost and quality. As a consequence, it can be stated that a methodology, and the associated procedures, are needed to assure that the best sequence of activities is established.

1.3 Difficulties

In most cases, global management implies a very precise synchronization among different activities. This can be considered as a real challenge. The main difficulties that have to be faced and managed are:

- There are different organizations involved in this process: the Engineering Department, the Operations Department, training, simulation, other subcontracted organizations, suppliers, etc.
- Efficient global management requires many interactions among the above stated organizations in several phases of the process.
- Experience indicates that very often the schedule for designing, validating and implementing plant modifications is very demanding and the engineering path is the most critical, it being difficult to find gaps to connect other activities.
- The availability of design documentation in advance is another critical aspect that implies additional difficulty in some “collateral” activities. For example, to design the HMI or to update the simulator in advance, a lot of design documentation is required in a moment of the project when the design is normally not yet finished.

The most powerful tool that can be used to face these difficulties is planning. All the necessary activities have to be considered from the initial point, taking into account input that
is required and output for each task. All the different players have to be identified, involved from the beginning, being aware of their role, and being prepared to perform what is expected of them on time. Of course, effective leadership is crucial to coordinate and follow up on all activities from a global point of view.

2. Experience in the new EJ system at Vandellos NPP

2.1 Scope of the Plant Modification

In 2009 Vandellos NPP (a 3-loop PWR – Westinghouse reactor) finished the implementation of a very important modification: the EJ replacement. The EJ is the Essential Water Service and the modification consisted of the design and installation of a completely new system. The system is designed to provide cooling capacity for the normal operation during start-up and shut-down and during accidental conditions. EJ is a safety system that has two redundant and independent divisions. The system has a reservoir as a cold source with two cooling towers with a total of 8 fan coils.

This important modification implies relevant changes in the local and main control room panels, equipment and buildings, procedures, and general and specific operation.

2.2 Global activities contracted to Tecnatom

Tecnatom was subcontracted to perform many different activities, most of them related to plant operation and maintenance, for example the review of the Operating Experience to support the designing tasks, the operation analysis for designing the HMI interface for the operators, the development of simulation models to support the design and validation of the engineering design and HMI design, the development of the operating procedures, and the associated training, including the development of the training materials and the delivery of the training.

The synergies between these activities and the experience in the operational field allowed Tecnatom to perform the services in an efficient and effective way.

2.2.1 Operating Experience Review (OER)

The purpose of the Operating Experience Review (OER) of the EJ system was to identify and analyze the problems that had been detected in previous or similar design modifications. This identification and analysis allowed the avoidance of those negative aspects associated with preceding designs, and at the same time incorporate the positive aspects. The OER started with a search of operational experience in national and international databases that keep record of these types of issues, such as:
• Institute of Power Operations (INPO)
• Word Association of Nuclear Operators (WANO)
• Data of Spanish NPPs (DACNE – Datos Centrales Nucleares Españolas)
• Description of Operative Incidents (DIO’S - selected significant events of the industry).

The OER continued with a selection and analysis of the operational experience found, grouping these in several design categories, for distribution among the different design teams:

- Mechanical Design Area
- Electrical Design Area
- Instrumentation and Control Area
- Chemistry Area
- Operational Area
- Maintenance Area

2.2.2 Operational Analysis and HMI Design

The Operational Analysis includes the following NUREG-0711 elements:

- Functional Requirements Analysis and Function Allocation
- Task Analysis

The result of these analyses was data related to control devices, instrumentation and alarms, etc. needed for the Human System Interface (HSI) design, and specifically:

a. Functional Requirements Analysis and Function Allocation

In the Functional Analysis for the design of the EJ System, the functions involved and the associated operating modes and mode changes were defined first. Later, the processes, process elements and associated system components for these functions and operating modes were defined, as well as the functional requirements of the system components, including their support requirements for their operation.

In the Function Allocation analysis of the EJ System, each operating mode and all mode changes defined previously were evaluated to allocate them to man, machine or a combination of both.

b. Task Analysis

In the Task Analysis, the data collected in the previous stages of the system analysis were used to identify and define the different activities that the operation personnel must perform with the system components to execute the different Operating Modes and Mode Changes for each Function.

The Task Analysis includes an evaluation of the execution of those tasks assigned to operation personnel to identify the requirements needed for their fulfillment. Task analysis provided:

- Operation sequences (each operating mode with plant conditions)
- Tasks (the different tasks into which an operating mode is divided)
- Activity (the different individual actions into which a task is subdivided)

For each of the operating modes of each function, the output of this Task Analysis allowed for the identification of:

- Necessary control devices
- Instrumentation requirements
- Alarms
- Communication systems
- Operator aids
• Recommendations for system design modifications

c. HSI Design
With the data obtained from the previous analyses, the HSI design was completed, following also the design guidelines included in the Human Factors Design Guidance included in NUREG-0700.
In order to comply with the HSI consistency guidelines, the final design of the EJ system took into consideration panel design criteria in which the design modification was going to be inserted. In this way the same type of conventional devices were installed in the panel.

2.2.3 Simulation and HMI V&V
a. Simulation Assisted Engineering (SAE)

New models were developed to simulate the new system including hydraulic and logic response. The new models were implemented in the full scope replica simulator and integrated with the rest of the models of the simulator. Additionally, new panels were built to be installed in the simulator.
This process was performed in three different phases:
• The first phase took place 16 months in advance and provided an initial product one year early. This initial product was used to support the engineering design and validation, and to become the first contact for the operators. At that point the operators provided very useful feedback on the design.
• The second phase consisted of the implementation of the final design before the implementation in the plant, some months in advance, and was used mainly for training, HMI final validation and procedure validation.
• The third and last phase consisted of the as-build review of the simulator.

b. HSI Verification and Validation
The HFE Verification and Validation of the EJ system is an activity also included in NUREG-0711 (element 10). The objective of this activity is to check that the design complies with HFE principles, and that it allows plant personnel to perform their assigned tasks successfully, achieving plant safety and operational goals.
Three types of evaluations were performed:
• HSI Task Support Verification
  For checking that HSI functions and components were correctly based on HFE analyses results (defined tasks could be completed with the designed HSI)
• HSI Design Verification
For checking that HFE guidelines of NUREG-0700 were met, considering human capabilities and limitations

- Integrated System Validation
  The safe operability of the design was confirmed, using a full-scope simulator and an almost complete HSI. Related system scenarios in all type of Plant conditions were observed, being observations based on a human performance model.

### 2.2.4 Operating Procedures

Operating procedures is a key aspect of the operation and has a strong relationship to many other activities, like task analysis, engineering design validation, training, etc. During this project the specific operating procedures for normal and abnormal operation were developed for this system, as well as the revision of other specific and global procedures that were affected by this change. It is important to point out that the result of the task analysis (from the HMI design) provided an important support to this activity. Moreover, the final validation was performed in the simulator with the participation of the operators.

### 2.2.5 Maintainability Review

The activities of maintenance personnel in NPPs can be jeopardized if the design of the facilities, systems and support materials does not allow the execution of these activities in an effective and efficient way. For this reason, the specification of the design requirements for facilitating the execution of these activities, and the verification of the correct implementation of these requirements is very important. The specific maintainability requirements of the EJ system were prepared following the criteria of several guidelines, EPRI-NP4350 “Human Engineering Design for Guidelines Maintainability” and the NUREG-0700 “Human Factor Review Guideline”. The Maintainability Verification was performed in the different buildings and associated equipment of the EJ design modification, considering aspects such as:

- Accessibility
- Habitability
- Protection against falls
- Environmental conditions
- Availability of cranes, hoists, etc.
- Fire protection
- Label design
- Standardization and modularization

### 2.2.6 Training

From the beginning, training was considered as a crucial issue. Due to this, the need was established to have specific training material together with the simulator to provide specific theoretical and practical courses.

Different courses were delivered to provide the operators with a deep knowledge of the system and to allow them to understand the response of the new system and the interaction with the other remaining systems under any operating condition. The first courses took place several months in advance, and were more intensively delivered near the start-up of the plant.
3. Conclusions

This important plant modification gave Tecnatom the opportunity to implement a global approach to managing all the activities related to the operation. As a consequence of this, most of the synergies were taken into account, and the services were performed in an efficient and effective manner, providing relevant benefits.

The main advantages of this approach were:

- The design was adjusted to meet the requirements in an efficient way
- The implementation phase was performed in a less risky way, due to the validation in advance
- The HMI design facilitates the safe operation of the plant in any condition
- The operation crews were trained on time

Main difficulties were related to the availability of the design documentation, coordination tasks between the different organizations and tight schedule. The experience of this project is being taken advantage of as lessons learned for other projects.
ABSTRACT

IRIS is an advanced PWR of integral type and 335 MWe power. Its compact containment (spherical, 25 m diameter) enables integrating it inside the reactor building (Fig. 1) to improve safety, security, and economics. Moreover, IRIS has instituted aggressive radiation reduction objectives. One of them is related to decommissioning and dismantling of concrete structures. Specifically, the concrete wall of the reactor vessel cavity in loop PWRs becomes activated. Hence, when the time for decommissioning comes, it must be dismantled/cut with all the necessary precautions, and then disposed of as radioactive waste (even if low-level), at high cost. The IRIS objective is that after 100 years of operation the reactor cavity structure (concrete walls) would remain below the free release level, i.e., would not be treated as radioactive waste. That would significantly simplify returning the site to green field status and reduce the cost of decommissioning.

A coordinated analysis was performed by several groups within the IRIS team; this paper describes the analysis performed by Georgia Institute of Technology using the SCALE6/MAVRIC methodology. The activation of the concrete in the reactor vessel cavity wall is driven by its impurities, primarily cobalt and europium. The neutron flux reaching the walls is significantly attenuated (more than ten orders of magnitude) in IRIS due to the inherent shielding provided by the wide downcomer, but the free release limit based on the IAEA guidelines is relatively low for the considered isotopes, 0.1 Bq/g. As a result, activation in the baseline case (reference design) exceeds the limit over the surface layer of a significant portion of the cavity wall. A design solution was developed in which the concrete wall is borated. This modified design reduced the activation anywhere in the cavity concrete wall to below the clearance limit. Thus, after removing the reactor vessel, decommissioning could proceed as with any other industrial facility. The simplified decommissioning offered by IRIS contributes to its overall sustainability and presents an attractive benefit to the reactor owner/utility.

1. Introduction

IRIS is an advanced PWR of integral type and 335 MWe power [1-3]. Its compact containment (spherical, 25 m diameter) enables integrating it inside the reactor building (Fig. 1) to improve safety, security, and economics. IRIS has instituted aggressive radiation reduction objectives. One of them is related to decommissioning. Usually, the concrete wall of reactor cavity (marked in Fig. 1) becomes activated. At decommissioning, it must be dismantled, which usually means cutting it in pieces with required precautions, and then disposing it off as a low-level radioactive waste, at high cost. The IRIS objective is that after 100 years of operation the reactor cavity structure (concrete walls) would remain below the clearance level, i.e., would not be treated as radioactive waste, and could be freely released. That would significantly simplify returning the site to green field status and reduce the cost of decommissioning, dismantling, and ultimate waste disposal.

A coordinated analysis was performed by several groups within the IRIS team, including ENEA (Italy), Westinghouse (USA), and Georgia Institute of Technology (United States). This paper presents the results of analyses performed by Georgia Institute of Technology. The MAVRIC sequence within the SCALE6 package was employed using the FW-CADIS methodology to efficiently perform this analysis.
2. Methodology and MAVRIC model

The wide IRIS downcomer inherently provides extra shielding and reduces the radiation field outside the vessel by several orders of magnitude compared to a loop PWR. This in-vessel attenuation enables setting up aggressive dose-reduction targets outside the vessel [4,5]. Shielding analysis of this large and complex problem is rather challenging. To increase the reliability of the obtained results, three groups within the IRIS team have performed analyses with different methods, allowing cross-validation of results. Fission source distribution was prepared [6] by F. Franceschini (Westinghouse) using the in-house NRC-licensed reactor physics codes. Extensive analysis was performed by K. Burn and L. Casalini (ENEA) [7] using MCNP [8] with Direct Statistical Approach (DSA) [9,10] for variance reduction. Deterministic analysis using the 3D Sn code TORT [11] was performed initially by M. Sarotto and extended by M. Ciotti (ENEA) to generate flux and dose distribution [12,13]. Monte Carlo analysis using the hybrid FW-CADIS methodology in MAVRIC sequence of SCALE6 package [14-16] was performed at Georgia Institute of Technology focusing on dose distribution and concrete activation [17,18]. This paper focuses on the latter.

Fig. 2 depicts the whole building SCALE6 model, together with the blow-up of the region of interest, lower portion of the reactor vessel, together with the surrounding concrete wall forming the reactor cavity.
The concrete wall has 8 m inner diameter and is 1 m thick. Its height is 12.7 m. The wall is made of reinforced concrete, assumed to be composed of “standard concrete” with rebar. A separate study has shown that the activation of concrete is driven by its impurities, primarily cobalt and europium, through the following activation reactions: $^{59}\text{Co}(n,\gamma)^{60}\text{Co}$, $^{151}\text{Eu}(n,\gamma)^{152}\text{Eu}$, and $^{153}\text{Eu}(n,\gamma)^{154}\text{Eu}$. Concentration of these impurities in concrete was conservatively assumed to be 10 ppm Co and 1 ppm Eu.

The deterministic portion of the hybrid MAVRIC model was characterized by the following parameters: $P_3S_8$ expansion; mesh size 7 cm radially in $x$ and $y$ directions and 17.6 cm axially; ~1.5M meshes; and, problem size 10x10x12.7 m$^3$. Simulation time on a unix workstation was 559 and 327 minutes for the forward and adjoint deterministic calculation, respectively, followed by 44 hours of Monte Carlo simulations with 8 million particles and using the variance reduction parameters automatically generated by the MAVRIC FW-CADIS method, based on the deterministic results. A representative neutron flux distribution together with associated uncertainties is shown in Fig. 3.

3. Results and Analysis

The activation of the concrete in the reactor vessel cavity wall is driven by its impurities, primarily cobalt and europium. The neutron flux reaching the walls is significantly attenuated (more than ten orders of magnitude) in IRIS due to the inherent shielding provided by the wide downcomer, but the free release limit based on the IAEA guidelines is also quite low (0.1 Bq/g) for the relevant isotopes. As the result, activation in the baseline case (reference design) exceeds the limit over the surface layer of a non-trivial portion of the cavity wall. An alternative design was developed in which the concrete wall is borated. This modified design reduced the activation over the whole cavity concrete to below the clearance limit. Thus, after removing the reactor vessel, decommissioning could proceed as with a standard industrial facility. The SCALE/MAVRIC package with its automated variance reduction method of Monte Carlo simulations, enabled effectively analyzing the considered design options. The simplified decommissioning offered by IRIS contributes to its overall sustainability and presents an attractive benefit to the reactor owner/utility.

Activation of the concrete wall forming the reactor vessel cavity is driven primarily by the activation of cobalt and europium in concrete, and cobalt in the rebar steel. Regarding the former, the variability in the concentration of these two impurities in the concrete aggregate and cement is substantial and will directly and proportionally impact the activation. Based on the previous work by other researches, conservative (relatively high) concentrations of 10 and 1 ppm for cobalt and europium, respectively, have been used as the base line values in this study, since there is usually little control on the selection of concrete aggregate. The Operating history of 100 years operation followed by 10 years of cooling time was assumed. Refs. 6 and 16 consistently predict the concrete activation level to exceed the free release limit by a factor of about 2. The areas where this occurs is indicated by color-coded areas in
Fig. 4, without (left) and with an uncertainty factor applied (right). The activation is driven by the $^{151}\text{Eu}(n,\gamma)^{152}\text{Eu}$ reaction.

Fig 1: Europium $^{152}\text{Eu}$ activation distribution in the reactor vessel cavity concrete wall. Colored areas denote activation above the free release limit. (Ref. 16)

Since the activation is not drastically above the free release limit, it was possible to devise design solutions that would enable reducing it below the free release limit. Two main approaches have been identified and examined:

1. Selection of the concrete aggregate and cement with low Co and Eu content. A recent study performed in Japan [20] reported on a systematic analysis of many samples aimed to identify low-activation materials. Several types of aggregates, gypsum and additives were identified that would have about an order of magnitude lower activation than the assumed base-line case (e.g., containing about 1 ppm Co and 0.1 ppm Eu). Such aggregates would reduce the activation below the free release limit everywhere within the cavity well concrete wall.

2. Boration of the concrete to reduce capture of thermal neutrons in Co and Eu. Such boration may be achieved either by: (a) adding boric acid to concrete (up to 0.7% considered acceptable); or, (b) using selected aggregates or cement with inherent boron content.

The effect of boration by boric acid (at concentration of $10^{21}$ at/cm$^3$ of boron in concrete) is shown in Fig. 9. Activation rate with and without boron is compared, for $^{151}\text{Eu}(n,\gamma)^{152}\text{Eu}$ and $^{59}\text{Co}(n,\gamma)^{60}\text{Co}$ reaction, as a function of depth into the cavity wall.

Figure 2: Activation rate, interaction per second per gram of concrete, vs. the depth into the concrete wall, for concrete without and with boron added (at $10^{21}$ atB/ccm).
Adding boron (at this concentration) shifts the maximum rate to the surface (from 10-15 cm inside), and reduces the overall maximum for each of the two reactions by about a factor of ten. This brings the activity concentration from about 2-3 times above to 3-5 times below the free release limit everywhere, providing a sufficient margin for uncertainties.

4. Conclusions
Several design options have been identified to reduce the activation of the cavity concrete wall in IRIS NPP below the free release limit. This will enable simpler and cheaper ultimate dismantling and disposition (compared to a typical loop PWR), and contribute to economic competitiveness of IRIS

5. References
02.06.2010

Wed 08:30 – 10:10

H1

Reactor Technologies
ABSTRACT

The European Lead Fast Reactor has been developed in the frame of the ELSY (European Lead SYstem) project funded by the Sixth Framework Programme of EURATOM, from September 2006 to March 2010. The project, coordinated by Ansaldo Nucleare, involved a wide consortium of European organizations. The ELSY reference design is a 600 MWe pool-type reactor cooled by pure lead. The ELSY project demonstrates the possibility of designing a competitive and safe fast critical reactor using simple engineered technical features, whilst fully complying with the Generation IV goals.

The paper focuses on the main aspects of the proposed design for the European Lead Fast Reactor highlighting the innovation of this reactor concept and overall objectives. Special attention has been dedicated to safety starting from the first step of the design development taking into account other important aspects such as the investment protection, compactness of the primary system as well as sustainability. The main safety features of the proposed innovative Decay Heat Removal systems are presented.

From the beginning of 2010, and for a duration of three years, the EC is financing a new project (LEADER – Lead European Advanced DEmonstration Reactor) as part of the 7th Framework Program. The project is the natural follow up of ELSY: on the basis of the full plant size design, a Demonstration Plant for the LFR technology will be developed with the aim to proceed, in the following phase, to a detailed design and construction of the facility. This paper highlights the main objectives of the LEADER project.
1. Introduction

A pool type configuration has been adopted for the ELSY primary system. This concept allows to contain all the primary coolant within the Reactor Vessel, avoiding problems related to the out-of-vessel circulation of the primary coolant. The primary system main components are the Reactor Vessel (RV), the Steam Generators (SG) and the Primary Pumps (PP).

From the beginning of the project it was decided to pursue a compact and simple primary circuit with the additional requirement that all internal components be removable to assure competitive electric energy generation and long-term investment protection. Simplicity is also expected to reduce both the capital cost and the construction time; these are also supported by the compactness of the reactor building (i.e., reduced footprint and height). The reduced footprint is possible due to the absence of the intermediate cooling system. General seismic behaviour is strongly improved by the embodied technical solutions, in particular the short-height vessel and the 2D anti-seismic supports below the Reactor Building slab. One of the main objectives of ELSY is the identification of innovative solutions to reduce the primary system volume and the complexity of the reactor internals. As a consequence, several components are characterized by strongly innovative solutions. RV, SGs and PPs are Safety Related components since they are part of the primary coolant pressure boundary, and are classified as Seismic Category 1. Primary system components have been designed following the rules of the ASME Code, Section III, Division 1, Class1.

2. Primary system arrangement

The primary system arrangement of ELSY is presented in Figure 1. Table 1 collects the main characteristics of ELSY.

![Figure 1 - Primary System](image)

The pool type design of ELSY reduces the risk of loss-of-coolant accidents and increases the grace time during heat-up accidents due to the large thermal inertia of the primary coolant. In the ELSY design the reactor vessel is in contact with the cold coolant and a minimum of components and structures are in contact with hot lead coolant.

The primary circuit is designed for good natural convection capability, which is achieved by a simple flow path, low core pressure losses (estimated core pressure loss is about 120 kPa at
nominal full power conditions), low primary circuit pressure losses and a large vertical distance between thermal centres of the core and Steam Generators (SGs). Furthermore, the thermal expansion of lead adds to such design features that enhance the natural circulation capability during loss of offsite power transients.

There are eight flat-spiral SGs, each one enclosing coaxially a primary pump. Together each pump and SG forms a SG unit, which is replaceable. The primary coolant is pumped upward by impellers coaxial to a vertical shaft. Then it turns in the radial direction and passes the SG tubes on the shell side. It exits the SG radially to the downcomer through perforated double-walled shells. The coolant continues through the downcomer and at the reactor vessel bottom it turns upward towards the core. Above the core the coolant flows to SG unit entrances, thus completing a full primary circuit flow path. This SG unit design together with the low pressure drop characteristics of the primary system allows the elimination of a pressurized core feed system, greatly simplifying the flow path.

A cylindrical inner vessel physically separates the hot plenum that feeds the primary pumps from the cold plenum outside. Between the primary coolant free levels and reactor roof is a cover gas plenum. The fuel assemblies extend above the fixed reactor cover and they are fixed at their upper end in the cold gas space. Thereby, the fuel assembly heads are directly accessible for handling under full visibility.

There are three DHR systems. The first safety grade system is the isolation condenser system consisting of four independent sub-systems, each one connected to a SG. The isolation condenser system has a nominal heat removal capacity of 30 MWt. Four water loops equipped with coolers immersed in the primary coolant constitute the second safety grade system which handles the DHR function. Its nominal heat removal capacity is 30 MWt. The ultimate DHR for long term heat removal is constituted by tubes fed by air exchanging heat by radiation with the reactor vessel, with a nominal heat removal capacity of 2 MW (RVACS, Reactor Vessel Air Cooling System).

<table>
<thead>
<tr>
<th>Electrical power [MWe]</th>
<th>600</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal efficiency [%]</td>
<td>43</td>
</tr>
<tr>
<td>Primary coolant</td>
<td>Lead</td>
</tr>
<tr>
<td>Primary system</td>
<td>Pool type, compact</td>
</tr>
<tr>
<td>Primary coolant circulation</td>
<td>Forced (Mechanical pumps)</td>
</tr>
<tr>
<td>Primary pumps</td>
<td>Mechanical pumps in hot collector</td>
</tr>
<tr>
<td>Primary coolant circulation for decay heat removal</td>
<td>Natural circulation + pony motors</td>
</tr>
<tr>
<td>Core inlet temperature</td>
<td>400 °C</td>
</tr>
<tr>
<td>Core outlet temperature</td>
<td>480 °C</td>
</tr>
<tr>
<td>Fuel</td>
<td>MOX at first stage; then MOX+minor actinides</td>
</tr>
<tr>
<td>Main vessel</td>
<td>Austenitic SS, hung, short height</td>
</tr>
<tr>
<td>Safety Vessel</td>
<td>Anchored to reactor pit</td>
</tr>
<tr>
<td>Steam generators</td>
<td>Integrated in main vessel</td>
</tr>
<tr>
<td>Secondary cycle</td>
<td>Water-superheated steam (180 bars, 335-450 °C)</td>
</tr>
<tr>
<td>Internals</td>
<td>All removable</td>
</tr>
<tr>
<td>Hot collector</td>
<td>Small volume, with free level above the core</td>
</tr>
<tr>
<td>Cold collector</td>
<td>Annular, free level higher than hot collector</td>
</tr>
<tr>
<td>DHR coolers</td>
<td>Immersed in the cold collector</td>
</tr>
<tr>
<td>Seismic design</td>
<td>2D isolator below reactor building</td>
</tr>
</tbody>
</table>

Table 1. Parameters of the ELSY plant
3. **Core**

The reference core consists of an array of open square fuel assemblies (FAs) each containing 428 fuel pins in square pitch (however a back-up option using hexagonal wrapped fuel elements has also been developed). Each FA contains, at its bottom end, a fuel pin bundle with structural grids similar to the grids of a PWR. The outer ring FAs are surrounded by reflector-assemblies (Dummy), these are closed square structures, containing lead, and shielding the cylindrical inner vessel. An additional structure connected with the inner part of the cylindrical inner vessel is shaped to match the square geometry of the reflector-assemblies. This structure, together with the reflector-assemblies, and the cylindrical inner vessel, provide the lateral support of the core. Fuel pins are arranged in a 21X21 open lattice with a total of 428 pins, the remaining 13 positions being allocated to four steel rods arranged at the corner and to a central square cross sectional tube with mechanical functions. The central tubes can be used also for insertion of control rods.

An alternative option with hexagonal closed fuel assemblies, similar to the one used for sodium fast reactors, is also studied. Neutronic analysis performed for ELSY have demonstrated that it is possible to burn all the generated minor actinides with an equilibrium content of MA, in the core, of about 0.9% of heavy metal.

4. **Decay heat removal systems**

ELSY normally relies on the secondary system (the water-steam system) to remove decay heat. The very high reliability required by safety for systems removing the core decay heat is achieved by redundancy and diversification in order to avoid the complete failure of the system. To meet the previous agreement, the DHR system was constituted by two independent, redundant and diverse systems: the Isolation Condenser System (DHR 2) connected to the Steam Generator and the W-DHR System (DHR 1) immersed in the reactor pool. The Isolation Condenser System is a four loop sub systems with:

- One heat exchanger: a vertical tube bundle with an upper and lower horizontal header.
- One water pool: the amount of water is sufficient to guarantee 3 days of operation.
- One condensate isolation valve and One water storage tank.
- The main steam line of the SG is connected to the upper header of the IC
- The lower header of the Isolation Condenser is connected to the main feed water line of a steam generator. This connection includes a condensate isolation valve and a hot tank maintained at the Feed water inlet temperature).

The Isolation Condenser System is designed to remove the required decay power of 22.5 MWt with three sub systems in service and one out of service; hence the system has a total decay power removal capacity equal to the conservative value of 30 MWt with four sub systems in service. The 30 MWt decay power level corresponds to the decay heat generated about 24 minutes after reactor shutdown and the 22.5 MWt decay power level corresponds to the decay heat generated about 1 hour and 30 minutes after reactor shutdown. The excess of decay heat generated and not removed by the isolation condenser system increases the system temperature without exceeding the temperature limits, owing to the thermal inertia of the system.

The four Water loops constituting the second safety grade system for the decay heat removal function (W-DHR) are equipped with coolers immersed in the primary coolant. The conceptual design of each of four W-DHR systems includes:

- A Cooling water storage tank.
- A water-lead Dip Cooler immersed in the cold collector of the primary system.
- Interconnecting piping.
- Steam vent piping to discharge steam into the atmosphere.

The water-lead Dip Coolers are immersed in the cold collector of the primary system. The W-DHR system is designed in order to evacuate a specified power of about 22.5 MWt with 3 out of 4 loops in operation; this value has been estimated to be sufficient to respect
the temperature limits on reactor vessel and fuel. The designed W-DHR system provides a total heat removal capacity of 30 MWt; three loops are sufficient to remove the required decay heat power hence one W-DHR loop represents the internal redundancy of the DHR1 system.

5. **The LEADER project – toward the Lead Demonstration Reactor**

From September 2006 to March 2010 the ELSY project developed a very innovative pre-conceptual design of an industrial plant for electricity production able to close the fuel cycle. The LEADER project, financed in the frame of EU-FP7, started on April 2010 from the results achieved by ELSY with an in depth analysis of the actual reactor configuration. With reference to the industrial scale Lead cooled Fast Reactor configuration, the design of a low cost and fully representative demonstrator (European LFR Technology Demonstrator Reactor – ETDR) will be performed. The objectives of the project activities for the ETDR are:

- to define the main suitable characteristic and design guidelines for the facility;
- to design a scaled demonstrator fully representative of the industrial size reactor;
- to use components/technologies already available in the short term to be able to proceed in the near future to a detailed design followed by the construction phase;
- to evaluate safety aspects and perform a preliminary safety analysis;
- to minimise the cost of the demonstrator.

The demonstrator shall confirm that the developed and adopted materials, both structural and innovative fuel materials, are able to sustain high fast neutron fluxes and temperatures.

6. **Conclusions**

ELSY and LEADER represent the basis of the EU implementation of the Lead cooled technology for fast reactors within the European Sustainable Nuclear Industrial Initiative (ESNII) of the SET-Plan. While ELSY focused on the definition of the industrial scale reactor, LEADER is dedicated to the development of a reduced scale demonstration reactor that will be one of the main step and milestone on the road to future Generation IV technologies.

**Acknowledgements**
The authors appreciate the effort of the EC Project Officer Ved Bhatnagar as well as of all the scientists and institutions involved and acknowledge the financial support of the EC.

7. **References**

SIMULATION TOOLS AND NEW DEVELOPMENTS OF THE MOLTEN SALT FAST REACTOR

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ABSTRACT

Starting from the Molten Salt Breeder Reactor project of Oak-Ridge, we have performed parametric studies in terms of safety coefficients, reprocessing requirements and breeding capabilities. In the frame of this major re-evaluation of the molten salt reactor (MSR), we have developed a new concept called Molten Salt Fast Reactor or MSFR, based on the Thorium fuel cycle and a fast neutron spectrum. This concept has been selected for further studies by the MSR steering committee of the Generation IV International Forum in 2009. Our reactor's studies of the MSFR concept rely on numerical simulations making use of the MCNP neutron transport code coupled with a code for materials evolution which resolves the Bateman's equations giving the population of each nucleus inside each part of the reactor at each moment. Because of MSR's fundamental characteristics compared to classical solid-fuelled reactors, the classical Bateman equations have to be modified by adding two terms representing the reprocessing capacities and the fertile or fissile alimentation. We have thus coupled neutronic and reprocessing simulation codes in a numerical tool used to calculate the extraction efficiencies of fission products, their location in the whole system (reactor and reprocessing unit) and radioprotection issues.

1. Introduction

The Generation-IV International Forum (GIF) for the development of new nuclear energy systems has established a set of goals as research directions for nuclear systems: enhanced safety and reliability, reduced waste generation, effective use of uranium or thorium ores, resistance to proliferation, improved economic competitiveness. Molten Salt Reactors (MSRs) are one of the systems retained in 2002 by this forum. Starting from the Molten Salt Breeder Reactor project of Oak-Ridge [1], we have performed parametric studies in terms of safety coefficients, reprocessing requirements and breeding capabilities. In the frame of this major re-evaluation of the molten salt reactor (MSR), we have developed an innovative concept based on the Thorium fuel cycle and called Molten Salt Fast Reactor or MSFR with no moderator in the core, leading to a fast neutron spectrum while ensuring excellent safety characteristics [2,3,4]. This reactor is associated to a chemical reprocessing unit located on-site. This MSFR configuration devised, developed and conducted by the Reactor Physics Group of LPSC in Grenoble has been officially selected by the Generation IV International Forum in 2009 for further studies. This concept of Molten Salt Fast Reactor together with its reprocessing unit will be presented in the next section.

Our reactor's studies of the MSFR concept rely on numerical simulations making use of the MCNP neutron transport code [5] coupled with a code for materials evolution which resolves the Bateman's equations giving the population of each nucleus inside each part of the reactor at each moment. Because of MSR's fundamental characteristics compared to classical solid-fuelled reactors, the classical Bateman equations have to be modified by adding two terms representing the reprocessing capacities and the fertile and fissile alimentations. This code, whose calculation scheme will be described in section 3, includes recent developments in order to take into account the fast spectrum of the MSFR and the coupling of the reactor with a salt control and on-site reprocessing unit. Finally some preliminary results of this neutronics-reprocessing coupling will be presented in section 4 in terms of fission products extraction efficiencies and location in the system and radioprotection issues.
2. The concept of Molten Salt Fast Reactor

2.1 Reactor Geometry

The standard MSFR is a 3000 MWth reactor with a total fuel salt volume of 18 m$^3$, operated between 650 and 850°C. As shown in Fig. 1, the core of the MSFR looks like a single cylinder (with diameter equal to the height) where the nuclear reactions occur within the flowing fuel salt. MSFR simulations have been performed using a binary fluoride salt, composed of LiF enriched in $^7$Li to 99.995 % and a heavy nuclei (HN) mixture initially composed of fertile thorium and either fissile, $^{233}$U or Pu. The (HN)F$_4$ proportion is set at 22.5 mole % (eutectic point), corresponding to a melting temperature of 565°C.

Fig. 1: Schematic view of a quarter of the MSFR. The fuel salt (not represented here) would be located within the orange lines

The external core structures and the fuel heat exchangers are protected by thick reflectors made of nickel-based alloys, which have been designed to absorb more than 80% of the escaping neutron flux. These reflectors are themselves surrounded by a 10cm thick layer of B$_4$C, which provides neutronic protection from the remaining neutrons. The radial reflector includes a fertile blanket (50 cm thick - green area in Fig. 1) to increase the breeding ratio. This blanket is filled with a fertile salt of LiF-ThF$_4$ with 22.5%- mole of $^{232}$Th initially.

2.2 Reprocessing Scheme

The on-site salt management of the MSFR combines a salt control unit, an online gaseous extraction system and an offline lanthanide extraction component by pyrochemistry [6]. This salt reprocessing scheme is presented in Fig. 2. The only continuous salt chemistry process is the gaseous extraction system. It consists first in injecting helium bubbles at the lower part of the core to trap the non-soluble fission products (noble metals) dispersed in the flowing liquid and also gaseous fission products. A liquid/gas phase separation is then performed on the salt flowing out of the core to extract gaseous species and dragged condensed particles. Following this “physical” process of purification, a small part of the gas is withdrawn in order to let the fission products decay and the remaining part of gas is sent back to the lower part of the core. An experimental forced flow loop of fluoride salt is under construction to evaluate the efficiency of this bubbling process in a fluoride salt.

The salt properties and composition are monitored through the online chemistry control and adjustment unit. A fraction of salt is periodically withdrawn and reprocessed offline in order to

*“FFFER project” in progress at LPSC Grenoble
extract the lanthanides before it is sent back into the core. In this separate, batch reprocessing unit, 99% of Uranium (including $^{233}$U) and Neptunium and 90% of Plutonium are extracted by fluorination and directly and immediately reintroduced in the core. The remaining actinides are then quickly extracted together with Protactinium and also sent back to the core. In the last step, the lanthanides are separated from the salt through a second reductive extraction and sent to waste disposal.

Fig. 2: Overall scheme of the fuel salt management including the online gaseous extraction (top) and the offline reprocessing unit (bottom) – The yellow boxes surrounded by a red line are enclosed in the reactor vessel

3. Simulation tools

Simulation of reactor evolution

Our numerical simulations rely on the coupling (see Fig. 3) of the MCNP neutron transport code [5] with a home-made materials evolution code REM [7,8]. The probabilistic MCNP code evaluates the neutron flux and the reaction rates in all the parts (called cells) of the simulated system. This requires a precise description of the geometry and the characteristics of all materials involved (temperature, density, elements, isotopes, proportions), together with the interaction cross-sections of each isotope constituting the reactor. These calculations are static, for a given and fixed state of the system. Following the reactor operation all along its life also requires simulating the temporal evolution of the system. The neutronic code thus has to be coupled with an evolution code.

The evolution code REM solves the Bateman equations to compute the evolution of the materials composition isotope by isotope within the cells as a function of the nuclear reactions and decays occurring in the system and of external parameters like reprocessing or fuel adjustment. The extraction by reprocessing of nucleus $i$ out of the core is implemented through specific removal constants $\lambda_{chem}^i$ equivalent to decay constants. The fuel adjustment of such reactors is performed during reactor operation through fertile or fissile alimentation. The classical Bateman equations have thus to be modified by adding two terms to take into account these MSR’s fundamental characteristics compared to classical solid-fuel reactors:

$$\frac{\partial N_i}{\partial t} = \sum_{j \neq i} \lambda_{j \rightarrow i} N_j + X_j <\sigma_j \phi> N_j - \lambda_i N_i - <\sigma_i \phi> N_i - \lambda_{chem}^i N_i + A_i$$  \hspace{1cm} (Eq. 1)
with $A_i$ the fertile or fissile alimentation of nucleus $i$. Our simulations consider several hundreds of nuclei (heavy nuclei, fission products, structural materials...) with their interactions and radioactive decays.

Fig.3. Coupling of the MCNP neutron transport code with the in-house materials evolution

The simulations of reactor evolution take into account the input parameters (power released, criticality level, chemistry...), by continuously adjusting the materials composition and thus the neutron flux of the system, via multiple interactions between the neutronic and the evolution tools. The REM code is indeed a precision-driven code, i.e. it has been designed to determine the reactor evolution while controlling the precision of the results at each step of this evolution. The resolution of the Bateman equations is constrained by several variables to keep the reactor’s simulated physical parameters constant during the evolution. These include the total power (with a one percent or so precision) and the reactivity (with a huge precision of some ten pcm, much smaller than the computational uncertainty of this parameter under MCNP). The numerical integration of the Bateman equations is finally done using a Runge-Kutta method.

**Simulation of the whole system: coupling of neutronics and reprocessing**

The method developed in the previous paragraph calculates each nucleus population only inside the core. In order to calculate the nuclei populations inside the whole process, we partitioned the whole system (reactor, reprocessing unit...) into elementary sub-systems characterized by transfer functions from one sub-system to another. Just like the core extraction, those transfer functions have to characterize the kinetic of the considered operations and the thermodynamic equilibrium. There is thus at each step a competition between nuclear decays and chemical extraction. For instance, let’s consider uranium just before the fluorination. There are three possibilities: nuclear decay, extraction by fluorine, or it could stay in the salt and thus could go to the next step which is the reductive extraction.

To couple the reprocessing and the core evolution, we add a dimension in equation 1. We have to add the location of each nucleus in the system as a new parameter. Consequently, the size of the matrix which was $N$ (the number of nucleus) becomes $N \times x$, where $x$ is the number of elementary operations done in the reprocessing unit. Equation 1 becomes equation 2 where ‘$B$‘ symbolizes the location of nucleus $i$ in the sub-system B and ‘$B\rightarrow C$‘ the transfer from sub-system B to sub-system C:

$$\frac{\partial N_i^B}{\partial t} = \sum_{j \neq i} \left( \lambda_{j\rightarrow i} + X_j <\sigma_{j\rightarrow i} \phi > \right) N_j^B \left\{ + \sum_{C \neq B} \lambda_{Chem}^{C\rightarrow B} N_i^C - \lambda_i N_i^B - <\sigma_i \phi > N_i^B - \sum_{C \neq B} \lambda_{Chem}^{B\rightarrow C} N_i^C \right\}$$

(Eq. 2)
We are thus able to calculate the evolution of matter in each process of the system and to know isotopes concentrations, gamma or neutron flux or the residual heat (fundamental data for radioprotection) everywhere as presented below in section 4.

The main issue in the reprocessing unit simulation is to determine the kinetic of each step of the process and consequently the transfer constants $\lambda_{Chem}^{B\rightarrow C}$. As technological choices have not been fixed yet, only the available thermodynamic data have been used [8].

We assume that the reductive extraction will be done thanks to counterflow exchangers. In such exchanges, the reaction is limited by diffusion issues in the salt. For example, as described in [8], the chemical transfer constant corresponding to transition from the metal to the salt is given by equation 3:

$$\lambda_{salt\rightarrow metal} = \frac{k}{V_{salt} / S} \quad \text{(Eq. 3)}$$

With $S$ the exchange surface, $k$ the transfer coefficient being equal to $10^{-5}$ m.s$^{-1}$ and $V_{salt}$ the concerned salt volume.

4. Preliminary results of the neutronics-reprocessing coupling

Extraction efficiencies

The extraction efficiencies are evaluated from the ratio between the input flow of matter in the reprocessing unit and the output flow sent to storage. The difference between the input and the output corresponds to the matter reinjected in core. For the actinides (also called transuranian elements or TRU in the following), this ratio is rather viewed as reprocessing loss, while it really corresponds to an extraction efficiency for lanthanides.

<table>
<thead>
<tr>
<th>Actinides</th>
<th>Lanthanides</th>
</tr>
</thead>
<tbody>
<tr>
<td>U</td>
<td>$&lt;10^{-7}$</td>
</tr>
<tr>
<td>Pa</td>
<td>$1.5 \times 10^6$</td>
</tr>
<tr>
<td>Np</td>
<td>$1.0 \times 10^{-7}$</td>
</tr>
<tr>
<td>Pu</td>
<td>$3.8 \times 10^{-5}$</td>
</tr>
</tbody>
</table>

Tab 1: Extraction efficiencies

The calculations have been performed with an extraction step for the actinides carried out through two cycles, while it requires around 20 cycles for the lanthanides. The fuel salt volume reprocessed is equal to 40 litres, corresponding to 200l of Bismuth and 400l of chloride.

The results obtained for the lanthanide extraction efficiencies and the actinide reprocessing losses are given in Tab. 1. The actinides losses during the reprocessing are lower compared to what is measured in current processes (PUREX...) due to the pyrochemical processes involved. However the results obtained are based only on thermodynamic considerations, neglecting possible technological limitations. They are thus optimistic.

Residual heat and neutron flux in the reprocessing unit

The coupling of the neutronic and reprocessing simulation tools allows the evaluation of the residual heat and the neutron flux present at each step of the reprocessing unit. The calculations have been performed with a 40 litres volume of fuel salt. The results are shown in Tab. 2.

The neutron flux is largely dominated by the spontaneous fissions of $^{244}$Cm, $^{252}$Cf, $^{246}$Cm, $^{242}$Cm, $^{258}$Cf, $^{248}$Pu, $^{248}$Cm and $^{240}$Pu. A total amount of 182 kW of residual heat has to be extracted from the reprocessing unit.
<table>
<thead>
<tr>
<th>Location in the reprocessing unit</th>
<th>Residual heat (kW)</th>
<th>Neutron flux (neutrons emitted per second)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Extraction</td>
<td>81</td>
<td>7.21 $10^7$</td>
</tr>
<tr>
<td>Fluorination (gaseous phase)</td>
<td>0.9</td>
<td>1.70 $10^6$</td>
</tr>
<tr>
<td>Fuel salt after fluorination</td>
<td>80</td>
<td>7.10 $10^7$</td>
</tr>
<tr>
<td>Bismuth bath (TRU extraction)</td>
<td>4</td>
<td>6.46 $10^7$</td>
</tr>
<tr>
<td>Fuel salt after TRU extraction</td>
<td>9</td>
<td>5.87 $10^6$</td>
</tr>
<tr>
<td>Bismuth bath (Ln extraction)</td>
<td>0.3</td>
<td>5.34 $10^5$</td>
</tr>
<tr>
<td>Fuel salt after Ln extraction</td>
<td>5</td>
<td>5.87 $10^5$</td>
</tr>
<tr>
<td>Chloride bath (hydrolysis)</td>
<td>2</td>
<td>5.86 $10^3$</td>
</tr>
<tr>
<td>Total</td>
<td>182.2</td>
<td>2.11 $10^9$</td>
</tr>
</tbody>
</table>

Tab 2: Residual heat and neutron flux in the reprocessing unit for 40 litres of fuel salt

5. Conclusions

Our simulations of the Molten Salt Fast Reactor concept rely on numerical tools making use of the MCNP neutron transport code coupled with a code for materials evolution which resolves the Bateman’s equations which give the population of each nucleus inside each part of the reactor at each moment.

Because of MSR’s fundamental characteristics compared to classical solid-fuel reactors, the classical Bateman equations have been modified by adding two terms representing the reprocessing capacities and an online alimentation. We have thus coupled neutronic and reprocessing simulation codes in a numerical tool used to calculate the extraction efficiencies of fission products, their location in the whole system (reactor and reprocessing unit) and radioprotection issues.

The very preliminary results presented in the paper, even if based on rough data of the pyrochemical processes involved, illustrate the potential of the neutronic-reprocessing coupling we have developed. We also show that these studies are limited by the uncertainties on the design and knowledge of the chemical reprocessing processes.

6. References

THE INTERNATIONAL PROJECT ON INNOVATIVE NUCLEAR REACTORS AND FUEL CYCLES (INPRO) – STATUS AND TRENDS

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ABSTRACT

The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) was initiated in 2000. As of April 2010, INPRO has 31 members and is implementing activities in five programme areas:

A: Nuclear Energy System Assessments (NESA) using the INPRO Methodology
Assisting Member States in performing Nuclear Energy System Assessments (NESA) using the INPRO methodology, in support of long-term strategic planning and nuclear energy deployment decision making.

B: Global Vision
Developing global and regional nuclear energy scenarios, on the basis of a scientific-technical pathway analysis, that lead to a global vision on sustainable nuclear energy development in the 21st century, and supporting Member States in working towards that vision.

C: Innovations in Nuclear Technology
Fostering collaboration among INPRO Member States on selected innovative nuclear technologies and related R&D that contribute to sustainable nuclear energy.

D: Innovations in Institutional Arrangements
Investigating and fostering collaboration on innovative institutional and legal arrangements for the use of innovative nuclear systems in the 21st century and supporting Member States in developing and implementing such innovative arrangements.

E: INPRO Dialogue Forum
Bringing together technology holders and technology users to discuss, debate and share information on desirable innovations, both technical and institutional, but also national long-term nuclear planning strategies and approaches and, on the highest level, the global nuclear energy system.

The paper presents main INPRO achievements to date, the current status of activities in these five programme areas and recent INPRO publications, in particular in support of nuclear energy system assessments (NESA) using the INPRO methodology.

1. Introduction

Concerns over energy resource availability, climate change, and energy security suggest an important role for nuclear power in supplying energy in the 21st century. Any major future increase in the use of nuclear power will be fostered by innovation in reactor and fuel cycle technologies, along with innovation in institutional arrangements and deployment, while meeting criteria of sustainable development. International cooperation is important to facilitate such innovation: both technical innovation involving research and development (R&D), and institutional innovation.

Dialogue between today’s technology developers and holders, and current and prospective technology users contributes to developing a joint understanding of the challenges and opportunities of nuclear technology to meet future energy needs in a sustainable manner.
The IAEA is in a unique position to provide a global forum for such cooperation on innovative nuclear energy systems and innovative deployment strategies. In response to related Member State requests, the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) was established in 2000 to help ensure that nuclear energy is available to contribute to meeting the energy needs of the 21st century in a sustainable manner.

INPRO provides a forum in which technology holders and users jointly consider innovations that would support the sustainable development of nuclear energy. The project brings together experts and policy makers from industrialized and developing countries to discuss and cooperate on sustainable nuclear energy planning, development and deployment. INPRO also offers Member States support in long term strategic planning and decision making on nuclear energy development and deployment, and enhances awareness of technology innovation options for the future.

INPRO is implemented with contributions from INPRO Members, all relevant Agency programmes, and in synergy with other international initiatives, such as the Generation IV International Forum (GIF) and the Sustainable Nuclear Energy Technology Platform (SNETP). The results of INPRO's activities are made available to all IAEA Member States. Since its establishment, INPRO has received continued strong support from Member States through resolutions of the IAEA General Conference and from world leaders, for example the G8: “The development of innovative nuclear power systems is considered an important element for efficient and safe nuclear energy development. In this respect, we acknowledge the efforts made in the complementary frameworks of the INPRO project and the Generation IV International Forum” (G8 Summit, Global Energy Security, St. Petersburg 2006).

2010 will be another important year for the project, as it marks its 10 year anniversary.

2. Membership

By April 2010, INPRO had 31 members, representing 75 percent of the world’s gross domestic product (GDP) and 65 percent of the world population. The members are: Algeria, Argentina, Armenia, Belarus, Belgium, Brazil, Bulgaria, Canada, Chile, China, the Czech Republic, France, Germany, India, Indonesia, Italy, Japan, the Republic of Korea, Kazakhstan, Morocco, the Netherlands, Pakistan, the Russian Federation, Slovakia, South Africa, Spain, Switzerland, Turkey, Ukraine, the United States of America and the European Commission.

![INPRO Members in 2010](image)

Figure 1: INPRO Members in 2010.
3. Nuclear Energy System Assessments (NESAs) using the INPRO Methodology

INPRO assists Member States in assessing existing or future nuclear energy systems in a holistic way to determine if such systems meet national sustainable development criteria. A Nuclear Energy System Assessment (NESA) using an internationally validated tool, the INPRO methodology, aids Member States in strategic planning and decision making on long term nuclear energy deployment. A NESA evaluates in a holistic manner all nuclear facilities in a given nuclear energy system, from mining through to final end states for all wastes and permanent disposal of high level waste, and all related institutional measures such as legal framework, regulatory bodies, etc.; it considers the complete lifecycle of the facility (‘cradle to grave’), i.e. design, construction, operation and decommissioning; and all assessment areas defined in the INPRO methodology, i.e. economics, infrastructure, waste management, proliferation resistance, physical protection, safety and environment. A NESA is targeted at:

- **Nuclear technology developers**, to identify possible gaps in research and development, and associated actions to fill those gaps;

- **Experienced nuclear technology users**, to assist with strategic planning and decision making concerning the development or expansion of a nuclear energy system;

- **Prospective first time nuclear technology users**, to identify issues that need to be considered for long-range and strategic decision making on planning their future nuclear energy system.

A ‘NESA Support Package’ is offered to IAEA Member States interested in performing a NESA. It includes a wide range of possible assistance including training materials, courses in using the INPRO methodology and assistance missions to Member States for data collection and evaluation. Integrating NESAs with other energy planning tools offered by the IAEA has been documented in a new publication *IAEA Tools and Methodologies for Energy System Planning and Nuclear Energy System Assessment* (2009). A comprehensive manual on applying the INPRO methodology (*Guidance for the Application of an Assessment Methodology for Innovative Nuclear Energy Systems: INPRO Manual, IAEA-TECDOC 1575 Rev.1*) and documentation on lessons learned from past NESA are available as IAEA

4. Global Vision, Scenarios and Pathways to Sustainable Nuclear Development

A long range and global analysis of possible future nuclear energy development scenarios has become an important element in the discussion of future nuclear prospects on a global level. By formulating scenarios and harmonizing visions for long term global nuclear development and deployment, INPRO aids newcomers and ‘mature’ nuclear countries alike to understand the potential of technical innovations and new institutional approaches for developing and building a sustainable nuclear ‘architecture’ in the 21st century, including possible transition scenarios. Activities include defining different options and scenarios for the global and regional development of nuclear energy over the next 50 years, identifying the corresponding type of nuclear fuel cycle, reactor technologies, and fissile materials, and the modalities for spent fuel recycling, storage and disposal. Institutional arrangements for the implementation of nuclear energy systems in developing countries are also addressed, including leasing of nuclear fuel, trade of fissile material and multinational approaches (MNA) for enrichment, reprocessing and disposal.

INPRO Collaborative Projects in this area include: Global Architecture of Innovative Nuclear Systems based on Thermal and Fast Reactors including Closed Fuel Cycles (GAINS); Fuel cycles for Innovative Nuclear Systems through Integration of Technologies (FINITE); Meeting Energy Needs in the Period of Raw Materials Insufficiency during the 21st Century (RMI) and Investigations of the 233U/Th Fuel Cycle (ThFC).

5. Innovations in Nuclear Technology

Ensuring the sustainability of nuclear energy might necessitate technology innovations and new and innovative institutional approaches. Technical innovations are being pursued nationally and through international initiatives such as the Generation IV International Forum and the European Sustainable Nuclear Energy Technology Platform. INPRO contributes to and complements these efforts. An important objective of performing assessments of nuclear energy systems using the INPRO methodology is to identify gaps in technologies and corresponding R&D needs. This area of work fosters collaboration among INPRO members on selected innovative nuclear technologies to bridge such technology gaps. The collaborative projects in this programme area have been selected so that they complement other national and international R&D activities. INPRO Collaborative Projects in this area...
include: Investigation of Technological Challenges related to the Removal of Heat by Liquid Metal and Molten Salt Coolants from Reactor Cores Operating at High Temperatures (COOL); Decay Heat Removal System for Liquid Metal Cooled Reactors (DHR); Advanced Water Cooled Reactors (AWCR) and Performance Assessment of Passive Gaseous Provisions (PGAP). In 2009, a new INPRO publication on Status and Trends of Nuclear Technologies (IAEA-TECDOC-1622) provided an overview on the history, present situation and future perspectives of nuclear fuel cycle technologies.

6. Innovations in Institutional Arrangements

Institutional arrangements are an important part of the nuclear energy system, including bilateral or multilateral agreements, treaties, national nuclear legislation and regimes, safety standards, new regulatory and licensing approaches, and international conventions. Deploying new reactor designs may require innovative approaches to such institutional measures, in particular for transportable, small and medium-sized reactors. This programme area fosters collaboration and supports Member States in developing innovative institutional and legal arrangements for deploying innovative nuclear energy systems in the 21st century. In 2009, a key study in the area of Innovations in Institutional Arrangements addressed legal and institutional issues related to the introduction of Transportable Nuclear Power Plants (TNPP). The main objective was to help developers learn about technologies that could simplify legal and infrastructure issues, and consider technical design scenarios and options of operation and ownership that might introduce additional problems. One INPRO Collaborative Project supports activities in this area, i.e. Implementation Issues for the Use of Nuclear Power in Small Grid Countries (SMALL).

7. INPRO Dialogue Forum on Nuclear Energy Innovations

The objective of the Dialogue Forum is to bring together technology users and technology holders from all interested Member States and facilitate discussion so that technology holders can better understand the needs and concerns of technology users, and users can better understand the possibilities and limitations of technology holders associated with the development and deployment of innovative nuclear energy systems. The Dialogue Forum addresses all concerned stakeholders, including governments, national and international organizations, regulators, vendors, operators and researchers. Open discussions between technology users and holders at an early stage of development of nuclear energy systems facilitates harmonization of practices, establishment of strategic partnerships between technology users and holders and future deployment of innovative nuclear energy systems in the technology user countries. Preceding this new activity was a two-year study, conducted in 2007 and 2008, which identified commonalities in the expectations of developing countries considering the introduction of nuclear power. The results were published in 2009 as Common User Considerations (CUC) by Developing Countries for Future Nuclear Energy System (IAEA Nuclear Energy Series No. NP-T-2.1, STI/PUB/1380 (2009). A first INPRO Dialogue Forum workshop was held in spring 2009, a second is planned for autumn 2010.

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NON-DIMENSIONAL PARAMETERS FOR OSCILLATORY INSTABILITY IN SUPERCRITICAL PARALLEL-CHANNEL SYSTEMS

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ABSTRACT

A study of oscillatory instability in parallel-channel systems at supercritical pressure is conducted with the aim of identifying the relevant non-dimensional parameters that govern the instability boundary. Valid non-dimensional parameters would have far-reaching applications. They would be useful for understanding the instability mode and provide a powerful tool for design engineers, scaling large systems into smaller less expensive systems, or for using different fluids such as CO₂ in place of H₂O. Many authors have proposed non-dimensional parameters for supercritical flow instability. The non-dimensional parameters derived here follow naturally from previous reported work of the first author, and they are different substantially from those proposed by other authors. Clearly, the determination of valid and proven non-dimensional parameters would be of immense value to new reactor designs that use supercritical light water on the primary side.

1. Introduction

Canada’s new Generation IV nuclear reactor would operate in the supercritical pressure range of light water with an inlet temperature greater than 300°C and an outlet temperature in excess of 400°C. At these conditions water behaves much like a single-phase fluid and the many problems associated with two-phase mixtures can be circumvented. The perceived advantage of using supercritical pressure and temperature conditions on the reactor’s primary side is a higher overall plant efficiency. Like two-phase flow, supercritical flow is prone to flow oscillations and flow instabilities are undesirable in any reactor design, as it can lead to fuel over-heating and, possibly, meltdown. A pervading view is that supercritical flow instability is, in fact, similar to two-phase flow instability. While this seems likely, it is too early to draw definite conclusions. The fuel bundles in a nuclear reactor comprise many parallel channels and parallel feeders ganged together by common inlet and outlet headers. Hence, understanding supercritical flow stability in parallel channels would be important for reactor design and licensing. The fuel bundle arrangement in Canada’s Gen-IV design would be horizontal, while the Euratom (Europe & Japan) Gen-IV version would be vertical. Regardless whether the reactor core is horizontal or vertical, any flow oscillations in the parallel channels of a fuel bundle, or between the feeders connecting the inlet and outlet headers, would be undesirable. Hence, this study looks at determining the non-dimensional parameters that govern supercritical flow instability in parallel channels. Dynamic instability has been considered in previous studies by Chatoorgoon [1] and Chatoorgoon, et. al [2].

2. Relevant Works

Zuber [3] did an extensive review of supercritical flow instability and the first in-depth analytical study of the various instability modes. The conclusions formed on the parameters that stabilize or destabilize a supercritical system are, in fact, the same as for a two-phase
The literature on supercritical flow instability in parallel channels is sparse, but progress is being made by many investigators. On the issue of static instability, Ambrosini and Sharabi [4] were the first to report its occurrence. While simulating supercritical flow in an up-flow parallel-channel system with the RELAP code, they identified a static, or excursive, instability. This was followed soon afterwards by Shah [5] and Chatoorgoon, et. al. [6], reporting that static instability was found to occur frequently in supercritical parallel-channel down-flow. Dashkiyev and Rozhalin [7] studied the stability of a system of parallel steam-generating tubes in the presence of various types of non-uniformities. Jain, et al. [8] discussed the scope of supercritical flow instability and its impact on the Generation IV nuclear power plant design. They described the separate experimental setups at the University of Wisconsin at Madison and Argonne National Laboratory (ANL). Ambrosini and Sharabi [4, 9-11] studied vertical up-flow in heated channels at supercritical pressures. Drawing on the analogy of two-phase flow, non-dimensional parameters for the instability were developed. Ortega Gomez, et al. [12] also studied heated channels with uniform heating. This study also proposed non-dimensional parameters for the instability, but these are not exactly the same as those proposed by Ambrosini and Sharabi [4].

Ambrosini and Sharabi [9] presented a further assessment and discussion of the dimensionless parameters originally presented in Ambrosini and Sharabi [4]. Stability maps presented show the quantitative trends of different operating parameters on the system stability. The study by Ambrosini [10] drew similarities between two-phase flow instability and supercritical flow instability, and concluded they were similar.

Shah [5] studied supercritical flow instability of both natural-circulation loops and parallel-channel assemblies using a linear model approach. He found that there can be significant disagreement between the linear and non-linear predictions of the instability boundary in natural-circulation loops. The situation was different, however, for parallel channel assemblies, which showed generally good agreement between the linear and non-linear predictions of the instability boundary. Further study is ongoing to explain these differences.

3. Geometry

The geometry is the same as was used by Ambrosini and Sharabi [4]. It consists of a single vertical channel of circular ID = 8.36 mm, length = 4.2672 m, wall roughness, \( \varepsilon = 2.5 \times 10^{-5} \) m.

![Schematic of Geometry](image)

Fig. 1: Schematic of Geometry

The flow rate in all cases is 0.05 kg/s, \( K_{in} = 20 \) in all cases, while \( K_{out} = 2, 5, 10 \) or 20. Twenty-four (24) cases were simulated using a linear stability code. These cases are defined in Table 1, which also shows the inlet temperature of each case. Three pressures were examined: 25, 30 and 40 MPa. The fluid was H\(_2\)O.
4. Software

The analytical results presented here were obtained using a linear stability program developed at the University of Manitoba (Shah [5]). The steady-state parameters required by the linear code were produced by SPORTS front-end by solving the steady-state 1-D conservation equations.

5. Software

The friction factors used were whatever was already implicit in the SPORTS code. For laminar flow the following familiar equation was used:

\[
f = \frac{64}{Re}
\]

(1)

For turbulent flow, there's a much more elaborate empirical formulation constructed to closely match the Moody diagrams.

6. Friction factor

The 4.2672m heated section was divided into 100 segments. Spatial convergence was confirmed. The initial inlet steady-state flow rate was 0.05 kg/s; however, for the stability simulation the inlet and outlet pressures were held constant, as well as the power, and the flow rate was allowed to vary as per those imposed constraints. Different powers were used until the stability threshold was attained.

7. Development of Non-dimensional Parameters

The work presented here extends the analysis presented by Chatoorgoon [6]. In that paper it was postulated that for horizontal flow (hence, no gravity effect), the instability boundary would occur close to the flow rate corresponding to

\[
\frac{\partial^2 \Delta p_{ch}}{\partial m^2} = 0,
\]

(2)

where \( \Delta p_{ch} \) is the channel frictional pressure drop. For vertical up-flow, the gravity effect can be important and Eq. (2) becomes less valid for diminishing frictional losses. The exception would be, however, when the K factors are significant and frictional effects overwhelm the gravity effect. In this study Eq. (2) is examined for vertical up-flow to assess at what K values the approximation works well and where it does not.

In Chatoorgoon [6], the following equation was derived to describe the instability boundary in horizontal flow:

### Table 1: Water cases (24) modelled

<table>
<thead>
<tr>
<th>( K_{in} )</th>
<th>( K_{out} )</th>
<th>( T_{in} )</th>
<th>( T_{out} )</th>
</tr>
</thead>
<tbody>
<tr>
<td>25 MPa</td>
<td>30 MPa</td>
<td>40 MPa</td>
<td></td>
</tr>
<tr>
<td></td>
<td>25 MPa</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1.5</td>
<td>1.0</td>
<td></td>
<td></td>
</tr>
<tr>
<td>2.0</td>
<td>2.0</td>
<td></td>
<td></td>
</tr>
<tr>
<td>10</td>
<td>10</td>
<td></td>
<td></td>
</tr>
<tr>
<td>20</td>
<td>20</td>
<td></td>
<td></td>
</tr>
<tr>
<td>285.91</td>
<td>340.0</td>
<td>270.76</td>
<td>336.22</td>
</tr>
<tr>
<td></td>
<td></td>
<td>235.06</td>
<td>322.43</td>
</tr>
</tbody>
</table>
\[
\frac{\Delta \rho_{ch}}{\Delta \rho_n} = \frac{h_c^2}{2 \rho_n m_s^2} \left( \frac{\partial^2 \rho_n}{\partial h_n^2} \right)_p + \eta \left( \frac{m_s^* + 1 - \eta}{(m_s^* + 1)^2} \right) \tag{3}
\]

(Eq. (22) in Chatoorgoon [6]). Note, there was a typographical error in the denominator of the last term of Eq. 22 in the sighted reference. The corrected form is given here.) Eq. 3 was derived analytically assuming a point heat source and the condition of Eq. (2). Eq. (3) is developed further in this paper and tested for the range of conditions given in Table 1 for vertical up-flow.

### 7.1 Further Development of Eq (3)

Since power (Q) is usually the variable in an experiment and not the mass flow rate, Eq. (3) can be re-written to express, instead, an explicit dependency on Q. With some algebra, Eq. (3) becomes

\[
\frac{\Delta \rho_{ch}}{\Delta \rho_n} = \frac{(Q / \dot{m})^2}{2} \left[ \frac{\partial^2 (\ln \rho_n)}{\partial h_n^2} - \left( \frac{\partial (\ln \rho_n)}{\partial h_n} \right)^2 \right] - (Q / \dot{m}) \frac{\partial (\ln \rho_n)}{\partial h_n} \tag{4}
\]

Three new variables are introduced.

\[
\xi = \frac{\Delta \rho_{ch}}{\Delta \rho_n} = 1 + \frac{\Delta \rho_c}{\Delta \rho_n} \tag{5}
\]

\[
\theta = -\frac{1}{\rho_n} \left( \frac{\partial \rho_n}{\partial h_n} \right)_p = -\left( \frac{\partial (\ln \rho_n)}{\partial h_n} \right)_p \tag{6}
\]

\[
\dot{Q} = \frac{Q \theta}{m} \tag{7}
\]

Where \( \dot{Q} \) is dimensionless, and is used from here on as the dimensionless power.

Employing Eqs (5) – (7), Eq. (4) becomes,

\[
\xi = \frac{\dot{Q}^2}{2} + \dot{Q} \tag{8}
\]

where \( \gamma = -\left( 1 + \frac{1}{\theta^2} \frac{\partial \theta}{\partial h_n} \right) \tag{9} \)

Note that \( \gamma \) is dimensionless. Eq. (8a) is a quadratic equation that can be solved to yield, taking the more positive root:

\[
\gamma \dot{Q} = \sqrt{2\gamma \xi + 1} - 1 \tag{10}
\]

### 8. Results

Eq. (10) is surprisingly simple and is fluid independent. It represents a unique relationship between the dimensionless parameters \( \gamma \dot{Q} \) and \( \gamma \xi \). The \( \gamma \dot{Q} \) term is related to the ‘power’, while \( \gamma \xi \) is relation to the ratio of the frictional pressure drop \( \frac{\Delta \rho_c}{\Delta \rho_n} \). Both \( \theta \) and \( \gamma \) are functions of channel outlet temperature only. \( \dot{Q} \) has dimensions similar to inverse enthalpy. Fig. 2 plots \( \gamma \) versus \( T_n \) (channel outlet temperature) for pressures of 25, 30 and 40 MPa.
\( \gamma \) can have both negative and positive values, as Fig. 2 shows. However, when \( T_h \) is less than \(~500^\circ\text{C}\), \( \gamma \) is negative.

![Figure 2: \( \gamma \) vs. channel outlet temperature, \( T_h \), for three different pressures](image)

#### 8.1 Determination of \( Q_b \) and \( Q_s \)

The results presented in Chatoorgoon [6] were for horizontal flow, a constant applied power, \( Q \), and the flow rate was varied until the instability boundary was determined. Also, a steady-state flow-rate sweep was performed to determine the minimum of the \( \frac{\partial \Delta p_{ch}}{\partial m} \) versus \( m \) characteristic, and \( m_b \) was the flow rate at the minimum. The corresponding power was \( Q_b \). In that paper the flow rate at the instability boundary, \( m_s \), was found to be close to \( m_b \) in all the cases examined.

In this paper we consider, instead, the situation where the flow is vertical upward, the flow rate is constant at 0.05 kg/s and the power, \( Q \), is varied until the instability boundary is attained. Thus, how to determine \( Q_b \)? \( Q_b \) was determined for a constant flow rate of 0.05 kg/s as follows: two steady-state flow-rate sweeps at two values of \( Q \) were performed. \( m_b \) was determined for each of the two sweeps. Knowing both \( m_b \) and \( Q_b \) for each case, a linear interpolation (or extrapolation) was performed to determine the value of \( Q_b \) corresponding to \( m_b = 0.05 \text{ kg/s} \). The values of \( Q_b \) derived are given in Table 2.

There was one case, however, for which a \( Q_b \) value could not be found because there was no point of inflection in the \( \frac{\partial \Delta p_{ch}}{\partial m} \) versus \( m \) characteristic. That case was for a system pressure of 40 MPa, \( T_{in} = 322.43^\circ\text{C} \), \( k_{in} = 20 \) and \( k_{out} = 2 \). The outlet temperature was 968 \(^\circ\text{C} \) – perhaps too high for nuclear reactor applications. The \( Q_s \) values were found by performing a linear instability analysis. Those values are also given in Table 2.
8.2 Reviewing $\dot{Q}_s$ versus $\dot{Q}_b$

Fig. 3 plots $\dot{Q}_s$ versus $\dot{Q}_b$ for the twenty-three cases that $Q_b$ could be found. It shows that all the linear instability predictions fall between the $\dot{Q}_s = \dot{Q}_b$ and the $\dot{Q}_s = \dot{Q}_b - 0.2$ lines. Two points fall slightly above the $\dot{Q}_s = \dot{Q}_b$ line. They correspond to 25 MPa, $T_{in} = 400 \, ^\circ C$, $K_{in} = 20$ and $K_{out} = 2$ and 30 MPa, $T_{in} = 336.22 \, ^\circ C$, $K_{in} = 20$ and $K_{out} = 2$. The common feature is $K_{out} = 2$. This indicates the low $K_{out}$ cases show a tendency to deviate from the higher $K_{out}$ cases. Even though the outlet temperature, $T_{out}$, is high in these cases, in excess of 700 $^\circ C$, it is possible that the flow instability for the low $K_{out}$ values may be different to the high $K_{out}$ cases. Any case that falls above the $\dot{Q}_s = \dot{Q}_b$ line is actually different, in type of instability, to the cases that fall below, or on, the $\dot{Q}_s = \dot{Q}_b$ line. Studies are continuing to shed further light on the matter. Undoubtedly, gravity becomes important when the $K$ factors are reduced.

<table>
<thead>
<tr>
<th>System Pressure MPa</th>
<th>$T_{in}$ ($^\circ C$)</th>
<th>$K_{in}$ = 20</th>
<th>$\dot{Q}_s$ (kW)</th>
<th>$\dot{Q}_b$ (kW)</th>
<th>$\dot{Q}_s$</th>
<th>$\dot{Q}_b$</th>
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<td>$K_{out}$</td>
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<td>108.3</td>
<td>1.4118</td>
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<td></td>
<td></td>
<td>5 102.11</td>
<td>92.3</td>
<td>1.5141</td>
<td>1.6005</td>
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<td></td>
<td></td>
<td>10 90.74</td>
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<td>1.0976</td>
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<td></td>
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<tr>
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<td></td>
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<tr>
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<td></td>
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<tr>
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<td></td>
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<td>106.2</td>
<td>1.1956</td>
<td>1.2111</td>
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<tr>
<td></td>
<td></td>
<td>20 100</td>
<td>87.3</td>
<td>1.3129</td>
<td>1.3715</td>
<td></td>
</tr>
</tbody>
</table>
Fig. 3: Plotting non-dimensional instability boundary Power versus \( bQ \).
Taking \( \bar{Q} = \bar{Q}_b \) in design would be conservative and safe when the \( K \) factors are high.

8.2 Plotting Eq (10)

To calculate \( \xi = \frac{\Delta \rho_{\text{ch}}}{\Delta \rho_h} \), \( \Delta \rho_{\text{ch}} \) is easily obtained from the system steady-state solution. Calculating \( \Delta \rho_h \) is less straightforward. It is obtained by defining the ‘hot’ side from the point where \( C_p \) is a maximum (the pseudo-critical point) to the channel’s outlet end. Hence, this too is obtained from the system’s steady-state solution.

Eq. (10) is plotted in Fig. 4 as a solid line. Also plotted are the twenty-four (24) instability boundary data obtained with the linear stability code. (No case was omitted.) It is interesting that all the data fall on the theoretical line. This may be a significant finding as the twenty-four cases are for different outlet \( K \)-factors, different inlet temperatures and different system pressures. All three \( K_{\text{out}} = 2 \) cases are also included, including the case mentioned above for which \( aQ_b \) could not be found. That one also fell plumb on the line described by Eq. (10). A logical explanation for this good agreement for low \( K_{\text{out}} \) values is not immediately obvious – it surprised us. (It could be just fortuitous.) Further study will continue to better understand the low \( K \)-factor situations.

It is surprising that a simple analysis developed for horizontal flow would apply this well for vertical up-flow. The relatively high \( K \) factors may be the reason, as low \( K \) factors would accentuate the gravity effect, thereby altering the stability characteristics. Clearly, further study is needed.

Eq. (10) will be tested for different fluids and reported another time. Eq. (10) also clearly shows the important non-dimensional parameters for the instability boundary. They are \( \gamma \bar{Q} \) and \( \gamma \xi \). The first term is a measure of the applied power, while the second term is a measure of the pressure drop in the ‘cold’ and ‘hot’ segments of the channel.

![Fig. 4: Plotting the 24 cases studied in the non-dimensional parameters proposed in Eq. 10](image-url)
Fig. (5) plots the twenty-four instability data in the non-dimensional parameters proposed by Ambrosini [11]. The effect of pressure is well correlated, but the effects of inlet temperature or $K$-factors are not well correlated. Ambrosini's non-dimensional parameters do not have any explicit dependence on the frictional pressure drop, which is usually important for determination of the instability boundary.

![Fig. 5: Plotting the 24 cases in Ambrosini non-dimensional parameters](image)

9. **Cautionary Note**

We do not mean to imply that all forms of oscillatory instability encountered in parallel channels will be covered by the theory presented here. It is unlikely to be that simple. For systems with reasonably significant $K$-factor losses compared to gravity losses, the parameters presented here are worth considering. Studies are ongoing to better define the situations when this theory does not apply. When gravity effects are dominant, it is expected that another non-dimensional term would be introduced.

10. **Conclusions**

Non-dimensional parameters for oscillatory instability in supercritical parallel-channel flow have been derived for situations when the $K$-factors are reasonably high. A theoretical equation was also derived. Twenty-four cases were examined for different $K$ factors, different inlet temperatures and three different pressures. All the results followed well the theoretical curve. The important non-dimensional parameters are: $\frac{\Delta \rho_c}{\Delta \rho_l}$, $\bar{Q}$, and $\gamma$.

Ambrosini non-dimensional parameters correlated well the effect of system pressure, but not the inlet temperature or $K$-factors. The effect of different fluids has not been studied here; hence, no conclusion can be drawn about that at this stage.

**NOMENCLATURE**

- $f$ – friction factor
- $h_{ex}$ - flow enthalpy at channel exit
- $K_{in}$ – channel inlet restriction loss coefficient
- $K_{out}$ – channel outlet restriction loss coefficient
- $\dot{m}$ – mass flow rate (kg/s)
Q – applied power (kW)
Qs – power at the instability boundary (kW)
\( \hat{Q} \) - normalized power, defined by Eq. (7)
\( \rho_A \) - flow density at channel exit (kg/m³)
TIN – channel inlet temperature (°C)
\( \Delta p_c \) – frictional pressure drop from channel inlet to maximum \( C_p \) point.
\( \Delta p_h \) – frictional pressure drop from maximum \( C_p \) point to channel outlet.
\( \Delta p_{ch} \) – overall channel frictional pressure drop
NSPC – Ambrosini’s non-dimensional subcooling number defined in Ref 4 & 11.
NTPC – Ambrosini’s non-dimensional sub-pseudo-critical number defined in Ref 4 & 11.

REFERENCES
RAPHAEL – A V/HTR SUCCESS STORY

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ABSTRACT

The FP6 RAPHAEL Integrated Project on V/HTR technology concluded in April 2010 after 5 years of successful performance. 35 partners from 10 Member States, an overall budget above 18 MEUR and about 170 key deliverables are some important figures of the project.

RAPHAEL provides results in seven V/HTR technology areas: core physics, fuel, fuel cycle back end, materials, components, safety and system integration covering the major systems and components of a V/HTR. Major highlights include design, fabrication and testing of innovative helium components, improved fuel fabrication and fuel and materials irradiations, and safety testing and PIE of irradiated fuel. In the area of coupled reactor physics and core thermo fluid dynamics, benchmarks have been performed on core safety experiments on the AVR and HTR10 high temperature test reactors, and on the HFR EU1bis fuel burn-up experiment. The fuel cycle back-end activities cover characterisation of V/HTR-specific waste, disposal behaviour and conditioning & spent fuel performance modelling. The materials activities comprise vessel and high-temperature materials, the latter work in collaboration with EXTREMAT, and graphite irradiation and characterisation. Safety and licensing assessments of a V/HTR, and the system integration aspects with respect to plant reference data and R&D results complete the comprehensive scope of RAPHAEL.

Selected results will be made available as Euratom input for exchange within the GIF VHTR projects in negotiated procedures.

Two advisory groups (safety-SAG and industrial users-IUAG) accompanied the project and provided valuable input regarding adjustment of concept specifications. The recommendations of the Industrial Users Advisory Group, including major end-users, are used as input to EUROPAIRS, an FP7 support action aiming at integrating end-users into the R&D process towards a demonstrator for cogeneration. To address the key issue of knowledge transfer, RAPHAEL conducted three Eurocourses, with support of the IAEA, to transmit V/HTR physics and technology to young engineers and students. Furthermore, RAPHAEL was regularly present in conferences and has issued numerous technical publications.

RAPHAEL executed intensive international collaboration mainly in the areas of materials and fuel, in particular with Korea, and in safety. In addition, its representation and contribution was often requested in collaboration initiatives of Euratom with Russia and China, and in workshops organized by IAEA.

1. Introduction

The FP6 RAPHAEL Integrated Project on V/HTR technology implemented in April 2005 terminated in April 2010 after 5 years of successful performance. 35 partners from 10 Member States, an overall budget above 18 MEUR and about 170 key deliverables are some important figures of the project.

The main objectives of RAPHAEL are to consolidate generic HTR technologies and to explore advanced solutions to improve HTR performances (VHTR objective) as well as to assess the potential of HTR/VHTR for extending the scope of use of nuclear power to different industrial heat applications (such as hydrogen production, petrochemistry and steel manufacturing) or cogeneration of heat and power besides mere electricity generation.
RAPHAEL was implemented to explore the performance of fuel, materials and components in challenging conditions, qualifying reactor physics models, addressing nuclear safety and spent fuel disposal issues to assess the viability of a V/HTR. This shall ultimately enable the development of a V/HTR prototype possibly within the next 15 years.

Seven technical sub-projects have been implemented to realise the objectives of the project managed and coordinated by the sub-project Coordination/Project Management which in addition provided reporting to the Executive Board, the Steering Committee and the Commission’s offices including the presentation of results according to the Plan for Using and Dissemination of knowledge. Furthermore two advisory groups, one on safety and another one including industrial users of a V/HTR, and extensive international collaboration have been implemented to guide the project and to include it into the worldwide engagement of HTR development.

2. Technical Achievements [1]

The RAPHAEL Project included seven technical sub-projects: Coupled Reactor Physics and Core Thermo-Fluid Dynamics, Fuel Technology, Back End of the Fuel Cycle, Materials Development, Component Development, Safety and System Integration. Major achievements have been obtained in recovering, mastering, developing and qualifying essential HTR technologies: fuel and some aspects of fuel cycle technologies (including waste management), materials and components, safety and computer tools required for designing and licensing HTR, (reactor physics, system transient analysis, fuel performance and graphite oxidation). New paths have been explored for increasing operating temperature, fuel burn-up and economic competitiveness (VHTR prospects).

Unfortunately the sub-projects Fuel and Materials suffered from the unscheduled outage of the HFR at Petten. Irradiations could not be completed as scheduled and consequently the targeted irradiation levels not fully reached and the PIEs could not be fully performed. With data from the Melt-Wire Experiment HTA-8 in 1986, mean temperatures of AVR fuel pebbles together with their standard deviations were determined, contradicting speculations about uncertainty of AVR fuel temperatures and providing the necessary information for fission product release predictions.

**Core Physics**

A significant effort has been made in terms of qualification of HTR modelling tools, in particular in the areas reactor physics, system transient analysis, and fuel performance. For this purpose, joint code-to-code benchmarks as well as code-to-experiment benchmarks are organised. The physics data on AVR data, HTTR and HTR-10 first criticality, PROTEUS critical mock-up, and Fuel performance data with available fuel irradiation have been used for the benchmarks. New data have been generated in Reactor physics on isotopic analysis of a pebble irradiated in HFR to 11% FIMA.

**Fuel**

Laboratory scale fuel kernel fabrication by external gelation and coating of surrogate kernel coating allowed recovery of TRISO particle fabrication technology. Moreover, innovative routes are explored:

- Fabrication of actinide-loaded kernels by impregnation on inert matrices is tested at the JRC Institute for Trans-Uranium elements (ITU)
- A manufacturing process has been developed to substitute the silicon carbide coating of standard TRISO fuel by zirconium carbide.
- Manufacturing of uranium oxy-carbide kernels (UCO) has been addressed. Replacing UO2 by UCO reduces CO formation from UO2 interaction with carbonaceous coating layers, reducing the pressure build-up in the particle under irradiation, which gives margins for higher fuel burn-up or higher operating temperature.

Due to the key role played in the HTR safety demonstration by the integrity of the first barrier, due also to the high cost of the fuel quality control methods in past fabrications, the development of modern effective and cheaper characterisation methods is fundamental. This
comprises particle distribution in the fuel element, geometry of the particle and of its constituents, physical and micro-structural properties of the coating layers and detection of defects in these layers. The first test, HFR-EU1bis, at very high temperature (core outlet helium temperature 1000°C), has been run by JRC-IE in 2004-2005 and the second one, HFR-EU1, at a lower temperature (hot helium at 850°C), but with higher burn-up, was completed in February 2010. These tests, complemented by safety tests at JRC-ITU, have shown until now that there are significant margins for operating HTR fuel beyond the conditions for which German fuel has been licensed.

An important effort has been started to improve the physical quality of coating layer models and their relevance for newly fabricated fuel, with the “analytical” irradiation programme, PYCASSO, meant at measuring the evolution of coating material mechanical and thermal properties under irradiation. Two irradiation campaigns focusing on the pyrocarbon properties have been performed in RAPHAEL.

**Waste management**

Different strategies are possible for HTR irradiated fuel management, from direct disposal of blocks or pebbles to complete separation of fuel kernels for reprocessing and recycling. For these strategies to address comprehensively HTR waste management options, they must be completed by a strategy for irradiated graphite (the largest part of HTR long-lived waste volume) management that also includes options for disposal and recycling.

Different areas have to be addressed for assessing the feasibility of each of these routes:

- Behaviour of irradiated graphite and of coated particle materials in disposal conditions,
- Conditioning technologies (for blocks, compacts or pebbles, coated particles and graphite),
- Compact extraction from fuel blocks,
- Graphite matrix / particle/ kernel separation and assessment of fuel recovery fraction,
- Graphite decontamination,
- Graphite reprocessing and recycling.

The European programme already addressed critical aspects of these items.

A study of the behaviour of irradiated HTR fuel in the final repository environment has been initiated: the radio-chemical interactions of each fuel barrier with different possible geo-chemical environments are tested separately in long term experiments. With the input of test results, mathematical models are developed and integrated into a computer code. With this code, a very preliminary estimation of an irradiated TRISO particle lifetime is in excess of 10000 years. Even for direct disposal, separating coated particles from their graphite matrix has 2 advantages:

- Significant volume reduction,
- Better conditioning for delaying interactions of the fuel with aqueous phases present in the geological repository

Different techniques are assessed for separation. Pulsed current disintegration is one of the most promising. Once coated particles are liberated from their matrix, they are either conditioned for disposal or the kernel is extracted and reprocessed, if fully separated from carbonaceous wastes. Different matrices have been assessed for conditioning coated particles. SiC appears to be the best candidate. The feasibility of the conditioning processes has also been addressed, as well as the mechanical stability of SiC compacts with embedded particles. Owing to their thermo-dynamic stability, oxide matrices should also be promising candidates. Decontamination is an important element for separate irradiated graphite management. A method for graphite decontamination has been tested in FZJ. This method can be used not only for separating impurities, but also for most of 14C, originated from nitrogen, trapped in porosities close to the surface of graphite blocks.

**Materials**

Investigations on vessel materials (tests on pre-irradiated weld specimens, large scale experiments on welded tubes & plate), high temperature materials (composites for control rod, metals for IHX & Internals) and graphites (irradiation tests at 750°C & 950°C, oxidation...
tests and micro-structural modelling) have been performed. To be able to reach higher core outlet temperatures for VHTR application, it is desirable to have a vessel material withstanding higher operating temperature than 16MND5 steel presently used for the PWR (maximum ~ 350°C). Existing data on 16MND5 mechanical properties at temperature above 500°C should only be revisited for validation of the relevant code case justifying the vessel accidental behaviour. For higher normal operating temperatures than possible with 16MND5, modified 9Cr1Mo steel, which already benefits from industrial experience, has been identified and evaluated in France and European HTR projects as the most appropriate candidate for VHTR application. Thick welding feasibility has been demonstrated by an irradiation test in the HFR showing base metal and welding mechanical properties (tensile, fracture toughness, impact strength and creep tests performed at 550 and 450°C) are not significantly affected by the low dose received in a 60 years vessel lifetime. Today the large rings of a HTR/VHTR vessel cannot be forged and large scale shell tests (including a circumferential welding) have been performed in the RAPHAEL project under simulated loading (internal pressure (15 MPa) + temperature (625°C)) in the CEA AIRBUS facility to check weldment behaviour. The IHX heat transfer zone and hot collector are structures experiencing the highest temperature in the reactor (the core outlet temperature). Among materials with some existing industrial experience, two nickel base alloys, in 617 and H 230, have been considered as the most appropriate for such conditions. For higher operating temperatures, ceramics or Oxide Dispersed Steels (ODS) are promising solutions for the IHX. Such materials can only be considered longer term developments due to the lack of industrial experience feedback and the risks of using such materials is still high. Properties of present commercial graphite grades have been screened in FP5 and FP6 for selecting the most appropriate ones for HTR, through characterisation, irradiation and corrosion tests. The RAPHAEL irradiation tests currently provide the only available high dose information on currently available graphites (up to 21 dpa) and are expected to provide the basis for design and graphite selection for the next HTR Demonstrator. In the frame of collaboration between the RAPHAEL project and a non-Euratom FP6 project, EXTREMAT, selection and an irradiation of different carbon-carbon composites has been performed.

**Components**

Efforts have been focused on key components of a reactor for cogeneration of electricity and process heat: blowers and the IHX.

The IHX experiences very high stresses due to significant thermal gradients and transients. that reduce its lifetime. Plate and tube IHX designs have been investigated. The advantages of plate designs are lower cost and higher compactness for a given power. Modularity (small HX blocks connected together) is also easier to achieve. The actual plate IHX heat transfer performance, which determines the real benefit of such design in terms of compactness, has been checked through various tests. In particular within RAPHAEL project, a representative plate IHX mock-up has been tested in the HE-FUS3 helium loop. The main issue with plate designs is that stresses are generally higher at the plate bundle and collector junction. The lifetime is reduced accordingly. To estimate this effect, numerical studies have been performed on one specific concept, helping designers to understand phenomena involved in premature wear of high temperature heat exchangers, and paving the way for efficient and realistic numerical methodologies.

The concept of a powerful blower design has been investigated including compressor, helium joint, electric motor and the bearings and catcher bearings. The compressor has been designed for maximum efficiency. The motor cooling has been studied to provide a compact solution. An innovative hybrid magnetic bearing that preserves catcher bearings and increases stability has been simulated and tested.
Complementary experiments on tribology of graphite/graphite and graphite/metal pairs representative of reactor components (control rods…) have been performed to identify best candidates regarding wear and give some recommendations.

**Safety**

A safety approach has been defined jointly by the partners of the European HTR projects, in order to take benefit from the inherent safety features of the modular HTR. This allows for the simplification of the reactor design, while still applying internationally accepted standards, principles and methodologies (e.g. IAEA Safety Guides and EUR requirements) and providing a convincing demonstration of the safe behaviour of the system in any condition, including the demonstration of practical exclusion of severe core damage.

Dust appears to play a key role on the distribution of fission products and on their release outside the reactor. For the pebble bed reactors where a large quantity of dust is generated and covers most structural surfaces in the primary circuit, fission products deposited on these surfaces will be associated with this dust and can therefore be remobilised with it. On the other hand for block type reactors, where the dust production is much smaller, a significant fraction of the fission products will interact directly with metallic surfaces and will most likely become permanently embedded. Verification and validation of source term, release and atmospheric dispersion analyses considering the uncertainties on circuit contamination and mobilisation of radioactivity have been performed material qualification in the light of the high integrity safety targets for the primary circuit and Confinement / containment building concepts considering the probability of leaks and breaks in the high pressure helium circuit towards the required protection against external events.

**System Integration**

Although structured in generic technology areas the RAPHAEL project has been set up with a view to ensuring consistency among the sub-projects and with respect to the industrial partner’s V/HTR designs. In addition to that background mission the system integration sub-project conducted specific studies adressing issues related to the extension of the HTR scope to very high temperature for process heat applications. Thus practically four leading activities have been managed:

- Providing reference data and boundary conditions to sub-projects on the basis of disclosed information from industrial partner's designs. In that regard the ANTARES design promoted by AREVA NP has been the main support.
- Assessing the coupling of a V/HTR plant with a hydrogen production plant. Two case studies have been carried out: the coupling with a thermochemical process (Sulfur-Iodine based with input from HYTHEC FP6 project) from the system analysis point of view and the coupling with a High Temperature Electrolysis process with focus on process efficiency, economics and technology readyness.
- Assessing the issues related to core design for VHTR application. Scoping studies have been conducted to identify fuel blocks or pebbles design adaptations which could allow to cope with higher core outlet operating temperatures.
- Integrating sub-projects R&D results and identifying additional needs. This core activity has been practically continuously performed at Project Management level.

3. **Advisory Groups**

According to the objectives of RAPHAEL two advisory groups were established to monitor the evolution of the project and to provide guidelines for further work.

The Industrial Users Advisory Group (IUAG) included representatives of utilities, heat users, vendors of chemical plants, nuclear engineering companies and vendors of nuclear power plants. The major activity of the IUAG was to investigate how to enter into and to foster the dialogue between HTR promoters and end-users. The IUAG concluded that active involvement of end-users in HTR projects is the best way to raise interest and pave the way for mutual understanding of the benefits and the issues involved. The first step in realising the conclusions of the IUAG appears in the EUROPAIRS joint project between end-users and HTR promoters investigating cogeneration applications.
The RAPHAEL Safety Advisory Group (SAG) was composed of representatives from safety authorities, expert organisations and IAEA. Its major activities included the information exchange on licensing issues in other countries promoting V/HTR technology and to discuss the General Safety Approach for Modular HTR (Part 1) drafted by the Sub-project Safety. With respect to the coupling with process plants, the SAG recommended to investigate plant interactions and identifying the initiating events generated by such combinations. It pointed out the difficulties that can be met due to different legal and regulatory environments for these combinations.

4. **Education and Training**

In order to meet the objectives of education and training, the RAPHAEL project held three Eurocourses covering the whole spectrum of RAPHAEL and of HTR technology, the last two in collaboration with the PUMA project and supported by IAEA. Through this collaboration the range of experts could be broadened as well as the number and origin of the attending students and young engineers.

- 1st Eurocourse, hosted by IKE Stuttgart, Germany, on core physics and thermal hydraulics
- 2nd Eurocourse hosted by NRG, Petten, The Netherlands, on coated particle fuel
- 3rd Eurocourse hosted by CEA, Aix-en-Provence, France, on safety, materials and components.

In addition, universities and research centers, partners of RAPHAEL, provided opportunities for students and young researchers to take diploma and the results of which also benefitted RAPHAEL.

5. **Public Relations**

Besides the periodical RAPHAEL Journal, the public website and the various presentations at conferences RAPHAEL was present at a number of conferences stands, e.g. at the HTR-2008 conference in Washington, DC, and exhibited posters at European events and abroad to demonstrate the leading role of Europe in V/HTR development.

A major public relation event was the ‘European City of Science’ organised by the French Presidency of the EU in Paris in November 2008 where RAPHAEL was present with a large stand in the Energy Pavillon. This has also been the occasion to present a large HTR mock-up and to produce a RAPHAEL film.

6. **International Collaboration**

RAPHAEL was active in various areas of international collaboration. Technically the collaboration in the Pycasso experiments with Korea and Japan has to be highlighted as well as in materials with EXTREMAT and in waste disposal with PUMA.

RAPHAEL contributed to the IAEA Meeting of the Technical Working Group on Gas-Cooled Reactors and to the Workshop-2009 regarding EU-China Cooperation in Nuclear Fission Research. Various information exchanges took place with US and South African institutions interested in HTR technology.

The RAPHAEL deliverables will be made available for the Euratom contribution on V/HTR for a balanced exchange within GIF. Representatives of RAPHAEL partners are active in GIF committees to monitor and guide the document exchange for the benefit of GIF and the European V/HTR community.

7. **Conclusions**
The RAPHAEL Integrated Project terminated in April 2010 after five years of successful performance. Major important results have been obtained in the areas of core physics, fuel, waste treatment and disposal, materials, components, safety and system integration raising the worldwide interest of the V/HTR community. Europe confirmed its strength in V/HTR technology and development. GIF members are interested to receive access to the results in order to complete their database. The success of the project was supported through international collaboration, the support and information exchange with IAEA and dedicated countries. Advisory groups including potential industrial users of V/HTR and on safety provided important and beneficial advice to the project. Throughout its five year term RAPHAEL disseminated V/HTR technology results and objectives to young researchers and engineers through its Eurocourses, to experts at conferences and seminars and to the public at scientific events, the public website and its periodically published RAPHAEL Journal. RAPHAEL technology will benefit FP7 Support Action EUROPAIRS involving both end-users and V/HTR promoters and (if approved) in the newly proposed HTR project ARCHER of FP7. Its realisation may be expected in the subsequent demonstrator for cogeneration.

8. References
02.06.2010

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Civil society energy policy
ABSTRACT

This paper describes the outcomes from the joint report between the Nuclear Energy Agency and the International Energy Agency of the OECD on the projected costs of generating electricity. It is the seventh in the series and presents the latest data available on electricity generating costs for a wide variety of fuels and technologies, including coal, gas, nuclear, hydro, onshore and offshore wind, biomass, solar, wave and tidal. The study contains data on electricity generating costs for almost 200 power plants provided by 16 OECD member countries, 4 non-OECD countries and 4 industrial companies or industry organisations. The study presents the projected costs of generating electricity calculated according to common methodological rules on the basis of the data provided by participating countries and organisations. Data were received for a wide variety of fuels and technologies, including coal, gas, nuclear, hydro, onshore and offshore wind, biomass, solar, wave and tidal. Cost estimates were also provided for combined heat and power (CHP) plants, as well as for coal plants that include carbon capture. As in previous studies of the same series, all costs and benefits were discounted or capitalised to the date of commissioning in order to calculate the levelised costs of electricity (LCOE) per MWh, based on plant operating lifetime data.

1. Introduction

The competitiveness of nuclear power generation compared to other energy sources is a subject of considerable discussion and is highly variable between different countries. This variability depends on a number of assumptions that relate to government policies on security of supply and carbon pricing, the cost of financing, technology choices and operational performance.

A new joint study by NEA and the IEA, *The Projected Costs of Generating Electricity: 2010 Update*, which was released only on 25th March 2010 provides a number of important new insights on the competitiveness of nuclear power. The study, the seventh in a series, contains data on electricity generating costs for almost 200 power plants from 17 OECD member countries and four non-OECD countries. It was conducted under the supervision of an Expert Group with more than 50 experts from governments, industry, academe and international organisations.

The projected costs of generating electricity were calculated according to a common methodology on the basis of data provided by participants. The most important assumptions concern the choice of real discount rates (5% or 10% real), fuel prices and the use of a carbon price of USD 30 per tonne for the comparisons (see assumptions at end). The results cover a wide array of technologies, including nuclear, coal (both with and without carbon capture), gas, hydro, on-shore and off-shore wind, biomass, solar, wave and tidal. All costs and benefits are discounted to the date of commissioning in 2015 to calculate the levelised costs of electricity (LCOE) per MWh over the lifetime of each plant. A number of sensitivity analyses show the relative changes of LCOE in response to changes in key variables such as discount rates, fuel, carbon or construction costs, load factors and lifetimes.

2. The key results

The study reaches two key conclusions. First, at a 5% real discount rate, nuclear energy is the most competitive solution for baseload electricity generation followed by coal-fired plants without carbon capture and natural gas-fired combined cycle plants. Based on the data available for this study, where coal is low cost (such as in Australia or certain regions of the United States), both coal plants
with and without carbon capture (but not transport or storage, referred to as CC(S)) are within the zone of competitiveness in the low discount rate case. See Figure 1 below for a graphic display of a regional breakdown of the results at a 5% discount rate.

Second, at a 10% discount rate, coal without carbon capture equipment, followed by coal with carbon capture equipment, and gas-fired combined cycle turbines (CCGTs), are overall more competitive than nuclear energy under the assumptions of the study, although large regional differences remain. (In Asia, for instance, nuclear energy remains the most competitive technology even at a 10% discount rate). The results highlight the paramount importance of discount rates and, to a lesser extent, carbon and fuel prices when comparing different technologies. See Figure 2 below for a graphic display of a regional breakdown of the results at a 10% discount rate.

There is thus no technology that has a clear overall advantage globally or even regionally. Each technology has strengths and weaknesses depending on the circumstances:
**Nuclear** has the capability to deliver significant amounts of very low carbon baseload electricity at costs stable over time; it has to manage, however, high amounts of capital at risk and its long lead-times for construction. Permanent disposal of nuclear waste, maintaining overall safety, and evolving questions concerning nuclear security and proliferation remain issues that need to be solved for nuclear energy.

**Coal** is economically competitive in the absence of carbon pricing and neglecting other environmental costs. This applies in particular where coal is cheap and can be used for generating electricity close to the mine, such as in the western United States, Australia, South Africa, India and China. However, this advantage is markedly reduced where significant transport or transaction costs apply, or where carbon costs are included. The high probability of more generalised carbon pricing and more stringent local environmental norms thus drastically reduce the initial cost advantage.

**Coal with carbon capture** was shown to be competitive in countries with cheap coal and under certain assumptions about the cost of carbon capture. However it has not yet been demonstrated on a commercial scale for fossil fuelled plant. The costs provided in the study refer to carbon capture at plant level (an unproven rule of thumb says that transport and storage might add another USD 10-15 per MWh). Until a realistic number of demonstration plants have been operated for worthwhile time frames, total CC(S) costs will remain uncertain.

**Gas** has the great advantage of setting the price in competitive electricity markets thus hedging financial risk for its operators and having a lower CO\(_2\) profile; on the other hand, when used for base-load power production it has comparatively high costs. Progress in the extraction of lower-cost shale gas has eased the supply and demand balance and therefore improved the competitive outlook for natural gas.

**On-shore wind** is, for the first time, included among the potentially competitive electricity generation sources. On the basis of the dynamic generated by strong government support, it is currently closing the general competitiveness gap. Local conditions remain important. Its weakness is its variability and unpredictability, which makes system costs higher than plant costs. Other renewable technologies are currently not competitive for baseload power generation.

The current report on the *Projected Costs of Generating Electricity* has been produced in a period of unprecedented uncertainty. Costs rose rapidly over the last four years, before falling late in 2008 and 2009. The current economic and policy context is thus characterised, on the one hand, by the growing momentum of climate change policies, but uncertainty about timing of impact of policy measures and, on the other. This implies large dramatic changes in economic conditions affecting both energy demand and supply.

In addition to uncertainties described above, there are also other factors which cannot be adequately incorporated into a cross-country analysis but need to be acknowledged and are therefore dealt with in the study in a qualitative manner in dedicated boundary chapters:

- Integrating variable and intermittent renewable energies in most existing electricity systems;
- current cost of capital for energy projects and differences in tax treatment;
- issues in connection with the behaviour of energy markets (demand and price risk);
- cost of CC(S), a technology that can be key for the decarbonisation of the power sector, yet is still in the development stage;
- for renewable technologies, site-specific load-factors are frequently decisive.

### 3 Nuclear today: a sensitivity analysis of discount rates and carbon prices

One of the most interesting results concerning the competitiveness of nuclear energy against other technologies stems from the sensitivity analysis of cost variation due to the discount rates. Figure 3 shows that under the assumptions of the study, which include a carbon price of 30 USD per tonne of CO\(_2\), nuclear energy is the most competitive technology as long as discount rates are roughly below 7% per year. Given that this is a real figure, nominal figures in the market would add 2% for inflation.
To the extent that such general comparisons are meaningful, this overall picture suggests a relatively solid position competitive position for nuclear energy.

A similar conclusion can be drawn with respect to carbon prices. At a 5% discount rate, nuclear energy is competitive as soon as carbon prices rise above roughly 20 USD per tonne of CO2. It is always more competitive than gas, wind or coal with carbon capture.

The story changes when a 10% real discount rate is assumed. In this case, nuclear energy’s high capital costs would render it uncompetitive below carbon prices of USD 40 per tonne of CO2, when it would overtake coal with carbon capture, another high fixed cost technology. Given that the cost
The sensitivity to changes in carbon prices at a 10% discount rate.

Overall, the report shows that nuclear energy’s competitive position and the financing of NPPs are both linked to the cost of capital and carbon prices. The previous section explained that the cost of capital will depend to a large extent on the magnitude of the risks that are perceived by investors. On the basis of this analysis, governments will be able to assess the order of magnitude in costs shifts that their interventions will have to achieve in order to make the construction of NPPs again a reality.

**ANNEX 1 - Assumptions**

<table>
<thead>
<tr>
<th>Fuel prices</th>
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<tbody>
<tr>
<td><strong>Study Assumptions for Fuel, the Fuel Cycle and Carbon Pricing</strong></td>
</tr>
<tr>
<td><strong>Fuel prices</strong></td>
</tr>
<tr>
<td>Average OECD import price assumptions are comparable with the assumptions used in the 2009 World Energy Outlook (IEA, 2009). All prices apply to the plant gate:</td>
</tr>
<tr>
<td><strong>Hard coal (OECD member countries):</strong></td>
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<tr>
<td><strong>Natural gas (OECD Europe):</strong></td>
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<tr>
<td><strong>Natural gas (OECD Asia):</strong></td>
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<tr>
<td>For brown coal, the study assumed national assumptions for both price and heat content. In the case of Australia, Mexico and the United States, all of which are large coal and gas producing countries, where domestic prices can decouple from world market prices, the study has adopted national assumptions also for black coal and gas prices.</td>
</tr>
</tbody>
</table>

| Costs of the nuclear fuel cycle |
| Where domestic data was unavailable, data in terms of USD/MWh was defined on a harmonised basis: |
| **Front-end (Mining, enrichment, conditioning):** | USD 7 per MWh |
| **Back-end (Spent fuel removal, disposal and storage):** | USD 2.33 per MWh. |

**Carbon price**

The PCEG project works with a harmonised carbon price common to all OECD countries of USD 30 per tCO2 for all technologies.
White Paper

Title: Nuclear Operations Excellence: A Foundation for the Renaissance

Abstract:

The world is poised for a nuclear power renaissance. Countries with dormant or phased-out nuclear energy programs are reactivating their plans for building a new generation of safe and economical nuclear plants. Countries with no nuclear programs are actively working to establish them. The renaissance is being driven by a variety of factors, foremost of which are instabilities in the cost and supply of oil and gas, and increasing political pressures to prevent continued global warming as a result of the use of fossil fuels. Nuclear is now seen as a necessary component of a renewable, non-carbon producing energy portfolio along with hydro, wind and solar.

So what is required to turn these plans into reality? Most observers, both inside and outside of the industry would agree that the foundation of the renaissance is first and foremost the continued safe and economical operation of the world’s existing 436 reactors in 30 countries. Although operating experience varies across existing reactors, it can be confidently said that since the accident at Chernobyl, the vast majority of the world’s reactor operators have demonstrated continuous improvements in plant operations. If however, another Chernobyl or TMI type accident were to occur, it would have a devastating impact on nuclear programs all over the world, and seriously jeopardize the plans for a nuclear renaissance for many years to come.

Given the premise that continuing improvements in plant operations, or nuclear operations excellence, is a necessary first step in creating a sustained renaissance, what exactly is nuclear operations excellence, how is it measured, how is it achieved, and what are the requirements for the next generation of plants to be built? This paper will attempt to answer these questions, first in a broad sense based upon the 50+ years of combined nuclear industry experience of the authors, and also in a more focused exploration of information technology tools that are supporting the current generation of plants on their path to nuclear operations excellence.

Outline:

1. What is it?

2. How is it measured?
   a. Economics
      i. US_Electricity_Production_Costs_and_Components.xls
   b. Safety & Compliance
      i. US_Nuclear_Industrial_Safety_Accident_Rate.ppt
      ii. Significant Events at US_Nuclear_Plants.ppt
   c. Reliability
      i. US_Nuclear_Generating_Statistics.xls
      ii. US_Nuclear_Unplanned_Automatic_Scrams.ppt
   d. Operations and Maintenance
      i. NuclearOutageStats.xls

3. How is it achieved?
   a. Workforce training
   b. Performance improvement programs
   c. Technology and Innovation

4. What are the requirements for next-generation plants?
   a. EAM, EOM, ERP, PLM…
Nuclear Operations Excellence Defined

It is easy to recognize nuclear operations excellence when you see it. A few examples are contained in the sidebars. Precisely defining it, however, is a little more difficult and depends upon whom you might ask, and what their frame of reference happens to be. From the standpoint of a nuclear plant operator the term may mean the non-incidental performance and turnover of a duty shift. An outage manager may consider the term to mean the successful completion of a plant outage and a return to service, on-time and under budget. A site vice-president might look at an industry record such as the recent 705-day continuous operation run by TMI-1 as evidence of nuclear operations excellence. Of course, the true definition includes all of these, and more.

For the purposes of this paper the authors have chosen to define this term as it relates to supporting a global nuclear renaissance:

“Nuclear Operations Excellence is a continuous commitment by all levels of an organization to safe and economical plant operations as evidenced by a constantly improving set of global industry Performance Indicators.”

Having a broader, yet more precise definition of nuclear operations excellence, the task now is to understand how it is measured and how to achieve it.

Sidebars:

Exelon Companies
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Management
Newsroom
News Releases
Corporate
Power Generation
Recent
Archive
PECO
ComEd
PR 2009 10 12

News Releases
October 26, 2009 - Three Mile Island Sets Another World Record

LONGONDERRY TOWNSHIP, PA. (Oct. 26, 2009) Exelon Nuclear’s Three Mile Island Generating Station Unit 1 today broke the world record for longest continuous days of operation among all pressurized water reactors in the world when operators shut the plant down for a scheduled refueling outage.

The 870-megawatt plant operated for 705 days when it shut down for a scheduled refueling and maintenance outage earlier today. That beats the existing record of 692 days set earlier this year by the Calvert Cliffs Nuclear Power Plant located in Maryland. There are more than 230 pressurized water reactors worldwide.

TENNESSEE VALLEY AUTHORITY
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Browns Ferry Nuclear Plant Delivers Record Performance

April 10, 2002

ATHENS, Ala. — TVA’s Browns Ferry Nuclear Plant Unit 3 generated a world-record 17.8 billion kilowatt-hours of electricity, operating nonstop for 669 days. It was the second longest continuous run of a commercial power generating reactor in the United States, based on information available to the industry.

Browns Ferry workers followed that record performance by setting a U.S. record for the quickest refueling outage, returning Unit 2 to service in just 14 days and 16 hours early today.
Measuring Nuclear Operations Excellence

Several government and industry organizations, both domestic and international, are a good source of data for performance indicators (PIs) related to nuclear operations. Sources from which the data in this paper has been collected include:

- Nuclear Energy Institute (NEI – http://nei.org)
- U.S. Nuclear Regulatory Commission (NRC – http://www.nrc.gov)

So what categories of data are most important for first measuring nuclear operations performance and then extrapolating that as a necessary condition for a sustained future nuclear renaissance? The categories proposed herein are considered by most to be the cornerstone PIs covering economics, safety and compliance, and reliability:

**Economics**

Any decision for building new nuclear generation, be it by investor-owned utilities or government-backed entities, must first pass the test of economic viability.

To get an idea of the comparative costs of the various sources of electricity production, Figures 1 through 3 chart the per unit fuel, operations and maintenance (O&M) and total costs of U.S electricity production over the period 1999 to 2008 for the four major fuel sources – coal, gas, nuclear and petroleum.

![Figure 1: Source: NEI](image)

![Figure 2: Source: NEI](image)
Over the long-term, coal and nuclear have consistently had the lowest total cost of electricity production in the U.S. Although O&M costs for nuclear are generally higher than for all other fuel types, low and stable nuclear fuel costs compensate to provide the overall lowest production costs since 2001. In addition there has seen a steady decrease in nuclear O&M costs as a result of the maturation of the industry over the last 25 years. It should be noted that the data presented do not take into account the costs of climate change legislation in the form of a carbon tax, which would further bolster nuclear as the lowest total cost producer.

Given the fact that nuclear has consistently demonstrated low electricity production costs over the long-term, are there other economic factors that must be considered to make nuclear an economically viable alternative? The answer is the significant up-front capital costs that are required to license and construct a nuclear power plant before the first megawatt is ever produced. In the U.S., there are also risks associated with building in a deregulated market where cost recovery is uncertain. The cost for the first round of new reactors is estimated to be in the range of $8-$10 billion each.

In the U.S., the Energy Policy Act of 2005 and subsequent legislation was enacted to promote a new generation of commercial nuclear power plants encompassing next generation designs, regulatory reform, risk avoidance and financial incentives. A summary of funding for government nuclear energy programs as of October 2009 is provided in Table 1.

### U.S. Nuclear Energy Funding ($MM)

<table>
<thead>
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<tbody>
<tr>
<td>Nuclear Energy, Science And Technology</td>
<td>776.0</td>
<td>812.0</td>
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<td>261.0</td>
<td>273.0</td>
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<td>Fuel Cycle R&amp;D</td>
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<td>University Programs</td>
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<td>5.0</td>
<td>5.0</td>
</tr>
<tr>
<td>Innovative Technology Loan Guarantee Program</td>
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<td></td>
<td></td>
<td></td>
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<tr>
<td>Loan Cap, Nuclear Facilities</td>
<td>18,500.0</td>
<td>18,500.0</td>
<td>-</td>
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</table>
In October 2008, the DOE announced that it had received 19 applications from 17 companies for federal loan guarantees to support the construction of 21 nuclear reactors. Requested were loan guarantees in the amount of $122 billion, which significantly exceeded the $18.5 billion available. Even in the face of an administration which has not publicly embraced the nuclear option (reference the defunding of the Department of Energy’s Yucca Mountain nuclear waste repository), current climate change legislation proposals in the U.S. Congress continue to have a nuclear play a significant role in future energy policy.

There are sure to be setbacks on the road to new nuclear generation in the U.S., but just a handful of years ago few in the industry would have predicted that anyone would seriously be considering building a nuclear plant in the U.S.

Safety and Compliance

Safety and compliance are a gage of an industry’s commitment to a safe workplace and the effectiveness of internal procedures, processes and governing regulations.

WANO tracks industrial safety statistics for the global nuclear energy industry. A key indicator of worker safety is the industrial safety accident rate (ISAR), defined as the number of lost time, restricted work, or fatalities per 200,000 worker-hours. Figure 4 charts ISAR data for the global and U.S. nuclear energy industries during the period 1999 through 2008. Both show steady decreases in accident rates and are well below the WANO 2010 target rate of 0.2 accidents per 200,000 worker hours.

Table 1: Source NEI

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<tbody>
<tr>
<td>Loan Cap, “Front End” Fuel-Cycle Facilities</td>
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<td>2,000.0</td>
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<td>Administrative Expenses</td>
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<td>Nuclear Waste Disposal</td>
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<td>386.3</td>
<td>196.8</td>
<td>196.8</td>
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<td>Nuclear Waste Fund</td>
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<td>199.1</td>
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<tr>
<td>Defense Nuclear Waste Disposal Funds</td>
<td>98.4</td>
<td>187.2</td>
<td>98.4</td>
<td>98.4</td>
</tr>
<tr>
<td>U.S. Nuclear Regulatory Commission</td>
<td>1,060.0</td>
<td>926.0</td>
<td>1,016.0</td>
<td>1,056.0</td>
</tr>
</tbody>
</table>

Figure 4: Source: NEI, WANO
To gage worker safety in the nuclear industry compared to other industries, Figure 5 shows the average ISAR in the U.S. over the period 1999 to 2008 for the manufacturing, construction, and leisure industries. The data indicate that the nuclear industry consistently provides one of the safest industrial work environments from both an absolute measure and in comparison to other industries.

Another measure of safety in the nuclear industry is the collective radiation exposure indicator. WANO also tracks this PI on a global basis for all reactor types, including the most widely deployed boiling water reactors (BWRs) and pressurized water reactors (PWRs). Figure 6 trends the collective radiation exposure on a per-reactor basis over the period 1999 to 2008 for the different reactor types on both a U.S and global basis:

The generally lower trend in collective radiation exposure indicates the effectiveness of radiation protection programs across the global nuclear industry.

Turning now to regulatory compliance, the U.S. Nuclear Regulatory Commission’s (NRC) Reactor Oversight Process (ROP) evaluates and assesses plant performance by analyzing findings resulting from its inspection program and 16 performance indicators reported by the country’s 104 operating reactors. Based on this assessment, the NRC assigns each plant to a category (or color-coded column) of regulatory oversight in a quarterly Action Matrix Summary. Column 1 requires the least oversight and column 5 requires the most oversight:

1. Licensee Response
2. Regulatory Response
3. Degraded Cornerstone
4. Multiple/Repetitive Degraded Cornerstone
5. Unacceptable Performance
Figure 7 charts the number of plants in each column of the Action Matrix Summary during the period fourth quarter 2000 through second quarter 2009. Top performing plants are in column 1 (Green) which over the period has averaged over 77% of all U.S. plants. Similarly over 94% of plants over the period have been categorized in columns 1 or 2 (White), those requiring the least amount of regulatory oversight. No plant has ever been listed in column 5.

![NRC Action Matrix Summary](image)

Figure 7: Source: NRC

Clearly the vast majority of plants in the U.S. are determined by the NRC to be operating to the highest standards of safety and compliance which are sustained over the long-term.

Reliability

There are many PIs that provide a good measure of plant reliability, a few of which are reported here. Figure 8 shows the number of unplanned automatic scrams (emergency plant shutdowns) per 7,000 hours critical (approximately one year of operation) during the period 1999 to 2008 for both the global and U.S. nuclear industry:

![Unplanned Automatic Scrams](image)

Unplanned automatic scrams are a challenge to both plant safety systems and plant operators and trends show very low sustained rates in the U.S and a generally lowering trend downward globally. These rates are now significantly lower than in the early 1990’s where global industry rates averaged 1.8 scrams per 7,000 hours critical.

Another indicator of reliability is the duration of the refueling outage, which is typically required every 12 to 24 months depending upon the design of the reactor fuel. Figure 9 shows the average duration of refueling outages in U.S plants during the period 1999 to 2008:
The trend shows generally stable to slightly decreasing refueling outage durations on the order of 38 days in 2009. This trend is a significant improvement from the early 1990’s where refueling outage durations averaged over 100 days. In the late 1970’s pressurized water reactors in the U.S. began replacing their steam generators. The first steam generator replacement (SGR) outage at Dominion Nuclear’s Surry plant took over 300 days to complete in 1979. Most SGR outages are now completed in less than 60 days.

And finally, perhaps the best measure of reliability is the capacity factor or unit capability. The unit capability is the ratio of the actual output of the generating unit over a period of time and its output if it had operated at full capacity during that time period exclusive of planned outages. It is basically a measure the amount of time a plant is on line and producing electricity. Figure 10 shows unit capability during the period 1999 to 2008 for both the global and U.S. nuclear industry:

In the U.S, unit capability for nuclear generation has consistently been above the 90 percent level with global performance reaching into the mid to high 80 percent levels. This too is a significant improvement in overall industry performance which averaged in the mid-70 percent range for much of the 1990’s in the U.S. and lower 80 percent range globally.

For comparison purposes Figure 11 shows preliminary 2008 unit capability data for all sources of electricity generation in the U.S.:
The data presented here illustrate the significant advantages of nuclear electricity generation over all other sources from the standpoint of economics, safety, regulatory compliance and reliability. As the industry has matured both in the U.S. and globally it has produced a consistently improving set of operating results over a wide range of technical, financial and political environments.

So the question now becomes how did the industry achieve such high levels of operating performance, and how can this performance be extrapolated to the nuclear renaissance?
Achieving Nuclear Operations Excellence

Having defined and measured what constitutes nuclear operations excellence, the task is now to understand how the global nuclear industry is able to achieve and sustain such high levels of operating performance. One thing is clear when analyzing the components of success the industry has achieved over a relatively short period of time: there is no one single method, process or activity that can be identified as having been responsible for such performance. Rather it is a combination of many factors involving workforce training, performance improvement programs and adoption of new technologies that has come together to produce sustained performance over the long term.

Workforce Training

Clearly a major contributor to the improvements the industry has made over the last decade has been the high level of performance of the workforce responsible for operating and maintaining operating reactors throughout the world. This performance is a direct result of significant efforts by the industry to establish and maintain comprehensive personnel training and accreditation programs.

Soon after the accident at the Three Mile Island plant in 1979 the U.S. nuclear energy industry formed the Institute of Nuclear Power Operations (INPO®) to promote excellence in nuclear power plant operations, including the training of personnel.

Under INPO the National Academy for Nuclear Training was created in 1985 to integrate the training activities of INPO, all U.S. nuclear operators and the National Nuclear Accrediting Board (NNAB). The academy's activities range from conducting workshops for training managers to courses for senior plant management. Individual utility training programs covering operations, maintenance and technical training programs for all key positions at each plant are formally accredited through the independent NNAB every four years. The requirements of an accredited training program include:

- Self-evaluation based on industry-wide standards
- Peer review from a team of training experts
- Review by a panel of the NNAB

The NRC oversees virtually every aspect of nuclear power plant personnel training - setting regulatory requirements, providing regulatory guidance, inspecting programs and enforcing requirements addressed by 10 CFR 50.120 and 10 CFR 55.

Nuclear operating companies have invested thousands of work hours and hundreds of millions of dollars in training. In 1979, U.S. companies had only 12 control room simulators; today, there are 74 in use. Since 1979, the number of professional training personnel has jumped from 440 to about 5,000, and the space dedicated exclusively to training has increased eightfold.

Experience in training programs is shared globally through the collaborative efforts of organizations such as INPO and WANO.

Performance Improvement Programs

In 2005, INPO published INPO 05-005, Guidelines for Performance Improvement at Nuclear Power Stations. The foundation of the guideline is that the performance improvement process is a continuous cycle of analyzing, identifying, planning, implementing and monitoring solutions.
The nuclear industry has at its disposal a variety of resources to support the effort to continually improve processes critical to plant operations. Standards organizations such as the American Nuclear Society (ANS) have published nearly 150 different standards to establish safe practices. These technical documents are written, reviewed, and approved by close to 800 volunteers who provide their experience and expertise to the various levels of standards committees.

In the area of operating experience review, the NRC created a “clearinghouse” group to collect and evaluate experience every day, then appropriately share it throughout the agency and the industry. The clearinghouse applies the lessons learned to the NRC’s core functions of oversight, licensing, rulemaking, and incident response. In addition, the agency is using information technology to more widely share operating experience, including a central database for managing all reported events, and an operating experience gateway that consolidates a variety of information onto a single page on the NRC Web site: http://nrcoe.inel.gov/results/.

Technology & Innovation

Since 1994 the NEI has recognized industry excellence through its Top Industry Practice (TIP) award program. The program highlights technology and process innovations in the nuclear energy industry that have broad applicability and have increased the safety and/or economics of plant operation. Over 130 entries were received in 2009 from across the industry for innovations in the following categories:

- Community Relations
- Configuration Management
- Equipment Reliability
- Loss Prevention
- Management & Support Services
- Materials and Services
- Nuclear Fuel
These awards, summarized in Table 2 provide insight into the innovations and technology that have in part been responsible for the industry’s high level of performance over the last decade.

<table>
<thead>
<tr>
<th>Category</th>
<th>Innovation</th>
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<tbody>
<tr>
<td>Configuration Management</td>
<td>Solution for eliminating vibrations in main steam lines</td>
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<td>Methodology for scheduling inspections</td>
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<td>Steam dryer flow model to analyze vibration problems</td>
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<td>Multi-disc traveling water screen</td>
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<td>Design of a clamp device</td>
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<td>Permanent drywall shielding</td>
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<td>Integrated containment management system*</td>
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<td></td>
<td>Post accident sampling system elimination*</td>
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<tr>
<td>Equipment Reliability</td>
<td>Condenser tube aging management</td>
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<td>Fleet-wide plant performance monitor</td>
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<td>Remote-operated vehicle for reactor inspection</td>
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<td></td>
<td>Upper reactor head assembly installation</td>
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<td>Energize-to-actuate valve modification</td>
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<td>Cooling tower design technique</td>
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<td></td>
<td>Equipment reliability improvement programs</td>
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<td>PWR asset management system</td>
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<td></td>
<td>Integrated risk management process</td>
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<td>Phased-array ultrasonic inspection</td>
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<td>Stress corrosion assessment using eddy current testing*</td>
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<td>Noble metal application to reactor vessel internals*</td>
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<td>Method to demonstrate benefits of zinc*</td>
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<td>Primary water stress corrosion cracking assessment*</td>
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<td></td>
<td>Reactor vessel head low frequency eddy current technology*</td>
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<td>Testing process for leaks in control rod drive mechanisms*</td>
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<td>Automated crack detection system*</td>
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<td>Loss Prevention</td>
<td>Water diversion device</td>
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<td>Installation and disposal of steam dryers</td>
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<td></td>
<td>Improving human performance</td>
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<td>Electronic database for license renewal</td>
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<td></td>
<td>Exemption from inspections on equipment with low safety significance</td>
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<td></td>
<td>RPV foreign object vacuum system*</td>
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<td>Management &amp; Support Services</td>
<td>Team to explore cutting-edge technologies and concepts</td>
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<td></td>
<td>Tiered, team approach to improve efficiency and productivity</td>
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<td></td>
<td>Mobile radiation data software</td>
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<td>Legislative campaign</td>
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<td>10 CFR 50.59 standardized methodology</td>
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<td>Penetration resistant security door</td>
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<td>Supplemental workforce improvement initiative</td>
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<td>Movable 86-ton radiation shield walls</td>
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<td>Category</td>
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<td>NEI Top Industry Practice Awards 2000-2009</td>
<td>Mechanical stress improvement program*</td>
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<td>Data exchange for reactor fuel assembly management*</td>
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<td>Widespread use of a wireless network*</td>
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<td></td>
<td>Data management system for refueling*</td>
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<td>Spectroscopy technique for analyzing boric acid residue*</td>
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<td>Determining the cause of deposits on fuel cladding*</td>
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<td>Control element drive inspection and repair enhancements*</td>
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<td>Fleet-wide, long-term asset management plan *</td>
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<td>Diesel generator assessment model*</td>
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<td>High-density polyethylene piping</td>
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<td>Materials and Services</td>
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<td>Conversion to graphite pressure seals for valves</td>
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<td>Industry-wide updated process description and guideline</td>
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<td>Streamlining core reload design and licensing schedule</td>
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<td>Tungsten wall shielding innovation</td>
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<td></td>
<td>Recirculation jet pump riser pipe repair</td>
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<td>Control rod drive mechanism sample specimen*</td>
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<td>Innovations in condensate polishing*</td>
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<td>Chemistry instrument upgrade*</td>
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<td>Controllers to improve material handling during outages*</td>
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<td>Boiling water reactor (BWR) chemistry optimizer*</td>
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<td>Mechanical nozzle seal assembly*</td>
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<tr>
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<td>Decay heat valve canopy modification*</td>
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<td>Reactor startup and shutdown chemistry optimization*</td>
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<td></td>
<td>Laser-based remote inspection tool*</td>
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<td>Next generation reactor core design improvements</td>
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<td>Zinc addition with high boiling duty pressurized water reactors</td>
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<td>Nuclear Fuel</td>
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<td>Fuel assembly filters*</td>
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<td>Modeling of river basin</td>
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<td>World record 20% up-rate in thermal power</td>
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<td>Solution to limit piping corrosion by air injection</td>
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<td>Method to improve safety while conducting inspections</td>
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<td>Software to predict the levels of xenon in the reactor core</td>
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<td>Eliminating corrosion</td>
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<td>Software tool for worker qualification data</td>
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<td>Training model that focuses on human performance factors</td>
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NEI Top Industry Practice Awards 2000-2009

<table>
<thead>
<tr>
<th>Category</th>
<th>Innovation</th>
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<tbody>
<tr>
<td>Automated simulator training</td>
<td>Use of simulator technology for plant design</td>
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<td>Multi-tasking training model</td>
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<td>Troubleshooting renaissance process</td>
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<td>Knowledge management tool</td>
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<td>Vision and Leadership</td>
<td>International technical exchange program</td>
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<td>Offset consumptive water needs with water from mines</td>
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<td>Nuclear fuel assembly inspection program</td>
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<td>Use of co-op students to improve equipment reliability</td>
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<td>Work &amp; Maintenance Management</td>
<td>Tooling for the inspection of a BWR steam dryer</td>
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<td>Remediation of pipe corrosion using chemicals</td>
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<td>Alloy 690 replacement pressurizer</td>
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<td>Creation of a vertical pump alignment tool</td>
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<td>Techniques for replacement of reactor instrument tube nozzles</td>
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<td>In-vessel underwater robotic inspection system</td>
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<td>Simplified reactor vessel head assembly</td>
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<td>Steam generator high impact team</td>
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<td>Electronic safety analysis report</td>
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<td></td>
<td>Reactor vessel flange repair platform*</td>
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<tr>
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<td>Method to examine control rod drive mechanism nozzles*</td>
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<td></td>
<td>Outage management of major nuclear refurbishments*</td>
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<td></td>
<td>Replacement of pressurizer*</td>
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<td></td>
<td>Major plant modifications to facilitate a power up-rate*</td>
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<td></td>
<td>Refueling and reactor head assembly replacement*</td>
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<td>Integrated reactor head assembly*</td>
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<td>Auxiliary bridge modification / inspection deck*</td>
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<td>360-degree BWR vessel-servicing platform*</td>
</tr>
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<td></td>
<td>Leak repair and corrosion cracking*</td>
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</table>

Table 2: Source NEI® (* re-categorized from vendor award)

EAM and EOM Technologies Supporting Nuclear Operations Excellence

Over the years several of the TIP awards were related to asset management and tools and practices. In the following sections we will describe several practices in Enterprise Asset Management (EAM) and Enterprise Operations Management (EOM) software solutions which are a direct contributor to nuclear operations excellence.

EOM spans a wide range of processes and practices including interoperability with other enterprise applications and data. The major functional areas addressed by EOM covers a broad range of operations based processes:

- Plant Status Control
- Routine Duties Automation
- Safety & Compliance
- Workforce Qualification & Scheduling
- Change Management
- Communications, Records & Reporting
Some examples of the application of EOM solutions to enable best practices in nuclear plant operations include:

- **Electronic Equipment Lineup Procedures** - At the end of a nuclear plant outage, Operations begins the long process of re-establishing configuration control of plant systems by performing equipment lineup procedures. These procedures are critical path items in the return of the unit to service and shortening the time needed to perform these lineups will have a direct impact on the cost and schedule of an outage. By implementing a fully electronic procedure management system as contained in the EOM plant status control solution, plant operators have been able to achieve significant savings in both time and radiation exposure when performing these procedures.

- **Plant Status Control Electronic Deviation Records (DRs)** – Having established normal plant configuration with an electronic equipment lineup procedure solution, operators use electronic DRs to document and track many of the activities which cause plant equipment to be in an off-normal configuration or operate in a off-normal status. These activities may include abnormal equipment, temporary modifications, equipment out-of-service, abnormal equipment configuration, jumpers, operator aids, etc. DRs can be used to alert operators to restricted plant mode changes to prevent entering modes of operation that require deviated equipment to be in service.

- **Tag Sharing in the Clearance Process** - The tagout/lockout process is perhaps the most important and time-consuming function that a plant’s operation and maintenance staff performs because it is critical to both personnel and plant safety. Errors in the tagout/lockout process can result in work delays, safety stand-downs, regulatory fines, and personnel injury or death. By automating this process, significant gains can be realized in both efficiency and safety. An electronic clearance solution is designed to assist in implementing, controlling, and executing a plant’s tagout/lockout process. The system enforces the logic of plant-specific procedures to ensure personnel compliance and facilitates access and linkage to equipment isolation, work status, and schedule information through interfaces to work control, maintenance management, and scheduling software systems. Clearance solutions which incorporate tag sharing, i.e. sharing a single physical tag among multiple work activities has been proven to save a significant amount of manpower resources during both normal plant operation and outages, without compromising personnel or plant safety.
Requirements for next-generation plants

- Identification and Management of Risk
  - Demonstrating a compelling risk-adjusted, cost/benefit business case analysis is required to obtain needed project approvals and financial backing.
  - Taking more direct control of the project Licensing and QA aspects.
  - Maintaining comprehensive configuration management between the certified reactor design (reference plant), the COL(A), and as-built.
  - Managing ITAAC, Corrective Action, Employee Concerns, Safety (permits, tagging, MSDS, etc.) and other programs.
  - Dedicating commercial grade materials and components to compensate for the lack of available safety-related items.
  - Determining, qualifying, tracking, and renewing required personnel skills to deal with aging and expanded work force needs.
  - Eliminating the cost and complexity of turning over very large quantities of data just prior to plant startup.
  - Communicating with an extensive group of stakeholders and regulators.

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2. Nuclear Energy Funding, Nuclear Energy Institute, October 2009
3. U.S. Nuclear Industrial Safety Accident Rate, Nuclear Energy Institute, April 2009
   [http://www.wano.org.uk/PerformanceIndicators/PI_Poster.asp](http://www.wano.org.uk/PerformanceIndicators/PI_Poster.asp)
   [http://www.bls.gov/iif/osh_nwrl.htm#industry](http://www.bls.gov/iif/osh_nwrl.htm#industry)
7. U.S. Nuclear Refueling Outage Days (Average, 1990-2008), Nuclear Energy Institute, January 2009
9. INPO 05-005, Guidelines for Performance Improvement at Nuclear Power Stations
10. TIP Award News Releases 2000-2009, Nuclear Energy Institute
NEW APPROACH OF SECOND ROMANIAN NPP SITING

TRAIAN MAUNA
Romanian Nuclear Energy Association-AREN-
65 Polona, C.P. (PO Box) 22-102, 010494 Bucharest - Romania

ABSTRACT

The NPP sitting studies in Romania began before 1975. The first Romanian NPP CANDU 6 type reactor gone to erection in 1980 on Cernavoda site planned to have 5 units. Gained the experience from Cernavoda NPP sitting, the first mission of new multi-branch of specialists team was to choose new NPP sites adapting the NPP Cernavoda project to the new parameters of close water cooling circuit and hard less and no rock foundation strata. The studies were carrying out in different stages on the inner rivers Olt, Mureș, Someș in Transylvania historical region. This paper tries to reconsider shortly the old analysis according to the last IAEA Safety Standards, taking into account the new NPP generation requirement. Paper is focused on geological aspects and other local sites characteristics.

1. Introduction

After the finalized sitting characterizations for Cernavoda NPP the works began on site in 1980. This plant is based on CANDU 6 reactor type with natural uranium oxide as fuel and heavy water as moderator, as reactor cooling and heat transporter of primary circuit. Both uranium fuel bundle and heavy water are manufactured in Romania using national technology. Nuclear fuel fabrication is based on yellow cake preparation from Romanian ore extraction. The Cernavoda NPP was equipped with GE via Ansaldo turbine having cooling water open circuit. The 54 m$^3$/s cooling water flow for every 700 MWe unit come from Danube River via the western inner sector of Danube-Black Sea Canal designed to satisfy the whole water flow for 5 units. The former Romanian authorities had approve the Nuclear National Plan having as final target total 16 CANDU 6 reactor type units to built overall the country. One of big problem was to adapt the turbine parameter of cooling water from open circuit to the closed circuit. In this case the negotiation of turbine licence condenser was done with two ways cooling water for closed circuit. The studies for correlating steam parameters and turbine systems put to Romanian researchers and designers a lot of problems to be solved. By the other way was from 1982 started sites surveys for new NPP on the inner Romanian rivers oriented mainly on Transylvanian depression. Based on experience of Cernavoda NPP site selection the team of multi-branch specialists analysing the AECL site recommendations, US 10 CFR regarding nuclear sitting, the draft or first issued of IAEA Safety Sitting Standards, the analysing of existing NPP site get to issue their specific requirements for new Romanian close circuit reactor CANDU NPP type or alternative reactor types as second proposal. This team was concentrating their efforts to improve the project for closing cooling water circuit and type of cooling tower and induced changes. The main problem was the foundation type of soil because the CANDU reactor was regular inserted into hard rock type on site having less than 0.2g free field maximum expected acceleration. The Romanian state politics of 80 years was to increase the assimilation of equipments, reducing the import and to adopt building solution with less materials consumption. This permanent intrusion of administration or party people in technical solution created a big stress on the specialist peoples involved in NPP design and execution. The strong economy of fuels was enforced so the research and design in hard condition can't give too good results of NPP sites surveys and investigations.
2. Sitting requirements

The requirements for inner country rivers NPP site are based on a good safety culture as defined in [1] in order to carry out all procedures for site selection. For Romanian site evaluation of NPP harmonized with the requirements from [2] must take into account for acceptance the following aspects:

(a) To protect the public and the environment from the radiological consequences of radioactive releases in normal operation or accidents in terms of safety and below under limit established by regulatory body, due to internal or external events;
(b) To implement emergency measures and evaluate the risks to individuals and the population and also for environment;
(c) To have enough resources required for erection and operation of NPP, no safety related;

Not all requirements from IAEA Safety Guides or other type of publications have applicability to Romanian regions, regarding mainly to the extreme natural phenomena or volcanic activities.

In Romania was 4 zones surveyed and/or investigated carefully:

i. on right side of Danube River downstream from Cernavoda NPP, on the rocky strata, having enough water for open cooling circuit;
ii. on left side of Olt River at the north frame of Făgăraș Mountains, on the south part of Transylvanian Depression, on soft strata;
iii. on both side of Mureș river valley traversing the Apuseni Mountains, on the rocky strata, on the western part of Transylvanian depression;
iv. on the both side of Someș river at the north of Apuseni Mountains, on the soft rocky strata.

Fig 1 Romania inner zones surveyed or investigated for NPP sites

The sites survey and analyse was done using exclusion matrix method and factorized site range. So step by step was applied the “Site allowable” \( S_a \) formula for every site according to phase of analyze:
where:

\[ S_a = \sum_{i=1}^{i-n} r_i \times w_i \]  

- \( r_i \) = range safety
- \( w_i \) = weight factor in connection with level of satisfy of requirement and technical solution

2.1. Specific safety requirements for inner country NPP site

- Earthquakes, Romania have a many earthquakes regions like Vrancea and other active faults;
- Surface faulting is double along Olt valley, one having a small seismic activity;
- Floods due to precipitation can not affect the sitting for the upper level of the valleys;
- Floods and waves caused by biggest rain or failure of existing or planned water control structures don’t affect the sites;
- Behaviour of foundation materials rocked or soft must give very small settlement under load combinations;
- Aircraft crashes assessment can be make because all area are under aircraft corridor, mainly for local flying;
- Chemical explosions from the facilities must take into account on the Olt valley and near the underground gas extractions facilities or processing plant for all regions in Transylvania;

Fig. 2 Olt river NPP potential sites

- Ultimate heat sink can be realized by an adequate reservoir solution;
- Atmospheric dispersion of radioactive material is generally good in the all area;
- Dispersion of radioactive material through surface water is considered to be acceptable and can be kept under legal limits by adequate monitoring;
- Dispersion of radioactive material through groundwater was not studied;
- Population distribution there are acceptable, is not very big town and the villages are small along the roads and can be assured exclusion area 1000 m radius, low population area 2000 m radius, both measured from the reactor building center point;
- The site having slope instability, collapse, subsidence/uplift, soil liquefaction, caverns existence, karstic formations and human made features such as mines, water wells and oil wells strata foundations characteristics are not accepted.
2.2. Requirement not related to safety

− Water flow 5-20 m³/s need for the cooling water in closed circuit with low thermal river water pollution, Olt river having around 70 m³/s mean years flow Near Făgăraş, Mureş river around 150 m³/s, Someşul Mare 70 ÷120 m³/s;

![Fig 3 Mureş river NPP potential sites](image)

− The maximum level of water river flooding enforce the level of NPP site level preparation and earth moving;
− The allowable route solution for very big size and heavy equipments transport between factories and site taking into account the Carpathian Mountains ring;
− Solution for BOP building must be in accordance with EUROCODES;
− Road and railroad for easy access and intervention must be constructed;

![Fig 4 Someş river NPP potential sites](image)

− Temporary building and facilities can be as close as possible for the working points;
The dwellings for operating staff must be into a low population area in order to give a trust in NPP function and have easy permanent public acceptance;

Fig 5 Romanian electric network status and planned

- Built and endow the public information centre;
- Improve the environmental landscape architecture by an adequate gross plan taking into account high of natural draft cooling tower;
- Assure the good rein water control and drainage from the NPP platform;
- Assure the all type of utility and service according to site position, environment and local sources;
- Provide a systems for physical protection and strong access control into nuclear area;
- Develop a modern infrastructure for the close village around the NPP.

3. Conclusions

- The Cernavoda NPP is only one CANDU type reactor from Europe safest from Eastern Europe, meeting the all last and new requirements in connection with, being a good example for the next sites;
- The very detailed geological, geophysical, geochemical and geotechnical studies are imperative need to develop into the next two years in order to have as good as possible knowledge about underground strata behaviour, due to sudden change of properties strata probability;
- The old survey and investigation of NPP site can not be conclusive to make a decision so any political people announcement are not based on data from specialists;
- During last 20 years NPP site investigation was not done in Romania;
- After 1990 many Romanian societies involved in NPP survey sitting was split or change their work;
- None of NPP site surveyed or investigated as above mentioned can be declared as selected site ready for bid.
- NPP having CANDU type reactor can be implanted on the Danube River site only using the open cooling water loop;
- The site on inner rivers need huge cooling tower for closed cooling circuit of turbine condenser and other heat exchangers.
4. References (selected)

ABSTRACT

The development and negotiation of an Engineering, Procurement and Construction (EPC) contract is a multi-disciplined and time consuming process. Relative to the first wave on new nuclear build projects of the 1950s – 1970s, today’s EPC contracts are more complex for a variety of reasons including more demanding regulatory and environmental requirements, global supply chain versus localization issues and different world wide economic considerations. This paper discusses the impacts of some of these challenges on developing an EPC contract in today's Nuclear Renaissance.

1. Introduction

In the current era of new nuclear build, which has often been referred to as the “Nuclear Renaissance”, much has changed in light of past experience in terms of the technology, regulatory requirements and commercial approach to new projects. This paper examines some of the challenges in developing engineering, procurement and construction (EPC) contracts in today’s Nuclear Renaissance in the context of sometimes conflicting needs such as complying with country specific regulatory requirements while trying to retain the advantages of standardized designs. Adding to the complexity of the EPC contract is the fact that there are more parties to these contracts, given the EPC’s wide-ranging scope. One of the purposes of an EPC contract is to document how risk is allocated and managed during a project. A few risk mitigation strategies are discussed with particular attention as to how the overall project risks can be equitably distributed amongst the parties.

To better address the challenges of developing an EPC contract, it is important to understand the history of how we arrived at this point in the commercial nuclear power industry. During the first wave of nuclear power plant projects, starting in the late 1950’s through the early 1970’s, if there was one thing that was a constant, it was “change”. Regulatory requirements were changing as the industry was growing, and these changes often impacted the designs, even while plant construction was underway. Nuclear steam supply system technology itself was evolving, power output was increasing, and there was fairly widespread acceptance by the public of nuclear power around the world.

1.1 The First Generation Commercial Plants

The commercial nuclear power era began when the world’s first nuclear power plant was commissioned on June 27, 1954 in Obninsk, Russia. The Obninsk nuclear power plant featured one 5 megawatt electric (MWe) Atom Mirny (AM-1) or Peaceful Atom reactor that was graphite moderated and water cooled. The next commercial nuclear power application to be deployed was the 50 MWe Calder Hall-1 gas cooled reactor (GCR) plant in the United Kingdom, which was officially opened by Queen Elizabeth II on October 17, 1956. Nuclear power applications continued to expand and diversify as the 60 MWe Shippingport pressurized water reactor (PWR) in the United States was commissioned in December of 1957 and a light water cooled graphite moderated reactor (LGR) was started in September of 1958 at the 100 MWe Troitsk A plant in Russia. The next reactor design to achieve commercial operation was the boiling water reactor (BWR) which was deployed in July of
1960 in the 200 MWe Dresden-1 plant in the United States. The first generation of commercial nuclear technology was rounded out with a liquid metal cooled fast breeder reactor (LMFBR) being commissioned in August of 1966 at the 61 MWe Fermi-1 plant followed by a pressurized heavy water reactor (PHWR) at the 216 MWe Douglas Point plant in Canada in September of 1968. The commercial operation dates and megawatt output information were obtained from Reference 1. The growth of commercial nuclear power did not stop with these first generation plants, but continued to expand as electrical demand increased and technology advances made nuclear power plants an attractive option for utilities. By the time of the accident at Three Mile Island in 1979, the net electric output of some of the units being deployed had increased to over 1300 MWe. During this expansion period, the six first generation reactor suppliers spawned additional suppliers and manufacturers through licensing agreements, partnerships and technology transfers. This first period of nuclear power plant builds was ultimately supplied by approximately twenty four (24) different reactor designers / manufacturers.

With these kinds of evolutionary changes taking place in the first wave of nuclear plant building, standardization was not a priority. However, reactor suppliers did pursue developing “classes” of plants in an attempt to gain some advantages from standardization. Nonetheless, even within a class of plants, there were often significant differences as the utilities contracted with different architect engineers (AE’s) and constructors to build the plants. In some instances, the utility further diluted the benefit of being part of a class by selecting different turbine suppliers for the same reactor design on a single site. Calvert Cliffs is an example of this, where Unit 1 has a General Electric supplied turbine and Unit 2 has a Westinghouse turbine. Additional examples were also seen as a few utilities chose to deploy different reactor vendor designs at a single site, with the strategy that they would be less susceptible to having to shutdown all of their nuclear plants at that site in the event that a particular vendor’s technology had a systematic technical or licensing problem. The Ringhals site in Sweden is an example of this where an ABB-Atom BWR and two types of Westinghouse PWRs were built. In the US, the Millstone site is another example where the utility deployed a GE BWR, a CE PWR and a Westinghouse PWR. Limited application of standardization led to challenges and inefficiencies in construction, equipment supply, licensing, training, spare parts and operations.

2. What Has Changed in Today’s Nuclear Renaissance?

In the time period following the accident at Three Mile Island, the projection of reduced electrical demand along with other factors such as high interest rates and licensing uncertainty, led to a dramatic decline in the contracting for new nuclear plants. As a result, there was a consolidation of reactor suppliers through buyouts, partnerships and some just going out of business. For the surviving suppliers, there was a greater focus on standardizing designs. Today, there are ten major reactor suppliers actively pursuing the commercial nuclear power plant market. The technologies represented by these suppliers include PWRs, BWRs and PHWRs. These technologies feature enhanced, traditional safety system design approaches, such as the ABWR and EPR, or passive safety system designs such as the AP1000 and the ESBWR.

2.1 The Regulatory Environment, Then and Now

During the first wave of new nuclear build, the regulatory landscape was changing almost as rapidly as the technology was evolving. Regulatory agencies were evolving to address the new technology offerings as well as the growing list of operational and safety issues that were developing. Some examples of new safety related issues that surfaced in the late 1960’s and 1970’s include addressing the small break loss of coolant accident (SBLOCA) scenario, large break loss of coolant accident (LBLOCA), anticipated transients without scram (ATWS) and station blackout. In resolving some of these issues, many nuclear plants
had to install additional hardware such as high pressure safety injection pumps, safety injection tanks and an alternate reactor scram capability.

The licensing process used in the United States by the NRC for most of the first wave on new nuclear build is prescribed in the US Code of Federal Regulations, 10 CFR Part 50. This process is a two step process whereby the utility would apply for a construction permit followed by submitting an application for an Operating License. This two-step process had inherent risks in that a utility could embark on construction, and then have the Operating Licensing application delayed or possibly denied altogether. The Shoreham plant in New York is a prime example where a plant was built but never used. This regulatory environment led to uncertain schedules and often unanticipated hardware requirements, all of which led to indeterminate costs. This approach to licensing could not underpin new plant construction in the US. Fortunately, the NRC has adopted a new, one step licensing process which combines the construction and operating licenses through a COL application or COLA. In the United States, the ABWR and AP1000 have achieved initial Generic Design Certification, and the EPR, ESBWR and USAPWR are undergoing their initial design certification reviews.

While the one-step licensing process described above is a significant and necessary improvement over the previous two-step process for conditions in the US, not all countries have adopted this approach. In such countries, there is a potential conflict between offering a standard plant while still having to meet country specific licensing requirements. The licensing uncertainty and associated review process could result in requirements for design changes or otherwise cause delays in the project schedule, both of which would increase project costs. If there is a potential that the country-specific regulator could require design changes, the benefit of deploying a standard plant is reduced. One method that is often applied to mitigate this risk is to have a two-phase contract. The first phase addresses project execution prior to achieving licensing approval. The second phase addresses full project execution and construction once the required approvals have been received. The licensing process could be streamlined if more countries that are contemplating new nuclear projects would take credit for or apply the benefits from other regulator’s certifications.

2.2 Contracting Challenges - Standard Plant and Localization Requirements

Some of today’s EPC contracts require that a significant portion of the work scope (labor & equipment) be sourced locally in order to stimulate the internal economy and provide long term employment opportunities (commonly termed “localization”). While implementation of a standard plant supports these goals, localization requirements may negate some of the benefits of standardization from a cost and experience perspective. The vendors that offer standard plants have developed supply chain sources and often negotiated prices that reflect a certain assumed level of production volume. If the localization requirements limit the application of these benefits, this can increase the cost of the plant. Additionally, local suppliers may not have the experience that standard suppliers have which may add risk to the quality and schedule performance on the project. The challenge in developing the EPC contract is to construct contract provisions that balance risk to benefit and optimize the advantages of deploying a standard plant design. One way to mitigate some of the risks associated with localization is to support development of the local supply chain prior to the commencement of the EPC contract.

2.3 Phased EPC Contracts

One approach in developing an EPC contract is to negotiate and sign a single contract that addresses all aspects and phases of the project, including early project and site development activities. This approach may be desirable if there is a clear need to start the project in order to meet anticipated electrical demand or qualify for incentives and a reactor technology selection has already been made. Having a full EPC in place early in the
process supports going forward with activities such as long lead material purchases that may be critical to the overall plant schedule. A risk with the single contract approach is that the total scope, full cost and schedule details may not be known at the time the full EPC contract is signed, so that flexibility must be allowed in the EPC contract to adjust the contract as results and information from early project and site development efforts emerge. Reasonable contract off-ramps need to be considered in this EPC contract approach should information become available that differs significantly from expectations such that it would diminish the value or increase the risks of the project to the point that it is no longer viable for either party.

Another contracting approach that is often applied is to execute a separate Early Work Agreement (EWA), and in parallel, negotiate the specifics of the larger EPC contract. Typical EWA’s include work tasks to evaluate local site geological conditions and condenser cooling options which impact the balance-of-plant design, as well as developing site layout considerations for construction. By completing an EWA contract before the EPC is signed, a more complete understanding of the scope, costs and schedule considerations will be achieved to better configure the ultimate EPC contract and more equitably balanced the risk.

2.4 EPC Risk Allocation

New nuclear build projects are costly, multi-year endeavours with the potential for all key parties to experience significant risk. In developing an EPC contract, a fair and balanced risk allocation should be the goal of all parties. A good contracting principle is to have a particular risk issue held, at least predominantly, by the party best able to prevent, absorb or mitigate the consequences of that risk. The table below is an example of an EPC contract risk allocation matrix for a few of the key provisions of an EPC contract. The example risk allocation illustrated in Table 2.4 is not intended to be a definitive statement of industry practice but is representative of risk distribution that may occur on an EPC contract today. Other combinations of risk allocation and sharing can be supported, and each EPC contract will reflect the individual party’s risk tolerances which reflect the party’s experience, capabilities, history, size, corporate structure and applicable regulations.

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<td>Warranty</td>
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Table 2.4 – Example Risk Allocation

2.5 EPC Forms Contracts – Alignment of the Parties

In today’s nuclear renaissance, the EPC contract appears to be the most common form of project contracting structure. One of the key advantages for the owner is that an EPC structure provides for a single point of contact and simplifies project management coordination and facilitates a more prompt resolution of issues as there are fewer parties involved. On the supplier side of an EPC contract, the delivery team structure can take
many forms. A common structure is that the delivery team is comprised of a multi-party consortium with a reactor/plant designer and an AE/Constructor as the key participants. Other delivery team models include the use of partnerships or a prime contractor/subcontractor approach, and the choice of a specific structure is often driven by tax-related considerations. Other forms of new build contracting in use today are listed below.

1. Engineering and Procurement (EP) contract plus a separate construction contract
2. EPC to build, plus partial ownership (equity interest) by EPC contractor
3. EPC to build, plus partial ownership (equity interest) by EPC contractor, then EPC contractor sells equity interested after a pre-determined time frame.
4. EPC to build, plus partial ownership and EPC contractor operates
5. EPC plus partial ownership and EPC contractor operates for an agreed time, then EPC contractor sells equity interested and turns over operation of the facility.

Each of the above contracting forms engenders unique benefits, risks, and challenges that need to be addressed specifically in the contract. Every new build opportunity needs to be evaluated to determine what the optimal contract structure is for that application.

3. Conclusions

Relative to the first wave of nuclear build, today’s EPC contracts are justifiably more complex. Some of the factors leading to the need for greater complexity include increased regulatory and environmental requirements and associated reporting needs to various project stakeholders. EPC contracts performed in a regulated environment need to address the greater involvement and potentially unpredictable impacts from the local (state, region or country) regulatory agencies. Today, these agencies are showing greater oversight with respect to project cost overruns and schedule delays, resulting in a heightened concern from project owners that costs will not be recoverable unless they are deemed prudent by such agencies. Additional complexity is introduced as the manufacture and supply of equipment and materials are more broadly sourced through a global supply chain, while individual projects may desire a high percentage of local supply. Furthermore, the costs of projects today often exceed the capitalization of a single utility, making financing and cash flow critical, often resulting in a need for joint ownership. This aspect of joint ownership adds further complexity to the EPC contracting process and final terms and conditions. Another key point the project owners evaluate is how the nuclear power option compares to alternate forms of generation. Key issues include the consideration of possible government incentives such as loan guarantees or tax credits for technologies that promote clean or carbon free generation, versus a coal plant which may be penalized in an evaluation by the application of a carbon tax. These factors all point to different risk profiles for project owners and suppliers, resulting in more complex EPC contracts to put the risk impact where it can be best managed. All of these concerns lead to the need for managing more interfaces and establishing a greater certainty of the project bases than what was required in the first wave.

With these complexities and financial implications, EPC development and negotiation for a specific opportunity is not a quick process. It takes a great deal of time and substantial resource commitment by all parties. Our opportunity today is to capitalize on what we have learned to produce a new generation of reliable, efficient, environmentally sound, and safe nuclear power plants, and the development of a balanced EPC contract is a critical part of the overall project process.

4. References

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Plant operations
RADIOACTIVE SPENT ION-EXCHANGE RESINS CONDITIONING BY THE HOT SUPERCOMPACTION PROCESS AT TIHANGE NPP

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ABSTRACT

Besides safety precautions, the immobilization of spent ion-exchange resins requires special treatment and conditioning techniques to meet the acceptance criteria for disposal. Waste acceptance criteria define, among others, the quality of waste forms for disposal, and therefore will sometimes orient the choice of a specific process. In Belgium, for economical reasons, the Volume Reduction Factor (VRF) is a key criterion.

After Tractebel Engineering performed a technical and economical comparison of the industrially available systems, Tihange NPP decided to install a spent ion-exchange resins hot supercompaction unit. Tractebel Engineering is in charge of the project management and takes an active part in the process optimization.

With this process, the final matrix inside the pellets is a water free solid, but somewhat brittle, block. Volume Reduction Factors between 1.8 and 1.9 are achieved. Special care must be taken about the mortar quality if pellets are to be grouted.

1. Introduction

With the evolution of disposal facility acceptance criteria, it is now required that spent ion-exchange resins meet specific quality requirements prior to disposal. Where final disposal facilities exist, waste acceptance criteria define, among others, the quality of waste forms for disposal, and therefore will sometimes define appropriate treatment options. Basically, two of the main methods for the treatment of spent ion-exchange resins, following pre-treatment methods like dewatering, grinding, foaming or decontamination by activity stripping are [1]:

- Direct immobilization, producing a stable end product by using cement, bitumen, polymer or high integrity containers;
- The complete removal of the resin inner structural water by a thermal process followed by the supercompaction of the hot dried resins.

Part one of this paper will describe the principle of a reference resins hot supercompaction process. Part two of the paper will introduce the application of this process at Tihange NPP (Belgium) operating Pressurized Water Reactors (PWRs).
2. **Reference resins hot supercompaction process**

2.1 **Type of spent ion-exchange resins to be conditioned**

Hot supercompaction has been initially developed for a reference NPP with one Boiling Water Reactor (BWR) producing mostly spent powdered resins (condensate polishing) and one Pressurized Water Reactor (PWR) generating bead resins. The waste to be processed in the reference plant is a mixture of spent powdered resins (75–90%) and bead resins (10-25%).

2.2 **Treatment process**

Spent ion-exchange resins are first dewatered by a centrifuge (separator and decanter) system and filled in intermediate drums. After dewatering, the residual internal structural water inventory is about 50-60% of the resin total weight. Intermediate drums will be transferred to a thermal-oil heated, vacuum drying vessel where resins are poured by means of a special docking device.

In the drying and mixing unit, resins are heated to the necessary process temperature. After meeting the drying criteria, resins are discharged into special compactable steel drums, on which a lid is automatically placed. After these operations, the drums are immediately transferred to a high force compactor. Resulting pellets (compacted drums) are routed to a measuring unit, where the dose rate, height and weight are automatically measured and recorded.

The dried hot resin supercompaction unit (see Fig. 1) is remotely operated by a PLC (Program Logic Controller) system. There is thus no operator radiological exposure resulting from the unit operation.

![Diagram of Resins hot supercompaction installation for Tihange NPP](image)

2.3 **Process performances**

The Volume Reduction Factor (VRF = dewatered resins volume / volume of produced pellets) of the hot supercompaction process is very good and amounts to about 2.5. The overall VRF of the process depends on the overpack type. Some 3800 satisfactory pellets have been produced by the reference hot supercompaction plant [2].
The advantage of the process is that products suitable for final disposal will be generated and, at the same time, an important volume reduction for interim storage will be achieved. Moreover, with respect to the retrievability, the pellets can be very easily retrieved from the overpacks, or the overpacks themselves can be placed in larger final disposal packages.

3. Spent ion-exchange resins hot supercompaction process at Tihange NPP

3.1 Spent ion-exchange resins quantity and type

Tihange NPP, located in Belgium on the banks of the Meuse River, operates 3 PWRs with a power of about 1000 MWe each. Spent ion-exchange bead resins are 640 µm (mean wet particle size) and the average production amounts to some 8 m³/year.

3.2 Selection of the conditioning process

Until 2005, these spent ion-exchange resins were immobilized in an organic matrix (styrol, epoxy based materials) by use of a mobile unit provided by an external service supply company. The resins are immobilized in ONDRAF/NIRAS1 licensed overpacks (standard 400 litres drums), on the basis of one campaign (30-36 m³ of resins) every four years. The process overall VRF (= dewatered resins volume / volume of conditioned 400 litres drums) was found to be about 0.5.

Due to concerns linked to the high cost of the process, and to fire hazards, Tihange NPP requested from Tractebel Engineering in 2005 a complete survey and reassessment of the currently available and industrially proven spent ion-exchange resins conditioning processes with, as final aim, the recommendation of the best suited process taking into account the internal/external constraints prevailing at the plant.

Each surveyed process was assessed in accordance with the 6 following criteria: overall investment and operation cost, autonomy (fixed installation) versus dependence (mobile installation operated by external service companies), manpower requested qualification for the process implementation, qualification of the process, i.e. compliance of the end product with the ONDRAF/NIRAS Waste Acceptance Criteria (WAC), nuclear and industrial safety, industrial references.

For each process, each criterion was ranked from 1 (lowest) to 6 (best). A weighing factor was then attributed to each criterion, enabling to end up with an optimized proposal. Sensitivity analysis, including variations of the weighing factor numerical values, enabled the robustness of each process to be assessed against uncertainties.

This multi-criteria analysis recommended the installation of a fixed dried resin hot supercompaction unit, i.e. the by far best ranked process, under the constraints prevailing at Tihange NPP.

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1 ONDRAF/NIRAS: Belgian waste management agency, in charge of the collection, the conditioning, the interim storage and the final disposal of the radioactive waste produced in Belgium.
3.3 Process modifications

Tractebel Engineering and contractor Hansa Projekt Anlagentechnik (HPA) contributed both to set up and optimize the design of the new installation. With respect to the reference process, several evolutions have been developed in order to meet the constraints and standards at Tihange NPP. Solutions for concerns about springback, dust releases and grouting are described hereunder.

The most significant difference between Tihange NPP and the reference plant is the absence of powdered resins in the Tihange waste. A major impact of this absence is the excessive springback of the produced pellets.

However, hot supercompaction of mixed powder and bead resins in the reference plant does not give rise to detrimental springback, indicating that powdered resins play a key role in the process. A powdered additive is required to fill the gaps between beads, prevent excessive deformation and elastic springback, and bind both products together into a cohesive matrix.

In order to decrease the springback and to ensure the production of stable, reproducible, contamination free pellets, two options were successfully investigated (see Fig. 2):

- Addition of powdered ion-exchange resins (polystyrene - divinylbenzene based) in a proportion of 25-30% of the total mixed volume: this additive provides a much lower springback and finally satisfactory pellets. An interpretation of this successful difference is that ion exchanging functional groups seem to make polymers less compressible. The final matrix inside the pellets is a water free solid, but somewhat brittle, block;

- Use of double drums: placing the compactable drum inside a second, slightly larger drum, appears to provide clean, reproducible pellets, with a powdered additive proportion limited to 25% of the total waste volume. Without this safety wrapping, the additive proportion must be raised to at least 30% (lower VRF).

Fig. 2: (A) Double drum: the external drum is smaller in height, slightly larger in diameter, and ensures an efficient protection of the internal drum – (B) The addition of powdered ion-exchange resins and the use of a double drum design reduce the pellet springback when the piston of the supercompactor moves up
It must be pointed out that powdered resins, once dried, become a very thin dust (particle size ~ 50 µm) which can be somewhat radioactive after blending with spent beads inside the dryer. Hence, the following minimum precautions are mandatory to ensure the system operation is safe: good docking of the compactable drum below the conical dryer when the dried mixture is discharged, confinement of the discharge (drumming) and capping area with ventilation, assessment of dust explosion risks and subsequent preventive & protective measures.

According to waste acceptance criteria related to heterogeneous radioactive waste in Belgium, pellets containing dried, hot compacted resins must be inserted into standard 400 litres drums, cooled down and grouted.

Grouting tests have shown that the mortar recipe has a great influence on the compliance with waste acceptance criteria. A specific, ready made, premixed mortar (as used in sealing applications) was found to perform good grouting without seeping into the dried resin pellets.

4. Conclusions

Tractebel Engineering has initiated and is managing the construction of a new application of the hot supercompaction process in Belgium at Tihange NPP. Several developments were required to adapt the reference process and equipment to PWR spent ion-exchange bead resins and Belgian radioactive waste acceptance criteria.

In an environment of very limited space for interim storage and in the absence of an operating final repository site, or in the case of high final disposal costs, the hot supercompaction process enables to achieve a global Volume Reduction Factor (VRF) close to 1 (overpack included) for the interim storage instead of increased volumes observed with other currently available processes.

The hot supercompaction process produces a water free end product for interim storage (pellet), which can be retrieved and routed into alternative types of package later, if not initially grouted.

Moreover, the hot supercompaction process uses well proven standard technologies like drying and compaction and enables a flexible use of the system components for the supercompaction of other operational solid waste streams when not conducting resins conditioning campaigns.

5. References


PRESENTATION OF CEA MARCOULE LABORATORY DEPARTMENT

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ABSTRACT

The CEA Marcoule Laboratory Department initially dedicated to production support, has recently been renovated to perform a wide range of analyses to support dismantling of plant operation and process development units. It provides services to the operators of the CEA Marcoule Research Center in the fields of analytical chemistry, metallographic examinations, radioactivity measurements, in situ or online nuclear measurements, decontamination processes and industrial chemistry studies.

The facilities are designed to analyze all types of nuclear industrial and research samples both under inactive conditions and in glove box and hot cell environments. The Marcoule Laboratory today combines the skills of CEA researchers and AREVA NC analysts in support of production and R&D, process development and dismantling activities at Marcoule or other nuclear sites.

1. Historical Context

The Marcoule Laboratory Department was established in 1956 at the beginning of the site which was the birthplace of plutonium production in France. The laboratory was commissioned in 1958, in support of the Marcoule fuel reprocessing plant (Usine de Plutonium N°1 called UP1) and auxiliary units including the radioactive effluents treatment plant, the solid waste treatment plant.

Its routine activity involved two or three 8 hour shifts related to quality and process control. Three extensions were added to the initial facility building between 1960 and 1986 to meet the new needs and expectations of the site. The range of activities included the following:

- Online process control (chemical and radiological) via a pneumatic transfer network between plutonium production units and the Laboratory.
- Quality control on raw materials and finished products.

Since the shut down of UP1 in 1997, the Laboratory made some evolutions of its activities; they were shifted towards support dismantling activities (i.e. UNGG reactors), beginning with the removal of radioactive material and rinsing of the process loops. These activities require large numbers of samples with chemical preparation followed by chemical and physical characterizations. The analysis results obtained in this step are fundamental for further waste treatments to guarantee safe decommissioning operations and waste decategorization. The treatment and conditioning of small quantities of exotic radioactive wastes such as organic effluents or highly contaminated wastes also began at that time.

In addition to this huge analytical demand, the laboratory has been reconfigured at the same time in term of safety taking into account the new rules (i.e. fire and earthquake resistance) and in terms of capabilities to support research & development and dismantling.

This renovation will continue until 2010 with final approval by the safety authority in 2011. This paper describes the CEA Marcoule Laboratory Department.
2. Ressources and Capacities

The Laboratory Department facilities are currently composed of 5 interconnected building wings covering a total ground area of about 4200 m² (fig. 1):

The Laboratory is divided into:
- Unrestricted access zones dedicated for non nuclear areas including the locker room, storage areas, yards and offices (1700 m²).
- Restricted access zone, the workplaces are distributed over 95 fume cupboards, 40 glove boxes and 35 shielded cells (2500 m²).

Fig. 1: schematic view of the Laboratory building

The Laboratory Department is operated by AREVA NC/Nuclear Site Value Development Business Unit under CEA/Nuclear Energy Division responsibility. It is staffed by 109 persons: 96 persons from AREVA NC (working 3X8 and 2X8 hour shifts) and 13 persons from the CEA.

Since 2008, 2 CEA teams have joined the Laboratory because AREVA NC and CEA intend to consolidate and reinforce the analytical and process competences of the Site in a same location in order to promote dynamic synergies between the teams. This cooperation could contribute to meeting the challenge and expectations in the future for Marcoule or other nuclear industrial plants and research centers.

The Laboratory Department is composed of:
- AREVA NC contribution
  - Central management entity (management, safety, security, quality, environment unit): 7 persons
  - General Service (Utilities, Radioprotection): 19 persons
  - Analytical Chemistry Laboratory: 31 persons
  - Nuclear Methods and Measurement Design Laboratory: 10 persons
  - Industrial Chemistry and Projects Laboratory: 15 persons
  - Radioactive Measurement Laboratory: 15 persons

- CEA contribution
  - Metallography and Chemical Analysis Laboratory: 10 persons
  - Advanced Decontamination Process Laboratory: 3 persons
The analytical and study capacities of the laboratory could be illustrated by the following data for 2008: treatment of 5000 samples processes by 23550 analytical determinations requiring 95000 working hours. About 1/3 of this time is dedicated to the studies (31600 working hours).

3. Missions

With the renovation and upgrade program, the Laboratory Department is able to perform a wide range of analyses to support dismantling, plant operation and process development activities, combining the skills of CEA researchers and AREVA NC analysts. The Laboratory Department has an important technical and analytical means with competent personnel and is organized to respond to the following integrated production processes:

- **Full Service laboratory function**
  - Performance of analysis programs.
  - Multiple clients and diverse analytical activities: CENTRACO, MELOX, Phenix, EDF, etc…
  - Identification of complex samples (poorly controlled processes).
  - Task organization, response times, priority management.
  - Communication of results in various forms.
  - Complex approval of results (allowing for historical background).
  - Preparation of cost estimates, cost allocation and invoicing.

- **Production quality control laboratory function**
  - Regular sampling plans per unit, per sampling point, per campaign, etc.
  - Allowance for process operation (3X8-2X8 hour shifts, short response time, real-time communication with various contacts).
  - Statistical control of the analytical process.

- **Research and development laboratory function**
  - Multiple client profiles (from research to industrial development).
  - Projects, studies and characterization development.
  - Possible interpretation of requests for services.
  - Analysis process not predefined and subject to dynamic modification.
  - Insufficiently documented tests.
  - Tracking and long-term interim storage of aliquot samples.
  - Complex consolidation of results.
  - Incorporation of results in reports.
  - Complicated scheduling.
  - International collaborations (Europe, US).

In technical and scientific fields, the laboratory department is active in analytical chemistry, metallographic examinations, radioactivity measurements, *in situ* or online nuclear measurements, decontamination processes and industrial chemistry studies to meet the following analysis requirements:

- Characterization of raw materials, nuclear waste and finished products.
- Process control in continuous operation.
- Testing and examination in support to R & D programs.
- Treatment and conditioning of small quantities of exotic waste with high alpha and/or beta contamination.
- Fabrication and supply of working standards and reference materials.
- *In situ* nuclear characterization.
- Characterization and expert examination.

Studies are carried out primarily in response to requests for:

- Assistance to plant operators concerning specific process-related problems.
- Pilot testing to adapt new or redesigned processes.
- Participation in programs of general interest.
- Treatment of liquid and solid waste rich in alpha and/or beta emitters.
Decontamination of liquid effluents at prototype scale.

The Laboratory Department provides Studies and Analysis for many customers (mainly in the nuclear fields) including the CEA, AREVA NC, AREVA NP, MELOX, EDF, COMURHEX, and SOCODEI. It is also involved in collaboration with numerous academics partners (Universities of Montpellier, Barcelona, Bordeaux, Limoges, Nancy, CNRS), Nuclear Research Centers (ITU, PSI, SCK/CEN, ORNL) and industrial partners (CANBERRA, THERMO ARL) to develop instrumentation devices and analytical operating procedures.

The expertise of the Department is well recognized by i) the presence of 3 senior experts (instrumentation and nuclear analyses), ii) 3 international experts ISO TC 85/WG5 and ENTRAP/WGA and iii) two AFNOR-BNEN national experts in nuclear waste fields who participates in various standardization working groups.

Finally, many analytical specialists belong to 8 working groups of the CETAMA, the CEA / AREVA organization devoted to the promotion of good practices and method validation in analyses for nuclear needs. By it ISO 9001 certification, the CETAMA (Commission d'ETablissement des Methodes d'Analyses) allows to validate the quality of analytical methods by i) using prescribed method developed by the members of working group, ii) using reference certified materials and iii) participating to inter-laboratory comparisons to test regularly the analytical capabilities of the teams.

4. Analytical and study Fields

The analytical and study activity is divided into 5 topics:

4.1. Analytical Chemistry :

This activity covers a wide range of capabilities in analytical chemistry in support of research, development and dismantling applications:

- elemental analysis of cations and toxic elements,
- specific analyses: CN⁻, Cl⁻, F⁻, NO₂⁻, NO₃⁻, SO₄²⁻, PO₄³⁻, C₂O₄²⁻, formates, acetates, TBP, DBP, MBP, etc…
- Conventional chemical analysis: density, acidity, pH measurement, etc…
- Preparation and characterization: measurement standards, reactants.
- Pretreatment and preparation of samples before analysis: mineralization, dissolution (bitumen, concrete, etc.).
- Development of analytical procedures.
  - New XFR quantitative methods applied for multielement nuclear glass.
  - Characterization of the evolution of TBP under α irradiation.
- Waste and decontamination.

The analyses are performed by many devices in various environments:

- Monitored zone in conventional working conditions:
  - X-ray fluorescence (x 2).
  - ICP-AES (x 4).
  - ICP-MS (x 2) (fig.2).
  - C/S analyzer. Atomic absorption spectroscopy.
  - GC-MS and µ GC (x2).
  - Optical Asbestos identification.
  - Ignition point, flash point.
  - Liquid chromatography.
  - Ion chromatography.
  - FTIR.
Fig. 2: ICP-MS in conventional working environment

- Fume cupboard environments:
  - ICP-AES.
  - Liquid chromatography.
  - Atomic absorption spectroscopy.
  - UV-visible spectrometry.
  - Mineralization, dissolution.
  - TOC analyzer

- Glove box environments:
  - ICP-AES.
  - C/S analyzer.
  - UV-visible spectroscopy.
  - Particle size analysis.
  - Mineralization, dissolution (fig.3).
  - Liquid chromatography, GC-FID
  - XRD

Fig. 3: Mineralization in glove box

- Shielded line:
  - ICP-AES (fig.4).
  - Liquid chromatography.
  - UV-visible spectroscopy.
  - $\gamma$ Spectrometry, specific electrodes, etc...
  - Mineralization.
  - Balance, microwave oven, density, etc…
4.2. Metallographic Characterization

This activity provides expertise and characterization of materials at local scale (about one square micrometer) complementary with more global chemical analyses more global. It supports research, development and industrial studies and projects concerning Gen IV fuel and structural applications, front end:

- Quality control of materials, fuels (UO$_2$, USi, UMo, UCe) and components as received (bulk, deposit, powder): microstructure, adherence, homogeneity, welding.
- Examination of materials (steel, ceramic, glass, polymer, composite) after corrosion or aging tests: intergranular corrosion, reactivity mechanism by gas, liquid metal, molten salts or Uranium materials as UC, UN.
- Examination of industrial components: failure, corrosion by UF$_6$, Cl$_2$, CFC, HF.

The metallographic characterizations are performed by many devices in monitored zone in conventional working conditions:

- Metallographic preparation (cutting, polishing, etc…).
- Optical Microscopy: image and image analysis.
- SEM-FEG/EDS (fig.5) and SEM-low vacuum/EDS: image and local chemical analysis.
- Electron Microprobe (EPMA): quantitative chemical analysis.
- Macro-microhardness.
4.3. Industrial Chemistry and Project

This activity is dedicated to the analytical development and control in support of industrial processes. It has a strong interaction with the analytical units in the Department. The variety of the samples leads to work in 3 types of workstations:

- ventilated hoods for low contaminating samples
- glove boxes for contaminating and low irradiating samples
- shielded cells for the most active samples (fig. 6)

Fig. 6: sample preparation in hot cell environment

This activity is supported by its own analytical devices in various environments:

- Characterization of active samples:
  - Sampling : tools design and assistance to the operator
  - Macroscopic examination in shielded cells
  - Mechanical treatment (cutting, grinding, sieving…)
  - Chemical treatment (acidic dissolution, purification, separation…)
  - Specific analytical means (μcalorimetry, DTA/MS-DSC, GC)
  - Physical characterization (densities, water rate, graphite rate…)
  - Explanation and reporting of the results to the customer
- Back-up to Marcoules's nuclear facilities
  - Study and validation of new chemical processes
  - Recovery and recycling of Pu
  - Co-precipitation and reactivity tests for acceptability of big quantities of liquid effluents in Marcoule's Treatment Station
- Cement encapsulation
  - Validation of the formulation
  - Quantification of gas releases (GC)
- Destruction of active organic matter :
  - Supercritical water oxidation process (fig. 7)
  - Treatment of Pu-charged resins through silver(II)-electrodissolution and oxalic conversion.
- Specific work :
  - Dissolution of a irradiated UNGG fuel element
  - Qualification of a nuclearized analytical device (LIBS) for future in situ measurements
  - Decontamination of debased organic solvents
This activity also runs the laboratory of Marcoule’s pilot facility and is in charge of the analytical support to the dismantling of this building. Many up-to-date analytical means are settled
- in glove boxes: X-ray fluorescence, LC, potentiometry, acid-base titration, overall $\alpha$ and $\gamma$ counting, COD, foam testing, GC/FID…
- in shielded cells: collimated $\gamma$ spectrometry, spectrophotometry, LC, density measurement…

4.4. Radioactive measurements

This activity is dedicated to isotopic analysis, quantification of alpha, beta and gamma emitters;
- Uranium-plutonium balance.
- Isotopic compositions: U, Pu, B and Li.
- Determination of beta emitters: $^3$H, $^{14}$C, $^{36}$Cl, $^{55}$Fe, $^{59}$Ni, $^{63}$Ni, $^{90}$Sr, $^{94}$Nb, $^{99}$Tc, $^{129}$I, $^{161}$Sm.
- Fabrication isotopic standards.
- Development of specific measurements (e.g. $^{99}$Tc in bitumen-encapsulated waste, $^{129}$I by ICP-MS).

This activity is supported by its own analytical devices in various environments:
- 2 thermal ionization mass spectrometers.
- 2 liquid scintillators
- 6 gamma spectrometers including one with sample changer.
- 7 alpha spectrometers with grid chamber detectors.
- 16 alpha spectrometers with semiconductor detectors.
- 1 ICP MS in fume cupboard (fig. 8).
4.5. Nuclear methods and conception design

This activity is dedicated to in situ radiological characterization of used equipment items (glove boxes, tanks, crushing machine, etc…), decontamination and dismantling monitoring of industrial set up and plants, waste packages. A specific methodology has been developed at Marcoule adapted to industrial expectations (more particularly in UP1) since 1998 [1]. This topic was developed because the conventional radiological approach was not appropriate to the characterization of industrial devices due to large dimensions and complex shapes. So, if “samples” couldn’t be analysed in the laboratory, the laboratory has to find a solution to make measurements in situ on the site before dismantling.

This approach could be broken down as follows:

- In situ initial radiological survey of complex configurations (fig.9 and 10).
- Monitoring of dismantling operations
- Local characterization of the waste
- Modelling of the residual activity taking into account of the real geometry of the pieces to assess the dismantling strategy (fig. 11) in term of costs, schedule and waste specifications
- Estimation of the Pu mass distribution with a correct uncertainty (1g up to 100g of Pu)
- Characterization of historique waste not well characterized
- Assessment of waste package management.
Fig. 10: Hot spots localisation in a used evaporator: superimposed images of visible-light image and gamma image.

Fig. 11: modelling of the evaporator to evaluate the residual activity

The analyses are performed using portable devices and spectral simulation software:
- Gamma camera (CARTOGAM).
- Four portable Ge gamma spectrometry systems (coupled with ISOCS and PASCALYS software).
- Six portable Cd-Te gamma spectrometry systems (coupled with PASCALYS).
- NaI portable probe for fast hot spot survey
- Two dose rate measurement systems for pipes and other configurations.
- MERCURAD dose rate modelling software useful for dismantling scenario.

The specificity of this team is to group into a same entity most of the competences and know-how (radiologic and chemical measurements, modelling) needed to the monitoring of dismantling of industrial plant (about 100 equipments with Pu contamination treated at that time).

4.6. Example of the interest of the laboratory in the decommissioning and dismantling of UP1/Mar 200

A major program is in progress at Marcoule to dismantle the first French defence reprocessing complex UP1. This complex has been commissioned at the end of 50’s and operated for
40 years. This extensive dismantling and waste recovery program must therefore cope with a large variety of waste and radiological situations. In many cases, operating events and poor documentation lead to a strong need of characterization activities prior to the studies or initiation of the dismantling operations. Those characterization programs are supported by the tools, analyses and expertise of the Marcoule Laboratory Department. Most of the collected data are used in the safety reports and the measurement methods, calibration and qualification techniques are part of the licensing process of each dismantling or recovery operation.

**Example of the MAR200 dissolution unit:**

One of the examples that best illustrates this program is the cleanup and dismantling of the continuous dissolvers located in Building 117 of the UP1 complex. The building contained two parallel dissolution lines (A and B).

For the line A dissolver, the initial analytical measurements revealed the lowest level of fissile material content in the deposits; these results allow to define the filtration parameters and to confirm the possibility of incorporating of the residues in the standard glass containment matrix. Based on these results and data, the safety report was approved. The recovery of the residues is now completed and the dismantling will start in 2010. All the residues have been completely vitrified.

For the line B equipment, preliminary inspection confirmed the existence of sludge and deposits at the bottom part of the dissolver. The in cell counters confirmed the presence of Pu and provided a rough estimate of the activity. A specific program to collect of 18 active samples was then defined and carried out.

Difficulties appeared in the sample collection as the sludge was located under the bottom support plate of the vessel. This perforated plate was about one inch thick and the preliminary rinsing test showed that the residues were partially blocked under the plate. In order to ensure fully representative sampling, the plate was removed by plasma cutting. The results obtained by the laboratory were taken into account on line for the definition of the interventions and the corresponding safety reports. The assessment of Pu content and its chemical and physical forms were of prime importance for defining of the sludge recovery technique. They were also considered in the safety report prepared for the future elimination route for the resulting waste. The analytical laboratory provided the project with more than 200 of physical, chemical and radiological data items under approved Q.A. conditions delivered in acceptable time. The required analyses included $\alpha$, $\beta$, $\gamma$ spectroscopy, chemical analysis of U and Pu, quantification of hydrogen in Pu, Pu isotopic composition, $^{90}\text{Sr}$ measurement, graphite quantification, chemical analysis of cations, anions, acetate, formiate as Pb, B, Ni, Cr total, Cr VI, As, Sb, Hg, Be, Se, Cd, CN.

The wide variety of available analytical techniques in the Marcoule laboratory was necessary to address the uncertainties on the sludge content, chemical composition and physical behavior under transfer and filtration conditions. Based on these results the recovery and filtration of the sludge is currently in progress.

Analytical techniques of the Marcoule Laboratory are also used for on line monitoring of D&D operations. In such cases, they mostly contribute to the waste characterization and elimination process. For example, such techniques have been successfully implemented for sorting metallic waste previously stored in pits. The low level counting equipment was installed on site and operated remotely. This allowed a direct and qualified characterization of the waste as LLW (low level waste), the waste were grouted on line without no additional handling or transfer.

**5. Future prospects**

The high technical potential and important analytical capacity of Marcoule Laboratory Department in the field of analyses, examinations and studies could now be proposed to other nuclear entities. CEA and ANC are now preparing the future. The goal is to propose a service, based on the capacity of laboratory, the specific competence and the wide acquired experience during the life of a nuclear site of Marcoule since the fifties, to provid for the need of nuclear industry.
6. Conclusions

This presentation has described the high technical potential and important analytical capacity of Marcoule Laboratory Department in the field of analyses, examinations and studies. It allows to treat analytical topics between laboratory to industrial scale. They were initially developed for Marcoule activities but these activities could now be proposed to other nuclear entities.

7. References

EDF-R&D’s concept for using 2D drawings as an EAM/EOM solution to create, modify and visualize operation data

First prototype developed using Ventyx EAM/EOM solutions

Over the years, electronic technology has played a very important role in the area of plant maintenance and operation. Enterprise Asset Management (EAM) and Enterprise Operation Management (EOM) have been developed and enhanced to improve plant personnel efficiency, productivity and safety.

At the same time, CAD technology has been commonly used in plant maintenance and operation processes mostly in the area of design. In nuclear power plants, 2D CAD drawings (P&ID, electrical, mechanical, etc.) have long been used to represent not only the future plant configuration but also the "as built" plant configuration.

Over the last few years the opportunity to combine the two technologies (EAM/EOM and CAD) has emerged. The question regarding the benefits of displaying and/or updating EAM/EOM data such as clearances/Tagout/lockout, alignment checklist, radiation areas, scaffolding, chemical risks, …in a graphical format using 2D or 3D CAD drawings has often been asked. This is why EDF R&D and Ventyx have collaborated in developing a prototype that allows plant personnel to graphically display, create, and modify alignment procedures and clearances using an EAM (Asset Suite) and EOM (eSOMS) software.

This paper describes some of the methodology as well as the tools used to develop such a prototype.

1. CAD Drawings

1.1. Existing drawings versus new drawings

In a lot of cases, CAD drawings (P&ID, electrical diagrams, etc …) are already available. The question then becomes: should we use the existing CAD drawings or should we redevelop new sets of CAD drawings? The answer to that question will depend mainly on the following criteria:

- What is the format of the existing CAD drawings (P&ID, Electrical, diagrams, etc …)?
- Is it more economical to convert/update the existing drawings or to redeveloped new CAD drawings?

Either option should be viable.

1.2. A generic 2D CAD format

At EDF rather than transferring existing CAD drawings directly into the NGC format, we have decided to first replace a part of the existing out of date CAD system with a more up to date system (NGC Kernel). Our engineering division has opted to re-use existing CAD drawings and import them into the new solution. The new format is a free XML format, called NGC (Noyeau Generique de CAO). The structure of the XML NGC file is composed of two sections:

- The first section includes the definition of the business objects and all their relationships, for example the pipes and the valves as well as the link between the two,
- The second section includes the definition of the graphical elements (primitives) representing the visible part of the drawings (curves, lines, etc…).
This format is completely generic and therefore independent of the CAD software used.

With XML transformations or file format transformations, it is also possible to recover specific drawings and export them in a “kind of” NGC format. Then, the developed concept is not only applicable to EDF, it is also valid for other companies when they use published CAD software formats.

See diagram below cons.

The advantage of the EDF approach is that the solution is very generic independent of the type of drawings (mechanical, electrical …) and the type of format which makes the solution more adaptable to different type of business and also more economic. Most other CAD software do not allow interface between EAM/EOM databases and drawings.

1.3. Integration between EAM and CAD software

Since each business object is part of both the EAM and the drawings, it is now possible to integrate CAD drawings with EAM. As an example, a valve, a switch, a pump, a pipe, etc. are displayed on drawings as an object, but also displayed in the EAM/EOM as a record, the two being linked. Therefore, the link between the CAD world and the database world can be easily established.

2. The EDF solution

An important area of improvement is the ability to view or modify operation and or maintenance processes (alignment checklists, clearances, radiations zones, scaffoldings, chemical risks, etc.) directly on 2D drawings.

Plant operation and/or maintenance data (work orders, tag out, etc.) are generally stored and managed in an EAM/EOM database. The idea would be to develop a 2D “plug in” that would enable viewing and/or modifying the operation/maintenance data (at each steps of the workflow) from the EAM/EOM using the 2D drawing.

At EDF R&D, we have demonstrated that such a concept is both valuable and feasible. We have developed a prototype that enables users to graphically modify alignment checklists or tag-out created in Asset Suite or eSOMS directly from a drawing. From Asset Suite or eSOMS, a user has an option to access and display a graphical representation of the tag-out or alignment checklist. Once graphically displayed, the tag-out or checklist can then be modified directly onto the drawing. Once done, the user can then return to Asset Suite or eSOMS and any modifications made on the drawing are automatically transferred to Asset Suite or eSOMS.

The figure below shows how using the “graphical plug-in” a user can switch from Asset Suite to the drawing. An online help option will guide the user on how to use the 2D drawing interface as well as understand the type of data required.
To create a line-up procedure with the EAM and to manage its life cycle (1D)

To graphically modify the line-up procedure

To follow the workflow process

The graphical application contains three different views:

- a 2D graphical view of the piping and equipment along with the equipment position. Colors, Shapes and codes are used to indicate critical information such as the equipment position, positioning sequence, pipe highlight, etc ... Using that view, the user can modify the equipment position, and/or sequence directly on the drawing.
- a tabular view similar to the one in the EAM/EOM listing the equipment/required position.
- a tree view indicating the equipment positioning sequence when executing the checklist or the Tagout.

1.4 Future Enhancements

At EDF R&D we are looking at extending this concept to other areas such as I&C system in order to help with pre job-briefing activities or facilitate fields periodic testing, etc. We also plan to link this technology with maintenance and/or operation activities for better scheduling.

In the long term we plan to not only use 2D but also 3D representations of the EAM/EOM data. The goal is to give users a better and easier tool for EAM/EOM visualization.

In general, combining EAM/EOM data onto a graphical display can be useful in verifying and validating data and could be of great help for plant operation and maintenance personnel.
2. The Solution Architecture

2.1 A baseline of generic functions

In order to easily promote new operation tools using 2D drawing and operation data, a specific software architecture has been developed.

The first two applications are a generic module of drawing visualization and a module dedicated to alignment procedures and clearance design.

We also planned to develop new applications for the enhancement of plant room drawings (e.g., for radioprotection, maintenance, risks management, etc.)

The “foundation” of this architecture is named GEL (Graphical Enhancement Library).

Each application based on GEL has to manage its interactions with data input or outputs. For example, the alignment procedures and clearances module manages its interface with an EAM/EOM (or potentially with any other source of data) and with a 2D documents server.

GEL allows several common services to the final applications. GEL:

- reads XML NGC files,
- transforms the XML files into SVG format for visualization,
- transforms the XML files into a graph of business objects,
- permits visualization,
- permits interaction with graphical primitives to reach business objects,
- offers a set of primitives for graphical enhancement,
- enables users to export enhanced drawings (image file, SVG, PDF).

2.2 Usage of open source software

GEL and the applications developed on this baseline are massively using Java programming. The main libraries uses are:

- Apache BATIK: SVG engine (Apache license),
- JBIX: XML binding (BSD license),
- Apache CXF: web services for interacting with Asset-Suite (apache license).

2.3 Easy deployment of new applications

All the applications based on GEL (Java) can be deployed easily, without specific installation.

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1 The 2D documents can also be stored in the database of the EAM.
on each computer of the power plant. Only a web browser is needed. This is enabled by the use of Java Web Start. 

Java Web Start is a framework developed by Sun Microsystems that allows users to start application software for the Java Platform directly from the Internet using a web browser [Wikipedia]. Java Web Start maintains security and integrity of the deployed software.

2.4 Other advantages of using XML technologies

2.4.1 Bridging the gap between design and operation

The choice of using XML for 2D CAD usage, both for design and operation, has been done by EDF. The expected benefits are a long term maintenance for all applications, the possibility to bridge the gap between design and operation with 2D tools using massively XML, easy applications modification, data verification, data transformation, drawing migration, etc.

2.4.2 Other usages in (non real time…) control rooms

This same CAD approach can be reused in other areas where data from different system has to be exchanged between users. This is particularly true in modern control rooms.

For example, outage centers have to consolidate date from the control process, from the EAM, from the EOM, from the schedule, etc. One of today’s limitation and challenge is the capability to merge heterogeneous data from different systems, not always designed to work together in a same environment. The XML solution can provide a solution in not only facilitating the exchange of data but also and facilitating their dissemination.

In the oil and gas industry, the concept of operating center has been widely used. In such centers, many different groups have their own private offices however a centralized structure allows all experts point of views to be merged and discussed. As an example, such operation centers could bring together the expertise of such groups as the drilling group, the well management group, the geology group, the transportation group, etc. Today, one major challenge is how to improvement the exchange of information/data from heterogeneous inputs. Once again, XML can solve that challenge.

2.4.3 To formalize knowledge

Nuclear power plants were built to last for several decades. Knowledge retention is a very important part of the life of the plant. One solution consists of incorporating as much as possible the design data into the tools.

As the concepts of object modeling can be represented in XML, then the idea is to formalize as much as possible these objects within or outside the applications. XML can then allow facilitating the transformation of an object model (i.e. in UML) in XML files representing the code of applications. In other words, it could facilitate the migration from technical requirements to an applicative code.

3. Working in a rapid application design model

2.1. Working with end users

The first R&D prototype was developed in our R&D lab at Chatou (Paris) over a period of
about six months. Continuous improvements were made during that period based on the feedback of end user testing. Clearances managers as well as field operators were used during that period for the testing. After the testing period, it was decided to move the prototype to one site and see how it would perform under real conditions. The Penly Nuclear Power Plant (1300 MW) near Dieppe (North of France) was selected as the pilot project. Today the prototype has been deployed and is being used by the Penly plant personnel (mostly clearance managers and field operators) to create/modify/visualize clearances and alignment checklist using the graphical interface. Alignment checklists have been graphically created for eight elementary systems.

2.2. Working with the software supplier

An important effort has been to standardize the data exchange interface between the R&D prototype and the Ventyx software in order to be able to interact with both the Ventyx EAM (Asset Suite Tagout) and EOM (eSOMS Checklist) solutions. Web Services have been developed and used with Asset Suite. However the interface with eSOMS is based on an XML file. We expect to reuse the Asset Suite Web Services in future development of the Web Version of the eSOMS interface.

The development and improvement of the prototype is continuing this year. Additional interfaces are currently being developed that will bring together clearances and alignment checklist into one graphical view. Using Asset Suite, all conflicts will be resolved and a final validation will be done using the graphical visualization.

3. Short conclusion

The main advantage of this type of solution is that all existing workflows are unchanged and that the quality assurance is maintained. This solution is just an operation support to better understand and visualize many of the power plant processes.
IMPLEMENTATION OF DESIGN AND CONSTRUCTION OF AN INCINERATION UNIT FOR LOW LEVEL RADIOACTIVE WASTE

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ABSTRACT

In the frame of the G8 Global Partnership Against the Spread of Weapons and Materials of Mass Destruction, France represented by the Atomic Energy Commission (CEA), has decided to provide the Russian shipyard Zvyozdochka, located in Severodvinsk (Archangelsk Oblast), a new incineration unit for combustible low level radioactive waste (LLRW) arising from maintenance and dismantling of nuclear submarines. CEA asked AREVA TA to implement the project. In its first part, the project organization involving Russian and French partners and work sharing are briefly shown.

The paper describes the design phase which started in 2007, after a comprehensive engineering and radiological survey of the existing building in which the incineration unit was expected to be installed and operated.

The incineration unit comprises the incineration system itself (oven, post-combustion, ashes collection system), dry off-gas treatment, and HEPA filtration process. A cementation process aimed to solidify radioactive ashes with a cement matrix in 200 l metal drums is part of the incineration unit. The design also included selection and definition of radiation protection and monitoring equipment, I&C and unit supervision. The unit is designed to incinerate 25 kg of LLRW per hour.

AREVA TA developed a set of safety functional requirements and dispositions associated with nuclear safety, radiation protection for the workers and the public and protection of the environment. A Safety Options File gathered and developed the main safety requirements to be considered for the unit design, based on French principles and practices, with consideration given to Russian regulations and practices, as well as waste acceptance criteria for the incineration unit, e.g. limits in water and halogens content in combustible waste. Operational functional requirements were defined.

Specific constraints were set by the Russian side. The building where the incineration unit will be installed and operated already existed
and was to be refurbished and its layout led to select specific design options.

At the end of the detailed design phase, including the safety analysis report and the environmental impact assessment of the incineration facility, a comprehensive file was submitted to the Russian Safety and Certification bodies for expertise. Approval of this file made it possible to launch the construction phase of the several parts of the incineration unit which was completed fall 2008.

In December 2008, a delegation of Russian experts came to France for certification of the incineration unit necessary before its transportation to Zvyovdochka shipyard. This certification was successfully completed and the unit was transported to Zvyozdochka.

Installation of the incineration unit in the building is completed and final test program is presently ongoing, this incineration unit will contribute to protect the environment in reducing the accumulation of solid waste produced by the important dismantling operations carried out in the ZVEZDOTCHKA Shipyard.

Introduction: the G8 Global Partnership
The Global Partnership against the Spread of Weapons and Materials of Mass Destruction was launched by the heads of State and Government at the G8 summit held in Kananaskis in June 2002. The aim of this partnership is to "prevent terrorists, or those who harbour them, from acquiring or developing nuclear, chemical radiological and biological weapons, missiles, and related materials, equipment and technology."

Within the framework of the Partnership, the participants have agreed to support cooperation projects, starting with Russia, to promote non-proliferation, disarmament, nuclear safety and the fight against terrorism. The destruction of chemical weapons, the dismantling of decommissioned nuclear submarines, the disposal of fissile materials and the redirection of former weapons scientists are among the priority concerns expressed.

In an agreement signed on 18th November 2003 and renewed on 16th March 2009, the French Government gave CEA responsibility for running the bilateral cooperation actions that fall within the framework of French participation in the G8 Global Partnership.
Rosatom is the representative of the Russian Federation for bilateral projects implemented in the frame of the G8 Global partnership.
The project of renovation of an Incineration Unit for Low Level Radioactive Waste at “Zvyozdotchka” shipyard was one of the projects proposed by France to the Russian Federation in the frame of the nuclear window of the G8 Global Partnership.
Renovation of an Incineration Unit for Low Level Radioactive Waste at Zvyozdotchka shipyard

This project consists of renovating the incinerator for solid nuclear waste located on the premises of the Zvyozdotchka shipyard.

The Zvyozdotchka Federal State Unitary Enterprise was created in June 1946. It is situated on the Yagri Island in Severodvinsk (35 km to the west of Arkhangelsk). Zvyozdotchka is a leading Russian shipyard specializing in repairs and refitting of submarines, surface ships and vessels of any class and purpose. In recent years, the shipyard started building trawlers, shelf-hoisting float drilling rigs for oil and gas production, manufacturing of ships propellers and launching boards for the space industry. On the administrative standpoint, Zvyozdotchka depends on Rosprom, the Federal Agency on Industry.

A first incineration unit was used by Zvyozdotchka in the 90’s but its conditions deteriorated. This installation was shut down more than 12 years ago. Once the renovation of the incinerator would be completed, the processing of solid radioactive waste at the Zvyozdotchka site, which is a bottleneck in dismantling operations, will be improved.

Preliminary study: diagnosis of the existing incineration unit

A preliminary study was conducted by AREVA TA in 2004 and 2005. It consisted of performing an engineered and radiological survey of the existing incineration unit and the hoist building (building n° 163). It aimed also at defining the conditions for renovating the incinerator as part of the free technical assistance being provided by France to the Russian Federation. In particular, the preliminary study made it possible:

- To diagnose why the former incinerator was not working properly and had to be shut down (waste feeding, recovery of ashes, instrumentation and control);
- To determine the current needs of the Zvyozdotchka shipyard in terms of types and characteristics of solid waste that would be incinerated;
- To implement the Russian statutory Declaration of Intent (DON) drawn up and approved prior to launch the project.
Conclusions of the preliminary study were as follows (see pictures 1 and 2):
- No equipment of the existing incineration facility could be reused;
- Foundations and civil engineering of the building n°163 which Zvyozdotchka wanted to be reused for the new incineration facility was in good conditions. This building would need refurbishment, adaptations to host the new facilities and connection to auxiliary networks. Its internal circular geometry on two levels and its environment have led to select some specific design and implantation options.
- Thus, a new incineration unit and a cementation unit for conditioning of ashes would have to be designed, built and installed in the refurbished building.

A contract was granted to AREVA TA by CEA for the whole project. AREVA TA subcontracted to Zvyozdotchka for the refurbishment of the building, including its nuclear ventilation, the connection to utilities and auxiliary networks and supply of radiation protection and monitoring equipment.

As a result, Zvyozdotchka was both Beneficiary of the French Technical Assistance and subcontractor of AREVA TA. In order to define the role and responsibilities of each actor during the different phases of the project, a Convention (so called ‘Tripartite Convention’) was signed by Zvyozdotchka, CEA and AREVA TA in November 2006. This convention also provided precisions about expected
performances of the incineration unit, applicable safety principles and input data regarding characteristics of radioactive waste to be incinerated, which would serve as a basis for the design studies.

**Characteristics of waste and expected performances of the incineration unit**

As defined during the preliminary study performed in 2005 and confirmed in the abovementioned Convention, the characteristics of the waste to be incinerated, generated during maintenance and dismantling of nuclear submarines, are the following:

- wood: 30 to 40% w/w
- textile: clothes, tissues, paper scrap: 50 to 60% w/w
- plastic and rubber (shoes): 10% maximum
- water content: average 10%, 30% maximum
- chlorine (halogen) content < 1% w/w
- estimated lower calorific power: 15800 kJ/kg

According to Zvyozdotchka, incineration of liquid waste was out of scope of the project.

Waste will be conditioned in cardboard or paper bags, waterproof and sealed, with a maximum weight of 5 kg. Waste will be sorted out and cut and bags prepared in an existing glovebox located in an adjacent building. Good segregation of waste will be of great importance for a good functioning of the incineration unit.

With respect to radiological properties, waste to be incinerated is low-level waste, containing mainly beta and beta-gamma emitters ($^{60}$Co, $^{137}$Cs, $^{90}$Sr) with a total specific activity less than $10^5$ Bq/kg. Radiological characterization performed on several waste streams did not show significant presence of alpha emitters.

The capacity of the incineration unit was set to 25 kg/h, making it possible on two or three shifts to incinerate the existing stock of waste and the one to be generated in the coming years, during the minimum expected lifetime of the unit: 10 years.

**Safety principles considered in the design of the incineration and cementation units**

AREVA TA developed a set of safety functional requirements associated with nuclear safety, radiation protection for the workers and the public and protection of the environment. A Safety Options File gathered and developed the main safety requirements to be considered in the unit design, based on French principles and practices, with consideration given to Russian regulations and practices as well.

The main safety functions taken into consideration in the design of the incineration and cementation units were to prevent internal and external hazards:

- internal hazards:
  - dispersion of radioactive substances: aerosols in case of drop of a waste bag, ashes during maintenance operations;
  - external exposure of operators;
  - fire and explosion due to presence of kerosene in the facility;
  - and other industrial and chemical hazards.
External hazards: the main external risk taken into consideration was earthquake, with requirement of seismic resistance of the building but not of the process units. The main following safety functions were included in the design of the facility:

- dynamic containment of the incineration and cementation units;
- dynamic containment of the building n°163, to be ensured permanently;
- tightness of the oven which ensures a static containment: a $10^{-3}$ leaking rate was defined;
- Limitation of external exposure at the working place and in storage areas;
- Limitation of presence of operators in the process rooms;
- Easy to decontaminate paints and materials used for soils, walls;
- Recovery of water generated in the facility in case of extinction of fire, flooding;
- Radiation control of personnel leaving process areas and before leaving the facility.

Incidental situations were assessed. The major design basis accident is fire in the cementation process, as a total activity of 15 MBq would be present in ashes stored in a collection tank. Maximum radiological impact to workers would be an effective dose of 1.6 mSv, with the assumption of inefficient personal protective equipment (mask). Radiological impact to the populations of Yagri Island would be about 2 µSv.

**Main figures of the design of the incineration and cementation units**

Early 2007, AREVA TA launched the design of a waste incinerator and a cementation unit for ashes. Fours Mouratille, Hygetec and Robatel were French companies associated to the project for design and construction as sub-contractors.

The incineration unit comprises the incineration system itself (see figures 1 and 2):

- feeding system based on conveyors and airlocks;
- oven for pyrolysis phase followed by combustion phase: temperature 850°C;
- post-combustion at 1100 °C;
- combustible was kerosene as required by Zvyozdotchka.
- first filtration of combustion gas by means of ceramic filters (French supplier: Gosfume)
- dry neutralization of combustion gas with sodium hydrogen carbonate powder
- final filtration of off-gases with HEPA filters, designed by Onega, engineering company subsidiary of Zvyozdotchka.

A nuclear ventilation was designed for the incinerator and cementation processes, with a depression of the oven of 80 Pa compared to the room where it is located, and
200 Pa between the post-combustion chamber and the oven. These depressions were defined to comply with the Russian regulations.

This nuclear ventilation is independent from the nuclear ventilation of the building itself, which is supposed to work permanently even if the incineration and cementation units are not in use.

Ashes are extracted by means of three endless screws in stainless steel. Two screws are located under the oven, and the third one under the ceramic filters. Ashes are conveyed to a collection tank which feeds the cementation process.

The cementation unit aims to immobilize radioactive ashes with a cement matrix in 200 l metal drums. Formulation of cement matrix was provided by Zvyozdotchka. It is made of Portland cement, bentonite, and water added to ashes in a defined composition.

The cementation unit is composed of (figure 3):
- a collection tank where homogenization of ashes coming from the oven and the ceramic filters is performed with a paddle;
- a system for admixture ashes, aggregates (cement, bentonite) and water in the 200 l drum, with screw conveyors and pump
- a drum rotating device for mixing all components.

A sample of ashes is planned to be taken by Zvyozdotchka for each drum realized for traceability and in case some characteristics would need to be assessed in the future.

At the end of the detailed design phase, including the safety analysis report and the environmental impact assessment of the incineration facility, a comprehensive file was submitted to the Russian Authorities for State Expertise. Approval of this file made it possible to launch the construction phase in France.

The equipment was manufactured in France in 2008. Cold testing of the incineration and cementation units was carried out in France in December 2008, in presence of representatives of the Russian Certification Company Atomcertifica. Certification was obtained for both units and their transportation to Zvyozdotchka shipyard was carried
out in February 2009. Some unexpected delays have occurred due to the Russian customs, but without major impact on the project schedule.

Meanwhile, refurbishment of building n°163 was performed by Zvyozdotchka before to start mounting the equipment.

Present status of the project
Mounting of the incineration and cementation units are completed. Settings of the several components of the facility are currently going on in cold conditions. Radiation protection equipment has been supplied and installed. Non radioactive waste, but representative in terms of physical and chemical composition, has been conditioned in cardboard boxes and will be burnt in order to produce inactive ashes. Thus, both the incineration unit and cementation unit will be operated in cold conditions and will receive the approval of the Russian Authorities to start testing in active conditions, with radioactive waste. This last testing phase is expected to take place in the weeks to come, before Zvyozdotchka would start to operate its new incineration facility.
1. Introduction

The European chemometrics standard says (ref.1): “The placing of the analytical sensor in
these situations (process analysis within industrial production, and in situ measurements on
contaminated land) constitutes the taking of a sample”. Therefore, each clearance
measurement result must be considered, as a singular data pertaining to a statistical sample
then, the result of each sampling plan is a set of data which treatment is the support of the
decision making to select between the available options: waste management, unconditional
clearance or, if any, additional material treatment (decontamination).

Once all data are collected, they are statistically summarized and graphically analyzed.
However, those analyses usually do not include the data fitting to probabilistic models. This
paper will analyze the usual probabilistic models: normal distribution, lognormal distribution
and Weibull distribution.

2. Residual material management

During the lifetime of any nuclear installation, initially slowly but as the operational life
continue, and finally when the plant achieves the decommissioning phase, increasingly
amounts of residual materials are produced and Its management must be optimized. One
sustainable management option for metallic scraps, but can be used for other residual
material types, is the clearance option. Five years ago a general material clearance
approach based in the MARSSIM (2) approach was presented (3). In that previous paper,
additional elements to the MARSSIM (2) were introduced such as:

(a) The use of a derivate of the Unity Rule quantity: The Residual Activity Index (or
Clearance Index as nominated by others (see ref.5):

\[
\text{RAI} = \sum_{i=1}^{n} C_i / CL_i = \sum_{i=1}^{n} \sum_{j=1}^{l} C_j \left( \frac{1}{CL_j} + \frac{f_{ij}}{CL_i} \right)
\]

Where: \(C_i\) is the concentration of the \(i\) – radionuclide of concern; \(CL_i\) is the Clearance Level
of the \(i\) radionuclide and, \(n\) is the number of radionuclides of concern, which includes the
radionuclides difficult to measure (DTM nuclides); \(C_j\) is the concentration of the key
radionuclide of interest \(j\); \(CL_j\) is the Clearance Level of the key radionuclide \(j\), and \(l\) is the
number of key radionuclides of concern, \(m\) is the number of radionuclides difficult to
measure (DTM nuclides),

(b) The use, according the recommendations of R. O. Gilbert (4), of a robust estimator for
the scaling factor; \(f_{ij}\): The 95% UCL of the Median of the nuclide ratios in the Survey Unit.
This free-distribution estimator are not affected by the violations of the postulated
distribution type (normal or lognormal) and/or lineal correlation hypothesis (x-y or logx-logy).

To perform the surveys four techniques were considered:

- Simple geometry: Classic approaches using RP handheld devices
- Complex geometry: To deal with special cases with non radiological accessible
  surfaces
- Big Size Items solved by sampling and portable gamma spectrometers, and
- Potentially homogeneously Lots or Containerised Materials using gamma total box or
  portal multi spectrometer devices
The first and fourth techniques have been and continue being used in the Spanish nuclear installations. In the third, the clearance survey is performed using one or more portable gamma spectrometer combined with sampling to establish the scaling factors. Normally the sampling is performed during the previous surveys (initial, decontamination).

In respect the second technique, it is under development yet. Usually, a material process scheme comprises 3 steps or phases:

- Step 1: **Initial characterization and handling**
- Step 2: **Treatment (decontamination) and/or Storage for decay**
- Step 3: **Clearance or conditioning as waste**

Radiological and no radiological characterization data are obtained during the three phases,

3. Process simulation

There is a various seri to minimize the quantity and volume of potential radioactive wastes. However, many constraints are in the field to be managed. Some of they are, no enough previous experiences, inadequate premises (Elevated backgrounds, lack of space, routine operational activities interferences, etc.), no suitable equipment (as cranes) and supplies (electrical power, compressed air, HVAC capabilities, decontamination capabilities, etc.), no operational procedures (to perform intentional or probabilistic sampling), no available good training (for field SOP) and no availability of funds.

Conceptually, a simple treatment such segregation + decontamination can be envisaged as a simple option (Figure 1). Obviously, this simple process admits different options such as to clean the loose contamination before to segregate or to clean in depth before segregate. However, to deal with all the possible variants could be cumbersome for this paper. Now, we define Segregation as an activity in the process performing not only physical segregation, to prepare “relatively” homogeneous lots of materials but also to remove from the process the items with the higher radiological values. That means, in terms of the distributions of residual activity, to remove or cut, although in some sense randomly, the contamination distribution higher values tail.

The next step in the process is to decontaminate. If de decontamination factor DF is an assumed constant, defined as:
DF = Initial Activity (IA) / Final Activity (FA)

\[ FA = FD \times IA \]

Nevertheless, if the initial and final activities are random variables (6), assuming the two variables independence, expectations are:

\[ E[FA] = FD \times E[IA] \]

In addition, variances are in the equation:

\[ \text{Var}[FA] = FD^2 \times \text{Var}[IA] \]

As previously, the IAR is the dimensionless quantity used to represent the residual activity not only initial but also in all the process phases. Many experiences in residual materials and radioactive waste characterization use the lognormal distribution to the nuclide activity in NPP’s.

Usually, in initial characterization a short amount of data (less than 100) is taken and sample mean and standard deviation are determined so 10 and 40 IAR values are typically found in scraps characterization. According this and postulating a lognormal underlying population distribution, a initial lognormal distribution LN(10,40) and 60 size random sample was obtained; the IAR_{simul}. Now we perform the effect of radiological segregation on the data. Then, we assume that if the “a priori” decontamination factor value were 5, so, to process optimization, we would have an operational segregation procedure will permit to segregate the material with higher than IAR=5 values. However, to take in account the variability of the decontamination factor we allow that, in practical terms, the materials with lower values of 6 (in average) will pass forward in the decontamination shop. The new random variable is IAR_{segre}. The decontamination effect is simulated simply dividing the IAR_{segre} sample by 5 the foreseen decontamination factor. The IAR_{decon} distribution is in the left side of the previous and it takes a clear bell-shaped similar as the normal distribution but slightly skewed to the right (as Weibull distribution with shape factor higher 1).

The above and similar curves are presented in clearable or cleared residual material. Finally two probabilistic models or distributions appear as the most suitable candidates: the lognormal and the Weibull with shape parameter higher than 1 (e.g. 1.7) and scale parameter (e.g. 0.5). With this values the average is around 0.446 but the most important issue is more than 0.96% of the population is below the value 1.

4. Clearance rules and doses

The clearance levels, CL, recommended in the RP-89 (8), which based in 10 microSv/y radiological criteria, were obtained using the dose model described in the Radiation Protection 119 (ref. 9) for these volumetric concentration clearance levels. For compliance demonstration with those clearance levels different approaches are used to perform the clearance of residual materials. But the general rule can be expressed such as: If the decision unit, DU (or lot) concentration average is higher than the reference (clearance)
level then, the lot is refused and if the DU concentration average is lower than the reference (clearance) it is cleared.

In both options, the mentioned lot concentration average is a real and unknown parameter of the concentration population actually present in the lot. Therefore, those rules must be translated in rules based on the sample data statistics:

- If the estimated sample concentration mean is higher than the reference (clearance) level, the lot is refused or
- If the estimated sample mean is lower than the reference (clearance) level (CL), then:
  1. If all the data are lower than the CL the lot is conditionally (or unconditionally) cleared or
  2. An mean upper Chevichev’s confidence level is lower than the CL or
  3. If any value is higher than the reference (clearance) level the lot is cleared if this value or data subset is lower than a CL multiple defined as an outlier limit (Grubb’s test) or,

(3bis) according the MARSSIM approach, if the sample data accomplish the Sign test and any datum, is higher than the reference (clearance) level the lot is cleared if this value or data subset is lower than a model or lot size dose-based CL multiple namely the clearance level for elevated measurement comparison CL_{EMC}, as:  

\[
CL_{EMC} = MF \times CL
\]

Where the MF is an elevated level mass factor this factor can be calculated using the CL dose model or “a posterior” using the elevated measurement area vs. the survey unit area. In our case we will need use a Mass Factor, MF, based in the RP – 117 (ref. 9) dose models.

5. Probabilistic models and compliance tests

Three probabilistic models can be considered: Normal Distribution, Lognormal Distribution and Weibull Distribution model. The corresponding tests can be applied in the demonstration of compliance.

First, the tests are applied to the above three random variables, the results if t-test, Chevichev’s test, sign test, Kurtosis test and elevated measurement comparison are in table 6. Only is notorious the concordance of all the results in the variable IAR_{decon}. With this result of compliance we try to fit its values to the mentioned probabilistic models. Due the data are skewed they don’t fit the Normal distribution so it is not a valid model. However, the lognormal is not refused by the classical fitting tests and the Weibull is refused by the classical tests but it is considered because those tests are very conservative and the plots seem very similar.

The two distributions obtained were LN(-1.107566; 0.779028) and W(0.447151; 0.693239). Once we have produced two distributions of “potentially clean” material we obtain 100 random samples of size 6 from these two distributions. Now the t-Test on the 100 samples, the t-test to the naiperian logs of those data, also the Sign test are performed and we find the usual randomized results: When the Lognormal samples are tested the 34% is refused by the t-Test, the 30% is refused by the log t-test but only the 9% is refused by the Sign test and when the Weibull samples are tested the 16% is refused by the t-Test, the 25% is refused by the log t-test but only the 6% is refused by the Sign test. The above results support the use in samples with size as low as 6 of non parametric tests as Sign test as is recommended in MARSSIM (ref. 2). Also, it appears the behaviour of those test is better for Weibull samples than for lognormal samples possibly because the Weibull distribution, with shape factor higher than 1, is less heavily tailed than the lognormal distribution.

5. Estimating doses

When a probabilistic model is assumed is possible to estimate the doses from the residual material to the potential exposed person described in the dose model used in the standard. Now we assume the previous lognormal model. In this model the expected value is:
\[ E(X) = \exp(\mu + 0.5\sigma^2) = 0.45 \]

If the clearance level is 1 Bq/g as in the case of Cobalt 60 and the MF are as in the enclosed table 6. According this table, if we have in a SU mass of 500 t, with mean 0.447 Bq/g and there is a 2 t part with mean 2.44 Bq/g, the annual dose due to this SU will be (as in 10):

\[ D = 10 \text{microSv/y} \times \left( \frac{0.447 \text{Bq/g}}{1 \text{Bq/g}} + \frac{2.44 \text{Bq/g}}{5 \text{Bq/g}} \right) = 9.35 \text{microSv/y} \]

Using a Mass Factor, 5 for 2, obtained from the dose model used in ref. 9.

Also, in a container with 2 tons using a 6-measurement procedure one sixth volume/part with 10.6 Bq/g concentration mean could be accepted:

\[ D = 10 \text{microSv/y} \times \left( \frac{0.447 \text{Bq/g}}{1 \text{Bq/g}} + \frac{10.6 \text{Bq/g}}{19.8 \text{Bq/g}} \right) = 9.82 \text{microSv/y} \]

Using a Mass Factor, 19.8 for around 0.3 t, obtained from the dose model used in ref. 9.

However, an investigation should be initiated because the 10.6 Bq/g value has, according our model, a very low probability of occurrence.

6. Conclusions

The optimization of the clearance process depends strongly on the segregation and decontamination activities when they are necessary. However the lowering of clearance levels is less important when the probabilistic approaches were consistent.

The simulations of those activities permit us to understand the appropriate probabilistic models in this process and the non parametric test appears robust and more consistent with the applied probabilistic models.

Finally, the dose calculations permit us to understand the necessity to determine consistent mass factors avoiding unnecessary efforts and facilitating the waste volume and worker dose savings.

7. References

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ABSTRACT

We briefly present our atomistic kinetic Monte Carlo approach to model the diffusion of point-defects in Fe-based alloys, and therefore to simulate diffusion induced mass transport and subsequent nanostructural and microchemical changes. This methodology has been hitherto successfully applied to the simulation of thermal annealing experiments. We here present our achievements in the generalization of this method to the simulation of neutron irradiation damage.

1. Introduction

It is long known that steels harden and embrittle under neutron irradiation. Atomic collision cascades, induced by impinging neutrons, create Frenkel pairs (couples of vacancies and self-interstitials) that, by thermally activated diffusion, cause or enhance atomic species' segregation, precipitation, and also eventually crystallographic changes. In turn, the macroscopic response of the material is changed, because of the higher concentration of obstacles to the motion of dislocations. This is a problem that affects structural materials used in existing reactors, as well as those envisaged for future nuclear systems.

The development of predictive tools to model the kinetics of such diffusion processes is thus of great importance for the nuclear sector. This is a delicate issue, since it requires the accurate estimation of the mobility of defects redistributing atoms in a complex system undergoing phase transformation. The method we propose \[1\] uses a classical atomistic kinetic Monte Carlo (AKMC) paradigm. In this model, the material is described at the atomic level, all atoms being hard spheres, located on the positions corresponding to the crystallographic structure of interest (bcc, fcc, hcp,... depending on the material), generally on a rigid lattice. Solute atoms and defects (vacancies and/or interstitials) are also introduced. In the AKMC algorithm, point-defects migrate one at a time, thereby inducing mass transport. The probabilities for all individual migration events, as well as the corresponding time increments, are calculated with the mean resident time algorithm \[2,3\], based on the migration energy barriers calculated at absolute zero temperature. The quality of the AKMC algorithm relies therefore mostly on how the migration energies are calculated.

Many algorithms exist to calculate point-defects' migration energies, that are different compromises between speed and accuracy: the simplest algorithms demand very little CPU time but make strong assumptions (e.g. Kang-Weinberg-like (KW) empirical approaches \[4\], where all atoms remain at the perfect lattice positions, therefore completely ignoring atomic relaxation), whereas other algorithms are very rigorous but are practically inapplicable to most problems of interest, because of their prohibitive CPU time requirements (e.g. Henkelman's approach \[5\]). The methodology that we propose is a viable alternative to these extremes: on the one hand, we keep a rigid-lattice description of the system, thereby keeping the description of the possible events as simple as possible; on the other hand the migration energies are calculated with only a few approximations, for example with the nudged elastic
band (NEB) method [6], using an appropriate inter-atomic potential to calculate the atomic forces and energies, as a function of the distribution of surrounding chemical species and other nearby defects. The only required physical input is thus a reliable inter-atomic potential for the alloy of concern. As the use of on-the-fly calculated barriers would be unfeasible in practice, an artificial neural network (ANN) [7] is used instead, as a mathematical regression tool, trained on a set of calculated examples. The ANN input are on-site variables describing the migrating defect’s neighbourhood.

In a preliminary step, this ANN-based approach has been successfully applied to a simplified first-case problem, that is the simulation of thermal annealing experiments, as described in section 2. Our first achievements towards the generalization of this methodology to treat neutron irradiation are outlined in section 3.

2. Application to the simulation of thermal annealing experiments

The simulation of thermal annealing experiments is a simple application of AKMC methods to a real-life problem, where direct comparison with experimental data is possible. The simulation is conducted by considering a small periodic box of atoms. Solute atoms are introduced, initially at random position, with a concentration that corresponds to the alloy being studied. One vacancy is introduced as well, the only migrating point-defect in the system. At every step of the simulation, the only possible events are thus the migration of the vacancy to one of its first-nearest-neighbours (1nn) positions, and their frequencies of occurrence are calculated as thermally activated processes: \( \Gamma = \nu_0 \exp(-E_m/k_B T) \), where \( \nu_0 \) is a constant \((6 \times 10^{12} \text{ s}^{-1})\), \( T \) is the absolute temperature and \( E_m \) is the migration energy calculated at absolute zero temperature.

In our approach, the migration energy is calculated using an interatomic potential, as pictorially depicted in Fig. 1. Such a calculation is easy to perform on-the-fly, but remains highly CPU-time demanding. Therefore, we only calculate a few energy barriers \((\sim 10^4)\), and train an ANN to predict it, given a description (under the form of a string of integers) of the local atomic environment around the migrating vacancy as input variables. Obviously, the interest is that, once trained, the CPU time required by the ANN to make a prediction is trivial compared to the NEB itself.

A complete description of this methodology, including the mathematical and practical aspects about how the ANN is trained, and also very importantly about how ANN predictions for new cases are reliable, is outside the scope of this paper, but can be found in Ref. [1]. Fig. 2 shows an example of ANN quality of prediction of the migration energy, for the case of a binary Fe-Cr alloy. The accuracy of the predictions is very high, so the ANN can, in this case, be considered as a perfect substitute to the NEB.
The ANN shown in Fig. 2 has been used to perform the simulation of a thermal annealing at 500°C for an Fe-20%Cr alloy. About $1.3 \times 10^{10}$ AKMC events where computed in approximately two CPU months (on modern mono-processor machines). Fig. 3 shows how the average Cr precipitate radius and Cr precipitate density, as predicted by the AKMC simulation, compare with 3 different series of experimental data. For further comparison, the results obtained when the migration energies are empirically estimated in a Kang-Weinberg-like (KW) way [4, 8], i.e. based on the rigid-lattice calculated energy difference, without any atomic relaxation, are also shown. The simulation time has been rescaled to fit the first experimental data point of the average precipitate size. This is a normal procedure for the AKMC simulation of thermal annealing, because of the high vacancy concentration in the simulation box (more details can be found in [1] and [8]).

We can see that the ANN-AKMC results are in better agreement with the experimental data, because both the average precipitate size and average precipitate density follow the experimental trend, contrary to the KW-AKMC results, where the precipitate density is largely over-estimated. This supports the importance to take an approach such as ours, where on the one hand static atomic relaxation and long-ranged chemical interactions are taken into account, but on the other hand computing time remains affordable thanks to the use of a properly trained ANN to calculate the energy barriers.

Figure 2 – ANN quality of predictions of vacancy migration energies in a binary Fe-Cr alloy. Average errors are lower than 5% and the correlation coefficients $R^2$ are higher than 0.98. The number of input variables is 223.

Figure 3 – Thermal annealing of an Fe-20%Cr alloy, using 2 different AKMC models. Series of experimental data (Jacquet, Novy and Bley, see references in [1]) are shown for comparison.
3. Generalization to the simulation of radiation damage

The generalization of the AKMC algorithm to the simulation of radiation damage requires that the existence and formation of vacancy and self-interstitial clusters is accounted for. This is however not straightforward, because of various methodological and practical reasons. In particular, the fact that several mobile point-defects must be introduced in the same simulation box implies that several hypotheses, on which the AKMC algorithm described in section 2 is based, are not valid anymore:

- Not all possible states of the system are uniquely stable anymore, instead more than one metastable state (i.e. local minimum in the potential energy surface) may exist.
- It is impossible to pre-define all migration mechanisms because of: (a) the presence of other point-defects, that open the possibility for combined migrations to take place; (b) the possibility to encounter spontaneous migration events, i.e. having a zero energy barrier, that in turn also open the possibility for combined migrations to take place.
- The migration path between two given states is not unique anymore, this complicating the definition of it from the ANN point of view.

The problem is therefore mostly methodological: a new AKMC algorithm must be defined, as a compromise between realism and complexity, with the additional condition that an ANN can still be trained to calculate the migration energies to guarantee that static relaxation effects are taken into account, while keeping the CPU time requirements affordable.

The simplest algorithm considers point-defects as additional solute species that also migrate one at a time. The possible migration events are defined in advance (see for example Fig. 4), for all individual point-defects, but combined migrations are totally ignored. The advantage of this approach, despite its debatable simplicity, is that the generalization of the ANN technique to construct a mathematical regression of the migration energy is straightforward from the mono-vacancy case described in section 2. Point-defects in the vicinity of the migrating one are thus simply described by an integer, just like solute atoms are.

This simple AKMC algorithm has been first applied to the problem of vacancy clusters creation and migration. Fig. 5a shows the ANN quality of prediction of the 1nn migration energy of a vacancy in pure Fe. Up to 50 other vacancies are allowed in its vicinity, which is in fact the only factor that influences the migration energy, because of the absence of solute atoms. We can see that the ANN is no longer a perfect substitute to NEB (compared to the single-vacancy case shown in Fig. 2), but remains quite acceptable nonetheless, despite the high perturbation of the matrix that is provoked by the large number of vacancies in the system. This shows that the ANN approach is robust and certainly not limited to single-vacancy problems. A similar application to Fe-Cu alloys allowed the lifetime and diffusion coefficient of Cu-Vac clusters to be calculated [9, 10].

The application of the AKMC algorithm to the problem of SIA clusters migration is more delicate, mainly because of the high strain field around SIAs. The calculation of the energy barrier for a given migration event is especially delicate and not easy to automate. At the moment, good quality predictions were obtained for the relaxed energy difference (i.e. difference of energy between the relaxed final and initial states), as shown in Fig. 5b. This
allows, in a first step, the migration of SIA to be modelled, calculating the migration energy in a Kang-Weinberg way, while still taking static atomic relaxation into account.

4. Concluding remarks
We have shown that our ANN-based AKMC algorithm to model the kinetics of diffusion of point-defects is robust, and can be generalized to take the presence of other nearby point-defects into account. As a consequence, our ANN-based AKMC, in combination with other models that also belong to the family of multi-scale modelling, can potentially be very helpful to simulate with high accuracy neutron irradiation in steels, at the lowest possible CPU time cost. An important requirement, however, is the development of appropriate interatomic potentials that take, together, several of the most important chemical elements into account (Cu, Ni, Cr, Mn, Si, P, …).

5. References
COUPLED NEUTRONICS – THERMAL-HYDRAULICS PROGRAMS FOR SCWRS

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ABSTRACT

The Supercritical Water Cooled Reactor (SCWR) was chosen as one of the Generation IV reactors by GIF. At the moment, a number of concepts – thermal as well as fast ones – exist. The reference parameters for a thermal SCWR have been taken from the European High Performance Light Water Reactor (HPLWR). Since the pressure is higher than the critical pressure (22.1 MPa) there is no change in the phase of the water in the core. On the other hand, due to the significant changes in the physical properties of water at supercritical pressure, the system is susceptible to local temperature, density and power oscillations. This inclination is increased by the pseudocritical transformation of the water used as coolant. Thus, for modelling a system of this type coupled neutronics – thermal-hydraulics programs are required.

Such a program system has been developed with the following main features: great modularity which allows for easy modifications, thus several SCWR concepts can be studied; detailed assembly calculations (with MCNP) and full-core analysis (with SCALE) are supported; the differential equations of xenon poisoning are implemented to study xenon oscillations.

The program system was used to examine the assembly of the HPLWR, to design the assembly and the core of the Simplified Supercritical Water Cooled Reactor (SSCWR) and to model xenon oscillations in SCWRs.

1. Introduction

The Supercritical Water Cooled Reactor (SCWR) is a Generation IV reactor concept which combines the technologies of recent Light Water Reactors (LWRs) and supercritical fossil fired power plants. It is a high-temperature and high-pressure water-cooled reactor operating above the critical point of water. Depending on the core design, it can have fast or thermal spectrum. By now, several versions have been developed which include: the American SCWR with power channels ([1]), the Canadian CANDU-SCWR ([2]), the European HPLWR with a three-pass core ([3]), the Japanese Super LWR ([4]), the Korean SCWR with a cruciform type U/Zr solid moderator ([5]), etc. This paper focuses on the thermal concepts. The reference parameters for a thermal SCWR have been taken from the European High Performance Light Water Reactor (HPLWR): operating pressure of 25 MPa, coolant inlet and outlet temperature of 280°C and 500°C, respectively ([3]). Above the critical pressure (22.1 MPa) there is no change in the phase of supercritical pressure water, but the physical properties of the fluid vary strongly, e.g. the outlet density is about an order of magnitude lower than that of the inlet. Thus, in a thermal SCWR the use of extra moderator is necessary which can be in the form of water channels and/or rods or solid moderator like zirconium-hydride.

Due to the strong variation of the coolant density the system is susceptible to local temperature, density and power oscillations (this inclination is increased by the pseudocritical transformation of the supercritical pressure water). Thus, I developed a coupled neutronics – thermal-hydraulics code system (CNTH) to study SCWRs. This paper presents the set-up of this program system. Results obtained include detailed assembly calculations, design of a new assembly, full-core analysis and xenon oscillation.
2. **Set-up of the program system**

The flowchart of the CNTH program system is shown in Fig. 1. The heart of the calculations is the inner iteration which requires under-relaxation to speed up convergence. In the literature, usually a convergence limit for the temperature of the coolant is prescribed. On the other hand, in the pseudocritical region the density of the water decreases dramatically with increasing temperature (e.g. 1 or 2°C temperature difference causes 40 or 75 kg/m³ difference in density), and this alters the moderation capability of the water. Thus, for systems with supercritical pressure water as coolant the density is the most important parameter for the neutronics calculations, therefore it is better to prescribe limits for this quantity.

If calculations which change the atom density of the materials (e.g. xenon poisoning or fuel depletion) are carried out an outer iteration is also required, denoted with “Burn-up” in Fig. 1. Besides the geometry module, I wrote all other parts of the program system as universally as possible, thus a great set of problems can be examined without extensive changes to the code. The neutronics module is able to use MCNP for detailed calculations (e.g. one assembly) and also SCALE6.0 for larger models (e.g. a full-core).

![Flowchart of the developed CNTH program system](image)

3. **Detailed assembly calculations for the HPLWR**

The assembly of the HPLWR ([6]) was examined. I obtained the following results ([7]):

- The power distribution can be easily changed by modifying the enrichment profile. Due to the heat-up of the coolant the moderation varies axially which can be compensated with an appropriate axial enrichment profile.

- Changing the ratios of flow rate in the different water regions (coolant between the fuel pins, moderator in the moderator box and gap water between the assemblies) alters the heat transfer coefficients. This effect combined with the preceding point can be used to mitigate or even eliminate hot spots in the assemblies.

- In the literature there is still no general heat transfer correlation for supercritical pressure water, thus the effect of changing the formula for its calculation was also examined. The scattering of the results shows that solving this deficiency is very important.

4. **The SSCWR**

Zirconium-hydride (ZrH) can be used as extra moderator in thermal SCWRs. As pointed out in [8] the δ-phase with a hydrogen to zirconium ratio of 1.6 to 1.0 is the best suited material for SCWR conditions. Thus, a new SCWR concept with ZrH was proposed in [9] (Fig. 2).
4.1 The design of the assembly ([9])

As the density of the coolant becomes smaller, the number of ZrH rods should increase by replacing some of the fuel pins with such rods. Thus, some of the fuel pins and the ZrH rods will be part-length rods. I calculated the power and temperature distribution of an SSCWR assembly (Fig. 3a). The characteristics of the power distribution are the same as by other SCWRs, namely there is a peak in the bottom part of the assembly. This is acceptable since the coolant temperature is the smallest here. The somewhat different characteristics of the linear power distribution (Fig. 3b) compared with the power distribution are due to the increasing number of ZrH rods which is also responsible for the local maxima in the moderation (H- to U235-atom ratio). The periodic decrease of the moderation can be explained with the continuous increase of the coolant temperature.

![Diagram of the SSCWR assembly]

Fig 2. The assembly of the SSCWR

![Power and temperature distributions (left) and linear power and moderation distributions (right) of the SSCWR assembly]

Fig 3a-b. The power and temperature distributions (left) and the linear power and moderation distributions (right) of the SSCWR assembly

The preceding assembly was somewhat modified and used for further calculations which are shortly described in the next section. I also examined the hexagonal assembly and found only minor differences compared to the square one ([10]).

4.2 Full-core analysis

The previous calculations were carried out by MCNP which allows detailed modelling, but in turn they were computationally very expensive. For a full-core analysis a faster
computational procedure is required, so I chose the SCALE6.0 code package for this task. I developed a step-by-step cross section homogenization process which I validated with MCNP calculations ([10]). The prepared homogenized macroscopic cross sections are used by the Monte Carlo module KENOVI. Since the assemblies are homogenized a full-core calculation takes less than an hour on an average performance CPU. A preliminary core configuration was obtained by carefully changing the enrichment, burnable poison content and number of ZrH rods in the assemblies. I found that radial power peaking is more pronounced in the regions with low coolant densities. Further analysis indicates that an increase in the average fuel enrichment will be necessary.

5. Xenon oscillations

Due to the large dimensions of the SCWR cores (e.g. the active height of the HPLWR is 4.2 m) xenon oscillations must be taken care of. The migration length of neutrons is larger than in conventional LWRs, nevertheless detailed calculations are required. I carried out some calculations on test cores using MCNP in the neutronics module ([11]). These calculations indicated unstable behaviour with respect to xenon oscillations, i.e. without any external intervention the power peaks were getting larger and larger causing excessive cladding and fuel temperatures. Thus, I have investigated the problem more thoroughly using SCALE6.0 in the neutronics module with a similar cross section homogenization process as described in section 4.2.

Fig 4. Xenon oscillation for different cases (Level 1 is at the bottom of the core, where the coolant has the smallest temperature)

Calculations were carried out for a one-pass SCWR called PWR-SC ([12]). The results shown in Fig. 4 confirm the great effect of the temperature feedback (TF):

- Firstly, with an axially uniform enrichment profile and “turned off” TF (Case1) the oscillations are symmetric as expected, the power in the middle level (Level 3) oscillates with double frequency.
- Secondly, if the TF is “turned on” (Case2), the power distribution changes to one with a bottom peak, and the middle level (Level 3) oscillates with the two upper levels (Level 4 and 5). This is also the case in Case3 where the enrichment profile was modified to get a starting power distribution with a top peak.
Thirdly, the greater the peak in the starting power distribution, the larger are the amplitudes of the oscillations. Future calculations will be carried out to quantitatively determine the effect of the temperature feedback. Parallely, the three-pass HPLWR ([3]) will also be examined.

6. Conclusion

A versatile, modular coupled neutronics – thermal-hydraulics program system has been developed to study SCWRs. The program system can be used for several tasks, e.g. fuel assembly calculations, full-core analysis, xenon oscillations. Some of the results were presented in this paper, details can be found in the given references.

7. Acknowledgement

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A DOMAIN DECOMPOSITION METHODOLOGY THROUGH ALTERNATE DISSECTIONS FOR COUPLED NEUTRONIC AND THERMAL-HYDRAULIC ANALYSES IN COBAYA3

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ABSTRACT

Nowadays, coupled 3D neutron-kinetics and thermal-hydraulic core calculations are performed by applying a radial average channel approach using a meshing of one quarter of assembly in the best case. This approach does not take into account the subchannels effects due to the averaging of the physical fields and the loose of heterogeneity in the thermal-hydraulic modelization. Therefore the models do not have enough resolution to predict those subchannels effects which are important for the fuel design safety margins, because it is in the local scale, where we can search the hottest pellet or the maximum heat flux.

The UPM advanced multi-scale neutron-kinetics and thermal-hydraulics methodologies being implemented in COBAYA3 include domain decomposition by alternate core dissections for the local 3D fine-mesh scale problems (pin cells/subchannels) and an analytical nodal diffusion solver for the coarse mesh scale coupled with the thermal-hydraulic using a modelization of one channel per assembly or per quarter of assembly.

In this work, we address the domain decomposition by the alternate core dissections methodology applied to solve coupled 3D neutronic/thermal-hydraulic (N-TH) problems at the fine-mesh scale. The N-TH coupling at the cell-subchannel scale allows the treatment of the effects of the detailed TH feedbacks on cross-sections, thus resulting in better estimates of the local safety margins at the pin level.

If we try to perform a subchannel-by-subchannel coupled N-TH calculation for a full PWR core in order to get all the detailed effects, the need of memory and CPU calculation time will be huge and even in the case we achieve to converge an steady state scenario at nominal conditions, which is a non-trivial case, the main problem come to us when we introduce some perturbations in the system (i.e. insertion of control rods) that makes the problem asymmetric and by far more heterogeneous.

It would be very difficult to follow the propagation of the perturbations in any transient without convergence problems and in reasonable calculation times. A multi-scale approach comes up as the easiest way to achieve such types of calculations, mainly because this approach allows us to establish robust convergence methods at each scale.

Keywords: fine mesh diffusion, multi-scale, multi-physics, domain decomposition

1. Introduction

The development of nuclear technology has enabled more and more realistic vision of the phenomena that occur in a nuclear reactor, in particular in neutron and thermal-hydraulic aspects. This requires on the one hand increasing the mesh refinement of calculation used, so far involved the homogenization of large regions of the reactor core, and secondly, the resolution of the corresponding coupled equations.
To overcome those limitations, the domain decomposition methodology through alternate dissections [1] has been developed and applied to the analysis of PWR reactor cores. The work presented in this paper is focused on the implementation and validation of a neutronic/thermal-hydraulic coupling of these features, which has been performed as part of the PhD thesis of the first author. This methodology allows us to tackle the problem of calculating the full-scale core in the level of fuel rods and subchannels with a multi-scale approach. The core calculation at this level of detail allows us to consider the effects of detailed thermal-hydraulic feedback on cross sections, resulting in better estimation of local security limits in the fuel bundles.

The domain decomposition methodology was first applied in the local-global scheme used in the Spanish System of Analysis of Pressurized Water Reactors [2][3]. In the neutronic field, current to flux ratios based on the interface discontinuity factor diffusion formulation [4] are used as boundary conditions in the calculation of each subdomain problem. In the thermal-hydraulic field, cross-flows between adjacent subchannels are chosen as boundary conditions equivalents to the current to flux ratios in neutronic to connect different subdomains.

The methodology cited in the preceding paragraph has been applied to the OECD/NEA MOX Core Transient Benchmark [5], using the COBAYA3 code system [1][6] for making these multi-scale calculations. The results reported in this paper will show the goodness of this methodology to address such kind of problems.

2. Domain decomposition through alternate dissections

The core partition in quarters of assemblies is a physically meaningful way of domain decomposition that can be applied in both neutronic and thermal-hydraulic physics [4]. Figure 1 shows a graphical representation of the level of detail considered for the fine-mesh transport corrected diffusion solver COBAYA3 and the two phase-flow code COBRA-TF [7] which provides the thermal-hydraulic subchannel solution for the cross section feedback.

The neutronic boundary conditions (BC) are the fine-mesh current to flux ratios in each cell interface, because they are independent of the level of flux in the subdomain and leave the diffusion equation in a homogeneous form. Therefore, they are insensitive to the fission source levels supposed over the core in order to converge the k-eigenvalue problem. On the other hand, the domain decomposition is unable to produce the correct source level for the core, only can improve the fission source distribution inside the subdomain. The alternate dissections scheme makes it possible to reach the true fission source distribution by “mixing” the quarters of assemblies. In that sense, the nodal solution is in charge of improving the fission source levels for the full core problem which are used to normalize the fluxes of each subdomain in the pin-by-pin solution, and hence, speeding up the overall problem convergence.

The thermal-hydraulic BC are mass injections (or sinks) at the outer open surfaces of the subdomain. Those mass injections come from the previous subchannel calculation on a different core dissection, and are derived from the cross-flows obtained in the centerplanes gaps connecting quarters of assemblies.

During the iterative solution process, the BC on the subdomain and their returned updated values on the centerplanes at the end of each computation are the information to be transmitted mainly. But also volume information is passed to initialize the values for the next subdomain computation and keep track of the convergence to the full core solution.
To reduce computing times, each formed subdomain is solved simultaneously to other subdomains in different processors thanks to the parallelization of the solution method. The processors compute the k-eigenvalue and fission source distribution for each isolated subdomain, then new current to flux ratios are computed in the centerplanes. When convergence is reached in the full core, k-eigenvalue coincides for all the subdomains, the pin powers are stored and then used by the thermal-hydraulic code as a fixed source term.

The thermal-hydraulics solution uses a quasi-stationary method to reach the steady state conditions because, up to now, COBRA-TF does not have a steady state solution implemented. The solution initiates at zero time and a void transient is run till convergence is reached for each subdomain, then cross flows are computed in the centerplanes of the geometry which are subchannel boundaries where BC will be applied in the next dissection. Those BC are formed by the three fields exchanged between neighbouring subchannels in the centerplanes of the subdomains, namely mass, enthalpy, and momentum.

After each dissection, all the subdomains are said to be in the same instant of time, and a restart case is computed for the next dissection using the updated cross flows and the already computed fields as initial conditions. Then, a comparison is established between the temperatures and densities fields computed in different iterations to decide the moment of convergence. In nominal conditions only two to three different alternate dissections are needed for the thermal-hydraulics to converge the full core temperatures and densities distributions. Afterwards, the cell temperatures and densities are stored and used by the pin-by-pin solver to interpolate in the cross sections tables in a new pin power computation.

3. Implementation and performance in the COBAYA3 code

The master-slave solution method chosen uses one single reserved process to manage the information treatment; this process prepares the values to be sent to each subdomain and sends that information to one subdomain at a time. All the information is packed in a single buffer and then sent in a single hit to the MPI library, so as to avoid generating constantly all the wrap-up that accompanies message sending.
The master process sends sequentially each subdomain to free processors, meaning that they are not computing a subdomain at that time. When there are no more processors available, it waits until one finishes his computations and sends results, and then it continues sending new subdomains until the full core has been computed for the corresponding type of dissection.

Thereafter, when the fission source is being computed in a neutronic sweep, a nodal solution is performed in the master process and the interpolation values are generated. Again a new dissection, different from the previous one, begins where the updated nodal values are used to interpolate the boundary conditions in each slave process and the nodal fission source level is used to renormalize the subdomains.

The developed code has been tested in a single 8 processors machine, a cluster of 48 processors, and a supercomputer with more than 1000 processors available. Actually, the optimum number of processors needed clearly depends on the size of the problem, thus, having more processors than the maximum number of subdomains formed, does not reduce the overall CPU time. And, on the other side, having less available processors than the maximum number of subdomains can lead to a situation in which all the subdomains are computed except one, and the execution in the master process have to wait till the last one is finished before going for a new dissection.

The master-slave solution method has been designed to tackle the memory requirements by distributing the memory containers among all the available processors. The memory required for a single fuel assembly including the whole active height is about 300 MB when using 40 axial levels in the thermal-hydraulics and 160 axial levels in the neutronics. Therefore, for a typical PWR core with 157 assemblies, the amount of memory needed is about 46 GB.

![Figure 2. Convergence in a 3x3 minicore problem](image)

Regarding the convergence performance of the domain decomposition, as it can be seen in figure 2, the errors in the currents and fission source distribution are efficiently reduced among different dissections. Those convergence rates correspond to a full length 3x3 minicore configuration with reflective boundary conditions. In this scenario, only 10 TH dissections are needed the get full convergence against 30 N ones, but the total computing time is distributed half and half between N and TH calculations. Generally, the first TH sweeps are more expensive in terms of computing time than the N because of the void transient in COBRA-TF.

4. Results and numerical verification
The OECD/NEA/USNRC PWR MOX/\textit{UO}_2 Core Transient Benchmark [5] provides homogenized pin level cross sections but without cell discontinuity factors to account for cell heterogeneities, transport and mesh size effects. It is used to compute full core results in comparison to the ones submitted in the final report. All the pin-by-pin calculations used the 8G cell cross section libraries.

First, we present a pin-by-pin solution for the full core at HZP ARI as described in Part 3 of the benchmark specification, where each pin, cladding and surrounding coolant is homogenized in one cell. Table I shows a comparison of some parameters with DeCART transport solution which was taken as reference and the PARCS and ANDES nodal solution in 2 groups.

<table>
<thead>
<tr>
<th></th>
<th>DeCART</th>
<th>COBAYA3</th>
<th>PARCS 2G</th>
<th>ANDES 2G</th>
</tr>
</thead>
<tbody>
<tr>
<td>Critical Boron</td>
<td>1265 ppm</td>
<td>1313 ppm</td>
<td>1341 ppm</td>
<td>1343 ppm</td>
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<tr>
<td>PWE (%)</td>
<td>ref 1.17</td>
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<tr>
<td>EWE (%)</td>
<td>ref 2.79</td>
<td>3.49</td>
<td>3.44</td>
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</table>

Table I. HZP 3D steady state assembly power relative error (%) with DeCART

The COBAYA3 values are found to be in between both approaches, and they would be very much improved if the interface discontinuity factors [2] would have been included in the cell library of the benchmark because of the mesh, transport and heterogeneity effects to be corrected. Table II shows the corresponding results of COBAYA3/Cobra-TF using the domain decomposition for the HFP exercise of the benchmark compared against ANDES/Cobra-TF and PARCS nodal solutions.

<table>
<thead>
<tr>
<th></th>
<th>COBAYA3/CTF</th>
<th>ANDES/CTF 8G</th>
<th>PARCS 8G</th>
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<tr>
<td>Critical Boron</td>
<td>1655.0 ppm</td>
<td>1672.5 ppm</td>
<td>1672.0 ppm</td>
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<tr>
<td>F_{z}</td>
<td>1.437</td>
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<td>1.424</td>
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<td>F_{xy}</td>
<td>1.3233</td>
<td>1.395</td>
<td>1.384</td>
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<tr>
<td>A.O.(%)</td>
<td>-10.25</td>
<td>-10.30</td>
<td>-9.9</td>
</tr>
<tr>
<td>T_{Doppler} (ºC)</td>
<td>561.7</td>
<td>561.8</td>
<td>563.0</td>
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<tr>
<td>T_{fuel max} (ºC)</td>
<td>1770.4</td>
<td>1572.8</td>
<td>-</td>
</tr>
</tbody>
</table>

Table II. MOX-part2 3D HFP steady state comparison

There is a very good agreement on the results shown in Table II where the deviations in k_{eff} are in less than 170 pcm. Nevertheless, it can be seen how the maximum fuel temperature is almost 200ºC higher in the pin-by-pin/subchannel coupled calculation than in the nodal/channel solution. In Table III it is compared the assembly wise power distribution obtained from the pin-by-pin calculation compared with the solution provided by ANDES/CTF. The highest errors are concentrated in the periphery of the core where gradients within the assemblies are higher due to the reflector presence. Therefore, the lack of cell discontinuity factors in the cross sections library is more remarkable.

Just to give some numbers about computational times, the COBAYA3/CTF calculation took 2 days and 5 hours using 225 processors in the Magerit cluster, and the ANDES/CTF took 16 min. on a workstation. Without parallelization, the detailed calculation would take 497 days to complete on the same workstation.
Table III. HFP 3D steady assembly power relative error between COBAYA3/CTF and ANDES/CTF

<table>
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</table>

Result with CTF+DS   0,1890
Relative errors (%) with ANDES/CTF

Due to space constrains, no more results will be shown in this section. Nevertheless more calculations of COBAYA3/CTF and ANDES/CTF in minicore and full core configurations can be found in the literature references [4], [6], [8] and [9].

5. Conclusions

The domain decomposition through alternate dissections has shown its potential and performance in addressing the problem of computing a full core at the scale of pin cells and subchannels. The results obtained show a good agreement compared against transport reference solutions and increase the level of confidence of the pin-by-pin fine mesh finite difference solver implementation and also of the coupled subchannel calculation with COBRA-TF.

During the implementation of the domain decomposition method, it has been clearly identify that a neutronic nodal acceleration is very convenient in order to achieve full core convergence and to speed up the solution process. For the thermal-hydraulic solution, the use of a core channel calculation does not improve the convergence in the subchannel scale and therefore it is not used.

Also coupled transient calculations at the pin cell scale have been performed with COBAYA3/COBRA-TF. However, the computing times required in these calculations are quite large because, up to now, the nodal/channel acceleration is not implemented yet in this kind of transient problems. This work is envisioned for future improvements of the method.

A pin-by-pin transport corrected diffusion computation coupled with an nine equations thermal-hydraulic model yields a level of detail useful to take as a reference calculation for comparison with coupled nodal-channel codes or to generate nodal cross sections libraries in situations where the thermal-hydraulics distribution within the fuel assemblies could be of importance in off-nominal conditions. It provides directly more accurate results on the margins at the pin-by-pin scales.
6. Acknowledgments

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The authors thankfully acknowledge the computer resources, technical expertise and assistance provided by the Centro de Supercomputación y Visualización de Madrid (CeSVIMA) and the Spanish Supercomputing Network.

7. References

EXPLORING NEW COOLANTS FOR NUCLEAR BREEDER REACTORS

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ABSTRACT

Breeder reactors are considered the unique tool for fully exploiting the natural nuclear resources. In current LWR, only a 0.5% of the primary energy contained in the nuclei removed from the mine is converted into useful heat, with the rest remaining in the depleted uranium or in the spent fuel. The objective of resource-efficiency stimulated the interest in Fast-Reactor-based fuel cycles which can exploit a much higher fraction of the energy content of the mined uranium by burning U-238, mainly after conversion into Pu-239. Thorium fuel cycles would also offers several potential advantages over a uranium fuel cycle. The coolant initially chosen for most of the FBR programs launched in the 60’s was sodium, which still is considered the best candidate for these reactors. However, Na-cooled FBR have a positive void reactivity coefficient, which has been among others, a fundamental drawback that has cancelled the deployment of these reactors. Therefore, it seems reasonable to explore totally new options on coolants for breeders.

In this paper, a proposal is presented on a new molten salt ($F_2Be$) coolant that could overcome the safety issues related to the positive void reactivity coefficient of molten metal coolants. Although it is a very innovative proposal that would need an extensive R&D programme, this paper presents the very appealing properties of this salt, in the case of using a specific type of fuel, similar to that of pebble bed reactors. The concept will be studied over a typical MOX composition and extended to a Thorium-based cycle. The general analysis takes into account requirements for criticality (opening the option of hybrid subcritical systems); requirements for breeding; and the safety requirement of having a negative coolant void reactivity coefficient. A design window is found in the definition of a $F_2Be$ cooled reactor where the safety requirement is met, unlike for molten metal cooled reactors which always have positive void reactivity coefficients, both in the case of critical and subcritical reactors.

1. Introduction

Nuclear waste management and nuclear fuel cycle efficiency are the main goals defined by the GIV International Forum in the framework of sustainability [1] and, as such, have been discussed and reviewed several times by national and international programs like the INFCE (International Nuclear Fuel Cycle Evaluation) [2] initiative. Indeed, the resource-efficiency requirement, together with the waste management problem, have recently led to increased pressure to move to Fast-Reactor-based fuel cycles as evinced in the launching of programs such as Generation 4 [1] or SNE-TP [3]. All of these initiatives feature a change in motivation for developing fast breeder reactors, whose history is as old as that of thermal reactors, with respect to that of the early 1960s, an era of uranium shortage. Today, the emergent environmental consciousness has more of an influence in driving the scientific community to...
propose new initiatives on nuclear waste management and efficient resource utilization.

Nonetheless, these systems still have to face several drawbacks before a full deployment can be guaranteed since they are somehow hindered by its poor neutronic safety features. In fact, while the goal of sustainability is achieved, the goal of enhanced safety, also essential for meeting the challenges of future nuclear energy systems, cannot be disregarded. These concerns are not unfound and have featured the history of Fast Reactors from its very beginning when, back in the 70’s, the engineering and safety problems encountered in the Enrico Fermi Nuclear Generating Station [4], the first American breeder reactor, led to a failed license renewal. The closure, in 1983, of the high-profile Clinch River Breeder Reactor [5,6] demonstration project, the decommissioning of the French Superphénix reactor [7], or even the Chernobyl-4 [8,9] accident in 1986, highlighted the safety problems attached to breeder reactors. Note that although the latter was not a FBR, the accident evidenced the hazards of prompt supercriticality power surges produced by positive reactivity coefficients.

Section 2 will give a deeper insight on the main safety issues of nuclear breeder reactors. In section 3 a proposal is presented on a new molten salt ($\text{F}_2\text{Be}$) coolant that, when combined with a specific type of fuel, could overcome the safety issues related to the positive void reactivity coefficient of liquid metal coolant. Finally, in section 4, we will summarize the results and draw the main conclusions.

2. Safety issues of nuclear breeder reactors

Breeder reactors are those with conversion ratios higher than one and, when this happens, it is possible to achieve very high percentages of energy utilization since fissile material inventory becomes larger at the end of the operation cycle. It is worth pointing out that to have a breeder reactor we will need, at least, 2 neutrons per each absorption reaction in fissile nuclei. This number increases rampant with the incident neutron energy and it will be at higher energies were higher conversion ratios can be achieved [10].

Since in a fast reactor the fission chain reaction is sustained by fast neutrons, there must not be any moderator in them. Liquid metals are a good option but their chief disadvantage is the relatively high neutron absorption cross section that could lead to reactivity insertions if the coolant “voids” from the reactor. There are many outstanding candidates for metal-cooled fast reactors such as Sodium, Lead, Lead-Bismuth Eutectics or Tin [11], but all of them show to some extent the same safety problems, and the existing solutions handicap the core performance, useful volumes and breeding gains. In this framework, it seems reasonable to explore totally new options as coolant for breeders. A new proposal is presented in next section.

3. A new proposal for coolant

Our proposal is to use $\text{F}_2\text{Be}$ as molten salt coolant in combination with a graphite moderator. The reason for this is that F-19 and Be-9 have a very low capture cross section for the thermal region that when combined with its moderating features result in an excellent blend from the neutronic point of view. If we are able to use them in some form as a molten salt coolant and then combine them with a moderator in the smallest amounts possible to shift the spectrum toward the region where the capture cross section is smaller, we would produce a spectrum fast enough to enhance breeding ratios. At the same time, a thermal tail would dominate the behavior against depletion or loss of coolant accidents. In other words, the negative void coefficient peculiar to neutron moderators would fully compensate the typically positive void coefficient of neutron absorption. A sudden removal of such a coolant would harden the spectrum, reducing the reactivity to a greater extent than the increase in fissions produced by the subsequent lowering on the absorption rate. The idea is...
conceptually the same as that proposed by T. Sujimoto [12], where a moderating material was added to the core to soften the neutron spectrum.

$F_2Be$ has also nice thermal properties, is exceptionally chemically stable and also soluble in water. Beryllium fluoride is widely used in biochemistry so there is a well found experience with its use. Besides, as other molten salts, it shows to be compatible with graphite even at high temperatures and no significant problems with corrosion have been observed. Finally, when used as a coolant for fusion devices or even Fusion-Fission hybrids, would offer advantages to Tritium breeding issues without renouncing to the goodness of $(n,2n)$ threshold reactions in Beryllium.

The theoretical approach was supported by numerical calculations needed to assess the suitability of this idea over a simple but representative geometry. This model accounted for fuel-pin level heterogeneities. The goal was to study if for any combination of fuel-to-moderator ratio, coolant-to-moderator ratio and fuel composition it was possible to have a critical system with breeding ratios higher than one and negative void coefficients. The parametric study was performed with MCNPX [13]. Two types of fuels were studied. Several conclusions were drawn from the experience.

![Figure 1: Locus of points that lead to criticality, conversion ratios higher than one and negative void coefficients in black, red and blue, respectively, for the different fuel-to-moderator ratios and coolant-to-moderator ratios. The different coolant ratios marked (for a 52% fuel ratio) with dots, were used in the burn-up analysis.](image)

The most important result was that due to the enhanced neutronic properties of U-233, namely its higher neutron yield per neutron, it was possible to have configurations that satisfied the three conditions simultaneously. This effect is unique to this combination of moderating and low absorptive properties due to its rather low absorption cross section. This is contrary to what happens to other liquid metals that have poor moderation properties but higher capture cross sections. Indeed, the fact that this coolant moderates without barely absorbing neutrons is the most salient feature of the salt and the reason why we can find a
design window where the three conditions are fulfilled.

Results also proved to be very sensitive to the fissile isotope content of the fuel that is naturally built up in a breeder reactor. In fact, burn-up effects opened an additional dimension with new degrees of freedom that could constraint our solution’s space. Results showed that higher burn-ups for a critical system such as the one colored in green in Fig.1, could be achieved, but would not by caused by a positive net breeding rate. They would come explained by a higher core’s reactivity excess at BOL and a subsequent higher reactivity swing. Moreover, the maximum burn-up would be limited to values around those typically achieved by current operating reactors, reason by which for a critical and intrinsically safe reactor, it would be better to opt for the well-known water-cooled reactor technology.

On the contrary, from the viewpoint of an efficient use of the resources, when restricted to subcritical breeder reactors, our proposal offered several advantages. Although subcritical hybrid systems have inherent safety features because the neutron chain vanishes if the driving external source is switched off, a high neutron multiplication in the subcritical blanket would still be desirable since energy production and nuclear breeding depend on it. Coolants with positive or close-to-zero reactivity coefficients impose a constraint on this value and, hence, hinder the core’s performance. Moreover a lower subcritical degree would allow for higher burn-up cycles before the reactivity swing becomes too large to be counterbalanced by an increase on the external source power. This would enhance the reliability of the plant and open the possibility of using proliferation-proof self-sustained systems.

Concluding, this idea shows an excellent potential to exploit the natural nuclear resources by means of Hybrid reactors and, more particularly to exploit the large amounts of Thorium resources around the world. The thorium cycle has been successfully applied to design a promising long-life hybrid breeder reactor that could work without on-site refuelling, while providing excellent operational margins due to its intrinsically safe nature. Although a full understanding of the proposal would require an extensive R&D program, this analysis paves the ground for further studies.

4. Summary and conclusions

A new coolant candidate, promising to overcome many of the problems attached to FBR, is presented. The proposal is to use F$_2$Be as a molten salt coolant in combination with a graphite moderator. Its desirable thermal properties together with its very appealing safety features on reactivity feedback made it a perfect candidate. The F$_2$Be concept was studied over a thorium-based cycle. The general analysis took into account the requirements for criticality (opening the option of hybrid subcritical systems); the requirements for breeding; and the safety requirement of having a negative coolant void reactivity coefficient. In the study a design window was found in the definition of a F$_2$Be cooled reactor where the safety requirement was met, unlike for molten metal cooled reactors, which always have positive void reactivity coefficients, in the case of both critical and subcritical reactors. The study was complemented with burn-up analyses so as to prove that such a long-life hybrid breeder reactor could work without on-site refuelling.
References


02.06.2010

Wed 10:30 – 12:10

Auditorium

Fuel cycle
JRC RESEARCH ON HIGH TEMPERATURE REACTOR FUELS

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ABSTRACT

At the Joint Research Centre (JRC), a vigorous experimental program is being pursued with the goal to irradiate HTR fuels (in particular pebbles available from external sources) at high temperatures and to high burn-ups in the High Flux Reactor Petten and to measure and confirm low fission product release rates. Since 2002, two such irradiation tests have been completed and several others are under preparation. The irradiations are followed by non-destructive and destructive post-irradiation examinations including burn-up determination and measurements of the fission product release during out-of-pile simulation of accident conditions. JRC has also developed the capacity to produce Pu-bearing fuel particles including kernels and coatings and to perform the required quality control tests.

1. Introduction

Within the Generation IV International Forum (GIF), considerable efforts are devoted in Europe on R&D for the (Very) High Temperature Reactor system. This is one of the six reactor systems which GIF had selected and recommended for further development, in particular for nuclear cogeneration of industrial process heat. The JRC, through its Institutes in Petten (JRC-IE) and in Karlsruhe (JRC-ITU), disposes of an extensive capability for HTR fuel studies as required for the development, qualification and licensing of HTR fuels. This spans fabrication and quality control, irradiation, post-irradiation and safety testing.

At present, interest in HTR fuels concentrates on UO\textsubscript{2}, which will be the fuel deployed in the next HTR. Plutonium oxide fuel is considered in a later step, for the incineration of excess stocks of military Pu, and even potentially for transmutation of minor actinides (“deep burn” strategy). The development of a transuranium elements - bearing fuel fabrication and characterisation chain is being pursued at JRC-ITU. Future work may cover experimental production of “deep burn” fuel as well as thorium fuel. The R&D work performed follows the goals described in the Strategic Research Agenda of the Sustainable Nuclear Energy Technology Platform [1].
2. HTR Pebble Irradiation Programme in HFR and Post-Irradiation Examinations

2.1 HFR Irradiation Programme

After a pause of several years after the end of the German fuel qualification programme, JRC-IE resumed in 2004 new HTR fuel irradiations in the HFR Petten, with focus on determining the limits of old and newly produced fuel in terms of temperature and burn-up. Use in advanced pebble bed HTRs with very high coolant outlet temperature (up to 1000°C) and improved sustainability are key goals. A first experiment with 5 German AVR fuel pebbles (HFR-EU1bis, [2], [3]) was completed in 2005 and followed by a second (HFR-EU1,[4], [5]) completed in 2010, which investigated higher burn-up tolerance of existing German pebbles and of newly produced Chinese fuel.

<table>
<thead>
<tr>
<th></th>
<th>HFR-EU1bis</th>
<th>HFR-EU1</th>
</tr>
</thead>
<tbody>
<tr>
<td>pebble number and type</td>
<td>5 AVR</td>
<td>2 INET + 3 AVR</td>
</tr>
<tr>
<td>start</td>
<td>09.09.2004</td>
<td>30.09.2006</td>
</tr>
<tr>
<td>end</td>
<td>18.10.2005</td>
<td>19.02.2010</td>
</tr>
<tr>
<td>duration [efpd]</td>
<td>250</td>
<td>445</td>
</tr>
<tr>
<td>burn-up [%FIMA]</td>
<td>9.34 – 11.07 (measured)</td>
<td>11 – 14 (to be confirmed)</td>
</tr>
<tr>
<td>Temperature [°C]</td>
<td>1250 (central)</td>
<td>900 (INET) 950 (AVR)</td>
</tr>
<tr>
<td>R/B* at end of irradiation (based on ^85Kr release)</td>
<td>approx. 4×10⁻⁶</td>
<td>approx. 5.5×10⁻⁹ (INET) approx. 1.6×10⁻⁷ (AVR)</td>
</tr>
</tbody>
</table>

* Release over Birth (Production) rates

2.2 Comments on the high burnup HFR-EU1 irradiation

In this experiment, the device held 5 pebbles irradiated in 2 separately controlled capsules. Two of the pebbles were from recent Chinese production (INET), the other three from earlier German production (AVR). The fuel surface temperatures were kept constant at 900°C (INET) and 950°C (AVR). These conditions are more benign for the fuel, because increasing burn-up results in decreasing central fuel temperature with time. The burn-up reached was 11% FIMA (INET) and 14% FIMA (AVR) which is significantly higher than the license limit for HTR fuel (approx. 8% FIMA). Fission gas release in this experiment was measured with a newly installed on-line gamma spectrometry installation which allowed permanent monitoring of the capsules. Until the end of irradiation, the R/B values remained consistently low in both capsules indicative of the absence of particle failure even at these high burn-ups. The experiment was recently dismantled and is awaiting further PIE and KÜFA testing as part of a newly proposed European collaborative project.
2.3 High Temperature post-irradiation results

The KÜFA (cold finger apparatus) [6] is an out of pile testing device, now operated by JRC-ITU [7], and is used to study fission gas and volatile FP release from irradiated HTR fuel elements under simulated loss-of-coolant accident conditions. Since its installation, in 2006, 8 pebbles from irradiation campaigns performed in AVR and in HFR were characterized [8], [9], [10]. These KÜFA experiments performed to date on pebbles from controlled irradiation campaigns confirm that no release occurs at temperatures corresponding to normal operating regimes in pile (up to 1250°C) and suggests that the fractional release of all FP remains below 10^-4 in realistic accident scenarios.

An example of KÜFA results is shown in Fig. 2: the measured $^{137}$Cs and $^{85}$Kr release curves from pebble fuel element HFR K6/3 are plotted as a function of time together with the applied temperature programme. This pebble had been irradiated for 633 EFPD in the campaign HFR-K6, which also included steam injection [11]and had achieved a burn-up of 9.7% FIMA [8].

![On-line activity for Kr 88](image)

Fig. 1: Typical Kr-88 release plot hinting at undamaged particles in HFR-EU1
The results of the KÜFA test confirmed excellent coated particle (CP) performance with CP failure detected only during the second 1800°C heating period (indicated by the increased $^{85}$Kr release after ~450h) and a low Cs fractional release ($10^{-6}$) up to 1700°C.

The spectrum of available tools for PIE of HTR fuel at JRC-ITU is complemented by non-destructive and destructive techniques including burnup determination by gamma-spectroscopy and chemical analysis, fuel element sectioning and ceramography, and electrochemical deconsolidation of the fuel element for recuperation of individual coated particles and subsequent characterization. As a complement of KÜFA for experimental accident simulation studies on fuel elements, a furnace system to expose the pebbles to aggressive atmospheres at high temperature derived from the Kora (or "Korroisionsapparatur") concept conceived in the Jülich research centre [12] has been built and is operational for "cold" testing in view of application to irradiated fuel elements. Among the new tools being developed for application on irradiated HTR fuel, an irradiated microsphere gamma analyzer (IMGA [13]) system for the automatic sorting of intact and defective CP based on gamma-spectroscopy is currently being optimized for hot cell installation.

3. TRISO particles fabrication and characterisation

3.1 Fuel fabrication and characterisation equipment

A number of laboratories worldwide are concentrating on UO$_2$ and UCO fissile materials with a view towards immediate industrialisation. Within the JRC’s project “Safety of Advanced Nuclear Fuels” fabrication, characterisation and irradiation tests of advanced HTR fuels is being pursued to investigate the HTR's capability as a Pu (and eventually minor actinide) transmutation device. This could enable a more sustainable fuel cycle possibly based on the use of Th as a fertile material.

At the JRC-ITU both external and internal gelation methods have been tested for the production of kernels. In a novel adaptation of the internal gelation process the hot silicon oil as the means to transfer heat for the thermal decomposition of hexamethylenetetramine (HMTA) has been replaced by tetrachloroethylene using a continuous flow passing a two phase nozzle.
JRC-ITU is also in the process of commissioning a chemical vapour deposition (CVD) coater for the production of coated particles, mounted entirely in a glove box. The system is fully automated enabling personnel to observe the operation remotely. Size and sphericity is determined using an automated image analysis system (PICTIS, Sympatec GmbH, Clausthal-Zellerfeld, Germany) capable of measuring more than 100,000 particles per second. The isotropy of pyrocarbon layers is determined using the 2 modular general ellipsometry microscope (2MGEM) method originally developed at ORNL, but now commercialised by Hinds Instruments, Inc., Seattle, USA.

Fig. 3 CVD system at JRC-ITU for coated particle production

Fig. 4 Coated particles: first attempts (left) and optimized production (right)
4. Conclusion

Europe has set itself ambitious targets in terms of climate change mitigation and security of energy supply as laid down in the Strategic Energy Technology Plan. In the Strategic Research Agenda set up by the Sustainable Nuclear Energy Technology Platform, HTR is recognized as an energy system with a near-term potential market for cogeneration of heat and power (in particular for energy-intensive industries) and concomitant CO₂ savings of at least the same order of magnitude as the potential electricity market. HTR fuel R&D is subject of a number of international collaboration frameworks involving JRC, including the Generation IV International Forum and a number of active bilateral agreements in particular with countries such as China which have already embarked on ambitious demonstration projects.

The safety performance of HTR is tightly linked to the fission product retention capability of its fuel in normal and accidental situations. Consequently, for the successful demonstration and deployment of this technology, HTR fuel testing and qualification in view of licensing are rare and key capabilities. These JRC activities are fully integrated and complementary with European collaborative projects.

5. References

[1] www.snetp.eu


U$_{1-y}$Am$_y$O$_{2-x}$ (y=0.10; 0.15; 0.20; 0.30) COMPOUNDS OBTAINED IN VARIOUS SINTERING CONDITIONS

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KEYWORD
Americium, Transmutation, Fuel, Thermodynamical model, Sintering

ABSTRACT

A thermochemical study based on the model of Lindemer and Bessman was developed to determine the sintering conditions of U$_{1-y}$Am$_y$O$_{2-x}$ (y=0.10; 0.15; 0.20; 0.30) compounds. Hyperstoichiometric conditions at low temperature and reducing conditions at high temperature were investigated.

Hyperstoichiometric conditions sintering test were unsuccessful up to now because the compounds were multiphasic and exhibit low densities (<85%). In the case of the reducing conditions, the sintered densities are superior to 92% and monophasic compounds were obtained.
1. Introduction

Minor actinides (MA) significantly contribute to the radiotoxicity and heat generation of spent nuclear fuel. Two routes for MA recycling are studied [1]. The homogeneous recycling consists in adding MA up to a few percents in UPu fuel to be used in the whole core of the reactor while for the heterogeneous recycling, it is based on larger amounts in uranium oxide and concentrated into specific assemblies at the periphery of the core [2]. This work focuses on this latter case and especially on U$_{1-y}$Am$_y$O$_{2-x}$ “blanks” compounds.

Similarly to UOX and MOX fuels, Am doped fuels properties strongly depend on the final stoichiometry. However, these materials exhibit a significantly different thermodynamical behaviour in comparison to conventional fuels. As a consequence, a thermodynamical modelling is currently developed to determine optimum sintering conditions. Although the fabrication of U$_{1-y}$Am$_y$O$_{2-x}$ has been achieved at the laboratory scale, this step remains difficult due to the particular thermodynamical properties and high α activity of Am [3,4] which has also to be considered for material ageing.

The present work is devoted to the fabrication and characterisation of U$_{1-y}$Am$_y$O$_{2-x}$ ($y=0.10; 0.20; 0.30$) sintered in various sintering conditions. These compounds were sintered either in hyperstoichiometric conditions at low temperature or in reducing conditions at high temperature. Besides, two types of microstructure such as dense and tailored porosity, to favour the He release, were investigated.

2. Thermodynamical modelling of U$_{1-y}$Am$_y$O$_{2x}$

AmO$_{2-x}$ exhibit very high oxygen potentials compared to others actinides compounds such as UO$_{2+x}$. As a result, for conventional fabrication process, this may lead to an excessive reduction of the oxide even at low temperature. If this reduction is not properly controlled, volatilization of a significant amount of americium may occur. Therefore, an accurate control of the sintering atmosphere is necessary to achieve the targeted O/M value. A thermodynamical approach based on the model of Lindemer and Bessman [5,6] has been developed to determine optimum sintering conditions.

Figure 1 (a-b) gives the O/M ratio for U$_{0.9}$Am$_{0.1}$O$_{2±x}$ and U$_{0.8}$Am$_{0.2}$O$_{2±x}$ versus the temperature and ∆G$_{O_2}$. It can be seen that for achieving U$_{0.9}$Am$_{0.1}$O$_{2.00}$ and U$_{0.8}$Am$_{0.2}$O$_{2.00}$ at 1750°C, ∆G$_{O_2}$ must be respectively equals to -240 and -200 kJ.mol$^{-1}$. These results highlight the fact that for a given O/M ratio, ∆G$_{O_2}$ increases with the Am content. This phenomenon is more important in U$_{1-y}$Am$_y$O$_{2x}$ than in U$_{1-y}$Pu$_y$O$_{2x}$ fuel.

![Figure 1: Oxygen potential ∆G$_{O_2}$ versus the temperature and the O/M calculated by thermodynamical modelling of of (a) U$_{0.9}$Am$_{0.1}$O$_{2x}$ and (b) U$_{0.8}$Am$_{0.2}$O$_{2x}$](image-url)
Although this modelling still requires to be enhanced, it points out that the two sintering conditions presented in Figure 2 must be investigated:
- hyperstoichiometric conditions at low temperature [7];
- reducing conditions at high temperature with highly humidified gas.

![Figure 2: Sintering conditions determined with the thermodynamical modelling](image)

3. **Pellet fabrication**

Pellet fabrication was carried in the ATALANTE hot cells in neutral atmosphere. This atmosphere was maintained during storage in order to prevent any further oxidation at room temperature. Firstly, an AmO$_2$-UO$_2$ blend was ball milled for 30 min using an oscillating ball-mill with stainless steel container. Then, the remaining UO$_2$ was added to adjust the composition and the last milling step was carried for 30 min. Pellets were shaped by uniaxial pressing at 400 MPa and sintered either in reducing or hyperstoichiometric conditions. For the tailored porosity compounds, a pore former was used in order to favour the creation of an interconnected porosity.

4. **$U_{1-y}Am_yO_{2+x}$ ($y=0.1:0.2:0.3$) characterization**

4.1 **Pellet characteristics**

The pellet sintered densities are represented versus Am content in Figure 3. A visual inspection of pellets was performed to evaluate the possible occurrence of cracks. It indicates that the compounds sintered in hyperstoichiometric conditions were cracked and had low densities (<85%) whereas the compounds sintered in reducing conditions had regular shapes and higher densities (>88%).

![Figure 3: Sintered density (%) versus Am content (%) of dense compounds](image)
4.2 Optical microscopy

Optical microscopy was performed on as-polished samples using an OLYMPUS BX30M microscope.

Figure 4 shows microstructures of as-polished surfaces of pellet sintered in reducing conditions. It can be seen that the compound is homogeneous while in the case of the hyperstoichiometric conditions heterogeneous compounds were observed [7].

![Microstructure Image](image)

Figure 4: Microstructure (x5) of U_{0.90}Am_{0.10}O_{2-x} sintered in reducing conditions

4.3 XRD Analysis

XRD analyses were carried out using a Bruker D8 Advance analyzer (molybdenum Kα with a specific sample holder for radioactive materials. An internal measurement standard (Au) was added for 2θ calibration of the XRD peak position and intensity corrections. The EVA and FULLPROF [8] program were used for lattice parameter refinement and peak intensity.

![XRD Patterns](image)

Figure 5: XRD patterns of U_{0.90}Am_{0.10}O_{2-x} sintered either in (a) reducing or (b) hyperstoichiometric conditions (▲: Am_2O_3, ■: UO_{2.16}, ×: AmO_2, *: Au)
XRD analyses confirm that the compound sintered in hyperstoichiometric conditions was multiphasic and heterogeneous. With a minor \( \text{UO}_{2+x} \) phase, a major fluorite phase is present in the sample sintered in hyperstoichiometric conditions. The lattice parameter of this phase was refined as 0.5466\( \text{nm} \). Besides, a very small quantity of an additional fluorite phase is observed with a significantly larger lattice parameter of 0.5574\( \text{nm} \). The simultaneous presence of these phases is probably due to incomplete sintering reactions.

In the case of the reducing conditions, a monophasic and homogeneous compound was obtained. The lattice parameter of this fluorite single-phase is equal to 0.5469\( \text{nm} \).

5. Conclusion

This work focuses on the fabrication of \( \text{U}_{1-y}\text{Am}_y\text{O}_{2-x} \) fuels in various sintering conditions. These materials exhibit a significantly different thermodynamical behaviour in comparison to conventional UOX and MOX fuels due to the \( \text{AmO}_{2-x} \) very high oxygen potentials. As a consequence, a thermodynamical modelling based on the Lindemer and Bessman model was developed putting forward the necessity to study two types of sintering conditions:
- hyperstoichiometric conditions at low temperature;
- reducing conditions at high temperature with highly humidified gas.

Pellets fabrications were carried in the ATALANTE hot cells in the above mentioned conditions with two microstructures, dense and tailored porosity. It appears that hyperstoichiometric conditions sintering tests were unsuccessful up to now because multiphasic and heterogeneous compounds with low densities were obtained. Reducing conditions are more suitable for the fabrication. Indeed, the compounds sintered in these conditions are homogeneous and monophasic and present higher densities.

6. References


[3] A. Jankowiak & al., Structural study of \( \text{Pu}_{1-x}\text{Am}_x\text{O}_2 \) \((x = 0.2; 0.5; 0.8)\) obtained by oxalate co-conversion, Journal of Nuclear Materials, vol. 393 (2009), pp. 87-91.


POSSIBILITIES OF CLOSING NUCLEAR FUEL CYCLE WITH INNOVATIVE METMET FUEL

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ABSTRACT

A possibility is considered for use in thermal and fast reactors in place of the base container type UO$_2$ or MOX fuel, the dispersion type fuel elements (high uranium content fuel, U-Mo, U-Nb-Zr, U$_3$Si with Zr alloy matrix). The use of the novel fuel in thermal reactors might result in 500 additional effective days prolongation of the campaign of fuels or in a reduced enrichment of fuel which leads to natural uranium saving as well as to increasing the conversion factor. Additionally Pu content increases more that twice in comparison with uranium dioxide pellet that is very important for closing nuclear fuel cycle.

On the basis of METMET fuel combined U-PuO$_2$ fuel (an analogue of MOX) can be fabricated where depleted uranium alloy and dioxide plutonium powder have initially separate arrangement. Due to this as compared to MOX fuel the new one features higher thermal conductivity, higher uranium content, hence, high conversion ratio, does not interact with fuel cladding and is more environmentally clean. Combined fuel is intended for use in fast reactors as it features higher characteristics in comparison to metallic or MOX fuel.

Novel approach to reprocessing of combined fuel is demonstrated, which allows the separation of uranium from plutonium as well as the newly generated fissile plutonium from burnt one without chemical processes with repeated use in fast, PWR or CANDU reactors, which simplifies the closing of the nuclear fuel cycle. Hence in comparison to MOX we can multiply use of generated Pu in spent fuel and instead of partial recycling implement full recycling which drastically reduces the fuel waste.

Thus, the novel concept of using novel dispersion METMET fuel in thermal or fast reactors might in future replace the currently existent approach based on the application of pelletized UO$_2$ or MOX fuel.

1. Introduction

To accomplish a further qualitative ramp in novel generation fuel development and to provide for the competitiveness of electric power production we suggest to replace the container design fuel rod, the possibilities of which are practically exhausted, to dispersion type fuel elements (high uranium content fuel, U-Mo, U-Nb-Zr, U$_3$Si with Zr alloy matrix) [1-4]. The dispersion type fuel is known to have a high irradiation resistance and thermal conductivity, hence low operating temperatures and can reach high burn-ups [4].

Structurally the dispersion fuel meat consists of uniformly distributed higher density fuel granules of U-Mo, U-Nb-Zr or U$_3$Si alloys that are metallurgically bonded between themselves and to fuel cladding with specially developed Zr-based matrix alloys having the melting temperatures of 790–860°C (fig. 1). In this case a fuel meat retains controllable porosity in the range of 14- 22% to accommodate fuel swelling. As applied to thermal reactors the novel fuel will have much better neutronics characteristics which in the end will reduce the cost of produced electrical energy.
The same approach might be applied to designing fast reactor fuels using a more environmentally clean production process (fig. c) [5, 6]. In this case the conversion ratio of novel fuel is higher than that of MOX fuel and is at the level of more high density metal fuel.

Fig. 1. Micro and macrostructures of fuel compositions, produced by capillary impregnation method: (a) U-Mo + Zr matrix; (b) U-Zr-Nb + Zr matrix (c) combined U(Th)-PuO₂ fuel [4-6]

2. **Design and Fabrication Process of Fuel Element for VVER, PWR, RBMK, CANDU (HWPR)**

Dispersion type fuel element with high uranium content can be produced by capillary impregnation method. It involves vibrofilling a zirconium cladding of a fuel element with blended fuel and matrix granules (fig. 2a) and a capillary impregnation – a short-term (1-5 minutes) anneal at 840–910°C (fig. 2b). At those temperatures zirconium matrix alloy melts down and under capillary forces moves into gaps between fuel granules as well as the space between fuel and cladding resulting in metallurgical bonds which promotes a high thermal conductivity of a fuel meat.

Fig. 2. Schematic cross-section representation of fuel element fabricated by capillary impregnation method; (a) as vibroloaded; (b) as capillary impregnated and appearance of fuel elements produced by capillary impregnation method

Properties of fuel compositions is presented in table 1 and microstructures in fig. 3 [2-4]. In the fuel elements the volume fraction of the fuel is 66-72 %. Hence, with the use of high uranium content fuel the uranium content reaches 9.5-12.9 g·cm⁻³. High thermal conductivity of the fuel compositions which combines with the metallurgical bond with the cladding is available, ensures the low operating temperature in the fuel element centre (cold fuel). Fuel compositions due to Zr alloy matrix show high corrosion resistance. Aqueous corrosion rate at 330 °C is 0.02-0.05 g·m⁻²·h⁻¹.
<table>
<thead>
<tr>
<th>Fuel</th>
<th>U\textsubscript{3}Si</th>
<th>U-9Mo</th>
<th>U-1.5Mo-1.0Zr</th>
<th>U-5Nb-5Zr</th>
<th>U-3Nb-1.5Zr</th>
<th>UO\textsubscript{2} pellet</th>
</tr>
</thead>
<tbody>
<tr>
<td>U content in fuel composition at volume fraction of fuel</td>
<td>66%</td>
<td>9.6</td>
<td>10.7</td>
<td>11.9</td>
<td>9.8</td>
<td>11.34</td>
</tr>
<tr>
<td>Thermal conductivity at 500 °C, W·m\textsuperscript{-1}·K\textsuperscript{-1}</td>
<td>72%</td>
<td>10.4</td>
<td>11.7</td>
<td>12.9</td>
<td>10.7</td>
<td>12.37</td>
</tr>
<tr>
<td>Interaction layer after annealing at 750 °C for 6000 h, μm</td>
<td>19</td>
<td>22</td>
<td>24</td>
<td>18</td>
<td>21</td>
<td></td>
</tr>
<tr>
<td>Corrosion rate in water at 330°C (g/m\textsuperscript{2}·h)</td>
<td>7-10</td>
<td>10-15</td>
<td>15-25</td>
<td>0.03</td>
<td>0.05</td>
<td>0.02</td>
</tr>
</tbody>
</table>

Tab 1: Properties of fuel compositions [2, 3]

Fig. 3. Micro and macrostructure of modified fuel composition with higher uranium content (72% volume fraction of fuel under the cladding)

**Basic advantages of novel fuel for use in thermal reactors (PWR, BWR, and VVER):**
1. High uranium content (9.5 – 12.9 g/cm\textsuperscript{3}) within fuel element cladding, that is 15-50 % more than the uranium content of the standard PWR, BWR and VVER-1000 fuel rods, which allows the uranium enrichment of fuel to be reduced or the burn-up to be increased. It is also improves the neutronics characteristics and safety of a reactor and increases the conversion ratio. Generated of Uranium large quantities of Plutonium involved in the process of fission, leads to significant savings of enriched uranium, which in the end favourably affects the economics of nuclear power. The conversion ratio increases from 0.55 to 0.7. Therefore the prolongation of the company will be more than 30% in effective days (up to 500 additional effective days), or uranium enrichment can be reduced from 4.95 to 3.7% (fig.4-5).

![U content in a VVER-1000 fuel element, g](image)

1300 1725 2050 2150
UO\textsubscript{2} U\textsubscript{3}Si UZrNb UMoZr

**Initial enrichment 4.95%**

![Content of burnt and remained Pu, per fuel element](image)

1525 effective days, UO\textsubscript{2}

K\textsubscript{c}=0.55

K\textsubscript{c}=0.7

UO\textsubscript{2}

![Fissionable Pu isotopes](image)

68%

75%

METMET

![Fig. 4 Comparative assessed neutron-physical characteristics of novel METMET and dioxide uranium fuel for VVER-1000 reactor at burn-up of 65 MWD/kgU](image)
The advantage of the fuel higher reactivity might be used to increase the time between refueling, in other words, to increase the Unit Capacity Factor (UCF). In this case instead of a year or a year and a half fuel cycle at two year cycle becomes feasible.

At the life time end the plutonium content reaches 2% instead of 1.3% in a uranium dioxide pellet. Due to the higher density fuel the total Pu content in fuel element is 2.5-3 times higher and makes up 45 g per a single fuel element. In this case if fissionable Pu isotopes make up 68-70% in spent UO₂, their content in the novel fuel is 75-78%. In other words, after reprocessing the plutonium content will be sufficient for fabricating a factor of 3 more MOX fuels than in case of the standard fuel which is very important in closing the nuclear fuel cycle (fig. 4).

Aside from this due to the specific features of the design the novel fuel might be reused in RBMK, CANDU reactors after minimal reprocessing (without the chemical reprocessing of fuel).

Fig. 5. Advantages of using novel dispersion high density uranium fuel in VVER-1000 type reactor [2-4]

2. Low temperature of the fuel, which satisfies the requirements to cold fuel.
3. The porosity in the fuel meat will allow the accommodation of swelling up to the burn-up of 1.0 g-fiss/cm³, which in terms of the standard VVER-1000 fuel rod corresponds to 120 MW*d/kgU.
4. Metallurgical bond between the fuel and cladding makes fuel elements serviceable under severe transients, that leads to optimization of Nuclear Plant operation conditions and improvements of their operation reliability and safety.

These advantages of novel generation fuel might result in the increasing of economic efficiency and degrease of the cost of electric power.

3. Design and Fabrication Process of Alternative to MOX Combined U(Th)-PuO₂ Fuel for PWR, VVER, CANDU (HWPR)

The major approach to fuel element development consists in individual operations used to fabricate a fuel element with uranium meat and to fill it with plutonium dioxide powder which minimizes dust producing operations in a fuel element fabrication [3].

At the first stage under conventional conditions of a plant a fuel element frame of dump uranium bonded with Zr matrix alloy is fabricated using described above capillary impregnation method. The fuel meat open porosity to be increased up to 25 – 30%. In this way a porous uranium meat (frame fuel element) is created that has the uranium content of 9.0 – 9.5 g/cm³ and the thermal conductivity of 16-18 W/m-K at 450°C and 30 W/m-K at
700°C (fig. 6a). Then, through the fuel column via open pores the standard plutonium dioxide powder up to 100 μm fraction is introduced that produced by pyrochemical or other method. Then a fuel element is sealed. In this way plutonium dioxide granules located in the pores of a heat conducting uranium fuel frame (fig. 6c) [2, 4].

The advantages of this fuel element as an alternative to the MOX one are:
1. The process of the fuel element fabrication is environmentally more friendly due to degrease the quantity of technologic operations with plutonium.
2. The higher uranium content of a fuel element, hence, the higher conversion ratio.
3. Low temperature of fuel (cold fuel).
4. Serviceability under transient conditions.
5. Possibility of Pu isotopes separation.

![Fig. 6. Structure of fuel meat prior (a) and after (c) filling with PuO₂ powder manufactured by pyrochemical method (b)](image)

Microstructure of combined fuel, where UO₂ is used in place of PuO₂ is illustrated in Fig. 7. In this option granules of Zr matrix and UO₂ were loaded simultaneously into fuel element cladding before annealing

![Fig. 7. Microstructure of combined fuel (UO₂ is used in place of PuO₂) [2, 4]](image)

This approach can be also implemented with the use of Th granules in place of dump uranium alloy granules. In this case the conversation ration increased. For CANDU reactors it is exceeds 1. Therefore this type of reactor can reach the burn up of 100 MW·d/kgU. Then the fuel granules may be extracted out of a fuel element via crushing the fuel meat and reused without the chemical reprocessing of fuel.

Mechanism of combined fuel operation: first PuO₂ burns up in METMET fuel frame while Pu generates in the METMET fuel that first serves as a breeding blanket and then begins to burn up. That is why, the components of the combined fuel have different Pu isotope compositions at the cycle end. The plutonium dioxide powder will basically accumulate non-fissionable isotopes while the METMET fuel – fissionable ones (fig. 6). Hence, at the average MOX fuel burn-up of 45 MW·d/kgU, the METMET fuel shall contain 75 % fissionable isotopes while in PuO₂ there will be only 42 % (table 2). By the mechanical separation METMET frame from PuO₂ powder the other without chemical reprocessing the combined fuel may be used many times at a high efficiency in this way saving natural uranium. In MOX fuel at the above
burn-up the concentration of fissionable isotopes makes up only 55%, hence, its re-use is economically not beneficial.

It has also to be noted that in the combined fuel the burn-up is distributed about equally between the PuO$_2$ powder and the granules of uranium alloys therefore the mechanically extracted granules will contain about two times less fission fragments having a high parasitic neutrons capture.

<table>
<thead>
<tr>
<th>Initial fuel MOX or PuO$_2$ powder</th>
<th>Spent MOX fuel</th>
<th>Spent combined U-PuO$_2$ fuel</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Average</td>
</tr>
<tr>
<td>66%</td>
<td>55%</td>
<td>58%</td>
</tr>
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</table>

Tab 2: Assessed Quantities of Fissionable Isotopes in Combined and MOX Fuel of VVER-1000 Operated on Three Year Cycle as Applied to Fuel Cycle Closing

4. Design of Fuel Element for Fast Reactors and to Burn MA

For fast reactors the same design and process are used, but to enlarge the plutonium dioxide fraction in a fuel element the porosity is increased to 30 – 38% (fig. 8a) and in place of zirconium claddings the steel ones are employed. The zirconium matrix alloy coating that results from the capillary impregnation at the inner surface of a cladding protects it from interacting with both fuel and fission products [5-6].

Fig. 8. Structure of combined U-PuO$_2$ fuel meat for fast reactors and MA incineration (a), internal zirconium alloy coat at inner surface of steel cladding (b) [4-6]

For consumption of MA a powder of their oxides is filled in instead of plutonium dioxide. If the fuel meat does not contain fuel (zirconium granules are used) a version of an inert matrix fuel element results (IFM) to burn up both Pu and MA in thermal or fast reactors [7]. The combined fuel retains the advantages inherent in metallic and ceramic types of fuel. Since the contribution into the total burn-up is made not only by the metallic fuel at the final irradiation stage as plutonium is generated, but also by plutonium dioxide at the initial irradiation stage, the total swelling of the metallic fuel is reduced due to a less build-up of fission fragments in it.

The suggested U-PuO$_2$ combined fuel having a high conversion factor and thermal conductivity like metallic fuel does not interact with a fuel cladding since Zr matrix coats the cladding and protects against fuel-cladding interaction. Fabrication technology is simple and environment-friendly.

Microstructure of such combined fuel, where UO$_2$ is used in place of PuO$_2$ is illustrated in fig. 9. In this option granules of Zr matrix and UO$_2$ were loaded simultaneously into fuel element cladding before annealing (capillary impregnation).
Major advantages of combined U-PuO₂ fuel for fast reactors:
High contents of U and Pu compared to MOX fuel and, hence, high conversion ratio which makes it feasible to close nuclear fuel cycle. 
Lower damage of fuel by fission products and, hence, lower swelling compared to that of U-Pu-Zr fuel. 
Novel fuel is dispersion type fuel, therefore the existence of metallurgical bond between fuel and cladding not only decrease fuel temperature, but also protects fuel cladding from interaction with fuel and fission products. 
High fabrication adaptability and ecology of production – actually the main part of fuel element fabrication is carried on under conventional conditions and only the final operation of fuel element fabrication needs remote implementation. In this case PuO₂ is used as a powder not as pellets. All this minimizes process operations with Pu and makes the fuel element fabrication environmentally clean.

To burn up MA in IFM of thermal or fast reactors a fuel design is suggested having on isolated location of Pu or MA, fig. 10 [7]. Fuel element completed by fuel minielements. Fuel minielements is loaded with powder of PuO₂ or MA and sealed that comply the requirements for ‘Rock Fuel’. In this way the quantity of dust producing technologic operations is minimized. Fuel elements are fabricated by impregnation or capillary impregnation methods using a molten Al or Zr matrix alloys.

5. Reprocessing of METMET and Combined U(Th)-PuO₂ fuels

Novel fuel dissolves in hot nitric acid (typical reprocessing) as content of Zr matrix alloy in fuel meat is low, only 10-15% of fuel meat volume. In addition due to large volume fraction of intermetallic phases, available in the structure of Zr matrix alloy it is also dissolves in acid (but not so rapidly as uranium fuel itself).

The other option of fissile phase separation without chemical reprocessing is application DUPIC type process, but simpler one [4-6]. First, like in DUPIC via heating gas fission products are removed from fuel. Then a fuel element is subjected to a light deformation. Zr alloy matrix bridges between fuel granules are destroyed and fuel element content is emptied on to a sieve where finer fractions of zirconium matrix alloy are separated from coarser granules of metallic fuel that also contains generated Pu (Fig. 11) [4-6]. Evoked from cladding granules of U-Pu alloy can be used repeatedly for PWR, BWR, RBMR and CANDU reactors using capillary impregnation method as they have high content of generated Pu (2-3%).
The same approach can be implemented for combined fuel [6]. The basic difference of such fuel is the separated arrangement of a depleted uranium alloy and PuO₂ powder. Due to this fact the degree of their burn-up and their isotope composition will be differ. While PuO₂ will almost fully burn and its composition will not practically contain fissile isotopes the metal fuel will build up Pu with fissile isotopes prevailing and having a high power value. Hence, as distinct from the standard design of a fuel element with homogeneously distributed heavy atoms (MOX or U-Pu-Zr), where the separation of fissile isotopes in Pu is a complicated task, in the novel fuel they are separated initially. Therefore fissile elements separation results without chemical processes.

Thus, as distinct from the DUPIC the novel combined fuel does not require pulverization (pounding) of burnt pellets, an intricate process of manufacturing new fuel pellets from high activity powder. Moreover, in the new process the valuable in terms of the power plutonium isotopes (Pu239 and Pu241) are for the most part separated from Pu 242 and Pu 240. This makes fuel reprocessing more workable and environmentally friendly, which simplifies the closing of the nuclear fuel cycle.

6. Conclusion

A possibility is considered for use in thermal and fast reactors in place of the base container type UO₂ or MOX fuel, the dispersion type fuel elements (high uranium content fuel, U-Mo, U-Nb-Zr, U₃Si with Zr alloy matrix). The use of the novel fuel in thermal reactors might result in 500 additional effective days prolongation of the campaign of fuels or in a reduced enrichment of fuel which leads to natural uranium saving as well as to increasing the conversion factor. Additionally Pu content increases more that twice in comparison with uranium dioxide pellet that is very important for closing nuclear fuel cycle.

On the basis of METMET fuel combined U-PuO₂ fuel (an analogue of MOX) can be fabricated where depleted uranium alloy and dioxide plutonium powder have initially separate arrangement. Due to this as compared to MOX fuel the new one features higher thermal conductivity, higher uranium content, hence, high conversion ratio does not interact with fuel cladding and is more environmentally clean. Combined fuel is intended for use in fast reactors as it features higher characteristics in comparison to metallic or MOX fuel.

Novel approach to reprocessing of combined fuel is demonstrated, which allows the separation of uranium from plutonium as well as the newly generated fissile plutonium from burnt one without chemical processes with repeated use in fast, PWR or CANDU reactors, which simplifies the closing of the nuclear fuel cycle. Hence in comparison to MOX we can
multiply use of generated Pu in spent fuel and instead of partial recycling implement full recycling which drastically reduces the fuel waste. Thus, the novel concept of using novel dispersion METMET fuel in thermal or fast reactors might in future replace the currently existent approach based on the application of pelletized UO₂ or MOX fuel.

7. References


1 Introduction

Alkylated 2,6-ditriazinylpyridines (BTP) are the first N-donor extractants to extract Am(III) and Cm(III) selectively over the chemically similar lanthanides(III) from up to 1 M nitric acid [1]. This separation is essential to the partitioning and transmutation strategy [2], which aims at reducing the long-term radiotoxicity of spent nuclear fuel by separating minor actinides and transmuting them in dedicated nuclear reactors. Despite the experimentally proven high separation selectivity of BTPs a molecular level understanding of the driving forces is not yet available. We have previously investigated the extraction and coordination structure of Am(III) and Cm(III) with 2,6-di(5,6-dipropyl-1,2,4-triazin-3-yl)pyridine (n-C$_3$H$_7$-BTP) [3-8]. Information on the corresponding behavior of Np(III) and Pu(III) is missing, which lie between U(III), [4-5] representing a “softer” actinide, and the “harder” lanthanide-like actinides Am(III) and Cm(III). Extending our previous study on the complexation and liquid/liquid extraction of trivalent actinides and lanthanides with BTP [1, 3], we perform analogous studies with Pu(III). The Pu(III)-complex is investigated by UV-Vis and extended X-ray absorption fine structure (EXAFS) in organic solution, and the extraction behaviour of Pu(III) with n-C$_3$H$_7$-BTP is studied as a function of n-C$_3$H$_7$-BTP concentration.

2 Experimental section

2.1 Chemicals and sample preparation

All chemicals are of p.a. quality or better and are obtained from Merck (Darmstadt, Germany) or Riedel de Haen (Seelze, Germany). All of the experiments are conducted using de-ionized, “MilliQ” water (ρ = 18 MΩ•m). $^{242}$Pu, $^{241}$Am, and $^{243}$Am are from laboratory stock solutions. The purity and activity of $^{242}$Pu is checked by liquid scintillation counting (LSC) using the scintillation cocktail Ultima Gold XR (Packard). n-C$_3$H$_7$-BTP is dissolved at varying concentrations in kerosene/1-octanol (70:30 vol.; kerosene from Prochrom, France).

We use 0.01 M Rongalite [9] (sodium hydroxymethylsulfinate, HOCH$_2$SO$_2$Na) as a reducing agent for the preparation of $^{242}$Pu(III) from a Pu(IV, VI) solution. The Pu oxidation state is confirmed by UV-Vis/NIR spectroscopy [10].

A solution of 4.0 mM Pu(III) in 0.1 M HNO$_3$ + 1.9 M NH$_4$NO$_3$ is diluted to 1.1 mM Pu(III) by addition of 0.1 M HNO$_3$ + 1.9 M NH$_4$NO$_3$ solution. This solution is spiked with $^{241}$Am(III) (approx. 1 kBq/mL). 2 mL of this aqueous solution and 2 mL of 10 mM / 20 mM / 40 mM / 60 mM n-C$_3$H$_7$-BTP in kerosene/1-octanol (70:30 vol.) are contacted in glass vials by vigorous shaking for 15 min. After centrifugation, phases are separated and transferred to new vials. After spectrophotometrically confirming the trivalent oxidation state in the organic phase, aliquots from both the organic and aqueous phases are analyzed by gamma spectrometry ($^{241}$Am) and ICP-MS ($^{242}$Pu). The experiments are performed in duplicate. An aliquot of the separated organic phase containing Pu(III) in 40 mM n-C$_3$H$_7$-BTP is used for EXAFS measurement.
2.2 Instrumentation

UV-Vis/NIR spectroscopy:
For absorption spectroscopy, a high-resolution UV-Vis/NIR spectrometer Cary 5 (Varian, USA) is used. The samples are measured in 10 mm quartz cuvettes. Wavelength ranges 400–1200 nm for $^{242}\text{Pu(III)}$ and 400–900 nm for $^{243}\text{Am(III)}$ are used. The spectra are deconvoluted using the graphical software Gram5.0 (for details see [11]).

XAFS spectroscopy:
XAFS measurements are performed at the INE-Beamline for actinide research [12] at the Ångströmquelle Karlsruhe (ANKA), Karlsruhe Institute of Technology, Germany. The ANKA storage ring is operated at 2.5 GeV electron energy with a mean electron current of 120 mA. A pair of Ge(422) crystals ($2d = 2.310 \text{ Å}$) is used in the double crystal monochromator (DCM). Higher harmonic radiation in the incident beam is suppressed by the two mirrors in the optics of the INE-Beamline and by detuning the parallel alignment of the crystals to 70% of maximum photon flux peak intensity at the beginning of each scan. The incident flux is measured by an Ar-filled ionization chamber at ambient pressure and held constant by a digital MOSTAB feedback unit.

For the XAFS measurements, solutions are filled into 400 µl capped PE vials and mounted in a special air tight sample holder, which is connected to an Ar supply line at the experimental station to keep the samples under inert, near oxygen-free conditions during XAFS measurements. The spectra are calibrated against the first derivative X-ray Absorption Near Edge Structure (XANES) spectrum of a Zr foil, defining the energy of the first inflection point as $E(\text{Zr 1s}) = 17.998 \text{ keV}$. All Pu L3 XAFS spectra are measured in standard fluorescence yield detection mode. Up to 10 scan are collected and averaged for each sample. Extended X-ray absorption fine structure (EXAFS, $\chi(k)$) data analysis is based on standard data reduction and least squares fit techniques [13] using the ATHENA [14] and the UWXAFS [15] program packages. The initial data analysis involves subtraction of the pre-edge background absorption and normalization of the edge jump to unity. The $k^2$-weighted $\chi(k)$ are obtained following $\mu_0$ spline function fit and subtraction, conversion of energy to $k$ using the white line energy to define the ionisation energy ($E_0$) and final $k^2$-weighting. Least square fits of the EXAFS are performed in R-space between 1.2 and 3.7 Å for two shell fits and between 1.2 and 5 Å for the four shell model using the feffit software [15].

3 Results and Discussion

3.1 Extraction of trivalent plutonium

The extraction of Pu(III) and Am(III) from acidic nitrate solution into n-C$_3$H$_7$-BTP in kerosene/1-octanol solution is investigated as a function of n-C$_3$H$_7$-BTP concentration. Figure 1 shows the dependence of the distribution ratios ($D$) of Am(III) and Pu(III) on the BTP concentration (the extraction experiments are spiked with Am(III) to accurately determine separation factors). The slopes of the log$D_{\text{Am(III)}}$ vs. log[BTP] plots are near three (2.6), which confirms the formation of [M(BTP)$_3$]$^{3+}$ complexes in the organic phase. The deviation from the expected slope of 3 is likely due to BTP oligomerization in the organic phase (see [1]). The separation factor (i.e., the quotient of distribution ratios for Am(III) and Pu(III) under identical experimental conditions, $\text{SF}_{\text{Am(III)}$/Pu(III)} = D_{\text{Am(III)}}/D_{\text{Pu(III)}}$) is 1.27, meaning that Am(III) is extracted better than Pu(III). This behaviour is comparable to that of the lighter Ln(III) elements, where we observe that cation extractability increases with increasing atomic number from La(III) to Eu(III) [7, 8].
Fig 1. Extraction of Pu(III) and Am(III) by n-C₃H₇-BTP, distribution ratio as a function of n-C₃H₇-BTP concentration. Organic phase: n-C₃H₇-BTP in kerosene/1-octanol (70:30 vol.).

3.2 Characterization of Pu(III) in n-C₃H₇-BTP solution by UV-Vis spectroscopy

The absorption spectra of the Pu(III)/n-C₃H₇-BTP complex (Fig 2) all exhibit an absorption band at 939 nm with a distinct shoulder, irrespective of ligand concentration. We conclude that upon extraction Pu(III) forms one species, independent of the ligand to metal ratio (L/M). The spectra of the complexes are also similar to that of the Pu(III) aquo species, but they are significantly red shifted by 36 nm. Because of the similarity of the spectral features for both the Pu(III) complex and the Pu(III) aquo species, we compare Pu(III) results with those for Am(III), which is redox stable under the experimental conditions. We extract Am(III) in a similar way and compare the UV-Vis/NIR spectrum for the organic phase with that for the aquo species. The absorption spectrum of the [Am(n-C₃H₇-BTP)₃]³⁺ complex shows an absorption band at 523 nm with a distinct shoulder. Similar to Pu(III), this band is also red shifted compared to the aquo species spectrum, in this case by 19 nm. This bathochromic shift results from strong complexation to the three n-C₃H₇-BTP ligands.

Fig 2. Absorption spectra of 1 mM Pu(III) in acidic nitrate solution (0.1 M HNO₃ + 1.9 M NH₄NO₃) and of extracted Pu(III)/n-C₃H₇-BTP complexes.
3.3 Characterization of Pu(III) in n-C₃H₇-BTP solution by XAFS

The speciation of Pu(III) with BTP ligands in organic solution is successfully characterized by XAFS spectroscopy. The k²-weighted EXAFS and corresponding Fourier transform (FT) for the Pu(III)-BTP complex in organic solution is shown in Figure 3. EXAFS data are initially modeled using two coordination shells to determine the number of coordinating atoms (CN). The value for CN in these initial fits is found to be 10 ± 1 N atoms for the first coordination shell and approximately two times as many C/N atoms in the second shell. This indicates that likely three n-C₃H₇-BTP ligands are bound to Pu(III). The EXAFS data are then modeled considering four coordination shells comprised of nine nearest N atoms directly bound to the Pu(III), second and third shells of 18 C/N atoms each (designated C/N and C'/N', respectively), and a fourth, most distant shell of 9 C atoms, representing C atoms located para to the coordinating hetero N atoms of the pyridine and triazine rings. The results of the fits are listed in Table 1 and compared to results previously reported for U(III), Am(III), and Cm(III). No evidence is found for the presence of any coordinated nitrate groups in the FTs of the fit residuals. Generally, a bidentate-coordinated nitrate ligand should have O atoms near approximately 2.45-2.55 Å and the associated nitrate N atoms would be expected to be located near 3 Å and the distal O atom near 4.2 Å. FT peaks associated with these distances are not observed here. Therefore, we conclude that the Pu(III) complex does not have nitrate directly coordinated, formally required for the charge neutralization.

Fig 3. k²-weighted Pu L3 EXAFS (experimental, filtered, and fitted data) and corresponding Fourier transform (FT) for [Pu(n-C₃H₇-BTP)₃]²⁺. Experimental EXAFS is shifted along the y-axis for clarity. Experimental and filtered data are shown as lines and fit curves as symbols. Imaginary parts of the FTs are shown.

4. Conclusions

In this study we expand our understanding of the partitioning of An(III) from Ln(III) using soft N-donor ligand extractants. We show by UV-Vis/NIR spectrophotometry that Pu(III) forms a strong complex to the three n-C₃H₇-BTP ligands in organic solution. Slope analysis of liquid/liquid extraction data proves the formation of the 1:3 complex [Pu(BTP)₃]²⁺ into the organic phase, in agreement with the behaviour of the respective Am(III) and Cm(III) complexes. The [Pu(n-C₃H₇-
BTP)₃³⁺ complex is isostructural to the previously studied [An(n-C₃H₇-BTP)]₃³⁺ complexes. The average bond distance between Pu(III) and ligating N atoms in [Pu(n-C₃H₇-BTP)]₃³⁺ is 2.56 Å and the same as those for U(III), Am(III), and Cm(III) within the experimental uncertainty of the EXAFS method. In other words, the interatomic distance of the trivalent actinide cations to the ligating N atoms of the pyridine and triazine rings in BTP is independent of which actinide is used, i.e., independent of the trivalent actinide cation ionic radius. Let us explore the difference between this An(III)-N bond distance and ionic radius, using this difference as a measure of bond strength or covalence. This is summarized in Table 2. We observe that, although the trivalent actinide ionic radius decreases across the 5f series, the interatomic distance in the n-C₃H₇-BTP complex remains nearly constant. The difference between bond distance and ionic radius (R(N)-radius) is smallest for U(III)-BTP meaning that the bond strength or covalence of the metal cation-N bond is greatest for this cation. The largest value is observed for the actinide in the series with the smallest radius, Cm(III). The difference between the An(III)–N bond lengths and the An(III) ionic radius increases in the series, U(III)–N < Pu(III)–N < Am(III)–N ≈ Cm(III)–N. This indicates an increasing complex stability of the An(III) with decreasing atomic number, due to an increasing softness of the An(III) cation. However, the An(III) extractability rises in the opposite order: Np(III) < Pu(III) < Am(III) < Cm(III).

Table 1. Metric parameters from R-space fits to Pu L3 edge data shown in Fig 3. Estimated standard deviations are listed in parentheses and do not include systematic errors. Results for U(III)-, Am(III)- and Cm(III)-BTP complexes are shown for comparison.

<table>
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<tr>
<th>Sample</th>
<th>Shell:</th>
<th>Parameter</th>
<th>N</th>
<th>C/N</th>
<th>C'/N'</th>
<th>C</th>
<th>r-factor^a</th>
</tr>
</thead>
<tbody>
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<td>U(III)/n-C₃H₇-BTP [4]</td>
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<td>18</td>
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<td>Pu(III)/n-C₃H₇-BTP</td>
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^a Coordination number, held constant at given values. ^b Interatomic distance. ^c Debye-Waller factor. ^d Relative shifts in ionization potential. ^e Parameter describing goodness of fit = weighted sum of squares of residuals divided by the degree of freedom. ^f Fixed at given value.
Table 2. Comparison of metal cation-N distances, An(III) ionic radii, and their difference as measure of the ionic character of the ligand-An(III) bond for U through Cm. See text for details.

<table>
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<tr>
<th>Element</th>
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*from [16]

Acknowledgment
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5. References

VACANCY FORMATION ENERGIES IN (U,Zr)N

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ABSTRACT
The application of nitride fuels in fast spectrum Gen-IV systems would allow for better breeding performance and/or increased safety margins. Classical (U,Pu)N fuels however have safety issues related to dissociation of the compound into metal and gas at high temperature. Experimental investigations on (U,Zr)N and (Pu,Zr)N have however shown that the high temperature stability can be improved by the addition of zirconium. In order to better understand how the addition of zirconium affects the behaviour of nitride fuels under irradiation, we have calculated vacancy formation energies in (U,Zr)N using the DFT code VASP. In this contribution, we review the calculated vacancy formation energies and discuss their impact on diffusion processes.

1. Introduction
By separation and transmutation of plutonium, americium and curium, one may reduce the radiotoxic inventory of high-level nuclear waste by over a factor of 100 [Delpech 1999]. While in principle any fission reactor may be used to perform multi-recycling of transuranium elements, there are certain limitations related to the type of neutron spectrum and mode of operation. If curium would be multi-recycled in an LWR, the concentration of the neutron emitting nuclide $^{252}$Cf increases by 3-4 orders of magnitude in the spent fuel, as compared to present day UOX or MOX fuel. This would complicate reprocessing and fuel fabrication to the extent that it becomes economically prohibitive. Therefore, curium should be recycled in a fast neutron spectrum [Salvatores 2005]. As methods for separation of americium from curium on industrial scale presently are not available, americium and curium may have to be recycled together in fast neutron reactors. The introduction of americium into the fuel of fast reactors however is detrimental for major safety parameters such as Doppler feedback, coolant temperature coefficient and effective delayed neutron fraction [Fazio 2007]. Further, the presence of americium reduces thermal conductivity. The linear rating of fuels with inherently low thermal conductivity, such as MOX fuels, therefore must be reduced in order to permit critical fast reactors to function safely as burners of americium [Zhang 2010]. Nitride fuels on the other hand, will typically operate with a large margin to melt, thanks to their superior thermal conductivity [Nishi 2006, Minato 2009]. Therefore, the selection of nitride fuels might permit operation of Gen-IV reactors with a smaller power penalty.

A potential issue of nitride fuels is however the tendency for actinide nitrides to dissociate into liquid metal and gas at high temperature [Potter 1977, Ogawa 1995, Thetford 2003]. In particular, AmN would de-stabilise already at 1600 K. Hence, traditional sintering of americium bearing nitrides should be performed under nitrogen atmosphere, where the maximum permissible temperature is raised to 1800 K [Jolkkonen 2004].

In order to improve the high temperature stability of Gen-IV nitride fuels, the addition of zirconium nitride in solid solution with the actinide nitride may be considered. High temperature tests performed on (U,Zr)N have shown a considerably improved stability with respect to UN [Potter 1977], allowing to measure the melting temperature of the mixed nitride
Further, tests on (Pu,Zr)N performed during 20 minutes at 2470 in nitrogen and 2570 K in argon exhibited moderate weight losses of 2-3 percent [Skupov 2006].

An additional advantage pertaining to the introduction of ZrN is its very high thermal conductivity [Shkabura 2006], leading to further reduction in fuel operating temperatures. Diffusion processes will be correspondingly affected and there might hence be a better potential for avoiding fission gas bubble formation and subsequent release. In order to investigate the impact of ZrN on diffusion processes, we have calculated vacancy formation energies in mixed (U,Zr)N nitride [Pukari 2010], using the density functional theory code VASP. The method used and the major results obtained is reviewed and discussed below.

2. Methods

Both UN and ZrN have the rock-salt crystal structure, with the respective lattice constants $a_0(UN) = 4.88 \text{ Å} [Hagawa 1974]$ and $a_0(ZrN) = 4.58 \text{ Å} [Christensen 1977]$ at room temperature. The rock-salt crystal structure in our DFT-calculations is described by a supercell of either 64 or 128 metal (M) and nitrogen (N) atoms. The former is constructed from $2\times2\times2$ unit cells and is used due to the benefit of being easily visualised while offering a suitable supercell size. The 128-atom supercell, selectively used in calculations, is derived from a primitive face-centered cubic cell and expanded with translation vectors $(4\times4\times4)$.

A system representing solid solution between UN and ZrN with either U or Zr majority has been constructed by substituting one or two atoms of the dominant metal species with atoms of the other metal species. Thus, the notations used are $U_{31}Zr_{1}N_{32}$ for a mixed nitride of 96.87% U majority and $U_{1}Zr_{31}N_{32}$ for an equivalent Zr majority in the case of a 64-atom system. Analogue notations are used for the 128-atom system.

Vacancies are created by removing any one M or N atom from UN and ZrN. In the case of $U_{31}Zr_{1}N_{32}$ and $U_{1}Zr_{31}N_{32}$, nitrogen vacancies can be created at 4 unique distances in relation to the position of the substitutional metal atom in a 64-atom system. These distances correspond to $0.5a_0$ (I), $0.86a_0$ (II), $1.12a_0$ (III) and $1.5a_0$ (IV) in a cubic supercell of $2a_0$ side length and are indicated with corresponding roman numerals.

Vacancy formation energies $E_{f,va}$ are calculated according to

$$E_{f,va} = E_v - \frac{n-1}{n}E_b$$

where $E_v$ is the energy of a supercell containing a vacancy, $E_b$ the energy of a bulk supercell and $n$ the number of atoms in the bulk [Matsson 2002]. The use of Equation 1 to describe vacancies in compounds assumes the chemical potentials of all species to be equal to $E_b/n$.

The presented first-principles calculations have been conducted with the DFT plane-wave basis set computer code VASP 4.6 [Kresse 1996]. All calculations employ the scalar relativistic projector augmented wave (PAW) pseudopotentials [Kresse 1999] to describe the core electrons. The data presented here were obtained within the generalised gradient approximation (GGA) [Perdew 1992]. The plane wave cut-off energy is set to 500 eV in all calculations. A Monkhorst-Pack integration [Monkhorst 1976] set-up with a $3\times3\times3$ k-point mesh in the Brillouin zone showed sufficient convergence in supercell calculations. In general, volume relaxation is excluded from the current calculations, since previous work has shown even high vacancy concentration to have a negligible influence on lattice parameters [Kotomin 2007, Ashley 2007, Christensen 1977, Walter 2008]. All calculations were performed without spin-polarization, since both UN and ZrN are non-magnetic at room temperature.
3. Results

Table 1 summarises formation energies $E_{f, va}$ for metal and nitrogen vacancies in pure UN and ZrN, calculated with a 64 atom supercell. For nitrogen vacancies, values obtained with a 128 atom supercell are also provided. It is evident that nitrogen deficiency is more favourable than that of a metal, regardless of the potentials used. The same trend was noted by other authors [Kotomin 2007]. The phenomenon is in good agreement with experimental results, which claim that mononitrides of U and Zr have rather deficiency than excess of nitrogen [Christensen 1977, Benz 1970].

<table>
<thead>
<tr>
<th>Vacancy</th>
<th>UN</th>
<th>ZrN</th>
</tr>
</thead>
<tbody>
<tr>
<td>Nitrogen</td>
<td>1.87 (1.81)</td>
<td>1.31 (1.40)</td>
</tr>
<tr>
<td>Metal</td>
<td>3.24</td>
<td>1.59</td>
</tr>
</tbody>
</table>

Table 1: Vacancy formation energies in UN and ZrN calculated with a 64 atom supercell, in units of eV. Values in brackets are obtained with a 128 atom supercell.

The values in Table 1 suggest that a N vacancy is energetically more favourable in ZrN than in UN, which could also indicate a lower surface energy of ZrN, suggesting surfaces of the mixed nitride to be enriched in Zr. This poses a question of how the vacancy formation mechanism in UN and ZrN is altered if substitutional Zr and U atoms are introduced to the materials.

Thus, Table 2 summarizes nitrogen vacancy formation energies in $U_{63}Zr_{1}N_{64}$ and $U_{1}Zr_{63}N_{64}$. The larger supercell was here found to be necessary for avoiding artefacts from mirror images [Pukari 2010].

<table>
<thead>
<tr>
<th>Position</th>
<th>$U_{63}Zr_{1}N_{64}$</th>
<th>$U_{1}Zr_{63}N_{64}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>I</td>
<td>1.77</td>
<td>1.30</td>
</tr>
<tr>
<td>II</td>
<td>1.80</td>
<td>1.30</td>
</tr>
<tr>
<td>III</td>
<td>1.80</td>
<td>1.32</td>
</tr>
<tr>
<td>IV</td>
<td>1.88</td>
<td>1.29</td>
</tr>
</tbody>
</table>

Table 2: Calculated nitrogen vacancy formation energies in $(U,Zr)N$, in units of eV.

In the case of $U_{63}Zr_{1}N_{64}$, increasing distance between the substitutional metal atom and a vacancy has no measurable effect. As expected, the vacancy formation energies are comparable with a pure UN. Vacancy formation energies in $U_{1}Zr_{63}N_{64}$ on the other hand, bring up an unanticipated effect. Comparing to the nitrogen vacancy formation energy in pure ZrN calculated with the same supercell size, the values obtained in $U_{1}Zr_{63}N_{64}$ are all about 0.1 eV lower. This phenomenon can be interpreted in terms of elastic stresses caused by the introducing an oversized U atom in the ZrN lattice, leading to larger displacement of neighbouring atoms during relaxation. The introduction of a single U atom further causes significant distortions in the charge density on nitrogen atoms in the entire supercell, which is not the case for the introduction of a single Zr atom in UN [Pukari 2010].

4. Conclusions and discussion

The DFT calculations here reviewed show that the nitrogen vacancy formation energy in ZrN is lower than that in UN, in agreement with the experimental fact that ZrN tends to be produced with a larger hypo-stoichiometry. Surprisingly, the lowest vacancy formation energies were found in configurations with a single uranium atom substituted into ZrN. If confirmed, this suggests that mixed $(U,Zr)N$ nitrides with a few percent of uranium could be formed in even more hypostoichiometric states than pure ZrN. Further, our data indicate that vacancy assisted intragranular diffusion rates may be higher in the said $(U,Zr)N$ than in pure
ZrN. Whether the latter are significant in comparison with grain boundary diffusion rates, remains to be investigated.

5. Acknowledgements

Financial support from Vetenskapsrådet (contract 90399101, GENIUS) is acknowledged. The authors would like to thank SNIC (Swedish National Infrastructure for Computing) for allocation of computing resources and Pavel Korzhavyi for insights into the art of DFT calculations.

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02.06.2010

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A

Education, training and knowledge management
ABSTRACT

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 and lessons learnt 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 a renaissance of nuclear power production is ongoing, due to security of supply and climate change issues (green house mitigation measures), which are receiving more importance lately.

In order to avoid a possible loss of capability and knowledge in the EU action is taken already at European Council level with a Council Conclusion on 1-2 December 2008 [1] and a Council Directive on 23 June 2009 [2] recommending actions to be taken regarding preservation of nuclear knowledge, education and training. This is confirmed through various
other statements released by national and international organisations, such as: 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 [3,4]; the Working Party on Nuclear Safety (WPNS), which recommends the improvement of exchange of nuclear safety information [5]; the European Atomic Energy Community, which recommends that the expected fading out of nuclear knowledge qualifies for a potential European solution [6]; 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 [7]; 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 [8].

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 [9].

The Institute for Energy has started with a pilot project on the consolidation of the enormous amount of scientific publications in one of its priority areas, which is described hereunder.

2. Consolidation Circle
The Institute for Energy has developed a methodology for consolidation of nuclear knowledge (fig.1).

Fig 1. Nuclear Knowledge Consolidation Circle

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 [10], managed by the Institute of Energy. 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.

3. Pilot Study
There is a huge amount of information and knowledge in WWER Reactor Pressure Vessel (RPV) embrittlement available, either published or easily available, but there are also publications that are 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.

In the first round of the pilot study the reviewers received up to 20 papers each in their specific field. They had to review the content and present it for discussion and consolidation to the WWER Reactor Pressure Vessel Embrittlement experts during a dedicated 1st workshop in December 2007. Although more than 600 papers were already collected it was decided to start with a limited number as pilot study on voluntary basis. The reports and presentations from the reviewers were requested to follow a well defined structure.

For every scientific paper/report the following information should be given:

- The title of the paper/report, including the authors and the reference where published/prepared
- A summary/abstract of the main findings formulated by the expert reviewer
- A comment of the expert on the "today’s relevance" of the paper/report

For the whole review of the subfield the following information should be given:

- Missing reference papers/reports known to the reviewer
- Open issues in the area known to the reviewer

The scope of the exercise can be different depending on the time-spans available. For the short-term it is to reach consolidated conclusions during one workshop for each of the different subfields after presentation and discussion. For the medium-term it is a consolidated review in each of the different fields after having applied the consolidation circle as often as necessary, in order to cover all available papers/reports. 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.); i.e. when all necessary consolidation circles have been carried out in all subfields and a common agreement has been reached. It has to be mentioned that the last State-of-the-Art document
was produced in 1981 by Alekseenko, Amaev, Gorynin and Nikolaev [11,12], which needs upgrading.
In a brainstorming session at the beginning the predefined fields of expertise in WWER RPV Embrittlement were discussed and defined as described in fig. 2.

![Fig 2. Subdivision of WWER RPV Embrittlement Expert Fields](image)

The recommendations of the first two workshops were the following:

- 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 3rd Consolidation Workshop on WWER RPV Embrittlement should be organized in 2009 in preparation for a training workshop in 2010
- A Joint Training Workshop between the JRC and the IAEA should be organized in 2010 on WWER RPV Embrittlement
- A Special Issue in a Journal on the NKP&C Workshop should be prepared with a general paper and ten specific papers by the reviewers of the ten sub-fields for 2010
- The co-operation with the EC funded project NULIFE should be strengthened
- A Comparison of the consolidated conclusions and open issues of the three WWER RPV Consolidation Workshop should be carried out
- JRC and IAEA should inquire the possibility to continue the successful series of Specialist Meetings on Irradiation Embrittlement
- The final goal is to cover the State-of-the-Art of WWER RPV embrittlement issues in a book

4. Conclusions

It is evident that a structural shortage of nuclear experts can not be solved by initiating such a pilot-project and vice versa, that initiating such a pilot-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 (tacit) 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 analyzing the third 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). A wide dissemination to the interested public is guaranteed free of charge and to the benefit of engineers entering the nuclear field. It seems promising 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.

Therefore, in summary, the first pilot study on WWER RPV Embrittlement is being carried out with some encouraging preliminary results:

- 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 2009 the third and last consolidation workshop of the pilot study was organized, reviewing the remaining 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 consolidation workshops, 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.

5. References

[5] CS 15475/2/06 REV2
The RELAP/SCDAPSIM 1,2,3 code, designed to predict the behaviour of reactor systems during normal and accident conditions, is being developed as part of an international nuclear technology development program called SDTP (SCDAP Development and Training Program). SDTP involves more than 60 organizations in 28 countries. One of the important applications of the code is for simulator training of university faculty and students, reactor analysts, and reactor operations and technical support staff. Examples of RELAP/SCDAPSIM-based system thermal hydraulic and severe accident simulator packages include the SAFSIM simulator developed by NECSA for the SAFARI research reactor in South Africa, university-developed simulators at the University of Mexico and Shanghai Jiao Tong University in China, and commercial VISA and RELSIM packages used for analyst and reactor operations staff training. This paper will briefly describe the different packages/facilities.
1. Introduction

The RELAP/SCDAPSIM [1,2,3] 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 uses the publicly available SCDAP/RELAP5 [4,5] 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 SDTP members. Member organizations as well as ISS have developed new code models, training materials, reactor specific training simulators, and graphical analytic packages that are used to train new users.

2. Review of RELAP/SCDAPSIM-based Simulator and Training Packages/Facilities

The RELAP/SCDAPSIM based simulator packages being developed to train new analysts, and in some cases, reactor operators, include (a) the 3D orthographic interface included as part of RELAP/SCDAPSIM and (b) optional 3rd party developed advanced interactive plant analyzer graphical displays and/or facilities. The 3rd party advanced plant analyzer GUIs include VISA, developed by KAERI in Korea [6], RELSIM, developed by RMA in the United States [7], SAFSIM developed by South African Nuclear Energy Corporation (NECSA) for SAFARI research reactor training [8]. Optional 3rd party (university) simulator facilities using RELAP/SCDAPSIM in combination with advanced graphics packages and other analysis modules include those developed by the University of Mexico [9], and Shanghai Jiao Tong University [10].

2.1 3-D Orthographic Interface

The 3-D orthographic user interface included with RELAP/SCDAPSIM automatically builds graphical displays with information provided in the standard RELAP/SCDAPSIM input model. For example a 3D orthographic display screen is built by RELAP/SCDAPSIM by reading the thermal hydraulic nodalization input. This display allows the user to quickly qualify the input model and determine if the physical layout of the hydrodynamic system (flow paths) is consistent with the system being modeled. Figure 1 shows a 3-D orthographic display for a typical two loop PWR. The white and green boxes correspond to actual reactor coordinates of individual fluid volumes and their connecting junctions. Red boxes warn of errors in closed loops in the horizontal direction. Loop closure errors in the vertical direction are fatal errors and result in the termination of the run after input processing is completed. This 3-D image can be manipulated by rotation, scaling, etc. during a run. The display is not very artistically pleasing but is extremely useful when developing an input deck.

Figure 1 - Integrated orthographic 3-D display of a typical two loop PWR.

2.2 VISA-RELAP/SCDAPSIM

VISA is a visual interactive nuclear plant analyzer and graphic package developed at the Korean Atomic Energy Institute (KAERI) [6]. The package can interactively control a RELAP/SCDAPSIM run and display the calculated results using different graphics displays built by the user. The integrated VISA-
RELAP/SCDAPSIM package uses standard RELAP5 input models. Input models cannot include the SCDAP detailed fuel rod and other core component input options so any SCDAP-related input must be removed from the input models.

Use of the VISA package is flexible in that bitmap images can be used to build initial displays and then enhanced or animated to display changing conditions during a RELAP calculation. This allows the user to scan plant drawings, photographs, and report figures or use widely available imaging programs to develop the graphics display.

Figures 2 and 3 show two typical display screens used in VISA for a three loop PWR. Figure 2 shows an animated image of a PWR hydrodynamic nodalization screen. The nodalization diagram is animated by mapping each RELAP/SCDAPSIM hydrodynamic volume to the corresponding screen element (closed boxes). Once the mapping has been completed, changes in temperature, void fractions, and other selected RELAP5 computed values can be viewed by the user during the run by watching color changes in the volume. A bar at the side of the drawing shows the corresponding color representing the value of a computed parameter.

2.3 RELSIM-RELAP/SCDAPSIM

RELSIM is an interactive simulator graphical user interface developed by Risk Management Associates in the USA [7]. RELSIM consists of three components. Two of the components are used in the integrated package. One option allows the user to run RELAP/SCDAPSIM interactively using advanced GUIs. The other option allows the user to play back previous RELAP/SCDAPSIM simulations using those same advanced GUIs. The third component of RELSIM is used by the user to build the graphic displays or screens used to either run the code interactively or play back previous simulations.

The integrated package uses standard RELAP/SCDAPSIM input models including the detailed SCDAP core component options. As a result, a typical user can very quickly build advanced graphics screens and animate the screens to control a simulation or display the results of the calculations as they are performed. The package has full access to the RELAP5 and SCDAP data base so any calculated results in the data base can be displayed, including “control variables” or parameters created using the RELAP5 control systems input. The user also has access to RELAP5 interactive commands that allow the user to control the simulation. For example, the user can turn on pumps or open valves to simulate different types of piping breaks.

A wide variety of different types of graphics screens can be built using RELSIM to display the calculated RELAP/SCDAPSIM results and control the simulation. Figure 4 shows an example of a screen built to display the
results of a 3 loop PWR simulation. In this example, the screen includes tabular elements to display numerical values, i.e. pressures, temperatures along with color displays to represent water levels in the different components of the reactor cooling system. Figure 5 shows a screen with an animated thermal hydraulic nodalization diagram. In this case, the colors within each of the thermal hydraulic volumes represent fluid temperatures in that volume. Figure 6 shows a screen developed for a TRIGA research reactor input model and analysis. Figure 7 shows the TRIGA model running with package on a PC with four monitors and displays.

2.4. SAFSIM-PC Simulator SAFARI Research Reactor

SAFSIM-PC, developed by the South African Nuclear Energy Corporation (NECSA), is a simulator training package used as a training tool for operators of the SAFARI research reactor [8]. This simulator packages uses RELAP/SCDAPSIM as its simulation engine. Figure 8 shows the simulator running on a typical PC with dual monitors. Figure 9 shows a more detailed picture of portions of the control panel screen.
2.5 University of Mexico Training Simulator

The RELAP/SCDAPSIM code is being used as the simulator engine for the RELAP/SCDAPSIM Based Plant Simulation and Training Facility located in Jiutepec, Mexico, at the Campus Morelos of the National Autonomous University of Mexico [9]. The facility utilizes a networked real time plant simulator system with the RELAP/SCDAPSIM system thermal hydraulic code in combination with advanced graphical software and displays to perform real time plant simulation and analysis for the training of analysts, technical support staff, and reactor operators. The system uses the standard RELAP/SCDAPSIM system thermal hydraulic and severe accident models with plant specific RELAP/SCDAPSIM input models and graphic displays to simulate LWR and research reactor designs. This facility is used to training operators for the Laguna Verde Power Plant. Figure 10 shows the training facility.

The RELAP/SCDAPSIM based system for Laguna Verde Power Plant uses a flexible, yet detailed Laguna Verde input model and along with combinations of generic and Laguna Verde specific graphical displays. Data is presented through an advanced human-machine interface to enable the user to have a natural and quick assessment of the Reactor’s operating status. The simulation and training facility will also be used as the main development and testing platform for other applications such as operator performance evaluation, human factors engineering for instrument and control panel design and optimization, system specification, system development and system verification, the study of task environmental constraints, and the study of virtual reality environments. University staff and students have access to advanced reactor analysis software to support academic and research activities.

2.6 Shanghai Jiao Tong University Simulator

The Nuclear Reactor system simulator was developed by Shanghai Jiao Tong University for use in analyzing operational transients, plant safety, and operator response during severe accidents [10]. It is also used to train reactor operators by simulating the Nuclear Power Plant’s control panel and the response of the NPP to various signals in the main control room. The simulator can also be used to develop accident management guidelines and to improve control room design. RELAP/SCDAPSIM provides the primary simulator engine with additional software to provide additional capability to model complex control systems. Figure 11 shows the facility.
Figure 11 - Training facility at Shanghai Jiao Tong University

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EUROPEAN APPROACH FOR A PERENNIAL STORAGE OF SEVERE ACCIDENT RESEARCH EXPERIMENTAL DATA AS RESULTING FROM EU PROJECTS LIKE SARNET*, PHÉBUS FP AND ISTP*

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*) detached to IRSN/DPAM Cadarache

1. Introduction

During the last decades, the European Commission has sponsored a large number of experimental research programmes on Severe Accidents in nuclear light water reactors. Typical examples of these programmes are the Phébus Fission Product and its follow-up ISTP programmes. Since more than four years an EC network of excellence called SARNET has integrated national/international research efforts in the same field of activity with a.o. one final goal of the controlled dissemination and perennial storage of SA experimental data.

2. The STRESA Platform

The STRESA informatics tool has been developed at the Joint Research Centre JRC/Ispra of the EC in 2002. STRESA (Storage of Thermal-Reactor Safety Analysis Data) is a general-purpose web-based informatics platform used to store documents and data coming from any type of experimental facility as well as from code calculations.

The database has the following main features:

- The database is accessible via Internet and is presently written in Microsoft Access form.
- The access to the data is controlled, the protection of confidential data is an important feature that is taken into account the information security structure of the database.
- Access authorization to specific documents is performed locally by the data owners, access rights are handled through the rules of well established agreements linked to particular programme rules.
- Networking by connection of various STRESA structures with one portal and many nodes connected to the main portal.
- Possibility for on-line plotting and visualization of data to user’s preferences.

A number of nodes have been established, both for the up-loading of experimental data and for the use of the data for the analysis and benchmarking of the data, but also for information/educational purposes.

The STRESA architecture is indicated in Figure 1: each user can connect via Internet to a server by the so-called three-tier arrangement, in which the access to the data is done through the server, which is detached from the real data. The server also serves the important purpose of giving access via password and identification to the data, according a given access level defined by the data owner. The data files in the STRESA Database are stored in their original format on the computer mass storage devices: ©Microsoft Word files, ©Adobe .PDF-files, AVI or MPEG video files etc. The only exception is for the digital data files, which are stored with a method developed at JRC called WinGraf mode. If the data are stored with WinGraf mode it is possible to benefit of the online plotting facility. The possibility of downloading the WinGraf data in plain comma separated values (CSV) format for plotting through standard tools such as Microsoft Access is also available.

The management team of the database software will in the future have to assure that all node users are working with the same common software package.

**Data access management**: A Microsoft Access database is used to keep information about the physical position of the data files, the data file types, data attributes etc. The data can be accessed in hierarchical mode. e.g.:  

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where one document may be assigned to more than one test, even related to different facilities. The user interface is obtained by a series of user-friendly accessible web pages, which allow the retrieval of the information, the plotting of the experimental data point or visualization of video films and images. In addition the list of users, the authorization information, list of events and other metadata are stored in Access database files.

As a new user is registered in the database, he receives, via e-mail, a Single Sign On (SSO) password to avoid double login. This password gives access to the Database and also views the lists of existing documents. A user is automatically given a zero-level priority access. Higher priority access can be given upon request via e-mail.

**Database security:**

The responsible of data release, indicated on each web page together with the list of documents, can give authorization to the user allowing him to have access to: (a) one particular document, (b) all the documents of a particular test or (c) all the documents of a particular test facility. He assigns to the user a corresponding level of priority.

STRESA uses a loose security system, with clear-text passwords stored directly in the database and no easy way to integrate with existing corporate authentication systems. An upgrade recently patched the password storage system, allowing for the use of encrypted passwords, and is currently being tested.

The system relies mostly on the host network infrastructure for security.

On several occasions, STRESA has been found vulnerable to attacks targeted directly to the host operating system and web server. The system has not been thoroughly tested against typical SQL exploitation methods (such as SQL injection).

It was also recommended to use more alternative and more secure web servers such as Apache with the ChiliASP extension, instead of the default IIS web server which is used currently. In the same way, by introducing new security features, STRESA could become increasingly difficult to port to newer platforms due to incompatibilities originating from the age of the technologies upon which STRESA is built.

This, in the long term, poses the biggest threats to the continued use of STRESA in its present form.

**IRSN security approach:**

IRSN choose a 2 level approach for security aspects. A first access level is done by a set security software/server which is able to manage users and passwords to give an access to a so-called DMZ zone. The 2\textsuperscript{nd} level manages authorization access between users into the database (documents, facilities ...).

The advantage of this choice is that the first access is managed by the IRSN network team, and all the IRSN security policy can be directly applied. Once this first step completed, the user can access to the STRESA web site. To avoid a double login, IRSN uses “Sign & Go” software to permit a single sign on. This software manages groups of users and passwords on several applications, and the access to the DMZ zone.

This two-level approach has the advantage to distinguish the security aspect and the database management. Indeed, the 2 aspects concern different teams, different development rules, and different works.

High security is required for the first level, and so this topic is the responsibility of network team. The second level can be managed by the database because, at this step, all users have been already identified at the first level.

**Improvements to the database:**

The present version of STRESA was written in 2002 and built upon a software stack that was current at the time. It uses ASP/VBScript pages, MS Access as data storage and indexing back-end and Visual Basic 6 for development.
While the choice of these technologies made perfect sense in 2002, it is now cause for some security and maintainability concerns. Also, there has been a scope migration from the initial STRESA design to the use it serves today.

The overwhelming (and to some extent unexpected) success and widespread adoption of the application exposes some shortcomings in the initial design choices, especially concerning the use of MS Access for data storage and indexing. MS Access is a database engine developed with personal/single use in mind, and is not suitable for a large distributed enterprise project such as the one STRESA has become.

STRESA also shows its age in other ways: the web interface is based on frames, there is no web 2.0 functionality, and the UI usability is not up to modern standards.

Furthermore, STRESA is becoming increasingly difficult to maintain. The system was designed to run on Windows 2000 Server, and presents some incompatibilities with more modern server systems such as Windows 2003/2008.

In the course of the SARNET2 project (2009-2013) it is planned to transform and upgrade this platform into a more stable and secure programme, maintaining as a minimum the same features as before, but with a marked note for long term preservation and management of the knowledge.

The new system, while retaining the general principles of the present version, will encompass new functionality and use state of the art technologies. The main goals for this new version are:

- **maintainability:** the system will be re-written from scratch and be based on vendor-neutral tools that enjoy a widespread community of users, as well as professional support. This is to avoid the lock-in situations that arise when depending on a single vendor to supply support and security fixes.

- **open-source:** being STRESA a publicly-funded project with the final goal of preserving scientific knowledge and encouraging scientific research, the use of open source software constitutes a two-fold opportunity: on one side we have the STRESA application per se, a system that will help the scientific community in preserving, sharing and consolidating knowledge. On the other hand, using an open-source model encourages the scientific community to build upon the STRESA infrastructure and create knowledge preservation and dissemination systems in other scientific fields.

- **security:** the choice will fall on mature and proven software components, where security bugs are patched rapidly and effectively.

The development of the second version of STRESA is currently at its initial phase. Numerous technologies are being investigated and evaluated, and the choices so far have been narrowed down to the use of a Linux/Apache/MySQL/PHP (LAMP) server architecture stack to host the application, which will be built upon an open source Content Management System (CMS).

**Linux/Apache/MySQL/PHP (LAMP) server architecture:** LAMP is the current de-facto standard for web applications, and enjoys an enormous community and professional support for bug fixes and security. The ubiquitousness of LAMP installations makes it very easy to find on the market professionals with the right skills to provide support and further development. It is released under an open source license.

**Content Management Systems (CMS):** A CMS can be seen as a “meta-website” infrastructure. It provides a robust code-base that can be extended and personalized through the use of modules and plug-ins, to serve any specific purpose. CMS’s are used as base of many popular websites [4] such as Mazda, MTV Networks, Citibank, Sony, The White House, Warner Bros., 20th Century Fox, and many others. They are usually open-source, written in PHP and based upon MySQL for data storage and indexing. So far the best choices seem to be Drupal, Joomla and Knowledge Tree, all three of which are open source or have a community (open source) version.

**Training:**

The services of the STRESA platform within SARNET2 comprise the creation of new local nodes combined to training sessions for newcomers, maintenance, users’ support and upgrades of the data tool and management of access rights. The operation of a local node implies mainly the up-loading of
proper experimental data. This work requires particular training, following an enquiry among SARNET partners.

The training sessions can be individual or in small groups, and usually last three or four days. The training is divided in two parts: the first part explains how to install and set-up a local STRESA node, and should be attended by the person that will manage the installation. This should be a person with an IT administrator background. The second part explains STRESA from a user perspective, and should be attended by the person who will act as Node administrator.

3. Description of the EC SARNET Network of Excellence

The European Commission (EC) in the 6th Framework Programme (FP), 49 European organizations** involved in research on nuclear safety, have decided to join their efforts in a durable way to resolve outstanding severe accident safety issues for enhancing the safety of existing and future NPPs. Integrating their respective and complementary know-how and spreading out the gained knowledge was the main objective of the Severe Accident Research Network of Excellence SARNET

Within the EC 7th FP, it was decided to provide for a prolongation of the severe accident research effort in order to tackle the remaining research work. During this SARNET2 project, again coordinated by IRSN, the long-term duration of the network will be ensured by the creation of a legal entity which will allow the partners to work together in a legally organized structure and guarantee self-sustainability of the network. Several overseas partners have joined this European project: AECL, USNRC and KAERI.

While the main objective of SARNET2 research (Fig 2) is aiming at the performance of significant progress towards the closure of the remaining high priority issues in the domain of severe accident, knowledge management is one key activity of the SARNET management team, not only through knowledge integration into the ASTEC severe accident computer code system (jointly developed by IRSN, France, and GRS, Germany) but also through electronic communication tools (ACT) and scientific databases with long-term maintenance capacities. Both instruments, together with a public SARNET website make part of one working party called generally DATANET for the collection of the available experimental data in a common format in order to ensure preservation, exchange and dissemination of severe accident experimental data, including all related documentation on experimental procedures.

Within DATANET the STRESA data platform as described in §2 has been chosen. The data are both previous experimental data that SARNET partners were willing to share within the network and all new data produced within the current SARNET-2 project lasting until 2013.

4. Data resulting from Phébus Fission Product and ISTP experimental work

An important part of experimental data flowing into the SARNET network come from both the international Phébus Fission Product and its follow-up ISTP programmes as performed by IRSN Cadarache (France) and managed in conjunction with main partners like EC/DG RTD and DG JRC, USNRC, AECL, Japanese** and Korean research institutions and Switzerland. Yet the Phébus FP and ISTP**

The international Phébus Fission Product and its follow-up ISTP programmes as performed by IRSN Cadarache (France) and managed in conjunction with main partners like EC/DG RTD and DG JRC, USNRC, AECL, Japanese** and Korean research institutions and Switzerland.

The Phébus FP experimental part (comprising five integral in pile fission product release tests from highly irradiated fuel in SA conditions) of the programme is now terminated, yet the data retrieval and interpretation has still to be finalised. During the first years after consolidated experimental data reports have been issued, most of this data is propriety of the programme partners, but within a small number of years the knowledge will become open to the international nuclear safety community. The

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1 Not a Partner of ISTP
2 International Source Term Programme
3 Not a Partner of ISTP
ISTP [2] data base also contains a large amount of experimental data on iodine chemistry in reactor containment and circuit; ruthenium studies, cladding oxidation studies and B₄C rod degradation tests.

It was therefore thought that the STRESA data base, and even more its upgraded version resulting from SARNET2, should become the ideal platform for the access and preservation i.e. management of this unique severe accident knowledge. This database could ideally be used later by experts of all kinds, ranging from technical support organisations, research institutions, universities and others.

At a later stage it will be most useful to establish a link to the Data Bank of the OECD/NEA, a reference tool for the preservation and dissemination of consolidated international experimental data in the field of nuclear safety.

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European approach for a perennial storage of Severe Accident Research

Fig. 2. SARNET2: Joint Programme Activities
EDUCATION AND TRAINING OF FUTURE NUCLEAR ENGINEERS THROUGH THE USE OF AN INTERACTIVE PLANT SIMULATOR

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1. Introduction

The International Atomic Energy Agency (IAEA) sponsors the development of nuclear reactor simulators for education, or arranges the supply of such simulation programs [1]. Aware of this, the Department of Nuclear Engineering of the Universidad Politécnica de Madrid was provided in 2008 with the Interactive Graphical Simulator of the Spanish nuclear power plant José Cabrera, whose operation ceased definitively in 2006. According with the IAEA-TEC-DOC-1411 [2], the simulator is a Graphical Simulator, used for training of main control room personnel, technical support engineers, and operations management. This paper presents all the work performed at the Department to turn the simulator into a teaching/learning tool, to be use in the nuclear engineering studies following guidance found in [3].

2. Description of the Simulator

The experience gained in the last years by our Department in the simulation of the Nuclear Power Plants, mainly in PWR, has been included in the nuclear Nuclear Technology programme with optimization of manoeuvres, start-up, etc. In addition, the Nuclear Reactor Design programme has been focused on the understanding of the computational codes for nuclear reactor designs, starting with the nuclear data processing codes, then the core calculations codes, and finally the plant simulators codes (JANIS, NJOY, WIMSD, ORIGEN/ACAB, MCNP, COBAYA/SIMULA, COBRA, SIMTRAN, RELAP). But more realistic studies are also required to complete this general objectives, and in this sense the Simulator is the appropriate tool to be used.

The Fig.1 shows a diagram of the simulator architecture as installed in the José Cabrera NPP for years [4]. The Interactive Graphic Simulator SGI is the one that has been installed in the Nuclear Engineering Department, and it is an analytical graphical simulator that is especially useful for didactic purposes. It is an interactive tool, that allows the student to complete the teaching-learning methodology in the nuclear science and technology with more practical methodologies, as is recommended in the new engineering studies adapted to the Bologna rules.

The components and systems of the whole power plant are replicated in the Simulator, this includes the nuclear reactor, the pressurized vessel, the primary and secondary loops, the turbine, the condensator, the fluids systems, the instrumentation and control components, and the electrical systems, as well as the emergency systems that are automatic started when needed. Also the simulator has an alarm panel that provides information similar to the one provided in the power plant, showing the variables and parameters that are out of range, and shows if the operator has to take any action, or at less helps to identify the variables. The alarm panel is divided in three panels: primary circuit, secondary circuit, and post-accident conditions. The software is supported in a HP-735 workstation, and three SUN
SPARC 4 Work Station. The screens allow to click in a component and get the information in a graphical way. Fig.2 shows the SGI work stations and alarm panels as they are installed in the Department.

Fig 1. José Cabrera Nuclear power plant General Simulator Arrangement

The components are represented in different screens with diagrams where the different colours indicates the status of the component (open/close, on/off,…). The functions available are: Initialization with until 60 different initial conditions, run/stop the simulation, malfunctions with different severity and duration, function SPDS with a continuous showing of the safety parameters, and the graphic representation.

The nuclear power plant of origin was a PWR reactor with only one primary circuit. This makes the installation simpler in order to be used for teaching purposes, that other nuclear plants with 3 or 4 primary circuits.

The simulator provides the real plant responses and the physical behaviour during the normal operation, and simulates several maneuvers, a series of malfunctions, and operational transients, and also allows the training in the emergency procedures under accident situations. With the simulation of these situations the student is trained in the plant behaviour, and in the nuclear and thermohydraulic phenomenology in the nuclear reactor and in the components of the whole plant. For that purposes includes the TRAC-PWR and RELAP5/MOD3.2 codes as the software package, that simulates the whole plant behaviour.

Very illustrative screens, as the one in Fig.3, show all the plant systems, and allow to act directly on the system components. Alarm control panels, similar to the ones existing in the control room of a nuclear power plant, are also available to alert users of potential equipment problems or unusual conditions.
Fig 2. Aula José Cabrera with the Interactive Graphical Simulator (SGI)

Fig 3. One Screen with reactor vessel, injection system, and auxiliary feedwater system
3. Methodology

The Nuclear Engineering Department dedicates two technicians to be responsible of the installation, one for the hardware and equipment, and other for the use of the simulator, and the students assistance. The Department has also the support of the Gas Natural - Union Fenosa company through the assistance of the technician who was the power plant operator trainer in the José Cabrera power plant, in order to solve the doubts and problems that may appear during the use of the installation. Also the Tecnatom company that developed the whole installation assist in order to solve punctual hardware and software problems.

A Commission integrated by members of the Nuclear Engineering Department (2 persons) and the Gas Natural - Union Fenosa company (2 persons), has been created to follow the work performed in the Simulator, and make proposals to improve when necessary, the teaching-learning process.

Aware of the advantages that the use of a simulator as SGI can provide for an active and independent training of our students, different material is under preparation for the development of practical classes. The aim is to provide students with the tools necessary to be able to acquire, following an active methodology, scientific knowledge and technology related to the design, safety and economical operation of a nuclear power plant. The intention is to encourage the student giving him a greater role in their learning, by providing a virtual environment that allows to operate the plant as if an operator is involved.

In the preparation of this material contribute teachers and technical staff of the Department as well as students who are in different stages of their studies.

Three types of students can be described regarding the use of the simulator:

- Undergraduate students that use the Simulator for the practices period of the topics that are part of the Grade level curricula: Nuclear Power Plants, Nuclear Technology, and Nuclear Safety.
- Master degree students that work for a period of 6 months in the Simulator, normally supported by a fellowship of the Consejo de Seguridad Nuclear (CSN, the Spanish nuclear regulatory commission), and develop the Master Final Project in the Simulator. This project provides 15 ECTS (European Credit Transfer System) for the "Nuclear Science and Technology Master". Also the Industrial Engineering Master Final Project may be carried out in the Simulator.
- Visiting and collaborating students that spend part of their time learning the use of the simulator and afterwards helping in the development of the material needed for its productive used from the Teaching-Learning objective point of view. The first ones are coming from foreign universities, the second ones are students from the Naval and Industrial Engineering Schools.

The graduate students that use the Simulator should start with the identification and understanding of the existing documentation, and then they may contribute providing more detailed documentation, description of the screens and components, or simulation of different situations. They generate the related documentation with the analysis of the results that have been obtained. Also they may prepare standard and simple practices to be run by the undergraduate students during the teaching-learning period at the grade level studies. Each student has a tutor or director of the project, which analyze the developed material, in order to help, guide and evaluate the student during the learning period.

The students have access to the manuals that the power plant operators have used in the continuous training they have followed [5,6], and the Nuclear Regulatory Commission in
Spain demand. The documentation includes the description of the power plant systems, the emergency operation procedures, as well as the description of the Simulator, the initial conditions available, and the malfunctions that may be simulated.

The material that is being prepared for each class or group of classes that constitute a practice for the undergraduate students is divided into three parts according to their purpose in the development of work by the student. These are:

- **Practice Manual**: objectives and theoretical basis, systems involved in the maneuver and main variables to be monitored, guide implementation of the maneuver, with detailed actions that students must carry out.
- **Monitoring material**: that the student must complete during the practice: tables for data collection, graphical representation of the temporal evolution of the significant variables, graphical analysis.
- **Material self-assessment**: that the student must complete following the practice: issues related to the development of practical and theoretical foundations.

4. Main results and experience

The standard operational situations that have been prepared for the moment and run by the students are:

- Normal operation in nominal power.
- Nuclear power variations and turbine demand follow.
- Identification of the operational states in the plant: Cold-Zero-Power, Hot-Zero-Power, Hot-Full-Power, Nominal operation.
- Plant start-up, from Cold-Zero-Power to Full-Power.
- Plant down, from Full-Power to Cold-Zero-Power, and evolution during the Zero-Power period.

The simulator also allows the simulation in hypothetical accidents, those which are complex and with a very low probability to happen. This is used in the training, in order to understand the optimal way to drive the plant to an stable and safe situation. For the simulation of the accidents, the best-estimate and realistic codes are used. Codes that have been validated previously. The evolution is done in real time, reason why the student take conscience of the time and the risk of these potential situations, and the high reliability needed in order to limit the global risk.

These accidental and complex situations provide the student the detailed understanding of the head transmission and fluids mechanics, the kinetic reactor behaviour and the coupling among them. These situations are for the moment under testing. They should be carried out by the students when the simpler transients and maneuvers are completely understood.

The accident situations are very extend, and as a sample may be simulated the following:

- Loss of electric feed, with failure of external electrical feed and Diesel Generator.
- Steam generator tube break, with or without the safety injection system.
- Reactor scram signal with failure in the control rod insertion, and success boration.
- Main pump rotor stop, with pressurizer valves opening.
- Small LOCA with safety injection, 0,5” primary circuit break.
- Essential services water loss, and auxiliary feed-water system.
- Components cooling system loss, and auxiliary feed-water system.
- Main steam line break in the auxiliary building, with safety injection system failure.
Until now several projects have been performed by the postgraduate students, under four Collaboration fellowships, and three Master Final Projects, covering the following topics:

- Preparation of the SGI Documentation and User’s Manual (systems descriptions, transient and operational modes, systems identification, screens and alarm panel description)
- Preparation of the SGI Malfuctions Manual (in particular for the Loss of coolant accident)
- Transient analysis due to primary circuit changes (Simulation of Loss of coolant accident in cold leg, user’s guide preparation, analysis of the Emergency Operation Procedures)
- Transient analysis due to malfuctions in the valves (pressurizer shower valve, pressurizer relief valve)
- Optimized Plant Start-up and Initial conditions.
- Optimized Plant down and drive to the cold conditions. Identification of the Xenon peak during the stop period.
- Loss of coolant accident simulation with a guillotine break in the cold leg.

And the practices programmed for the undergraduate students until now have been the following:

- Nuclear Power Plants, with 50 students: Nominal operation simulation, and thermal power variation simulation
- Nuclear Safety with 40 students: Loss of coolant accident simulation

For these practices the students have the Practice Guide Manual, and as a sample the following documentation is available for the Nuclear Safety practice [7]: Practice Manual (description of the practice, systems involved, and variables to follow, and realization guide), Follow-up material (Tables to feel-up, Graphic representations to prepare), and auto-evaluation material (questions to answer).

The students are trained through the simulations in the interpretation of the screens that are showed in the workstations, and the plotted variables and its temporal evolution. The adviser professor examines the results obtained by students in order to assess if the simulation has been effective.

5. Conclusions

The experience obtained so far with the use of the simulator has been very successful. The graduate students involved in the development of the projects, practices and documents related with the simulator show a great interest for the work that they are doing making that the laboratory where the simulator is installed to be busy place. Regarding the undergraduate students, the practices in the simulator encourage them to follow the Nuclear Energy studies in the Engineering Schools, what is very rewarding for the Department professors.

The simulator has proved to be an optimal tool to transfer the knowledge of the physical phenomena that are involved in the nuclear power plants, from the nuclear reactor to the whole set of systems and equipments on a nuclear power plant. It is also a relevant tool for motivation of the students, and to complete the theoretical lessons. This use of the simulator in the learning-teaching process meats also the criteria recommended for the Bologna
adapted studies, as it helps to increase the private hands-on work of the student, and allows them to experience the work inside a team, in a practical and real installation.

It should be noticed that this type of simulator is only available in selected universities and Nuclear Engineering Departments in the world, and that it helps to reach the excellence in the nuclear engineering programs studies.

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Life science applications
CONSEQUENCES OF LONG-TERM RADIOACTIVE CONTAMINATION OF AQUATIC ENVIRONMENT BY ANTHROPOGENIC RADIONUCLIDES

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ABSTRACT

The distribution of the main dose-formed radionuclides in components of aquatic ecosystems within the Chernobyl exclusion zone and effects of chronic irradiation on biota during last decade was studied. The absorbed dose rate for hydrobionts registered in range from 1.3 mGy year\(^{-1}\) to 3.4 Gy year\(^{-1}\). The heightened chromosome aberration rate in the embryo tissue of snails (up to 27 %) and in the root meristems of higher aquatic plants (up to 18 %) was determined. In hemolymph of snails from contaminated lakes the quantity of death cells averages 36-44 %, the part of phagocytic cells averages 41-45 %, as well as decrease of the young amoebocytes quantity to 10-20 %. The high level of parasitic fungi and gall-producing arthropods of the common reed in the most contaminated lakes within the exclusion zone was registered. Above mentioned phenomenon may testify upon the decreasing of the parasitical stability of plant under impact of long-term radiation exposure.

1. Introduction

Though of the years, past after the Chernobyl NPP accident in 1986, self-purification of closed water bodies in the Chernobyl exclusion zone, defined as a roughly circular area of 30-km radius around the destroyed unit, is extremely slow process. Therefore, ecosystems of the majority of lakes, dead channels and crawls possess high level of radionuclide contamination of all the components.

The basic problems of radiation safety of the Chernobyl exclusion zone concern radionuclide wash-off with surface drainage water to river network, their export outside the exclusion zone and affection of the water quality in the Dnieper River. Undoubtedly one of the most important and still insufficient studied problems of aquatic ecosystems within the exclusion zone is research of chronic impact of ionizing radiation on non-human biota.

Thus the investigation of biological effects of long-term exposure on hydrobionts, inhabiting the radioactive contaminated water bodies, has a theoretical and applied importance for ecological hazard understanding, associated with changing of environmental radioactivity state due to human activity.
Our main tasks were: (1) analysis of species specific and dynamics of radionuclide accumulation by hydrobionts of the different taxonomy and trophic level and revealing of biological indicators of radioactive contamination; (2) study of radionuclide distribution in the main components of aquatic ecosystems; (3) dose rate estimation due to external and internal sources of irradiation for different groups and species of hydrobionts; (4) evaluation of cytogenetical, hematological and parasitological effects dynamics due to long-term radioactive impact on hydrobionts in conditions of water bodies within the Chernobyl exclusion zone.

2. Materials and methods

Our researches were carried out during 1998-2009. The water bodies of research were the flood plain reservoirs of the Pripyat River - Azbuchin Lake, Yanovsky, Dalekoye-1 Lake, Glubokoye Lake as well as Uzh River and Pripyat River within the Chernobyl accident exclusion zone. The results of the analyses compared to the data received for hydrobionts from the control lakes, located in the neighbourhood of the Kiev City: Vyrlitsa, Opechen', Pidbirna, Goloseevo.

External gamma irradiation dose rate was measured by DKS-01 dosimeter and by Na-I field radiometer SRP-68-03. The estimation of the internal dose rate for snails was carried with use of dose conversion coefficients [1]. The $^{137}$Cs content was measured by $\gamma$-spectrometry complex: PGT IGC-25 detector (France), “Nokia LP 4900 B” analyser (“Nokia4”, Finland), low-volt feeding source – crate NIM BIN, amplifier NU-8210 (“Elektronicus Merokeszulekek Gyara”, Hungary) and 100 mm thickness leaden protection. The $^{89}$Sr content was measured on low-background NRR-610 $\beta$-radiometer (“Tesla”, Czech). Minimal detectable activity was 0.04 Bq under 1000 s sample exposition. $^{239}$Pu and $^{239+240}$Pu content in electrolytic samples was determined by $\alpha$-spectrometric tract by NUC-8192 impulse analyser (“Elektronicus Merokeszulekek Gyara”, Hungary). The $^{241}$Am content was measured by x-ray-spectrometric line including x-ray detector EG&G Ortec LOAX-51370/20 CFG-SU-GMX (“EG&G Ortec”, USA) and analyser “Nokia LP 4900 B”. The results were measured in Bq/kg at natural humidity and the mistake of estimated radionuclide concentration fell within 15-20%.

The chromosome aberration rate was measured in embryo cells of gastropod pond snails (Lymnaea stagnalis) by the standard anaphase method [2] and in the apical root meristems of the four species of higher aquatic plants: common reed (Phragmites australis), arrowhead (Sagittaria sagittifolia), flowering rush (Butomus umbellatus) and manna (Glyceria maxima) by the modified method [3]. Haematological studies were carried out in mantle liquid of gastropod snails by the analysis of dead cells, young amoebocytes and phagocytic cells quantity [4, 5].

3. Results

3.1 Chromosome aberration rate

The freshwater snails have great importance for the processes of radionuclide biogenic migration in aquatic ecosystems. Due to ability to accumulate practically all of radionuclides which registries in water these invertebrates can be considered on the one hand as bio-indicators of radioactive contamination of aquatic ecosystems and from another hand - as organisms which can take active part in the processes of radionuclide redistribution in hydrobiocenosa. In this part studies wes evaluated the cytogenetical effects level in embryo tissue of gastropod snail as chromosome aberration rate, considering it as reaction of snails on radiotoxicological condition of environment.

The absorbed dose rate for snails from lakes within the Chernobyl exclusion zone was registered in the range from 0.8 to 3.4 Gy year\(^{-1}\). The highest rate was found in snails from lakes Dalekoye-1 and Glubokoye located within the dammed territory on the left-bank flood lands of the Pripyat River, the lowest - from the Pripyat River and Uzh River. Molluscs from the control lakes were characterised by absorbed dose rate about 0.3 mGy year\(^{-1}\). The rate
of chromosome aberration was found in snails from lakes Dalekoye-1 and Glubokoye - 21-22 %. In the embryo tissue of snails from Yanovsky Crawl the chromosome aberration rate was about 18 % and in Azbuchin Lake - about 23 %. About 3.3 % of aberrant cells were registered in snail’s embryo from the Pripyat River and in snails of Uzh River the aberration rate was about 2.3 %. The rate of chromosome aberration for snails from the control lakes was about 1.1-2.0 % (Fig. 1).

During 1998-2008 a tendency to decrease of chromosome aberration level in mollusks from all lakes of the exclusion zone was registered. The probabilistic prediction of the chromosome aberration rate for gastropod snails in lakes of the Chernobyl exclusion zone have shown that spontaneous mutagenesis level (2.0-2.5 %) [6] can be reach in Azbuchin Lake and Yanovsky Crawl in 2020s-2030s and in Dalekoye-1 Lake and Glubokoye Lake – in 2060s-2070s.

The studies of the different species of plants within the exclusion zone have revealed a numerous morphological anomalies as repeated organs, gigantism or dwarf, underdevelopment or sterility of reproductive organs, excessive branching, growth inhibition of the secondary points of growth etc. All of this variety of plant anomalies of development is testify that the vegetation within the exclusion zone has undergone to strong damage of the genotype, which consequence is the long genetic instability and thus increased variability of many species. The spontaneous rate of aberrant cells for higher aquatic plants from the control lakes does not increase 2.0-2.5 %. We found 7.5 % aberrant cells in common reed of Yanovsky Crawl. The highest rate of chromosome aberration was registered in plants from Azbuchin, Glubokoye and Dalekoye-1 lakes - 9.0 %, 10.8 % and 17.8 % respectively. In comparison, the data received for reed from Goloseevo Lake amount to 1.0 %. The rate of chromosome aberration in reed from closed water bodies within the left-banked flood plain of the Pripyat River was in 5-8 times higher than spontaneous mutagenesis level.

As shown in Fig. 2. the single fragments were the frequently occurring aberration in meristem cells of reed - 57.1 % of all aberrations. Rate of single bridges were 40.7 %, multiple aberration, including different variation of fragments and/or bridges (pair bridges or fragments, bridge and fragment, bridge and two fragments, three fragments), were 2.3 %.

Figure 1. Chromosome aberration rate in snail embryos in water bodies within the Chernobyl exclusion zone and control lakes within the Kiev City area.

As shown in Fig. 2. the single fragments were the frequently occurring aberration in meristem cells of reed - 57.1 % of all aberrations. Rate of single bridges were 40.7 %, multiple aberration, including different variation of fragments and/or bridges (pair bridges or fragments, bridge and fragment, bridge and two fragments, three fragments), were 2.3 %.
Figure 2. Spectrum of the main chromosome aberration in cells of the common reed from water bodies within the Chernobyl exclusion zone.

It seems that spectrum of the main types of chromosome damages in plants of water bodies of the right-bank flood plain of the Pripyat River (Azbuchin Lake and Yanovsky Crawl) rather determines by the chemical mutagens (up to 69 % of single fragments). The type of chromosome damages distribution in plants from lakes within embankment territory on the left-bank flood plain of the Pripyat River (Dalekoye-1 Lake and Glubokoye Lake) points to practically equivalent effects of radiation and chemical factors - 44-49 % of bridges, 43-50 % of fragments and 6-8 % of multiple aberrations.

The positive correlation between absorbed dose and rate of aberrant cells of apical root meristems of the common reed and embryos of the gastropod pond snail was found. According to our data this dependence more corresponds to a power-mode function (Fig. 3).

![Figure 3](image.png)

Figure 3. The dependence of chromosome aberration rate from absorbed dose rate in cells of the common reed (a) and embryos of the gastropod pond snails (b) from water bodies within the Chernobyl exclusion zone.

3.2 Haematological effects

Molluscs is one of the first group of animal that have got cell immunity during the evolution. The protective function has the cells of mantle liquid (hemolymph), which peculiar to high proliferation, heterogeneity and sensitivity to radiation impact. The most functional and active
elements of hemolymph of the snails are the granular amoebocytes, which have morphological and quantitative reaction on the physiological changes of organism. The decrease of adsorbed dose rate causes the decrease of young amoebocytes quantity in hemolymph. The decrease of total agranulocyte quantity in cell population occurs due to decrease of amoebocytes quantity and due to increase of granulocyte quantity, which involves in phagocytic reaction.

![Figure 4](image-url)

**Figure 4.** Ratio of different types of amoebocytes in the mantle liquid of snails in water bodies within the Chernobyl exclusion zone and control lakes within the Kiev City area.

In hemolymph of snails from Dalekoye-1 Lake, Azbuchin Lake and Glubokoye Lake the quantity of dead cells averages 36.2 %, 39.2 % and 43.8 % respectively, the part of phagocytic cells averages 44.3 %, 41.2 % and 45.0 %, as well as decrease of the young amoebocytes quantity to 13.2 %, 20.1 % and 9.5 % respectively (Fig. 4).

The insignificant quantity of abnormal cells and micronuclei has been observed here as well. In conditionally “clean” water bodies the part of dead cells averages from 2.2 to 5.3 % and the quantity of phagocytic was at level 3.0-4.2 %. The quantity of young amoebocytes has increased here to 79.7-89.6 %.

### 3.3 Arthropode-borne and parasitical fungi diseases

In reservoirs with the increased level of radioactive contamination (lakes of the dammed territory of the left-bank flood plain of the Pripyat River) the high level of the common reed damage by gall-producing arthropods, in particular by mites *Steneotarsonemus phragmitidis* since 2000 is observed. This phenomenon for the first time was registered at the territory of Ukraine within the Chernobyl exclusion zone [7]. We suppose the scales and speed of distribution of this phenomenon in reservoirs of the exclusion zone deserve the special attention. As the common reed is almost cosmopolitan it is quite logical to predict wide moving of mites in other reservoirs, which the Polesye region (woodlands) is so rich. Thus if in 2000 and 2001 the damaged individual registered only in one of seven reservoirs of sampling in the exclusion zone - Dalekoye-1 Lake, during 2002-2009 the damaged individuals of the common reed began to meet in all other researched water objects. At that if in some reservoirs we fond a single affected individuals only, in Yanovsky Crawl and Azbuchin Lake the described phenomenon quickly has received distribution and in 2005 the share of damaged plants was accordingly 74 % and 32 % (Fig. 5).

The highest percent of affected plants observed in Dalekoye-1 Lake (practically 100 %), located on the dammed territory of Krasnensky flood-lands. This territory is characterized by the maximal density of radioactive contamination in the exclusion zone. The specific activity of radionuclides in the reed’s tissue (at natural humidity) during the researched period
reached to 10000 Bq kg\(^{-1}\) for \(^{137}\)Cs and 2000 Bq kg\(^{-1}\) for \(^{90}\)Sr [8]. At that the absorbed dose, caused by external gamma-radiation and radionuclides incorporated in meristem tissue of plants, was reached more than 4.0 Gy/year during last 10-15 years after Chernobyl accident. As have shown cytogenetical studies the rate of chromosome aberration in cells of meristem tissue of the common reed in Dalekoye-1 Lake reaches to 18 % and is maximal among the investigated reservoirs of the exclusion zone. The damage events of common reed by larvae of gall fly of family Chloropidae, genus Lipara has been registered as well.

Figure 5. The common reed in Dalekoye-1 Lake, affected by the gall-producing mite Steneotarsonemus phragmitidis. The total view the reed bed without panicles at the end vegetation (left) and “corkscrew” galls on the damaged plants (right) in October 2007.

The influence of parasitic fungi Claviceps purpurea (ergot) lesions on the seed production of the common reed was determined. The positive correlation between registered abnormalities and parasitic fungi lesions of the common reed with levels of radiation exposure on plants, sampled from lakes within the Chernobyl exclusion zone was registered. Registered high level of parasitic fungi lesions ratio for the common reed plants sampled from the lakes of the left-bank dammed floodplain of the Pripyat River, which are the most radioactive contaminated water bodies of the exclusion zone. Such significant damage of the common reed’s panicles by the parasitical fungi in lakes of the Krasnensky flood lande was observed on the background of the most low parameters of seed production in water bodies of the exclusion zone.

The increase of the parasitic fungi lesions ratio simultaneous the enhancement of absorbed dose rate of ionizing radiation, which may conclude of chronic low-dose radiation influence upon the populations of the common reed within the Chernobyl exclusion zone, was determined. At that the Pirson correlation coefficient for the parasitic fungi lesions ratio and absorbed dose rate of ionizing radiation was 0.913 and error coefficient was 0.030.

Relation of potential seed production and effective seed output is one of the significant parameter characterizing rate of reproductive performance suppression of plant. Thus for the cultural cereal explorers observed complete or partial sterility [9, 10] but for the common reed we found that panicle sterility ratio does not exceed 72 percents. In normal case 1 panicle produces about 5,000 viable seeds [11] but in our case even potential seed production was much lesser.

Above mentioned phenomenons may testify upon the decreasing of the parasitical stability of the common reed under impact of long-term irradiation

4. Conclusion
Different radiation effects of ionizing radiation on hydrobionts within the Chernobyl exclusion zone have been registered in post-accident period. Some of these effects appear shortly, while an increasing importance is expected by the remote consequences - genetic damages induced by a long-term irradiation. These remote consequences are a long-drawn realization of changes in molecules of heredity, where the initial molecular damages have a latent period without any display and can be transferred through the many generations of cells to be a reason of genome instability in future.

The absorbed dose rate for hydrobionts of litoral of the researched water bodies within the Chernobyl exclusion zone during 1998-2009 was registered in a range 1.3 mGy year\(^{-1}\) - 3.4 Gy year\(^{-1}\). The highest levels were registered in lakes of the dammed territory of the left-bank floodplain of the Pripyat River - Glubokoye Lake and Dalekoye-1 Lake, the lowest one - for running water objects. The ratio of doses, caused by an external and internal irradiation of hydrobionts in different reservoirs, essentially varies and depends on the contents of gamma-emitting radionuclides in bottom sediments of littoral zone and soils of shoreline. At their high level of radionuclide contamination up to 95 % of radiation dose can form due to external sources and only 5 % due to radionuclides, incorporated in the tissues. The rate of external dose on hydrobionts of the same water body can change in range of tree exponents and depend from radioactive contamination level of ecological niche. At present time the main dose-formed radionuclide in the majority water bodies within the exclusion zone is ⁹⁰Sr, which part of total internal radiation dose amount 90-95 %.

The long-term impact of low dose irradiation in aquatic ecosystems, especially in lakes within the inner (10-km) Chernobyl exclusion zone exhibits in increased level of chromosome aberrations and, connected with it, in reproductive death of cells. Cytogenetical and haematological research of hydrobionts in the exclusion zone testify to a high level of chromosome aberration in embryo tissues of gastropod snails and in root meristems of higher aquatic plants, and also to essential changes of hemolymph structure of molluscs in the most contaminated lakes. Chromosome aberration rate in tissue of hydrobionts from the closed water bodies of the exclusion zone repeatedly exceeds a spontaneous mutagenesis level for aquatic biota and can be display of radiation-induced genetic instability.

For the common reed in lakes of the inner exclusion zone, the high level of affection by parasitic fungi *Claviceps purpurea* and by gall-producing arthropods, in particular by mites *Steneotarsonemus phragmitidis* (sometimes up to 100 % of a vegetative population of lake), was discovered. During 2002-2009 affection by mites quickly propagated in other closed reservoirs of the exclusion zone, essentially reducing rates of growth, seed efficiency and bioweight of plants. Now there are no bases finally to approve, that the damage of a reed causes by the impact of ionizing radiation, however we are anxious about the fact, that this mite species for the first time is registered in Ukraine within the Chernobyl exclusion zone, in the territory which is the most contaminated by radionuclides. In this connection it is supposed, that one of the possible reasons of a total plants disease can be a loss of them a parasitical stability in conditions of chronic radiation influence.

5. **Acknowledgement**

This study was financially supported by the Ukraine Ministry for Emergency Situations and by the National Academy of Sciences of Ukraine (Projects 0101U004987, 0102U004665 and 0102U003541). The authors also wish to thank the personnel of Radioanalytical Laboratory of the State Specialised Scientific and Production Enterprise “Chernobyl Radioecological Centre” for the radionuclides measuring procedure.
6. References


DOSE ASSESSMENT FOR POPULATION IN CASE OF DESIGN BASIS ACCIDENT AT NPP

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ABSTRACT

Radiation protection of the population in case of a reactor accident utilizes reference levels or generic criteria which are mainly based on dose values. Therefore adequate provisions for effective and timely dose assessment for population in case of design basis accident (DBA) at NPP are important. Radiological consequences for the DBA at NPP with a PWR-1200 reactor were studied. Radionuclide concentrations downwind from the NPP and projected doses due to the accident were estimated using the software InterRAS. In the result of calculations and its analysis versus international recommendations it was found that after the DBA at NPP with a PWR reactor there is no need for evacuation or providing sheltering for the public because the total effective dose will not achieve currently recommended generic criteria. Besides, there is no need for recommendation to the public to avoid eating potentially contaminated food or milk.

1. Introduction

The use of nuclear energy in practice is growing daily. Despite of safety precautions in design and operations, reactor accidents do occur. One of the most important aspects of managing a nuclear emergency is the ability to promptly and adequately determine and take actions to protect members of the public and emergency workers. Reactor accident assessment must take account of all critical information available at any time and must be an iterative and dynamic process aimed at reviewing the response as more detailed and complete information becomes available. Accident assessment process relates reactor plant information and environmental monitoring data to the appropriate protective actions, covering the entire course of an accident. Classification of the accident is most important during emergency. All changes in plant conditions or radiological conditions must be evaluated immediately to determine if the classification should be changed. Based on accident classification it is possible to determine public protection actions and emergency worker recommendations. These actions should be taken primarily during the initial response to an accident and do not cover the recovery phase in detail, when more detailed assessments based on accident specific information can be carried out.

2. Material and methods

Design basis accident at NPP with a PWR-1200 reactor (or WWER-1200 in case of a Russian design) was studied. Design basis accident is an accident conditions against which a facility is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits [1]. It was assumed that the NPP works with maximum heat power of 3200 MW(th) (content of fuel in the reactor is 100%).
The release input data were the following: release route were dry containment leakage, unfiltered, primary system leak for 2.5 hours (vessel water level below top of active fuel and decreasing), time core is uncovered were assumed 15-30 minutes (rapid fuel cladding failure and local fuel melting), 100 % gap release. Release passes into the atmosphere through a containment volume that is sprayed (see figure 1). It means that on the way to the atmosphere that will remove a large fraction of the iodine and particulate.

Fig 1. Release from the containment in case of DBA

Leak rate were 0.004 %/hour (or 0.1 %/day), which is the maximum leak rate that could take place in case of DBA). Hold up time assumed to be zero [2]. Release through the containment assumed as a ground-level release, not isolated release location. In this case building wake effects need to be considered. If the wind speed was zero the release would be considered as a release from the definite height [3].

Real meteorological data which typical for winter and summer seasons in Belarus accordingly were used for calculations (see Table 1). All plant-specific and meteorological data have been selected for the purpose of the assessment of the worst case scenario of DBA.

Tab. 1: Meteorological data for “winter” and “summer” scenarios of DBA

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Winter</th>
<th>Summer</th>
</tr>
</thead>
<tbody>
<tr>
<td>Wind direction</td>
<td>west, south-west</td>
<td>south-west</td>
</tr>
<tr>
<td>Surface wind speed</td>
<td>5.5 - 11 m/s</td>
<td>6.4 - 6.7 m/s</td>
</tr>
<tr>
<td>Atmospheric pressure</td>
<td>1008.0 GPa</td>
<td>993.7 GPa</td>
</tr>
<tr>
<td>Air temperature</td>
<td>-2.5 - -1.5 °C at night and in the morning</td>
<td>20 °C</td>
</tr>
<tr>
<td></td>
<td>3.7-1.8 °C – in the day time</td>
<td></td>
</tr>
<tr>
<td>Cloudiness</td>
<td>0 %</td>
<td>100 %</td>
</tr>
<tr>
<td>Mixing layer height</td>
<td>1.2 - 1.5 km at night</td>
<td>0.6 km</td>
</tr>
<tr>
<td></td>
<td>0.5 - 0.3 km in the day time and at evening</td>
<td></td>
</tr>
<tr>
<td>Stability class</td>
<td>F</td>
<td>D</td>
</tr>
<tr>
<td>Precipitation rate</td>
<td>от 1 до 4 mm/h,  snow cover 1-1,5 cm height</td>
<td>no rain</td>
</tr>
</tbody>
</table>

Source term estimation in case of DBA:
\[ \text{SourceTerm}_i = FPI_i \cdot CRF_i \cdot \prod_{j=1}^{N} RDF_{(i,j)} \cdot EF_i \] 

(1)

where

- \( FPI_i \) – Isotope inventory;
- \( CRF_i \) – Amount of isotope \( i \) released out of core/core inventory of isotope \( i \);
- \( RDF_{(i,j)} \) – Fraction of the isotope \( i \) activity available for release following reduction mechanism \( j \);
- \( EF \) – Fraction of activity available for release that is released [2].

Activity of radionuclides and doses for the public in case of the DBA were calculated using The International Radiological Assessment System (InterRAS) software, which was developed for use by personnel who responds to a nuclear emergencies. The InterRAS is a set of personal computer-based tools. InterRAS Version 1 is based on the U.S. NRC’s RASCAL Version 2.1 code (NRC94) but was modified to allow assessment a greater range of accidents and to conform to the guidance in the IAEA Basic Safety Standard (IAEA 96) [3, 4].

The “Source Term to Dose” (ST-DOSE) model was used. This model contains tools to estimate the distance at which urgent protective actions may be needed based on nuclear power plant conditions or release rates. The ST-Dose model will first generate a time-dependent “source term”, the release rate for each radionuclide from the facility as a function of time. The time-dependent release rate (the “source term”) than provides the input to an atmospheric dispersion and transport model.

The model estimates radionuclide concentrations to downwind, both in the air and in the ground deposition. The calculated concentrations than are used to estimate projected doses. The dose pathways are: cloudshine from the plume, inhalation from the plume, and groundshine from deposited radionuclides (assuming 4 days of exposure to groundshine).

Radionuclide concentrations downwind from the NPP and projected doses due to the accident were estimated at distances of 1, 2, 5, 25 and 50 km from the site.

It is difficult to estimate doses from air, ground and food samples in time for protective actions to be effective. That is why operational intervention levels are used during an emergency. However, there may be a need to calculate the doses which may have been received once the detailed isotopic composition of the source term (air samples), ground composition and food contamination is known [2].

Consumption of water and locally produced food after the accident could increase the dose significantly, especially during the first vegetation period after the release. Therefore it is important to assess promptly and adequately contamination level of foodstuffs and dose for public in case of consumption of locally produced food for the purpose of initiation protective actions and/or food restriction measures.

The dose from the consumption of food contaminated by Cs-137, as one of the critical radionuclides during the emergency, were calculated based on measured concentrations in milk and leaf vegetables during an emergency period (first 30 days after the release). Dose assessment was made for rural population only.

Committed effective dose from consumption of food or soil were calculated by using the equation:

\[ \sum_{j} \left( C_{f,j} \times U_{\beta} \times DI_{i,j} \times CF_{s,j} \right) \times \prod RF_{f} = E_{\text{ing}} \]

(2)

where:

- \( E_{\text{ing}} \) – Effective dose from ingestion, mSv;
- \( C_{f,j} \) – Activity concentration of radionuclide \( i \) in food \( f \) after processing, kBq/kg;
- \( U_{\beta} \) – The amount of food \( f \) consumed by the population of interest per day, kg/d or L/d. [5];
- \( CF_{s,j} \) – Ingestion dose conversion factor, mSv/kBq [6];
- committed effective dose from ingestion per unit intake of radionuclide \( i \)
- \( DI_{i,j} \) – Days of intake is the period food is assumed to be consumed, d. If \( T_{1/2} > 21 \) days use 30 days, if \( T_{1/2} < 21 \) days use the mean life \( (T_m) \) of the radionuclide:

\[ T_m = T_{1/2} \times 1.44 \]

(3)
where:

- $T_{1/2}$ – Radiological half-life;
- RF – Reduction Factor is the fraction of the contamination remaining after decay or some process $j$ used to reduce the contamination before food is released for consumption. In case of this study RF assumed to be 1, which means that the contamination of food is not reduced (conservative approach) [2].

3. **Results**

Results of calculation of total activity of the accidental release in case of DBA at NPP with a PWR reactor are presented in the Table 2.

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>Activity, Bq</th>
<th>Nuclide</th>
<th>Activity, Bq</th>
</tr>
</thead>
<tbody>
<tr>
<td>Kr-85</td>
<td>1,10E+11</td>
<td>Kr-85m</td>
<td>4,40E+12</td>
</tr>
<tr>
<td>Kr-88</td>
<td>1,30E+13</td>
<td>I-131</td>
<td>4,70E+11</td>
</tr>
<tr>
<td>I-132</td>
<td>6,70E+11</td>
<td>I-133</td>
<td>9,50E+11</td>
</tr>
<tr>
<td>I-135</td>
<td>8,30E+11</td>
<td>Xe-131m</td>
<td>1,80E+11</td>
</tr>
<tr>
<td>Xe-133m</td>
<td>1,10E+12</td>
<td>Xe-135</td>
<td>6,10E+12</td>
</tr>
<tr>
<td>Cs-134</td>
<td>4,20E+10</td>
<td>Cs-136</td>
<td>1,70E+10</td>
</tr>
<tr>
<td>Rb-88</td>
<td>1,30E+13</td>
<td>Ba-137m</td>
<td>2,70E+10</td>
</tr>
<tr>
<td>Kr-87</td>
<td>8,90E+12</td>
<td>Xe-133</td>
<td>3,20E+13</td>
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<td>Xe-135m</td>
<td>1,30E+11</td>
<td>Xe-138</td>
<td>3,20E+13</td>
</tr>
<tr>
<td>I-134</td>
<td>1,00E+12</td>
<td>Cs-137</td>
<td>2,70E+10</td>
</tr>
</tbody>
</table>

Total activity of the accidental release for all the scenarios of DBA was $1,1 \times 10^{14}$ Bq.

In case of “winter” scenario the highest value of total effective dose for population living in the near zone was $1,1 \times 10^{-1}$ mSv, bearing $70^\circ$ (south-west) at a distance of 1 km from the site, the highest value for the population living in the far zone was $1,3 \times 10^{-2}$ mSv, at a distance of 10,5 km at the same direction. Maximum values of cloud shine dose, ground shine dose and effective inhalation dose were $2,1 \times 10^{-2}$ mSv, $1,9 \times 10^{-2}$ mSv and $6,8 \times 10^{-2}$ mSv accordingly and observed at a distance 1 km from the site.

In case of “summer” scenario the highest value of total effective dose for population living in the near zone was $5,6 \times 10^{-2}$ mSv, bearing $70^\circ$ (south-west) at a distance of 1 km from the site. The highest value for the population living in the far zone was $3,2 \times 10^{-3}$ mSv, at a distance of 10,5 km at the same direction. Maximum values of cloud shine dose, and effective inhalation dose in summer were $1,0 \times 10^{-2}$ mSv and $3,0 \times 10^{-2}$ mSv accordingly and were observed at a distance 1 km from the site, ground shine dose were zero.

Class stability F (“winter” scenario) is the most unfavorable for radioactive release dispersion, as in this case the highest doses were observed. But dose reduction factors (e.g., building shielding factor) were not taken into account in calculations. In real situation doses for public in winter period would be much less that calculated numbers. Contrary to that, if the accident would happen in summer period, the doses for public could increase due to consumption of locally produced food which will appear to be contaminated.

Maximum ingestion doses for public were $1,13$ mSv from the consumption of contaminated milk and $0,39$ from the consumption of contaminated leaf vegetables at a distance of 3 km from the site. According to the Standards of Radiation Safety-2000 of Belarus there is no need recommendation to the public to avoid eating potentially contaminated food or milk [7].

4. **Discussion**

For facilities in threat category I (e.g., NPP) arrangements shall be made for effectively making and implementing decisions on urgent protective actions to be taken off the site.
These arrangements shall include the specification of off-site emergency zones for which arrangements shall be made for taking urgent protective action. These emergency zones shall be contiguous across national borders, where appropriate, and shall include:

(a) A precautionary action zone (3-5 km), for which arrangements shall be made with the goal of taking precautionary urgent protective action before a release of radioactive material occurs or shortly after a release of radioactive material begins, on the basis of conditions at the facility (such as the emergency classification) in order to reduce substantially the risk of severe deterministic health effects.

(b) An urgent protective action planning zone (25 km), for which arrangements shall be made for urgent protective action to be taken promptly, in order to avert doses off the site in accordance with international standards.

(c) Food restriction planning radius (300 km) – this is the area where preparations for effective implementation of protective actions to reduce the risk of stochastic health effects from the ingestion of locally grown food should be developed in advance.

In general, protective actions such as relocation, food restriction and agricultural countermeasures will be based on environmental monitoring and food sampling.

The sizes are shown in terms of a radius of a circle centered at the source of the potential release or criticality. However, the actual boundary of the zones should not be a circle but should be established to conform the geographical features such as roads, rivers or political boundaries.

In the result of our calculations and their analysis versus international recommendations it was found that after the DBA at NPP with a PWR-1200 reactor there is no need for providing sheltering for the public because the total effective dose will not achieve currently recommended generic criteria (10 mSv in 2 days). Besides, there is no need for evacuation, urgent decontamination, restriction of food, milk and water consumption (the total effective dose will not achieve 50 mSv in 1 week).

The absence of protective actions may be than revised based on prevailing accidental conditions, environmental measurements and dose values.

The results of the study could be used for emergency planning purposes.

5. References

COMPUTATIONAL MODELING OF RADIONUCLIDE RESUSPENSION

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ABSTRACT

Freshly fallen radionuclide materials after an energetic release from a Radiological Dispersal Device (RDD) or by a nuclear accident similar to Chernobyl can be resuspended back into air by wind or human activities. Resuspended microscopic radionuclides can be a threat to humans who are in the immediate vicinity of the fallout. Wind tunnel experiments were conducted at the Wehrwissenschaftliches Institut für Schutztechnologien (WIS) facility in Germany and Computational Fluid Dynamics (CFD) analysis using FLUENT software was conducted at the University of Ontario Institute of Technology (UOIT) in Canada to determine the resuspension factors of radioactive lanthnum oxide ($^{140}$La$_2$O$_3$) powder which was used as a surrogate for RDD fallout. Both experimental and computational results show a linear increase in resuspension factor from 0.25 \( \mu \text{m} \) to 1 \( \mu \text{m} \) particle sizes and exponentially decrease in resuspension factor from 1 \( \mu \text{m} \) to 6 \( \mu \text{m} \) particle sizes. Results found using FLUENT software over-predicted the resuspension factors for particle sizes between 1 \( \mu \text{m} \) and 3 \( \mu \text{m} \). The over-prediction was attributed to surface roughness effects, and suggests the need to include natural surface profile with microscopic surface roughness elements into full CFD model to simulate physical behavior of the particles trapped in between roughness elements.

1. Introduction

1.1 Background

Particle resuspension is a natural phenomenon. It is a common observation during the winter in Canada to see how freshly fallen snow resuspended off the ground during a snow storm. These snow particles resuspend off the ground by gusting wind or by mechanical lift forces imposed on them by rotating tires of moving vehicles.
Very close to the ground; kinetic energy of gusting wind or mechanical forces acting on microscopic particles provides the energy required for escape of particles from the surface against their static equilibrium. Particle size, density, Van der Waals forces, relative humidity of air, surface roughness, electrostatic forces are some of the factors dictate the static stability of a particle resting on a surface. The type of flow (laminar or turbulent), presence of turbulent burst-sweep events (Biswas & Eswaran, 2002) and mean flow acceleration/deceleration are some of the factors affecting particle detachment from the surfaces (Ibrahim, Dunn, & Qazi, 2008).

Numerous models have been proposed by researchers in the past to quantify the process of resuspension of microscopic particles. The most common models are Rock’n’Roll (R’n’R) model (Reeks & Hall, 2001), Force Balance model (Cleaver & Yates, 1973) and Energy-Accumulation model (Reeks, Reed, & Hall, 1988). In the R’n’R model particles oscillate about pivot points of asperities on the ground surface due aerodynamics forces and adhesive forces acting on them (Reeks & Hall, 2001). In the Force Balance Model particles are resuspended off the ground surface by turbulent bursts events when lift force generated by the bursts is greater than the adhesive force (Goren, 1970 and Zimon, 1964). The rate of removal of microscopic particles in the Force Balance Model is calculated by observing the spatial distribution and frequency of turbulent ejections close to the ground where particles were deposited. In the Energy-Accumulation Model particles are resuspended off the deposited surface when particles absorbed enough vibration energy to overcome the surface adhesive barrier at points of detachment. This model is very similar to Force Balance model, but in the Energy-Accumulation Model particles can be resuspended off the deposited surface even when the lift force is smaller than the adhesive force (Reeks and Hall, 1988). The Energy-Accumulation Model estimates the resuspension of particles based upon the adhesion of small elastic particles on an elastic structure under the influence of Van der Waals forces.

These models predict resuspension factors for a mono layer of microscopic spherical particles and also use experimental values to calculate some parameters in the models. Microscopic particles that would be generated after a Radiological Dispersal Device (RDD) may not be spherical if radioactive powders were generated using grinding, crushing or milling operations.

1.2 Objectives of the project

The objectives of the project are to find experimental resuspension factors of radioactive microscopic Lanthanum Oxide powder ($^{140}$La$_2$O$_3$) under controlled conditions and to compare the experimental resuspension factors with results found from Computational Fluid Dynamics (CFD) modeling of the resuspension process.
1.3 Journey of particles to average human breathing height

The journey of re-suspended radioactive microscopic particles up to the human breathing level can be divided into three main stages. The first stage is to bring freshly fallen microscopic particles up to the viscosity dominated boundary layer across the laminar sub-layer. The second is to bring particles from the edge of laminar sub-layer to the end of the turbulence boundary layer as depicted in Figure 1.

For smooth surfaces, particles resides within the laminar sub-layer (k<< thickness of laminar sub-layer) where k is average thickness of roughness elements on the surface. For rough surfaces particles reside outside the laminar sub-layer since roughness elements protrude far into turbulent flow (Sippola & Nazaroff, 2002). The final stage of the resuspension process is to bring particles from the edge of the boundary layer up to target average human breathing height (h) at a distance (d) downstream in the direction of free stream velocity as illustrated by Figure 2.
Resuspended microscopic particles quickly reach terminal velocity since the drag force acting on these microscopic particles is significantly larger compared to the forces due to gravity and buoyancy. The particles follow the streamlines especially in the free stream velocity region.

Most of the environmental boundary layers are fully developed and their thickness can easily be measured using a hotwire anemometer or can be calculated using empirical relationships. The value of Reynolds number downstream will indicate the nature of the flow within the boundary layer; whether the flow is laminar or in transition from laminar to turbulent or turbulent. Velocity distributions across the different regimes of the boundary layer can be calculated using the Blasius relationship for laminar flows and Log Law relationship for turbulence boundary layers (Muralidhar & Biswas, 1999). Values such as local Reynolds number and the distance from the fallout to the point of interest are important parameters that would assist emergency response managers to quickly find the resuspension factors from tabulated tables. These resuspension factors can then be used to assess airborne contamination levels from surface readings, and allow determination of vehicle and personnel movements, personal protective equipment and stay times in contaminated regions.

2. Resuspension experiment

2.1 Contaminated plate preparation for the experiment

The objective of the experiment was to find resuspension factors of freshly fallen radioactive material as a result of high energy dispersion by a RDD. Radioactive $^{140}$La$_2$O$_3$ powder was produced in a research reactor facility near Hamburg, Germany, by irradiating 1 g of non-
radioactive microscopic $^{139}$La$_2$O$_3$ power in a high-flux thermal neutron field. The nominal activity of 2 Ci (74 GBq) was achieved during the day of active experiment after mixing measured amount of active powder with inactive La$_2$O$_3$ powder to get total starting mass of 30 g. The mixture was dispersed inside environmentally controlled chamber at the Wehrwissenschaftliches Institut für Schutztechnologien (WIS) facility, simulating a high-energy dispersion by a RDD event as depicted in Figure 3. The powder was collected onto 2 mm thick 50 cm square plates by gravitational settling until required activity was achieved.

![Figure 3: View into the Contamination Chamber (through a lead-glass window)](image)

Particle distribution of freshly fallen La$_2$O$_3$ powder onto painted plates was measured using both a Malvern Spraytec laser diffraction system and Grimm aerosol spectrometers. The Malvern Spraytec laser diffraction system uses the laser diffraction principle to calculate ensemble particle size distribution. The cumulative mass fraction of La$_2$O$_3$ powder against average particle diameter was fitted into standard Rosin-Rammler (Wilbur & W., 1995) distribution as shown in Figure 4.
Equation (1) shows the standard Rosin-Rammler particle size distribution. This distribution normally occurs in solid powder materials generated by milling, grinding and crushing operation.

\[ R = \exp \left[ - \left( \frac{d}{d_m} \right)^n \right] \]  

(1)

Here, \( R \) is the retained mass fraction of particles with a diameter greater than \( d \), \( d_m \) is the mean particle diameter and \( n \) is a measure of the spread of particle size. The maximum particle diameter was measured as 12.5 \( \mu \)m; the mean diameter was calculated as 3.62 \( \mu \)m and minimum diameter was measured as 0.25 \( \mu \)m with a spread factor of 3. The total mass of the powder settled on to contaminated plates were calculated multiplying the time corrected activity by the specific activity of La-140.

After the contamination process, the plates were carefully transferred to a multi-probe radiation measuring system to make sure a homogeneous contamination was achieved during the gravitational settling process and confirmed the total amount of 5 MBq of La-140 per plate was achieved.

### 2.2 Experimental setup of the wind tunnel

Contaminated plates were brought to the test chamber of the wind tunnel and placed on the floor as depicted in Figure 5.
The test section of the wind tunnel at the WIS facility had a hexagonal cross section. The test section was symmetrical across the vertical plane running at the centers of horizontal sides. The wind tunnel test chamber was mapped using constant temperature hotwire anemometers prior to active trials. The average steady state velocity of the wind tunnel was measured to be 5.48±0.15 m/s. The relative humidity and temperature of the test chamber were measured to be 40±2% RH and 25°C respectively. Flow behavior close to the wind tunnel floor was visualized by injecting liquid nitrogen along the top surface of the floor to determine the most effective location to mount Grimm inlet ports to capture majority of resuspended La₂O₃ powder. The Grimm 1 and Grimm 2 were positioned at 2.5 cm and 115 cm downstream from the plates. Grimm spectrometers measure airborne aerosol particle distributions using 90° laser scattering techniques."
3. Computational Modeling

The CFD process can be divided into 3 main parts: mesh generation, solver and post processing of results. Mesh generation was performed by the Gambit 2.4.6 software and the Fluent 6.3.26 double precision solver was used as solver and postprocessor (Fluent Inc., 2010).

3.1 Mesh Generation

CFD simulation for the flat plate was performed in 2D due to the symmetric nature of the flow field in the area of the test section of the WIS experimental setup. The 2D mesh generated for the experimental setup had 946541 quadrilateral mesh elements. A very fine mesh close to the contaminated plate and the floor of the test chamber was selected to capture the viscous effects of the wall surfaces, where microscopic particles resides on contaminated plates. Figure 6 shows the mesh distribution and boundaries of the computational domain.

Figure 6: 2D Computational domain of WIS facility boundary conditions

Locations of upstream and downstream hotwire anemometers were selected as velocity inlet and pressure outlet boundary conditions respectively. Wall boundary conditions were introduced to the floor, wetted plate surfaces and ceiling. Turbulence intensities, I of the boundaries were derived from the relationship as shown in Equation (2).
\[ I = \frac{u'}{u_{avg}} \]  

Here \( u' \) is the fluctuating velocity component around the mean velocity, \( u_{avg} \). Fluctuating velocity components of the boundaries are equal to the standard deviations of the mapped velocity data (Santos, 2006). Percentage turbulence intensities of velocity inlet and pressure outlet were calculated to be 2.81% and 1.53% respectively. FLUENT recommends using turbulence length scale to introduce turbulence properties to velocity inlet and pressure outlet boundaries for non-circular cross-sections. Turbulence length scales were calculated using Equation (3). Turbulence length scale is a physical quantity, which defines the maximum size of the turbulence at boundaries (Fluent Inc., 2010)

\[ l = 0.07L \]  

The hydraulic diameter, \( L \) of the boundaries can be calculated using Equation (4)

\[ L = \frac{4P}{A} \]  

Here \( P \) is the perimeter of the wetted cross-section of the boundary and \( A \) is the cross-sectional area of the boundary.

3.2 CFD modeling of flow fields

The Continuity equation, momentum equations coupled with famous Shear Stress Transport (SST) k-omega turbulence model with transitional flow option was solved to find the flow fields of 2d computational domain. The SST k-omega turbulence model was selected to model the turbulent transport because of its capabilities of modeling turbulence transport parameters down to the wall surfaces (Fluent Inc., 2010) where the resuspension process initiates. Flow fields were solved until 1E-5 residual convergence criterion for continuity, momentum and turbulence transport parameters were achieved. Particles were then introduced to the converged flow at 100 \( \mu m \) vertically above the ground surface to avoid the effects of surface roughness of the contaminated plates.

3.3 Particle trajectories

Particle trajectories were calculated using a Lagrangian tracking scheme. This scheme assumes that the presence of particles in the flow does not affect the structure of the turbulent flow. This is a valid assumption since the mass concentration of suspended
radioactive La$_2$O$_3$ particles in the wind tunnel test chamber is very small compared to the
density of air (Sippola & Nazaroff, 2002).
To understand the paths of particles from the ground to breathing height; it is required to
look into the forces acting on them. The location of the particles in a 2D computational
domain using a rectangular Cartesian coordinate system can be defined by x and y
coordinates and time (t). Using Newton’s second law of motion, the rate change of velocity of
a particle in the x direction when drag force, gravity, buoyancy and the other forces such as
lift force acting on particles can be written as in Equation (5).

$$\frac{du_p}{dt} = F_d(u - u_p) + g_x \left( \frac{\rho_p - \rho}{\rho_p} \right) + F_x$$  \hspace{1cm} (5)

Here $F_x$ represents the other forces such as Saffman lift force, subsequent drift of the
particles across streamlines and Brownian motion. In Equation (8), $u$ is the continuous flow
velocity at x location of the particle, $u_p$ is the particle velocity and $F_D$ is the drag force per unit
mass acting on the particle. This drag force can be written using Equation (6).

$$F_D = \frac{18 \mu}{\rho_p d_p^2} \frac{C_D \text{Re}}{24}$$  \hspace{1cm} (6)

Here $\mu$ is the continuous flow dynamic viscosity and $\rho_p$ is the particle density. Reynolds
number, $\text{Re}$ and coefficient of drag $C_D$ of the particle can be written using Equation (7) and
Equations (8) respectively.

$$\text{Re} = \frac{\rho_p d_p |u_p - u|}{\mu}$$  \hspace{1cm} (7)

$$b_1 = \exp(2.3288 - 6.4581\phi + 2.4486\phi^2)$$
$$b_2 = 0.0964 + 0.5565\phi$$
$$b_3 = \exp(4.9050 - 13.8944\phi + 18.4222\phi^2 - 10.2599\phi^3)$$
$$b_4 = \exp(1.4681 + 12.2584\phi - 20.7322\phi^2 + 15.8855\phi^3)$$  \hspace{1cm} (8)

Here $b_1$, $b_2$, $b_3$ and $b_4$ are constants that depend on the shape factor of microscopic La$_2$O$_3$
powder particles. The Reynolds number, $\text{Re}_{\text{ sph}}$ in Equation (7) was computed with the
diameter of a sphere having the same volume. Shape factor $\phi$ is defined as in Equation (9).

$$\phi = \frac{s}{S}$$  \hspace{1cm} (9)
Where \( s \) is the surface area of a sphere having the same volume as the particle, and \( S \) is the actual surface area of the particle. For preliminary calculations, the particles are considered as spheres having shape factor of 1.

4. Results and discussions

The classical definition of the resuspension factor, \( K \) is defined as the ratio of airborne concentration, \( C \) at a reference location to surface contamination, \( SC \) where the particles were resuspended. This can be expressed as in Equation (10).

\[
K(m^{-1}) = \frac{C(\mu g m^{-3})}{SC(\mu g m^{-2})}
\]  

(10)

For this work, the resuspension factor of particle bins according to particle diameter can be written as Equation (11).

\[
K_B = \frac{N_G}{N_{CP}}
\]  

(11)

Where \( K_B \) is the resuspension factor for selected bin size. \( N_G \) is the total number of particles in bin \( B \) that crosses the face of the Grimm inlet, \( N_{CP} \) is the number of particles of bin \( B \) on contaminated plates at the beginning. Resuspension factors (\( K_B \)) were calculated for 24 bins of particle diameters from 0.25 \( \mu m \) to 12.5 \( \mu m \). The resuspension factors calculated for both experimental and computational (FLUENT) methods are depicted in Figure 7.
Both experimental and computational resuspension factors show similar trends up to 6 μm particle size. Particle resuspension factors increase from 0.25 μm to 1 μm almost linearly and decrease exponentially down to 6 μm, beyond this point experimental results started increasing exponentially while Fluent results stay almost constant.

5. Conclusion

Fluent results for particle size between 1 μm and 3 μm over predict the resuspension factors. This may be due to the fact that particles were introduced into the computational domain beyond the region where the effects of surface roughness place a major role in microscopic particles resuspension process. Further numerical studies are required to confirm this prediction. The difference between the experimental results and the numerical results of resuspension factors should provide the resuspension factor within laminar sub-layer. The calculation of resuspension factors within laminar sub-layer using the CFD method will be investigated in the future. Three dimensional CFD modeling of the total resuspension factor
from contaminated surfaces to the average human breathing height with the effects of laminar sub-layer and surface roughness is not possible with the computer capabilities currently available; instead one can place the particles at the edge of the laminar sub-layer if the resuspension factors of the laminar sub-layer is known. It is also demonstrated that the SST k-omega turbulence model with transitional flow option is an appropriate turbulence model to calculate turbulence properties close to the contaminated surface where turbulence plays a major role in initial resuspension of microscopic particles. Future work will be done to build a data base of particle resuspension factors from the ground surface to the end of laminar sub-layer for a range of surface roughness that will help to model the resuspension process in 3D computational domain with the computer capabilities available at present.

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Bibliography


DOEZOR2: A CODE TO ASSESS THE RADIOLOGICAL IMPACT OF EFFLUENT DISCHARGES

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ABSTRACT

DOEZOR2 (DOsis al Exterior en ZORita v. 2) comprises a suite of models and data management tools which can be used to perform the radiological impact assessments of routine and continuous discharges from nuclear power plants and nuclear fuel cycle facilities for regulatory purposes. The Code is based on Radiation Protection 72 and the Regulatory Guide 1.109 (USNRC) methodology. The code development was carried out by SOCOIN (Gas Natural Fenosa Group) and is implemented in José Cabrera NPP (or Zorita NPP). DOEZOR has been operating during ten years and is about to be updated in order to consider realistic doses methodology. The new software differs from its predecessor in a number of ways, including a new user interface, new radionuclides and several new capacities (realistic dose).

The model is developed to estimate radiological consequences of emissions from nuclear power plants. Internal exposure via inhalation and ingestion, external exposure from clouds and radioactivity deposited on the ground are included in the model. DOEZOR2 has been developed in Visual Fortran, using user friendly windows environment and modular architecture (easy to implement for other uses and installations).

The main features of the code include:

• Annual discharge to the atmospheric or river environment can be modelled.
• Dynamic systems dispersion of radionuclides released to the river (two reservoirs).
• A comprehensive list of exposure pathways.
• A suite of environmental transfer models to estimate the transfer of radionuclides through the environment.
• Site specific data (parameters of life habits).
• Results in terms of individual doses (three age’s groups), using effective dose as defined in Council Directive 96/29/EURATOM and dose coefficients from U.S. Federal Guidance Report 13.
• Results in “MS Word form” for periodic information to authorities.

This software could be implemented anywhere, considering the particular characteristic of the site for the plant or facility.

1. Introduction

The Council Directive 96/29/Euratom of 13 May 1996 laying down basic safety standards for the protection of the health of workers and the general public against the dangers arising from ionizing radiation specify in the Article 45:

“The competent authorities shall ensure that dose estimates from practices referred to in Article 44 are made as realistic as possible for the population as a whole and for reference groups of the population in all places where such groups may occur”
The assessment of dose to a population due to discharges from NPP is an important part of the system of radiological protection.

In assessing the radiological impact of discharges of radionuclides, any methodology must give consideration to the scenarios for release, the pathways for transport through the environment and the modes of exposure of the receptor (humans). Thus, a source-pathway-receptor analysis is appropriate. It will be cover all stages of an assessment of doses to members of the public. The five typical stages are, in sequence: identification and quantification of sources, modelling the transfer into the environment, determination of exposure pathways, identification of reference groups, and estimation of doses to the reference groups.

DOEZOR2 (DOsis al Exterior en ZORita version 2) comprises a suite of models and data management tools which can be used to perform the radiological impact assessments of routine and continuous discharges from NPP and others nuclear fuel cycle facilities. The Code is based on Radiation Protection 72 and the Regulatory Guide 1.109 (USNRC) methodology. The code development was carried out by SOCOIN (Gas Natural Fenosa Group) and has been implemented in José Cabrera NPP (Zorita).

2. Methodology

For DOEZOR2 the sources are atmospheric and liquid discharges, with the liquid discharges being routed to Tajo River.

The equilibrium approaches are employed for chronic releases. Therefore, for example, uptake by plants from soil uses a concentration ratio approach, and uptake by terrestrial animals uses a transfer factor in which the radionuclide concentration in a particular food product is taken to be directly proportional to the rate of intake of that radionuclide by the animal.

2.1 Atmospheric discharges

For discharges to atmosphere the transfer and dispersion of the radionuclides between the release point and the receptor location was determined by sectorised Gaussian plume dispersion modelling. (Results obtained from XOQDOQ model), assuming cautious values for critical group locations (Farms, Villages) and taking into account the range of meteorological conditions that occur in the course of a year.

Pathways considered:
- Ingestion of radionuclides in terrestrial food
  - Milk (Cow, Goat&Sheep )
  - Meat (Beef, Lamb)
  - Vegetables (Fresh, No Fresh)
- Inhalation of radionuclides in the atmosphere;
- External irradiation from radionuclides in the atmosphere.
- External irradiation from radionuclides deposited on the ground.

It is important to consider the agricultural practice around the nuclear installation. The calculations of doses from meat and milk are generally based on cows. Local information indicates that there are no cows in the vicinity of the site but that sheep or goats are present, then such pathways have been considered.

2.2 Aquatic discharges

Discharges to the aquatic environment can result in significant different doses for the same release rates depending on the receiving water body. The doses received from discharges to a system of water body would be dependent on the volumetric flow of the river. Dynamic systems
The dispersion of radionuclides released to the river has been implemented in DOEZOR2 taken into account two reservoirs (Zorita, Almoguera) and the monthly flow of the Tajo River.

**Pathways considered:**
- Ingestion of water
- Ingestion in the main aquatic foods (fish)
- Ingestion of radionuclides in terrestrial food irrigated by receiving water body.
  - Milk (Cow, Goat, Sheep)
  - Meat (Beef, Lamb)
  - Vegetables (Fresh, No Fresh)
- External irradiation from radionuclides on river sides or sediments.

![Fig. 1. Pathways considered in aquatic discharges](image)

**2.3 Identification of reference groups**

Reference groups are intended to be representative of individuals likely to receive the highest doses. Given that the definition of a reference group requires that the habits of the group are reasonably uniform. Generic habit data can be used where site-specific information is not available.

**2.4 Dose Coefficients**

For dose assessments the dose coefficients for inhalation and ingestion was published in the Council Directive 96/29/EURATOM and transposed to Spanish law in RPSCRI (Ref. 3) and should be used. The Directive 98/C 133/03/EURATOM recommends that it is sufficient to consider three age groups: 1-year olds, 10-year olds and adults. The dose coefficients for external exposition to soil and sediment are from Federal Guidance Report 13 “Cancer Risk Coefficients for Environmental Exposure to Radionuclides: CD Supplement” (Ref. 8).

**3. DOEZOR Code**

DOEZOR2 has been developed in Visual Fortran, using ergonomic windows and modular architecture (easy to implement for other uses and installations). The program is developed to estimate radiological consequences of routine emissions from José Cabrera NPP. Internal exposure via inhalation and ingestion, external exposure from clouds and radioactivity deposited on the ground and sediments have been implemented. There are two interfaces depending on the users (Normal user, Expert user). The "normal" user (not expert) can introduce the monthly data discharges in easy menus in order to obtain the results in "MS
Word form", and electronic format for periodic information to regulatory body. In this case the user does not be allowed to change any parameter code.

The menus for “Expert user” are more complex than “Normal user”. In “Expert user” mode, the code allows the user to change the parameter code; the Dose Conversion Factors, The Foodchain Transfer Factors, etc. In this case the code allows the user to do exercises in order to identify the input parameters which have the greatest influence on the doses. This is done by performing a sensitivity analysis, in addition, the code provides additional information in the results and the user can analyze this information in a graphic mode.

The code has been validated with alternative calculations.

4. Applications
4.1 Retrospective Assessment
The primary application of DOEZOR2 is for retrospective assessments due to the discharge of radioactive effluents from Zorita NPP during normal operations. Retrospective assessments consider doses that are currently being received or that were received in the past:

Calculation of monthly doses to man from routine releases of reactor effluents. In this case the code provides impress reports for regulation purposes.

Realistic assessments of doses in order to perform epidemiological studies carry out by the regulatory body (Consejo de Seguridad Nuclear [CSN]). In this case we have been used a methodology reaches a consensus with CSN taking into account the “Guidance on the assessment of radiation doses to members of the public due to the operation of nuclear installations under normal conditions” (Ref 7). Zorita NPP became in operation in 1968 and has more than 60 municipalities in the 30 km around the plant. The task of calculation of realistic doses could be difficult if you have not a tool of calculation like DOEZOR2. In the case of the calculation of doses for epidemiological studies Figure 3 shows an example of the evolution of the annual dose for a specific municipality (Village 1) taking into account the real atmospheric discharges from the plant. In this figure we can see the evolution of annual dose in three groups of people and how the dose has been reduced in recent years. This reduction is due to different policies followed by the plant in procedures and equipment: effluent control; emission reduction as fuel change, improvement in inspections of fuel, etc.
4.2 Prospective Assessment

**Prospective assessments** consider doses that may be received in the future, e.g., from planned discharges. In this case judgement is needed on what may happen in the future, e.g., regarding changes in land use.

**Generalised Derived Constraints (GDC)** represent the annual discharge of a particular radionuclide by a single mode of discharge that is assessed to give an effective dose rate (of 0.02 mSv·y⁻¹ for discharges to atmosphere of 0.08 mSv·y⁻¹ for discharges to river) to the critical group.

**Screening factors** for atmospheric and surface water discharges have been calculated using the code. The factors used represent the maximum effective doses that could be received by a hypothetical critical group member for external and internal radiation from a unit concentration of a radionuclide discharged into the air or water.

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Plant operations
THE NUMERICAL SIMULATION OF THE WWER440/V213 REACTOR PRESSURE VESSEL INTERNALS RESPONSE TO MAXIMUM HYPOTHETICAL LARGE BREAK LOSS OF COOLANT ACCIDENT

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ABSTRACT

The reactor internals are designed to ensure cooling of the fuel, to ensure the movement of control assemblies under all operating conditions including accidents and facilitate removal of the fuel and of the internals following an accident. This paper presents preliminary results of numerical simulations of the WWER440/V213 reactor vessel internals (RVI) dynamic response to maximum hypothetical Large-Break Loss of Coolant Accident (LBLOCA). In the case of this LOCA accident it is assumed rapid “guillotine” break of one of the main coolant pipes and rapid depressurization of the primary circuit. The pressure wave spreads at the speed of sound, enters the reactor pressure vessel and causes deformation and stress in reactor vessel internals.

The purpose of this analysis is to determine reactor pressure vessel internals response and structural integrity due to rapid depressurization and to prove no permanent (plastic) deformations occur in the RVI which would prevent timely and proper activation of the control assemblies.

The finite element model was created by MSC.PATRAN and dynamic response was solved using MSC.DYTRAN. The model consists of reactor vessel internals (Lagrangian solid elements) and water coolant (Euler elements). Arbitrary Lagrangian Eulerian coupling was used for simulation of the fluid-structure interaction. The calculation assumes no phase change in the water.

1. Introduction

The functions of the reactor internals are to support the core, to hold the fuel assemblies in place, to direct coolant flow, to hold and protect control assemblies in normal operation conditions and accidents conditions. The design of the reactor core must secure, that all the internal components are designed, manufactured and assembled in such a way so that they can withstand static and dynamic loads during normal operation, abnormal operation and during project accidents to the extent necessary for securing of safe shutdown of the nuclear reactor, for maintaining the sufficient cooling of the reactor core.

In the case of a LOCA accident of a pressurized water reactor it comes to sudden drop of the pressure from nominal value to the pressure of saturated steam at the given coolant temperature in the point of pipeline rupture. A pressure wave is generated and it propagates at the speed of sound in the primary circuit. This wave enters the reactor pressure vessel and causes a pressure difference affecting the reactor vessel internals. It comes to unsymmetrical loads of the core barrel due to gradual propagation of pressure wave in the space between the core barrel and the reactor pressure vessel. These loads are only important in the initial phase of the accident – within a time interval of tenths of second after the occurrence of the accident. After this interval, the pressure comes to balance and the dynamic load of the reactor vessel internals disappears.

The numerical simulation of the WWER440/V213 reactor vessel internals response to LOCA accident was performed by three-dimensional finite element code MSC.Dytran.

2. Model

The main components of the WWER reactor internals are the core barrel, the core basket and the block of guide tubes. The core barrel supports the core basket and the block of guide tubes, and separates the cold leg from the hot leg. In its upper part the core barrel is fixed by elastic tube elements placed between the RPV cover and the core barrel flange.

In its lower part, the core barrel is fixed by keys which are welded to the reactor pressure vessel cylindrical part. These keys are important for safety because they restrain large
transverse motion of the core barrel while allowing unrestricted radial and axial thermal expansion. The bottom of the core barrel consists of the upper and the lower forged lattice, the vertical cylinder and thirty-seven guide tubes. These tubes contain the fuel part of the emergency control assemblies when they are in the bottom position e.g. in the case of a LOCA accident. The block of guide tubes of the reactor is held down by the RPV upper block and therefore it prevents axial displacement of the core, the core basket and the core barrel bottom in all operating conditions. It consists of the lower round plate for connection with the basket and the upper plate which serves as a support structure for the spring blocks.

The reactor internals structural calculation model consists of the core barrel, the core barrel bottom, the core basket and the block of guide tubes. The fluid model includes the whole volume of primary coolant inside the reactor. The geometry and the finite element mesh were created by means of MSC.PATRAN by using 150 000 eight-node hexahedral Lagrange solid elements and 270 000 eight-node hexahedral Euler elements with a mapped meshing. Arbitrary Lagrangian Eulerian coupling was used for simulation of the fluid-structure interaction between the structure and the fluid.

Following assumptions were considered in the model:

- the reactor pressure vessel was considered as a rigid boundary for the fluid elements and for all contacts defined in the model
- contacts pairs were modelled between following parts of the reactor internals:
  1. the core barrel flange – the shoulder of the reactor pressure vessel (RPV)
  2. the core barrel horizontal sealing ring between cold and hot legs – RPV
  3. core barrel grooves – keys welded to inside surface of the RPV cylindrical part
- the telescopic spring blocks pushing the block of guide tubes and the core basket in axial direction were modelled by type SPRING element
- the gravity of the reactor internals and primary coolant was considered, the mass of the fuel was distributed to the fuel assembly elements
- no phase change in the fluid elements was assumed

![Fig 1. Finite-element mesh of the structural model and the fluid inside the reactor](image)
3. **Material**

The reactor vessel internals are manufactured from titanium stabilized austenitic stainless steel 08CH18N10T. The bilinear elastic-plastic material model was used for structure elements.

The slightly compressible fluid material was used for primary coolant inside the reactor. The density and the bulk modulus were used as input data for the fluid material model. It was assumed, that no phase change occurs during the short analysed period of the LOCA accident.

4. **Load**

In a LOCA accident of a pressurized water reactor the pressure in the point of pipeline rupture drops suddenly. The generated pressure wave enters the RPV and causes dynamic loads of the reactor internals. The global thermo-hydraulic calculation of the LOCA accident was performed by means of the RELAP5/Mod 3.2.2 code by using the six loops model of the WWER440 reactor cooling system. This code is dedicated to analyses of transients and accidents in cooling systems of light water reactors. It is based on the model of a two phase non-homogeneous non-steady thermo-mechanical system. In the calculation the break opening time 1ms was assumed.

The pressure time history calculated by the RELAP code for the ruptured cold nozzle and for the remaining cold and hot nozzles is shown in the Fig 2. This pressure time history was used as boundary condition at RPV nozzles cross-section in the dynamic analysis performed by means of the MSC.Dytran code.

5. **Calculation**

The numerical simulations of the WWER440/V213 reactor vessel internals dynamic response to the maximum hypothetical LOCA accident was solved by using the three-dimensional analysis code MSC.Dytran for analyzing the dynamic, nonlinear behaviour of solid components, structures, and fluids. This code is particularly suitable for analyzing short, transient dynamic events that involve large deformations, high degree of nonlinearity, and interaction between fluid and structures.

The MSC.Dytran uses an explicit solver which determines a stable time step based on the mesh size, the speed of sound, and the velocity. The dynamic unsymmetrical load of the reactor internals during the LOCA accident takes a few tenths of second and therefore the necessary analysis time was only 0.4 s. After this time the dynamic load of the vessel internals disappears and the monitored parameters are sufficiently stable. On the other side, due to the small element dimensions in the fuel assembly, the solver determined relatively small time steps so the calculation consisted of around 1.5 million cycles.

6. **Results**
The load of the reactor vessel internals during the LOCA is proportional to the pressure drop from the nominal pressure in the reactor to the pressure of saturated steam at the given temperature of the coolant in the ruptured nozzle. Therefore more severe conditions will occur if the LOCA accident is assumed in the cold leg of the primary circuit because the pressure drop is higher than in the case of the hot leg rupture.

The calculation has been performed for both described variants of a LOCA accident but only main results of a LOCA accident on the cold leg are presented in this paper because the length of the manuscript was limited. The local fluid pressures and velocities, reactor vessel internals displacements and stresses, and contact forces were obtained from the numerical simulation of the RVI response to the LOCA accident.

Immediately after the pipe break the pressure difference affects the core barrel wall opposite the broken nozzle. The pressure wave propagates from the ruptured nozzle down through the space between the reactor pressure vessel and the core barrel. This pressure wave causes asymmetrical load of the reactor vessel internals. The pressure time history opposite the broken nozzle and inside the core barrel is shown in the Fig 4. The maximal pressure difference $\Delta p = 5.5$ MPa was determined at the time of 5ms. Fig 3 shows the pressure distribution in the coolant and Fig 5 shows the stresses in the core barrel at various time steps. Immediately after the pipe break the stress maximum in the core barrel wall appears opposite the broken nozzle. Then the stress maximum moves towards the contact between the horizontal sealing ring and RPV. The maximum Von Misses stresses were found at the time of 5ms as it is shown together with the deformation in Fig 6. The contact forces between each core barrel groove and key were evaluated because they represent the maximal hypothetical load for detailed structural analysis of the key consoles and welds.

![Fig 3. Pressure (MPa) distribution at time steps 2ms, 4ms, 6ms, 10ms and 30ms](image)

![Fig 4. Pressure difference impacting the core barrel wall opposite the broken nozzle](image)
7. Conclusion

The nuclear power plant safety analysis guidelines define basic requirements and conditions for accident analyses. The most important acceptance criteria for the reactor vessel internals demands that RVI ensure the movement of the control assemblies under all operating conditions including accident. The numerical simulation of the WWER440/V213 reactor vessel internals response to the maximum hypothetical LOCA accident showed that during LOCA accident no such deformations will occur which could prevent unrestricted movement or proper activation of emergency control assemblies.

References

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RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS ACCIDENTS
FOR FRENCH PWRS

SOME IRSN VIEWS

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ABSTRACT

The radiological consequences of Design Basis Accidents (DBA) were evaluated in the first safety reports of French Pressurized Water Reactors (PWR), but only some DBA have been accounted for and disparate rules and assumptions have been adopted in the calculations. Since the 90’s, EDF (the French utility) and IRSN (the French TSO – Technical Safety Organisation - for the Safety Authority) have joined continuous efforts to define the radiological requirements, rules, methods and assumptions for the analyses.

A standard for radiological consequences assessment was issued in 2004 by EDF based on a conservative approach to assess the radioactive releases outside the containment buildings and on a realistic methodology to evaluate their impact on the population and on the environment. The IRSN review of this standard has been achieved in 2006 and the main conclusions have been used for the radiological consequences analyses performed for the 3rd periodic review of the 900 MWe reactors and the 1st periodic review of the 1450 MWe reactors. A second IRSN review based on the update of this standard has been achieved in 2009 for its application during the 3rd periodic review of the 1300 MWe reactors and for the EPR safety report.

The paper summarizes the current situation in regards to the radiological requirements, the general approach and the main rules to evaluate the radiological consequences of DBA. Moreover, it presents some outcomes of the work, including both suggestions for plant improvements and detection of the needs for future research. To illustrate these points, two examples of DBA are provided: Steam Generator Tube Rupture and Loss Of Coolant Accident.

The final part of the paper provides the IRSN’s vision for future, when reactors belonging to the Gen III (EPR) and Gen II (current PWRs) generations will be in simultaneous operation in France, as a consequence of the plant life extension underway. The design requirements regarding the amplitude of hypothetical release in case of accident are highly more ambitious for the EPR and justify, for IRSN, future works to improve the situation for Gen II reactors, for both DBA and severe accidents.
1. Introduction
The radiological consequences of Design Basis Accidents (DBA) were evaluated in the first safety reports of French Pressurized Water Reactors (PWR), but only some DBA have been accounted for and disparate rules and assumptions have been adopted in the calculations. Since the 90’s, EDF (the French utility) and IRSN (the French TSO – Technical Safety Organisation - for the Safety Authority) have joined continuous efforts to define the radiological requirements, rules, methods and assumptions for the analyses.

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The paper summarizes the current situation in regards to the radiological requirements, the general approach and the main rules to evaluate the radiological consequences of DBA. Moreover, it presents some outcomes of the work, including both suggestions for plant improvements and detection of the needs for future research. To illustrate these points, two examples of DBA are provided: Steam Generator Tube Rupture (SGTR) and Loss Of Coolant Accident (LOCA).

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Nota: In this paper, the French Nuclear Safety Authority (ASN) requests are based on the IRSN analysis which was approved by the French Advisory Committee for Reactor Safety.

2. Current situation
2.1 Radiological requirements/objectives
In France, there are no radiological requirements fixed by ASN for DBA. Nevertheless, ASN asked EDF to define some radiological objectives. These ones are defined for any DBA categories in term of qualitative and quantitative aspects. They encompass the following two periods of time:

- **Short term** (from some hours to some days) for which three exposition pathways are considered: exposure to the plume, inhalation and exposure to the deposits;

- **Long term**, based on the methodology called “4 exposition pathways – 50 years” for which, in addition of the short term exposition pathways, the ingestion is also taken into account.

In most cases, the EPR radiological objectives are more drastic than those of GEN II reactors. For example, for EPR DBA category 4, there shall be no necessity of protective measures for people living in the vicinity of the damaged plant (no evacuation, no sheltering) and, for GEN II reactors DBA category 4, there shall be no necessity of constraining protective measures for people living in the vicinity of the damaged plant (no evacuation).
In a general manner, IRSN deems that radiological objectives cannot be considered as criteria to assess the acceptability of consequences of accidents. IRSN underlines that the assessment of radiological consequences of accidents has to be integrated in a global process of enhancing the safety linked to the mitigation of the accidental exposure due to radiological releases for the persons of the public, the workers and the environment. All measures have to be taken to reduce as much as reasonably achievable these releases and their impact on the population and on the environment.

ASN asked EDF to implement, for each periodic safety review, a global approach aimed at increasing the plants safety, by identifying not only the accidents leading to the most important consequences for the population and the environment, but also the paths of releases and the main radiological exposition pathways in order to study possibilities to implement technical devices and arrangements aimed to reduce the consequences of accidents.

2.2 General approach and rules

The radiological consequences assessment is based on a conservative approach to assess the radioactive releases outside the containment buildings and on a realistic methodology to evaluate their impact on the population and on the environment. Nevertheless, for the last item, some assumptions have a certain degree of conservatism.

Any DBA categories presented in the safety report have to be assessed either by detailed study or by argument. In this last case, it is necessary to demonstrate that radiological consequences of the investigated accident are encompassed by those of an accident, with the same occurrence frequency, which has been studied in detail. It is necessary to focus on two kinds of accident:

- **Accidents related to the core** (e.g. LOCA, SGTR, LOOP): the radiological consequences of each accident are assessed on the basis of the thermal-hydraulic study presented in the safety report, i.e. with the same single failure criterion and the same rules related to systems operation (including filtration and venting systems), in accordance with the rules of DBA studies.

- **Accidents not related to the core** (e.g. failure in liquid or gaseous processing system, fuel handling accident): the same rules that those used for DBA studies, related to the operation of filtration and venting systems are applied. A single failure criterion have to be sought regarding to the containment function.

The main parameters of any DBA categories presented in the safety report have to be identified. For some of these parameters, a sensitive study has to be done and referenced in the safety report. The sensitive studies leading to significant consequences have to be presented in the safety report.

3. Analysis outcomes

3.1 Steam Generator Tube Rupture

The investigated SGTR accident, defined as a category 4 DBA, depends on the reactor type:

- **For 900 MWe and 1300 MWe reactor series**: a guillotine break of one steam generator tube plus one main steam safety valve locked in open position during its solicitation in water are considered;

- **For N4 reactor series**: a break guillotine of two steam generator tubes located on the same steam generator is considered.

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1 LOOP: Loss Of Off-site Power
These two kinds of accidents are due to the different reactor series designs. In particular, 900 MWe and 1300 MWe reactor series are composed of a single main steam relief train while N4 reactor series have two main steam relief trains.

The single failure criteria used for the thermal-hydraulic study of these accidents are chosen with the objective to maximize the liquid releases to the environment:

- For 900 MWe and 1300 MWe reactor series: the mechanical lockage in close position of one main steam relief train of the affected steam generator is considered;
- For N4 reactor series: the loss of one diesel is considered.

Main consequences of SGTR are related to the risks of contamination of the secondary side inventory (mainly affected steam generator) by leakage of radioactive coolant coming from the primary side (as the primary coolant is more or less contaminated by corrosion and fission products, in the limit of technical specifications, corresponding to continuous operation with a limited amount of defective fuel rods) and possibly discharge of activity (either in steam or liquid phase) to the atmosphere via the main steam relief trains and the main steam safety valves.

The three releases pathways are the following ones:

- Steam release by affected steam generator;
- Liquid release by affected steam generator;
- Steam release by non affected steam generator.

Even if the accident investigated for the 900 MWe, 1300 MWe and N4 series is a little bit different due to the design, the SGTR accident is the DBA leading to the most important radiological consequences. As a result, and in accordance to the IRSN view presented in chapter 2.1, ASN asked EDF to define a strategy to reduce as much as reasonably achievable the radiological consequences of this DBA. EDF has to present the goals, the principles and the technical solutions (retained and not retained) of this strategy during the periodic safety review of the 1300 MWe reactor series starting in 2010.

### 3.2 Loss Of Coolant Accident

For the 900 MWe reactor series, the investigated LOCA, defined as a category 4 DBA, is a guillotine break of the reactor coolant system. In comparison with the radioactive fission products released from the fuel rods, getting defective as a consequence of the accident, all other activity releases into the containment are assumed negligible.

The two main release pathways are the following ones:

- Leak of the containment reactor building;
- Leak of systems used during the accident and located outside the reactor building (i.e. safety injection system, spray system…).

During the 900 MWe reactor series periodic safety review, IRSN focussed on the release pathway leading to a leak through the reactor cavity and spent fuel pit cooling and treatment system tank. This release pathway leads to a direct leak to the environment i.e. without any filtration. The IRSN analysis has shown that 19 % of the total $^{131}$I release come from this release pathway for the reference study and 38 % for a sensitive study increasing the leakage flow rate into the tank. As a result, and in accordance to the IRSN view presented in chapter 2.1, ASN asked EDF to implement a specific device in order to reduce as much as reasonably achievable the radiological consequences of this DBA. Some technical solutions were proposed by EDF and are still under analysis at IRSN.
3.3 Need of Research activities

The second IRSN review of the new standard for radiological consequences assessment has been achieved in 2009 and led to new issues still needing future research. This part of the paper focus on the iodine species released in case of SGTR.

As a matter of fact, even if there is no damage to the core, the reactor coolant water contains dissolved iodine, with at least the initial activity during normal operation increased with a possible iodine spike activity generated by the accident or a prior initiating transient. During the accident, primary coolant water flashes at the break and a part of the dissolved iodine is released into the gaseous phase, the remnant remaining in the liquid phase. From a prospective study performed by IRSN, the iodine chemistry involved during the SGTR leads to a rapid conversion of iodide ions (I⁻) into iodate (IO₃⁻) and hypoiodous acid (HOI) due to radiolysis and low iodine concentrations (~10⁻⁹ mol.L⁻¹). The HOI molecule becomes volatile in the temperature range (250-300°C) of interest. As a result, iodine release could be higher than other fission product releases like caesium, except noble gases, because of the carry over phenomena, the partitioning into the steam phase due to flashing and the mass transfer from liquid steam generator side to gaseous phase.

After analysing such DBA, IRSN noticed that it remained a lack of information related to iodine chemistry. Indeed the partitioning of iodine between the liquid and the gaseous phases does not only depend on thermal-hydraulics but also on chemical processes. These processes take place in conditions (pressure, temperature, iodine concentration and radiation level) far from the ones presently investigated in severe accident R&D and for which the existing models are validated. A new experimental database is consequently needed to extend, to DBA and in particular in case of SGTR, the validation domain of iodine chemistry models used for severe accident source term evaluation.

4. Plant life extension

In near future, the EDF request for plant life extension of current reactors beyond 40 years will be subject to ASN and IRSN investigations. Plant life extension of current reactors will lead in France to simultaneous operation of Gen III reactors (EPR) and Gen II reactors during a long period of time. The design requirements regarding the amplitude of hypothetical release in case of accident are highly more ambitious for the EPR and justify, for IRSN, future works to improve the situation for Gen II reactors, for both DBA and severe accidents. The IRSN view presented in chapter 2.1, i.e. all measures have to be taken to reduce as much as reasonably achievable the releases, related to an accident, and their impact on the population and on the environment, will be already applicable but, in the framework of plant life extension, the releases and their impact should be drastically decreased. As a result, IRSN deems that EDF has to reconsider the approach of the risk of nuclear accident for GEN II reactors. In particular, IRSN considers that the qualitative radiological objectives defined for EPR could be applied on GEN II reactors. Moreover, IRSN deems necessary that EDF defines more drastic quantitative radiological objectives than those presently existing. This could be one of the key issues in the decision to allow the operation of current reactors beyond 40 years.

IRSN underlines that this view is applicable for both DBA and severe accident. In order to reduce the releases; in particular iodine releases, some technical solutions have to be discussed with EDF. In case of severe accident, it could be interesting to review the current emergency filtered containment venting system. In this goal, some technical devices, like a scrubber filter, should be investigated. This device could also be used to reduce the radiological consequences of a Steam Generator Tube Rupture.
5. **Conclusion**

This paper presents the approach adopted by IRSN to assess the radiological consequences of DBA for French PWRs.

After a short overview of the current situation in regards to the radiological requirements, the general approach and the main rules to evaluate the radiological consequences of DBA, the paper presents some suggestions for plant improvements and needs for future research. Moreover, a vision for future, when reactors belonging to the Gen III (EPR) and Gen II (current PWRs) generations will be in simultaneous operation in France as a consequence of the plant life extension underway, is provided.

The next step of the work will be the 3rd periodic safety review of the 1300 MWe reactors and the EPR safety report. In these frameworks and in particular for the 1300 MWe reactors, some design improvements could be proposed by EDF or underlined by IRSN as for the 3rd periodic safety review of the 900 MWe reactors.

Moreover, the planned plant life extension will be demanding the definition of higher safety requirements and investigation of practical, efficient and reasonable devices allowing a much better control of the accident consequences for both DBA and severe accident than presently achieved.

6. **References**


ANALYSIS OF LOSS OF HEAT REMOVAL ACCIDENTS IN RBMK-1500 REACTOR AND SPENT FUEL POOLS DURING DECOMMISSIONING

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ABSTRACT

Ignalina NPP is the only nuclear power plant in Lithuania consisting of two units, commissioned in 1983 and 1987. Both units are equipped with channel-type graphite-moderated boiling water reactors RBMK-1500 with nominal electrical power 1500 MW. Unit 1 of Ignalina NPP was shutdown for decommissioning at the end of 2004 and Unit 2 – at the end of 2009. The design of RBMK, as channel type reactors, allows to change the fuel assemblies on-line. This online refuelling puts specifics to the accidents during refuelling but the integral reactor core characteristic remains almost constant during the reactor operation. Thus, in RBMK reactors the effect of integral reactivity dependence on fuel burn-up is minimized. Because of this, in RBMK-1500 reactor the fuel assemblies remain for long time inside reactor core after the final shutdown. Only after few years the fuel (part of which is relatively fresh) can be removed into spent fuel storage pools. For this purpose each reactor unit is equipped with a system of spent fuel pools. Even at shutdown conditions the decay heat is generated in reactor core and in the spent fuel pools. The heat should be removed to keep the temperature below the water saturation temperature. This paper discuss the results of accidents, leading to failure of heat removal from the fuel assemblies in reactor core and in the spent fuel pools. The analyses of most probable Beyond Design Basis Accidents, related to the loss of heat removal from the fuel in shutdown RBMK-1500 reactor and spent fuel pools, were performed by employing best-estimate system thermal hydraulic code RELAP5. The results of performed analysis demonstrated, that in any case the fuel heat-up processes in reactor and in spent fuel pools are slow. The available reserve of time for operator’s actions is demonstrated and possible mitigation measures are discussed.

1. Introduction

Ignalina NPP is the only nuclear power plant in Lithuania consisting of two units, commissioned in 1983 and 1987. Both units are equipped with channel-type graphite-moderated boiling water reactors RBMK-1500 with nominal electrical power 1500 MW. Unit 1 of Ignalina NPP was shutdown for decommissioning at the end of 2004 and Unit 2 – at the end of 2009. The detailed description of Ignalina NPP with RBMK-1500 reactors is presented in book [1]. The design of RBMK, as channel type reactors, allows to change the fuel assemblies on-line. This online refuelling puts specifics to the accidents during refuelling but the integral reactor core characteristic remains almost constant during the reactor operation. Thus, in RBMK reactors the effect of integral reactivity dependence on fuel burn-up is minimized. Because of this, in RBMK-1500 reactor the fuel assemblies remain for long time inside reactor core after the final shutdown. Only after few years the fuel (part of which is relatively fresh) can be removed into spent fuel storage pools.

At shutdown reactor the pressure in Drum Separators (DS) of Reactor Cooling System (RCS) is atmospheric. Even at shutdown conditions the decay heat is generated in reactor core and the heat should be removed to keep the temperature below the water saturation temperature (according to technological regulation of RBMK-1500, a cooled reactor is...
considered as subcritical reactor with water temperature not above 80 °C in the RCS and graphite stack temperature not above 100 °C). Normally at shutdown RBMK reactor RCS is filled with water up to connection of Steam Water Piping (SWP) to DS. In this case the heat from Fuel Channels (FC) is removed by natural circulation of water (Fig. 1). The hot water from FCs (6) passes in SWPs (7) and DSs (1). DSs and SWPs are elevated ~15 m higher reactor core and placed in DSs compartments. The 1661 pipes of SWP (7) with external diameter 76 mm and approximately length 27.4 m each create the significant heat transfer area (Fig. 1). From SWP the heat is removed by air employing ventilation system of DSs compartments. Such type of reactor cooldown in technological regulation of RBMK-1500 is called “cooldown in natural circulation mode” and it allows to maintain the temperature of water in RCS below 100 °C. Because the acceptance criterion for fuel cladding is 700 °C [2], the intactness of fuel cladding is assured.

![Diagram of RBMK-1500 reactor cooling system](image)

**Fig. 1. Schematic representation of RBMK-1500 reactor cooling system during water natural circulation:** 1 – DSs, 2 – Main Circulation Pumps (MCP) suction header, 3 – tripped MCPs, 4 – MCP pressure header, 5 – group distribution header, 6 – FC, 7 – SWP, 8 – steam lines

Each reactor unit is equipped with a system of spent fuel pools. All process operations related to handling of the spent fuel are performed in central hall or in the spent storage pools hall. Spent Fuel Pools (SFP) of Ignalina NPP are designed for:
- storage of non-cut Spent Fuel Assemblies (SFA) in deep compartments of storage pool (Rooms 236/1 and 236/2);
- storage of spent nuclear fuel in shipping casks in shallow compartments of the storage pool after cutting SFAs (Rooms 336, 337/1, 337/2, 339/1, and 339/2).

The spent fuel assemblies, prepared to be cut in the “hot” cell, are accumulated in separate pool (Room 234). The loading of the shipping casks is performed in two pools (Rooms 338/1 and 338/2). Also there are transport corridor (Room 235) for the transportation of SFAs and shipping casks between the pools and transport corridor (Room 157) for transportation of fuel assemblies between spent fuel hall and reactor hall. The whole complex of storage pools of the spent fuel storage and handling system comprises 12 pools (Fig. 2). The reloaded fuel assemblies remain in the pool for at least a year, after which they may be removed to be cut (in “hot” cell) and then to be loaded into the 102 placed shipping casks. The shipping casks with spent fuel assemblies are stored in the storage pools until they are loaded into the protective casks CASTOR or CONSTOR to be further transported to the dry spent fuel storage facility. The detailed description of spent fuel pools in Ignalina NPP is presented in [3, 6].
The main problem in nuclear energy is providing of safety at all stages of lifetime of nuclear installations in conditions of normal operation, accidents and at shutdown. Prevention of radioactive substances release in an environment or minimization of releases is the primary goal of the organizations operating nuclear power plants. Usually the means of safety improvement are based on the analysis of transients, design and Beyond Design Basis Accidents (BDBA). Consideration of BDBA development as a result of which the failure of nuclear fuel can occur, is actually not only for accidents with the core degradation, but also for accidents in SFPs as well. The main goal of this paper is to discuss the consequences of the most probable beyond design basis accident station blackout, which leads to loss of heat removal from the fuel in shutdown RBMK-1500 reactor and in SFPs of Ignalina NPP Unit 2. For both cases the analyses were performed by employing RELAP5/MOD3 code [4]. The RELAP5 model of RCS of RBMK-1500 is presented in [5], the model of SFP - in [3].

2. Results of analyses

2.1. Cooling of shutdown reactor in station blackout case

For the analysis of consequences of beyond design basis accident – station (i.e. power plant) blackout during reactor cooling in coolant natural circulation mode it was assumed that the reactor core is loaded by uranium – erbium fuel of 2.6 % 235U enrichment at average depth of burnup 25 MW days/kgU (the averaged parameters for reactor core) The decay heat of fuel assembly in one Fuel Channel (FC) at such condition (1 day after reactor shutdown) is already reduced from approximately 2500 kW (at normal reactor operation) down to 13.99 kW. The constant decay heat level was assumed into calculations as conservative assumption. Decay heat of a reactor, conservatively was accepted as total decay heat in 1661 average loaded FC. It was assumed that at the moment of time t = 0 seconds the operation of ventilation system in DSs compartments stops. The make-up of reactor cooling system by water using design means is unavailable due to total loss of electric power supply (station blackout). At the initial stage temperature of water in FC is 110 – 115 °C and temperature of graphite stack is 120 °C (Fig. 3). After switching-off of ventilation system the water in SWP will not be cooled. Water with temperature 115 °C passes into DSs, where the pressure is atmospheric. Water in DSs boils and the steam is discharged through opened steam discharge valve (this valve is opened at shutdown reactor to maintain atmospheric pressure in RCS). After switching-off of ventilation system the water flow rate through FC slightly increases (Fig. 4), because the heat removal rises upwards (from SWP passes into DS), that increases driving force. As the steam is discharged into steam condensing pools and there is no additional make-up of RCS by water, the volume of water in DS decreases (Fig. 5). DSs are emptied in 8 hours after loss of power supply, simultaneously coolant natural circulation is terminated (Fig. 4), because after emptying of DSs the circuit of circulation interrupts. After the termination of coolant natural circulation the
boiling of water in FCs begins, as it is seen from Fig. 3 (temperatures of the core components match temperature of water saturation). After the beginning of water boiling in FCs (8 hours later after loss of power supply), pressure of water column in the FC – DS path starts to decrease, because the part of water is evaporated. This leads to decrease of pressure in fuel channels (Fig. 6). After the water boiling beginning in FCs the temperatures of the core components increases a little and remain within the limits of 130 – 100 °C. At water boiling in FC the temperatures of core components even decrease (Fig. 3) because of pressure drop in FC (Fig. 6) and reduction of water saturation temperature. It will take additional approximately 10 hours, till all water from fuel channels will be evaporated.

Thus, in case of station blackout during reactor cooling by coolant natural circulation mode, the dry out of fuel assemblies can occur not earlier than 18 hours after the beginning of accident. Before this, the fuel is reliably cooled by coolant natural circulation mode and water boiling in fuel channels with steam removal through steam discharge valve. The operators have enough time to find the possibilities to provide make-up of RCS by water, using non-regular, non-designed water sources.

### 2.2. Processes in spent fuel pools in station blackout case

It was assumed during the modelling, that totally there are 7901 spent fuel assemblies with storage time from 8 days to 5 years after removal from reactor. The shipping casks with fuel assemblies are placed in SFP in two layers one above the other, according to the real shipping casks loadings in the SFP. The non cut assemblies with shorter storage time in SFP are placed in lower level of water pools. The total decay power in SFP is equal 4253 kW. The initial temperature of water inside SFP is 50 °C and outside temperature is equal 20 °C. The water level in SFP is 16.9 m from bottom of the pool.

After loss of heat removal from pools, due to decay power of fuel assemblies, the water temperature in SFP starts to increase. The characteristic periods of accident are:

<table>
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<tr>
<th>Station blackout during reactor cooling by coolant natural circulation mode</th>
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<td><strong>Fig. 3. Behaviour of temperature of the core components</strong></td>
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<td><strong>Fig. 4. Behaviour of coolant flow rate through FCs of one RCS loop</strong></td>
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<td><strong>Fig. 5. Behaviour of water volume in both DS of one RCS loop</strong></td>
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<td><strong>Fig. 6. Behaviour of pressure in RCS</strong></td>
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After loss of heat removal from pools, due to decay power of fuel assemblies, the water temperature in SFP starts to increase. The characteristic periods of accident are:
- Heat up of water in the pools from 50°C up to 100 °C (up the boiling condition of water is reached);
- Water boiling in the pools, evaporation of water and associated decrease of water level, down to uncovering of fuel assemblies starts;
- Overheating of uncovered fuel assemblies, up to loss of intactness of fuel rod claddings;
- Release of radioactive fission product, oxidation of zirconium and hydrogen generation;
- Melting of zirconium and fuel.

The time to reach water boiling can be find from simple energy balance equation:

\[ t_b = V (h_b - h_i) / (Q v'), \quad (1) \]

where: operating data for the Ignalina NPP are as follows:
- \( Q = 4253 \text{ kW} \) – decay (residual) power,
- \( h_b = 419 \text{ kJ/kg} \) – water enthalpy for the boiling (for water 100°C),
- \( h_i = 209 \text{ kJ/kg} \) – water enthalpy in SFP at the beginning of loss of SFP cooling 50 °C,
- \( V = 5070 \text{ m}^3 \) – SFP water volume,
- \( v' = 0.001 \text{ m}^3/\text{kg} \) – specific volume of water in the pool.

Thus, from equation (1): \( t_b = 69 \text{ h} \).

The evaporation rate of water after boiling start is:

\[ w_s = Q / (h_s - h_b), \quad (2) \]

where: \( h_s = 2676 \text{ kJ/kg} \) for the saturated steam at pressure 1 bar;
Then, from (2): \( w_s = 1.884 \text{ kg/s} \). Assuming that heat from the SFP is removing only by evaporation of water, time of the SFA uncovering start after the boiling water start in the SFP can be calculated by the formula:

\[ t_u = \Delta V / (v' w_s), \quad (3) \]

where: \( t_u \) – time of the SFA uncovering start after the boiling water start;
- \( \Delta V = 1230 \text{ m}^3 \) water volume above SFA (SFP cross section area 300 m², water level above SFA in SFP – 4.1 m).

Then, from (3), the time interval between start of water boiling in SFP up to decrease of water level, down to uncovering of fuel assemblies: \( t_u = 181 \text{ h} \).

The modelling of loss of heat removal by employing RELAP5 code was performed only for the first characteristic periods of accident (water heat up, evaporation and beginning phase of SFA dryout. At temperature higher as 800 °C the specific phenomena of severe accident (steam – zirconium and air – zirconium interactions and etc.), for which the RELAP5 code is not qualified, would occur. These phenomena were analysed using code for severe accidents ATHLET-CD (see [6]). During the modelling, the real compartments of SFP were modelled by simplified one cylinder and 4 concentric rings model. First two central rings modelled the water pools with SFA, the third ring models the gaps between walls of pools and grids with spent fuel assemblies. The last ring models the walls of SFP from ferroconcrete and the outside air, which is around the SFP [3]. As it is shown in Fig. 7, at first after loss of heat removal, the water temperature in SFP start to increase. It takes approximately 127 h to heat up of water from 50 to 100 °C. After start of water boiling, it takes additional more than 250 h when the water level decreases below the top of fuel assemblies (see Fig. 8) with longest storage period. The start of uncovering of fuel assemblies may be indicated by fast increase of fuel temperature (Fig. 7). The calculated, by employing RELAP5, time intervals of first two characteristic periods of accident (water heat up and level decrease) are significantly longer, comparing to “hand” calculations (see \( t_b \) and \( t_u \)). This is because, in RELAP5 model the heat losses through the SFP walls are evaluated. The RELAP5 calculations demonstrate, that heat transfer from SFP to environment is limited due to the heat removal from outside walls to the air. The calculated amount of heat, which can be removed as the heat loses from outside area of concrete SFP building walls at water boiling in SFP condition, is equal to \( \sim 580 \text{ kW} \). The total decay heat in SFP 4253 kW
(according assumed initial conditions). Thus, the part of heat loses is small, comparing to the total decay heat, but it allows to slowdown the fuel heat up process.

<table>
<thead>
<tr>
<th>Station blackout in spent fuel pools</th>
</tr>
</thead>
<tbody>
<tr>
<td><img src="image1.png" alt="Graph showing fuel and SFP wall temperatures" /></td>
</tr>
<tr>
<td><img src="image2.png" alt="Graph showing water level in SFP" /></td>
</tr>
<tr>
<td>Fig. 7. Behaviour of fuel and SFP wall temperatures</td>
</tr>
</tbody>
</table>

3. Conclusion

The performed analysis using RELAP5 code shows that the RBMK-1500 reactor after shutdown can be cooled by natural circulation of water. In the loss of cooling system make-up and loss of heat removal case (station blackout) during reactor cooling by coolant natural circulation mode, the dry out of fuel assemblies can occur 18 hours after the beginning of accident. In the case of station blackout the increase of water temperature in the spent fuel pools from 50°C up to 100 °C (to reach boiling conditions) will takes approximately 127 hours. The evaporation of water volume above the spent fuel assemblies level, when uncovering of fuel starts will takes approximately 250 additional hours. Thus, the operators have enough time to find the possibilities to provide make-up of RCS and spent fuel pools by water, using non-regular, non-designed water sources.

4. References

APPLICATION OF REALISTIC (BEST-ESTIMATE) METHODOLOGIES FOR LARGE BREAK LOSS OF COOLANT (LOCA) SAFETY ANALYSIS: LICENSING OF WESTINGHOUSE ASTRUM EVALUATION MODEL IN SPAIN

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ABSTRACT

When the LOCA Final Acceptance Criteria for Light Water Reactors was issued in Appendix K of 10CFR50 both the USNRC and the industry recognized that the rule was highly conservative. At that time, however, the degree of conservatism in the analysis could not be quantified.

As a result, the USNRC began a research program to identify the degree of conservatism in those models permitted in the Appendix K rule and to develop improved thermal-hydraulic computer codes so that realistic accident analysis calculations could be performed. The overall results of this research program quantified the conservatism in the Appendix K rule and confirmed that some relaxation of the rule can be made without a loss in safety to the public. Also, from a risk-informed perspective it is recognized that conservatism is not always a complete defense for lack of sophistication in models.

In 1988, as a result of the improved understanding of LOCA phenomena, the USNRC staff amended the requirements of 10 CFR 50.46 and Appendix K, “ECCS Evaluation Models,” so that a realistic evaluation model may be used to analyze the performance of the ECCS during a hypothetical LOCA. Under the amended rules, best-estimate plus uncertainty (BEPU) thermal-hydraulic analysis may be used in place of the overly prescriptive set of models mandated by Appendix K rule. Further guidance for the use of best-estimate codes was provided in Regulatory Guide 1.157

To demonstrate use of the revised ECCS rule, the USNRC and its consultants developed a method called the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology as an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifying the uncertainties in a LOCA analysis. More recently the CSAU principles have been generalized in the Evaluation Model Development and Assessment Process (EMDAP) of Regulatory Guide 1.203.

ASTRUM is the Westinghouse Best Estimate Large Break LOCA evaluation model applicable to two-, three- and four loop Pressurized Water Reactor as well as passive plants (AP1000). This method follows the steps in the CSAU methodology. The uncertainty analysis (Element 3 in the CSAU) follows a direct Monte Carlo sampling procedure combined with non-parametric order statistics procedure. ASTRUM methodology is based on the use of WCOBRA/TRAC computer code which was validated against a large set of Separate and Integral Effects Tests (SETs and IETs).

The objective of this paper is to discuss the main features of the ASTRUM Evaluation Model and its compliance with the Spanish regulatory requirements.
1. INTRODUCTION

The Westinghouse realistic large-break LOCA evaluation methodology using the automated statistical treatment of uncertainty method (ASTRUM) has received final USNRC approval in November of 2004, and is documented in Reference [1]. Since receiving approval, the ASTRUM methodology has been applied or is in the process of being applied to more than 25 US, European, Asian and South American nuclear power plants.

ASTRUM represents the second generation of best-estimate large-break loss-of-coolant accident (LBLOCA) evaluation models developed by Westinghouse. Westinghouse’s previously approved best-estimate LBLOCA methodology is described in Reference [2] for Westinghouse designed 3- and 4-loop plants with emergency core cooling system (ECCS) injection into the cold legs. Both methodologies are patterned after the Code Scaling, Applicability, and Uncertainty (CSAU) methodology developed under the guidance of the US Nuclear Regulatory Commission (USNRC) (Boyack et al., 1989) [4]. In this paper these previously approved methodologies will be collectively referred to as the Code Qualification Document (CQD) methodology.

“First generation” CQD methodologies are characterized by an approximate approach, based on the response surface technique, to the estimate of the 95th percentile peak cladding temperature (PCT), local maximum oxidation (LMO) and core wide oxidation (CWO). For all practical purposes, these methods result in the determination of the 95th percentile, that is not associated with a specific confidence level.

ASTRUM represented an improvement to the uncertainty methodology of these first-generation best-estimate methods and is applicable to all Westinghouse designed 3- and 4-loop plants with emergency core cooling system (ECCS) injection into the cold legs; to Westinghouse designed 2-loop plants with upper plenum injection (UPI); to Combustion Engineering (CE) plants and passive plants (AP1000).

Like the CQD methodologies, ASTRUM is also patterned after the CSAU methodology. The only significant difference from the CQD methodology is in the application of the uncertainty analysis to the pressurized water reactor (PWR) large break LOCA scenario. In ASTRUM, a statistical approach to the estimate of the 95th percentile PCT, LMO and CWO is adopted and the confidence level achieved in the NRC-approved methodology is of a 95% confidence that the calculated maximum values for PCT, LMO and CWO are higher than the actual 95th percentile PCT, LMO and CWO.

The engine of the ASTRUM methodology is the WCOBRA/TRAC thermal-hydraulic computer code. WCOBRA/TRAC has been shown to adequately model LBLOCA phenomena, a detailed assessment was made through comparison to experimental data extended to more than 150 Separate Effect Tests (SETs) and Integral Effect Tests (IETs).

WCOBRA/TRAC [5, 6, 7] combines two-fluid, three field, multidimensional fluid equations used in the vessel with one-dimensional, 5-equations, Drift Flux formulation used in the loops to allow a complete and detailed simulation of a PWR. This best-estimate computer code contains the following features:

• Ability to model transient three-dimensional flows in different geometries inside the vessel.
• Ability to model thermal and mechanical non-equilibrium between phases
• Ability to mechanistically represent interfacial heat, mass and momentum transfer in different flow regimes.
• Ability to represent important reactor components such as a fuel rods, steam generators, reactor coolant pumps, etc.
A typical calculation using WCOBRA/TRAC begins with the establishment of a steady-state, initial condition with all loops intact. Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into one of the loops. The evolution of the transient proceeds continuously, using the same computer code. Containment pressure is modeled with a time dependent pressure table. Containment pressure is calculated using the COCO code [8] and mass and energy releases from WCOBRA/TRAC calculation.

The ASTRUM analysis flow chart in which all uncertainties of the LOCA parameters are accounted for to estimate a PCT, Local Maximum Oxidation (LMO) and Core Wide Oxidation (CWO) at 95-percent probability is summarized below:

- Plant Model Development

In this step a WCOBRA/TRAC model of the plant is developed. A high level of noding detail is used in order to provide an accurate simulation of the transient. However, specific guidelines are followed to ensure that the model is consistent with models used in the code validation. This results in a high level of consistency among plant models, except for specific areas dictated by hardware differences, such as in the upper plenum and lower plenum, or the ECCS injection configuration.

- Assessment of Uncertainty

The ASTRUM methodology is based on order statistics. The technical basis of the order statistics is described in Reference [1]. The determination of the PCT uncertainty, LMO uncertainty and CWO uncertainty relies on a statistical sampling technique. According to the statistical theory, 124 WCOBRA/TRAC calculations are necessary to assess against the three 10CFR50.46 criteria (PCT, LMO, CWO) [9].

ASTRUM Uncertainty Contributors considered in the analysis are the following:

- ECCS Parameters
  Accumulator liquid volume, Accumulator pressure, Accumulator liquid temperature, Accumulator surge line hydraulic resistance, Safety Injection temperature

- RCS Parameters
  Pressurizer pressure, Average fluid temperature

- Power Distribution Parameters
  Core Power, Power Distribution Shape, Peaking Factors, Burnup

- Global Models
  Break Type, Break Size, Break flow model, Downcomer Condensation, Break flow path resistance (nozzle and pump resistance)

- Local Models
The uncertainty contributors are sampled randomly for their respective distributions for each of the WCOBRA/TRAC calculations. The list of uncertainty parameters, which are randomly sampled for each time in the cycle, break type (split or doubled ended guillotine), and break size for the split break are also sampled as uncertainty contributors within the ASTRUM methodology.

Results from the 124 calculation are tallied by ranking the PCT from the highest to the lowest. A similar procedure is repeated for LMO and CWO. The highest rank of PCT, LMO, and CWO will bound 95 percent of their respective populations with 95-percent confidence level.

2. SPECIFIC MODIFICATIONS TO EXTEND ASTRUM APPLICABILITY TO SPANISH PWRs

While the ASTRUM methodology has been approved by the USNRC, specific modifications were required to extend the applicability of this methodology to the analysis of Spanish PWRs. In particular, the Gadolinia Implementation.

Other modifications and adaptations were required as a result of the regulatory review performed by CSN [10]. In this category it is highlighted the FQ considerations and the Total Oxidation calculation including the effect of time in life.

These modifications are discussed in this paper.

2.1 Gadolinia Implementation

Two ENUSA supplied $\text{UO}_2$-$\text{Gd}_2\text{O}_3$ Gadolinia Products [11],[12] are considered for the Spanish Westinghouse designed PWRs namely the Gad 3% and Gad 8%. Meanwhile the Gad 8% product includes a UO2 enrichment cutback, the Gad 3% product does not. These products are not typically used in the US. Other neutron poison product called IFBA is used instead.

Although the Gadolinia and IFBA poisons certainly have different neutronic characteristics, for LBLOCA purposes, there are no primary differences, with both fuel types differing from their un-poisoned base product as follows:

Different Fuel Temperatures and Rod Internal Pressures fuel performance characteristics and Reduced net reactivity that translates in reduced peaking factors (Gad 8% and IFBA not for Gad 3%), this general variation is mainly due to the different thermal conductivity with respect to the un-poisoned base product and to the UO2 enrichment cutback.

As for the IFBA product, the gadolinia dopped pellets assessment is treated as a local effect using the boundary conditions from the non-gadolinia study. Since the differences between UO2 rods and UO2 with gadolinia rods are due mainly to the material properties, and the limited amount of Gadolinia rods present in some fuel assemblies neither the initial conditions not the thermal-hydraulic transient obtained for a homogeneous core of UO2 will be affected due to the presence of gadolinia.

The main updates made were twofold, first to introduce new Gad fuel thermal conductivity properties and second to allow for burnup dependent Gadolinium Radial Pellet Power Profiles.
2.2 FQ Considerations

Steady-state axial distributions are established by core loading pattern and burnup. The axial distribution tends to vary widely as a result of changes in external controls such as boron or control rods.

The maximum of the radial distribution times the maximum of the axial distribution is the total core peaking factor FQ (i.e. maximum linear heat rate divided by the core average linear heat rate).

Predictions of FQ are accurate to within 5% at > 95% probability. Local variations in pellet and subchannel geometry are also considered for the hot rod.

Transients are simulated in the core design process, yielding a wide range of possible power distributions and FQ. Beside FQ, the axial distribution is further specified by defining the average relative power in the bottom and middle third of the core.

Consistently with the USNRC RG 1.157 [3], in the Westinghouse best-estimate LOCA methodology there are therefore two components to the total core peaking factor, a nominal term plus an uncertainty term. The treatment of each of these two terms is discussed in detail in the following part of this Section.

Nominal Total Core Peaking Factor

The most common mode of operation for a nuclear power plant is "baseload" (i.e., full power, all control rods out). In baseload operation, FQ varies slowly with time. However, the Technical Specifications allow for transient operation.

Load Follow maneuvers and other operational transients introduce relatively short lived skewed power shapes with higher peaking factor compared to equilibrium, steady state conditions.

Limiting transient power distributions are generated during the core design analysis to confirm that maximum values will remain below limits established in the plant technical specifications.

The calculated maximum peaking factor is obtained from the envelope that is generated by these distributions. Peaking factors encountered during normal steady-state operation will be far below these calculated values.

In the Westinghouse best-estimate LOCA methodology, the Nominal Total Core Peaking Factor FQ is sampled from a uniform distribution that is bounded by the maximum transient total core peaking factor (upper bound, which is defined as the technical specification limit minus the uncertainty) and by the maximum nominal baseload steady state total core peaking factor (lower bound). These values are defined on a plant specific basis.

In the sampling approach used in the USNRC-approved ASTRUM methodology, it will be assumed that for a given reactor state, the nominal FQ is any value from the baseload value to the maximum predicted transient value, with equal probability. The plant is therefore essentially assumed to be always in transient conditions, and nominal FQ is always sampled to be higher than the calculated maximum baseload value.

As discussed above, this is a clearly conservative assumption, that the sampling range is clearly conservative with respect to the FQ histogram. Moreover, there are two additional conservatisms included in this evaluation.
First, all FQ in the histogram are assumed to be equally probable, while the lower FQ values, corresponding to baseload conditions, have a significantly higher probability than the higher transient FQ value, since plants traditionally operate at steady state condition (lowest nominal FQ value) much more frequently than at transient operation conditions (load follow, abnormal events, etc) (up to highest nominal FQ value).

Second, the plant is always assumed to be operating at full power conditions, without consideration for the fact that the very consideration of transient FQ values requires transient conditions to exist.

**Treatment of Uncertainties in the Total Core Peaking Factor**

Uncertainties in the total core peaking factors are then added to the nominal value sampled as described above. These uncertainties are composed of several independent subcomponents and are used in the ASTRUM methodology as a hot rod and hot assembly global uncertainty and a hot spot local uncertainty.

**Regulatory Considerations:**

CSN accepted the USNRC-approved ASTRUM treatment for the probability distribution function (pdf) but emphasized that the Probability distributions assigned for input variables should not underestimate the uncertainty of these input variables, at least in the range of values considered as conservative and also required a demonstration of the compatibility of the pdfs with the operation limit conditions permitted in the Tech Specs. Furthermore, the plant should demonstrate that such a probability distribution functions conservatively bounds the operation values so the plant is covered by the LBLOCA analysis.

CSN requires that all the pdfs for input uncertain parameters permit an adequate exploration in the limit of Tech Spec specially in the conservative area.

This practically translates into eliminate the lower bound of the Nominal Total Core Peaking Factor FQ pdf and concentrate the sample in the upper limit defined as the FQ technical specification limit minus surveillance uncertainties.

**2.3 Local Maximum Oxidation Calculation including effect of time in life**

There is not a standardized methodology for the addressing of total oxidation (transient + pre-transient pre-LOCA) in either the ASTRUM topical report or its Best Estimate predecessors, or in any other generic correspondence with the USNRC. The pre-LOCA oxidation is accounted for beyond ASTRUM analysis. In order to verify the compliance with the 10CFR50.46 Local Maximum Oxidation limit of ECR = 17% an evaluation can be performed separately that account for both LOCA and Pre-LOCA oxidation.

**3. REASONS TO MOVE TO BEPU METHODOLOGIES: EMERGING REGULATORY ISSUES AND MARKET DRIVERS**

The Westinghouse Best-Estimate LOCA analysis methodology was first approved by the US NRC in 1996. These methods are capable of addressing some of the more complex thermal-hydraulic behavior associated with a loss-of-coolant accident. Also, from a risk-informed
perspective it is recognized that conservatism is not always a complete defense for lack of sophistication in models.

The need to understand the complex phenomenon associated with a loss of coolant accident are highlighted when the power rating of a nuclear power plant is increased such as will be the case for uprating projects. The older Appendix K methods are not sufficiently sophisticated to address some of the issues that have been raised by the regulators in response to plant power upratings. Therefore Westinghouse utilizes Best Estimate LBLOCA methods when performing a comprehensive power uprating project or any other major project. Appendix K remain valid for assessments pursuant to the reporting requirements of 10 CFR 50.46; and, evaluations to support minor plant, fuel design, or other input changes that would normally be handled under 10 CFR 50.46 and/or 10 CFR 50.59.

As examples of emerging issues can be cited: Downcomer boiling effects, Burnup effects relative to UO2 thermal conductivity degradation and Embrittlement due to hydrogen pickup, Pre-transient oxidation added to transient oxidation for comparison with 17% limit of 10CFR50.46, Stable and sustained quench calculation. These emerging issues may need margin generation obtained with the use of advanced methods for LOCA analyses.

In addition, there are significant plant operating benefits to be realized with the application of the Westinghouse Best-Estimate Methodology for analyzing the large break LOCA design basis accident. A Best-Estimate analysis does not require that each plant operating parameter be combined at its most limiting condition in a single calculation of the PCT. Instead, a large array of expected operational conditions is considered in multiple calculations. A statistical treatment of the results allows the determination of the PCT at the 95th percentile with 95% confidence, for a defined plant operating space. The analysis statistically combines the results of many different plant operating conditions and physical model uncertainties so the PCT at the 95th percentile with 95% confidence is anticipated to be substantially less than that currently calculated with different Evaluation Models (i.e. Appendix K approach). The calculated oxidation is also anticipated to be reduced. Since additional margin to the USNRC 10 CFR 50.46 regulatory limit is anticipated, the application of the BEPU method allows plants to use this margin to relax or change certain plant operating parameters that would otherwise be prohibited by other LBLOCA methods. These can include: plant power uprates, improvement in core design with increases in core peaking factors and longer fuel cycles, cope with equipment degradation for example minimum safety injection flows and increases in steam generators tube plugging, Relaxation in Technical specification limits as accumulator specifications, pumped ECCS delay times, Tavg, and uncertainties, among others and/or Margin.

4. SUMMARY AND CONCLUSIONS

ASTRUM methodology has been favourable informed by the CSN with minor conditions and has been successfully applied for first time in Spain.

This methodology relies on extensive automation with significant quality improvements and is a relatively simple engineering process that creates margin and permits the LBLOCA analysis for power uprates.

ASTRUM is now a proven technology that has been applied or is in the process of being applied to more than 25 US, Asian European and South American nuclear power plants and recently in Spain.

The methodology is well positioned to address emerging issues in the new regulatory environment.
REFERENCES


ABSTRACT

CIEMAT, former Junta de Energía Nuclear (JEN) started nuclear research at the 60th decade, focussed on the development of pacific uses of Nuclear Energy. At that time, CIEMAT research and pilot plants developed involved the whole nuclear fuel cycle steps. It means from the uranium recovery to the spent fuel reprocessing. With this scope a plutonium research laboratory was constructed and operated from 1961 to the 90’s focussed on chemistry of plutonium studies, separation processes and radiochemical analyses, in order to assist the working pilot plants at the Centre.

Thereafter, as the result of the changes on the research objectives of CIEMAT, the plutonium laboratory suffered several modifications and finally it was safety stopped due to the obsolescence of its equipments and auxiliary systems.

Present paper shows the D&D activities performed and techniques developed to avoid alpha emitter contamination. In every dismantling phase there were established the measures of operational radiological protection adapted to the radiological risk. Dosimetric controls realized during dismantlement showed that incorporation of radionuclides was not detected.

Radiological final control was performed applying the derivate levels of declassification to request the installation decommissioning.

1. Introduction

CIEMAT, former Junta de Energía Nuclear (JEN), was created in 1951 with the aim of promoting nuclear energy and its peaceful uses in Spain. Progressively the construction of Pilot Plants linked to Nuclear Fuel Cycle was promoted. JEN never came to conduct research, or any pilot plant was built, in order to enrich natural uranium.

Research Centre has 19.41 ha, with a total of about 76 buildings.

During 1951-1984 period, 59 nuclear and radioactive facilities were built related to benefit of uranium ore, radiochemical laboratories, fabrication of fuel for research reactors, two research reactors (JEN-1 and CORAL), hot cells, reprocessing plant for
the spent nuclear fuel in the JEN-1 experimental reactor, a laboratory for manufacturing and dispensing of isotopes for use in medicine, etc.

In 80’s decade energy research policy changed and Centre objectives opened to other energy resources. This fact, together with the obsolescence of facilities and laboratories promoted a progressive dismantling programme.

In January 2000, the so call PIMIC Project was created with the following objectives:
- D&D of old and stopped nuclear facilities
- D&D of obsolete radioactive laboratories in order to adapt the radiological condition of these to the current legislation
- Site restoration

In this frame several facilities has been dismantling such as fuel fabrication pilot plant, checking tanks for liquid waste control, laboratories related to uranium recovery from raw materials, radiochemical laboratories associated to pilot plants, etc.

Paper shows dismantling works carried out to get the closure of a Pu research laboratory.

2. Installation description and initial radiological characterization

Pu installation was form by several laboratories and places; two of them used as radiochemical laboratories and two more dedicated to the development of radiometric techniques for Pu and several alpha emitters analyses. The others were those containing auxiliary systems (ventilation system and tanks for liquid waste control), offices, radiological dressing room, etc.

Only one of the two chemical laboratories contained obsolete equipment at the time of the action. It was equipped with five glove boxes for Pu and Am handling and a shielded box for samples excitation made of steel and lead.

Laboratory was completed with work surfaces, an specific globe box ventilation systems, facility ventilation system, the liquid effluent system, intake pipes of various gases as well as distilled water, electrical system, pressure meters, fire control system. No equipment were located in the rest of laboratories at the beginning of decommissioning activities an exception of the auxiliary systems

Initial radiological control showed that main alpha contamination values were at glove boxes and its ventilation system. Although alpha contamination was found in the fluid system and in the filters of the facility ventilation system

On having carried out the measurement of smears taken inside the boxes on a surface of 100 cm$^2$ the following values were obtained:

<table>
<thead>
<tr>
<th>Box of Gloves</th>
<th>Am-241(Bq)</th>
<th>Pu(238+239+240)(Bq)</th>
<th>Pu-241(Bq)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Nº 1</td>
<td>4,66 10$^4$</td>
<td>8,82 10$^3$</td>
<td>1,17 10$^4$</td>
</tr>
<tr>
<td>Nº 2</td>
<td>1,07 10$^5$</td>
<td>8,33 10$^1$</td>
<td>2,35 10$^2$</td>
</tr>
<tr>
<td>Nº 3</td>
<td>8,96 10$^3$</td>
<td>5,21 10$^2$</td>
<td>6,46 10$^2$</td>
</tr>
<tr>
<td>Nº 4</td>
<td>1,33 10$^3$</td>
<td>2,93 10$^3$</td>
<td>3,55 10$^3$</td>
</tr>
</tbody>
</table>
Biggest risk for the personnel during the dismantling was the risk of internal contamination by incorporation (inhalation), for being radionuclides of high radiotoxicity (low LDCA).

It was required by the Spanish regulatory commission (CSN) to approve in advance the works procedures and radiological protection instructions of each dismantling step.

3. Dismantling activities
3.1. Glove boxes
Dismantling considered the following steps:
- Glove boxes decontamination before dismantling.
- Fixing the remaining contamination by especial thick paint
- Development of a confined cut system to avoid contamination spread and to reduce handle tools contamination.

To avoid the dispersion of the contamination during the dismantling of the boxes, is conditioned a confined enclosure, split into three areas and a committal system was defined for the courts.

- Court area. Area potentially with major contamination risk, it had a system air extraction (with filters HEPA)
- Area of undressed
- Area of conditioning wastes

The confined enclosure was also used in the remaining decommissioning activities.
3.2. Dismantling of auxiliary systems.

Confined philosophy was also implemented to carry out these works. In case of liquid waste system it was required to take residuary liquid waste out before cutting pipe circuits.

Fig. 2. a) HEPA filters boxes of installation ventilation system. b) Filters box dismantling detail

Fig. 3. Pipes sealing and cutting. Right: Liquid waste tanks at the facility.

Once equipments and systems were completely taken out, walls were treated for final radiological control with the objective of reaching the clearance levels.
4. Radiological protection

For every specific activity specific instructions of Radiological Protection were prepared, the meetings ALARA were supported and the Radiation Work Permit were expressed fulfilling the procedures of the CIEMAT Radiological Protection Service (RPS)

The whole involved personnel was classified like exposed category worker A and during the whole dismantling it was submitted to the control of:

- External Dosimetry: TL allocation with monthly reading
- Internal Dosimetry: CRC (entire organism control and lung) often half-yearly and bioelimination control (urine’s analysis to determine Pu/Am) often every two months.

Related to radiological classification of dependencies two transit points was established. The dependency where tasks of court and conditioning of generated waste were performed it was classified like of Limited Permanence Zone. Rest of dependencies was classified as Survey Zone.

The radiological control of the personnel was carried out with help of the portable monitors before its exit of transit’s points. Operational dosimeter control was carried out to the personal and environment

The personal protection equipment used in the dismantling is shown in figure 5. It included cotton / poplin coverall, Tyvek, respiratory protection mask of compressed air PA90, two pairs of fibre / latex or nitrilo gloves, integral protection Tychem, suit, two pairs of hoses, safety footwear.

A procedure of specific work was established for the dressed and the undressed

Fig. 5. Left: Detail of waste treatment; Right: personal protection equipment
5. Waste generation

Volumes of waste are summarized in Table 1, taken into account the materials type and radioactive waste classification (low level radioactive waste or waste at the clearance levels).

<table>
<thead>
<tr>
<th>Material</th>
<th>No compactables</th>
<th>Compactables</th>
<th>Sludge</th>
<th>Liquid</th>
</tr>
</thead>
<tbody>
<tr>
<td>RBMA (m³)</td>
<td>14,82</td>
<td>28,82</td>
<td>1,56</td>
<td>7,9</td>
</tr>
<tr>
<td>At clearance level (m³)</td>
<td>53,32</td>
<td>1,54</td>
<td>--------</td>
<td>--------</td>
</tr>
<tr>
<td>% At clearance level</td>
<td>78%</td>
<td>5%</td>
<td>--------</td>
<td>--------</td>
</tr>
<tr>
<td>Total Activity (MBq)</td>
<td>1690*</td>
<td>91,3</td>
<td>--------</td>
<td>1,52</td>
</tr>
</tbody>
</table>

*sludge is included

6. Conclusions

The dismantling of the installation was carried out in accordance with those indicated in the operational procedures, no significant operational deviate from them occurred.

Then, despite the most important risk was the superficial contamination, no personal contamination took place. Of the results of internal dosimetry none of the involved persons got radionuclides incorporation.

The radiological final survey was carried out using non spectrometric portables monitors. These surveys verified the fulfilment of derivate clearance levels.

All the surveys units were in accordance withe the Sign Test. This implies that the residual activity value was below of derivate clearance levels established for the facility and therefore such surveys units were cleared.
02.06.2010

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Reactor Technologies
EUROPAIRS PROJECT: CREATING AN ALLIANCE OF NUCLEAR AND NON-NUCLEAR INDUSTRIES FOR DEVELOPING NUCLEAR COGENERATION

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ABSTRACT

Developers of High Temperature Reactors (HTR) worldwide acknowledge that the main asset for market breakthrough is its unique ability to address growing needs for industrial cogeneration of heat and power (CHP) owing to its high operating temperature and flexibility, adapted power level, modularity and robust safety features. HTR are thus well suited to most of the non-electric applications of nuclear energy, which represent about 80% of total energy consumption. This opens opportunities for reducing CO2 emissions and securing energy supply which are complementary to those provided by systems dedicated to electricity generation.

A strong alliance between nuclear and process heat user industries is a necessity for developing a nuclear system for the conventional process heat market, much in the same way as the electro-nuclear development required a close partnership with utilities. Initiating such an alliance is one of the objectives of the EUROPAIRS project just started in the frame of the Euratom 7th Framework Programme (FP7) under AREVA coordination.

Within EUROPAIRS, process heat user industries express their requirements whereas nuclear industry will provide the performance window of HTR.

Starting from this shared information, an alliance will be forged by assessing the feasibility and impact of nuclear CHP from technical, industrial, economical, licensing and sustainability perspectives.

This assessment work will allow pointing out the main issues and challenges for coupling an HTR with industrial process heat applications. On this basis, a roadmap will be elaborated for achieving an industrially relevant demonstration of such a coupling. This roadmap will not only take into consideration the necessary nuclear developments, but also the required adaptations of industrial application processes and the possible development of heat transport technologies from the nuclear heat source to application processes.

Although only a small and short project (21 months), EUROPAIRS is of strategic importance: it will generate the boundary conditions for a rapid demonstration of collocating HTR with industrial processes as proposed by the European High Temperature Reactor Technology Network (HTR-TN).

1. Introduction

On 1 September 2009 started a most challenging European project called EUROPAIRS (End-User Requirements fOr industrial Process heat App lications with Innovative nuclear Reactors for Sustainable energy supply), and funded through a grant of the European Commission. EUROPAIRS is the logical consequence of the recommendations of the Industrial Users Advisory Group (IUAG) implemented within the RAPHAEL Integrated Project. The Industrial Users Advisory Group (IUAG) was composed of representatives of utilities, heat users, vendors of chemical plants, nuclear engineering companies and vendors of nuclear power plants. The major activity of the IUAG was to investigate how to enter into and to foster the dialogue between HTR promoters and end-users. The IUAG concluded that active involvement of end-users in HTR projects is the best way to raise the interest and to
pave the way for mutual understanding of the benefits and the issues at each side. The objectives and the work programme of EUROPAIRS have been defined in order to implement the recommendations of the IUAG with the aim to define a promising roadmap for the coupling of a nuclear reactor designed for cogeneration with a heat consuming plant in order to replace the conventional heater and thus reducing significantly the CO₂ emission. Consequently EUROPAIRS shall identify the boundary conditions for the viability of nuclear cogeneration systems connected to conventional industrial processes and to initiate the partnership of nuclear organisations and end-user industries, which would be deployed in a further step to develop a Demonstrator. The boundary condition framework will define technical, industrial, economical, licensing and safety requirements for the nuclear system, the processes that can consume the energy generated, and the coupling system. Finally EUROPAIRS could deliver valuable input to the start of the “Confirmation of Key Technologies” phase, as defined in the SNETP vision report. SNETP recommends alternative uses of nuclear systems and more specifically the heat and power application of (V)HTR systems, as described in the SNETP Vision Report.

2. Technical Scope
EUROPAIRS will exchange information between end-users and nuclear industry to determine the viability of (V)HTR deployment in a concrete industrial environment for mid-term applications. Considering the viability of the combination of heat and electricity supply by a (V)HTR system to an industrial environment, the following key issues can be identified:

- Technical and industrial compatibility between the nuclear system and the industrial processes, as well as economic viability of the coupling of both types of systems
- Safety aspects and licensing
- Sustainability of an extensive deployment of (V)HTR for process heat applications and cogeneration

The project will be coordinated by its Coordination Team. Apart from its normal role of maintaining the consistency of the work in the whole project, organising a harmonious progress of work of all its components and reporting to the Commission, the Coordination Team will have a special role in this project: to develop the strategic alliance between nuclear industry and research on the one hand, and process heat user industries on the other hand, in order to initiate the development of a Demonstrator in the next phase with a strong and consistent team including representatives of the two communities sharing common goals. Additionally, the Coordination Team is expected to establish links with other communities, relevant organisations and other international activities (for example SNETP, end-user technical platforms, HTR-TN, IAEA, NGNP, GIF VHTR, PBMR).

![EUROPAIRS Structure of Work Programme](image-url)

**Fig. 1: EUROPAIRS Structure of Work Programme**
The project will seek the advice of an informal advisory network of end-users in order to review its objectives and the results obtained.
The EUROPAIRS work programme will be implemented in three technical work packages:

**Work package 1: Viability Assessment**

Work package 1 deals with the compatibility between energy supply by a (V)HTR and the industrial processes of the end-users involved.

**Figure 2: Commercial Process Heat Cogeneration Facility – Basic Configuration**

Industrial activities present very different processes which can be very simple or very complex. To evaluate this diversity, EUROPAIRS project regroups 13 different end-users coming from 6 countries. Resulting from data received from the end-users on different processes, three process families are proposed. The objective of this classification is to have a general view of heat utilization in the industry and to be able to use it for other purposes than the one studied in EUROPAIRS.

The three process classes were defined as follow:

1. **The steam class** because heat is transport via steam media. The temperature is between 150°C and 600°C.
2. **The chemical class** because heat is the driver of chemical reactions and is consumed as reaction enthalpy at constant temperature. The temperature is between 600°C and 900°C. Heat is supply by combustion reaction in burner and marginally by electrical heating.
3. **The mineral class** because heat is used to melt solid or to drive reactions between solids. The temperature is in general above 1000°C.

The classes will be examined to provide the requirements for the (V)HTR for connection to an industrial plant. Although the potential of this combination has always been regarded as high, the viability of directly coupling a (V)HTR system to industrial processes has not been experienced yet and is still to be assessed. For this assessment, the determination of the capabilities of the nuclear system and the expression of end-user requirements are crucial. Only with this information available in sufficient detail a successful match can be potentially achieved. A key factor in providing a representative analysis in this respect is the active
contribution of potential end-users. Information exchange will consider the characteristics, technological limits and constraints, requirements, adaptability and flexibility potential of both end-user industrial processes and (V)HTR nuclear system. This information will determine the operating window(s) of combined systems, identify issues to be resolved and define requirements considering infrastructure, coupling system, heat transport, licensing requirements and economics of the coupling.

The (V)HTR appears as leading technology candidate for the next generation of commercial nuclear reactors, following the advanced LWR designs being marketed today. Key attributes of the modular (V)HTR that distinguish it from the LWR are its fuel and its helium coolant. These characteristics, associated with limitations on power level and power density, allow to envisage the development of a modular concept favoring inherent safety characteristics (no need of engineered high reliable and redundant safety system). Such simplification in the safety design leads to investments savings. Furthermore, because of extremely low radiological consequences levels, the emergency planning zone may be limited to the footprint of the plant site (approximately 450 m radius from each module). This is an important advantage of the (V)HTR that allows it to be sited where the market of its energy is located.

One benefit of the much higher operating temperatures in (V)HTRs is the high efficiency power production. Based on modular concept, the nuclear cogeneration plant may consist of multiple reactor modules to answer power demand of industrial sites. This approach results in costs advantage from a modular design and construction approach. It also allows an end-user more flexibility to match load growth.

Work package 2: Safety and licensing

The designers of modular (V)HTR systems base their safety demonstration on inherent physical characteristics of the reactor and on the use of passive safety systems for keeping the reactor in safe conditions and with only negligible release of radioactive products to the environment. As a consequence they consider that the safety approach commonly used for water reactors has to be adapted for (V)HTR systems. The safety approach for (V)HTR systems has already been developed in Europe for the pre-licensing of the German HTR-Modul and reviewed by the German Safety Authorities. The licensing issues related to modular (V)HTR design have been examined again in projects of the 5th Framework Programme (HTR-L) and of the 6th Framework Programme (RAPHAEL) and by industry in recent projects. The assessments made in these frames have been discussed with different Safety Authorities and TSO.

So far the application of (V)HTR in an industrial environment has not been addressed yet and may lead to specific safety aspects that need to be examined. Therefore the main scope to be addressed in this task of EUROPATRS is mainly the coupling between a nuclear plant and an industrial site: what are the safety related impacts in different normal operating conditions and accidental conditions of each of the two coupled systems on the other system? What are the safety requirements to be met by the nuclear system for taking into account the fact that it will be located at an industrial site (chemical risk, fire risk...)? Another issue will be the potential radioactive contamination of the products. These are the kind of questions among others that Safety Authorities and TSOs involved in work package 2 will address.

Work package 3: System Deployment Outlook

Solutions are presently developed in Europe and in other parts of the world, most particularly through the international partnership of the Generation IV International Forum (GIF), for making electricity generation by nuclear energy compatible with the objectives of sustainable development. If on top of this, a significant penetration of HTR on the market of industrial process heat is assumed, can nuclear energy remain sustainable?

For answering this question, based on the present nuclear fleet dedicated to electricity generation and fuel cycle scenarios and on their possible evolution in the next decades (progressive introduction of fast reactors in iso-generation or breeding modes, extension of
closed cycle, minor actinide burning), scenarios of penetration of HTR on the market of industrial process heat will be assessed, considering different types of possible fuel cycles for HTR in symbiosis with the fuel cycles used for the remaining of the reactor fleet.

A preliminary work programme in the form of a detailed roadmap for the development of a large scale demonstration of the coupling of a (V)HTR with an industrial process heat application and for further industrial deployment of such coupled systems will be defined, addressing the following items:

- Based on the work already performed in FP5 and FP6, and in different national programs, based also on HTR-TN roadmap and on SNETP “Strategic Research Agenda”, as well as from the inputs of EUROPAIRS, the complementary work (design and R&D) for developing a (V)HTR system answering end-user needs will be identified.
- The possible needs of evolution of end-user processes for adaptation to the use of a (V)HTR as a heat source will be expressed and the corresponding developments identified.
- A state-of-the-art of the industrial heat transport technologies available in the range of conditions which are relevant for (V)HTR industrial applications will be produced by industrial end-users from the experience obtained from existing technologies used on industrial sites and potential development needs in this area will be assessed.
- A schedule for the demonstration project will be established, based on the assumption that the required resources are available at the right time and that the project will progress at the pace allowed by the necessary duration of all technical steps and by their logical sequencing.
- A rough estimation of the funding needs in the different phases of the Demonstrator project will be provided.

3. International Collaboration
EUROPAIRS will develop its action in an international context marked by the existence of several industrial (V)HTR prototypes, NGNP, PBMR and HTR-PM, to mention the most mature projects only. It is also marked by a new interest of large industrial process heat and steam users for these projects such as DOW, CHEVRON, CONOCOPHILLIPS and POTASHCORP grouped in the “Public Private Partnership” for NGNP development in the US, SASOL in South Africa, etc. EUROPAIRS will endeavour to cooperate with these international projects in order to benefit from a larger feedback on experience of partnership with end-users.

4. Conclusions
The challenging European Project EUROPAIRS will present the investigation of cogeneration of power and heat from a (V)HTR coupled to a conventional user plants. It shall foster the dialogue and the partnership between industrial end-users, nuclear industry, and European research organizations for addressing this new challenge. The EUROPAIRS work programme will provide results on:

- Technical and industrial compatibility between the nuclear system and the industrial processes, as well as economic viability of the coupling of both types of systems
- Safety aspects and licensing
- Sustainability of an extensive deployment of (V)HTR for process heat applications and cogeneration

International collaboration will be established with projects such as NGNP, PBMR and HTR-PM and with European technology platforms.

The results of EUROPAIRS will be published in newsletters, on a dedicated website and in a big workshop at the end of the project assembling experts and public audience.
THE GUINEVERE PROJECT AT THE VENUS-F FACILITY

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ABSTRACT

Within the framework of the ECATS (Experimental activities on the Coupling of an Accelerator, a spallation Target and a Sub-critical blanket) research domain of the FP6 IP-EUROTRANS program, the GUINEVERE (Generation of Uninterrupted Intense NEutrons at the lead VEnus REactor) project was launched in 2006 in order to check in the experiments open questions for the techniques used in the MUSE programme (CEA Cadarache, France, 2000-2004), related to the online reactivity monitoring, sub-criticality determination and operational procedure of an Accelerator Driven System. For this purpose, the VENUS light water critical reactor at the SCK-CEN site of Mol (Belgium) was modified into a subcritical fast core (VENUS-F) and the GENEPI accelerator, designed for the MUSE experiment was up-graded to the new GENEPI-3C accelerator. The VENUS-F coupled with the GENEPI-3C and a TiT target will provide a unique facility in Europe for fast sub-critical and critical reactor physics investigations. This paper describes the present status of the facility.

1. Introduction

The GUINEVERE project (Generator of Uninterrupted Intense NEutrons at the lead VEnus REactor) is part of the EUROTRANS Integrated Project (6th EURATOM FP) which gathers feasibility and design studies of an ADS prototype as well as its possible extend to an industrial transmutation installation. GUINEVERE aims at providing a zero power experimental facility to investigate reactivity on-line monitoring and absolute measurement which are major issues for ADS safety.

The new facility hosting the GUINEVERE experiments is located at the VENUS critical facility at the SCK-CEN site of Mol (Belgium). The VENUS reactor was a zero power thermal neutrons water moderated critical facility up to 2007. From then on the critical facility and its site have been modified in order to host a fast lead core, further on referred to as VENUS-F, and the GENEPI-3C accelerator. The GENEPI-3C machine cumulates specifications of the first GENEPI accelerator, i.e. pulsed mode operation with new continuous mode specifications.
Both reactor and accelerator are now in full commissioning phase. The start-up of the reactor in critical mode is foreseen in the summer 2010 and the first coupling experiment in the following autumn. After GUINEVERE project realization the VENUS-F system will provide a unique facility in Europe for fast sub-critical and critical reactor physics investigations.

2. The VENUS-F critical facility

The VENUS-F core consist of Fuel Assemblies (FA) arranged in a cylindrical geometry (~80 cm in diameter, 60 cm in height), and composed of a 5x5 pattern mixture of fuel and solid lead rodlets (Fig. 41) to figure out the presence of a fast system coolant. Lead plates are added around the pattern to decrease the fuel/coolant ratio (and hence increase the core size) and keep a symmetry that avoids core loading error issues. The outer section of a FA is 80 mm and the pattern chosen is shown in. The fuel is 30% $^{235}$U enriched metallic uranium (provided by CEA). The upper part of the FA contain

![Fig. 1 Cross section of one fuel assembly of the VENUS-F core.](image)

The core is surrounded by two 40 cm axial and radial lead reflectors. According to the preliminary calculation 88 assemblies will be necessary to obtain a critical configuration (SCO). By removing the four central assemblies a subcritical core (SC1 with $k_{eff}$~0.97) can be obtained and a stainless steel shaft for the insertion of the accelerator thimble can be inserted. A small lead buffer is foreseen to fill the gap between the target tube shaft and the 160x160 mm$^2$ central hole. A radial view of the reactor (inner part of the vessel) at mid-plane, in the SC1 configuration ($k_{eff}$~0.97) is shown in Fig.2 b).

![Fig. 2 Cross section (mid-plane) of the SC0 (a) critical core configuration and sub-critical core configuration SC1 (b). In purple the position of the safety rods, in red the control rods.](image)

Every component of the new FA structure are manufactured and preassembled.

3. The GENEPI-3C accelerator
The GENEPI-3C (GEnérateur de NEutrons Pulsé Intense-3Continu) accelerator is the third of a series designed for neutronic experiments [1]. It is designed and built by a CNRS/IN2P3 collaboration and after been tested at LPSC Grenoble for beam characterization measurements has been transferd and reassembled in VENUS-F reactor building in October 2009.

The GENEPI machines are 250 kV deuteron accelerators ended by copper targets with titanium-tritium (TiT) or titanium-deuterium (TiD) deposits, providing 14 MeV or 2.5 MeV neutrons via T(d,n)4He or D(d,n)3He reactions (see Fig.3).

The new GENEPI-3C machine cumulates specifications of the first GENEPI accelerator, designed for the MUSE experimental programme [2] at MASURCA reactor (CEA Cadarache, France, 2000-2004), i.e. pulsed mode operation with very sharp and intense beam pulses (1 μs, 50 mA peak current), with new continuous mode specifications summed up in Tab. 1. In this new DC mode, it will be also possible to operate beam interruptions (so-called “beam trip”) with a programmable duration and a low repetition rate for the needs of the foreseen experiments.

<table>
<thead>
<tr>
<th>Mean current</th>
<th>160 μA to 1 mA</th>
</tr>
</thead>
<tbody>
<tr>
<td>High Voltage</td>
<td>250 kV</td>
</tr>
<tr>
<td>Beam trip rate</td>
<td>0.1 to 100 Hz</td>
</tr>
<tr>
<td>Beam trip duration</td>
<td>~ 20 μs to 10 ms</td>
</tr>
<tr>
<td>Transition (ON/OFF)</td>
<td>~ 1 μs</td>
</tr>
<tr>
<td>Beam spot size</td>
<td>20 to 40 mm in diameter</td>
</tr>
<tr>
<td>Maximum neutron production</td>
<td>~5×1010 n/s</td>
</tr>
<tr>
<td>Pulse stability</td>
<td>~1%</td>
</tr>
</tbody>
</table>

Tab. 1 Specifications of the GENEPI-3C accelerator installed at the VENUS-F site

The accelerator itself consists of an ion source sited in the 250 kV high voltage head, followed by a horizontal beam transport line section of 3 m at the exit of the accelerator tube.
Beam transport is ensured with electrostatic quadrupoles. A first group of 4 quadrupoles transports and focuses the beam at the 90° dipole magnet, which deflects the beam downwards. A quadrupole doublet is located at the exit of the magnet chamber and two quadrupole triplets located on the vertical beam line above the reactor core focus the beam onto the target. The target is located at the end of a short optic free thimble inserted into the reactor core.

4. **The coupling**

For the coupling of the accelerator with the reactor, important civil engineering work has been necessary. A new room (Fig. 4) above the reactor bunker was constructed for the hosting of the accelerator and of a new ventilation system, not present in the past but required now because of the use of a Tritium target for the deuterium beam. Those modifications have been started at September 1st, 2008 and completed at the end of 2009.

![Fig. 4 Cross section of the VENUS site and picture of the insertion of the accelerator vertical beam line in the VENUS_F core. The upper part of the building has been added to the existing one for the GUINEVERE project. Here are located the accelerator room and the extraction of the HVAC system.](image)

5. **Conclusion**

In order to investigate reactivity on-line monitoring and absolute measurement which are major issues for ADS safety a low power ADS systems has been constructed by coupling the fast subcritical core VENUS-F, the GENEPI-3C accelerator and the TiT target.
6. References


SUPERCritical WATER-COOLED NuCLEAR REACTORS:  
THERMODYNAmIC-CYCLES PLANT LAYOUTS AND 
THERMAL ASPECTS OF PRESSURE-CHANNEL DESIGN

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ABSTRACT

Research activities are currently conducted worldwide to develop Generation IV nuclear-reactor concepts with the objective of improving thermal efficiency and increasing economic competitiveness of Generation IV Nuclear Power Plants (NPPs) compared to modern thermal power plants. The Super-Critical Water-cooled Reactor (SCWR) concept is one of the six Generation IV options chosen for further investigation and development in several countries including Canada.

Water-cooled reactors operating at subcritical pressures (7–16 MPa) have provided a significant amount of electricity production for the past 50 years. However, the thermal efficiency of the current NPPs is not very high (30-35%). As such, more competitive designs, with higher thermal efficiencies, which will be close to that of modern coal-fired thermal power plants (45–50%), need to be developed and implemented.

Previous studies have shown that direct cycles with no-reheat and single-reheat configurations are the best choices for SCWR concepts. This paper presents three SCW NPP concepts based on direct single-reheat and no-reheat regenerative cycles, and indirect single-reheat regenerative cycle. In general, there are many technical challenges associated with the single-reheat and no-reheat SCW NPP configurations.

The direct single-reheat cycle requires nuclear steam-reheat, thus increasing the complexity of the pressure-tube reactor-core design. The nuclear steam reheat seems to be possible only in a pressure-tube reactor. Conversely, the major technical challenge associated with the no-reheat cycle is the high moisture content in the low-pressure-turbine exhaust. The direct no-reheat cycle can be implemented in NPP with pressure-vessel or pressure-tube reactors. The gross thermal efficiency of the direct cycles was determined to be about 50%.

The indirect single-reheat cycles utilize heat exchangers (steam generators) to transfer heat from the reactor coolant to the secondary-loop working fluid. The indirect cycle has the safety benefit of containing the potential radioactive particles inside the primary coolant. Also, these cycles are applicable to any Generation IV reactors including pressure-vessel and pressure-tube SCWRs. However, the heat-transfer process through heat exchangers reduces the maximum temperature of the secondary loop coolant, thus lowering the thermal efficiency of the cycle.

The SCWR-core concept investigated in this paper is based on a generic pressure-tube reactor cooled with supercritical water. The considered reactor concept is based on a horizontal pressure-tube configuration with the following operating parameters: electrical power of 1200 MW, pressure of 25 MPa, reactor inlet temperature of 350°C, and reactor outlet temperature of 625°C.
1. Introduction

Thermal power plants lead the power industry in terms of high thermal efficiencies. Combined-cycle gas-fired thermal power plants have reached gross thermal efficiencies of about 50–55% and SuperCritical (SC) coal-fired thermal power plants – 45–50% range. Therefore, Nuclear Power Plants (NPPs) are not very competitive with thermal power plants in terms of thermal efficiencies. For water-cooled NPPs gross thermal efficiencies are within a range of 30–36% and for gas-cooled (carbon-dioxide-cooled) NPPs in UK – within a range of 38–42% (however, all these gas-cooled reactors will be shut down in the near future). Therefore, the major task is to develop next-generation reactors and corresponding to that NPPs with significantly higher thermal efficiencies within the range of that for coal-fired thermal power plants.

Currently, there are six Generation IV nuclear-reactor concepts under development worldwide [1]:

1. Gas-cooled Fast Reactors (GFRs) (helium, 7 MPa, 485–850°C, primary thermodynamic cycle – a direct cycle based on the Brayton cycle (gas-turbine cycle), back-up – an indirect cycle based on the Rankine steam cycle through heat exchangers);
2. Very High-Temperature gas-cooled Reactors (VHTRs) (helium, 9 MPa, 500–1000°C, primary thermodynamic cycle – a direct cycle based on the Brayton cycle (gas-turbine cycle), back-up – an indirect cycle based on the Rankine steam cycle through heat exchangers);
3. Sodium-cooled Fast Reactors (SFRs) (520–550°C, primary thermodynamic cycle – an indirect cycle based on the Rankine steam cycle through heat exchangers, back-up – an indirect cycle based on the Brayton cycle (SC-carbon-dioxide gas-turbine cycle));
4. Lead-cooled Fast Reactors (LFRs) (up to 550–800°C, primary thermodynamic cycle – an indirect cycle based on the Brayton cycle (SC-carbon-dioxide gas-turbine cycle));
5. Molten Salt-cooled Reactors (MSRs) (sodium-fluoride salt with dissolved uranium fuel, up to 700–800°C, primary thermodynamic cycle – an indirect cycle based on the Rankine steam cycle through heat exchangers); and
6. SuperCritical Water-cooled Reactors (SCWRs) (25 MPa, up to 625°C, primary thermodynamic cycle – a direct cycle based on the SC Rankine steam cycle, back-up – indirect cycle based on the SC Rankine steam cycle through heat exchangers).

Currently, the only one known operating Generation IV power-nuclear reactor is an SFR or Liquid Metal Fast Breeder Reactor (LMFBR) – BN-600 in Russia (Beloyarsk NPP, Zarechnyy, Swerdlovsk). The rest of Generation IV reactors are in a stage of experimental reactors or just concepts. Common features for all these reactors are quite high outlet temperatures, which are beyond the critical temperature of water (374°C), and due to that a possibility of using SCW technology with high thermal efficiencies in the nuclear power industry is a very beneficial way.

In terms of SCWRs, developing reactors with SCW parameters is considered to be a conventional way and the ultimate pass for water-cooled reactors (see Fig. 2). The main objectives for developing and utilizing SCWRs are: 1) Increase thermal efficiency of current NPPs from 30–35% to approximately 45–50%, and 2) Decrease capital and operational costs and, in doing so, decrease electrical-energy costs.
Fig. 1. Single-reheat-cycle 600-MW$_{el}$ Tom'-Usinsk thermal power plant (Russia) thermal layout [2]: Cyl – Cylinder; H – Heat exchanger (feedwater heater); CP – Circulation Pump; TDr – Turbine Drive; Cond P – Condensate Pump; GCHP – Gas Cooler of High Pressure; and GCLP – Gas Cooler of Low Pressure.
SCW NPPs will have much higher operating parameters compared to current NPPs (i.e., pressures of about 25 MPa and outlet temperatures up to 625°C) (Fig. 2). Additionally, direct-cycle SCW NPPs will have a simplified flow circuit in which steam generators, steam dryers, steam separators, etc., will be eliminated. Furthermore, SCWRs operating at higher temperatures can facilitate an economical production of hydrogen through thermochemical cycles or direct high-temperature electrolysis [3].

The SCWR concepts [1, 4–6] follow two main types: (a) A large reactor pressure vessel (PV) (see Fig. 3) analogous to conventional Light Water Reactors (LWRs); or (b) Distributed pressure tubes (PTs) or pressure channels (see Figs. 4–7) analogous to conventional Heavy Water Reactors (HWRs).

Within these two main classes, PT reactors are more flexible to flow, flux and density changes than PV reactors. This makes it possible to use the experimentally confirmed, better solutions developed for these reactors. The main one is channel-specific flow-rate adjustments or regulations. A design whose basic element is a pressure tube or channel (see Figs. 6 and 7), which carries a high pressure, has an inherent advantage of greater safety than large vessel structures at supercritical pressures.

To decrease significantly development costs of SCW NPPs and to increase their reliability, it should be determined whether SCW NPPs can be designed with a steam-cycle arrangement similar to that of SCW coal-fired thermal power plants, which operate successfully and efficiently for more than 50 years.

2. General Considerations Regarding SCW NPP Cycle

2.1 Supercritical “steam turbines

SC-“steam” turbines of medium and large capacities (450 – 1200 MWₑₑ) [5] have been used very successfully at many coal-fired thermal power plants worldwide for more than fifty years. Their gross steam-cycle thermal efficiencies have reached about 45–50%, which is equivalent to a net-plant efficiency of approximately 40 – 43% on a Higher-Heating Value (HHV) basis.

An analysis of SC turbines [5] showed that:

- The vast majority of the modern and upcoming SC turbines are single-reheat-cycle turbines.
- Major “steam” inlet parameters of these turbines are: The main or primary SC “steam” – pressure of 24–25 MPa and temperature of 540–600°C; and the reheat or secondary superheated steam – pressure of 3–5 MPa and temperature of 540–620°C.
- Usually, the main “steam” and reheat-steam temperatures are the same or very close (for example, 566/566°C; 579/579°C; 600/600°C; 566/593°C; 600/620°C). And
- Only very few double-reheat-cycle turbines were manufactured so far. The market demand for double-reheat turbines disappeared due to economic reasons after the first few units were built.
2.2 Direct, indirect and dual-cycle options

Since "steam" parameters of an SCW NPP are much higher than those of current NPPs, several conceptual designs have been investigated to determine the optimum configuration. As such, direct, indirect and dual cycles have been considered [3, 5–10]. In the direct cycle (see Figs. 8 and 9 and Tables 1 and 2) SC "steam" from a nuclear reactor is fed directly to
an SC turbine. This concept eliminates the need for complex and expensive equipment such as steam generators. From a thermodynamic perspective, this allows for high steam pressures and temperatures, and results in the highest cycle efficiency (about 52%) for the given parameters.

Fig. 6. 3-D view of SCWR ceramic-insulated fuel-channel (based on AECL design [1, 4]).

Indirect and dual cycles utilize heat exchangers (steam generators) to transfer heat from the reactor coolant to a turbine. The indirect cycle (see Fig. 10) has a safety benefit of containing the potential radioactive particles inside the primary coolant. However, the heat-transfer process through heat exchangers reduces the maximum temperature of the secondary-loop working fluid, thus lowering the efficiency of the cycle. Also, indirect cycles are applicable to any Generation IV reactors including PV and PT SCWRs.

Since increasing the thermal efficiency is one of the main objectives in development of SCW NPPs, the direct cycle is considered to be the primary choice for SCWRs.

A preliminary investigation of SCW NPP reheat options [10] revealed the following:

- The no-reheat cycle offers a simplified SCW NPP layout, contributing to lower capital costs. However, the efficiency of this cycle is lower compared to the direct single-reheat cycle.
- The single-reheat cycle has the advantage of higher thermal efficiency (compared to that of the no-reheat cycle) and reduced development costs due to a wide variety of single-reheat SC turbines manufactured by companies worldwide. The major disadvantage is an increased design complexity associated with the introduction of Steam-ReHeat (SRH) channels to the PT-reactor core. And
- While the double-reheat cycle has the highest thermal efficiency, it was deemed that the complicated nuclear-steam-reheat configuration would significantly increase the design and construction costs of such a reactor.
As such, configurations based on the single-reheat and no-reheat cycles should be considered as a basis for future developments.

---

**Fig. 8. Single-reheat direct cycle A for 1200-MW$_{el}$ SCW NPP (PT SCWR) [7].**

**Fig. 9. No-reheat direct cycle B for 1200-MW$_{el}$ SCW NPP (PT or PV SCWR) [7].**

**Table 1. Selected parameters of proposed SCW NPP Cycles A and B [11].**

<table>
<thead>
<tr>
<th>Parameters</th>
<th>Unit</th>
<th>Description / Value</th>
<th>Description / Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cycle type</td>
<td>–</td>
<td>Single-Reheat (A)</td>
<td>No-Reheat (B)</td>
</tr>
<tr>
<td>Reactor type</td>
<td>–</td>
<td>Pressure Tube</td>
<td></td>
</tr>
<tr>
<td>Reactor spectrum</td>
<td>–</td>
<td>Thermal</td>
<td></td>
</tr>
<tr>
<td>Parameters</td>
<td>Unit</td>
<td>Description / Value</td>
<td>Description / Value</td>
</tr>
<tr>
<td>----------------------------------------</td>
<td>--------</td>
<td>--------------------------------------</td>
<td>--------------------</td>
</tr>
<tr>
<td>Fuel</td>
<td>–</td>
<td>UO$_2$ (ThO$_2$)</td>
<td></td>
</tr>
<tr>
<td>Cladding material</td>
<td>–</td>
<td>Inconel or Stainless steel</td>
<td></td>
</tr>
<tr>
<td>Reactor coolant</td>
<td>–</td>
<td>H$_2$O</td>
<td></td>
</tr>
<tr>
<td>Moderator</td>
<td>–</td>
<td>D$_2$O</td>
<td></td>
</tr>
<tr>
<td>Power Thermal</td>
<td>MW$_{th}$</td>
<td>2300</td>
<td>2340</td>
</tr>
<tr>
<td>Power Electrical</td>
<td>MW$_{el}$</td>
<td>1200</td>
<td>1200</td>
</tr>
<tr>
<td>Thermal Efficiency</td>
<td>%</td>
<td>52</td>
<td>51</td>
</tr>
<tr>
<td>Pressure of SCW at inlet</td>
<td>MPa</td>
<td>25.8</td>
<td>25.8</td>
</tr>
<tr>
<td>Pressure of SCW at outlet (estimated)</td>
<td>MPa</td>
<td>25</td>
<td></td>
</tr>
<tr>
<td>$T_{in}$ coolant (SCW)</td>
<td>°C</td>
<td>350</td>
<td>350</td>
</tr>
<tr>
<td>$T_{out}$ coolant (SCW)</td>
<td>°C</td>
<td>625</td>
<td>625</td>
</tr>
<tr>
<td>Pressure of SHS at inlet</td>
<td>MPa</td>
<td>6.1</td>
<td>–</td>
</tr>
<tr>
<td>Pressure of SHS at outlet (estimated)</td>
<td>MPa</td>
<td>5.7</td>
<td>–</td>
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<tr>
<td>$T_{in}$ Coolant (SHS)</td>
<td>°C</td>
<td>400</td>
<td>–</td>
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<td>$T_{out}$ coolant (SHS)</td>
<td>°C</td>
<td>625</td>
<td>–</td>
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<tr>
<td>Power thermal SCW channels</td>
<td>MW$_{th}$</td>
<td>1870</td>
<td>2340</td>
</tr>
<tr>
<td>Power thermal SRH channels</td>
<td>MW$_{th}$</td>
<td>430</td>
<td>–</td>
</tr>
<tr>
<td>Power thermal / SCW channel</td>
<td>MW$_{th}$</td>
<td>8.5</td>
<td>8.5</td>
</tr>
<tr>
<td>Power thermal / SRH channel</td>
<td>MW$_{th}$</td>
<td>5.5</td>
<td>–</td>
</tr>
<tr>
<td># of fuel channels (total)</td>
<td>–</td>
<td>300</td>
<td>270</td>
</tr>
<tr>
<td># of SCW channels</td>
<td>–</td>
<td>220</td>
<td>270</td>
</tr>
<tr>
<td># of SRH channels</td>
<td>–</td>
<td>80</td>
<td>–</td>
</tr>
<tr>
<td>Total flow rate of SCW</td>
<td>kg/s</td>
<td>960</td>
<td>1190</td>
</tr>
<tr>
<td>Total flow rate of SHS</td>
<td>kg/s</td>
<td>780</td>
<td>–</td>
</tr>
<tr>
<td>Flow rate / SCW channel</td>
<td>kg/s</td>
<td>4.37</td>
<td>4.37</td>
</tr>
<tr>
<td>Flow rate / SRH channel</td>
<td>kg/s</td>
<td>10</td>
<td>–</td>
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Table 2. Selected parameters of proposed SCWR fuel channels [11].

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<tr>
<td>$T_{max}$ cladding (design value)</td>
<td>°C</td>
<td>850</td>
</tr>
<tr>
<td>$T_{max}$ fuel centerline (industry accepted limit)</td>
<td>°C</td>
<td>1850</td>
</tr>
<tr>
<td>Heated fuel-channel length</td>
<td>m</td>
<td>5.772</td>
</tr>
<tr>
<td># of bundles / fuel channel</td>
<td>–</td>
<td>12</td>
</tr>
<tr>
<td># of fuel rods per bundle</td>
<td>–</td>
<td>43</td>
</tr>
<tr>
<td>Bundle type [12]</td>
<td>–</td>
<td>CANFLEX</td>
</tr>
<tr>
<td># of heated fuel rods</td>
<td>–</td>
<td>43</td>
</tr>
<tr>
<td># of unheated* fuel rods</td>
<td>–</td>
<td>–</td>
</tr>
<tr>
<td>Diameter of heated fuel rods (# of rods)</td>
<td>mm</td>
<td>11.5 (35) &amp; 13.5 (8)</td>
</tr>
<tr>
<td>Diameter of unheated fuel rod</td>
<td>mm</td>
<td>–</td>
</tr>
<tr>
<td>$D_{hy}$ of fuel channel</td>
<td>mm</td>
<td>7.52</td>
</tr>
<tr>
<td>$D_{oh}$ of fuel channel</td>
<td>mm</td>
<td>9.04</td>
</tr>
<tr>
<td>Heated area of fuel channel</td>
<td>m$^2$</td>
<td>9.26</td>
</tr>
<tr>
<td>Flow area of fuel channel</td>
<td>m$^2$</td>
<td>3625</td>
</tr>
<tr>
<td>Pressure tube inner diameter</td>
<td>mm</td>
<td>103.45</td>
</tr>
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Average parameters of fuel channels in single-reheat (A) and no-reheat (B) options

<table>
<thead>
<tr>
<th>Parameters</th>
<th>Description / Value</th>
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<tbody>
<tr>
<td>Heat flux in SCW channel (A&amp;B cycles)</td>
<td>kW/m$^2$</td>
</tr>
<tr>
<td>Heat flux in SRH channel (A cycle)</td>
<td>kW/m$^2$</td>
</tr>
<tr>
<td>Mass flux in SCW channel (A&amp;B cycles)</td>
<td>kg/m$^2$/s</td>
</tr>
<tr>
<td>Mass flux in SRH channel (A cycle)</td>
<td>kg/m$^2$/s</td>
</tr>
</tbody>
</table>
Conclusions

The following conclusions can be made:
1. The vast majority of modern supercritical turbines are single-reheat-cycle turbines. Just a few double-reheat-cycle supercritical turbines have been manufactured and put into operation. However, despite their increased efficiency double-reheat-cycle turbines have not been considered economical.
2. Major inlet parameters of the current and upcoming single-reheat-cycle supercritical turbines are: the main or primary supercritical "steam" – pressure of 24–25 MPa and temperature of 540–600°C; and the reheat or secondary superheated steam – pressure of 3–5 MPa and temperature of 540 – 620°C.
3. In order to maximize the thermal-cycle efficiency of SCW NPPs it would be beneficial to include nuclear steam reheat through direct or indirect cycles. Advantages of a single-reheat cycle in application to SCW NPPs are:
   a. High gross thermal efficiency (up to 52% in case of direct cycle), which is the current level for supercritical coal-fired thermal power plants and close to the maximum thermal efficiency achieved in the power industry at combined-cycle thermal power plants (up to 55%).
   b. High reliability through proven state-of-the-art supercritical turbine technology; and
   c. Reduced development costs accounting on a wide variety of supercritical turbines manufactured by companies worldwide.
4. The major disadvantage of the direct single-reheat-cycle implementation in pressure-tube SCW NPPs is a requirement for significant changes to the reactor-core design due to addition of nuclear steam-reheat channels at subcritical pressures.
5. Based on the abovementioned analysis, the direct and indirect single-reheat cycles with heat regeneration and the corresponding arrangement appear to be the most advantageous as a basis for SCW NPPs with the co-generation of hydrogen. Indirect supercritical cycles can be implemented in all 6 concepts of Generation IV reactors.

Fig. 10. Indirect single-reheat cycle for 1200-MWe SCW NPP (PT or PV SCWR) (same cycle can be used for all Generation IV nuclear reactors) [13].
ACKNOWLEDGEMENTS

Financial supports from the NSERC Discovery Grant, NSERC/NRCan/AECL Generation IV Energy Technologies Program and Ontario Research Excellence Fund are gratefully acknowledged.

NOMENCLATURE

<table>
<thead>
<tr>
<th>Symbol</th>
<th>Description</th>
<th>Subscripts</th>
<th>Abbreviations</th>
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</thead>
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<tr>
<td>D</td>
<td>diameter, m</td>
<td></td>
<td></td>
</tr>
<tr>
<td>P</td>
<td>pressure, Pa</td>
<td></td>
<td></td>
</tr>
<tr>
<td>T</td>
<td>temperature, ºC</td>
<td></td>
<td></td>
</tr>
<tr>
<td>el</td>
<td>electrical</td>
<td></td>
<td></td>
</tr>
<tr>
<td>h</td>
<td>heated</td>
<td></td>
<td></td>
</tr>
<tr>
<td>hy</td>
<td>hydraulic-equivalent</td>
<td></td>
<td></td>
</tr>
<tr>
<td>in</td>
<td>inlet</td>
<td></td>
<td></td>
</tr>
<tr>
<td>pc</td>
<td>pseudocritical</td>
<td></td>
<td></td>
</tr>
<tr>
<td>th</td>
<td>thermal</td>
<td></td>
<td></td>
</tr>
<tr>
<td>HTR</td>
<td>Heater</td>
<td></td>
<td></td>
</tr>
<tr>
<td>HWR</td>
<td>Heavy Water Reactor</td>
<td></td>
<td></td>
</tr>
<tr>
<td>IP</td>
<td>Intermediary Pressure</td>
<td></td>
<td></td>
</tr>
<tr>
<td>LP</td>
<td>Low Pressure</td>
<td></td>
<td></td>
</tr>
<tr>
<td>LWR</td>
<td>Light-Water Reactor</td>
<td></td>
<td></td>
</tr>
<tr>
<td>NPP</td>
<td>Nuclear Power Plant</td>
<td></td>
<td></td>
</tr>
<tr>
<td>PC</td>
<td>PseudoCritical</td>
<td></td>
<td></td>
</tr>
<tr>
<td>PT</td>
<td>Pressure Tube (reactor)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>PV</td>
<td>Pressure Vessel (reactor)</td>
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<td></td>
</tr>
<tr>
<td>PWR</td>
<td>Pressurized Water Reactor</td>
<td></td>
<td></td>
</tr>
<tr>
<td>RFP</td>
<td>Reactor Feedwater Pump</td>
<td></td>
<td></td>
</tr>
<tr>
<td>SC</td>
<td>SuperCritical</td>
<td></td>
<td></td>
</tr>
<tr>
<td>SCW</td>
<td>SuperCritical Water</td>
<td></td>
<td></td>
</tr>
<tr>
<td>SCWR</td>
<td>SuperCritical Water Reactor</td>
<td></td>
<td></td>
</tr>
<tr>
<td>SH</td>
<td>Sheath</td>
<td></td>
<td></td>
</tr>
<tr>
<td>SHS</td>
<td>SuperHeated Steam</td>
<td></td>
<td></td>
</tr>
<tr>
<td>SRH</td>
<td>Steam ReHeat</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Abbreviations:

- BWR: Boiling Water Reactor
- CANDU: CANada Deuterium Uranium
- CANFLEX: CANDU FLEXible (fuelling)
- CEP: Condensate Extraction Pump
- HHV: Higher Heating Value
- HP: High Pressure
- HTR: Heater
- HWR: Heavy Water Reactor
- IP: Intermediary Pressure
- LP: Low Pressure
- LWR: Light-Water Reactor
- NPP: Nuclear Power Plant
- PC: PseudoCritical
- PT: Pressure Tube (reactor)
- PV: Pressure Vessel (reactor)
- PWR: Pressurized Water Reactor
- RFP: Reactor Feedwater Pump
- SC: SuperCritical
- SCW: SuperCritical Water
- SCWR: SuperCritical Water Reactor
- SH: Sheath
- SHS: SuperHeated Steam
- SRH: Steam ReHeat

References


THE GIF PROLIFERATION RESISTANCE AND PHYSICAL PROTECTION (PR&PP) EVALUATION METHODOLOGY: OVERVIEW AND PERSPECTIVES

*European Commission-Institute for the Protection and Security of the Citizen Ispra, 21020 – Italy

ON BEHALF OF PR&PP WG MEMBERS & CONTRIBUTORS

Generation IV International Forum

ABSTRACT

The Generation IV International Forum (GIF) was established in 2000 to perform research and development for a new generation of nuclear energy systems (NES). The so-called Generation IV. Generation IV NES are expected to excel in sustainability, safety, economics, proliferation resistance and physical protection robustness [1]. In 2002 GIF created the Proliferation Resistance and Physical Protection Working Group (PR&PP WG) with the primary task to develop a methodology for evaluation of PR&PP aspects of GEN IV nuclear energy systems. The current version, of the methodology, Revision 5, was published in 2007 for open distribution on the GIF web site together with an accompanying volume of appendixes [2, 3]. In order to develop the methodology a series of development and demonstration case studies were carried out, based on a notional sodium cooled fast neutron nuclear reactor system, named the Example Sodium Fast Reactor (ESFR). The 2007-2009 ESFR case study analyzed the response of the entire ESFR nuclear energy system to different proliferation and theft strategies [4]. Three proliferation strategies were considered: Concealed diversion [5], Concealed misuse [6], and Breakout [7]. One objective of the case study was to investigate whether the PR&PP methodology could capture differences in proliferation resistance among varied design configurations. In 2008 the PR&PP WG began a closer interaction with the GEN IV System Steering Committees (SSCs). A series of workshops were carried out and white papers are currently being jointly prepared by the PR&PP WG and by the GEN IV SSCs for all six Gen IV concepts, emphasizing the PR&PP aspects and identifying areas for R&D work. This paper will briefly recall the PR&PP evaluation methodology, and then summarize the main findings of the 2007-2009 case study. Finally, it will present some of the work started in collaboration with the GEN IV SSCs and perspectives for the future activities of the group.

1. GIF Goals and PR&PP evaluation methodology

The GIF technology roadmap highlighted Proliferation Resistance and Physical Protection as one of the four goal areas along with Sustainability, Safety and Reliability, and Economics and indicated that: “Generation IV nuclear energy systems will increase the assurance that they are a very unattractive and the least desirable route for diversion or theft of weapons usable materials, and provide increased physical protection against acts of terrorism.” [1]

GIF tasked the Proliferation Resistance and Physical protection Working Group (PR&PP WG) to develop an evaluation methodology. The PR&PP WG has defined Proliferation Resistance and Physical Protection Robustness as follows:
“Proliferation resistance is that characteristic of an NES that impedes the diversion or undeclared production of nuclear material or misuse of technology by the Host State seeking to acquire nuclear weapons or other nuclear explosive devices. Physical protection (robustness) is that characteristic of an NES that impedes the theft of materials suitable for nuclear explosives or radiation dispersal devices (RDDs) and the sabotage of facilities and transportation by sub-national entities and other non-Host State adversaries.” [2].

According to the methodology developed by the PR&PP WG, Proliferation Resistance and Physical Protection share a common evaluation paradigm similar to safety evaluations [2] (Figure 1). A system is challenged by the threats and responds according to its proliferation resistance intrinsic and extrinsic features. Estimation of the outcomes allows the system assessment. In figure 2 a more detailed Evaluation Framework is presented, highlighting its pathway oriented nature.

![Fig 1. PR&PP Evaluation Paradigm [2].](image1)

![Fig 2. PR&PP Evaluation Framework [2].](image2)

The methodology is organized as a progressive approach applying alternative methods at different levels of thoroughness as more design information becomes available and research improves the depth of technical knowledge. The PR&PP WG has also worked on the safeguardability concepts to identity qualitative features and characteristics which facilitate the deployment of the safeguards system in the plant [3].

2. The 2007-2009 ESFR case study: hints and main lessons learned

A notional sodium-cooled fast-neutron nuclear reactor system, named the Example Sodium Fast Reactor (ESFR), was used as a case study for the development and demonstration of the GIF Proliferation Resistance and Physical Protection (PR&PP) evaluation methodology. The 2007-2009 case study demonstrated that the PR&PP methodology can provide useful feedback to designers at various levels of detail including pre-conceptual design and considered a baseline design as well as design variations. The Study analyzed the response of the entire ESFR nuclear energy system to different proliferation and theft strategies [4-7]:

1. Concealed diversion of material;
2. Concealed misuse of the facility;
3. Breakout and overt diversion or misuse;
4. Theft of weapons usable material or sabotage of facility system elements.

The ESFR is a hypothetical nuclear energy system consisting of four sodium-cooled fast reactors of medium size, co-located with an on-site dry fuel storage facility and a Fuel Cycle Facility with pyrochemical processing capability of the spent fuel and re-fabrication of new ESFR fuel elements. Fuel is metallic (U and TRansUranic, TRU). The baseline design is an actinide burner of thermal power of 800 MWth. LWR spent fuel elements are processed on-site to provide needed transuranic feeds. The system, the related safeguards details, and all
the design variations are reported in detail in a project report [4] that has been published by GIF in 2010. Hereafter only the main features of the baseline design are described.

The baseline design is an actinide burner, with a transuranic (TRU) conversion ratio (CR) of 0.64. The core is divided into two enrichment zones and contains 102 Driver Assemblies, each one containing 271 pins, for a total of 88 kg of heavy metal (U+TRU). 42 low-enriched fuel assemblies (around 22 % in TRU) are loaded in the inner reactor core; the remaining 60 highly-enriched assemblies (around 27 % of TRU) are located in the outer reactor core. The total core loading is 8970 kg of HM. Due to the burner core configuration, the system has a TRU consumption rate of 80 kg/year, corresponding to a need of an external feed of material from about 56 Light Water Reactor (LWR) spent fuel elements per year. The LWR spent fuel elements are processed on site together with the ESFR spent fuel elements for the re-fabrication of the ESFR fresh fuel elements. The system is considered to be under international safeguards, and a notional safeguards approach was developed by JRC for the benefit of the project. All design variations (DVs) are associated to reactors with thermal power of 1000 MWth and correspond to different core configurations and overall objectives (Burner: CR 0.22; Deep Burner: CR 0.22; Equilibrium core: CR 1.00; Breeder: CR 1.12) [4].

Each threat was assigned to a project sub-group that was responsible for analysing the response of the baseline design and, to different extents, of its DVs in order to show how PR&PP evaluations can capture differences in the design. For PR, where the actor is assumed to be a state, common characteristics have been assigned to the actor’s description in terms of capability (Industrialised country) and objective (acquisition of one Significant Quantity of Pu), in order to have a fully stylised description of the actor. The same was done for the description of the sub-national actor relevant for PP for the theft strategy.

As reported in the final report of the case study [4]:

“The case study exercise illustrated a practical approach for applying the PR&PP methodology in a traceable way, leading to accountable and dependable results for evaluating PR pathways at a qualitative level and PP pathways at qualitative and quantitative levels. Basic lessons learned from the case study included the following:

- Each PR&PP evaluation should start with a qualitative analysis allowing scoping of the assumed threats and identification of targets, system elements, etc.
- Detailed guidance for qualitative analyses should be included in the methodology.
- Access to proper technical expertise on the system design as well as on safeguards and physical protection measures is essential for a PR&PP evaluation.
- The use of expert elicitation techniques can ensure accountability and traceability of the results and consistency in the analysis.
- Qualitative analysis offers valuable results, even at the preliminary design level.
- Greater standardization of the methodology and its use is needed.

Completeness in identifying potential diversion pathways is a key evaluation goal. Targets and potential pathways can be systematically identified for each specific threat, and plausible scenarios can be systematically found to describe the potential proliferant host state’s strategies to divert target material. A set of diversion pathway segments can be developed, and the PR measures, i.e. the high level PR qualifiers defined by the PR&PP methodology for each pathway, (Technical Difficulty, Proliferation Time, Proliferation Cost, Material Type, Detection Probability and Detection Resources Efficiency) can be estimated.

The diversion threat pathways analysis can also provide a variety of useful information to stakeholders, including regulatory authorities, government officials, and system designers. This information includes how attractive the material is to potential proliferators for use in a weapons program, how difficult it would be to physically access and remove the material, and whether the facility can be designed and operated in such a manner that all plausible
diversion pathways are covered by a combination of intrinsic features and extrinsic measures.

The misuse threat pathways analysis requires consideration of potentially complex combinations of processes to produce weapons-useable material (i.e., it is not a single action on a single piece of equipment but rather an integrated exploitation of various assets and system elements). Given a proliferation strategy, some measures are likely to dominate the others, and within a measure some segments will dominate the overall estimate over the whole pathway.

The breakout threat pathways analysis found that breakout is a modifying strategy within the diversion and misuse threats and can take various forms that depend on intent and aggressiveness, and ultimately the proliferation time assumed by a proliferant state. Furthermore, PR measures can be assessed differently within the breakout threat, depending on the breakout strategy chosen. Note that some additional factors related to global response and foreign policy were identified as being relevant to the characterization of the breakout threat, but those factors are not included in the PR&PP methodology.

A substantial base of analytic tools already exists for theft and sabotage pathway analysis. The case study verified that these tools can be used within the PR&PP methodology framework.

The theft and sabotage threats pathways analysis found that multiple targets and pathways exist. The most attractive theft target materials appeared to be located in a few target areas. Specifically, for the ESFR, the most attractive theft target areas with the most attractive target materials were found to be the LWR spent-fuel cask parking area, LWR spent-fuel storage, the fuel services building staging/washing area, the FCF air hot cell, and the FCF inert hot cell.

The case study generated a number of additional insights. In particular, subgroups noted that during the evaluation process the analyst must frequently introduce assumptions about details of the system design, for example the delay time that a door or portal might generate for a PP adversary. As the study progressed, the working groups realized that, when these assumptions are documented, they can provide the basis for establishing functional requirements and design bases documentation for a system at the conceptual design stage. By documenting these assumptions as design bases information, the detailed design of the facility can be assured of producing a design that is consistent with the PR&PP performance predicted in the initial conceptual design evaluation (or, if the assumptions cannot be realized in detailed design, the original PR&PP evaluations must be modified appropriately).

The PR&PP methodology therefore has the potential to be a powerful tool that can be applied at the conceptual design stage for nuclear energy systems, to generate the design bases for detailed system design. Future work will include efforts to further exercise this approach and demonstrate its utility in guiding the design of Generation IV nuclear energy systems.”

3. Current & future activities of PR&PP WG

The PR&PP WG of GIF will update the methodology in the course of 2010-11 taking into consideration the feedback from the case study. In particular the Measures characterising the pathways (For PR: Proliferation Technical Difficulty, Proliferation Cost, Proliferation Time, Fissile Material Type, Detection Probability, Detection Resource Efficiency; for PP: Probability of Adversary Success, Consequences and Physical Protection Resources) and related metrics will be re-examined and refined or revised where needed. The use for
formalised techniques for the expert elicitation will be discussed by pointing how the PR&PP evaluation methodology can be seen as an expert elicitation process.

In addition, the PR&PP WG is developing, in close collaboration with GEN IV Systems Steering Committees (SSCs), white papers on all six Gen IV systems: Very High Temperature Reactor, (VHTR); Lead Fast Reactor, (LFR); Sodium Fast Reactor (SFR); Molten Salt Reactor, (MSR); Supercritical Water Reactor, (SCWR) and Gas Fast Reactor, (GFR). These white papers will collect general information about the six Gen IV concepts with emphasis on their PR and PP features introduced at the level of design. To this aim a template was prepared covering the following aspects: 1) Overview of Technology; 2) Overview of Fuel Cycle; 3) PR&PP Relevant System Elements and Potential Adversary Targets; 4) Proliferation Resistance Features (relevant for: a) Concealed diversion or production of material; b) Breakout; and c) Production in clandestine facilities); 5) Physical Protection Features (relevant for: a) Theft of material for nuclear explosives and b) Radiological sabotage); 6) PR&PP Issues, Concerns and Benefits. All white papers will be completed in the course of 2010 and will be published highlighting common features of the association fuel cycle and cross cutting PR&PP horizontal issues fro all Gen four systems.

For existing plants, proliferation resistance features and evaluation have traditionally not been part of the design process. Generation IV nuclear energy systems offer the opportunity to consider proliferation resistance as well as physical protection features beginning with the earliest design phases. This involves the consideration of the safeguards system through the so called Safeguards by design Concept [8] and the proliferation resistance and physical protection evaluation methodology offers a tool to iteratively evaluate and enhance the process.

4. References


02.06.2010

Wed 10:30 – 12:10

H2

Civil society energy policy
FUTURE NUCLEAR ENERGY SCENARIOS FOR EUROPE

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ABSTRACT
Nuclear energy is back on the agenda worldwide. In order to prepare for the next decades and to set priorities in nuclear R&D and investment, market share scenarios are evaluated. This allows to identify the triggers which influence the market penetration of future nuclear reactor technologies. To this purpose, scenarios for a future nuclear reactor park in Europe have been analysed applying an integrated dynamic process modelling technique. Various market share scenarios for nuclear energy are derived including subvariants with regard to the intra-nuclear options taken, e.g. introduction date of Gen-III (i.e. EPR) and Gen-IV (i.e. SCWR, HTR, FR) reactors, level of reprocessing, and so forth.

The assessment was undertaken using the DANESS code which allows to provide a complete picture of mass-flow and economics of the various nuclear energy system scenarios. The analyses show that the future European nuclear park will exist of combinations of Gen-III and Gen-IV reactors. This mix will always consist of a set of reactor types each having its specific strengths. Furthermore, the analyses highlight the triggers influencing the choice between different nuclear energy deployment scenarios. In addition, a dynamic assessment is made with regard to manpower requirements for the construction of a future nuclear fleet in the different scenarios.

1. Introduction
The world’s energy demand is growing rapidly reflecting the growth of economy worldwide. However, this growth in energy demand puts a large pressure on natural resources and especially on the environment. Renewable energy technologies and nuclear energy are the main technological options to answer to the energy demand without putting too much pressure on natural resources and the environment. Therefore, nuclear energy is back on the agenda worldwide. In Europe, the European Commission recently presented the Vision Report of the Sustainable Nuclear Energy Technology Platform for the role of nuclear fission energy to the European transition towards a low-carbon energy mix by 2050 [1].

In order to prepare for the next decades, to set priorities in nuclear R&D, manpower, and investment, scenarios are evaluated. This allows to identify the triggers which influence the market penetration of future nuclear reactor technologies. To this purpose, the future nuclear reactor park mix in Europe has been analysed applying an integrated dynamic process modelling technique.

2. Well Reputed Data
All considered scenarios are performed in an intra-nuclear mode without considering full energy market competition (e.g. fossil fuels, renewables, etc...). The scenarios are based on a given (nuclear) energy demand, the existing nuclear reactor park and its foreseen phase-out, the introduction of Gen-III and Gen-IV reactors, various fuel cycles, stocks of uranium and plutonium, and unlimited fuel cycle facility capacity, i.e. assumption is made that investment in new fuel cycle facilities would happen given the growth prospects for nuclear energy. It is evident, that the credibility of the scenarios is closely dependent on the reliability
and objectivity of the above mentioned input data. Therefore, this section will describe the important input data in more detail.

An important boundary condition for the integrated dynamic modelling is the energy demand scenario. Based on data from [3, 4], a realistic energy demand was determined up to the end of the century and has been applied as input for the analyses. There are many well reputed studies all over the world, which determine the (nuclear) energy demand up to 2030 or 2050. However, the need for implementing a nuclear energy demand up to the end of the century, creates a restriction. Therefore, the primary energy demand up to 2030 was taken from [3] as this is a study focussed on Europe. The (low) average growth (~0.1% per year) of the period 2020-2030 has been extrapolated from 2030 up to 2100. The share of nuclear energy as a part of the total energy demand was taken from [4] for the period 2000-2100. This results in a significant growth in the nuclear energy demand during the 21st century. In fact, the derived energy demand scenario projects a growth in nuclear energy demand from 2006 to 2030 of about 82%. This compares well to the projections of ~80% as presented in the World Energy Outlook 2008 [5]. However, one of the considered scenarios described in this paper, deals with a constant nuclear energy demand scenario as presented in [1].

In the analyses, it is assumed that different types of new reactors may be built, taking into account the introduction date of Gen-III and Gen-IV reactors. The following reactors have been modelled: a European Pressurized Reactor (EPR) [6], a Supercritical Water Reactor (SCWR) [7], a medium sized (~600 MWe) Fast Reactor (FR) with a conversion ratio of about 1.25 [8, 9, 10, 11], and a High Temperature Reactor (HTR) for which the effect of cogeneration is included by assuming increased (about 20%) thermal efficiency for electricity production [12, 13]. The construction costs of a Gen-III LWR are taken from a literature survey [14] (assuming equal exchange rates for dollars and euros which will be updated in future simulations). The HTR construction costs are based on the comparison with Gen-III LWR construction costs in [15]. The construction costs for FR are based upon the comparison with Gen-III LWR costs provided in [9]. Finally, the construction costs for an SCWR have been compared with a typical Gen-III LWR in [16]. The main reactor characteristics are summarised in table 1.

<table>
<thead>
<tr>
<th></th>
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<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Gen-III LWR</td>
<td>1600</td>
<td>70% MOX 30% UOX</td>
<td>35.5</td>
<td>91</td>
<td>1</td>
<td>60</td>
<td>2013</td>
<td>2400</td>
</tr>
<tr>
<td>SCWR</td>
<td>1000</td>
<td>MOX</td>
<td>44</td>
<td>90</td>
<td>1.5</td>
<td>60</td>
<td>2035</td>
<td>~80% Gen-III</td>
</tr>
<tr>
<td>FR</td>
<td>600</td>
<td>FR-MOX</td>
<td>40</td>
<td>90</td>
<td>1.5</td>
<td>50</td>
<td>2035</td>
<td>~120% Gen-III</td>
</tr>
<tr>
<td>HTR</td>
<td>200</td>
<td>UOX</td>
<td>40 + 20</td>
<td>85</td>
<td>0.25</td>
<td>40</td>
<td>2023</td>
<td>~105% Gen-III</td>
</tr>
</tbody>
</table>

Decommissioning and operation & maintenance cost estimates are based on the approach of [9]. These costs are taken as 3% and 8% of the construction costs respectively. For the present analyses, no deep burn of MOX fuel in an HTR is considered. As an assumption for all MOX fuel fabrication costs, the required depleted uranium is supplied without additional costs. This mitigates the influence of increasing uranium-prices on reactors employing 100% or partly MOX-fuel. It is important to realise that for the current analyses, it is assumed that the fuel cycle facility capacity is unlimited. However, the associated costs for the different steps in the fuel cycle are considered and taken from [17] and for particle fuels from [18].

Limitations in fuel stocks are assumed for the EU27 countries. For the European natural uranium availability, 30% of the world reserves is assumed taken from [19]. For depleted uranium, reprocessed uranium and plutonium stocks, estimates are based on reports from [20, 21]. The uranium price is largely volatile. In 2007 the price peaked to a value about...
tenfold of the price in 2001 [19]. In the current analyses, the uranium price is modelled by an initial price of 10 $/kgU in 2000 and an escalation rate of 2.5% in the base scenario.

In order to evaluate the manpower requirements for the construction of a future fleet of nuclear reactors in Europe, manpower requirements are based on [22, 23, 24, and 25]. The literature data is interpreted and from this data, profiles have been derived for the construction of Gen-III and Gen-IV reactors. For the scenarios, the time evolution of the determined manpower requirements will be presented as a 10 year walking average, not distinguishing between the different professional sectors.

3. Scenarios
For the assessment of the nuclear fuel cycle strategies, the DANESS code (“Dynamic Analysis of Nuclear Energy System Strategies”) version 4.0 [2], was used to simulate the flows of fissile material, fresh fuel, spent fuel, high level waste as well as all intermediate stocks and fuel cycle facility throughput. Four scenarios have been considered which are summarized in table 2. Scenario 1 is the base scenario. The base scenario assumes an escalation rate of 2.5% for the uranium price development. It applies the nuclear energy demand scenario based on [3] and [4] as explained in section 3. Scenario 2 deals with increased uranium prices. The escalation rate is varied from 2.5% in the base case scenario, to 3.2% and 4.9%. The respective uranium prices at the end of the century in the year 2100 will be 118$/kgU, 233 $/kgU and 1195 $/kgU, representing approximately a double and tenfold uranium price at the end of the century. Scenario 3 deals with increased costs for geological disposal of spent fuel and high level waste. In the base scenario 1000 $/kgHM [17] is considered. Scenario 3 considers a double and fivefold price. Scenario 4 shows the influence of the energy demand scenario. A constant nuclear energy demand is assessed, like described in [1]. This energy demand scenario is provided by EDF. The authors assume that it is therefore referring to the French situation. As the share of nuclear energy in the total electricity production is high in France, an assumption of constant nuclear energy demand may be valid. However, a constant nuclear energy demand scenario for all EU27 countries is considered as the lower boundary.

<table>
<thead>
<tr>
<th>Scenario</th>
<th>Escalation Rate</th>
<th>Disposal Costs</th>
<th>Nuclear Energy Demand Scenario</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 'Base'</td>
<td>2.5%</td>
<td>1000 $/kgHM</td>
<td>DG-TREN / IIASA WEC</td>
</tr>
<tr>
<td>2 'Uranium'</td>
<td>a. 3.2 %</td>
<td>base</td>
<td>base</td>
</tr>
<tr>
<td></td>
<td>b. 4.9%</td>
<td></td>
<td></td>
</tr>
<tr>
<td>3 'Disposal'</td>
<td>Base</td>
<td>a. double</td>
<td>base</td>
</tr>
<tr>
<td></td>
<td></td>
<td>b. fivefold</td>
<td></td>
</tr>
<tr>
<td>4 'Constant Energy Demand'</td>
<td>base</td>
<td>base</td>
<td>Constant Demand</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(SNE-TP)</td>
</tr>
</tbody>
</table>

3.1 Base Scenario
Evidently, all scenarios show the phase out of the existing nuclear park. In order to fill the gap between nuclear energy demand and nuclear reactor park capacity, the only choice for the near future (up to 2023) is to construct Gen-III reactors like e.g. an EPR. It is estimated that for the replacement of the current reactor park, depending on the considered scenario, about 100 EPRs have to be constructed in order to fill the capacity gap. From this year, it is assumed that the technological development of HTR systems has resulted in a commercial availability of such reactors. From 2035, also SCWRs and FRs are commercially available. The reactor order decision making model allows from then the construction of both Gen-III and Gen-IV reactors based on the levelized costs of the different reactors and forecasted energy demand.

Figure 5 shows the time evolution of the nuclear reactor park as determined by the base scenario. The data is presented as 5 year walking average. It shows an evolution towards a reactor park in which there is a significant share for each reactor type. The limitations on the
fuel availability have forced a higher share for FRs than would be the case when fuel availability would be unlimited.

![Graph](image)

**Fig. 5:** Reactor capacity distribution and manpower requirements for scenario 1

Figure 5 also shows that the construction of the Gen-III EPR fleet will already require somewhat more than 25 000 fte around the year 2025. Furthermore, a peak in manpower requirements of about 150 000 fte can be observed around 2070-2090. This is caused by the fact that on the one hand new nuclear plants are constructed to keep up with the rising energy demand, and additionally, the reactor fleet constructed about 40 to 60 years before has to be replaced. Note that the peak manpower requirements as shown in figure 7 are largely influenced by the large amount of HTR modules to be constructed. The manpower requirements for construction are similar in scenarios 2 and 3.

### 3.2 Influence of Increased Uranium Prices

The resulting reactor capacity distributions in the year 2100 are shown in figure 8 (left). It is clear that the difference between scenarios 1 and 2a is not very large, however, when the escalation rate rises to 4.9%, the influence of the increased uranium prices is significant. In such cases, the benefit of FRs used in breeder mode is clearly demonstrated. Their share is increased drastically at the cost of the market shares of Gen-III reactors and HTRs.

![Graph](image)

**Fig. 8:** Installed reactor capacity distributions in 2100 for scenarios 1, 2 (left), 3 (right) and the unlimited fuel availability reference case 0.

### 3.3 Influence of Increased Disposal Costs

The resulting reactor capacity distributions in the year 2100 are shown in figure 8 (right). It is clear that the difference between scenarios 1 and 3 is not very large. It can be observed that the share of FRs and HTRs is slightly increased at the cost of the share of SCWRs. This is caused by the high burn-up characteristic of the FRs and HTRs assumed in the analyses.

### 3.4 Constant Energy Demand Scenario

Due to the increased efficiency of Gen-III and Gen-IV compared to the Gen-II reactors of the current reactor park, a decrease can be expected in the installed reactor capacity. This is clearly demonstrated in figure 12. The current reactor park is basically replaced by about 60 EPRs. In addition, some Gen-IV reactors are build up to 2070. Most of them are HTRs because the HTR is commercially available early enough. This scenario clearly demonstrates that in constant nuclear energy scenarios, the development of Gen-IV systems should be
very fast in order to compete with other reactor types before the current reactor park is replaced. If this development is not fast enough, there will be plenty of time to evaluate the Gen-IV designs and build demonstration reactors, because commercial deployment will only be feasible when the near future reactor park is taken out of operation, which is around 2070. By that time, the distribution completely changes and evolves to a situation in which each reactor type has a significant share. The distribution is not affected by limitations in fuel availability as in scenarios 1, 2, and 3 because there is plenty of fuel available from the assumed stocks for EU27.

![Graph](image)

**Fig. 12: Reactor capacity distribution and manpower requirements for scenario 4**

Compared to the base scenario, at the end of the century in the constant nuclear energy demand scenario, the shares of SCWRs and HTRs is increased at the cost of the shares of Gen-III reactors and FRs. The manpower requirements for the construction of the future nuclear fleet, shown in figure 12, differ in this scenario from the requirements in the other scenarios. The peak manpower requirement occurs again around 2070-2090. However, the peak requirement is now in the order of 50 000 fte compared to about 150 000 in the other scenarios. In order to construct the fleet of about 60 EPRs to replace the current Gen-II reactor park, somewhat less than 25 000 fte of manpower is required. Compared to the base scenario, this is about the same order of magnitude.

### 4. Conclusions

The analyses show that given the considered nuclear energy demand and given a limited number of available Gen-III and Gen-IV reactor types, the future European nuclear park will exist of combinations of Gen-III and Gen-IV reactors. This mix will always consist of a set of reactor types each having its specific strengths.

The analyses highlight some triggers influencing the choice between different nuclear energy deployment scenarios:

- The gap between nuclear energy demand and park capacity can only be filled by Gen-III for the near future (up to about 2023). It is estimated that this requires the construction of about 60 (in the constant nuclear energy demand scenario) to 100 (in all other scenarios) Gen-III reactors with the size of an EPR.
- The constant nuclear energy demand scenario shows that the nuclear reactor park will consist mainly of Gen-III reactors up to 2070. After 2070, the Gen-III reactor park is replaced by a mix of Gen-III and Gen-IV reactors.
- Fast reactor breeders as applied in the current analyses are efficient in uranium consumption. Therefore, FRs are attractive in scenarios with increased uranium prices.
- The analyses show that HTRs are sensitive to increased uranium prices. HTRs loose a part of their market share to FRs and SCWRs when uranium prices increase. However, still a considerable share is maintained.
- The economical attractiveness of the SCWR, achieved by reducing capital costs and increasing efficiency, is identified by the analyses showing a considerable market share for SCWRs in every considered scenario.
The requirements of manpower for the construction of the nuclear fleet in the short term (~2025), i.e. to construct the Gen-III fleet which replaces the current reactor park, are about 25 000 fte. In the long term, a peak in manpower requirements of about 150 000 fte might be expected around 2070-2090.

5. Acknowledgement

The work described in this paper was funded by the Dutch Ministry of Economic Affairs. Furthermore, the authors would like to acknowledge the continuous support of Luc Van Den Durpel and Argonne National Laboratory for the use and support of the DANESS code.

6. References


HARMONIZING NUCLEAR PRACTICES IN THE EU –
CONTRIBUTION OF EUROPEAN NUCLEAR LICENSEES

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ABSTRACT
The European nuclear industry recognized that with the deregulation of the electricity market, diversity of national regulations could seriously distort competition. Therefore harmonizing regulatory practices is the best way of ensuring that the industry can evolve within a stable legal framework. In order to pool resources, the licensees launched mid 2005 ENISS (European Nuclear Installation Safety Standard Initiative) under the umbrella of FORATOM. The principal mission of ENISS is to bring together decision-makers, operators and specialists from the nuclear industry with national regulators in order to identify and possibly agree upon the scope and substance of harmonized safety standards. ENISS currently represents the nuclear utilities and operating companies from 16 European countries with nuclear power programs. ENISS above all provides the nuclear industry with the platform that it needs to express its views, provide expert input and interact fully with regulators throughout the harmonization process. The work of ENISS is a good example of how dialogue and results-oriented participation with stakeholders can help identify optimal solutions to the problems that nuclear industry faces today.

1. Introduction
1.1 Importance of Nuclear Energy in the EU
The spring 2007 European Council endorsed the Commission proposal to cut greenhouse gas emissions by 20%, to increase energy efficiency in the EU by the same amount by 2020 as well as the renewable energy in the energy consumption. The publication by the Commission of the second Strategic Energy Review and the updated PINC (5th Illustrative Nuclear Program based on Article 40 of the Euratom Treaty) in November 2008 underline the progress that has been made in signposting the future of nuclear energy in the EU. It further endorses the important role that nuclear energy plays, along with other low-carbon energy sources, “in the transition to a low-carbon economy.”

Today, nuclear energy generates approximately 935 TWh/a which presents two thirds of the EU low carbon electricity and makes an important contribution to mitigation of global climate change. The current operation of the 144 NPPs in the EU avoids round about 631 million tonnes CO₂ per year.

The contribution of nuclear energy to the overall electricity supply will substantially decrease, unless new plants are built in time and NPPs in service are operated longer (so called “long-term operation” or LTO). Insufficient base load capacity may jeopardise the stability of the EU's electricity network unless countermeasures on a large scale are introduced. According to current projections, the nuclear generation capacity in the EU Member States that is in 2010 approximately 130 GWe would fall by as much as 35
GWe by 2020. This figure takes into account original technical design basis lifetime\textsuperscript{1}, already agreed life time extensions, firm decisions on new NPPs and the currently announced phase-outs. If this loss of base load capacity is not replaced by nuclear, a significant part of it will be generated by CO\textsubscript{2}-emitting gas or coal-fired plants. Renewable energy sources will increase their share but other energy sources will be needed, since possibilities to store electricity are limited and demand has to be met at all times. Therefore LTO of existing reactors and licensing and construction of new reactors as well is an urgent need to achieve the EU GHE targets. In order to contribute to these targets in a sustainable manner on the EU level, the European nuclear power plant operators support the harmonisation of safety requirements for the operation of existing reactors and the standardisation efforts for design and construction of new built.

1.2 Public acceptance
Public opinion regarding nuclear energy production appears to be strongly divided in the European Union. Nearly identical shares of respondents express support for nuclear energy (44\%) and opposition to it (45\%).

Nuclear safety is mirrored in the public perception: a majority of EU citizens (59 \%) believe that nuclear power plants can be operated in a safe manner, rising to almost 80\% in some of the Member States with NPPs. On the other hand, a majority (53\%) of European citizens also still feels that the risks associated with nuclear power as an energy source outweigh its advantages, apparently mainly caused by the lack of progress in the management of radioactive waste and fears about plant security.

The results of the Euro barometer on nuclear safety, published by the EC in February 2007, also reveal that a vast majority of EU citizens thinks that the EU should play an active role in harmonizing legislation between EU Member States (87\%).

2. Regulatory activities
2.1 Nuclear Safety and Harmonisation – Which initiatives have been reached / are on the road?
To achieve a harmonized level of nuclear safety various organizations and initiatives have developed sets of requirements and instruments for supervision over the last two decades. A broad overview is shown in the following table.

The Convention of Nuclear Safety (CNS) and the IAEA Safety Standards, divided in Fundamentals, Requirements and Guides are highly recognised in the nuclear community as the leading sets of requirements, due to the holistic approach, the hierarchy applied and the detailing. The IAEA Safety Standards are developed with the help of expert committees and often used as the base of Member States national regulations. Furthermore the Western European Nuclear Regulators Association (WENRA) aims harmonising nuclear regulatory systems in the EU countries with nuclear programmes and Switzerland by 2010.

\textsuperscript{1} Technical design basis lifetime is only an assumption used during original plant design to base design calculations of systems, structures and components for technical issues such as fatigue. This is not a technical limit.
Activities at the European Level  Activities worldwide

In addition, WANO is a platform for the exchange of operating experience, professional and technical development amongst the NPPs as well as technical support and exchange of information. NPP licensees implemented the European Utilities Requirements (EUR) defining and benchmarking safety requirements and operational targets for new reactor designs.

The Multinational Design Evaluation Programme (MDEP) is a multinational initiative to develop innovative approaches to leverage the resources and knowledge of the national regulatory authorities who will be tasked with the review of new reactor power plant designs. Today these include: Canada, China, Finland, France, Japan, Korea, Russian Federation, South Africa, the United Kingdom and the United States. IAEA will take part in the work of MDEP. The OECD Nuclear Energy Agency (NEA) is performing the Technical Secretariat function.

The nuclear industry established the WNA Working Group on “Cooperation in Reactor Design Evaluation and Licensing” (CORDEL) with the aim of stimulating a dialogue with nuclear regulators on the benefits of globally standardised designs for new reactors. Among others, all greater European NPP operators are supporting this group. Achieving reactor design standardization will require the combined efforts of industry, regulators, policy makers, governments and international institutions.

The CORDEL Group proposed a conceptual three-phase programme introducing a mutual acceptance and eventually internationally valid design approvals for standardized reactor designs. But such an evolution towards internationally valid design approvals would necessarily occur in a manner consistent with each country's sovereignty over its own regulatory framework. Each country's regulator would remain responsible for a
comprehensive licensing and oversight process, with a streamlined design approval simply being one part of it. No aspect of the CORDEL proposal is meant to imply that any national regulatory process would be subordinated or limited by foreign decisions.

2.2 Community legislation
None of these above mentioned approaches has reached the status of a legal binding European Safety Requirements until now. To achieve a harmonised high level of nuclear safety in Europe these approaches have to be considered. The EU community initiative to legalize common nuclear standards for nuclear installations can only be developed in context with these sets of requirements and instruments of supervisions.

The Euratom Treaty does not explicitly address the particular aspects of nuclear installation safety. While the European legislation provides for a binding framework as far as authorization, inspection and enforcement are concerned, the responsibility for implementing and enforcing such a European legislation lies with the Member States. Such directives are legally binding for Member States and shall be transposed into national regulations. The new Council Directive 2009/71/EURATOM establishing a Community Framework for Nuclear Safety adopted in June 2009 provides binding legal force to the main international nuclear safety standards, namely the Safety Fundamentals established by the International Atomic Energy Agency (IAEA) and the obligations resulting from the Convention on Nuclear Safety.

Regarding radiation protection, the Council Directive 96/92/EURATOM is currently under revision.

2.3 The WENRA process
Nuclear license holders supported the work of the Western European Nuclear Regulators Association (WENRA) on the harmonization of European safety standards for existing nuclear power plants, as well as for radioactive waste/spent fuel storage and for decommissioning. WENRA Reference Levels, mainly based on IAEA Safety Standards, were published begin of 2006.

Regarding the Reactor Safety Reference Levels WENRA countries have benchmarked and scrutinized thoroughly by comprehensive processes their national regulations against all Reference Levels.

![WENRA Countries - Legal Evaluation WENRA RLs (2006)](image)
The figure above summarizes the legal requirement data from all 17 WENRA countries. It indicates that the Reference Levels are formally required for over half of the assessments, and 4% have differences that can be justified (grey coloured). Hence, for harmonization on the legal side, there is a need for a significant number of additional, formally issued, generic national requirements. However, the work carried out by the regulators harmonizing national regulations by 2010 is voluntary.

WENRA finalised in December 2009 and published on January 2010 a first version of a pilot study titled “Safety Objectives for new Power Reactors” produced by their Reactor Harmonisation Working Group (RHWG). Though the study is generic in nature, it already shows the directions along which WENRA intends to develop their safety objectives for new plants. WENRA has called for stakeholder comments by the end of June 2010.

3. Activities of Nuclear Licenses
3.1 The ENISS Initiative

The European nuclear industry recognized that with the deregulation of the electricity market, diversity of national regulations could seriously distort competition. Therefore, harmonizing regulatory practices is the best way of ensuring that the industry can evolve within a stable legal framework. In order to pool resources, the licensees launched mid-2005 ENISS (European Nuclear Installation Safety Standard Initiative) under the umbrella of FORATOM. ENISS, which represents the European nuclear power plant licensees, has become a major stakeholder in the consultation processes on harmonising safety standards.

The ENISS Objectives are summarized as follows:

- To establish a common licensee view with respect to the “WENRA RLs” and to present the industry position in discussions with WENRA in a proactive way
- To support an exchange of information about the interaction of license holders with their national regulators, in order to achieve a harmonised set of new regulations
- To create an information platform for the European nuclear license holders with respect to new national and international regulatory activities
- To strengthen the influence in the revision work of the IAEA Safety Standards
- To cooperate with the European Institutions on regulatory issues in the area of nuclear safety, radiation protection, waste management and decommissioning
- To collaborate with international associations dealing with regulatory issues

ENISS currently represents the nuclear utilities and operating companies from 16 European countries with nuclear power programmes and AREVA NC as well. In two cases (Spain, Switzerland) national nuclear associations are on the interest of the licensees.

**ENISS – Membership 2010**

- Belgium (Tractebel)
- Finland (Fortum, TVO)
- Germany (EON, RWE, EnBW
- Italy (SOGIN/ENEL)
- Spain (UNESA)
- The Netherlands (EPZ)
- Switzerland (Swiss Nuclear)
- Czech Republic (CEZ)
- Hungary (PaksNPP))
- Slovakia (Slovenske Elektrarne)
- Bulgaria (Kozloduy NPP)
- United Kingdom (BE)
• France (EdF, AREVA NC)                        • Romania (Nuclearelectra)
• Sweden (EON-Se, Vattenfall AB)             • Slovenia (Krško NPP)

All ENISS Members are representing licensees.

The ENISS main activities are covering the following areas:
- Commenting WENRA RLs and Safety Objectives for New Reactors
- Involve in the IAEA Safety Standard revision work
- Provide comments on the EU Basic Safety Standards
- Participate in the ENEF process
- Feedback of Experience Programme
- Cooperation with EUR
- Cooperation with CORDEL

The structure of ENISS Organisation reflects the needs acting as counterpart in regulatory issues regarding the
- WENRA harmonisation process,
- IAEA Safety Standards revision work,
- European Institutions regulatory framework (nuclear safety, radiation safety, waste safety)

Both the ENISS Reactor Safety Group (RSG) and the Waste/Decommissioning Safety Group (WDSG) are representing expertise and competence in national and international regulatory requirements, licensing and oversight, design, operation, interim storage and decommissioning of nuclear installations across Europe. All comments and documents submitted to third parties are elaborated on a consensual way. If necessary and agreed by the Steering Committee, both Safety Groups are supported by Expert Groups temporary introduced. The Radiation Protection Expert Group is in particular responsible for commenting draft IAEA BSS and draft EURATOM BSS.

As an example the ENISS activities on WENRA Safety Reference Levels will shortly be addressed. ENISS provided end of May 2006, WENRA with official comments:
Regarding Reactor Safety Reference Levels ENISS has made a lot of comments at different stages. Several meetings have been hold with the WENRA RHWG in a constructive mood. ENISS has been able to explain its position and to better understand the WENRA ones.

WENRA issued January 2007 and March 2007 a revised set of Reactor Safety Reference Levels, in addition an explanatory note on PSA (March 2007). In July 2007 an adjustment of Safety Issue C (Quality Management) was issued. This safety issue is now dealing with Management Systems considering IAEA GS-R-3.

### WENRA Reactor Safety Reference Levels

**Comparison Version 1.0**  
**January 2008 – January 2006**

<table>
<thead>
<tr>
<th>Issue</th>
<th>Nb RLS</th>
<th>ENISS Comments</th>
<th>Mod RLS</th>
</tr>
</thead>
<tbody>
<tr>
<td>A: Safety Policy</td>
<td>6</td>
<td>4</td>
<td>3</td>
</tr>
<tr>
<td>B: Operating Organisation</td>
<td>15</td>
<td>11</td>
<td>2</td>
</tr>
<tr>
<td>C: Management System</td>
<td>16 (Version 2006)</td>
<td>9</td>
<td>16</td>
</tr>
<tr>
<td>D: Training and Authorization of NPP staff</td>
<td>15</td>
<td>10</td>
<td>7</td>
</tr>
<tr>
<td>E: Design Basis Envelope for Existing Reactors</td>
<td>23</td>
<td>20</td>
<td>11</td>
</tr>
<tr>
<td>F: Design Extension of Existing Reactors</td>
<td>18</td>
<td>16</td>
<td>11</td>
</tr>
<tr>
<td>G: Safety Classification of Structures, Systems and Components</td>
<td>8</td>
<td>6</td>
<td>6</td>
</tr>
<tr>
<td>H: Operational Limits and Conditions</td>
<td>20</td>
<td>9</td>
<td>7</td>
</tr>
<tr>
<td>I: Ageing Management</td>
<td>6</td>
<td>6</td>
<td>5</td>
</tr>
<tr>
<td>J: System for Investigation of Events and Operational Experience Feedback</td>
<td>16</td>
<td>5</td>
<td>5</td>
</tr>
<tr>
<td>K: Maintenance, In-service inspection and Functional Testing</td>
<td>20</td>
<td>9</td>
<td>5</td>
</tr>
<tr>
<td>L: Emergency Operating Procedures and Severe Accident Management Guidelines</td>
<td>14</td>
<td>6</td>
<td>3</td>
</tr>
<tr>
<td>N: Contents and updating of Safety Analysis Report</td>
<td>16</td>
<td>5</td>
<td>1</td>
</tr>
<tr>
<td>O: Probabilistic Safety Analysis</td>
<td>18</td>
<td>10</td>
<td>6</td>
</tr>
<tr>
<td>P: Periodic Safety Review</td>
<td>9</td>
<td>7</td>
<td>3</td>
</tr>
<tr>
<td>Q: Plant Modifications</td>
<td>15</td>
<td>8</td>
<td>3</td>
</tr>
<tr>
<td>R: On-site Emergency Preparedness</td>
<td>18</td>
<td>7</td>
<td>4</td>
</tr>
<tr>
<td>S: Protection against Internal Fires</td>
<td>10</td>
<td>15</td>
<td>11</td>
</tr>
<tr>
<td><strong>TOTAL</strong></td>
<td>291</td>
<td>165</td>
<td>119</td>
</tr>
</tbody>
</table>

During the WENRA/ENISS interaction process altogether 118 Reference Levels out of 291 were modified. ENISS considers that the last version issued by WENRA in January 2008 of the Reactor Safety Reference Level represents a good and delicate equilibrium between the regulator’s and industry’s positions, with which most NPP operators in Europe could live with.

ENISS will comment advanced versions of WENRA Reference Levels regarding waste/spent fuel storage and decommissioning of nuclear installations. New versions can be expected in 2010. In cooperation with the EUR group a common set of comments on the WENRA pilot study "Safety Objectives for new Power Reactors" is under preparation and will be provided to WENRA mid of 2010.
IAEA Safety Standards are not legally binding in Member States but are used as reference in a lot of national regulations and EU regulations. ENISS as an NGO is actively involved in the revision of the IAEA Safety Standards in commenting drafts and takes part in IAEA Drafting Groups. With its observer status ENISS is representing its position in IAEA Safety Standard Committees (NUSSC, WASSC, RASSC).

ENISS will review and comment the revised EU BSS when the draft will be agreed by the EURATOM Article 31 Committee and issued for public consultation.

3.2 The European Nuclear Energy Forum (ENEF)

The creation of ENEF was initiated by the March 2007 European Council, when Member States suggested “that broad discussion takes place among all relevant stakeholders on the opportunities and risks of nuclear energy.” The Forum’s creation shows that nuclear will now be treated on a level footing with other major energy sources that have had similar forums for some time.

ENEF should provide advice to European policy makers, mainly in the European Institutions on: security of energy supply, incentives for investment, EU legislative issues, public opinion, R&D, knowledge management, safety and waste management. The Forum aims to promote an inclusive, transparent and non-ideological debate on nuclear between all the relevant stakeholders. It gathers for the first time a broad range of stakeholders -the nuclear industry, public authorities, the financial community and various sections of civil society- in a debate on the future of nuclear energy in Europe.

At the first meeting in Bratislava (Slovakia), it was agreed to establish three working groups that are in charge of analysing three main issues: the opportunities of nuclear, the risks of nuclear and information and transparency. The working groups are drafting proposals in order to enable ENEF to provide a roadmap for the continued development of nuclear energy in the European Union. Every working group has created subgroups in order to dwell on specific topics.

The ENEF Risks Working Group established in January 2008 the Sub-Working Group (subWG) Harmonisation, led by ENISS, to undertake an investigation into nuclear safety harmonisation in the EU. The subWG Harmonization analyzed the EU’s legal basis for legal instruments relating to nuclear safety and prepared a document entitled “Considerations on a potential EU Directive based on Common Fundamental Safety Principles for Nuclear Installations”. This paper was presented at the ENEF Bratislava plenary meeting (November 2008) as the ENEF contribution to the new Safety Directive which was being prepared by the European Commission. The paper recommended that an EU Directive regarding safety of nuclear installations should be based on common Fundamental Safety Principles with more detailed requirements in the annex.

Following the publication of the Commission Proposal, the subWG was mandated by the Working Group Risks to complete a thorough analysis of the proposal. The comments and suggested changes put forward by the SG were adopted as the ENEF Working Group Risks response to the Commission Proposal.
The Risks Working Group at its March 2009 meeting agreed to the subWG proposal to enlarge the scope of its activities to tackle in addition nuclear safety issues. Given the wider scope, the subWG was renamed from “Harmonisation” to “Nuclear Installation Safety”. This proposal was endorsed at the ENEF Prague plenary meeting (May 2009).

The work of the subWG is aimed at contributing to further convergence of nuclear safety practices and harmonisation of nuclear safety in the EU. In its work programme, the SWG has been mandated to consider possible criteria and safety objectives for long-term operation of NPPs, including the risk-informed approach.

Most European nuclear utilities filled in the questionnaire on the long-term operation of nuclear power plants that was put together by the Nuclear Installation Safety Subgroup. Following the information it received the subWG has written a paper entitled: Considerations on a potential EU Recommendation on harmonised conditions for safe long-term operation of Nuclear Power Plants in the EU. In conclusion, the SWG agreed to put forward the idea of calling for a European Commission (EC) Recommendation on the long-term operation of nuclear power plants. The paper will be approved by the SWG members and sent to ENEF’s Risks Working Group for approval and endorsement before being discussed at the ENEF plenary meeting in Bratislava, in late-May.

The ENEF Risks Working Group established also in January 2008 the subWG Waste to develop a roadmap to successful implementation of geological disposal in the EU. The Roadmap, adopted by the Working Group Risks in October 2009, presents the basic elements that Member States should consider when developing a national waste management programme, especially for geological disposal of high-level waste.

In the context of the upcoming publication of a possible EU legal instrument for spent fuel and radioactive waste management, the ENEF Sub-Group Waste decided to participate to the stakeholders’ consultation process launched by the European Commission on 31st March 2010. The sub-WG put forward in a position paper the principles which support the idea that the EU needs to develop a common legislative framework governing the management of spent fuel and radioactive waste. The position paper defines, namely, the scope of the possible legally binding instrument as well as the essential elements to be developed in such an instrument. It also tackles what is required to be included in the EU’s national programs for the management of spent fuel and radioactive waste, and it finally gives support to the fact that deep geological disposal is recognised as the only proven, practical solution for the disposal of high level waste and spent fuel.

Due to the international character of new built a small number of standardized designs was deployed worldwide. Manufacturing of nuclear components is perfomed in a global suppy chain. In Europe major utilities emerged as multinational utilities establishing cross-border joint ventures. These new market conditions need to establish international elements in licensing procedures across Europe to achieve standardisation. That means for instance international cooperation of regulators in design assessments and its mutual acceptance based on a common set of codes and standards used for safety demonstration, in the long term intergovernmental agreements on acceptance of design approvals. This issue is handled in the ENEF subWG Legal Roadmap.

For the nuclear industry greater harmonization of licensing procedures in the EU is a major topic. The Legal Roadmap subWG issued on October 2008 a position paper titled
“The Importance of New Approaches in Licensing”. First proposals were provided how to facilitate the licensing of new nuclear power plants by harmonizing and streamlining licensing procedures and encouraging cooperation among Member States in design assessment and design certification. The subWG will support the Commission in launching a study titled “Survey of licensing procedures of EU Member States”. Best practices should be recommended and ways towards international assessment of standard designs should be shown.

4. Conclusion
The future development of nuclear safety practices in the EU is a major challenge for all stakeholders – Member States, European Commission, nuclear industry – each interested party has the responsibility of ensuring the sustainable development of nuclear energy. The European nuclear utilities in order to support that process, launched

- ENISS to drive harmonisation of nuclear regulations on the EU level and their compliance with national regulations
- EUR to drive standardisation in designing new reactor concepts

In addition the nuclear industry contribution to the activities of the European Nuclear Energy Forum provides guidance to European Institutions to facilitate further steps in preparing EU regulations in the nuclear field.
LIMITING CONDITIONS FOR NUCLEAR POWER PLANT COMPETITIVENESS VS. FOSSIL AND WIND PLANTS

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Unska 3, 10000 Zagreb - Croatia

ABSTRACT

The aim of this paper is to compare potential energy options for future electricity generation. The paper considers comparison of discounted total cost of electricity generated by nuclear power plant and by combined natural gas and wind plants, having in total equal electricity generation. Large uncertainty in the future fuel costs makes planning of optimal power generating mix very difficult to justify. Probabilistic method is used in the analysis which allows inclusion of uncertainties in future electricity generating cost prediction. Additionally, an informative functional relation between nuclear plant investment cost, natural gas price and wind plant efficiency, that determines competitive power generation between considered options, is also shown. Limiting conditions for nuclear power plant competitiveness vs. fossil and wind plants are presented.

1. Introduction

The cost of electricity generation is certainly a very important factor in accepting a power plant project. In order to make an appropriate calculation of the cost of generated electricity, it is necessary to include plant costs and load factors during the plant lifetime. This is achieved by calculating the levelized lifetime busbar electricity cost. It is a fictitious average cost depending upon the discount rate and the predicted changes of costs and load factor during plant operation. The advantage of using levelized electricity costs for a comparative competitiveness assessment of different power producing technologies is in the possibility to investigate the impact of changes of some cost items having significant influence on total generation costs, such as changes of fossil fuel cost, during plant lifetime.

The objective of this paper is to summarize the methodology of the probabilistic analysis and to compare levelized cost distribution for a gas fired power plant jointly with a wind power plant, and a nuclear power plant. Coal plant analysis is not included due to uncertain cost of potential plant design used (classic or advanced) and insufficient cost data for CO2 sequestration and storage.

2. Probabilistic method to compare competitiveness of power producing technologies

Cost item used in the calculation has a certain uncertainty margin and a certain probabilistic distribution within this margin. Additionally, the changes of selected cost items could be correlated positively or negatively with absolute value of correlation factors between 1 (fully correlated) and 0 (non-correlated). After defining the cost items with ranges, a distribution within the ranges and correlations, probabilistic distributions of levelized electricity costs could be obtained, Ref.[1]. The standard method to achieve this is to calculate electricity cost using random values (Monte-Carlo method) of each cost item within a given uncertainty range. Large number of such calculations is needed to obtain a meaningful probabilistic distribution of the results. Use of an appropriate computer code capable of handling the necessary number of variables is indispensable in obtaining valid distributions of levelized electricity costs. The parameters used in probabilistic analysis to compare competitiveness of the considered power technologies are: overnight specific investment cost, years of plant construction, constant operation and maintenance cost, variable operation and maintenance
cost, fuel cost, plant efficiency, capacity factor, years of loan repayment, years of plant lifetime, discount rate, average interest rate for loan repayment, expected fuel price change during the plant lifetime. The analysis is performed in the following steps:

1. The expected range of uncertainty for plant performance and the cost variables that affect the levelized cost of electricity are defined;
2. Probabilistic distributions within the ranges and correlation factors for each key input variable are predicted;
3. Monte-Carlo analysis is used to generate random inputs for the calculation of levelized electricity costs on the basis of the described uncertainties;
4. A reasonably large number of calculations are performed to obtain meaningful probabilistic distributions of the results.

The above procedure generates the cumulative probability distributions for levelized costs of electricity for each of the compared technologies. The process includes the generation of a large number of random input variables and the corresponding values of the levelized electricity generation costs. The probability distributions for the levelized electricity generation costs are obtained by counting the number of results obtained within each levelized cost interval (in example which follows 50 intervals have been used), Ref.[2].

One of key issues to consider when comparing long term competitiveness of power plans is the fuel costs (in considered case natural gas cost and uranium cost) and rates of cost increase. Average annual cost increase of nuclear fuel during plant lifetime in analysis which follow is assumed in the range of 2-4 %, and for natural gas in the range of 2-5 %. The foreseen average cost increase rates for natural gas, on the basis of the recent experience, seem to be on the optimistic side. Underestimation of gas cost increase rates would favour gas fired plants as compared to nuclear plants, because of a larger impact of gas fuel cost on the cost of the produced electricity.

3. Example of probabilistic analysis of levelized electricity generation cost

As part of the studies carried out for electrical system expansion analysis in Croatia for the period 2010-2030, probabilistic method has been applied to compare best-estimated differences in levelized lifetime costs of electricity generated by selected power producing technologies, Ref.[3]. As mentioned in introduction, these technologies include combined cycle gas fired plant jointly with wind plants and nuclear power plant with light water reactor. Since the wind plant cannot operate in base load mode alone, combination of wind and gas plant is analysed. Gas fired power plant, as a flexible power generator, is used to promptly replace the missing capacity. Therefore, levelized costs of generated electricity in wind plants is analyzed jointly with the gas fired plant by taking these plants as a single power generator. Investment cost is the sum of the investment costs of a wind and a gas fired plant. Fuel consumption is equal to the gas consumption of gas fired plant for the period it operated instead of wind plants.

In reality, assumed combination of gas and wind plants would generate additional costs to power generating system due to larger cost of power transport and additional costs in gas plant caused by the necessity to permanently accommodate its output power to fill the difference between wind power generation and system demand. Such costs are not considered in the analysis. It is obvious that the inclusion of these costs would improve the competitiveness of nuclear plant.

There are numerous computer codes available to handle the described problem. The presented study made use of the Stochastic Analysis of Technology Systems (STATS) computer code developed in ANL (USA).

The STATS model was originally developed to estimate composite uncertainty distributions for various systems and technologies. It provides a convenient method for treating the uncertainties and correlations between cost and performance components. The method has
the capability to provide improvements in cost comparisons based on the combinations of the best estimated input data. Additional information obtained by uncertainty analysis is useful for considering relative risks and benefits of the selected technology in system expansion analysis.

The code calculates composite uncertainty distributions based on user-defined component uncertainties. It uses the Monte-Carlo method for problem solving. Random number generator is applied to calculate a value for each component; that is, each uncertainty variable is represented as a relative probability function. The code then calculates a composite value by using these selected values for all of the components, as prescribed by the user-specified functional relationship between these values.

Selection of the correct probability distribution of each cost related item is possible if there is reliable information about the ranges of uncertainty of the variables from historical data. If historical data is not available, an expert judgment should be applied, based on the past knowledge about the range of variation of particular input data.

Levelized busbar cost of generated electricity during selected period (usually plant lifetime) is defined as:

\[
LEC = \frac{\sum_{t=1}^{LT} (Fac + Vac)_t}{8760 \sum_{t=1}^{LT} \left(\frac{CF_t}{1 + dr}\right)}
\]

where:
LEC - levelized electricity cost; Fac - annual fixed plant cost in year t; Vac - annual variable plant cost in year t; CF - plant capacity factor in year t; LT - plant life time; dr - discount rate

Input data used in the analysis of levelized costs for wind-gas synthetic power generator and nuclear plant are indicated in Table 1.

<table>
<thead>
<tr>
<th>Plant type</th>
<th>Nuclear</th>
<th>Combined cycle gas</th>
<th>Wind</th>
</tr>
</thead>
<tbody>
<tr>
<td>Overnight plant investment US$/kW</td>
<td>3100</td>
<td>3300</td>
<td>850</td>
</tr>
<tr>
<td>Construction period years</td>
<td>5</td>
<td>3</td>
<td>3</td>
</tr>
<tr>
<td>Equity %</td>
<td>15</td>
<td>25</td>
<td>15</td>
</tr>
<tr>
<td>Fixed annual O&amp;M costs US$/kW</td>
<td>80</td>
<td>100</td>
<td>10</td>
</tr>
<tr>
<td>Fuel Cost (uranium, natural gas)</td>
<td>0.0015 $/kWh</td>
<td>0.0025 $/kWh</td>
<td>7 $/GJ</td>
</tr>
<tr>
<td>Assumed fuel cost increase rate %</td>
<td>2%</td>
<td>4%</td>
<td>2%</td>
</tr>
<tr>
<td>Balance of nuclear fuel cycle cost $/kWh</td>
<td>0.004</td>
<td>0.005</td>
<td>-</td>
</tr>
<tr>
<td>Decommissioning cost $/kWh</td>
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<td></td>
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<td>0.002</td>
</tr>
<tr>
<td>External cost</td>
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<td>Capacity factor %</td>
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<td>85-22</td>
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<tr>
<td>Plant lifetime years</td>
<td>60</td>
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<td>30</td>
</tr>
<tr>
<td>Investment repayment period years</td>
<td>20</td>
<td>15</td>
<td>15</td>
</tr>
<tr>
<td>Equity repayment period years</td>
<td>10</td>
<td>10</td>
<td>10</td>
</tr>
</tbody>
</table>

Table 1. Estimated cost data with their expected ranges of uncertainty

Assumed order of capacities of considered power plants are for nuclear 1000 MW, for gas 300 MW and for wind 1-3 MW. The data are partly based upon EIA Energy Outlook 2009,
Ref.[4]. Each of the indicated cost parameters, rates and durations of loan repayment is characterized by an uncertainty range and a distribution within the range, as also given in Table 1. These data are based on the best engineering judgment relative to the expected ranges and distribution within the ranges. The cost calculation is made on constant currency value for the plants entering commercial operation around 2020. The estimated width of the ranges and internal distribution allowed by the code (even, triangular or 5 point) depend upon reliability of data. For data with large degree of uncertainty (such as data on future costs of power plants investment, maintenance and fuel) it is most justified to use wide ranges and even distributions within ranges. Such approach is applied for input data shown in Table 1.

4. Results of levelized cost calculations

Two thousand levelized cost calculations for each plant type were carried out. The results obtained with the STATS code were re-arranged using the EXCEL spreadsheet to produce curves shown in Figure 1.

![Distribution of levelized cost of generated electricity in nuclear power plant and in system containing natural gas and wind plants](image)

Figure 1. Distribution of levelized cost of generated electricity in nuclear power plant and in system containing natural gas and wind plants

The figure provides information on the number of random calculations of costs for each of 50 discounted cost intervals. Cost interval with the largest number of random calculation results is considered as most probable cost interval. It can be seen that, on the basis of the input data given in Table 1, nuclear power plants, considering the total plant life period, produce less expensive electrical energy than a natural gas combined cycle plant jointly with wind plants. This is mainly caused by present and expected future increase of gas cost and increased total investment cost respective to gas plants alone, due to wind generators. It could be seen from the graphs that obtained distribution shows that the most probable range of levelized busbar costs for the system of combined cycle gas plants and wind power generators is in the range of 0.115-0.125 US$/kWh, while for nuclear power plant is in the range of 0.068-0.078 US$/kWh.

5. Relationship between values of most influenced costs items for which competitiveness of considered options would be achieved

Analyses show that dominant influence on plant competitiveness have nuclear power plant investment cost, natural gas cost and wind plants capacity factor.
On the bases of this conclusion a functional relation between given parameters was elaborated for which levelized cost for both options would be equal. Despite possibility to apply also in this case probabilistic method, the calculation for simplicity and in order to obtain first estimates, was performed by using deterministic approach. Result is shown in figure 2.

![Figure 2. Relationship between nuclear power plant competitive overnight investment cost, natural gas cost and wind plants capacity factor for which levelized electrical power generating cost would be equal](image)

It can be seen from the Fig. 2 that nuclear plant with overnight investment cost in the range given in Table 1 (3100-3200 US$/kW) would be competitive if natural gas is higher than about 6 $/GJ with combined cycle gas plants only, but with same natural gas cost limit competitiveness could raise to about 4300 $/kW if power system contains also wind plants. With present natural gas cost of 7-8 $/GJ the competitiveness is achieved if nuclear plant overnight investment cost is around 4300 $/kW compared with gas plant only, but competitive nuclear plant overnight investment could increase to 5000$/kW if wind plants are included. These values are slightly dependent upon wind plant capacity factor in the indicated range. With higher wind plants capacity factor, competitive NPP investment must be lower. The influence of wind plants capacity factor increases with gas cost (because fuel saving is more valuable). The value of wind plants electricity generation also increases if CO2 emissions are limited, or if cost of damage caused by CO2 emissions becomes larger.

6. References

4. EIA Annual Energy Outlook 2009- Assumptions
DEREGULATION AND INTERNATIONALISATION - IMPACT ON THE SWEDISH NUCLEAR INDUSTRY

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ABSTRACT

The deregulation of the Swedish electricity market in 1996 was well known in advance, and the nuclear power plants in Sweden, as well as their main suppliers, made early preparations for this new situation. In a study performed by the author at Malardalen University in Sweden - it is concluded that the electricity industry, including the nuclear power plants, was fundamentally transformed in conjunction with market liberalisation. Two large foreign companies, E-on and Fortum, entered the Swedish market and became part-owners of the nuclear plants. After deregulation, the electricity market in Sweden is dominated by these two companies and the large national company Vattenfall. Similarly, Vattenfall has recently grown into an international energy company, acquiring generation capacity in Northern Europe outside of Sweden, including nuclear power plants in Germany. Restructuring of the nuclear industry on the supplier side started in the 1980s, when the Swedish company ASEA and BBC of Switzerland merged to become ABB. Several years later the Swedish nuclear plant supplier ABB-Atom became part of Westinghouse Electric Company, today owned by Toshiba. The Swedish experience thus confirms an international trend of mergers and consolidation in the nuclear industry.

1. Introduction

When studying the history of nuclear power in Sweden, it becomes apparent that the development can be divided into three distinct phases:

- 1945 – 1970: A monopolized national development program
- 1975 – 1985: Construction of twelve commercial power reactor units and a public debate that culminated in a national referendum in 1980
- 1990 – present: Deregulation and internationalisation
While several studies in history and political science have been made for the two first phases of development, very little effort seems to have been devoted to the role of nuclear power in deregulated markets. The present study is probably the first to be made in Sweden. The study was finalised in 2009 by the author while studying history at Malardalen University in Vasteras, Sweden.

2. Background

2.1 Historical development

When development of nuclear energy was started in Sweden directly after World War-II, this new energy source was immediately monopolized by the state. The reason was that Sweden at that time was considering developing a nuclear weapon, and the preferred technology was a dual purpose solution, heavy water reactors that could operate on domestic natural uranium and produce both electricity and weapons grade plutonium. A state owned research and development company called AB Atomenergi was formed and made responsible for the nuclear design of the plants that were to be operated by the state owned electricity producer Vattenfall. Two heavy water plants were built in the 1960’s, but only the first one was taken into operation. This was a pressurized heavy water reactor, that for some years supplied electricity and hot water for district heating to a community outside Stockholm. The second plant, Marviken, was a larger boiling heavy water reactor, that was never taken into operation for safety reasons. The development of heavy water reactors was immediately stopped and AB Atomenergi dismantled. The reasons were partly the safety problems at Marviken, but also the political decision not to develop a Swedish nuclear weapon. Sweden was to ratify the Non Proliferation Treaty (NPT).

A privately owned industrial company, ASEA, was early selected as equipment supplier for the nuclear plants that were to be designed and operated by the state. ASEA started intimate cooperation with AB Atomenergi and Vattenfall and thus became experienced in the field of nuclear plant projects. ASEA, however, did not fully believe in heavy water reactor technology and had an ambition to become an independent supplier of nuclear power plants. The company secretly studied the development of light water reactors in the U.S., in particular the boiling water reactor (BWR) by General Electric. ASEA developed its own version of a BWR, without any license from GE. At the same time, a group of small power producers decided to
build a nuclear power plant outside the town of Oskarshamn in Sweden. They asked for bids on BWRs both from ASEA and GE. In 1965 they formed a joint nuclear power company, OKG, and the very same day Oskarshamn -1 was ordered from ASEA as a turn-key plant. The breakthrough of commercial nuclear power in Sweden was thus made by privately owned companies both on the supplier and customer side. However, the Swedish state decided to take part in this new development. A joint venture company, ASEA-ATOM, was formed by the state and ASEA. In 1969, this new company won the order by Vattenfall for its first nuclear power unit, to be built at Ringhals. Westinghouse Electric Company in the U.S. got the order for a pressurized water reactor (PWR) unit to become Ringhals-2. Westinghouse was later on to receive orders for two more PWR units at Ringhals. The first two BWR plant units by ASEA-ATOM were soon to be followed by new orders from the private power industry for Oskarshamn-2 and two units at Barseback. After having built five plant units based on a conventional BWR design, ASEA-ATOM developed a much more advanced design with features like internal recirculation pumps. This new design was to be built as three BWR units at Forsmark in Sweden and two similar units at Olkiluoto in Finland. All ASEA-ATOM plants were built on a turn-key basis. In international comparisons of eg. capacity factors they have shown good performance. No ASEA-ATOM plants were built outside Sweden and Finland.

Somewhat contrary to the intense construction programme for new reactors, an intense public debate started over nuclear safety, in particular the issue of waste handling. This debate was detrimental to the ruling social democrats and contributed to the formation of a non-socialist government in 1976, which was dissolved after two years in power, due to internal disagreement over the nuclear power plant programme. The nuclear plant units were for many years a sensitive political issue; even after the social democrats came back in office, and unlike the early days of development, the party was now divided over nuclear power. A national referendum over nuclear power was held the year after the 1979 reactor accident at Three Mile Island in the US. In 1984, as a result of the referendum, Parliament decided to limit the number of nuclear plant units to the ongoing programme of twelve units. Building of new nuclear reactors was prohibited.
2.2 The 1990’s: Deregulation and internationalisation

The 1990’s was a period of market deregulation and liberalisation in many business sectors, such as finance, telecommunications and electricity trading. The origin of this new policy may be traced back to the politics of the governments of Margaret Thatcher and Ronald Reagan in the late 1980’s. In some European countries like Sweden, these liberal policies were quickly embraced by political parties on both sides of the political scale, but it was not until after the deregulation in Great Britain and Norway that Sweden decided to deregulate its electricity market and join Nord Pool, the electric power exchange in Oslo. After careful preparations, such as division of the former public utility Vattenfall into a large business oriented company for electricity generation and trading, called Vattenfall AB, and a smaller public company for operating the main electricity grid, the Swedish deregulation was implemented as of 1. January, 1996. During the last few years before deregulation took place, there was a consolidation of Swedish electricity producers into fewer and larger power companies. Furthermore, two large foreign companies, Eon of Germany and Fortum Power and Heat of Finland acquired power plants in Sweden and became substantial part-owners of the nuclear power plants. The newly formed Swedish company Vattenfall AB simultaneously expanded in northern Europe and became a large owner of German nuclear power plants. Vattenfall is today a far larger power company than the public utility it used to be before deregulation.

Looking at the other category of companies, nuclear technology companies that are main suppliers to the nuclear power plants, the transformation of the industry beyond national borders has been different from that of power producers. The nuclear plant and fuel businesses are now truly global, contrasting the electricity producers, that are local or regional. In Sweden, the ownership of ASEA-ATOM was first changed when the state decided to withdraw from its part-ownership of ASEA-ATOM after the referendum over nuclear power. In 1987, ASEA merged with BBC of Switzerland to create ABB, a global technology company that soon decided to withdraw from the power generation business entirely. The nuclear business, including ABB-Atom was acquired by British Nuclear Fuels plc., that immediately integrated it into Westinghouse Electric Company, a subsidiary of BNFL. However, in 2005 Toshiba Corp. acquired Westinghouse from BNFL.
3. Swedish Nuclear Power Plants Today

Twelve nuclear power plant units were built in Sweden, and ten are still in operation (2010). The two units at Barseback were prematurely shut down by the government for political reasons. A "shut-down law", empowering the government to stop the operation of nuclear power plant units, was passed by Parliament in 1997. After the new law was first applied to the Barseback plant, the Swedish Supreme Administrative Court called for inhibition of the government’s decision, a unique and controversial situation. Barseback was eventually shut down after negotiations, and the owner (at that time Sydkraft AB, which is today part of Eon) was compensated by part ownership of Vattenfall’s Ringhals plant. Shares in the nuclear power plants have also been used for equity when restructuring the power industry; Eon became part-owner also at Forsmark when Vattenfall acquired its nuclear generation capacity in Germany. Fortum is a minority owner both at Oskarshamn and Forsmark. State owned Vattenfall is still the largest owner of Swedish nuclear power plants with majority at Ringhals and Formark, while Oskarshamn is owned by two foreign companies, Eon and Fortum, that are both registered at the stock exchange.

The three large power companies Vattenfall, Eon and Fortum are dominating the trade of electricity in Sweden, a classical oligopoly situation. One may ask how they can jointly own large nuclear power units and still be competitors. The answer is that the nuclear plants are not making any profit because they are producing electricity at cost. Power is supplied to their owners in proportion to their share of the plant. Naturally, the owners have to share budgets, including new investments in the plants. According to their websites, the level of new investments is very high at all Swedish plants and include measures for power uprates and plant life extension. Thus the estimated plant lifetime may be increased from 40 to 50 or 60 years.

4. Conclusion

Deregulation and internationalisation has had a major structural impact on the Swedish nuclear industry, giving Swedish companies access to foreign markets while at the same time increasing domestic competition. Electricity supply in Sweden and Northern Europe tends to expand outside national borders to neighbouring countries, while main vendors of nuclear technology products tend to integrate in a few large global companies with local presence in Sweden.
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Plant operation
OPTIMIZED LASER SYSTEM FOR DECONTAMINATION OF PAINTED SURFACES

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ABSTRACT

Laser systems have long been seen as potentially very interesting for removing contamination from surfaces. The main expected advantages are the possibility of remote process and the absence of secondary waste. However these systems were unable to find their way to an industrial deployment due to the lack of reliability of the laser and the difficulty to satisfactory collect the (contaminated) ablated matter. In this contribution we report on a compact, reliable and efficient laser decontaminating system called ASPILASER®. It is adapted to the constraints bound to a nuclear environment. It takes advantages of the recent progress made by the fibre lasers which have now a lifetime longer than 20000 hours without maintenance. The collecting system collects all the removed matter (gases and aerosols) on nuclear grade filters. The fully automated system has been successfully tested on a vertical wall of a stopped nuclear installation. It has demonstrated an efficiency of 1 m²/hr which is in the same order of other classical techniques but with a much lower quantity of waste and the ability to work continuously without human intervention. Measurements performed after the laser treatment have shown that the contamination was completely removed by removing the paint and that this contamination was not re-deposited elsewhere on the wall. The system will also be used in highly contaminated hot cells to decrease the radiation and allow maintenance or refurbishing in safe working conditions.

1. Introduction

After final shutdown all the nuclear facilities have to be decommissioned. Radioactive contamination can be found on various parts such as piping, tanks, instruments, painted walls or floors and have to be decontaminated. Increasing societal, environmental and economical constraints put strong requirements on the decontamination procedures for the site owners. In particular this includes the limitation of personnel exposition, the reduction of the produced waste and the respect of the time schedules. This leads to the development of new decontamination methods.

The decontamination methods can be subdivided into two main categories. The mechanical methods include scabbling, milling, grinding as well as water-jetting or sand blasting. They are widely used because of their simplicity, their low price and versatility. However they tend to produce a large quantity of waste and need important manpower. The second class of process is the chemical methods. In this category spray or foam reagents, gels ... can be found. They are well adapted to metallic surfaces, are moderately cost effective and produced large amount of secondary waste which could be difficult to handle.

The laser enters the first class of process. It involves the heating of the surface and the removal of the contamination either by fusion of the layer (paint) or the ablation of this layer (oxide layer, paint, radio-active deposit). Since the nineties it has been seen as a potentially very promising tool for removing surface contamination. As the main expected advantages, it has been mentioned the fact that they were truly non-contact methods with the possibility of remote process and the absence of secondary waste. Furthermore they can be very selective and remove only the contaminated layer of a surface. Nevertheless they couldn’t find their way to an industrial deployment mainly due to the complexity and the investment
cost of a laser system, the lack of reliability and the running constraints (dangerous gases for example). Furthermore less attention was paid to the removed matter and, in some cases the laser system spreads it more widely within the facility in a less tractable form (very small particles, dangerous gases).

We present here a compact and efficient laser decontamination system called ASPILASER®. It relies on pulsed fibre lasers emitting in the near infrared and producing mJ pulses at high repetition rates. Those lasers made tremendous progress in terms of performance and reliability. The laser is associated to a scan head to scan the laser beam over the surface to decontaminate. The collecting system can collect all particles and gases on nuclear grade filters.

This system was tested in a shutdown nuclear facility, previously used to handle spent nuclear fuel. The experience was realized in real working environment. It consisted of removing some alpha contamination on walls by removing the paint. The fully automated system was able to completely decontaminate 6 m² at a rate of 1 m²/hr. All the ablated matter was collected on filters and there were no need for the workers to wear any particular protection. Co-activity was also possible during the laser run.

2. Principles

In nuclear facilities paints are often used as a protective layer to the surface building whose role is to avoid the contamination to migrate into the wall. The contamination is mainly due liquid projections or particles deposited on the wall surface. At the end of the facility’s lifetime, the decommissioning strategy consists of removing the paint from the wall. Removing paints with lasers has long been seen as a promising technique, especially with pulsed lasers (for a review see [1]). Since the emergence of high power laser diodes, continuous-wave lasers were also tried [2]. From these previous studies and our know-how we came to the conclusion that a decontaminating system based on the laser technology should have the following characteristics:

- the system should be simple to use and reliable (no optical alignment, adapted to harsh environments), safe (no emission of dangerous gases or small particles, thus allowing co-activity)
- it has to be efficient with efficiencies at least in the same order of other “classical” techniques and with no need for a second operation like vacuum cleaning,
- it should request a minimum of operating requirements (no assisting gas, no cooling water, …) with a good wall-plug efficiency,
- it should be cost effective and reusable (i.e. the noble parts could be cleaned and decontaminated).

2.1 Choice of the laser source

Different laser types were used to study the paint ablation ranging from the eximer laser emitting in the UV to the Nd:YAF laser in the near infrared (1.06 µm) and the CO₂ laser in the mid-infrared (10.6 µm). To reach the ablation domain the power deposited on the surface should be in the order of some MW/cm². This regime can only be obtained with pulsed lasers. In the CW regime the layer is melt (in case of metals) or burnt (in case of paints).

To be able to safely decontaminate large areas, it is preferable to choose a laser wavelength which can propagate through an optical fibre. This practically restricts the wavelength of the laser to the visible and the near-infrared domain.
Experiments were conducted on both wavelengths 1064 nm and 532 nm [3] which revealed that the paint removal efficiencies were approximately the same. As near-infrared lasers, i.e running on the fundamental wavelength, are simpler and more common we chose this kind of lasers.

An important progress was made recently in the development of a new laser technology which benefited from the telecom boom in the late 90’. Sealed, industrial and reliable fibre lasers appeared on the market with excellent properties: diffraction limited beam, nanosecond pulse length, very long lifetime, high repetition rate, good wall-plug efficiency. Although the mean output power is somewhat limited, this laser is optimum for the ablation regime we are looking for [3].

The laser characteristics we used to develop our system are the following:
- Energy per pulse : 1 mJ
- Average output power : 50 W
- Repetition rate : 50 kHz
- Pulse width : 140 ns
- Wavelength : 1070 nm
- $M^2 < 1.6$ (practically diffraction limited)
- Wall-plug efficiency : > 10%

2.2 Optimisation of the laser-matter interaction

The ablation efficiency (ablated depth versus the laser fluence in J/cm$^2$) of paints with a laser is a typical S curve [4]. An example is given on figure 1. The ablation threshold is in the order of a few tenths of J/cm$^2$ and the saturation begins between 2.5 and 3.5 J/cm$^2$. To get these saturation fluences the laser beam has to be focused typically on a 150 µm diameter spot. Furthermore, to decontaminate a large surface, the laser beam has to be scanned over the desired area at high speed with precision.

![Fig. 1: Ablated depth per pulse versus the laser fluence at two different repetition rates (20 Hz and 10 kHz) [3]](image)

Both actions (scanning and focusing) are realised with a galvanometric scan head similar to the one found in industrial laser marking systems. In our case the scan head can scan over 30 by 30 cm$^2$ and focus the laser beam on a 140 µm spot. The way of scanning is of prime importance in order to avoid redeposition and to allow a perfect cleaning of the surface [5]. The main advantage of this scanning system is the fact that there is no moving part during the laser processing: This facilitates the collection of gases and particles produced during the process.
2.3 Collecting gases and particles

In a simplified approach, laser ablation in the nanosecond regime could be split into 3 different phases: First, the laser light is absorbed by the material and this leads to a fast increase of its temperature. In a second step the matter which has reached the vaporization temperature is ejected from the surface and form gases or condensed nanoparticles. Then, due the constraints in the material generated by the rapid temperature rise, some larger particles are also ejected from the surface. Figure 2 gives a typical size distribution of the particles in the case of paint ablation. The nanometric aggregates are mainly coming from the decomposition of the polymer and the larger particles contains mainly the pigments [6].

![Image](image.png)

Fig.2: Typical size distribution of the particles ejected from the surface in case of paint ablation; the smaller particles are mainly aggregates from the polymer, larger ones are mainly pigments [6].

Those gases and particles contain the contamination and have to be totally collected. It should be noticed that even without contamination the gases and the nanoparticles could be dangerous for the health.

The system was tested in our laboratory and we found that there was no gas or particles escaping the aspiration cone.

3. Results obtained in a nuclear facility

The system was tested in a nuclear facility, currently being dismantled at the CEA centre of Cadarache (France). The wall surfaces were polluted with alpha contamination mainly due liquid projections or aerosols. Those hot spots have to be removed before building destruction. The laser system was put on a XY moving system allowing an automatic displacement of 3x2 m² step by step (in our case a step is approximately 30 cm). This surface can be easily increased; the photograph below (figure 3a) shows the system in the nuclear environment. The system is based on four 50-W fibre lasers and is fully automated to treat 6 m². Measurements of the radioactivity made before and after the laser process showed a complete decontamination, much below the required threshold of 0.8 Bq/cm² in α contamination. The measured values were always below the detection limit of the apparatus (figure 3b), showing the absence of any contamination especially due to a re-deposition. Furthermore we made complementary measurements three months after the experiment: Again we found no activity demonstrating also the absence of “bleeding” (contamination coming back to the surface from the concrete). We estimated the process yield to be approximately 1 m²/hr. It should be pointed out that no other operation was required (vacuum cleaning of the floor, further cleaning of the wall). Along with the collected particles.
(= 100 g/m²) the only waste generated was the pipe of the aspiration system. The system is safe for the workers and allows co-activity.

![Photograph of the laser system in a nuclear environment.](image1)

![Before-after measurements showing the absence of radioactivity after the laser process.](image2)

**Fig. 3:** (a) Photograph of the laser system in a nuclear environment. (b) Before-after measurements showing the absence of radioactivity after the laser process

### 4. Conclusion and perspectives

We have designed a novel laser system for surface decontamination. We used industrial pulsed fibre lasers and an optimized way of scanning preventing any re-deposition. We also put some efforts on the collecting system in order to collect all the gases and the aerosols emitted by the laser ablation. The automated system was tested in a nuclear installation and proved the expected performances: High decontamination factor, very limited production of nuclear waste (almost no secondary waste), safety and simplicity of use, and an hourly yield of \( \approx 1 \text{m}^2/\text{hr} \) which is comparable to other techniques but does not need a second operation and could work 24hr/day, 7 days/week. Commercial products are expected to be available at the end 2010, and will be industrialized and commercialized by the company ASTRIANE (Manosque, France).

### 5. References


ABSTRACT

The paper describes the modeling of WWER-440 reactor and determination of weighting factors and axial functions of ex-core detectors by forward MCNP calculation. Universal and detailed whole-core WWER-440 model was developed for precise 3D Monte Carlo calculations in MCNP5 code. Calculation of weighting factor distribution on the level of fuel-pins was performed for a selected source range ex-core detector. A fitting procedure based on weighted least squares method was designed and applied to obtain the axial weighting functions for representative core region fuel pins. The results show the improvement in finding precise distribution of the weighting factors and the shape of axial functions of ex-core detectors.

1. Introduction

The contribution of fuel assemblies to the ex-core detector response depends not only on the power, but also on the position of the given assembly in the core. The weight of the inner assemblies is several orders of magnitude lower than the outer ones. Consequently, the detector response for a given reactor power is strongly influenced by the spatial power distribution and indirectly by the parameters determining the distribution, such as core loading pattern, time elapsed, position of control assembly, coolant temperature, etc. Precise knowledge of the spatial weighting functions can be very beneficial for the solution of various reactor physical, operational, and safety problems. The work was based on the previous study of Csom, Czifrus, and Feher [1]. In this paper, a developed calculational model and method is described to determine the spatial weighting functions of ex-core detectors for the WWER-440 reactor taking into account different operational parameters.

2. Weighting function problem

The spatial weighting function gives a relationship between the spatial power distribution in the core and the ex-core detector response. The weighting function can be defined in various ways in dependence on the problem to be solved. In general, it is position-dependent contribution of a given part of the core to the ex-core detector response. In this particular case the weighing function value represents the average number of reactions occurred in the ex-core detector per one source neutron created in one-twentieth of a fuel pin height. It follows from the above definition that real ex-core detector response is obtained by convolution of two spatial functions - the power distribution function and the weighting function, and integration of the response function over the whole core region in interest.
3. Computational method

Considering the complicated geometry of the space between the core volume elements and the ex-core detector, reliable neutron transport calculation can only be performed using a technique enabling the treatment of complex 3D geometry. Therefore the MCNP5 Monte Carlo code was chosen [3]. Calculation of the spatial weighting functions of ex-core detectors using the Monte Carlo method is not novel. The application of the MCNP code for determination of weighting functions is reported in the literature [1] and [2]. For several reasons, the forward method was applied contrary to the “time-saving” adjoint method. For example, forward mode of run makes possible to eliminate errors due to the homogenization of the assembly as well as the use of group-wise nuclear data. Neutron transport calculation was performed by the so named fix-source method. This method enables to compute single weighting factor values from separate fuel pin elements of the source region.

4. Computational model

Regarding the objective of the work to determine the weighting functions with high accuracy and reliability, geometric and material part of the reactor model had to be created in the finest possible details. Precise 3D whole-core model of the WWER-440 was developed in MCNP5. The model has a wide range of applications in various areas of reactor-physical, operational and safety calculations and analyses. Horizontal and vertical section of the WWER-440 model is shown in Fig. 2. The next step was focused on optimization of the universal model in order to the effective neutron transport calculation from defined core source region to the given ex-core detector.

Fig 1. Placement of the WWER-440 ex-core detectors

Fig 2. Horizontal and vertical section of the WWER-440 model in MCNP5

Spatial source region boundary depends on the required accuracy of the calculation. However, reduction in uncertainty below a given threshold can be unreasonable because of high computation time demands. The geometric bounds of the source region set in the model cover all of those fuel pin elements whose weights exceed 0.5% of the calculated maximum.
weighting factor value. The total number of fuel assemblies to be taken into account is 54, see Fig. 1. The calculation of weighting functions presented in this paper was performed for a boron-lined proportional counter CPNB44 installed at the 3rd unit of NPP Jaslovske Bohunice. This source-range ionization chamber is used for the startup reactivity measurements.

5. Computational results

The paper presents selected calculational results of the axial and horizontal weighting factor distribution in the source region of the core. Approximation of analytical function was performed in the axial direction so far. Weighting factor distribution in the horizontal direction is shown for two selected core layers.

5.2 Axial weighting function

Axial weighting factor distribution was calculated in the source fuel pins of the core. In the calculation, 193 source pins was considered all together. These pins are distributed in a quasi-regular grid which covers 54 fuel assemblies of the core, see Fig. 1. Axial functions of four selected fuel assembly source pins are presented in this paper only.

The objective of the axial approximation was to find an analytical function which properly describes weighting factor distribution in vertical direction that is along the source fuel pins. Different approximation functions were investigated and the following polynomial function was proposed as the most appropriate one:

\[ w_a(z) = a_0 + a_1 z + a_2 z^2 + \ldots + a_M z^M, \]  

where \( w_a(z) \) is axial weighting function of ex-core detector related to a given pin, \( z \) - height coordinate of a pin element (fractional distance between the central plane of symmetry of the core and the center of pin element, the value is positive above the plane of symmetry), \( a_0, \ldots, a_M \) are approximation constants.

Degree of the polynomial function depends on the position of given source pins in the core. It changes from a degree of 8 for closer peripheral pins up to a degree of 4 for distant ones. Another application of the axial approximation is smoothing the calculated MCNP weighing factor values for a horizontal approximation. The aim is to improve the quality of approximation in horizontal direction (in a given layer of the core). In terms of this procedure the horizontal approximation is applied not on the calculated weighting factor values, but on the axially smoothed values.

Weighted least square method (WLSM) was chosen for the vertical fitting [4]. Fitting program based on the WLSM uses Single Value Decomposition method (SVD) to determine optimal approximation constants values arising from Eq. 1.

The shapes of approximation curves correspond to the theoretically expected weighting factor distribution in vertical direction. As it follows from Fig. 3, the axial weighting factor distribution in the closest pin (-11,4,070) to the ex-core detector as well as in the selected close pin (-10,4,065) is described by a polynomial function of the 8th order, which is a typical “bell” curve. With respect to the distant pin (-4,-7,127), axial distribution is expressed by a polynomial function of the 4th order with the shape of “arc” curve. Distribution in the medium-distant pin (-8,6,065), given by a polynomial function of the 6th order, shows transient character between the above two curve shapes. Extension of the curves that is increase in the FWHM (Full Width at Half Maximum) with increasing distance of a given fuel pin from the ex-core detector is mainly caused by geometrical reasons. In the case of distant pins, the distance ratio of middle and upper/lower pin elements from the ex-core detector is smaller.
than for closer pins. Axial curves are symmetrical in respect of the middle plane of symmetry of the core because of symmetrical ex-core detector positioning.

![Graph showing relative axial weighting factors of four selected fuel assembly pins.](image)

Fig 3. Relative axial weighting factors of four selected fuel assembly pins

5.2 **Horizontal weighting factor distribution**

The paper presents computational results obtained for two selected layers of the twentieth of fuel pin height. The first layer (No.11) is situated immediately above the central plane of symmetry of the core. The second one (No.20) is the upper peripheral layer of the core. As it was mentioned earlier, the twentieths of fuel pins were selected in such a way that they cover the model in a quasi regular grid. At average, 3 or 4 twentieths of fuel pins (elements) were calculated for each fuel assembly, with the exception of peripheral core region being closest to the ex-core detector where 5 - 7 elements per assembly were selected. All together 193 fuel pin elements were calculated.

The horizontal weighting factor distribution for two selected layers of the core is shown in Fig. 4 and 5. From the obtained distributions results the following:

- the major portion of the ex-core detector response is determined by a few peripheral fuel assemblies located in the closest vicinity of the ex-core detector,
- the weighting factor values vary significantly within a given fuel assembly,
- in the closest fuel assembly region to the ex-core detector, in the direction from the maximum weight position to the core centre, approximately eight-fold decrease of weighting factor values is observed,
- based on the weighting factor distribution in the given layers it can be expected that the horizontal function will "warp" in the upper and lower part of the source region, i.e. their values will not be mainly determined by the distance from the ex-core detector,
- weighting contribution of the closest 20 peripheral fuel assemblies to the ex-core detector signal represents 92% from the sum of the core weighting factor values,
- it can be expected that the horizontal weighting function will have a character of exponential polynomial function of the 3rd - 8th order of magnitude, generally expressed by the following equation:

\[
    w_h(x, y) = \exp \left( \sum_{i=0}^{K} \sum_{j=0}^{L} a_{ij} x^i y^j \right),
\]

(2)
where \( w_h(x, y) \) is the horizontal weighting function of the ex-core detector related to a given core layer,

\( x, y \) are the coordinates of a pin element (fractional distance between the central plane of symmetry of the core and the center of pin element),

\( a_y, \text{resp. } a_y, \ldots, a_{i3} \) - approximation constants.

Fig 4. Weighting factor distribution in the central layer of the core No. 11

Fig 5. Weighting factor distribution in the upper peripheral layer of the core No. 20

6. Conclusion

Present practise still points out the need for the determination of spatial weighting functions of ex-core detectors. The exact knowledge of these functions can be very useful for the solution of various reactor physical, operational and safety problems especially for interpretation of reloads start-up measurements. The objective of the work was to present the methodology for determination of weighting factors and functions at the twentieth of fuel pins level using Monte Carlo computational approach. This can be achieved even within a statistical uncertainty of 1% at acceptable CPU time. The calculational results show that the major portion of the WWER-440 ex-core detector response (more then 90%) is attributed to less than 20 fuel assemblies being located closest to the given detector.

7. References

Some commercial nuclear power plants have been permanently shut down to date and decommissioned using dismantling methods. Other operating plants have decided to undergo an upgrade process that includes replacement of reactor internals. In both cases, there is a need to perform a segmentation of the reactor vessel internals with proven methods for long term waste disposal.

Westinghouse has developed several concepts to dismantle reactor internals based on safe and reliable techniques. Mechanical cutting has been used by Westinghouse since 1999 for both PWR’s and BWR’s and its process has been continuously improved over the years.

Detailed planning is essential to a successful project, and typically a “Segmentation and Packaging Plan” is prepared to document the effort. The usual method is to start at the end of the process, by evaluating the waste disposal requirements imposed by the waste disposal agency, what type and size of containers are available for the different disposal options, and working backwards to select the best cutting tools and finally the cut geometry required. These plans are made utilizing advanced 3-D CAD software to model the process. Another area where the modelling has proven invaluable is in determining the logistics of component placement and movement in the reactor cavity, which is typically very congested when all the internals are out of the reactor vessel in various stages of segmentation.

The main objective of the segmentation and packaging plan is to determine the strategy for separating the highly activated components from the less activated material, so that they can be disposed of in the most cost effective manner. Usually, highly activated components cannot be shipped off-site, so they must be packaged such that they can be dry stored with the spent fuel in an Independent Spent Fuel Storage Installation (ISFSI). Less activated components can be shipped to an off-site disposal site depending on space availability. Several of the plants dismantled to date in the US have repackaged the less activated waste back into the reactor vessel and shipped the entire assembly to the disposal site. Decisions like these can be driven by many factors such as disposal...
costs, transportation logistics, licensing fees, etc., but will have a significant impact on the segmentation and packaging plan so must be considered early in the planning phase. All segmentation tools are remotely controlled since the mechanical segmentation projects that Westinghouse has executed, so far, have been performed under water due to the high radiation levels. ALARA and personal safety is the number one priority during the site work. The complexity of the work requires well designed and reliable tools. Westinghouse has optimized the technologies from its experiences accumulated over the years. Its main focus has always been to improve tool handling and cutting speed, water cleanliness, fail-safe and safety aspects. Different band saws, disc saws, tube cutters and shearing tools have been developed to cut the reactor internals. All of those equipments are hydraulically driven which is very suitable for submerged applications.

The purpose of this paper will be to provide an overview of the Westinghouse mechanical segmentation process, based on actual experience from the work that has been completed to date.

Introduction
Some commercial nuclear power plants have been permanently shut down to date and decommissioned using dismantling methods. Other operating plants have decided to undergo an upgrade process that includes replacement of reactor internals. In both cases, there is a need to perform a segmentation of the reactor vessel internals with proven methods for long term waste disposal.

For more than twenty years, Westinghouse has developed several concepts to dismantle reactor internals based on safe and reliable techniques, including plasma arc cutting (PAC), abrasive waterjet cutting (AWJC), metal disintegration machining (MDM), or mechanical cutting.

In recent years, the cost of control and disposal of highly irradiated secondary waste has continued to rise. Therefore, Westinghouse has streamlined its segmentation strategy by selecting mechanical cutting methods over abrasive and thermal methods. Using mechanical cutting also reduces the risk of a secondary waste spill which can significantly impact final pool clean up issues and increase personnel exposure due to the spread of contamination.

Project Experience
Since 1999, Westinghouse has segmented all types of reactor internals in Boiling Water Reactors (BWR’s) using mechanical cutting methods. The projects have been performed in the Nordic region in Forsmark 1, 2 and 3, Oskarshamn 1, 2 and 3 and Olkiluoto 1 and 2. Figure 1 shows the BWR internals that Westinghouse has cut, so far, with the types of mechanical tools that have been used.

Fig. 1. Internal parts segmented by Westinghouse
More recently, Westinghouse has been awarded a contract from EDF to dismantle the Chooz A reactor vessel. In those projects, the internals are always cut under water due to the high radiation level and all tools are therefore remotely controlled. ALARA and personal safety is the number one priority during the site work. The complexity of the work requires well designed and reliable tools. Westinghouse has optimized its mechanical cutting techniques from the experience gained from the various segmentation projects. Examples of areas where improvements have been made are handling means, cutting speed, cleanliness and safety. The Westinghouse personnel is experienced and well trained on mock-ups and in an environment similar to the site work.

Radiation Challenge
The assessment of the radiation level that the personnel is expected to face is always performed before any site activity starts. If this assessment shows that the dose rate will be too high, measures are taken to reduce it. The actual dose rate taken by the personnel is continuously followed throughout the work. Careful planning results in man dose levels no more than the normal levels achieved when working inside the reactor containment. All site activities are planned in detail and in close cooperation with the customers. Special areas for tool maintenance are set up in the reactor containment. All tools are designed to be used and controlled remotely. Training on mock-ups in a non hostile environment before going to site is a very important step in order to optimize all procedures and thereby minimize the dose rate exposure on site for the personnel.

The waste containers that are filled with cut pieces, have restrictions regarding the allowable dose rate for shipping to the waste disposal facility. Westinghouse has the experience from expected radiation level of the internals cut on previous projects. In addition, Westinghouse has also the competency to perform advanced dose rate calculations in order to find out the specific activity for each component. The dose rate calculations, in combination with our experience, are used to optimize the waste container packing plan. Figure 2 shows the dose rate around a filled waste container.

Packing Experience
When performing a segmentation project, it is very important to plan the cutting and packing activities thoroughly. An important part of every project is therefore to create a cutting and packing plan. In the cutting plan all cuts of the internal parts are studied thoroughly to obtain cut objects with dimensions to receive the highest filling ratios inside the storage containers. The packing plan is a detailed report with good accuracy of how many containers that will be needed and also works as an instruction to the site personnel on how to position the cut pieces in the best way inside the containers. The packing plan specifies which type of waste containers and how many that must be ordered for the whole project execution.

As the storage cost is significant for our customers it is very important that the actual packing corresponds to the forecast. Westinghouse has never exceeded the number of containers that we have stated in the packing plans. Figure 3 shows a container filled with cut pieces from a packing plan. Figure 4 shows a packed container from a segmentation project in Oskarshamn.
Segementation Project in Forsmark (1999-2000)

Westinghouse first segmentation project was performed in Forsmark in connection with the exchange of Core Shroud and Core Support Grid for Forsmark 1 and 2. The cutting of the two Core Shrouds was performed with a band saw and the two Core support grids were cut with a heavy plate shearing tool (see fig. 5) and a band saw.

Experience collected from one segmentation work improves all work significantly when performing a similar segmentation. In this Forsmark segmentation, the repeated work in the second reactor was executed with 30 % less time on site.

Segementation Project in Forsmark (2003-2004)

Westinghouse has segmented the Core Spray System (CSS) in all three reactors in Forsmark (F1, F2 and F3). The first two CSS (in F1 and F2) were removed from the Core Shroud Cover (CSC) with Electrical Discharge Machining technique (EDM) before the CSS was cut with different shearing tools (see fig. 6). As the Core shroud cover was to be reused in the reactor operation this cutting procedure and handling had to be performed with the highest accuracy, quality and cleanliness.

The cutting of CSS in F3 was performed during two months in 2004. Westinghouse used different shearing tools and a specially developed type of band saw (see fig. 7). The project execution on site was done according to the timetable and with the full satisfaction for the customer.
Westinghouse performed segmentation of a number of internals in Oskarshamn 1 and 2 during an upgrade project in 2003-2004, see table 1. The Core Shroud Covers (CSC) complex design required a number of special developed cutting tools. A tube cutting tool was designed in order to cut the steam pipes on the CSC. When the pipes were cut the CSC was turned upside down in order to reach the Core Spray System. The turning of the 15 ton CSC (see fig. 8) was a very critical operation that required extensive mock-up testing and structural verifications before the start of the actual site work. After turning the CSC, it was then segmented with different shearing tools before we could start to cut the remaining CSC with band sawing technique (see fig. 9). The segmentation time on site was 1.5 month for O1 and 7 months for O2. Westinghouse made a lot of testing on full scale mock-ups before going on site. Improvement work on procedures and tools was also done on site in order to gain the best results. The adjustments and the experience we gained from the first CSC improved the site work for the second CSC. The second execution time was about 30% shorter.

<table>
<thead>
<tr>
<th>Internal Part</th>
<th>Power Plant</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core Shroud Cover (2 pcs)</td>
<td>O2</td>
</tr>
<tr>
<td>Core support grid</td>
<td>O1</td>
</tr>
<tr>
<td>Core support grid</td>
<td>O2</td>
</tr>
<tr>
<td>Feed Water spargers (6 pcs)</td>
<td>O2</td>
</tr>
<tr>
<td>Core spray riser pipes (8 pcs)</td>
<td>O1</td>
</tr>
<tr>
<td>Core spray riser pipes (8 pcs)</td>
<td>O2</td>
</tr>
<tr>
<td>Test channels (9 pcs)</td>
<td>O1</td>
</tr>
<tr>
<td>Test channels (3 pcs)</td>
<td>O2</td>
</tr>
</tbody>
</table>

Table 1: Internal parts cut in Oskarshamn – 2003-2004
Segmentation Project in Olkiluoto (2004-2006)

In 2004-2006, a segmentation project was performed in Olkiluoto (OL1 and OL2). The cutting in OL2 was made in the time period between the outages 2004-2005 and for OL1 between outages performed in 2005-2006. The internal parts that were cut in this project are described in table 2 below. We had experience from cutting CSC and Core support grid, so these activities gained from this experience. The new activity for this project was cutting of steam separators which were segmented with a band saw (see fig. 11). Overall all project procedures were performed with the full quality. New experience was the sawing in inconel material which is very hard. During this project, we also introduced a new cutting technique for some support beams on the CSC. We used a disc cutting tool that proved to be very well suited for this kind of cut. Figure 10 shows cutting of steam tubes in Olkiluoto.

![Fig. 10. Cutting of steam tubes in Olkiluoto](image)

![Fig. 11. Cutting of Steam separators in Olkiluoto](image)

<table>
<thead>
<tr>
<th>Internal Part</th>
<th>Power Plant</th>
</tr>
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<tbody>
<tr>
<td>Core Shroud Cover</td>
<td>OL2</td>
</tr>
<tr>
<td>Core Shroud Cover</td>
<td>OL1</td>
</tr>
<tr>
<td>Core support grid</td>
<td>OL2</td>
</tr>
<tr>
<td>Core support grid</td>
<td>OL1</td>
</tr>
<tr>
<td>Steam Separators (19 pcs)</td>
<td>OL2</td>
</tr>
<tr>
<td>Steam Separators (19 pcs)</td>
<td>OL1</td>
</tr>
</tbody>
</table>

Table 2: Internal parts cut in Olkiluoto – 2004-2006

**Mechanical Cutting Technique**

Mechanical cutting has a number of advantages compared to other cutting techniques:

- The technique produces a minimum amount of secondary waste.
- The visibility during cutting is very good because the cutting produces only a negligible amount of micro particles.
- Chips from the cutting process falls down to the bottom of the cutting pool and are easy to collect.
- No gases are produced that can cause airborne contamination.
- The technique is safe and reliable.
- All reactor internal sizes, materials and thicknesses can be cut.

**Ongoing Projects and Testing**

Westinghouse has ongoing segmentation projects in Forsmark 1, 2, 3 in Sweden, Olkiluoto 2 in Finland, Chooz A and BCOT (EDF maintenance facility) in France. Testing and qualification is performed on full scale mock-ups in a specially designed pool for segmentation purposes (see fig. 12).
Conclusion
Westinghouse’s experience in mechanical cutting has demonstrated that it is an excellent technique for segmentation of reactor internals. Westinghouse continues to develop this technique to address the market needs. Based on its extensive experience, Westinghouse has found that the slight sacrifice in cutting speed is more than made up for in the savings in both cost and schedule gained by eliminating the need for special waste handling equipment and processes. These are typically required when large volumes of secondary waste are created as a by-product of the cutting process. Using mechanical cutting also reduces the risk of a secondary waste spill which can significantly impact final pool clean up issues and increase personnel exposure due to the spread of contamination.
THE POTENTIAL OF HIGH POWER LASERS IN NUCLEAR DECOMMISSIONING

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ABSTRACT

Contaminated concrete and pipework present major nuclear decommissioning challenges in terms of the total volumes of material to be treated, the radiation levels present and the number of facilities affected. A number of concrete decontamination techniques have so far been evaluated and whilst water-jetting or mechanical scabbling tend to be the favoured options, each has drawbacks such as significant secondary wastes or the need for extensive control and deployment systems. Concrete decontamination by means of laser scabbling has the potential to avoid many of the above drawbacks. Whilst scabbling has been demonstrated at laboratory scale, to date, no representative scale demonstration has been provided which would give industry confidence in the technique. Although cutting of pipes has been performed on numerous occasions, most of the techniques employed are slow to operate or are not suitable for remote deployment in highly radioactive cells. Remote laser cutting of pipework also needs to be adequately demonstrated, before active deployment will be seriously considered. This paper will describe the results of using optical fibre delivered laser power to remove the surface of representative concrete samples. In addition, the capability of the laser for single-sided tube cutting will be described. In this latter process, a finely focused laser beam is used to sever a tube by moving the laser beam in a single axis, across the stationary tube.

1. Introduction

This paper describes the results of a project to demonstrate the potential of high power lasers in aspects of nuclear decommissioning. In March 2009, the UK’s Nuclear Decommissioning Authority awarded TWI a contract to develop prototype equipment, in order to effectively demonstrate the twin processes of concrete scabbling and tube cutting and how these technologies might be implemented for remote use in nuclear decommissioning environments. The goal of the project was to allow Site Licence Companies and supply chain companies to evaluate the technology for any particular decommissioning activity, in terms of both process capability and operating costs, in the knowledge that the necessary underlying technical and engineering issues had already been addressed.

Contaminated concrete and pipework present major decommissioning challenges in terms of the total volumes of material to be treated, the radiation levels present and the number of facilities affected. The topics are highlighted on many occasions in the Lifetime Plan (Technology Baselines and Underpinning R&D) documents for Sellafield, Dounreay and Magnox North. A number of concrete decontamination techniques have so far been evaluated and whilst water jetting or mechanical scabbling are favored options each have drawbacks such as significant secondary wastes or the need for extensive control and deployment systems. Concrete decontamination by means of laser scabbling has the potential to avoid many of the above drawbacks. However, whilst the technique has already been demonstrated at a laboratory scale, see for example [1], to date, no representative scale demonstration has been provided which would give industry confidence in the technique. Although cutting of pipes has been performed on numerous occasions, most of the techniques employed are slow to operate or are not suitable for remote deployment in highly active cells. Laser cutting is well suited to remote deployment due to light and compact process heads, the lack of reaction force between the head and the tube and limited fume generation. [2]. However, as with laser...
scabbling, the process needs to be adequately demonstrated before active deployment will be seriously considered.

2. Industrial Lasers
A key parameter in most laser processes is the power density in the beam applied to the surface of the material in question. The two processes of concern in this paper are unusual in that laser cutting requires a very high power density in the beam, whereas laser scabbling requires a relatively modest power density. Because of the former requirement, the need for an efficient, robust and compact laser source and the need, in remote applications, for optical fibre delivery of the laser beam power, for this work an industrial fibre laser was chosen. This laser has an output power of 5kW, adequate to demonstrate both processes, but the same type of laser is commercially available with powers up to 50kW. In a fibre laser, the laser light is generated inside a small diameter optical fibre, some tens of metres in length. This fibre is connected to the beam delivery fibre, which is of the ‘plug and play’ type and easily interchangeable. The delivery fibres are well protected in a flexible metallic armored sleeve. Such fibres can be manufactured up to several hundred metres in length, without appreciable losses in delivered power. The fibre laser produces light with a wavelength of about 1micron, ie invisible to the human eye, being in the near infra-red part of the spectrum. The performance of the laser was monitored using a laptop computer, which also provided detailed information about the operating status of the laser. Control of the laser is from the controller of the deployment system in use, in this case an articulated arm robot.

3. Single-Sided Laser Cutting
Laser cutting is a very well established manufacturing process which accounts for the largest use of high power lasers. The majority of work performed is the cutting of material up to about 20mm thick, with exceptional quality of the resulting edge. Tube cutting is also performed commercially, but almost all of these systems rotate the tube under a stationary laser beam. For single-sided tube cutting with a laser beam, alternative systems are required. The laser light arriving at the cutting head down the optical fibre first expands as it leaves the fibre and is then made parallel by a lens. Below this lens, a second lens then focuses the laser light to a very small spot to create the power density needed for cutting. The system used in this work is unusual in that its focusing lens had a focal length of 500mm. The effect of this is to produce a very narrow beam of light, with a large depth of focus. This large depth of focus is a major contribution to the process of single-sided tube cutting. The laser beam was enclosed by a cutting nozzle and a nozzle tip with an exit diameter of about 5mm. In contrast to conventional laser cutting, for tube cutting, the laser beam focus is positioned about 90mm below the tip of the nozzle, allowing tubes up to 170mm in diameter to be cut from one side. The cutting process was also assisted by a jet of air at high pressure, which exited the nozzle concentric with the laser beam. This compressed air was necessary to blow away material in the kerf of the cut melted by the laser beam, and is of particular importance for single-sided tube cutting, in order to achieve separation of the tube. The cutting head can be seen in Figure 1. This cutting system could also be equipped with a camera which looks directly through the cutting nozzle, focusing at approximately the same point as the laser beam. This is useful in remotely positioning the cutting head above the tube to be cut. For the work described here, the cutting head was manipulated by an articulated arm robot. All movement of the process head and hence the laser beam, switching of the compressed air and control of the laser, was achieved through the single robot controller.

Using this equipment various options for single-sided tube cutting were possible. Stainless steel tubes from 25 to 170mm in diameter, with a range of wall thicknesses from 1.5 to 11mm, were cut using single pass, two pass and multiple pass techniques. Generally speaking, a two pass technique proved the most efficient. If the cut edge of a tube is examined, it is clear that the quality of the cut at the side closest to the cutting head is much cleaner than the opposite side. This is because, on the first pass, most of the energy in the laser beam and the assist gas, are used to cut material originally contacted by the beam. Only laser energy and gas which have passed through the upper section of the cut are available to address the lower
section and this is cut less effectively. For the second pass, a kerf has been previously opened in the top section and now the majority of the laser energy passes through this and acts more effectively on the lower section of the tube. As an example of performance, Figure 2 shows the dependence of the maximum cutting speed at which the tube is severed as a function of laser power, for a tube of 155mm diameter and 1.5mm wall thickness, for two pass cutting. Note that the cutting speed appears to be linear with applied power, at least up to 5kW. The optimum assist gas pressure appeared to be about 8bar. Figure 3 shows cut sections from 60mm diameter tube, with wall thicknesses from 1.5 to 11mm, again for two pass cutting. Process parameters can be found on the figure caption. The largest tube to be cut had a diameter of 170mm with a 7mm wall thickness. Using a three pass technique this tube was severed in a time of 7min, using 4.8kW of laser power. Another possibility demonstrated was the cutting of concentric tubes. For example, a 25mm diameter tube located inside a 60mm diameter tube. In this case a two pass technique was effective in severing both tubes at once.

![Figure 1](image1.jpg)

**Figure 1**
The cutting head

![Figure 2](image2.png)

**Figure 2**
Cutting speed as a function of laser power for a 155mm diameter tube with a 1.5mm wall. Two pass cutting, assist gas pressure 8bars

4. Concrete Scabbling

In the laser scabbling process, the laser beam is applied to the surface of the concrete and its energy is absorbed, heating the concrete matrix and the concrete aggregate. Expansion of residual water vapor, probably in both the matrix and aggregate and differential expansion between aggregate and matrix, causes the concrete to break up in a highly energetic fashion, leaving a rough scabbled surface, consisting of matrix and aggregate. In any effective use of this process for decontamination, clearly the laser beam must move with respect to the concrete surface and the ejected debris must be contained. In this work, the former was achieved by the use of an articulated arm robot and the latter by enclosing the process and using a large pump and filtration system to recover the debris. A photograph of the scabbling head showing its major components can be seen in Figure 4.
In the scabbling system, the laser light was fed, via an optical fibre, to a set of optics similar to that used for laser cutting, although in this case the focal length of the lens used was much shorter. The laser light was brought to a focus at a small diameter aperture and then allowed to expand to a diameter of about 60mm at the base of a debris recovery tube. This tube, about 150mm in diameter, was terminated round its circumference by a steel wire brush, in contact with the concrete surface. The aperture and the region through which the beam passed below the focusing lens were both protected by jets of compressed air. On this system the air pressure and any possible contamination of the optical elements were continuously monitored. If contamination occurred a warning signal was automatically generated. If the compressed air failed, the laser beam could not be released.

The top of the aluminium tube was connected to a long flexible hose and hence to a pumping system which removed the concrete debris as it left the surface of the material. The complete scabbling head was mounted on the arm of an articulated robot, which was itself mounted on a linear gantry some 6m in length. The scabbling process and effective debris removal requires the process head to be at all times roughly perpendicular to and at a constant distance from the concrete surface. The 6 axes of motion offered by the robot allow this to be achieved. However, the scabbling head was also equipped with its own vision system. A combination of low power lasers and a camera were mounted on the side of the scabbling head. The information recorded by the camera is interpreted by software and the results fed back automatically to the robot controller. In this way, once a scabbling area has been set, the vision system and its feedback to the motion controller of the robot, will automatically maintain both the attitude of the head perpendicular to the concrete surface and a constant stand off distance, as the scabbling process proceeds. A large 16kW motor powered the vacuum system which removed the concrete debris. Air was sucked in at the base of the scabbling head, through the wire brushes. This air draws the concrete debris into the flexible tube and down to the first stage of an enclosed separation process. Concrete particulate matter was deposited in a first container and concrete dust was collected via a filter, in a second
container. The body of the pumping unit also contained two additional filter housings capable of containing HEPA filters. The efficacy of the debris removal system was high, with hardly any scabbled material remaining on the concrete regardless of its orientation.

For a given laser spot size on the concrete, the main process parameters are the laser power and the travel speed. Work performed has indicated that removal rate is proportional to laser power, at least up to the 5kW of power available with the laser being used. At 5kW power, this system has removed 1m² of material, to a depth greater than 10mm, in a time of 110min. A single pass of the process results in a scabbled ‘trough’, lenticular in section. This shape is related to the energy distribution in the incident laser beam, which at the concrete surface is Gaussian in form. A slower process speed will generally result in a deeper scabbled section. For concrete containing limestone aggregate, the deepest section has been measured at 22mm, using a laser power of 5kW and a travel speed of 100mm/minute. For removal of large surface areas, a track overlap of 50% proved to be the most effective for producing a uniform scabbled profile. The process appeared independent of the attitude of the concrete.

Re-scabbling over an existing track is possible, and does result in an increased removal rate. However, in multi-pass processing of the same track, the amount of concrete removed was seen to drop at each successive pass. For example, at 5kW laser power and 300mm/minute travel speed, the maximum depth of scabble recorded for three successive passes of the beam was 10mm, 18mm and 22mm, respectively. Surface contaminants such as grease and paint (Figure 5) had no effect on the scabbling process.

5. Conclusions
- For concrete with a limestone aggregate, a 5kW laser will remove 1m² of surface to a minimum depth of 10mm in under two hours. Coverage can be increased by either reducing the depth of removal or by increasing the laser power. For other types of aggregate, which show less reaction to the laser than limestone, scabbling is not as effective and further work is required to optimise performance.

- A very effective and efficient system for cutting of stainless steel pipes and other fixtures/fittings has been developed. The cutting head is both lightweight and has a significant stand-off tolerance and so is relatively simple to remotely deploy and operate.

6. References

7. Acknowledgements
The authors are grateful to the Nuclear Decommissioning Authority for funding the work reported in this paper and for giving permission for its publication. The assistance of Matt Spinks and Paul Fenwick in conducting the trials is also acknowledged.
02.06.2010

Wed 10:30 – 12:10

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ENEN PhD Event
EXPERIMENTAL AND COMPUTATIONAL ANALYSIS OF STEAM CONDENSATION IN THE PRESENCE OF AIR AND HELIUM

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ABSTRACT
Among the different phenomena expected to occur within nuclear reactor containments during a postulated loss of coolant accident, condensation on containment walls plays a major role, since it represents an important heat sink for evacuating the energy released by the discharge of the primary water. Nevertheless, condensation strongly affects other relevant phenomena, like containment atmosphere mixing, that influences the distribution of noncondensable gases hypothetically delivered in severe accident conditions. In this scenario, the role of condensation is not obvious, since it can locally aid the hydrogen produced by the oxidation of the core claddings to concentrate and reach flammability limits, providing a dangerous effect instead of a positive one.

The understanding of condensation in the presence of air and hydrogen is therefore a fundamental task for the safety analyses of reactor containments. This research has been carried out with the aim to contribute to the understanding of these phenomena. A double strategy has been adopted, including complementary experimental and computational activities. Novel data have been made available by the CONAN facility, investigating the effects induced by light noncondensable gases in experimental configurations that were scarcely investigated in past studies. Computational fluid dynamics (CFD) condensation models have been developed and validated. The suitability of helium as a substitute for hydrogen in experimental activities has been investigated by theoretical and computational analyses allowing to establish simple criteria for the scaling of condensation tests in the presence of a light noncondensable gas.

1 Introduction
One of the main goals to be achieved by the adoption of CFD codes for containment safety analysis is the reduction of uncertainties and therefore an increase of confidence in the safe design of nuclear power plants. However, before a massive use of CFD codes can be considered advisable, the evidence of their reliability must be provided. In this aim, experimental data useful for the validation of these codes have to be produced. One of the most important benchmarks carried out in the field of condensation modelling in the last decades is probably the International Standard Problem 47 (ISP47) [1]. Analysing the results of the benchmark exercise, a general need was felt for achieving a better understanding of the effect of light noncondensable gases and improving the predictive capabilities of CFD models; in this purpose, smaller scale experimental analyses are advisable than those used for the ISP47. In this direction, research activities in the field of condensation modelling have been coordinated by University of Pisa and CEA in the frame of the SARnet network on severe accident [2, 3]. Moreover, the separate effect test (SET) CONAN facility operated by University of Pisa has been modified and adopted to produce data on steam condensation in atmosphere of air and helium [4,5,6]. The aim of this paper is to summarize the main findings of these computational and experimental investigations, focusing on those aspects that can constitute the basis for further improvements in the understanding of condensation phenomena [6].

2 CFD models
A common strategy for modelling wall condensation in CFD codes consists in assigning volumetric sinks in the cells adjacent to the condensing interface. Volumetric sources of mass, energy and momentum are implemented in the balance equations, respectively defined as

$$S_m = \dot{m}_\nu \rho / (2 \Delta c) \quad S_h = S_m h_v \quad S_q = S_m v$$
In order to evaluate the condensation mass flux \( \dot{m}_{v,i} \) and thus the sources, different approaches can be adopted. In this work, two models have been developed adopting different strategies and having different purposes.

2.1 The HMTDM model (Heat & Mass Transfer Diffusion Method)

This model is capable to evaluate the mass transfer rates on the basis of concentration distributions in the near wall region, without requiring any specific closure law; however, it requires a very fine space meshing since its accuracy depends on the concentration profiles next to the condensing wall. The condensation mass flux is calculated as

\[
\dot{m}_{v,i} = \frac{j_{v,i} \cdot \bar{n}_i}{1 - Y_{v,i}}
\]

The mechanistic character of the model consists in the way in which the diffusion fluxes are calculated. The development of an appropriate model basing on the principles of irreversible thermodynamics was necessary, where the diffusion mass flux of the generic \( k \)-th species in a mixture of \( n \) species is given by

\[
\overline{j_k} = \rho \sum_{j=1}^{n-1} D_{kj} \nabla Y_j
\]

Due to its computational cost, the HMTDM model is hardly applicable to large scale geometries. However, it is very useful for achieving a better understanding of condensation and relevant information for the development of coarser models for large scale analyses.

2.2 The HMTAM model (Heat & Mass Transfer Analogy Method)

This model estimates the condensation mass transfer rates on the basis of the heat and mass transfer analogy. The condensation mass flux is assigned as

\[
\dot{m}_{v,i} = \dot{h}_{m,0} \left( F \frac{Y_{v,i} - Y_{v,b}}{1 - Y_{v,i}} \right) = \dot{h}_{m,0} \cdot \ln \frac{Y_{w,b}}{Y_{w,i}} = Sh_{0,x} \frac{\rho D_{sm}}{x} \cdot \ln \frac{Y_{w,b}}{Y_{w,i}}
\]

where the Sherwood number is obtained by the heat and mass transfer analogy as

\[
Sh_{0,x} = Nu_{0,x} \left( Sc / Pr \right)^{1/3}
\]

The key point of the model is the choice of the appropriate correlation for the Nusselt number, which can be troublesome for complex geometries and not well defined phenomenology. However, since the use of relative coarse meshing is possible, a significant reduction of computational resources is achieved, making this model applicable to large scale or even to full containment scale analyses.

3 Experiments and Computations

Two main physical properties differentiate a noncondensable light gas and air: molecular diffusivity and molecular weight. In the aim to investigate the effect of helium on steam condensation, the analysis has been thus divided into two parts: forced convection condensation, focusing on the effect of diffusivity, and natural convection condensation, combining both diffusivity and buoyancy effects. Two different experimental campaigns and computations have been carried out with a multiple purpose: qualify experimental data, validate CFD models and achieve information on the involved phenomena.

3.1 Main features of the CONAN facility

The test section of the CONAN facility consists in a roughly 2 m long, 0.34 m side channel having square cross section, in which a mixture of steam, air and helium is circulated in downward flow. One of the lateral surfaces is cooled on the back side by water. Condensation occurs on the inner surface of this cooled plate, whereas other surfaces are insulated to avoid that condensation occurs over them. The overall condensation rate is measured with an uncertainty of ±1%. Local temperature measurements are available at different locations and depths along and in the thickness of the aluminium plate where condensation occurs, permitting to estimate local heat fluxes (uncertainty ±700W), on whose bases the local mass fluxes are deduced. Tests have been performed having different steam generator power (5 to 25 kW), different mixture velocity (0.5 to 3.5 m/s) and relative
concentrations of the noncondensable gases (0% to 100% of helium in the noncondensable mixture).

3.2 Forced Convection
A useful way for analysing experimental results consists in comparing local experimental Sherwood numbers deduced by measurements to those predicted by the analogy adopting the appropriate correlation, that in turbulent forced convection is the Schlichting’s.

\[
\text{Experimental: } Sh_{0,x} = \frac{m^*}{\rho D_{inm}} \left( \ln \frac{Y_{im,\dd}}{Y_{im,\dd}} \right)^{-1} \\
\text{Analogy: } Sh_{0,x} = 0.0296 \text{Re}^{0.8} \text{Sc}^{0.33}
\]

In Fig 1 (left) the results of this analysis are shown for tests at 20 kW. In forced convection conditions, a remarkable agreement is obtained for Reynolds number higher than \(10^5\). As a matter of fact, for fully developed forced convection condensation the analogy is capable of providing an appropriate description of heat and mass transfer phenomena. Relevant information provided by the analogy and confirmed by experiments (see Fig 1, right) is also that, for a given inlet velocity, helium concentration has a minor effect on the overall condensation rate. The increase of the steam diffusivity and the condensation driving forces are in fact counterbalanced by a reduction of the mixture density, which implies a decrease of the maximum attainable Reynolds and Sherwood numbers. As result, the condensation rate does not change noticeably with helium.

3.3 Buoyancy Effects
In condensing mixtures of steam and air, the fluid close to the interface is heavier than in bulk, since the condensing interface is cold and rich of air. Natural convection regimes are established if buoyancy forces are strong enough to overwhelm inertia forces. In the presence of large quantities of helium, however, the non-equilibrium between interface and bulk density can be reduced. For certain helium concentrations the density difference can be even annealed and forced convection regimes can be thus experienced even at low velocities. This phenomenon has been observed in CONAN tests with helium concentrations in bulk around 62% (see Fig 2 left). While for low concentrations of helium a natural convection regime is experienced (see Fig. 2 right, blue dashed line), for tests at high helium concentration (around 60%), the comparison between experimental asymptotic Sherwood numbers and correlations for forced (Schlichting) and natural (McAdams) convection pointed out the presence of a forced convection regime (black dashed line). The transition between the steam-air natural convection regime and the steam-air-helium mixture convection regime is investigated in Fig 3 (left). Experiments and computations confirm a progressive reduction of the Sherwood number from 0% to 60% of Helium, due to a reduction of buoyancy forces. However, the most interesting phenomena have been observed with helium concentrations larger than 62%, for which an inverse density gradient establishes, since the density at the interface is lower than the bulk one. This phenomenon has been called \textit{buoyancy reversal}, in which buoyancy and inertia forces act in opposite directions. If reversed buoyancy forces are strong enough, a local inversion of the velocity field can be experienced. This phenomenon, named \textit{flow reversal}, is clearly reproduced by the HMTDM model, e.g., for tests at 90% of helium (see Fig 3 right), but experimental data suggest its presence also for tests at lower...
concentrations (see Fig 3 left). Indeed, the phenomenon is associated to a sharp increase of the mass transfer coefficient and the condensation rate. As shown in Fig. 2 (right, red dashed line), a natural convection regime establishes with improved mass transfer characteristics. Detailed computations show that the occurrence of buoyancy reversal and the subsequent flow reversal involves a significant increase in turbulence in the near-wall region, which is suggested as the cause of improved mass transfer with respect to steam-air natural convection.

**Fig 2.** Experimental data of condensation tests at low velocity

**Fig 3.** Buoyancy reversal and flow reversal phenomena in CONAN tests at low velocity

### 3.4 Overall performance of the CFD models

An overview on the performance of condensation models is given in Fig. 4, where the overall condensation rates calculated by both models are compared to the experimental ones. The overall behaviour of the models is satisfactory for both forced convection and low velocity cases. A relatively poorer prediction of local mass transfer rates has been ascertained in the entrance region, which could have caused a slight underestimation of overall condensation rates. Moreover, uncertainties on the boundary conditions can play a relevant role, particularly for the tests at low condensation rates, which are more scattered.

**Fig 4.** Comparison between calculated and experimental condensation rates

### 4 Scaling of condensation tests for hydrogen safety analyses

Basing on the molar formulation of the heat and mass transfer analogy, simple scaling criteria can be found, estimating the ratio of the condensation rates in the presence of
hydrogen and helium in mixtures having the same velocity and temperature boundary conditions. These criteria are only functions of the properties. In forced convection, one gets

\[ \frac{m^*_{c}(H_2)}{m^*_{c}(He)} = \left( \frac{D_m(H_2)}{D_m(He)} \right)^{1/3} \left( \frac{\rho(H_2)}{\rho(He)} \right)^{1/15} \left( \frac{\mu(H_2)}{\mu(He)} \right)^{7/15} \]

Whereas, for natural convection, it is

\[ \frac{m^*_{c}(H_2)}{m^*_{c}(He)} = \left( \frac{D_m(H_2)}{D_m(He)} \right)^{2/3} \left( \frac{\mu(H_2)}{\mu(He)} \right)^{1/3} \left( \frac{\Delta \rho(H_2)}{\Delta \rho(He)} \right)^{1/3} \]

The theoretical correlations proposed above have been tested against the prediction of the HMTDM model, proving the consistency of the analysis (see Fig 5). The results of this analysis allow concluding that mixtures of hydrogen and helium having the same molar concentrations give similar mass fluxes in forced convection. In natural convection, the same phenomena described in paragraph 3.3 for helium tests must be expected, but buoyancy and flow reversal would occur in hydrogen tests for lower molar concentrations of the light gas, since hydrogen is lighter than helium.

![Fig 5. Scaling analysis for force (left) and natural (right) convection condensation](image)

5 Conclusions

Condensation phenomena in the presence of light noncondensable gases have been investigated. The synergy between experimental and computational analyses has allowed consolidating knowledge about forced and natural convection condensation, focusing on peculiar phenomena occurring in the presence of buoyancy effects. Indeed, a significant improvement has been achieved in the understanding of buoyancy and flow reversal effects, made possible by simultaneous experimental and computational investigations.

Two different CFD models have been developed and validated, adopting different modelling strategies and having different purposes. Both models are capable to reproduce correctly the phenomena involved in steam-air-helium condensation tests, confirming also the reliability of experimental results. An experimental database is therefore made available, for the validation of codes. Finally, simple criteria for scaling of condensation tests in the presence of a light gas have been provided, demonstrating the suitability of helium as substitute for hydrogen and thus reinforcing the confidence in present and previous studies.

6 References

MODELLING OF SODIUM BOILING FOR COUPLED NEUTRONIC/ THERMAL-HYDRAULIC TRANSIENT ANALYSIS OF THE GEN-IV SFR

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ABSTRACT

The Generation IV Sodium-cooled Fast Reactor (SFR) is an advanced fast-spectrum reactor concept, for which safety demonstration is one of the key challenges. An important SFR feature in the latter context is the strong interaction, during hypothetical accident events, between the core reactivity and two-phase thermal-hydraulics. In order to enable the accurate simulation of SFR core behaviour under such conditions, a new computational tool is being developed in the frame of the FAST code system at PSI through the coupling of the thermal-hydraulic code TRACE, the reactor kinetics code PARCS and the fuel behaviour code FRED. For this purpose, appropriate extension of the TRACE code is being carried out.

The present paper reports on the development and application of the new calculational tool for transient single- and two-phase flow modelling of the SFR, coupled to 3D neutron kinetics. The first part presents the extension of the non-homogeneous, non-equilibrium two-fluid models, which are available in TRACE for steam-water, to sodium two-phase flow simulation. The conventional correlations for ordinary gas-liquid flows are used as basis, along with optional correlations specific to liquid metal where necessary. A number of new models for representation of the constitutive equations specific to sodium have been implemented and compared with the original closure models. Validation of the extended TRACE version has been carried out via experiments simulating (i) a loss-of-flow accident in an SFR, and (ii) steady-state boiling in tubes and pin-bundles. The first set of experiments enabled one to study the influence of the physical models on the calculated results and to show the capability of the extended code to predict sodium-boiling onset, flow regimes, pressure evolution, dryout, etc. Detailed analysis of the second set of experiments has, in turn, allowed proper choice to be made of correlations for the accurate prediction of pressure losses across fuel bundles, for both single- and two-phase sodium flow. The second part of the paper reports on the integration of the extended TRACE code into the FAST code system and its application to a 3D coupled neutronic/thermal-hydraulic assessment of SFR core transient behaviour. In the latter context, the development and validation of the TRACE and PARCS input models needed for the coupled simulation are addressed. First results are then presented for a specific SFR transient, viz. an unprotected loss-of-flow event with boiling onset.

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1. Introduction

The Sodium Fast Reactor (SFR) is a fast reactor concept proposed by the Generation IV International Forum and is considered, by several countries, as the prime candidate for the large-scale implementation of breeder reactor technology in the medium-term future. Since the void distribution in an SFR core is one of the principal parameters controlling the reactivity during hypothetical transients, and also since boiling may lead to dryout and the melting of core materials, the accurate modelling of sodium two-phase flow is an important requirement in safety studies for the SFR. For this purpose, and under the framework of the FAST project at PSI [1], a new computational tool – involving the extension of the TRACE thermal-hydraulics code [2] – is being developed to enable accurate transient simulation of coupled neutronics/thermal-hydraulic/thermal-mechanic phenomena in the SFR core.

This paper first presents (Section 2) the extension of the TRACE code, which has been carried out in relation to two-phase sodium flow implementation. The main basis for this has been changes made in the logics of the existing models for steam-water two-phase flow. Additional equations-of-state for liquid and gaseous sodium have also been implemented and tested.

For validation of the extended code, numerical simulations have been carried out of (i) steady-state boiling experiments in tubes and pin-bundles [3], and (ii) loss-of-flow (LOF) experiments in an electrically heated, 37-pin bundle in a sodium loop, under conditions close to those of a fast reactor [4]. The experimental data have provided useful insights into the modelling of two-phase sodium flow phenomena with the extended TRACE. The comparison of experimental results with predictions made using the best-estimated set of correlations is presented in Section 3.

The integration of the extended code into the FAST code system aims at a 3D coupled neutronic/thermal-hydraulic assessment of SFR transient behaviour through the coupling of TRACE with the neutronics code PARCS [5]. The development and validation of the TRACE and PARCS input models for the reference Gen-IV SFR is presented in Section 4, together with the first stage of analysis of a hypothetical unprotected loss-of-flow (LOF) event with boiling onset. Finally, conclusions are drawn in Section 5.

2. Extension of the TRACE code to sodium two-phase flow

The two-fluid, six-equation model used in TRACE for steam-water flow representation [2] has been considered an appropriate basis for the current developments for sodium two-phase flow simulation. The approach has been, first, to add all necessary equations-of-state for liquid and vapour sodium to the code, and second, to use as much as possible the internal logics of the TRACE models developed for steam-water flow, replacing where necessary the water-specific correlations by those for liquid metal.

TRACE uses a non-homogenous, non-equilibrium model, considering each phase separately in terms of two sets of conservation equations. Closure of the set is obtained using flow regime-dependent correlations, representing the various transfer mechanisms between phases, as well as between phases and the wall. As the accuracy of the two-fluid model strongly depends on its constitutive equations, these relations have been studied in detail for the sodium two-phase flow analysis [6, 7]. The comparison of selected experimental data (presented in Section 3) with those obtained using different models – implemented as options in the extended TRACE code – have enabled recommendations to be made for sodium two-phase flow modelling. Most of the constitutive equations resulting from these studies are listed in Table 1.

The interfacial area needs to be specified for closure of the set of field equations, specifically to define the heat transfer rate and momentum exchange between phases. A simple correlation, based on the consideration that the annular flow regime dominates in sodium two-phase flow, is used [8]. On the same basis, the Wallis model [8] is considered the most appropriate for the interface momentum exchange. The coefficient of interface heat transfer from liquid to interface is based on the interfacial resistance model [9], and a large value is used for the interface-to-vapour heat transfer in order to keep the vapour temperature close to the saturation value, the interface being assumed to be always at saturation. The correlation used for heat transfer through liquid-phase convection is from Mikityuk [10], and...
the Dittus-Boelter correlation is adopted for single-phase vapour convection. Finally, the axial pressure drop in two-phase flow is correlated to the liquid flow pressure drop via the two-phase flow multiplier \( \varphi \) from Kaiser [7, 11].

<table>
<thead>
<tr>
<th>Interfacial transfer mechanisms</th>
<th>Wall-to-fluid transfer mechanisms</th>
</tr>
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<tbody>
<tr>
<td>Interfacial area: ( a_i = \frac{4 \sqrt{\alpha}}{D_h} )</td>
<td>Wall heat transfer - for liquid phase: ( Nu = 0.047 \left( 1 - e^{-3.8(\frac{P_e}{P_{e,0}} - 1)} \right) \left( P_e^{0.77} + 250 \right) )</td>
</tr>
<tr>
<td>Interfacial momentum exchange: ( c_i = \frac{a_i}{2} \rho g f_i [1 + 150(1 - \sqrt{\alpha})] ), with ( f_i = 0.005 )</td>
<td>- for vapour phase: ( Nu = 0.023 R e_{\varphi}^{0.8} P_e^{0.4} )</td>
</tr>
<tr>
<td>Coefficient of interfacial heat transfer: ( h_{i\varphi} = \frac{2\sigma}{2 - \sigma} \sqrt{\frac{M}{2\pi R}} \frac{h_{i\varphi}^2 M P_e (T_i)}{R T_{2,5}^2} )</td>
<td>Two-phase flow pressure drop multiplier: ( ln(\varphi) = 1.48 - 1.05 ln(\sqrt{X_{LM}} + 0.09 \left( ln(\sqrt{X_{LM}}) \right)^2 )</td>
</tr>
</tbody>
</table>

Tab 1. Constitutive equations currently used in the extended TRACE code for sodium boiling
(The various symbols are defined in the nomenclature section of the paper)

3. Validation against sodium boiling experiments

Two types of sodium-boiling experiments have been considered currently. The first set was performed at the Joint Research Centre, Ispra, Italy, and investigated quasi steady-state boiling and two-phase flow characteristics in one-dimensional geometries [3]. The second set emphasizes the spatial dependency of sodium boiling in a simulated reactor fuel subassembly and was performed at Kernforschungszentrum Karlsruhe (KfK), as part of a series of LOF transient tests using an electrically heated 37-pin bundle [4].

3.1. Steady-state sodium boiling

The most relevant comparison of computed results with data from the Ispra steady-state sodium boiling experiments is with respect to the total pressure drop \( \Delta P \) as function of the velocity at the test section inlet (with specified constant inlet temperature, outlet pressure and applied power). These total pressure drop data effectively serve as a practical tool for checking the applicability and predictive capability of the physical models used in the sodium two-phase flow modelling. It can be seen from Figure 1, which shows the comparison of TRACE predictions with the experimental data for different powers, that the extended code yields satisfactory results for the prediction of the two-phase sodium pressure drop. A sensitivity study of the calculated results to the physical models, in particular the two-phase flow pressure drop multiplier, has been presented in [7].

Fig 1. Two-phase flow pressure drop measurements from the Ispra experiments and comparisons with TRACE
3.2. Loss-of-flow boiling experiments in a 37-pin bundle

Experiments simulating a LOF transient under reactor conditions have been used to study the influence of the physical models on the calculated results and thus further validate use of the extended TRACE code for SFR transient analysis. Because pin-bundle sodium boiling experiments show strong spatial dependence, the use of two-dimensional representations of the 37-pin bundle test section is preferred to one-dimensional calculations. As shown in Figure 2a, the two-dimensional expansion of the boiling region is well predicted by TRACE. Boiling onset is predicted at 6.5s, compared to 6.1s in the experiment. Immediately after the start of boiling, the boiling region grows radially, starting from the central subchannels, and then expands gradually in the axial direction with some flow oscillations. Figure 2b compares the calculated void fraction in the central and peripheral regions of the bundle with the experimental results for the central subchannel. It is seen that boiling inception in the outer subchannel is predicted at 8 s, i.e. 1.5 s after boiling onset at the centre of the bundle. As soon as the boiling front reaches the test section wall, the pressure rises abruptly (see Figure 2c) and the flow decrease is accelerated. These results show the capability of TRACE to predict and reproduce the course of LOF boiling comprising initial local boiling, radial expansion of the boiling region during the first stage of boiling, flow excursion and dry-out. Further results have been presented in [6].

Fig 2. (a) Extension of the boiling region, and (b) absolute pressure and (c) void evolution, in the LOF experiment

4. Coupled 3D neutron kinetics/thermal-hydraulics analysis of SFR

4.1. Development and validation of the input model

Prior to the analysis of SFR core behaviour under transient conditions, the TRACE and PARCS models were first developed in stand-alone mode and then used for coupled calculations within the frame of the FAST code system [5]. The PARCS model [12] is based on macroscopic cross-sections at a reference point and uses derivatives with respect to state variables (for each cross-section type) to account for the feedbacks at any other set of core conditions. The input was prepared using the cell code ECCO (part of ERANOS [13]) for computation of the macroscopic cross-sections, a FORTRAN procedure being applied to obtain the appropriate format. The thermal-hydraulic model developed using TRACE has been limited to the core region, and the thermal-hydraulic and neutronic standalone models have been coupled by an external mapping scheme linking the corresponding nodes appropriately. Finally, coupled simulations were performed to validate the model and obtain steady-state and null-transient solutions for different states of the SFR core. Effective multiplication factors and reactivity feedback coefficients, computed using the coupled PARCS/TRACE model, have been validated against static ERANOS results. Figure 3 illustrates the good agreement obtained for the reactivity feedback coefficients.
Fig 3. Comparison of PARCS/TRACE and ERANOS calculated feedback coefficients for the Doppler effect (pcm), the coolant density (pcm/(g/cm$^3$)), as also the axial and radial expansion effects (pcm/cm).

4.2. Simulation of a hypothetical unprotected loss-of-flow transient

The FAST code system – with the extended TRACE integrated into it – is currently being used to analysis the behaviour of the SFR core during transients in which boiling is anticipated. An unprotected loss-of-flow (ULOF) event has been selected for a first-of-its-kind application of the new tool to coupled spatial neutron kinetics/two-phase thermal-hydraulics. As an illustration of the first results being obtained, Figure 4 demonstrates the capability of the code to predict boiling onset in the hottest channel (located close to the core mid-radius for the analysed core), followed by expansion of the boiling region towards the upper part of the core.

Fig. 4: Distribution of the void fraction in the SFR core 16s after onset of the ULOF

5. Conclusions

An overview has been presented of the doctoral research being carried out in the frame of PSI’s FAST project to enable reliable transient analysis of SFR core behaviour. Thereby, focus is placed on hypothetical boiling events, through the consideration of coupled neutronic/thermal-hydraulic phenomena.

The extension of the thermal-hydraulics code TRACE, originally limited to steam-water flow, to the simulation of sodium two-phase flow, is first addressed. The accuracy of the selected wall-fluid, as well as interfacial, momentum exchange models is demonstrated on the basis of sodium boiling experiments. The comparison of TRACE predictions with the test data have shown the capability of the extended code to accurately predict the most important boiling phenomena, viz. boiling onset, boiling front extension, evolution of pressure and temperatures, and dryout.

The development and validation of a TRACE/PARCS model for the coupled thermal-hydraulics/3D kinetics analysis of an advanced sodium-cooled fast reactor has been briefly presented. The model is currently being used to perform the analysis of hypothetical
unprotected SFR transients in which boiling is anticipated. This is expected to contribute to an in-depth assessment of SFR safety issues and, as such, could potentially lead to recommendations for design changes aimed at improving SFR safety.

Nomenclature

- \( a_i \): interfacial area concentration (m\(^{-1}\))
- \( c_i \): interfacial friction coefficient
- \( D_h \): hydraulic diameter (m)
- \( h \): heat transfer coefficient (W/m\(^2\).K)
- \( h_{fg} \): heat of vaporization (J/kg.K)
- \( M \): molar mass (g/mol)
- \( Nu \): Nusselt number
- \( P_s \): saturation pressure (Pa)
- \( Pe \): Peclet number
- \( Pr \): Prandtl number
- \( P/D \): pitch-to-diameter ratio
- \( R \): universal gas constant
- \( Re \): Reynolds number
- \( T_s \): saturation temperature (K)
- \( X_{LM} \): Lockhart-Martinelli variable

Greek symbols

- \( \alpha \): void fraction
- \( \varphi \): two-phase flow multiplier
- \( \rho \): density (kg/m\(^3\))
- \( \sigma \): coefficient (recommended value: 1.)

Subscripts

- \( g \): gas
- \( i \): interfacial
- \( l \): liquid

References

DEUTERIUM TRAPPING IN CARBON FIBER COMPOSITES UNDER HIGH FLUENCE

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ABSTRACT

The paper is devoted to investigation of deuterium trapping in CFC, dance graphite MPG-8 and pyrolitic graphite (PG) under plasma ion- and electron irradiation. Number of specific features of deuterium trapping and retention under plasma ion and electron irradiation is presented and discussed. In particular it is shown that 1) deuterium trapping takes place even when energy of impinging ions approaches zero; 2) deuterium is trapped under irradiation by plasma electrons; 3) under irradiation at equal fluences deuterium trapping is higher, when ion flux is smaller. High energy ion penetrating the surfaces are trapped in the traps created at the expense of their kinetic energy. The process may be named “kinetic trapping”. Under low energy (smaller than 200 eV) electron and/or ion irradiation the energy of inelastic interaction on the surface provides creation of active centers, which initiate dissociation of deuterium sorbed on the surface, penetration of deuterium atoms into graphite and their trapping in specific low energy traps. The term “potential trapping” is proposed for this type of trapping. Under high energy irradiation such atoms can fill the traps formed through kinetic mechanism. Origination of moveable deuterium atoms from the layer of surface sorption seems to be time dependent process and it is a reason of increase of trapping along with irradiation time. New features of deuterium trapping and retention in graphite evaluated in this study offer new opportunities for analysis and correct estimation of hydrogen isotope trapping and retention in tokamaks having graphite tiles.

1. Introduction

Fuel retention is a crucial issue for tokamaks having plasma facing components made from Carbon Fiber Composite (CFC). It is also important for ITER, because tritium inventory in vessel should be limited. Hydrogen isotope trapping and retention in graphites and CFC have been measured in a number of experiments performed in tokamaks [1] and in laboratory ion beam- and plasma devices [2,3]. At the same time, important parameters and peculiarities of hydrogen trapping under plasma irradiation have not been measured and analyzed properly. Among them there are flux, energy and temperature dependencies of hydrogen trapping, influence of plasma impurity, Mechanisms of hydrogen trapping has not been satisfactory explained.

The paper presents results of investigation of deuterium trapping in CFC, pyrolitic graphite (PG) and MPG under plasma ion- and electron irradiation. Influence of impinging ion energy and flux density on deuterium trapping has been studded. Driving forces and mechanisms of trapping are discussed.

2. Experimental

The experiments were performed in the thermal desorional stand presented in [3]. The stand is intended for sample irradiation by plasma ions and electrons and for measurement of the amount of retained gases by the method of thermal desorption spectrometry (TDS). Deuterium discharge is switched in the electrode system composed from heated cathode, preliminary anode and anode. For annealing and TDS analysis the investigated sample of 10×10×1 mm³ is heated up to 1650 K by thermal irradiation from heated tungsten wire
installed behind its backside. The sample temperature is measured with W – W-Re thermocouple installed in the hole made in lateral surface of the samples. The sample temperature during ion or electron irradiation is \( 450 \pm 10 \text{K} \).

Irradiated surface of PG samples was oriented in parallel to hexagonal planes. The samples of all used materials after machining were cleaned in alcohol in ultrasonic bath and annealed in vacuum.

Negative and positive biasing of the samples provided their irradiation by plasma ions or electrons. Experiments were performed at ion fluence \( 5 \times 10^{23} \text{at/m}^2 \) and at electron fluence \( 1 \times 10^{25} \text{e/m}^2 \). TDS analysis is made with the help of monopole mass-spectrometer of MCX-4 type. The temperature ramp during analysis was equal to 5 K/s. TDS spectrum of \( \text{D}_2 \) was recorded.

3. Results and discussions

3.1 Hydrogen trapping under high energy ion irradiation

Fig 1. Total deuterium desorption from CFC (a) and PG (b) as function of energy of impinging ions and electrons. Difference between desorptions from CFC samples irradiated with different ion flux densities is shown (a).

Figure 2. Spectra of thermal desorption of deuterium as \( \text{D}_2 \) from CFC (a) and PG (b) irradiated with ions \( (j_1 =1 \times 10^{20} \text{at/m}^2 \text{s}) \) and electrons.

Figure 1 shows energy dependence of thermal desorption of deuterium from CFC and PG. The figure 2 presents TDS spectra of deuterium as \( \text{D}_2 \) from CFC and PG irradiated by deuterium ions with different energies.

It is seen (fig.1) that deuterium trapping decreases in both materials when irradiation ion energy drops from 1000 to 200 eV. Difference between the materials is not substantial.
because of development of surface relief and destroying the structure of PG during ion bombardment.

The main peaks of D$_2$ spectra of high ion energy (500, 800, 1000 eV) irradiated samples (HEIS) locate in the temperature ranges 900-1000 K and 1050-1150 K with shoulder between 1200 and 1300 K. In case of high energy hydrogen irradiation of MPG main peaks of H$_2$ spectra also locate in the high temperature region (fig.3), but peak in the range 700-800 K is higher.

![Figure 3. Spectra of thermal desorption of hydrogen as H$_2$ from MPG irradiated with ions (j$_i$ =3.5×10$^{20}$ at/m$^2$s) and electrons](image)

The authors of paper [4] had analyzed a number of papers dealing with measurements of TDS spectra of graphite and CFC and came to conclusion, that positions of the peaks of TDS spectra depended on irradiation conditions, and to some extent reflected the structure of near surface layer of irradiated materials. In particular, they associated the peak at 900-1000 K with the traps created by fast ions inside stopping zone, i.e., in the region destructed and modified by implanting ions. Then, they showed that high temperature peaks and/or shoulders had appeared in the spectra as a result of hydrogen release from the traps in non-destroyed regions inside or behind stopping zone.

Both parts of TDS spectra of D$_2$ are increased along with ion energy (fig. 2). Thus basing on the results of [4] one can say that deuterium trapping in HEIS both in stopping zone and behind them depend on kinetic energy of implanting ions. This type of trapping could be named “kinetic trapping”.

### 3.2 Hydrogen trapping under low energy ion and electron irradiation

In the low energy region (200, 100, 50 eV ) the decrease of deuterium trapping stops (fig. 1). A noticeable desorption from CFC, MPG-8 and PG samples was registered even when they were exposed in plasma under floating potential of approximately 10 eV, or irradiated by low energy (12 eV) electrons.

The main peak of TDS spectra of LEIS (low energy irradiated samples) locates in the temperature range 700-850 K (fig. 2). Difference of the shapes of TDS spectra of LEIS and HEIS indicates that the traps formed in the LEIS differ from those of HEIS. Low energy ions or/and electrons cannot create hydrogen traps through knock out collisions with carbon atoms. Then, analysis shows that in conditions of our experiments only deuterium gas can act as the source for trapping in the electron irradiated samples. Similarity of TDS spectra of both of electron- and of ion irradiated samples allows concluding that trapping in the lasts is also provided by deuterium gas.

To explain peculiarities of deuterium trapping under low energy irradiation, one can propose, that the energy of inelastic interaction of impinging particles with surface provides creation of
active centers, which initiate dissociation of deuterium sorbed on the surface, penetration of deuterium atoms into graphite and their trapping in specific low energy traps. The term “potential trapping” is proposed for this type of trapping. The low energy ions penetrating the surface are trapped in the same traps. Efficiency of low energy trap formation and filling seems to be at its maximum, when low energy ions stop in the near surface region, and decreases, when ion penetration depth increases. Energy of inelastic interaction on the surface provides hydrogen trapping in some other cases. (For example, hydrogen is trapped at the expense of energy of water molecule interaction with graphite exposed at the air [5]). The penetration depth of deuterium ions with energies of some tens of eV is around 1 nm. Deuterium retention under such irradiation is (2-6)×10^{20} at/m^2 (fig.1). It is higher than the amount of carbon atoms (∼1×10^{20} at/m^2) in “flat” stopping zone. To explain this phenomenon one can propose that: 1) the thickness of the trap containing layer gradually enhances and becomes thicker, than ion stopping zone as irradiation time goes on; 2) the retention capacity of the trap containing layer is increased under ion bombardment due to increase of surface roughness: 3) deuterium concentration in the near surface layer far exceeds ratio D:C=0.4 usually considered as a saturation concentration of stopping zones, i.e., some kind of soft hydrocarbon film is formed in near surface layer. The peak or shoulder in the range 500-600 K exists in deuterium spectra of both of LEIS and of HEIS (fig. 2). That is why its nature cannot be established basing on results of these experiments. However the peak was found in TDS spectra of CFC exposed at the atmosphere [5], thus one can assume that relevant traps are formed by potential mechanism.

Some atoms penetrating CFC surface diffuses into the bulk and is trapped there. It is not possible in the perfect structure of PG. As a result the peak at the temperature range 1200-1400 K presents in TDS spectra of CFC and MPG-8 and is absent in similar spectra of PG (fig.2a, 2b and 3).

3.3 Time dependence of hydrogen trapping

Longer exposition of CFC samples in the plasma during irradiation with smaller ion flux density but up to the same fluence leads to retention increase for all investigated ion energies (fig.1). Trapping of impinging ions does not depend on duration of irradiation at the constant fluences. Contrary to that, a probability of deuterium molecule dissociation and penetration the surface by potential mechanism increases when time of deuterium interaction with graphite increases. Thus, one can believe that time dependence of trapping either in LEIS or in HEIS is provided by atoms originated from the layer of surface sorption. Sharp growth of D_2 desorption from LEIS with increase of irradiation time shows that trapping of deuterium of surface sorption constitutes presumable part of entire deuterium trapping. The shape of TDS spectra of HEIS does not change principally when irradiation time increases. It means that deuterium penetrating graphite by potential mechanism is trapped in the traps created by fast ions in the stopping zone. Ions with higher energy create obviously bigger amount of deuterium traps in the stopping zone of HEIS. But the amount of deuterium additionally trapped in longer experiments is practically the same for ions with different energies (fig. 1). Thus, one can believe, that dissociation of deuterium molecules and penetration of deuterium atoms into graphite is the limiting stage for filling the traps in the stopping zone.

3.4 Features of hydrogen isotope trapping in tokamaks having CFC (graphite) tiles

New features of deuterium trapping and retention in CFC (graphite) evaluated in this study, including time dependence of retention, trapping under low energy irradiation, ion/electron activated trapping of neutrals, etc., offer new opportunities for analysis and correct estimation of hydrogen isotope trapping and retention in tokamaks having CFC (graphite) tiles.
In particular, the following considerations about hydrogen retention in tokamaks can be made. Duration of both of tokamak discharges and of entire campaign should be taken into account, when inventory of hydrogen isotopes in graphite tiles is estimated. In particular, deuterium retention capacity of the tiles during long discharges could be higher, than retention capacity during short ones, if irradiation fluences are similar for both discharges. Retention capacity of tokamak tiles is estimated usually basing on the results of laboratory measurements. The present results show, that such a way of estimation is not quiet correct, if impinging particle flux, time of irradiation and neutral gas composition in vicinity of irradiated surfaces are differ in modeling laboratory stand from the tokamak.

4. Conclusion
The paper presents the results of experimental investigation of energy and flux dependences of hydrogen trapping in CFC, PG and MPG. Driving forces and mechanisms of trapping are analyzed.

The CFC and PG samples were exposed to plasmas where incident ion energy was varied by biasing from floating potential from 12 eV to 1 keV. Experiments were performed at fluence $5 \times 10^{23}$ at/m$^2$ for ion flux densities $2 \times 10^{19}$ at/m$^2$s and $1 \times 10^{20}$ at/m$^2$s. Positive biasing was also used for sample irradiation by electrons with flux density $1 \times 10^{20}$ at/m$^2$s at fluence $1 \times 10^{25}$ at/m$^2$. The samples were kept at ~450 K. TDS spectra of D$_2$ was recorded.

Number of specific features of deuterium trapping and retention under plasma ion and electron irradiation is found. In particular: 1) hydrogen trapping takes place even when energy of impinging ions approaches zero; 2) hydrogen is trapped under irradiation by plasma electrons; 3) Under irradiation at equal fluences deuterium trapping are higher, when ion flux density is smaller (Irradiation time is longer).

High energy ion penetrating the surfaces are trapped in the traps created at the expense of their kinetic energy. The process may be named “kinetic trapping”.

Under low energy electron and/or ion irradiation the energy of inelastic interaction on the surface provides creation of active centers, which initiate dissociation of deuterium sorbed on the surface, penetration of deuterium atoms into graphite and their trapping in specific low energy traps. The term “potential trapping” is proposed for this type of trapping. Under high energy irradiation such atoms can fill the traps formed through kinetic mechanism. Origination of moveable deuterium atoms from the layer of surface sorption seems to be time dependent process and it is a reason of increase of trapping along with irradiation time.

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UNDERSTANDING OF HYDRIDING MECHANISMS OF ZIRCALOY-4 ALLOY DURING CORROSION IN PWR SIMULATED CONDITIONS AND INFLUENCE OF ZIRCONIUM HYDRIDES ON ZIRCALOY-4 CORROSION

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ABSTRACT

Zirconium alloys are widely used as fuel claddings in Power Water Reactors. As they represent the first containment barrier to fission products, their mechanical integrity is essential for nuclear safety. During their corrosion in primary water, some of the hydrogen involved in the oxidation reaction with water ingresses into the alloy through the oxide layer. In the metallic matrix, once the solid solution limit is reached at the irradiation temperature, hydrogen precipitates as Zr hydrides mainly located just under the metal/oxide interface due to the thermal gradient across the cladding. As these hydrides may contribute to a larger oxide thickness and to a more fragile behaviour of the cladding, the minimization of hydrogen pick-up is required. Accordingly, since the Zircaloy-4 (Zr-1.3Sn-0.2Fe-0.1Cr) alloy is known to be sensitive to this phenomenon, the understanding of its hydriding mechanism and of the influence of zirconium hydrides on its corrosion behaviour is needed. Regarding the study of the hydriding mechanism, isotopic exchanges were carried out in D\textsubscript{2}O environment at 360° C and led to the localization, in the oxide scales, of the limiting step for the hydrogen diffusion. To estimate an apparent diffusion coefficient of hydrogen in the oxide formed on Zircaloy-4, we firstly based on SIMS profiles and penetration depth of deuterium in the dense part of the oxide film. Secondly, ERDA estimation of the hydrogen content in zirconia and fusion measurements of the hydrogen content in both metal and oxide were used to estimate a hydrogen flux absorbed by the alloy and hence to deduce an apparent diffusion coefficient. Finally, these two methods lead to quite similar values (between 2.10\textsuperscript{-14} cm\textsuperscript{2}/s and 6.10\textsuperscript{-14} cm\textsuperscript{2}/s) which are in accordance with bibliography. Concerning the impact of hydrides on the corrosion of Zircaloy-4, several pre-hydried and reference samples were corroded simultaneously in primary water at 360°C. The characterization of the pre-hydried samples revealed some changes compared with the reference ones, such as the presence of the Zr\textsubscript{2}O sub-oxide at the inner metal/oxide interface, a lower fraction of \(\beta\)-ZrO\textsubscript{2} in the oxide and a faster diffusion of oxygen species through grain boundaries of zirconia (TEM, \(\mu\)-XRD, \(^{18}\)O isotopic experiments). Moreover, during oxidation, the hydrogen initially present in the hydride phase remains in the metallic matrix and leads to the allotropic transformation \(\delta\text{-ZrH}_{1.66} \rightarrow \epsilon\text{-ZrH}_{2}\).
1 Introduction

In Pressurized Water Reactors, zirconium alloys used as fuel claddings are exposed to aggressive aqueous environment (285-325°C, 155 bars, 0.7 to 2.2 ppm Li and 10 to 1200 ppm B). In these conditions, the corrosion kinetics of Zircaloy-4 (Zr-1.3Sn-0.2Fe-0.1Cr) shows a drastic acceleration phase referred as “phase III” which occurs for fuel burnups above 40 GWd/tU [1]. To explain this acceleration, three major hypotheses were proposed by authors. The first advanced cause could be the tin content in the alloy [1] [2] [3]. A second assumption assigned the accelerated corrosion to the impact of the dissolution of Zr(Fe,Cr)\(_2\) precipitates under irradiation [4] [5], whereas, several authors suggested that this enhanced corrosion rate would be rather due to the precipitation of massive hydrides at the metal/oxide interface, in the metallic part of the cladding [6] [7] [8] [9]. Indeed, during their corrosion in primary water, some of the hydrogen involved in the reaction with water ingresses into the alloy through the oxide film and finally precipitates as Zr hydrides heterogeneously located in the underlying metal, just beneath the metal/oxide interface. Many studies tried to understand the nature of the link between the precipitation of massive hydrides in the cladding and its further oxidation rate [6] [7] [8] [9] [10] [11]. However, the extent of the impact of a hydride rim under the metal/oxide interface on the corrosion kinetics is still under debate. In relation with these investigations, several studies tried to determine the hydriding mechanisms through the oxide scale [12] [13] [14] [15] [16] in order to explain the hydrogen pick-up fraction (HPUF) of the Zircaloy-4 cladding, which is between 15% and 20% in PWRs [17].

Presently, nuclear operators wish to enhance fuel burnups up to 60 GWd/tU, and the corrosion and hydriding of Zircaloy-4 claddings have become the limiting factor in the lifetime of the fuel assembly made of this alloy. In this purpose, the understanding of the hydriding mechanisms of the alloy and the impact of hydrides on its corrosion behaviour is required.

2 Understanding of hydriding mechanisms of Zircaloy-4

2.1 Experimental procedures and techniques

The Zircaloy-4 samples were corroded in static autoclaves at 360°C and 18.7 MPa in light primary water (H\(_2\)O, 2 ppm Li, 1000 ppm B). To understand the hydriding mechanisms of these samples, isotopic exchanges of hydrogen were carried out. In a first step, all the samples were corroded with specific times of exposure in order to obtain oxide thicknesses in the pre or in the post transition phase of Zircaloy-4 oxidation kinetics. Then, in a second step, the specimens were corroded in heavy primary water during short or long time exposures, in order to investigate the hydriding mechanisms of the alloy. In parallel, some samples were used to follow the corrosion kinetic behaviour of the alloy. After exposure, some fusion measurements were carried out to estimate the hydrogen uptake of the samples. The experimental procedure and techniques have been detailed in a previous article [18].

2.2 Corrosion tests results

The corrosion behaviour at 360°C in PWR conditions of Zircaloy-4 was measured: the kinetic curves of two samples are presented in figure 1. According to literature data, it is well known that the pre-transition stage of Zircaloy-4 corrosion is characterized by the growth of a dense and protective oxide layer via a diffusion process of oxygen vacancies [19]. The kinetic curve of this oxidation follows a sub-parabolic law shaped as \(x = kt^n\) with an \(n\) parameter included between 0.3 and 0.5 [20]. The experimental points on both samples are reproducible and were fitted according to the following law (dotted line):

\[ x = 0.47t^{0.3} \quad \text{(eq. 1)} \]
with \( x \) representing the oxide thickness (\( \mu m \)) and \( t \) the corrosion duration (days).

![Graph showing corrosion kinetics of Zircaloy-4 samples during corrosion in primary water at 360\(^\circ\) C and 18.7 MPa. The kinetic transition appears around 2 \( \mu m \) and 125 days of corrosion.](image)

When the alloy reaches the kinetic transition, around 2 \( \mu m \), the oxide cracks [21] [22] and becomes porous [23] [24] [25] whereas a new protective film is forming at the interface. Hence, in the post-transition stage, the oxide is divided into two sub-layers: the external one, which presents a lot of defects as pores and cracks, and a quite dense internal layer near the metal/oxide interface. In this post-transition region, a new corrosion cycle occurs and can be fitted with the same previous law (eq. 1).

The hydrogen contents absorbed by the alloy during corrosion were determined, before and after the kinetic transition. The results seem to be in agreement with bibliography with an average hydrogen pick-up fraction of 12%.

### 2.3 Estimation of the apparent diffusion coefficient of hydrogen in the oxide scale at 360\(^\circ\) C

After the corrosion tests, the samples were analyzed using SIMS technique. The hydrogen profile obtained after 50 days of corrosion in light primary water in the pre-transition stage is shown in figure 2. As seen in this figure, the pre-transition oxide is divided into two sub-layers, regarding the hydrogen content: an external one whose thickness is around 300 nm, and an internal one, with a thickness close to 1.3 \( \mu m \). In literature, the same sub-division is reported and associated with the microstructure of the oxide scale. In particular, the outer layer is permeable to hydrogen whereas the inner part of the film is dense [12] [26]. These results were confirmed with a re-oxidation of the sample in heavy primary water during a long exposure.

In the other side of the interface (fig. 2a), in the metallic part, the hydrogen profile indicates a high heterogeneous content which is linked to the hydrogen absorbed during the corrosion process. After the cooling, at the ambient temperature, the hydrogen solubility is very low [27] and its precipitation leads to the formation of zirconium hydrides. The crater image obtained after the abrasion of the sample by the ion beam is presented in figure 2b and highlights the over-concentrated hydrogen zones in the metal.
To estimate the apparent diffusion coefficient of hydrogen in the dense (internal) part of the oxide, the same sample was re-oxidized in heavy primary water during a short exposure so that the isotope did not reach the metal/oxide interface. Indeed, entering a bulk diffusion coefficient and a grain boundary diffusion coefficient in the Cast3m code ($D_b = D_{gb} = D_a$), we tried to describe the experimental curve obtained after the 6 hour-exposure in $D_2O$. An estimation of $D_a$ can be made between $2 \times 10^{-14}$ cm²/s and $6 \times 10^{-14}$ cm²/s, and we showed that it can be considered as equal for the dense part of the pre-transition and the post-transition oxide films.

![Graph and image showing SIMS profiles and crater analysis](image)

Fig. 2. a) SIMS profiles of hydrogen and oxygen obtained after a 50 day-corrosion in PWR conditions of a Zircaloy-4 sample in the pre-transition region. b) Corresponding image of the crater after the SIMS analysis. Bright zones correspond to the high level contents of hydrogen in the metallic part of the alloy.

In complement of SIMS analyses, the Elastic Recoil Detection Analysis (ERDA) technique can be used to quantify the hydrogen contents obtained in the pre-transition and in the post-transition oxides and to estimate, via another way of calculation, the apparent diffusion coefficient of hydrogen in the dense part of the corrosion films formed on Zircaloy-4 alloy. The figure 3 presents the scanning electron micrograph of the post-transition sample (tapered cross-section with a bias angle of 10°) coupl ed with the corresponding hydrogen cartography obtained by ERDA. Based on these data and on the hydrogen content obtained by fusion measurements, the expression of the hydrogen flux (in steady state) absorbed by the alloy during the time of corrosion can be integrated and $D_a$ can be deduced. A more precise demonstration has been proposed in another article [18]. Finally, at 360°C, the apparent diffusion coefficient of hydrogen deduced from the ERDA results in the pre-transition and in the post-transition oxides are in good agreement with the estimation based on the SIMS / CAST3M experiment. These values are similar and validate the hypothesis of a same apparent diffusion coefficient of hydrogen in the oxide, no matter if the kinetic transition is reached or not. Based on these results, an average value of $2 \times 10^{-14}$ cm²/s can be proposed for the apparent diffusion coefficient of hydrogen in oxides grown on Zircaloy-4.
Fig. 3. a) Scanning electron micrograph of a tapered cross-section of the post-transition oxide. b) Corresponding hydrogen cartography obtained by ERDA. Heterogeneous hydrogen distribution in the metallic part of the sample can be associated with the figure 2b and the presence of zirconium hydrides.

3 Impact of zirconium hydrides on Zircaloy-4 corrosion in primary water

3.1 Experimental procedures and techniques

To estimate the impact of zirconium hydrides on Zircaloy-4 corrosion, several samples were cathodically charged with hydrogen in order to precipitate a dense $\delta$-$\text{ZrH}_{1.66}$ phase on the outer surface, which is called “rim”, of the specimens. The hydriding protocol was similar to the one qualified by EDF R&D laboratory [8]. In a second step, these samples were corroded simultaneously with reference un-hydrided samples in light primary water at 360°C in order to compare their oxidation kinetics, their oxide microstructure, and the diffusion of oxygen in the oxide film (through isotopic exchange with $^{18}$O). All the experimental procedures and techniques have been detailed elsewhere [28].

3.2 Impact of zirconium hydrides on oxidation kinetics of Zircaloy-4

The corrosion behaviour at 360°C in PWR conditions of pre-hydrided Zircaloy-4 samples were analyzed and compared with reference specimens (fig. 4). For each group of specimens, the reproducibility of kinetic curves is quite satisfactory and an impact of cathodic charging on the corrosion rate of Zircaloy-4 is clearly evidenced. For the pre-hydrided samples, the impact of the cathodic treatment is expressed through higher $k$ and $n$ parameters in the kinetic law ($0.54$ and $0.38$ respectively). Moreover, the greater oxidation rate of hydrided specimens is also associated with an earlier kinetic transition, both in time and thickness, with transition thicknesses close to 1.5 µm compared with 2 µm for the reference samples, as mentioned earlier.
3.3 Impact of zirconium hydrides on oxide microstructure

For a same oxide thickness in the pre-transition stage, the characterization of pre-hydrided and reference samples revealed no impact of hydrides on morphology and orientation of zirconia grains in the oxide scale, as previously observed [9]. However, on the pre-hydrided sample, a higher density of micro-cracks has been observed inside the oxide film and a new phase indexed as the sub-oxide Zr$_3$O located at the interface between zirconia and the massive hydride phase $\delta$-ZrH$_{1.66}$ has been identified (figure 5).

![Fig. 4](image)

**Fig. 4.** Corrosion kinetics of pre-hydrided and reference Zircaloy-4 samples during corrosion in primary water at 360°C and 18.7 MPa.

![Fig. 5](image)

**Fig. 5.** (a) Micrograph in STEM mode of a pre-hydrided sample corroded 7 days at 360°C. Under the oxide region, a massive hydride phase is still present. (b) Presence of sub-oxide grains between the oxide scale and the hydride phase indexed as Zr$_3$O (c, d, and e) at the metal/oxide interface. Red dotted lines represent the metal/oxide interface.
To precise the impact of hydrides on the oxide microstructure, the corrosion films formed on pre-hydrided and reference samples were analysed using synchrotron microdiffraction. The local X-ray analysis allowed us to build a profile from the internal interface (metal or hydride/oxide) towards the external interface (oxide/water). The results obtained for the pre-hydrided sample corroded in primary water are presented in figure 6. Besides the major presence of the monoclinic zirconia and the δ-ZrH$_{1.66}$ hydride phase, a few diffraction picks confirmed the appearance of the sub-oxide Zr$_3$O and revealed the presence of a new hydride phase indexed as ε-ZrH$_2$ at the internal interface. Indeed, another experiment based on the corrosion in heavy primary water of pre-hydrided and reference samples pointed out, thanks to SIMS analyses, that during the corrosion process, the hydrogen initially present in the sample stays in the hydride phase and does not diffuse towards the oxide film. The accumulation of hydrogen in the vacant tetragonal sites of the δ-ZrH$_{1.66}$ phase could explain the progressive transformation into the new ε-ZrH$_2$ hydride phase.

Finally, a qualitative determination of the polymorphic oxide phases of zirconia, based on the empirical model proposed in [29] was carried out in order to determine the evolution of tetragonal phase through the corrosion film. In both types of samples, a higher fraction of tetragonal phase is observed near the internal interface, which confirmed previous results [30]. However, on the pre-hydrided sample, whatever the distance from the interface being considered, a lower proportion of tetragonal zirconia is always observed compared with the reference sample.

![µ-DRX profile obtained on a pre-hydrided sample corroded in light primary water at 360°C (oxide thickness of 7.4 µm).](image)

**3.4 Impact of zirconium hydrides on oxygen diffusion through the oxide scale**

To complete this study, the impact of zirconium hydrides on oxygen diffusion through the oxide scale was investigated using the isotopic exchange with $^{18}$O. The pre-transition and the post-transition phases were explored, with short and long time exposures [28] in order to compare the oxidation mechanisms between the pre-hydrided and the reference samples. In the pre-transition stage, after the exposure of 6h and 24h in H$_2$-$^{18}$O, a deeper penetration of $^{18}$O is observed in the oxide scale formed on the pre-hydrided samples, leading to a twice
higher diffusion coefficient of oxygen (2.4.10^{-15} \text{ cm}^2/\text{s} compared with 1.3.10^{-15} \text{ cm}^2/\text{s} for the reference sample, with this last result in accordance with bibliography [31]).

Moreover, in both kinetic domains (pre- and post-transition), a higher concentration of \textsuperscript{18}O in the fast diffusion paths (grain boundaries, probably) can be observed on the pre-hydrided samples, as well as a faster enrichment of \textsuperscript{18}O near the internal interface, which is the sign of a higher part of diffusion through the grain boundaries in corrosion films formed on hydrogen-containing samples [32]. A model based on a different structure of grain boundaries in zirconia formed on the pre-hydrided and the reference samples has been proposed [33].

4 Conclusion

Finally, the results mentioned above are useful to clarify the hydriding mechanisms of Zircaloy-4 during corrosion in reactor. Chronologically, once the hydrogen reaches the solid solution limit, zirconium hydrides precipitate in the metallic matrix and our results clearly indicate that they may impact the behaviour of Zircaloy-4 corrosion. Even if the potential impact of other factors proposed in literature can not be excluded, the precipitation of massive hydrides takes a real part in the appearance of the “Phase III acceleration” observed in PWRs.

5 References
