TRANSACTIONS

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STATUS REPORT OF THE UTILIZATION OF RESEARCH REACTORS IN THE NEW AND CANDIDATE MEMBER STATES OF THE EU

ISTVÁN VIDOVSZKY
KFKI Atomic Energy Research Institute
H-1525 Budapest, POB 4, Hungary

ABSTRACT
The nuclear profile of Europe changed in 2004, when the Central and Eastern European countries joined the EU. New types of NPPs and research reactors appeared. These reactor types are not unknown for the nuclear community, as e.g. papers describing them were presented on previous RRFM meetings. Even some of the related nuclear programs have been presented, however an overview of these research reactors as well as of the programs related to these reactors could be of interest. Four of the ten new member states (Czech Republic, Hungary, Poland, Slovenia) operate research reactors, further two (Latvia and Slovakia) operated research reactors in the past. Two candidate member states (Bulgaria and Romania) operate research reactors as well. The paper describes the programs of research reactor utilization in the mentioned countries and tries to give insight into the future perspectives as well.

Introduction
Four of the ten new member states and two candidate member states operate research reactors. The paper describes the research reactors in these six countries and the research and application programs related to them. One has not to forget about the countries, where research reactors have never been operated. Lithuania is one of them, and one can mention, that more than 70% of its electricity is produced by NPPs. However for the audience of RRFM the countries, where research reactors are in operation are of main interest, so further I will concentrate on them.

Bulgaria
The Institute of Nuclear Research and Nuclear Energy (INRNE) operated the only research reactor of the country. INRNE belongs to the Bulgarian Academy of Sciences. The IRT-2000 reactor was first put into operation in 1961. The power of the reactor was 0.5 MW that time, it was upgraded to 1 MW in 1965 and to 2 MW in 1970. The reactor was designed in the former Soviet Union, who supplied the fuel as well. During the period 1982 – 86 there was a project, supported by the Kurchatov Institute (Moscow) to upgrade the reactor to 5 MW. The project was terminated at the time of the Chernobil accident. Later the future of the reactor was uncertain. In 2001 the government of Bulgaria has taken a resolution for the refurbishment of the reactor into a low power reactor up to 200 kW. The reconstruction project is performed with international cooperation (Kurchatov Institute, Skoda, SCK CEN, etc). The reactor is currently in extended shut-down.

The scope of the reconstruction is:
- new seismic/safe adjacent building for reactor operations,
- modification of existing reactor hall,
- addition of a new reactor vessel and replacement of reactor internals,
- new primary cooling system and reconstruction of secondary cooling system
- new reactor protection system and I&C system,
- new radiation monitoring system,
- new electrical distribution system including two diesel generators.
The proposed new utilization of the reconstructed reactor is: education (students in nuclear engineering, physics, biology, radiation protection), training of NPP operators, neutron activation analysis, production of radioactive isotopes, boron neutron capture therapy (BNCT), testing of instruments and calibrating standards.

**Czech Republic**

There are three research reactors in the Czech Republic. Additionally in ŠKODA Works, at Plzeň there was a test reactor, which is already dismounted.

The LVR-15 in Řež near Prague is the most significant research reactor of the country. LVR-15 is operated by the Nuclear Research Institute (NRI). NRI is partly owned by ŠKODA. LVR-15 was first put into operation in 1957. The power of the reactor was 2 MW that time; it was upgraded to 4 MW in 1964. The fuel is supplied from Soviet Union, later Russia. The first fuel (1959 – 1974) of LVR-15 was the 10% enriched EK-10. In 1974 during a reconstruction the fuel was changed for 80% enriched IRT-2M. This time the primary heat exchangers were replaced by two 9 MW units. The last reconstruction started in 1988 and was finished in 1991. During this reconstruction all structures and components were replaced except the heat exchangers. The designed thermal power of the reactor is 15 MW, but the licensed power level is 10 MW. In 1995 – 98 the reactor was converted to 36% enrichment, the type of fuel assemblies remained IRT-2M. The reconstructed reactor was designed for 30 years, so the operation is foreseen up to 2018.

The reactor is used for:
- material testing experiments of reactor pressure vessels at water loops and in irradiation rigs,
- activation analysis using a pneumatic rabbit system,
- experiments at beam ports in the field of basic and applied physics,
- irradiation for medical and radio-pharmaceutical purposes,
- irradiation of silicon monocrystals,
- BNCT at the former thermal column.

The LR-0 low power reactor is also operated by NRI in Řež. LR-0 serves reactor physics and related nuclear fields. It was critical for the first time in December 1982, regular operation started in 1983. The fuel of LR-0 is LEU, the shortened variant of VVER type NPP fuel, with the only difference in length of the fuel rods. The maximum allowed power is 5 kW, but the reactor is usually operated on much lower level. The core can be changed in a very flexible way, so different core configurations can be investigated.

At the Czech Technical University in Prague, at the Faculty of Nuclear Sciences and Physical Engineering (FNSPE), a 1 kW reactor, the VR-1, also called VRABEC (Sparrow) is in operation. The reactor was first critical on December 3, 1990. The reactor was designed by Chemoproject Praha, based on calculations and analyses performed at FNSPE in cooperation with ŠKODA Plzeň. The major components of the reactor were manufactured by ŠKODA Plzeň. The reactor has been in permanent operation since January 1992. The fuel used is IRT-3M, enrichment 36%, production of Russia. The 1 kW reactor can be operated at 5 kW level for short periods. The reactor is used in the following areas: in the education of technical university students, in scientific research, in respecting reactor parameters and requirements of the so called clean reactor core (free from a major effect of the fission products) in the education and training of specialists in the power industry, in information science and in promotional activities in the field of nuclear power.

The operation of Sparrow is foreseen for the long term.

**Hungary**

Hungary has two research reactors.
The Budapest Research Reactor (BRR) is operated by the KFKI Atomic Energy Research Institute (AEKI). AEKI belongs to the Hungarian Academy of Sciences. BRR was first put into operation in 1959. The power of the reactor was 2 MW that time, it was upgraded to 5 MW in 1966 and to 10 MW in 1992. The reactor was designed in the former Soviet Union. The upgraded, reconstructed reactor (1992), which can be considered as a new reactor from the technical point of view, was entirely designed and built by Hungarian companies. The fuel is supplied from Soviet Union, later Russia. The first fuel (1959 – 1966) of BRR was the 10% enriched EK-10. From 1966 the 36% enriched VVR-SM later VVR-M2 fuel is used. A beryllium reflector was added to improve neutron economy during the 1966 reconstruction. The spent fuel of the entire operation is still kept at the site in controlled water pools. The older fuel (used before 1986) is canned in aluminium capsules filled with inert gas (Nitrogen), the more freshly irradiated spent fuel is in contact with water. The water quality of the storage pools is continuously checked, no corrosion problems occurred. BRR is equipped with a cold neutron source (CNS). The CNS was put into operation in 2001. BRR is operated for about 4000 hours annually. The reconstructed reactor was designed for 30 years, so the operation is foreseen up to 2023. There is sufficient fuel for 4 – 5 years of operation. The further fuel supply is ensured by a long term agreement with the vendor. The conversion for LEU is considered, it can be done within the coming few years. The 20% enriched VVR-M2 type fuel will be supplied by the the same vendor as the earlier variant was.

BRR is used for research and for practical purposes as well. The reactor is acknowledged as a center of excellence by the EU. EU supports the work of scientist from member states by covering the costs of 20% of the operational hours. The main fields of research are: material studies, neutron activation analyses, prompt gamma activation analysis, time of flight spectrometry etc. The practical applications include neutron radiography (e.g. investigations of helicopter rotors), radioisotope production (mainly for medical purposes), neutron activation analysis (e.g. identification of traces for the police).

The training reactor of the Budapest University of Technology and Economics is operated by the Institute of Nuclear Techniques (INT) of the Faculty for Science. The reactor was first put into operation in 1971. The power of the reactor was 10 kW that time, in 1986 it was upgraded to 100 kW. The reactor was designed in AEKI, the fuel is of the EK-10 type (the spare fuel of the first period of BRR is used). There is no spent fuel there, as the original core is still in use. The reactor is foreseen for long term use.

The main use of the reactor is training. Both undergraduate and graduate students are trained at the reactor. The reactor is also open for students from abroad (e.g. France, Slovakia, Czech Republic). INT regularly organises base-level and advanced-level courses in isotope-techniques and radiation protection. INT organises the continuing education 2-years course for nuclear reactor engineers too. A large proportion of the Hungarian nuclear experts were formed by these courses.

A radiochemical laboratory is operated at the reactor. The research work is concentrated in reactor physics, radio-chemistry, noise diagnostics, health physics and nuclear electronics.

Poland

Poland had two research reactors EWA and MARIA, both on the site of the Institute of Atomic Energy in Swierk near Warsaw. EWA was similar to the Budapest Research Reactor, it used first EK-10, later VVR-SM fuel, it was shut-down earlier, MARIA is still in use. The biological shield of the former EWA reactor is foreseen for the interim storage of spent fuel.

The MARIA research reactor was first put into operation in 1974. In 1985 it was shut down for modernization. Since 1993 MARIA is back in operation. In January 2004 the reactor was temporarily shut-down due to the lack of fuel. Solving of the fuel problem is in progress, so the restart of MARIA can be expected in the near future.

MARIA is a pool type multipurpose research reactor moderated by water and beryllium (as a matrix, in which the core is inserted) with graphite reflectors. The fuel elements are of VVR-C type, Russian
(Soviet) origin. The fuel elements are of the concentric cylindrical type, they are located inside pressurized (approx. 1.8 bars) fuel channels. Due to this design the reactor has two totally independent primary systems, one for the pressurized channels and the other for the reactor pool. The fuel enrichment was originally 80%, in the period 1999-2001 the reactor was converted to 36% enrichment.

The main use of MARIA is isotope production for medical and industrial purposes.

**Romania**

Romania had two research reactors, one in Magurele near Bucharest, one in Pitesti. The reactor in Magurele was similar to the Budapest Research Reactor in its early phase, i.e. EK-10 fuel was used at 2 MW power. It was shut-down earlier while the other reactor in Pitesti is still in use.

The dual core TRIGA ROMANIA consists of two reactors. The steady state reactor is a 14 MW research reactor fuelled by mixed HEU and LEU. The reactor is reflected by beryllium and placed in light water pool. The annular core pulsed reactor is a 500 kW reactor, fuelled by 20% enriched 12U-Zr-H fuel, placed in the same pool. The maximum power of the pulses is 20000 MW with a minimum period of 1.2 msec, pulse width 4.6 msec half pulse. Despite of the excellent facilities for material tests, isotope production and beam experiments available at the reactor, there is a lack of utilization requests causing a reduced use of the facility.

Up to now the facility was used for:
- tests on pressure tubes from CANDU reactor;
- neutron activation analysis on different probes from Cernavoda NPP (lubricants, resins, etc.);
- radioisotope production for pharmaceutical and industrial use,
- neutron radiography for fuel rods and for different other devices,
- training and assistance for Cernavoda NPP.

In the future
- determination of B and Gd concentration (in the moderator from Cernavoda NPP),
- development of prompt gamma activation analysis,
- neutron diffractometry;
- new technologies for radioisotopes production,
- test on Slight Enriched Uranium fuel for the NPPs;
- reactivity insertion accident test on spent fuel from Cernavoda NPP could be of interest.

Despite the high level of knowledge and capacity in the institute and the good conditions of the reactor and of associated facilities, the extremely reduced stock of LEU fuel elements and the highly increased price of the nuclear fuel in the market, is preventing for the development of an efficient utilization program.

The reactor is currently continuing the core conversion started in the year 1992, and started to send the spent HEU back to the United States.

**Slovenia**

The Josef Stefan Institute (IJS) in Ljubljana operates the only research reactor of the country. The TRIGA MARK II reactor is of the well-known TRIGA type. The reactor was first put into operation in 1966. The reactor was delivered by GA, the reactor tank and body were built by Slovenian companies. In 1991 it was reconstructed and equipped for pulse operation. The power of the reactor is 250 kW.

219 spent fuel elements of the reactor were shipped for final disposal to the United States in 1999. The reactor has sufficient fuel for further operation.
The reactor has been playing important role in developing nuclear technology and safety culture in Slovenia. The most significant role the reactor played in training, as practically all nuclear professionals in Slovenia started their career or attended practical training courses at the TRIGA MARK II reactor. All NPP Krsko reactor operators and other technical staff pass training courses on the reactor. On-the-job training for IAEA trainees from developing counties (in average two per year) is important as well. Research is mainly related to neutron activation analysis. The production of radioactive isotopes is also significant both for medical and industrial (including the needs of Krsko NPP, e.g. for turbine testing by radioactive sodium) purposes.

Addition: Former German Democratic Republic

In the eastern part of Germany, on the territory of the former German Democratic Republic in Rossendorf near Dresden a research reactor very similar (same fuel, same power) to the Budapest Research Reactor was in operation from 1957 to 1990. After the unification of Germany the rules did not allow the Rossendorf Reactor to be in use. The significant amount of fresh fuel and a few other devices of the reactor (e.g. the semi-automatic target loading device for radioactive isotope production, called DORA) were offered to potential users. AEKI purchased these items by a symbolic price of 1 DM (approximately 0.5 €) each shipment. So the Rossendorf Reactor survives, at least partly in Budapest.

Summary

The described reactors differ quite significantly from each other, there are well known types among them (TRIGA in Romania and Slovenia), the majority is of Russian origin with Russian fuel (EK, IRT, VVR). The backgrounds and present utilizations are different as well. All these reactors are foreseen for further use, however technical problems (supply etc.) may hinder the utilization in some cases. The need for research reactors is significant in all these countries.
THE INTERNATIONAL PROJECT ON INNOVATIVE NUCLEAR REACTORS AND FUEL CYCLES (INPRO): GENERAL DESCRIPTION AND IMPLICATIONS FOR THE RESEARCH REACTOR INFRASTRUCTURE NEEDED FOR R&D

YURY A. SOKOLOV
Head, Department of Nuclear Energy
International Atomic Energy Agency
1400 Vienna, Austria

ABSTRACT
The substantial growth in 21st century energy supplies needed to meet sustainable development goals has been emphasized by UNCSD, WSSD, IPCC and others. This will be driven by continuing population growth, economic development and aspiration to provide access to modern energy systems to the 1.6 billion people now without such access, the growth demand on limiting greenhouse gas emissions, and reducing the risk of climate change.

A key factor to the future of nuclear power is the degree to which innovative nuclear technologies can be developed to meet challenges of economic competitiveness, safety, waste and proliferation concerns. There are two major international initiatives in the area of innovative nuclear technology: the IAEA’s International Project on Innovative Nuclear Reactors and Fuel Cycle (INPRO) and the Generation IV International Forum. With INPRO some scenarios of future energy needs were identified and the methodology for holistic assessment of the innovative nuclear energy systems (INS), which can be developed to meet these scenarios, was developed. The current status of the INPRO project and details of the INPRO methodology will be reported.

The research needs identified due to Agency’s activities on innovative nuclear system development assume the use of research reactors. The areas crucial for the development of INS which critically dependent of the RR experiments and following requirements addressed to the RR will be discussed. These areas include the development of advanced fuel and core materials for proposed innovative power reactor concepts.

1. Nuclear power: 21st century vision

At the outset of the 21st century, the Millennium Development Goals and the Johannesburg Plan of Implementation have both set global objectives for sustainable development (SD) that give high priority to the eradication of poverty and hunger, universal access to plentiful fresh water and energy, and environmental sustainability. These goals have been established by governments.

Several plausible energy development scenarios to meet these SD goals have been developed. One is the ‘A1T Scenario’ of the Special Report on Emissions Scenarios (SRES) published in 2000 by the Intergovernmental Panel on Climate Change [1]. The A1T Scenario assumes strong positive advances in international cooperation, rapid technical progress and a low global population trajectory.

A second scenario is the ‘SD Vision Scenario’ published in 2003 by the OECD International Energy Agency (IEA) in Paris. The SD Vision Scenario predictions are very similar to those of the A1T Scenario. However, it is less technologically optimistic in that it assumes that activist government policies will be needed to push the world beyond business-as-usual trends. It therefore assumes among other things access to affordable electricity by at least 95% of the world’s population by 2050.
Figure 1 shows the resulting primary energy supply mix through 2050.

For nuclear energy the significant result is that both these SD scenarios predict global growth in nuclear energy by a factor of 14.

In contrast, medium-term (business-as-usual) IAEA projections for nuclear power (NP) are much lower, as shown in Figure 2.

The IAEA’s low projection (dark green bars) assumes no new NPPs beyond what is already being built or firmly planned, plus the retirement of old NPPs on schedule. The high projection (light green bars) takes into account additional reasonable new proposals. The medium term projections in 2030 are 2-3 times lower than in the SD scenarios.
To succeed in future markets and close this gap, nuclear energy will have to innovate. Innovation, in turn, requires R&D, and R&D needs are different for developed and developing countries. Developing countries, for example, are expected to have a greater interest in small and medium sized reactors (SMRs), which may be better suited to their generally smaller national electrical grids and their generally smaller financial resources. Such divergence in R&D needs necessitates broad cooperation and coordination in R&D across national borders, and this requirement is further reinforced when additional nuclear applications are added to the mix – for seawater desalination or hydrogen production. Research reactors, both in developed and developing countries, can contribute significantly to the necessary international cooperation and to innovation development.

2. International cooperation and innovation development

Currently there are two principal multinational initiatives promoting such cooperation. One is the U.S. initiated Generation IV International Forum (GIF). The other initiative is the IAEA’s International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO). It is open to all IAEA Member States and currently has 22 members. INPRO has the following overall objectives:

- To help to ensure that nuclear energy is available to contribute in fulfilling, in a sustainable manner, the energy needs of the 21st century.
- To bring together all interested Member States, both technology holders and technology users, to consider jointly the international and national actions required to achieve desired innovations in nuclear reactors and fuel cycles that use sound and economically competitive technology; are based, to the extent possible, on systems with inherent safety features; and minimize the risk of proliferation and impacts on the environment.
- To create a process that involves all relevant stakeholders that will have an impact on, draw from, and complement the activities of existing institutions, as well as ongoing initiatives at the national and international level.

INPRO’s initial focus has been on identifying the prospective needs of future NPP buyers and developing ‘user requirements’ in the areas of economics; sustainability and the environment, including waste management; safety; proliferation resistance; and infrastructure development.

The INPRO methodology which was documented in 2003 in IAEA-TECDOC-1362, Guidance for the evaluation of innovative nuclear reactors and fuel cycles [2] was then tested and upgraded through several case studies, as documented in 2004 in IAEA-TECDOC-1434, Methodology for the assessment of innovative nuclear reactors and fuel cycles [3]. The methodology is now ready to be applied in the assessment of innovative nuclear system (INS) in national and multinational studies.

3. INPRO methodology

The objective of sustainable energy development is comprehensively integrated into the upgraded INPRO methodology. The methodology thus ensures that the assessment of a given INS takes into account all four dimensions of sustainability, i.e., economic, environmental, social and institutional. Such an assessment can then be used to help define an overall research strategy, including short-, medium- and long-term research, development and demonstration (RD&D) plans. The definition of an INS in INPRO includes both evolutionary advanced systems with moderate modifications that use proven solutions and truly innovative designs that incorporate radical conceptual changes to achieve performance breakthroughs in selected areas.
The INPRO methodology can be used:

- to screen an INS for its compatibility with the energy needs of the 21st century and sustainability considerations;
- to compare different INSs; and
- to identify the RD&D required to improve and validate the performance of an INS.

The assessment must include in the evaluation all components of the system to achieve a holistic view and ensure that the overall system is sustainable.

INPRO has defined a set of basic principles, user requirements, and criteria (with each criterion consisting of an indicator and an acceptance limit). The highest level in the INPRO structure is a basic principle (BP), which is a statement of a general rule that provides broad guidance for the development of an INS. Basic principles are formulated for all areas: economics, safety, environment, waste management, infrastructure and proliferation resistance. For each basic principle there are then several user requirements (UR), which are the conditions that should be met to achieve acceptance of a given INS by investors, designers, plant operators, regulatory bodies, local organizations and authorities, national governments, NGOs, the media, and the end users of energy (e.g., the public and industry). The INPRO methodology thus seeks to ensure that each INPRO assessment takes into account the interests and views of all stakeholders. Finally, to determine whether and how a given UR is being met criteria (CR) were formulated, with each criterion consisting of an indicator (IN) and acceptance limit (AL).

For some ALs, INPRO has proposed specific values, e.g., in the area of safety where limits are internationally accepted and applied. In the long term, it is expected that internationally agreed ALs will be proposed also in the areas of proliferation resistance, environment, and waste management. An INPRO manual is under preparation that will provide IAEA Member States more detailed information on the selection of indicators and acceptance limits.

In the area of economics there is a single BP, specifically that energy from an INS has to be affordable and available. This translates into several URs. First, the price to the consumer must be competitive relative to alternative systems, the INS must be able to compete successfully for investment, and the investment risk must be acceptably low. It is recognized that government policies (e.g., to assure energy supply security) will have a significant influence on investor decision making in all countries and that in some countries governments may participate directly in such investments. All costs must be taken into account, including capital costs, operating and maintenance costs, and fuel costs (which reflect the capital and operating cost of mines, conversion, enrichment, fuel fabrication, and reprocessing) and the costs of decommissioning and long-term waste management. There is also an economic UR requiring sufficient flexibility to accommodate market changes and growth.

In the area of safety the BPs and URs developed within INPRO are based on an extrapolation of current trends. The fundamental safety functions are to control reactivity or sub-criticality, to remove heat from the core and decay heat, and to confine radioactive materials and shield radiation. To ensure that an INS will fulfill these fundamental safety functions, INPRO has set out BPs that the levels of protection in defence-in-depth shall be more independent from each other than in existing installations, that the emphasis on inherently safe characteristics and passive systems shall be increased, that the risk from radiation exposures shall be comparable to that of other similar industrial facilities, and that the knowledge of plant characteristics and the capability of analytical methods used for design and safety assessment of INS shall be at least within the same confidence level as for existing plants.

The URs address the prevention, reduction and containment of radioactive releases to make the health and environmental risk of an INS comparable to that of industrial facilities used for similar purposes. Thus for an INS there will be no need for relocation or evacuation measures outside the plant site, apart from those generic emergency measures developed for any industrial facility. RD&D must be carried out before deploying INSs using, e.g., large-scale engineering test facilities including, possibly, research reactors, pilot plants and prototype plants to bring the knowledge of plant characteristics and the capability of codes used for safety analyses to the same level as for existing plants. Safety analyses will involve a combination of deterministic and probabilistic assessments.
Protection of the environment is a central theme within the concept of sustainable development. Nuclear power has the potential to support sustainable development by providing much needed energy with a relatively low burden in terms of the atmosphere, water resources, and land use.

INPRO has two BPs dealing with the environmental impacts of nuclear energy. These translate into a number of URs addressing the following points. All potential adverse effects of the nuclear fuel cycle must be prevented or mitigated effectively. Both radiological and non-radiological effects need to be considered. An INS should use all resources (water, fuel, materials, land, etc.) at least as efficiently as acceptable alternatives. The environmental performance of a proposed technology must be evaluated as an integrated whole, taking into account the likely environmental effects of the entire collection of processes, activities and facilities in the energy system.

Waste management is especially important. Because it involves longer time scales than other environmental impacts and, in many cases, different source terms and pathways, INPRO considers waste management separately. The starting point is the nine fundamental principles for radioactive waste management already established by the IAEA [4]. These translate into four BPs: that the generation of radioactive waste shall be kept to the minimum practicable; that radioactive waste shall be managed to protect human health and the environment that waste management will not place undue burdens on future generations; and that waste generation and management must be optimized for operational and long-term safety. These principles in turn lead to user requirements: to minimize long-lived wastes that would be mobile in repository environments; to limit exposures to radiation and chemicals from waste; to specify permanently safe end states for all wastes and to move wastes to these end states as early as practical; to classify wastes and to ensure that intermediate steps do not inhibit or complicate the achievement of the end state; to ensure that the accumulated liability at any stage of the life cycle is covered. A number of specific areas of needed RD&D are identified, including the partitioning and transmutation of long-lived fission products and minor actinides.

Recognizing the potential of all nuclear energy systems to be misused for the purpose of producing nuclear weapons, INPRO provides guidance on incorporating proliferation resistance into INSs. Proliferation resistance is a combination of intrinsic features and extrinsic measures. Intrinsic features are those that result from the technical design of INS and reduce the attractiveness of nuclear material for misuse during production, use, transport, storage and disposal. Intrinsic features include material characteristics such as isotopic content, chemical form, bulk and mass, and radiation properties. Intrinsic features can also prevent the diversion of nuclear material by limiting the points of access to such material, by facilitating detection and by preventing the undeclared production of direct-use material. This includes reactors designed to prevent undeclared target materials from being irradiated; reactor cores with small reactivity; and fuel cycle processes that are difficult to modify. Extrinsic measures are based on States’ commitments, obligations and policies of States; as well as on agreements between exporting and importing States on the exclusive use of nuclear energy systems for agreed purposes; on arrangements that control access to nuclear material and technology; on verification measures and on legal and institutional measures to address any violations of the measures defined above.

INPRO’s BPs and URs require that proliferation resistance features and measures be implemented throughout the full life cycle of an INS and that both intrinsic features and extrinsic measures be utilized, with neither being considered sufficient by itself. The URs require that: the commitment and obligations of States be adequate; the attractiveness of nuclear material with respect to its suitability for conversion into nuclear explosive devices be low; the diversion of nuclear material be difficult and detectable; that multiple features and measures be incorporated in an INS covering plausible acquisition paths of fissile material for a nuclear weapons programme; and that the combination of intrinsic features and extrinsic measures be optimized during design and engineering to provide cost-effective proliferation resistance. RD&D is needed in a number of areas, in particular, in developing a process to assess the proliferation resistance of a defined INS.

The maturity of the nuclear power infrastructure is an important factor for the successful deployment and operation of nuclear power plants. Globalization and the importance of developing countries in future world energy markets point to the need to adapt infrastructures, including legal, institutional, industrial,
economic and social features and substructures to facilitate the deployment of nuclear power systems in
developing countries.

In a world with a growing need for sustainable energy, the harmonization of regulations and licensing
procedures could facilitate the application of nuclear technology. For developing countries the existence
of regional or international licensing and regulatory mechanisms should not require a highly developed
nuclear knowledge base and infrastructure. International licensing and regulatory mechanisms shall make
it possible to adopt an INS for the supply of energy without making an excessive investment in national
infrastructure. Nonetheless, the associated URs recognize the need to establish an adequate national legal
framework, the importance of an industrial and economic infrastructure adequate to the task of installing
an INS, the necessity of adequate measures being taken to secure public acceptance, and the necessity of
adequate human resources. Globalization, of course, brings with it the opportunity to draw on a much
broader pool of resources across the many disciplines of science and engineering associated with the
range of technologies relevant to nuclear energy systems.

In performing an INS assessment, the assessor must base the analysis on a reference national energy
scenario, but must also take into account global and regional demands for uranium, fuel production,
reprocessing capacity, etc. The assessment must thus use elements of regional or global scenarios and
take advantage of modelling tools that can calculate the resources, both financial and material, required
for a given combination of reactors to meet a specified supply of nuclear energy as a function of time.
Such modelling tools help assess the practicality of a proposed system in terms of material balances, such
as uranium demand, waste arisings, plutonium re-cycling, etc. The further development and application of
such modelling tools is seen to be an important part of energy planning and of INPRO, and the use of
such tools will be integrated into the INPRO methodology as it is further developed.

Finally, an INPRO manual is currently being completed, which will provide a detailed description of
assessment methods in the different INPRO areas.

4. INPRO: Further steps

The INPRO methodology, as described in IAEA-TECDOC-1434, is now ready to be applied for the
assessment of INSs in national and multinational studies. Several INPRO members have begun specific
applications as follows.

a) A joint assessment of an INS based on a closed fuel cycle with fast reactors. The partners for this study
are China, France, India, the Republic of Korea and the Russian Federation with Japan as an observer.

b) Studies of transitions from current LWRs to Generation 3+ with fast neutron reactors and an advanced
fuel cycle in France.

c) An analysis of the introduction of a block of 700 MWe for nuclear electricity production in countries
with limited demands, based on either the ACR700 or CAREM300, in Argentina.

Several studies are also planned in India and other INPRO countries. All these activities are expected to
identify possible frameworks and options for collaborative RD&D for INS that might be performed in
later phases of INPRO, and these include also research tasks that would make use of available research
reactors.

5. Research needs for INS development

Future stages of INPRO will involve the selection of promising INS concepts and the identification of key
R&D needs. Developing and validating such INSs will require that their key features are demonstrated in
reasonable smaller experiments. The certification of all new materials, fuel assemblies and reactor cores
will require a substantial amount of irradiation testing in dedicated research reactors, including in special
loops. While the INPRO methodology for the assessment of INSs does not indicate non-traditional
research directions beyond fuel certification, structural material (SM) development and integrated tests
(including tests of safety aspects of failures), nevertheless the holistic consideration of INSs that is
implemented in the INPRO methodology, incorporating sustainability, economics, safety, environment, and non-proliferation perspectives, adds the requirement that all goals be achieved harmoniously for guaranteed performance. The sustainability requirements emphasize, with practical unanimity, the relevance of fast reactor technology with a closed fuel cycle, which will definitely create a high demand for research reactor experiments. In addition to the traditional characterization of fuel and structural materials from the mechanical and thermophysical points of view, and in addition to appropriate code development and validation for complex modelling of fuel and material behaviour, the major challenge will come from the flexibility required to accommodate high burn-up (economic requirement), high levels of minor actinides (waste minimization – environmental concern) and high levels of fission product impurities (economics-fabrication simplification, non-proliferation) for different types of fuel (metallic, nitride, oxide, etc.). Moreover, it will be necessary to assess the impact on fuel and SM performance of selected fuel reprocessing technologies (pyro, CO₂ extraction, aqueous, etc.), fabrication processes (pellet, vibro, casting...) and other options.

In the area of structural materials, beyond the characterization of material properties under high flux radiation, the challenges will concern compatibilities with coolants (heavy liquid metals, gases, molten salts, etc.), with different physical and chemical forms of fuel (metal, nitride, oxide, etc.), and with the impurities in the fuel after reprocessing and fabrication.

Integrated RR experiments will be needed to validate both operational performance and safety related characteristics, i.e. the ability to work with low beta-effective and limited effects of cladding failure and fission product release, etc.

In addition to the INPRO methodology and project outlined above, this assessment of research needs is also based on experience based on the development of existing reactor lines, i.e. L(H)WR, GCR, and LMFR. The Agency constantly monitors these developments and leads many activities on topical problems. Our latest analysis shows that more than 55 innovative reactor concepts and designs are being developed worldwide and many of them target the use of advanced materials for fuel, structure, coolant or non-conventional processes that need to be tested in research reactors. Examples are:

- advanced MOX fuel with vibrocompacted nitride and enriched with MA for FRs,
- high burn-up fuel (up to 100 GWd/tHM) for water cooled reactors,
- composite ceramic fuel (pebble bed or prismatic block) for HTGRs, and
- fuel and SM for long-life core operation (up to 30 years) without reloading and reshuffling fuel.

6. Problems facing research reactors

One group of issues is connected with ageing RRs and the associated problem of decommissioning. The median age is 37 years for research reactors. For those research reactors where there are currently safety concerns, no plans for refurbishment and less than full self-sufficient use, the problem of decommissioning is pressing – entailing safety concerns, security concerns, a lack of funds for decommissioning, and the loss of experienced staff. Many research reactors that are shut down but not decommissioned still have fresh and/or spent fuel on-site.

Regional cooperation can be a tool to improve utilization, and cooperation on the back-end of RRs could include management, storage, reprocessing, waste management and final disposal. The optimum prospect from the Agency’s point of view would be the replacement of ageing and under-utilized research reactors with more regional, state-of-the-art, high flux, multipurpose research reactors with integrated interim storage and facilities for final disposal. But this route has to overcome the second problem, converting from HEU to LEU fuel.

Users of high flux RRs need a high-density reprocessable fuel. This work is now supported by the U.S. RERTR programme (Reduced Enrichment for Research and Test Reactors), which has recently been extended to 2016 as a part of the GTRI (Global Threat Reduction Initiative). RERTR has as its goal the reduction and eventual elimination of all commerce in HEU for research reactors. High-density U-Mo fuel is under development as a replacement, although problems have been encountered with high burn-up. There is an additional U.S. programme, the Foreign Research Reactor Spent Nuclear Fuel Acceptance
Program, the objective of which is to recover spent research reactor fuel of U.S. origin containing HEU.

7. Conclusion: Role of IAEA

The Agency shares the view that research reactors must necessarily play a key role in the further development of the peaceful use of atomic energy, in the education and training of scientists and engineers, in fundamental research and science, in the nuclear industry and in applications such as isotope production, medicine treatment, radiobiology, etc. The IAEA will continue to assist Member States as before in the effective and safe utilization of RRs, in education and training, and in effective use for applications. But reflecting the new challenges with which we are faced, Agency assistance will increasingly address the problems of aging facilities and personnel, accumulated spent fuel, waste management, decommissioning, and correcting the under-utilization of some reactors. Within its available resources, the Agency has to focus on vulnerabilities in the areas of non-proliferation, safety, physical protection of material and personnel. The IAEA will support international initiatives on the conversion from HEU to LEU, repatriation of HEU to the country of origin, and the development of dense LEU fuel.

8. References


THE EUROPEAN FUSION PROGRAM AND THE ROLE OF THE RESEARCH REACTORS

R. LÄSSER, R. ANDREANI, E. DIEGELE
EFDA Close Support Unit
Boltzmannstr. 2, D-85748 Garching, Germany

ABSTRACT
The main objectives of the European long-term Fusion Technology Program are i) investigation of DEMO breeding blankets options, ii) development of low activation materials resistant to high neutron fluence, iii) construction of IFMIF for validation of DEMO materials, and iv) promotion of modelling efforts for the understanding of radiation damage.

A large effort is required for the development and performance verification of the materials subjected to the intense neutron irradiation encountered in fusion reactors. In the absence of a strong 14.1 MeV neutron source fission materials research reactors are used. Elaborate in-pile and post-irradiation examinations are performed. In addition, the modelling effort is increased to predict the damage by a ‘true’ fusion spectrum in the future.

Even assuming that a positive decision for IFMIF construction can be reached, the operation of a limited number of materials test reactors is needed to perform irradiation studies on large samples and for screening.

1. Introduction

One of the dreams of human mankind is to have an almost inexhaustible, cheap and environmentally friendly energy source available similar to the sun, but controlled by humans.

The fusion community worldwide is presently making an attempt to build such a fusion reactor. The main difficulties are that i) such a power producing fusion reactor is still a few decades away, despite the promise of fusion experts in the past to harvest this energy source soon, and it will be a large (even if it is orders of magnitude smaller size than the sun) device combining high technologies, ii) its construction requires an international effort to share cost and resources, iii) fusion reactor relevant plasmas with a $Q_T$ far higher than one must be achieved and controlled and iv) materials are needed capable of withstanding the harsh fusion environment to exclude frequent replacement.

In Europe the long-term fusion programme is to a large extent directed by the outcome and the priorities given in the Power Plant Conceptual Studies (PPCS) [1] addressing the credibility of the fusion power plants, their design, safety, environmental and economic viability, sustainability, etc. Four different fusion reactor types, labelled A-D, have been studied in the past and are re-examined in the light of new developments. Recently also a further type was added, called model AB, with the intention to combine the advantages of the reactors A and B. For comparison these reactors have the same electrical power (1500 MW) output. They are based on different extrapolations in physics and technologies, e.g. different breeder blanket (BB) concepts, and, consequently, cover a wide range in thermal efficiency. Model AB and B employ as reference concepts the Helium Cooled Lithium-Lead (HCLL) and the Helium Cooled Pebble Bed (HCPB) BBs, respectively, whereas the “dual-coolant” and the “self-cooled” Lithium-Lead BB concepts are part of Model C and D, respectively. Model A is technologically the least advanced one and, therefore its construction involves the lowest risk. Model D, extrapolating to even more advanced technologies, requires a large R&D effort and implies a high development risk.

The European long-term fusion strategy is based on two pillars: i) the construction of ITER and its operation after the year 2015, a decision which is now fully overdue and awaiting an agreement of the six possibly contributing countries (CH, EU, JA, KO, RF, and US) and ii) the International Fusion
Materials Irradiation Facility (IFMIF) presently organised as an international cooperation under the Implementing Agreement between the four contributing countries: EU, JA, RF and US.

The purpose of ITER in this strategy is twofold: i) to test the physics aspects of fusion power like plasmas with respect to control, ash removal, reliability, safety, and ii) to demonstrate fusion technology issues of relevance for future power fusion reactors such as interaction of the plasma with the first wall, performance of BBs, achievement of a tritium breeding ratio larger than 1, processing of the kg quantities of tritium and its repeated recycling even during long pulses.

The second pillar of the EU strategy covers the necessity for improvement, development, full characterisation and full code qualification of radiation resistant fusion materials with low residual activation. For verification and validation of these materials to be used first in DEMO the decision to build a fusion relevant neutron source (IFMIF) with large enough flux in a reasonably sized volume to allow accelerated irradiation of a large number of samples is fundamental. A further precondition presently not fulfilled is that a certain number of countries must be interested in IFMIF and willing to share the cost. After a positive decision, IFMIF will enter a five-year phase of engineering validation and engineering design activities (EVEDA) of the major components for IFMIF and then the construction and decommissioning phase (CODA). Recently also an accelerated realisation of IFMIF within 10 years by aggressively combining EVEDA and CODA is discussed offering possible overall cost reductions. Under these assumptions ITER and IFMIF would be built and operated almost in parallel.

In the following the European Fusion Programme and the role of materials research reactors will be presented with emphasis on the components exposed to the neutron and gamma environment present in the vicinity of fusion plasmas.

2. Materials used in future fusion power reactors

2.1 Structural materials

The main requirements for structural fusion materials include good tensile and fracture mechanical properties, good creep strength, high fatigue resistance, low ductile to brittle transition temperature (DBTT), e.g. below 250°C for near term blanket concept, good compatibility with LiPb, low embrittlement due to hydrogen isotopes and high resistance against tritium permeation, thermal and dimensional stability (no recrystallisation and minimal swelling), and, mandatory, low activation and low level waste properties.

The purpose of the European Fusion Materials Development (MD) is to develop in the long-term the materials required by the models AB, B, C and D of the PPCS and in the shorter-term to produce and characterise the materials for the Test Blanket Modules (TBMs) of the HCLL and HCPB reference BBs. These TBMs will be tested in ITER from the first day of operation and exposed to the most realistic fusion environment for decades. They shall be DEMO relevant as far as possible and will be the first nuclear components fully built using the so-called EUROFER, the European reference material with the composition 9 CrWVTa belonging to the class of Reduced Activation Ferritic Martensitic (RAFM) steels. EUROFER will be also the material for the first generation of BBs in the models A, AB and B of PPCS, and the basis for structural material of model C.

Oxide dispersion strengthened (ODS) EUROFER based materials and nano-composited ferritic (NCF) steels are being developed for advanced BBs to increase the upper temperature limit from 550°C to 650°C and even to 750°C, respectively, for the purpose of achieving a higher thermal efficiency. The goal is to improve the (creep) strength at the higher temperature and at the same time to keep the good low temperatures properties at about 300°C.

SiCf/SiC composites, fabricated from silicon carbide fibres (SiCf) embedded in a silicon carbide (SiC) matrix, are planned to be used for the advanced BB concepts, either as functional materials to act as thermal barrier and electrical insulator in the form of flow channel inserts (model C of PPCS) or as BB structural material in the case of model D. The attractiveness and interest in the SiCf/SiC materials stem from the low activation properties and the high operational temperature window of 600°C up to
1100°C. The EU programme is focused on the industrial production of multi-directional SiC/SiC samples (3D with 2D as an intermediate step) of larger dimensions (20x20 cm²) with good mechanical and thermal properties as requested by designers. In addition the further improvement of the SiC/SiC properties and of the manufacturing technologies at laboratory scale is promoted.

Tungsten alloys are developed in the EU long-term programme for the use in advanced gas-cooled divertor concepts. Key issues are the low ductility and the high DBTT. Various routes are being pursued, e.g. use of nano-structured materials with low interstitial content, fabricated either under severe plastic deformation or by mechanical alloying. The materials being studied are WL-10 (W with 1% La₂O₃) and WVM (W doped with K). Drawbacks of nano-structured materials are the recrystallisation of the lattice, the connected loss of ductility at the high temperatures and long exposure times required for operation.

2.2 Functional materials

All functional materials to be used inside a fusion machine must be developed and characterised in accordance with their specific requirements, e.g. the materials used for first wall armour (CF/C, Be, W), tritium breeding (Li-based ceramics, liquid LiPb), neutron multiplication (Be), windows (CVD diamonds) of diagnostics and heating devices and in superconducting magnets (NbTi or Nb₃Sn as superconductors; Cu as stabiliser; fibre reinforced plastics (e.g. epoxy reinforced with a two dimensionally woven glass-fabric) as insulation materials).

3. Radiation damage

In a fusion environment due to the presence of the 14.1 MeV neutrons created by the fusion of deuterium and tritium the evolving radiation damage is different from conventional fission reactors although the basic atomic defect production follows the same physical processes initiated by the neutrons, the knock-on atoms and the displacement cascades. The difference is caused by the high-energy fusion neutrons capable of producing multiple displacement cascades and the much higher probability for the generation of transmutation products, especially of gaseous atoms such as hydrogen and helium isotopes. Their presence has a large influence on the production, stability, coalescence of vacancy clusters, voids and bubbles, the evolving microstructure and finally on the mechanical properties of the chosen material. Due to the dissimilar neutron spectra the comparison of the radiation damage produced e.g. in fission reactors, spallation sources, fusion devices and accelerators cannot be based only on the displacement per atom (dpa) values alone, but needs to include the production rate of gaseous transmutations through characteristic parameters such as the ratios of helium production rate to dpa and hydrogen production rate to dpa.

4. Materials Research Reactors

In the absence of any strong fusion neutron source the best-suited irradiation devices for fusion studies are fission neutron reactors despite the above explicatied different characteristics of radiation damage created by fission and fusion neutrons. Especially dedicated “materials research reactors” or “material test reactors (MTRs)” provide high fluxes of neutrons in relatively large-sized irradiation positions and have the advantage to offer good (irradiation temperature) control, instrumentation and additional measurement devices. Other irradiation facilities are spallation sources, cyclotrons, accelerators, dual and multiple ion beam irradiation devices. All of them have their own advantages and usefulness, but suffer from providing inappropriate energy spectra and, more important, are limited in irradiating large samples. Small scale dedicated 14.1 MeV neutron sources exist (in Frascati, Italy, or Rossendorf, Germany, where the fusion reaction is initiated by the injection of an accelerated deuteron beam on a tritium doped target) but are used for other applications because of their low flux.

The neutron sources most frequently used in the present EU long term fusion programmes are the High Flux Reactor (HFR) in Petten (Netherland), BR2 in Mol (Belgium) and OSIRIS in Saclay (France). Recently Russian reactors (BOR60) are being used for achieving high irradiation dose up to 80 dpa (in steels) within less than 4 calendar years.
Among the various research reactors in use in Europe only a few details shall be mentioned for the HFR in Petten, owned by JRC and operated by NRG. The HFR is a 45 MW pool type reactor, the core lattice is a 9 x 9 array containing 33 fuel assemblies, 6 control rods, 19 experimental positions and 23 Be reflector elements. The active height is 60 cm. The reactor runs ~11 cycles, each of 25 full power days (FPDs), totalling to 250 FPDs per year.

5. Irradiation and characterisation of irradiated fusion materials

The characterisation of the physical and mechanical properties of the structural and functional materials under radiation up to the foreseen lifetime of the components in ITER (~3 dpa) and in DEMO (80 to 150 dpa) is a fundamental prerequisite before their use.

With respect to EUROFER, various irradiation campaigns were performed in the past (30 dpa have been achieved in 2004) and are being carried out (up to 80 dpa) to determine the degradation in mechanical properties. Characteristic properties studied as a function of irradiation dose and temperature include the increase of yield strength due to irradiation hardening, the reduction of total elongation, DBTT shifts, stress relaxation and creep. The near-term irradiation programme addressing TBM issues have highest priority; TBM fabrication processes using EUROFER must be qualified. The joints produced by different techniques (TIG, electron beam, laser welding or HIP (hot isostatic pressure/diffusion bonding) need to be characterised for possible deterioration of the mechanical properties in the bonded and heat affected zones under irradiated and unirradiated conditions.

With respect to Li based ceramic pebbles, their tensile, creep and fracture behaviour needs to be studied as a function of the Li-burn-up. Further issues are the tritium release (also for Be), possible changes of the packing factors in the pebble beds, change of flow resistance for the purge gas. Materials databases are created where the results of the experimental campaigns are stored. In November 2004 the irradiation campaign called HIDOBE [2] at HFR was finished after 12 cycles producing 2 dpa in EUROFER, 8 \(10^{22}\) tritium atoms, 2-3% Li-burn-ups and a huge amount of interesting data. A cross-section of the experimental set-up simulating a simplified breeder unit with the breeder material lithium orthosilicate in the middle and the multiplier material Be on top and bottom and equipped with thermocouples and diagnostic tools is shown in Fig. 1. In addition to the large number of data collected during the almost two years of irradiation Post Irradiation Examination (PIE) of the whole set-up will give further information on structural aspects and compatibility issues between the installed materials under radiation.

For advanced structural materials such as ODS, SiC\(_f\)/SiC and W-alloys up to now only a limited number of irradiation studies has been performed. After the very recent successful production of EU reference materials of EUROFER ODS and 3D SiC\(_f\)/SiC, extensive irradiation campaigns will be
started to obtain information on their irradiation stability. Especially the irradiation stability of the grain size as controlled by the added oxides, the interphase between the SiC fibres and the SiC matrix and the nano-sized microstructure of W-alloys are of concern.

With respect to ITER the joints of the plasma facing components need to be studied for possible deterioration of the thermal conductivity and for unexpectedly large stress relaxation in bolts and nuts.

Also the components of superconducting magnets are affected in a radiation environment by the type of radiation, flux density, total dose, lifetime fluence, operating temperature and operating conditions. In NbTi and A15 compounds radiation damage reduces the transition temperature through an increase in disorder (especially in metals and ordered compounds, not so much in alloys), changes the upper critical field $H_{c2}$ and increases i) the normal state resistivity $\rho_n$ through the creation of additional scattering centres (again significantly in metals and ordered alloys and very little in alloys) and ii) the critical current density $J_c$ through the generation of pinning centres. A Residual Resistivity Ratio (RRR) of approximately 100-150 is required for a good stabilizer function of Cu. Irradiation enhances $\rho_n$ and decreases the RRR. The most radiation sensitive part of the superconducting magnets in ITER is the insulator. Not only the electrical properties, but also the intrinsic mechanical material parameters are affected and also X-rays can introduce damage and the mechanical properties of standard insulation materials are close to the ITER fluence requirement at $1 \times 10^{22}$ n/m$^2$ (E>$0.1$ MeV). First mechanical screening tests on newly developed fusion magnet insulation materials based on cyanate-ester clearly demonstrate improved mechanical properties under ITER relevant conditions.

MTRs are best suited to determine these properties under neutron irradiation. A matter of concern remains; the irradiation damage created in these reactors is most probably less severe than in fusion reactors and thus not fully representative. With a fluence that typically produces 2-5 dpa damage per annum in steels, MTR are well suited to accumulate dose levels as anticipated for ITER, but are too slow to accumulate 100 dpa typical for DEMO and beyond.

6. The International Fusion Materials Irradiation Facility

The conclusion of various workshops and strategy discussions in the past is that IFMIF [3] is the neutron source best suited to simulate the D-T neutrons offering a high neutron flux with the possibility of accelerated testing. Between 20 and 50 dpa/fpy are generated in the High Flux Test Module (HFTM) in a reasonable sized irradiation volume of about 500 cm$^3$. Although the IFMIF neutron spectrum is harder than in fusion, reaching up to 60 MeV, the hydrogen and helium to dpa ratios as well as the fraction of damage energy produced by the Primary Knock on Atoms (PKA) as a function of PKA energy are very similar to the values of a fusion reactor. IFMIF is considered to be the key experimental device for the second pillar of the Fast Fusion track.

Present investigations in Small Sample Test Technology (SSTT) will finally result in better use of the available irradiation volumes in IFMIF as well as in other irradiation facilities. Due to the already achieved progress, the aim to irradiate up to 800 samples at the same time in the HFTM of IFMIF is very reasonable.

7. Modelling effort of irradiation effects

Since 2003 increasing activities are performed in irradiation modelling with the main emphasis to develop an understanding of the radiation damage in RAFM steels produced in various irradiation facilities, to employ the developed tools for a combined description of the experimental observations and finally to use the verified tools and validated theories for development of new materials and extrapolation into parameter spaces where no experiments were performed. Clearly also here special irradiation experiments (multiples ion beam irradiations and other dedicated experiments) are required to verify and validate the physical models, codes and simulations. A further benefit of this activity could be to give directions for the production of new irradiation resistant materials and for the optimisation of the number and type of samples to be irradiated in limited irradiation volumes. So, in the long term, these modelling activities could help to reduce the number and cost of expensive
experimental irradiation campaigns. The required progress in modelling of such complex alloys as required for fusion will very likely be not achieved within this decade.

8. Conclusions

The development, characterisation and code validation of the materials to be applied in future power producing reactors seems today to be the key issue for their safe and economic operation. Neutron irradiation facilities are the most useful units for the determination of the radiation damage in these materials. Irradiation modelling is underway with good progress, but the consideration of all irradiation effects including transmutations, phase changes, impurities and major alloy elements present in the materials will need time. Also the gap between observations on microstructure and predictive conclusions to mechanical properties will need further large efforts and perseverance to give credible results in the case of multi-component alloys such as EUROFER and ODS steels. Therefore, the irradiation in the presently used fission facilities must be continued. It is the main pillar of the experimental characterisation programme at least for another decade. As some MTRs may reach their end of life within a decade, the planning and construction of new MTRs is now becoming an urgent issue. These reactors offer high irradiation volumes and are needed for many different purposes. The construction of at least one new dedicated European materials research reactor should be addressed now and pushed forward independently from a possibly positive decision for construction and operation of IFMIF.

References

PRODUCTION OF Mo$^{99}$ IN EUROPE: STATUS AND PERPECTIVES

H. BONET AND B. DAVID

National Institute for Radioelements (IRE)
6220 Fleurus, Belgium

B. PONSARD

Nuclear Research Centre (CEN-SCK)
2400 Mol, Belgium

ABSTRACT

Nuclear Medicine relies on Mo$^{99}$/Tc$^{99m}$ for 80% of the 25 million diagnoses made annually all over the world, for tracking diseases in cancerology, cardiology, neurology … Other isotopes such as I$^{131}$, Y$^{90}$, … are also used for therapy procedures which are currently promising significant developments.

Mo$^{99}$ together with I$^{131}$, Xe$^{133}$ and Sr$^{90}$ are fission products generated by irradiating HEU targets in high flux test reactors and then purified by chemical process in a hot cell facility. This involves a complex process which is only mastered by four major producers in the world, two of them located in Europe are producing about 50% of the world demand, making use of four reactors in four different European countries.

To ensure continuous and reliable supply of Mo$^{99}$, isotope with a short half life precluding any significant storage, sufficient irradiation capacity has to be available 365 days per year and requests appropriate coordination of these reactors operating schedule as well as mastering many other obstacles related to procurement of material, transportation, waste management, … and licenses.

We will review all these aspects in the present situation and the perspectives of development of the production of fission isotopes in Europe and in the world, addressing in particular the possible approaches for securing tomorrow the necessary irradiation capacity.

1. Introduction

Radioisotopes used in Nuclear Medicine are produced either in test reactors (by activation of stable isotopes or by fission of uranium-235) or in particle accelerators facilities (by activation). Reactors and accelerators are production tools complementary and not competing in most of the cases. Radioisotopes selected for production in reactors for medical application will have generally longer half-life than cyclotron isotopes because of production and logistics characteristics differences.

Main applications are related to radiodiagnostic for tracking diseases in cancerology, cardiology, neurology … , to radiotherapy mainly in cancerology but also in cardiology, arthritisme… and for cancer palliative care.

Most of these radioisotopes having a short half life, it is mandatory for satisfying Nuclear Medicine requirements in particular for diagnostics in emergency, to ensure a security of supply for 365 days per year. A reliable network of production facilities including appropriate distribution capabilities is a key strategical factor for the health of million of people in the world.
<table>
<thead>
<tr>
<th>Origin</th>
<th>DIAGNOSTIC</th>
<th>THERAPY</th>
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<tbody>
<tr>
<td></td>
<td>Isotope</td>
<td>Half life</td>
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<tr>
<td>Cyclotron</td>
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<tr>
<td></td>
<td>Fluor-18</td>
<td>2 minutes</td>
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<tr>
<td></td>
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</tr>
<tr>
<td></td>
<td>Thallium-201</td>
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<td>Reactor</td>
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<tr>
<td></td>
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<td></td>
<td>Iodine-131</td>
<td>8 days</td>
</tr>
<tr>
<td></td>
<td>Iodine-125</td>
<td>60 days</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Note (*) Decay daughter [Tc-99m (6 hours); Rhenium-188 (17 hours) Yttrium-90 (2.7 days)] used in medicine

Tab. 1 : Examples of radioisotopes for medical applications

Today Nuclear Medicine is using Radiopharmaceuticals based on a few key radioisotopes for performing about 25 million diagnostic procedures and 2 million therapy procedures per year in more than 10,000 hospitals in the world. For cancer therapy only, in industrialized countries 1/3 of the treatments are relying on radiotherapy and for developing countries, for economical reasons, the ratio reaches 50 %, whereas the number of cancer identified increases by 5 and 7.5 % per year respectively.

Today, 80 % of the nuclear medicine diagnostic procedures are based on the Tc⁹⁹m scintigraphy relying on the supply of Mo⁹⁹ fission product. Therefore, we will review the current status and the perspectives for future development of the production capacity of Mo⁹⁹.

2. Current production of fission radioisotopes in Europe and in the world

Production of fission radioisotopes is the most critical for Nuclear Medicine, as today there is only four major producers in the world relying on 6 reactors for which location and technical characteristics are appropriate whereas the production of other isotopes is much more diversified (more than 30 reactors and a large number of cyclotrons) but supports only 20 % of the medical procedures. This is illustrated in Tab.2.

<table>
<thead>
<tr>
<th>Reactor</th>
<th>Producer</th>
<th>Operation d/year</th>
<th>Production % Mo⁹⁹</th>
<th>Production % I⁰³¹</th>
<th>Capacity Mo⁹⁹ production</th>
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<td>-</td>
<td>30</td>
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<td>TYCO-H IRE</td>
<td>± 115</td>
<td>5</td>
<td>-</td>
<td>15</td>
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<td>IRE</td>
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<td>20</td>
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<td>IRE</td>
<td>± 210</td>
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<td>10</td>
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<td>NTP</td>
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<tr>
<td>TOTAL (%)</td>
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Tab. 2 : Tentative sharing of Mo⁹⁹ and I⁰³¹ fission radioisotope products by reactor and producer
The other producers in the world are generally supplying only the local market in many countries (Australia, Russia, India, China, Indonesia, Korea, Argentina, …). Europe is supplying about half of the world market and the USA and Japan are not producers although they are very important users of these products.

Although the peak capacity of Mo\textsuperscript{99} production might exceed 2.5 time the World demand, the security of supply might be jeopardized during some period of time if any unexpected shutdown is occurring when different reactors are shutdown for refueling/maintenance. For that reason, AIPES “Association of Imaging Producers & Equipment Suppliers” has set up a working group for having the Producers and Reactor Operators optimizing the different reactors operating schedules for limiting or eliminating the period of time at risk for a reliable isotope production. In particular, the CEN.SCK is willing to propose a maximum flexibility for scheduling the BR2 cycles during the most critical periods and possibly to extend or move slightly the operating cycles.

Although IRE has proven to be a reliable supplier of Mo\textsuperscript{99} for about 30 years, due to the risk of unexpected shutdown of aging multipurpose reactors, IRE signed mutual back-up agreements with the other producers. Nevertheless, today Reactor availability in the world has to be reassessed for the future in particular because of recent decisions to shutdown FRJ-2 and Studsvik reactors in Europe and delay to start up Maple reactors in Canada.

3. **Optimum characteristics for a future reactor network for Mo\textsuperscript{99} production**

For securing the future supply of Mo\textsuperscript{99} and production of other radioisotopes required by the medical community, a future European reactor should meet the following criteria:

- High availability: long cycle with minimum outages (3-4 days).
- High reliability: high safety and low risk of SCRAM or operation perturbation.
- Solutions based on diversity more than redundancy for reactor design and operations.
- Large irradiation capacity for producing 50% of world demand by 2015.
- High n thermal flux (1-1.5 \texttimes 10^{14} \text{ n/cm}^2.\text{s}) in most of irradiation positions, well predictable and stable.
- Low investment cost and low operating cost (including fuel cycle).
- Easy unloading of targets at full power and easy transportation from reactor to processing facility.

Considering that all existing reactors are not expected to remain in operation much later than 2015, and that JHR might have started operation at that time, a leftover network of multipurpose reactors could still be expected to contribute to production of fission radioisotopes but mainly in a back-up role. In particular, a minimum network of multipurpose reactors including new installations as FRM-II and some aging but very performant reactors such as BR2 (flux > 1.10^{15} n/cm.s) or HFR with BNCT facility should be maintained for meeting specific needs such as radiotherapy or production of specific radioisotopes (e.g: W-188).

Therefore, at least one new reactor should be made available by 2010-2020 and for satisfying the hereabove criteria it should be reactor fully dedicated to radioisotope production both for commercial supply and for supporting R&D for emerging radioisotopes applications. Typical characteristics and design parameters for such reactor should include:

- A pool reactor dedicated to radioisotope production, without neutron beams tubes, test loops in core or fancy experimental features.
  A very simple design instead of complex multipurpose facility will significantly increase safety, reliability and availability of the reactor.
  Availability might reach 360 days/year, and no perturbation by experiments will guarantee predictable flux level and minimize risk of SCRAM.
  Fuel and Reflectors elements layout might provide for a large number of RIGS dedicated to
irradiation of targets properly cooled in order to meet the expected production capacity.

- The design should be proven, not only for the fuel and the reactor core but also for all auxiliary systems. A prototype facility or a first of the kind advanced design will not provide any guaranty to be operational within a 5 years time period and will induce unacceptable financial risk. This is illustrated not only by the MAPLE project, but by many other projects, which were hardly realized.

- By eliminating the complex testing facilities and in particular penetration if neutron beam tubes, safety might be enforced by simple safety features and investment costs might be reduced by a factor 2 to 3. Operating cost related to fuel cycle, waste management and maintenance might also be significantly reduced. In such case with an investment not exceeding 100 million € and operating cost not exceeding 10 million € per year, the economy of irradiation for radioisotope production might be financially acceptable by itself. This could be improved by additional funding coming from irradiation of material (eg : Silicon doping) or by subsidies related to R&D programs asking for new radioisotopes or to education programs.

- A network of multipurpose reactors designed for reactor research and advanced physics testing should be kept available for providing the minimum back-up possibly needed at some period of time. Nevertheless, depending on the evolution of the world radioisotope market and production capabilities, a second dedicated reactor might be required in Europe by 2020. For the sake of security of supply, diversity in the design and sitting should be considered as mandatory. The lessons learned from the MAPLE project are obvious with this respect.

4. Conclusions

For meeting the needs of Nuclear Medicine within the next 10 to 15 years, a new simple and cost effective Reactor fully dedicated to radioisotope production should be installed within the network of multipurpose reactors used for other nuclear applications. For such important investments, although contribution of commercial suppliers might be foreseen, the support of national and international authorities is required for securing an appropriate funding.
IRRADIATION OF FUELS AND MATERIALS IN THE JULES HOROWITZ REACTOR: THE 6th EUROPEAN UNION JHR CO-ORDINATION ACTION (JHR-CA)

DANIEL IRACANE
CEA Nuclear Energy Division, DSOE/DIR
Building 121, CEA Saclay
F - 91191 Gif Sur Yvette
daniel.iracane@cea.fr

DANIEL PARRAT
CEA Nuclear Energy Division, DEC/SA3C
Building 315, CEA Cadarache
F - 13108 Saint Paul Lez Durance Cedex
daniel.parrat@cea.fr

ABSTRACT
The FEUNMARR thematic network in the 5th FP pointed out the need for a new MTR facility in Europe to answer the continuous need of irradiation capabilities for fission power reactors and fusion facilities and to face the ageing of present MTRs. The Jules Horowitz Reactor (JHR) Project in Cadarache copes with this context, as an international service-oriented user-facility. In the field of nuclear fuels and materials irradiation experiments, a 6th FP co-ordination action, called JHR-CA, has started at the beginning of 2004 for 2 years. The main objective is to network existing expertises on development of a new generation of experimental devices, through definition of conceptual designs, instrumentation and related in-reactor services. This paper presents the outline of the JHR project, the organization of the JHR-CA programme, and a choice of irradiation device conceptual design results.

1. Introduction: Situation of the Material Test Reactors in Europe

European Material Test Reactors (MTRs) have provided essential support for nuclear power programs over the last 40 years. Associated with hot laboratories for the post irradiation examinations, they are structuring research facilities for the European Research Area in the fission domain. They address the development and the qualification of materials and fuels under irradiation with sizes and environment conditions relevant for nuclear power plants in order to optimise and demonstrate safe operations of existing and coming power reactors as well as to support future reactors design.

However, in Europe, MTRs will be more than 50 years old in the next decade and will face increasing probability of shut-down due to their obsolescence. Such a situation cannot be sustained on the long term since “nuclear energy is a competitive energy source meeting the dual requirements for energy security and the reduction of greenhouse gas emissions, and is also an essential component of the energy mix” [1].

Renewing the experimental irradiation capability meet not only technical needs but important stakes such as i) increasing the integration of the European MTRs community which is now fragmented due to its history, and ii) maintaining a high scientific expertise level by training of new generations of searchers, engineers and operators. This answers the shared concern in Europe about the availability of competences and tools in the coming decades.

This analysis was made by a thematic network programme of the Euratom 5th FP, called Future European Union needs in Material Research Reactors (FEUNMARR). This programme involved experts and industry representatives, in order to answer the European Commission question on the
need for a new Material Testing Reactor (MTR) in Europe. The survey addressed the irradiation needs for the studies of material and fuel for commercial generation 2 and 3 up to generation 4 reactors, for back-end cycle requirements with dedicated breeders or accelerator driven systems, and for fusion. The survey dealt also with nuclear medicine and fundamental research. Cross cutting topics like education and training, operation best practices were addressed.

A consensus has been drawn on the following recommendations provided in the final synthesis report released in October 2002 [2], [3]:

- “Given the age of current MTRs, and anticipating continued R&D demand in the 21st century for material and fuel tests in support of nuclear energy production, there is a strategic need to renew material test reactors in Europe.

- Considering the lead-time before a new system can become operational, a decision to build a first new MTR in Europe is required in the very near future.

- The initiative to build the Jules Horowitz Reactor (JHR), and to organise an international programme around it, is an important contribution to the joint development of a new European Material Test Reactor.

- A new MTR, such as the proposed JHR, should in due time establish robust technical links with existing MTRs, aiming to provide a broad and efficient network of facilities at service of the international nuclear community. Programmes should be devised to reach a worldwide range of customers.

- A new MTR should support education and training for future teams of nuclear scientists and engineers, and help providing the European member states with the expertise that will be needed in areas such as nuclear reactor engineering and plant safety.

- There is an increasing reliance by the medical and pharmaceutical professions on research reactors to produce radioactive isotopes. Cooperation between at least three reactor sites within Europe is required in order to ensure a stable supply. If there is a risk that stability of supply cannot be ensured, the building of a new dedicated facility should be considered.”

2. The Jules Horowitz Material Test Reactor

The Jules Horowitz Reactor Project copes with this context for the study of material and fuel behaviour under irradiation.

The Commissariat à l’Energie Atomique (CEA), with the support of EDF, has launched the JHR project as a new European MTR to be implemented in Cadarache (south of France); start of operation is foreseen in 2014.

Many initiatives are on going to support the establishment of the JHR as an international R&D infrastructure, such as bilateral contacts with European and international industry and the launch of the International Advisory Group within the OECD/NEA framework to assess the project and to promote it as an international users-facility.

Because building a new MTR offers an opportunity to structure the European research area in the field of fission and because developing research infrastructure of European interest is one of the 6 major objectives of the 7th FP, it is expected that the European Commission supports the JHR, as an Infrastructure of European interest, for the joint development of experimental devices and for the construction. This will open the JHR experimental capability to European laboratories, and in particular the new Member States laboratories. The EC participation has an important leverage effect to consolidate the European status of the JHR project i) for European utilities and ii) for European laboratories.

2.1 JHR project objectives

JHR will offer modern irradiation experimental capabilities for studying material & fuel behaviour under irradiation. JHR will be a flexible experimental infrastructure to meet industrial and public needs related to generation 2, 3 and 4 power reactors and to different reactors technologies in operation in Europe.

JHR is designed to provide high neutron flux (twice larger than the maximum available today in MTRs), to run highly instrumented experiments to support advanced modelling giving prediction
beyond experimental points, and to operate experimental devices giving environment conditions (pressure, temperature, flux, coolant chemistry, ...) relevant for water reactors, for gas cooled thermal or fast reactors, for sodium fast reactors, etc

This irradiation experimental capability will address

- Power plant operation of existing and coming reactors (Gen 2 & 3) for material ageing and plant life management,
- Design evolutions for Gen 3 power reactors (in operation for all the century) such as performance improvement and evolution in the fuel cycle,
- Fuel performance and safety margins improvements with a strong continuous positive impact on Gen 2 & 3 reactor operating costs and on fuel cycle costs (burn-up and duty-cycle increase for UOX and MOX fuel)
- Fuel qualification in incidental or accidental situation
- Fuel optimisation for High Temperature Reactors
- Innovative material & fuel development for Gen 4 systems in different environments (very high temperature, fast neutron gas cooled systems, various coolant such as supercritical water, lead, sodium, ...). These systems raise challenging breakthroughs to be addressed by a modern experimental irradiation infrastructure like JHR.

These objectives require representative tests of structural materials and fuel components as well as in-depth investigations with separated effects experiments coupled with advanced modelling.

For example, the JHR design accommodates improved on-line monitoring capabilities such as the fission product laboratory directly coupled to the experimental fuel sample under irradiation. This monitoring can be used to get key information on the fission gas source term during transients related to incidents. It can also provide time-dependent data on the fuel microstructure evolution during the irradiation, which is of course a valuable input for modelling developments.

As a modern research infrastructure, JHR will contribute to the expertise & know-how training with a positive impact on safety, competitiveness and credibility.

The JHR design is optimised for the above technical objectives. As an important secondary objective, in connexion with other producers, the JHR will contribute to secure the production of radioisotope for medical application. This is a key public health stake.

2.2 JHR planning and funding

The JHR construction schedule is the following:

- Completion of definition studies in 2005 (typically 100 persons are working on definition studies since 2003)
- Decision for development & construction: Second half 2005
- Development studies: 2006-2007
- Construction phase & tests: 2008-2013
- Public consultation: 2005
- Preliminary safety analysis report: 2006
- Construction permit delivery: 2007
- Start of operations: 2014

The JHR construction cost is 500 M€ for the period 2006-2014.

The JHR project, as a flexible research infrastructure, meets at the same time i) middle term needs for the industry (utilities, vendors) and ii) long term public issues related to sustainability and energy policy. For that reason, a balanced financing scheme is proposed between industry and public authorities:

- CEA: 50%
- EDF: 20%
- AREVA: 10%
- 20% to be funded by European partners (European Commission, European industry) for an infrastructure of European Interest
2.3 Access rules to the JHR facility

A Consortium Agreement will be established between Members contributing to the financing of JHR construction:

- A Member has guaranteed and secured access rights to experimental locations in the reactor
- A Member (or a group of Members) can use his access right to perform Proprietary Experimental Programs for his benefit
- A Joint Program is built with access rights given by Members to address issues of common interest

Research institutes will participate in experimental programs:

- Together with industrial Members to implement proprietary programs
- Through the Joint Program; this program, addressing topics of common interest between European countries, can use a significant share of the JHR experimental capability. An International Advisory Group [IAG] for the Jules Horowitz Reactor was set up for 2003 within the OECD/NEA and with the EC to support the establishment of the JHR as an international R&D infrastructure for the nuclear industry,
- Through the European Commission membership; as a Member, the EC keep a lever effect with his access rights and implement programs strategic for Europe, through subsequent EURATOM FPs and by gathering when necessary complementary access rights among Members.

2.4 Experimental capability characteristics

To meet above needs, the JHR nuclear power will be 100MW. The JHR facility will allow performing a significant number of simultaneous experiments in core (~ 10) and in reflector (~ 10).

In core experiment will address material experiment with high fast flux capability ranging from $2.5 \times 10^{14}$ n/cm²/s up to $5 \times 10^{14}$ n/cm²/s (perturbed fast neutron flux) depending on the location.

In reflector experiments will address fuel experiment with perturbed thermal flux ranging from $5 \times 10^{13}$ n/cm²/s up to $5 \times 10^{14}$ n/cm²/s (perturbed thermal neutron flux). Experiments can be implemented in static locations, but also on displacement systems as an effective way to investigate transient regimes occurring in incidental or accidental situations.

These performances are to be understood as providing a flexible experimental capability where the flux can be used to create 16 dpa/year for in-core material experiments (with 260 full power operation days per year) or 850 W/cm (on 2% U5 enriched fuel) for in reflector simple fuel experiments.

One important JHR design feature is to accommodate several independent loops. This is mandatory to meet concurrently needs from different reactor technologies (light water reactors, sodium or gas fast reactors, high temperature reactors, …), from different reactor generations (Gen 2, 3, 4).

This objective requires optimising not only the core design but also the overall facility to effectively manage several loops.

For this purpose, an experimental area around the core pool, in the reactor building, will allow to settle out of pile loop components in dedicated casemates. This experimental area will accept about 10 loops either under irradiation or in preparation for coming irradiation.

The Fission Product Laboratory will be settled in the reactor building experimental area to be connected to several fuel loops. The on-line measurement of fission product gas and/or coolant contaminated with fission product will be an effective standard JHR capability providing very valuable data for fuel modelling.

Last but not least, the experimental process will make use of two hot cells to manage experimental devices before and after the irradiation with non destructive examination posts. Safety experiments are an important objective for JHR and require, as a common feature, an “alpha cell”; this cell is designed to manage devices with failed experimental fuel without causing contamination. A fourth cell will be used for the management of spent fuel, of waste package, of targets for medical application.

2.5 General lay out

The future site of JHR facility has been selected at the CEA Cadarache centre.
The global lay out of JHR facility consists in:
- a central area, called nuclear island, comprising the reactor building (RB) and a nuclear auxiliaries building (NAB),
- buildings related to reactor operation such as electrical support and cooling,
- buildings related to facility operation, such as, staff offices and cold assembly workshop for experimental devices.

JHR reactor core is laid in a reactor pool inside reactor building. Reactor primary circuit is completely located inside the reactor building. Parts of experimental devices which are not placed in reactor pool are located in the reactor building experimental area, adjacent to the reactor pool.

Hot cells, laboratories and storage pools are located out of the reactor building, in the nuclear auxiliaries building.

Transfers, between reactor building and nuclear auxiliaries building, of experimental devices are performed underwater. This leads to the implementation of a monolithic water block linking reactor pool to experimental or storage areas in NAB.

The reactor building is divided into two zones. The first zone contains the reactor hall and the reactor primary cooling system. The second zone hosts the experimental areas in connection with in pile irradiation (eg., loop support systems, gamma scanning, fission product analysis laboratory etc.).

This experimental area is laid in such a way that the interface with the reactor pool is maximised to facilitate the distribution of underwater experimental connections between reactor pool and the experimental zone. Bunkers and laboratories are located below the level of reactor hall. Three different levels in the experimental area are provided below the reactor hall level. At each level, a significant free space (in the order of 300 m²) is provided for experimental device implementation.
The JHR set of pools and hot cells is organised on the general principle of underwater transfers for used fuel elements and irradiation devices, and more generally, irradiated components. Pools are distributed along the central water channel that runs from the reactor building to the nuclear auxiliaries building.

Pools in the reactor building are limited to the reactor pool (including neutronography capability for experiments) and an intermediary deactivation pool (for temporary storage of fuel elements, reflector elements or replaced core mechanical structures). During reactor shutdown, experimental devices can be temporarily stored in a dedicated rack in the reactor pool.

The main pools are located in the nuclear auxiliaries building: one used fuel storage pool, one experimental devices storage pool and one mechanical components working and storage pool (including cutting out of waste).

Two water channels from the central water channel enable underwater access to the hot cell area in the nuclear auxiliaries building. This area includes 4 hot cells with the following functions: a $\beta,\gamma$ hot cell for general material or fuel experiment operation, a specific $\alpha,\beta,\gamma$ hot cell for experimental programs with broken experimental fuel samples, a $\beta,\gamma$ hot cell for waste management, 1 $\beta,\gamma$ hot cell dedicated to the transit of radioisotope for medical application and to the dry evacuation of used fuel. Furthermore, material and fuel experiment hot cells are connected to non destructive examination stations for irradiated samples.

**Beyond reactor irradiation, experimental process involves hot cells, pools and laboratories**

2.6 Core features

JHR core design is based on a tank pool concept. The general core options are the following:

In order to create an intense fast neutron source, the core is of compact size (600 mm fuel active height). The core is cooled and moderated with water.

The core is designed to be operated with a high density low enriched fuel ($^9$U enrichment lower than 20%), density 8 g/cm$^3$, requiring the development of UMo fuel. The fuel element is of circular shape (set of curved plates assembled with stiffeners) and comprises a central hole. The UMo fuel is under development within an international collaboration (UMo/Al dispersion solutions and monolithic UMo solution). As a back-up solution, the JHR may be started with an U3Si2 fuel with a larger enrichment (typically 30%).
The core area is surrounded by a reflector which optimises the core cycle length and provides intense thermal fluxes in this area. The reflector area is made of water and beryllium elements. Irradiation devices can be placed either in the core area (in a fuel element central hole or in place of a fuel element) or in the reflector area.

The 60 cm height core is in a Φ 740 mm pressurised tank; 34 fuel elements are placed in a 37 locations core rack.

~20 experiments can be located in the core and in the reflector

- High thermal neutrons flux (5 × 10^14 n/cm²/s) to produce fission rates relevant for fuel studies
- Beryllium reflector to improve core performances
- High fast neutrons flux (5 × 10^14 n/cm²/s) to simulate material ageing

The JHR Fuel element:

- Fuel mix: U-8Mo, Al à 8 gU/cm³, 19.75%
- or (decision 2007): U3Si2, Al à 4.8 gU/cm³, 36%
- Active height: 600 mm
- Fuel rod thickness: 6.61 mm
- Water channel: 1.54 mm
- Cladding: Aluminium alloy
- Structure: 38 plates
- External diameter: 95 mm
- Internal diameter: 41 mm
- Int. diam. of protection tube: 37 mm
3. JHR Co-ordination Action programme

3.1 JHR-CA objectives and organization
The JHR Co-ordination Action programme (JHR-CA) in the 6th FP aims to contribute to the joint development of innovative experimental devices. This innovation process is driven with the objective to implement a new generation of experimental devices in the JHR; it has also a strong added value for existing MTRs by cross-fertilization.

The JHR-CA has started on 1st January 2004 for two years with two main objectives.

3.1.1 To structure a European collaboration on the definition of the experimental devices, and related processes and services: The Experts Group

The technical definition of the JHR experimental devices and the related experimental processes and the induced services will benefit from an international shared development.

The JHR-CA gathers the European expertise for providing specifications and conceptual designs of the JHR experimental devices with a careful consideration on the related overall processes and induced services. The integration of these conceptual designs in the JHR is assessed through several criteria such as:

- the performance of the devices versus the scientific and technological state of art,
- the flexibility and efficiency of the overall experimental process (not limited to the irradiation unfolding),
- the foreseen quality of the service for industry and research institutes (flexibility, short time-to-result, cost, quality of the data).

The Expert Group (EG) drives the definition of the JHR experimental devices design to meet current industrial demands (issues on existing or under development reactors such as fuel performance and safety, mechanical behaviour under irradiation, corrosion, ageing assessment) and to meet emerging needs (e.g. gas and high temperature loop, fast transient experiments, …).

The JHR-CA allows to address through the European collaboration identified breakthrough for key technical stakes (hot temperature management, on-line instrumentations and control, variable neutron shield, etc). This provides an effective side-product for developing innovative programmes on existing research reactors programs to the benefit of MTRs users.

3.1.2 To involve end-users in the design process: The Users Group

The Co-ordination Action aims to involve vendors, utilities, public stakeholders in the design of the experimental devices and MTR next generation. These end-users will make sure that the JHR Project provides a service-oriented irradiation platform responding to their needs.

Through the Co-ordination Action, information on the detailed design and performances versus needs are broadly shared, which supports the convergence process toward the construction decision since:

- It is a decisive step for a renewing policy of research infrastructures for fission, covering the creation of new facility and the functioning of existing ones and access to them,
- It structures a European collaboration on the definition of the experimental devices with users involved from the beginning.

3.2 JHR-CA participants

The EG is constituted by following institutes:
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<tr>
<th>Participant name</th>
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<th>Country</th>
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<td>Commissariat à l’Énergie Atomique</td>
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<td>SCK•CEN</td>
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Following companies are members of the UG:

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<td>EDF</td>
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<td>KKL/UAK</td>
<td>Switzerland</td>
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<td>IBERDROLA</td>
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4. JHR-CA Work Plan

4.1 Expert Group work packages

Five scientific work packages (WPs) have been identified as proposing challenging exercises for the scientific community. The aim is to build up a conceptual design of an irradiation device capable of performing sophisticated in-situ measurements or controlling precisely the environment of the sample. The three first ones concern the materials and the two others the nuclear fuel:

- **WP1: Materials behaviour under high temperature conditions**
  The objective is the conceptual design of an experimental helium gas loop designed for irradiation of HTR materials in the JHR core, at high temperature (700-1200°C) and high fast neutron flux (from 1,4 to 5,2 * 10^{14} n/cm².s). This loop is located inside a JHR fuel assembly, and is dedicated to separate effects experiments on selected materials, such as SiC/SiC, Oxide Dispersed Strengthened Steel (ODS) and ZrC.

- **WP2: In-pile mechanical testing devices**
  The objective is the conceptual design of an in-pile mechanical testing device with on-line environment, stress and strain control. As a challenge, one aim to perform an on-line control (axial and bi-axial) load, with a precise mechanical and temperature monitoring on a single-axial device.

- **WP3: Corrosion under irradiation**
  The exercise addresses the in-pile irradiation assisted cracking growth rate measurements, thanks to the local electric potential drop measurement.

- **WP4: Current fuels**
  This WP addresses end-of-life scenarios for PWR fuels, and notably the fuel thermal-mechanical behaviour and the fission gas release, thanks to a cluster of instrumented rodlets (central thermocouples, pressure gauges and fission gas sweeping lines) placed in a PWR loop.
• WP5: Gas system fuels
  This WP addresses high pressure and high temperature gas rig designed for the irradiation of a 8 HTR/VHTR (High/Very High Temperature Reactor) compact stack in the JHR reflector. The stack is swept by an inert gas at low flow rate to route the released fission gases to the fission product laboratory for quantitative measurements.

Three other work packages deal with subjects important for the definition of the JHR experimental capability:

• WP6: Medical applications
  This addresses the technical and strategic optimisation of the European isotopes production for medical applications, taking into the increase of the demand and the securing of this production by networking two or three research reactors.

• WP7: Operation optimisation
  This cross-cutting topic will provide conditions for an optimum irradiation device fleet operation and for a good management of the experimental programmes.

• WP8: Integration assessment
  This important cross-cutting topic is performed by the JHR team responsible for designing JHR core and facility. Taking benefit from the studies performed in WP1 to 6, this team identifies the interfaces between the experimental devices and the facility, and study the critical points, in order to integrate the devices versus the reactor design. The interaction of the WP8 with the other ones can be summarized by this diagram:

4.2 Users Group work packages
Four topics are investigated by the User Group:

• Assessing the on-going design versus needs
  This topic comprises the participation to the IAG meetings, for assessing the overall JHR design, and the assessment of the experimental devices conceptual design proposed by the EG.

• Defining organization and operation rules suitable for a new European MTR
  The recommendations are supported by the WP7 conclusions.

• Relevance of JHR safety standards versus experimental needs
  This addresses the assessment of the JHR safety standards from the users point of view with a possible release of requirements.

• Testing capacity of JHR on safety tests
The objective is to formulate needs and specifications for experiments dedicated to safety programmes.

5. JHR Co-ordination Action: Examples of study results

Among a lot of results gained in 2004, three examples of conceptual designs can be shortly described in this paper: two on materials, and the third one on fuels.

5.1 WP1: HELIOS test loop for in-core material irradiation at high temperature and high dose rate

The HELIOS test device is an experimental Helium gas loop at 7 MPa designed for material in-core irradiation at high temperature and high dose rate:

- Typical material: SiC based ceramics and composites, ODS (Oxide Dispersion strengthened Steels) and ZrC for High Temperature Reactors
- Fast neutron flux range: $1.5 \times 10^{14}$ to $5 \times 10^{14}$ n.cm$^{-2}$.s$^{-1}$
- Maximum dose rate: 16 dpa/y
- Nominal temperature: up to 1200°C
- Available space for sample holder: diameter 25 mm - length 600 mm,
- Fluid surrounding the samples: circulating Helium.

It is dedicated to separate effect experiments such as dose accumulation, temperature and time dependency tests, and environment dependency test (helium with controlled impurities). Different sample holders can be designed to reach specific experimental objectives, depending on the type of samples and instrumentation.

The HELIOS test loop should be placed in one of the standard in-core experimental emplacements with the highest fast neutron flux. The temperature control of the samples is possible by means of a small in-pile loop of circulating gas such as Helium. Indeed, an integrated circulator situated at the head of the in-pile part and far above the active zone allows the fluid flowing down. Then, after being pre-heated by means of electric heater the gas travels through the active zone around the samples, the fluid is cooled down by a heat exchanger situated at the bottom of the rig. Finally, the fluid returns up in the gap between the pressure tube’s inner wall and the sample holder shell.
Schematic view of a sample holder in the HELIOS test device
FIGURE A

Radial cross section

with upper part of sample holder - type E

FIGURE B

Sample

3.0 x 3.0 mm

Gas gap 3

4.0 mm

Gas gap 1

0.2 mm

Gas gap 2

2.8 mm

Stainless steel

Stainless steel

Molybdenum

Molybdenum

Sample fillet
3.5 x 3.5 mm

32 x 1.0 mm

20.0 x 1.0 mm

18.0 x 1 mm

29.6 x 2 mm

HELIOS gas loop: Conceptual design results:

Figure A: Radial cross section with sampler carrier (cuboid-shaped specimens)

Figure B: Temperature distribution in the sample carrier (2D calculation with CAST3M)
5.2 WP2: In-core test device for material irradiation at high dose rate under controlled stress

This device is dedicated to experiments at very high fast neutron flux such as:

- dose accumulation or irradiation growth of cladding and structural materials with on-line strain measurement (diameter, length ...),
- axial stress relief test on blade sample with on-line load measurement,
- loading tests with strain measurement like diametral creep test under monitored internal pressure,
- in-core controlled dynamic tensile test,
- creep test with in-core stress increments, control of the biaxiality ratio and on-line measurement of axial and diametral strain,

Main features are:

- Fast neutron flux range: $2.5 \times 10^{14} - 5 \times 10^{14} \text{ n.cm}^{-2}.\text{s}^{-1}$
- Maximum dose rate: 16 dpa/y
- Nominal temperature: up to 600°C
- Available space for sample holder: diameter 25 mm - length 800 mm,
- Maximum temperature discrepancy over the samples: 7.5°C,
- Typical sample material: Steel, Zirconium alloy, Nickel alloy, Titanium, Aluminum alloy

The accurate temperature control of the samples is possible by means of a small in-pile loop of circulating fluid such as NaK. Indeed, an annular electromagnetic pump situated above the active zone allows the fluid flowing down after being previously pre-heated by the mean of electric heater situated just above the pump. Then the fluid is cooled down by a heat exchanger situated at the bottom of the rig.

Pneumatic material testing irradiation device: Sample multi-axial loading and axial / diametral strain measurements (VTT concept)
**5.3 WP4: Fuel rod cluster irradiation under PWR conditions**

This test device is an experimental pressurized water loop designed to LWR fuel rod cluster testing in the JHR’s reflector. Typical samples are 6 to 8 segmented or re-fabricated fresh or pre-irradiated rods with a fissile length up to 600 mm and an external diameter up to 12.5 mm. It is dedicated to separate effect experiments on comparative characterization of fuel rods irradiated in the same PWR conditions: microstructure evolution, fission gas release and fission product distribution. It is designed for steady state irradiation, medium power transients as well as first phase of loss of coolant experiments (no high temperature and no quenching).

Because of this piping system stiffness, the in-pile rig is not adapted to displacement system of the reactor's reflector. Thus, the test rig should be placed in one specific experimental emplacement equipped with a variable thermal neutron screen that allows fuel rod power adjustments and medium speed transients.

Main features of this device are:

- Fast neutron unperturbed flux (> 1MeV): up to 2.5 $10^{13}$ n.cm$^{-2}$.s$^{-1}$
- Thermal neutron unperturbed flux: up to 4.3 $10^{14}$ n.cm$^{-2}$.s$^{-1}$
- Mean linear power of fuel samples:
  - nominal value: 200 W.cm$^{-1}$
  - maximum value: 400 W.cm$^{-1}$
- Inlet coolant temperature: 300 +/- 20 °C
- Maximum outlet coolant temperature: 340°C
- Nominal coolant pressure: 155 bar
6. Conclusion

The output and the added value of the JHR-CA is a consensus shared between European laboratories and user on a first level of specifications for the next generation of irradiation experimental devices and programmes. The technical work performed in 2004 in the Experts Group allowed to define together irradiation specifications and an irradiation device conceptual design for each workpackage. This work is now assessed by the Engineering team in terms of integration in the JHR facility, through two main objectives:

- to define the interfaces between the device and the facility (e.g. instrumentation and fluid lines, location within the core or in the reflector,…),
- to identify the critical points regarding the safety aspects.

After this assessment phase, an iteration will be done between the irradiation specifications, the conceptual design and the assessment results, to build-up a feasible experiment. Another output of this iteration is the determination of irradiation devices or components set, or instrumentation means to study in priority.

For that reason, the following step after the JHR-CA has to address the detailed design of some of these devices and the reinforcement of the European collaboration on this field. For that purpose, it is proposed to launch an Integrated Infrastructure Initiative with the following objectives:
• to drive technological innovation in the MTR field, with a mutual benefit for existing European MTRs and JHR,
• for the research activity, to develop some irradiation devices and/or key components, including test definition on existing MTRs,
• for the Transnational access, to offer access in existing MTRs for testing innovations; this contribute to the growth of present MTRs technological offer,
• For the networking activities, to address cross-cutting topics such as gauge calibration and conditioning, professional training.

7. References

STATUS REPORT ON THE COST AND AVAILABILITY OF ENRICHED URANIUM FOR RESEARCH REACTORS

HANS MÜLLER and JÜRGEN LAUCHT
Research Reactor Fuel Cycle Services
RWE NUKEM GmbH, Alzenau, Germany

ABSTRACT
Availability and price development of enriched uranium contained in fuel elements for research reactors plays an important role with regard to reliability and economic and planning reasons. The leading price factors of LEU (19.75% enriched uranium metal), are the contained natural uranium equivalent in the form of UF6 (feed component), the separative work of the enrichment (SWU), conversion of the enriched uranium into metal form and associated services, such as transportation. World market price of feed material for enrichment was more or less stable in the last decades. After very moderate feed price increases between 2001 and mid-2003, the price gained momentum and almost doubled in the short period between the 2nd half of 2003 and year-end 2004.

This paper tries to give an answer to the question how the increased feed and SWU price affects the price and availability of LEU for research reactors. This paper deals also with the historic development of supply sources and origins of LEU, present status of supplies, price formulae, available stocks, value and advantages of fuel origin and return possibilities of spent LEU.

1. Introduction
Mandatory for the operation of research reactors is the safe supply of fresh fuel and a viable option for the disposal of spent fuel. In the 1990s both the supply of fresh LEU and the take-back of spent fuel were adversely affected by the U.S. side due to a supply problem at the Y-12 plant and the lack of timely renewal of the U.S. Return Program for spent fuel. In the European Union (EU) the shortfall of supply from the US could be bridged by the use of no longer needed, considerable stocks of enriched uranium. This supply was organized by RWE NUKEM.

The question is now whether a supply gap will again occur and whether research reactor operators can rely on a stable pricing for the needed LEU.

2. Historical development
As in the past the major suppliers of LEU were Russia and the U.S. From the 1960s until the mid-1980s, the material supplied by the U.S. for production of fuel elements for research reactors inside and outside the USA was mainly highly enriched uranium (HEU). In the same period, the Soviet Union supplied HEU with enrichments from 36 to 90% U-235 in the form of fuel elements to Soviet-designed research reactors inside and outside the Soviet Union.

In 1977, U.S. President Carter expressed concern about the widespread use of weapons-grade uranium in research reactors worldwide. In the same year, the USA launched a program to reduce enrichment in research reactors in Western countries in order to eliminate proliferation risks and misuse of HEU. The International Fuel Cycle Evaluation (INFCE) under Working Group 8 [10] recommended that the ideal U-235 assay for research reactor fuel should be less than 20%, namely 19.75%, in order to address the non-proliferation concerns of the U.S. In 1978, the U.S. installed a program to reduce the enrichment of research and test reactors (RERTR Program) under the lead of the U.S. Department of Energy (USDOE) and Argonne National Laboratories with the goal to convert all reactors with HEU fuel to LEU fuel and to reduce the annual export of U.S. HEU as a first step from 450 kg/a to 150
kg/a. After the end of the Cold War, Russia joined the RERTR Program with the same objective to reduce enrichment in Russian-supplied research reactors.

Following the success and progress of the worldwide conversion program to replace highly enriched uranium (HEU) by low enriched uranium (LEU) the standard fuel for research reactors is uranium with an enrichment of 19.75% U-235 in the form of metal pieces.

3. Supply sources of LEU

Today, LEU is mainly provided by the U.S. by using excess military stocks of HEU. The U.S. has discontinued the production of fresh HEU, and has major programs underway to reduce their surplus stockpiles of HEU. In Russia, HEU may still be produced for civil purposes (fast reactor and research reactor fuel).

Through another international program Russia is selling uranium for power reactors below 5% U-235 derived from 500 tons of Russian surplus HEU to the USA, and the USA has a separate program to eliminate 174 tons of U.S. HEU that has been declared surplus. Together these programs are currently supplying up to 15% of the uranium demand for power reactors worldwide.

The United Kingdom, France, and South Africa have discontinued all HEU production.

4. Annual demand for LEU

The annual quantities of HEU needed to produce LEU for research reactors are relatively small. It is estimated that at present among the Western countries about 1 ton of LEU is being used to fuel already converted reactors. This quantity could increase to 2-3 tons if high flux reactors in the USA and the EU would be converted to LEU as well. The production of 3 tons of LEU would be equivalent to a dilution of approximately 600 kg of HEU (90% U-235).

5. LEU situation in the USA

The U.S. surplus disposition program for 174 tons of HEU has allocated up to 10 tons of HEU for the production of LEU for research reactors for the period of 2002 to 2016. The LEU is generated by blending down of the HEU to LEU by melting in induction furnaces. The quantity of 10 tons of HEU is sufficient to produce more than 45 tons of LEU.

6. LEU situation in Russia

No exact figures are available, but we have an information that the annual production of HEU fuel in Russia for research reactors inside and outside Russia was in the year 2002 approximately 850 kg with U-235 assays ranging from 20 to 90%.

7. Price factors for LEU

The main price factors for enriched uranium for power reactors are the contained natural uranium equivalent in the form of UF6 (feed component), the separative work of the enrichment (SWU), conversion of the enriched uranium into metal form and associated services, such as transportation. Two factors, namely the natural uranium price and the SWU price, are market related prices and subject to changes. These changes are mainly due to the supply and demand situation, but partly also due to psychological factors, such as the mere perception of supply surplus or scarcity. The recent price shifts in the uranium market show that low prices for LEU - if really determined upon the market prices of the individual cost components - can be no longer taken for granted.
8. World market price developments for natural uranium

Starting in the late-1970s until the mid-1990s, natural uranium prices experienced an almost continuous decline. However, there was a temporary peak in uranium prices around mid-1996, with prices exceeding the level of US$ 16/lb U3O8. But thereafter prices decreased again almost irresistibly and bottomed out late in 2000 and early in 2001, at levels of even slightly below US$ 7.00/lb U3O8. But - as stated earlier - from that time on natural uranium prices increased again almost continuously.

And these were the reasons for this price development:

The fall of the iron curtain around 1990 heralded the gradual evolvement of a global nuclear fuel market. Massive new nuclear fuel quantities became available, first from the Soviet Union and then from the Commonwealth of Independent States (CIS). In the mid-1990s, the release of uranium derived from Russian nuclear weapons was launched. Shortly thereafter, the U.S. government decided to liquidate most of its civil nuclear inventories, through the privatization of the United States Enrichment Corporation (USEC).

This massive influx of uranium from the new supply sources into the Western market depressed prices to extreme low levels - as just stated - causing a cut-back in uranium production and exploration efforts. In effect, inventory holders sold most of their excess material at the cost of the cheapest uranium mines, not noticing that they were driving higher-cost uranium mines out of the business, thus preventing investment in existing and new uranium production centres.

However, more recently a number of events triggered and stimulated the perception of an approaching uranium supply shortage, thereby accelerating uranium price increases throughout 2003 and 2004:

- Secondary nuclear fuel supply sources feeding the uranium spot market, such as utility excess inventories, have largely dried out.
- A fire in the solvent extraction area of the Australian Olympic Dam mill late in 2001 led to a sharp cut of the project's output scheduled for 2002.
- In the first half of 2003, the rapid increase of water flowing into the world's largest individual mine, McArthur River in Canada, led to the suspension of production for about three months.
- The accidental UF6 release from Converdyn's Metropolis operations late in December 2003 caused a temporary interruption of part of the light water reactor fuel supply chain in North America as well in Europe. Most importantly, the temporary shutdown of the Metropolis facility was about to adversely effect the provision of HEU feed under the US-Russian HEU-LEU Agreement.
- And - last but not least - according to the amendment of the contractual stipulations of the HEU-LEU Agreement in spring 2004, the Russians were conceded to take back almost 30% of the HEU feed, thereby reducing the Western market's availability of UF6 originating from this deal over its remaining term by about 25,000 tons U.

Despite little new generating capacity being brought on line on a worldwide basis, uranium demand is steadily increasing. Part of this increase is due to NPP capacity factor improvements, upgrading of NPPs, and reactor lifetime extensions. This is happening primarily, but not exclusively, in the USA. As a result, world uranium requirements are continuing their upward trend.

In 2004, natural uranium production worldwide was about 38,500 tons U, compared to the actual demand of around 66,700 tons U. Thus, just below 60% of the total demand was covered by fresh production; the balance of slightly more than 40% came from secondary sources. The prospects concerning the future uranium supply situation are currently not so good:

- There are just a few new uranium production projects in the pipeline.
- There is only limited ability to squeeze additional uranium out of existing projects.
• Lead times of new mining and milling projects are lengthening, partly due to time-consuming Environmental Impact Assessments.
• Producers' willingness to invest in the extension of existing or the start-up of new mines is adversely affected by uncertainties over the timing of potential governmental sales of additional nuclear fuel, such as military uranium and plutonium.
• Exploration activities were extremely low in previous years. However, they buoyed up under the impression of recent uranium price increases.

9. What is the influence of increased natural uranium prices for LEU for research reactors?

Although LEU is mainly produced by blending of HEU, the following pricing parameters could apply if one would adopt the pricing system for low enriched uranium for power reactors:

1 kg of 19.75% enriched uranium contains 47.324 kg of natural uranium equivalence and 37,783 separative work units (SWU), if proceeding on a tails assay of 0.3% U-235.

LEU Price = 47.324 x Unat price (US$/kg) + 37.783 x SWU price (US$) + conversion of UF6 to U metal (US$)

Prices of natural uranium (Unat) and separative work (SWU) are subject to market developments:

- Average Unat price for the period 1997-2001: approx. 22 US$/kg UF6
- Current Unat price: approx. 65 US$/kg UF6
- Average SWU price 1997-2001: approx. 70 US$
- Current SWU price: approx. 100 US$

Calculation of LEU price based on current market:

\[ \text{47,324 kg U (in UF6) x 65 $/kg U (UF6) = US$ 3,076} \]
\[ \text{37,783 SWU x 100/SWU = US$ 3,778} \]

The resulting price is US$ 6.854 per kg LEU in the form of UF6.

Additional cost for conversion from UF6 to U-metal applies when LEU is produced from UF6. However, while the conversion of enriched UF6 to UO2 as part of the power reactor fuel fabrication - with product assays below 5% U-235 - is done in industrial-scale facilities, the conversion of UF6 with assays up to 19.75% U-235 to U-metal is performed in very small conversion facilities with safe geometry and is subsequently very expensive. Thus, the price for this latter type of conversion is not at all comparable with the one for the conversion of UF6 to UO2 for power reactors.

Prices for such LEU conversion are not published and are subject to local conditions. According to RWE NUKEM’s experience, such conversion prices are in the range of US$ 2,000/kg U. This cost factor is not very much market related and could be assumed as constant cost (GDP related only).

As a result, the theoretical price of 1 kg LEU metal (19.75% U-235) would be US$ 8.854.

Since this price consists of 1/3 part of the cost for natural uranium, the LEU price will be depend on uranium market price movements. Quite some analysts predict that the current natural uranium price has still the potential of further growth, but that it should at least remain at its current high level.

10. How are the major LEU producers facing the new uranium prices?

As a matter of fact, Russia and the U.S. being the major LEU producers have different ways of production: While the Russians use mostly fresh uranium for enrichment, the U.S. uses existing military HEU stocks for down blending. Accordingly, we have two different price calculation scenarios:
Russia has to take into account the elevated cost of natural uranium and separative work since they have to source the feed material and to provide enrichment services. Even conversion cost increases more in Russia than elsewhere since their GDP is higher than in Western states. As a result, the price of Russian LEU is relatively close connected to market prices.

The US producer uses material from shelf which has been paid already in the past under military programs. Additional cost for down blending with the help of induction furnaces is non-market related internal cost. Since money earned from LEU sales has to be returned to the treasury (treasury owns the 10 tons HEU source for research reactor use), the sales price does not necessarily be at market price. Since USDOE has mostly agreed upon long-term requirements contracts with research reactor operators there is only little clearance for price increases (escalation clause). However, new contracts will certainly at least partially reflect the increased market prices for feed and SWU.

11. Conclusion

Recent market price increases have influenced the market of power reactor uranium as well as for LEU for research reactors. The calculated market price for LEU with an assay of 19.75% U-235 as needed for research reactors rose within the last few years from about US$ 6,000 to US$ 8,800 now.

This price is an indicator only. Reactor operators may see cheaper LEU prices, depending on quantity and availability of the requested material. Most important may be the date of commitment to a (long term) contract. All contracts concluded in the years before 2003 are earmarked with low prices and low escalation. New contracts may follow – at least partially – the uranium market price.

Russian material follows close to the market price development of feed and SWU component contained in the LEU, since production is mainly made from fresh material.

US-origin LEU has a higher value than Russian-origin LEU (the difference is in the range of at least US$ 500 /kg U), since its return possibility is still a most interesting option. In spite of this known fact US material does not follow the market price factors (plus surcharge) since the LEU is produced from existing, paid HEU stocks. It might be part of the USDOE policy to offer LEU for attractive prices to keep control on the RERTR program and to attract the use of LEU rather then HEU.

In principle Russian and US prices of LEU should be close together – with a premium of approximately 500 USD/kg for US material as long as return of spent fuel to the US is possible.

About one thing the experts from uranium trading are quite sure: There are currently no indications for a price relaxation. A moderate scenario would be that prices remain at the current high level. As a consequence the LEU price for research reactors should follow this development unless policy demands for discounted prices.

Well set are those reactor operators who have made long-term commitments in contracts which are not subject to the spot price development.
Summary

Phénix is the only fast breeder reactor in Europe available today for irradiation experiments. The neutron flux in the core is about ten times higher than those reached in other European research reactors (OSIRIS, HFR, BR2, R2,...) and has a fast neutron energy spectrum. The experimental conditions are described in this paper: flux, temperatures, monitoring and instrumentation.

Phénix also provides irradiation devices such as irradiation rigs, containing for example up to 19 pins, able to be inserted inside special carrier sub-assemblies. Different carriers have been developed: fuel, breeder, steel assemblies. More recently some moderating carriers have been designed in order to optimise transmutation conditions for some heterogeneous target experiments.

A hot cell, adjacent to the reactor is equipped with a wide range of non-destructive post irradiation examinations equipment for use with sub-assemblies and pins, for instance dimension measurements, eddy current testing, neutronography.

The core characteristics make the reactor particularly suitable for experiments in the following fields:
- transmutation tests which require high flux of high energy,
- fuel and material tests for the future reactors developed in the Generation 4 framework, especially all the fast reactors such as GFR,
- and also all the material tests for fusion programme (ITER), for Accelerator Driven Systems and even for Light Water Reactor (studies on internals for lifetime extension up to 60 years).

The Phénix experimental programme involves irradiations in these various fields. It is presented here together with the related schedule with emphasis on the future experiments. They will include irradiation of high minor actinide content fuels, structure materials and fuel for Gen IV GFR concept. The experiments are conducted in the frame of a strong international co-operation specially with Japan and USA.

The operation of the plant in 2004 is also described, and the future operating schedule of the plant.

As a conclusion the Phénix programme demonstrates the very good complementarity of the reactor with the other research reactors in Europe. It shows also a large international scope involving customers from many R&D organisations.

Introduction

The French fast reactor prototype Phénix, with a nominal 145 MWe power rating (350 MWth), was started up again in July 2003 for 6 cycles, following a major safety upgrading and renovation.

The specific experimental possibilities of a fast breeder reactor enable a wide scope of experiments. The Phénix experimental programme involves irradiations in the main following fields:
- Transmutation tests which require high neutron flux of high energy.
- Fuel and material tests for the future reactors developed in the Generation 4 framework, especially all the fast breeder reactors such as GFR.
To begin with, this report describes the main experimental conditions, the irradiation devices as well as the non-destructive post irradiation examinations equipment.

The current status of the Phénix experimental program, as well as the operation of the plan in 2004 are also presented.

**Experimental conditions at Phénix**

**Neutron flux**

Phénix stands out from the other experimental reactors (OSIRIS, HFR, BR2 type …) due to the very high neutron flux (more than ten times higher than the reactors mentioned above) with a relatively hard spectrum (high energy) characteristic of the fast neutron reactors.

The speed, or dose rates obtained are thus much higher than those in other reactors. Furthermore, the timely use of moderating materials can locally slow down the neutrons, thus benefiting from the high flux inherent to FBR, while increasing the effective cross-section of the neutrons. The following table summarizes the possibilities:

<table>
<thead>
<tr>
<th></th>
<th>Fast flux (n/cm²/s)</th>
<th>Thermal flux (n/cm²/s)</th>
<th>Dose rate</th>
</tr>
</thead>
<tbody>
<tr>
<td>PWR power reactor</td>
<td>1.3.10ⁱ⁴</td>
<td>0.9.10¹⁴</td>
<td>2 dpa/year</td>
</tr>
<tr>
<td>OSIRIS, HFR, BR2 type research reactor</td>
<td>2 to 4.10¹⁴</td>
<td>4 à 7.10¹⁴</td>
<td>3 dpa/year</td>
</tr>
<tr>
<td>Phénix reactor</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fast spectrum</td>
<td>4.4.10¹⁵</td>
<td></td>
<td>18 dpa/6-month cycle</td>
</tr>
<tr>
<td>Type 1 moderate flux*</td>
<td>3.6.10¹⁵</td>
<td></td>
<td>12 dpa/cycle</td>
</tr>
<tr>
<td>Type 2 moderate flux*</td>
<td>1.5.10¹⁵</td>
<td>3.7.10¹³</td>
<td>4 dpa/6-month cycle</td>
</tr>
</tbody>
</table>

* Types 1 and 2 correspond to the type of moderating carrier used

**Temperatures**

The irradiation of the inert materials (containing no fissile material) generally takes place at sodium temperature, between 380°C at the entrance to the core and 550°C average temperature at core outlet.

However, to cover the new needs shown by gas-cooled reactors, an innovative device is being developed which allows for irradiating specimens up to 1000°C. This device is based on the use of the power left in the materials (samples, sample-holders and heating boxes) by the radiation and the neutrons, and the dimensioning of the gas gaps providing the thermal insulation between the heating boxes and the sealed containers.

**Measurements in the reactor**

The experiments are placed in the reactor core after passing through the sodium storage tank, referred to as the drum. In theory, there is no possible connection with the outside. This constraint sharply limits the possibilities of instrumentation and measurement.
The instrumentation commonly used includes measurements of sodium temperature at the entrance to and exit from the experimental set-up, radiological analysis of the sodium at the exit and mass spectrometry or gamma spectrometry analysis of the leakage gases.

Two temperature measurement devices will be used in the high temperature setups currently being developed. The first is based on phenomena of the SiC reconstitution which occurs when the SiC is heated, after irradiation, to a temperature above or equal to its irradiation temperature. The second determines the maximum temperature reached during radiation, using a post-irradiation examination. This measurement device uses a stack of materials with increasing fusion temperature, arranged so that their successive fusion causes the stack to gradually decrease in height.

**Experimental setups**

The experimental sub-assemblies
These are sub-assemblies similar to the standard fuel assemblies in the Phénix core, to which different types of changes have been made:

- Nature of the structure materials (cladding, hexagonal wrapper, etc. . .),
- Nature or composition of the fuel,
- Geometry of the pin or pellet,
- Production process for a component, etc.…

Most of these sub-assemblies are fissile, however, there can be breeder sub-assemblies and even control rods.

This type of setup provides irradiation conditions which are very close to actual operation, yet which do not allow for any significant deviation from the standard design. The experimental sub-assemblies have primarily been used for experiments related to the development of the Fast Neutron Reactor system.

The irradiation rigs
The irradiation rig is complementary to the experimental sub-assembly. It is generally made up of a 40-mm diameter tube equipped with a sodium feed at the base and a locking and handling head which can contain a wide variety of experimental objects:

- Phénix or other geometry fuel pins, able to hold standard or experimental fuel
- Absorber pins
- Specimens of materials intended for mechanical tests (tensile strength, resilience, toughness, fatigue, creep, etc…)
- Transmutation target, etc…

The rings are irradiated inside special assemblies called “carriers”.

The special feature of these sub-assemblies is their central channel into which the rig is inserted. There can be different types of carriers depending on the irradiation conditions being sought (fissile, fertile, steel, etc…). A new type of carrier has just been designed and made for the irradiation program. This carrier creates a zone of moderate neutrons around the rig, in the core or core periphery. The goal is to improve the transmutation performances of certain experiments.

Highly innovative experiments can be conducted with these irradiation rigs. They offer a wide range of irradiation conditions both with respect to the neutron flow and its spectrum, and the irradiation temperature range. In addition, since their assembly and dismantling are possible in cell at Phénix, pins which have already been examined after one irradiation can be irradiated again.
Non-destructive testing in Phénix hot cell

The hot cell is equipped with very comprehensive equipment for the non-destructive testing of sub-assemblies and pins. The beds are designed to accept a large variety of pins. Although the hot cell is a sealed and shielded chamber, it is not intended for the examination of fuel pellets by direct means, i.e. rupture of the pin claddings must be avoided so as not to disseminate fuel and fission products.

Examination of the structure of sub-assemblies
Several measuring beds have been developed to measure the length and deformation of irradiated sub-assemblies and in particular:
- A bed for measuring the characteristic dimensions of the assemblies.
- A bed for measuring the flatness of wrapper tube faces.

These measurements are complemented by metallurgical tests on other CEA laboratories after cutting off samples into the cell.

Pin examinations
Theses analyses are to determine the behaviour of the pin cladding and the fuel pellets ant the interaction between them. The irradiated pins are also subject to several complementary tests.

- **Pin measurements**
  Deformations in the cladding, which has been subjected to neutron flux and high temperature, are measured.

- **Eddy current testing**
  This test enables the detection of cracks in the cladding as well as interactions between cladding and fuel. This bed consists of annular coils which move along the pin’s axis.
• **Neutronographic examination**
  This test, analogous to radiography, gives an image of the column of fuel pellets inside the pin. The neutron source is actually a tiny nuclear reactor based on uranyl nitrate. A burst of neutrons is produced for each examination by the supercritical effect of a mobile reflector in beryllium oxide.

• **Examination by spectrometry**
  The distribution of gamma emitters contained in the pins is measured.

**Phénix operation**

The French fast breeder reactor Phénix was put on commercial operation in July 1974.

Recent years have been marked by successive safety reevaluations which generated significant renovation work and in-depth inspections of the plant components and structures [ref. 1].

This renovation work resulted in obtaining authorization to start up again in January 2003 for six cycles of 120 EFPD. During these cycles, two of the three available steam generators are in operation. The plant thus operates at 2/3 rated power, which is approximately 140 MWe.

The reactor was started up in July 2003, for the 51st cycle, which ended on 8 August 2004, after 131 EFPD operating time. The diagram below details reactor operation in 2004. The total number of operating days is 202. The availability factor during the year was 73.6%.

Two planned shutdowns took place: in February for refuelling and from August to November for refueling, statutory inspection and maintenance.

During the later, the main outage were:

- Defects found on steam tube spacer grids of two Evaporator modules and one bimetallic welding of a SG steam header which required extension of module controls and welding repair.
- Replacement of Complementary Shutdown system and control rod n° 6 mechanisms.

At the beginning of 2005 the reactor is operating at steady nominal conditions. The remaining operating time (five 120 EFPD operating cycles) should be achieved by 2009 when the reactor will be shutdown.
Experimental irradiation program

The experimental irradiation program conducted at Phénix has a two-fold purpose. It responds to the separation-transmutation directive 1 in the 30 Dec. 1991 law on radioactive waste management, and it supports the research on future system design. Figure 1 shows the position of the main experiments in the reactor core during the 51st cycle.

Figure n°1
Position of the main experiments in the 51st cycle core
Figure 2 shows the status of the experiments in core at the end of the cycle.

Status on 24/02/2005

Figure n°2

Two experiments have reached the final irradiation objectives:

- The Matina 1A [ref 1] experiment, devoted to researching the behavior of inert matrices for transmutation, which was discharged.
  The results of the first non destructive testing (visual inspection, pin dimension measurement, neutronography) show no visual defect and no swelling of the pins.
- The Metaphix 1 experiment studying the behavior of metallic fuels, which was removed during the 2nd shutdown and is currently cooling prior to a series of non-destructive tests.

The irradiation period for the Caprix 1 experiment (high plutonium fuel: 45%), was increased from the initial period of 180 EFPD to approximately 350 EFPD (10 % at burn up).

Preparation of the future Camix-Cochix (heterogeneous mode minor actinide transmutation), Matina 2-3 (inert matrix irradiation) and Profil-M (neutronic data measurement) was continued during 2004.
The first design elements of the three Futurix experiments were completed:

- Futurix/FTA (FUels for Transmutation of transUranien elements in Phénix/Fortes Teneurs en Actinides – High Actinide Content) irradiation experiment is part of a collaboration between the DOE, ITU, CEA and JAERI (Japanese Atomic Energy Research Institute). The DOE seeks to irradiate four fuels, including two zirconium-based metallic alloys, with and without uranium (U-Pu-Am-Np-Zr and Pu-Am-Zr) and two nitride fuels, also with and without uranium ([U,Pu,Am,Np]N and [Pu,Am,Zr]N).
  The ITU plans for two Ceramic-Metal type fuels, with a macromass base of (Am,Pu,Zr)O2 dispersed in a molybdenum (Mo) metallic matrix.
  The CEA proposes two Ceramic-Ceramic fuels of the (Pu,Am)O2 particles type micro-dispersed in a magnesia (MgO) matrix.

- Futurix/MI experiment to evaluate the behavior under irradiation and in temperature of inert materials (with no fissile matter) identified as potential candidates for the structure materials and the fuel elements of future fast reactors with gas coolant. These materials are made of carbide type ceramics (SiC, TiC, ZrC, …) or nitrides (TiN, ZrN, …), and refractory metals (Mo, Cr, …).

- Futurix/Concept: study of the behaviour of new fuel concept for the gas fast reactors.

In addition, 2004 was the year of the decision to launch MATRIX, a new irradiation experiment whose objective is to study material behavior under irradiation and at low temperature of the structural materials (metals, ceramics) planned for the future systems (GCR, ADS, ITER). The CEA, the US Department Of Energy (DOE) and the European Union work together on this experiment. In 2004, work was primarily achieved on the definition of the experimental grid and design of the experimental setup.

Conclusion

51st Phénix operating cycle was successfully achieved in August 2004. 52nd cycle started in December after a planned shutdown for refuelling, maintenance and control.

The first series of irradiation experiments devoted to minor actinide and long life fission products transmutation is well underway.

Two experiments have completed their irradiation: Metaphix 1 (minor actinide transmutation with metal fuel) and Matina 1A (inert matrices). Their non destructive testing is being carried out.

The second series of experiments which includes testing of materials and fuel for Generation IV systems is being designed and manufactured.

Reference

ABSTRACT

Imperial College London (IC) operates commercially a 100 kW research reactor, and as site licensee is responsible for funding both operations and eventual decommissioning. With long lead times ahead urgent decisions on the future business options have had to be made in 2004/5 including choices on whether to move to early decommissioning, recognising the high costs entailed, or to pursue continuing operations involving life extension measures such as refuelling.

To develop a coherent overall approach strategy a financial risk driven programme was initiated to help define a robust transparent business and termination case for the reactor. This study was carried out in collaboration with a UK firm of financial risk experts, PURE Risk Management Ltd (PURE), working within a dedicated IC London reactor project team. This work evaluated immediate closure options due to financial constraints or life limiting failures, and options for continuing operation extending to 2028. Decommissioning and clean up were reviewed. Bespoke financial models created single value cost outputs and ranges of probabilistic net present values (NPV) for decommissioning costs and financial provisions to meet those costs at various levels of risk acceptance and regulatory compliance.

1. Introduction

Many changes have occurred to the UK nuclear industry over the past 15 years. Nuclear research groups have closed, fast reactor programmes ceased, and the United Kingdom Atomic Energy Authority (UKAEA) changed emphasis to decommissioning. However, there is now at least some understanding that more investment is required in nuclear skills at all academic levels. Unfortunately, requirements for research reactors in this programme are not seen as the highest priority. Most UK research reactors and associated facilities have now closed. In 1997, the 100 kW CONSORT pool-type reactor [1] became the last civil nuclear research reactor surviving in the UK. The Reactor provides a research tool to IC and other UK research establishments and has an important commercial role in providing services to a wider market such as environmental analysis, isotope production (including essential provision for the national measurements system of primary standards), fission chamber power measurement calibration (supporting 16% of all UK energy generation), pharmaceutical development and medical analysis (supporting development of better drugs across the world). The Centre currently also provides nuclear engineering training to Mechanical Engineering undergraduates, postgraduates at three other UK educational establishments, the Health and Safety Executive, and others, such as (in 2003) the Russian nuclear safety regulator. This places IC in an unrivalled position in the UK, and because of the customer base, before a decision is made to move to decommissioning, the transition will have to be handled with the care to preserve IC’s reputation.
Despite the above, IC has identified that the Reactor is not core to its current research and teaching activities. This leaves IC facing a stark choice - refuel in 2008-10 and continue business operations, or plan for earlier decommissioning. There are commercial considerations. The Reactor currently costs c. £600K per annum to operate - partly funded by providing commercial services to third parties. Despite every effort to close the revenue/costs the annual income is only c. £200K. The £400K deficit is met from College resources and is now considered unsustainable. Life extension refuelling would cost in the order of £1M and needs business justification. Other factors are that IC, as licensee, is legally obliged to pay for the decommissioning, demolition, cleanup and disposal an operation that will take some 4-5 years to complete. Cost estimates of £6.5-8M have been made. Part of PURE financial risk based objectives was refinement of these figures to demonstrate an understanding of the business impact and other risks on decommissioning strategy.

2. Influence of UK regulation

The Conditions of the 1994 Nuclear Site Licence granted by the UK safety regulator, the Nuclear Installations Inspectorate (NII) of the Health and Safety Executive (HSE) requires IC to develop and enhance QA documentation related to all lifetime phases including decommissioning. Although it is recognised that the low power and low overall source term of CONSORT make it a special case, it nevertheless operates under the same 36 Site Licence Conditions (SLCs) as all other UK nuclear installations including BNFL Sellafield. Under approved arrangements [2] for the SLCs, detailed arrangements for SLC35 on Decommissioning are not required for this relatively low-hazard plant. Any work, which is done during the operational phase, is covered under arrangements for SLC 22 (Modifications). Radioactive discharges for operations and decommissioning are regulated by a separate agency called the Environment Agency (EA).

3. Decommissioning planning

CONSORT is not unique in attempting detailed decommissioning planning and financial assessment. In UK law, there are a number of reasons why a proactive approach many years before ultimate shutdown, is adopted. The experience of CONSORT is illustrative.

In 1999, Imperial College undertook a Periodic Review of Safety (PSR), required by the NII every 10 years, and by following College internally imposed arrangements [2]. The latter followed the format and recommendations of IAEA’s guidelines on Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report [3]. Chapter 19 of the PSR report required an assessment of decommissioning plans. In addition, UK regulation places a number of Safety Assessment Principles (SAPs) [4], which are not relevant for this paper, except that of course, that safety and financial impact are interlinked. Pre-decommissioning strategy documentation is also required under a UK Government White Paper “Quinquennial Review of waste management on Nuclear Licensed Sites” [5] which requests operators to draw up strategies for the decommissioning of their redundant plant, including justification of the timetables proposed and the adequacy of the financial provision. Government policy is for HSE, in consultation with the EA, to review these strategies on a 5-yearly basis to ensure they remain soundly based. HSE aims to assess the proposed decommissioning strategy to determine whether it is adequately comprehensive, technically practicable and appropriately timed. In particular it considers whether the strategy is consistent with Government policy that ‘decommissioning should be undertaken as soon as it is reasonably practicable to do so, taking account of all relevant factors’, and also that the ‘hazards presented by the plant (or site) are reduced in a systematic and progressive way’. It also considers whether arrangements are in place to quantify the costs of decommissioning and to make available funds to undertake the work to the proposed timescales in order to assess ‘the adequacy of the financial provision being made to implement the strategy’.

IC document [2] gave an overview of the facility and its operational activities, described the (then current) broad plans for decommissioning the reactor in 25 years time. This report described safety and management regimes expected for the decommissioning phase of the reactor, disposal routes for
radwaste, and gave the best-estimate time scales for decommissioning and associated threats to such
timescales. At the previous issue, it was not aimed to provide the detailed technical assessment, which
will be done when the decision to plan decommissioning on a short-term horizon is made. Financial
discussions, together with some confidential aspects of radwaste disposal planning, were located in
separate confidential Appendices. It is recognised that this document will go into more detail about
what decommissioning will mean for IC at next issue. A subsequent 2002 report set out the findings
of a review of this decommissioning strategy by NII, in consultation with the EA.

HSE concluded in 2002 that cost estimates prepared by IC were indicative values based on its
knowledge of recent decommissioning projects at other research reactors. HSE believed that IC has a
broad understanding of the magnitude of its decommissioning costs but supports its undertaking to
refine the estimates. IC has noted a contingent liability in its accounts and was considering how to
translate this into a firm financial provision. HSE believed that IC’s financial position gives
confidence that decommissioning can be taken forward on shorter timescales if necessary.

4. Other drivers for decommissioning planning

Cost estimates of £6.5-8M had been made, but refinement was required, to include an understanding
of the impact of business risks. An initial contract was placed in 2002 to determine the true costs of
decommissioning, and break those down into work streams and cost the streams accordingly, to meet
the HSE requirements. These confirmed that there were a significant number of items that were
outside of IC’s control, and that had a significant ability to increase costs from a best estimate figure at
the lower end of this range. A small risk register was created at this stage.

IC is investigating the options for managing the spent reactor fuel, for which no UK options exist,
other fissile material, and intermediate level waste that will need to be transferred off the site and the
destination for this material needs to be established. In addition, decommissioning will generate low-
level radioactive waste, and Imperial College will need to ensure that there is an authorised disposal
route for this material. There are many uncertainties associated with these areas that have the potential
to impact the overall costs. A way of assessing the impact on costs was sought.

Early in 2004 IC agreed that the impact of financial risks on future business operations and overall
termination strategy was essential to help identify the best strategic option and level of costs involved
to deliver that option top the satisfaction of shareholders, regulators and other stakeholders.

5. The 2004 review of financial cost / Financial risk of nuclear business services and
operations and a refuelling and decommissioning strategy for consort

CONSORT financial risks affecting 5 strategic options each with full plant decommissioning were
identified and probabilistically modelled. This required creation of a comprehensive risk register that
described key income and cost components of the business model. In addition a further 4 strategic
options were assessed using a ‘no refuelling’ condition. Dynamic model outputs delivered Net Present
Value forecasts at a 3% discount rate. Least cost strategic options were assessed. A decommissioning
module delivered cost distributions, and enabled annual financial provisions necessary to meet those
costs to be calculated. From these results College senior management were able to assess a range of
see acceptable costs figures consistent with acceptable risk management, financial accounting and
insurance standards. Financial models are highly flexible and transparent. Insights provided critical
risk information that could lead to unplanned decommissioning occurrence.

The PURE/ICRC project led to 3 key findings essential to ICRC CONSORT reactor planning:

• A general strategy with no refuelling and efficiency improvement through additional cooling and
optimisation is shown to be financially dominant
A mid-term planned closing date of 2017 is shown to give the best net present value consistent with income assumptions (from ICRC business and external funding opportunities for decommissioning) and technical life of the reactor with cooling; the approach is flexible, subject to review of assumptions.

An improved basis for covering the costs of decommissioning has been presented, through a rationale for provisioning for planned decommissioning which has been agreed with the Imperial College London auditors, and a risk model being used in discussions with potential nuclear insurers to cover further identified and quantified financial risks related to closure earlier than planned.

6. Risk Register & risk mitigation

A comprehensive risk register was created with 130 separate defined risks. 70 risks were operational risks and 60 risks associated with the business plan which are probability-based input variables and added to the deterministic model using CrystalBall® software to create a stochastic, risk based model.

Four risks defined by the Risk Register and the modelling runs could cause early unplanned decommissioning. These risks have low probabilities of occurrence. But if they do occur, earlier decommissioning will be inevitable, without sufficient opportunity to fully provision for decommissioning costs. These risks are: loss of licence; significant accident; safety case obsolescence, and ‘Green’ pressures leading to early closure. ICRC is mitigating these risks.

7. Main findings of the financial risk assessment

The overall conclusions and commentary are as follows:

- A robust strategy for the CONSORT reactor is to continue commercial operations, not to consider refuelling as an option, to investigate the use of artificial cooling and optimisation to improve reactor operational reliability, and to plan to decommission in the mid term, beginning around 2017.
- Decommissioning costs for a target date of 2017 and the financial provisions required to fund them at the 50 percentile expected cost are approximately £8.0 million in 2004 pounds. This assumes the decommissioning would be completed in 2021. Annual provisioning to meet this figure is approximately £400K based on an expected net fund growth rate of 3%. Provisioning up to the 65% level of the cost distribution would require an annual set aside of £412K.
- The cost of decommissioning will be accounted for by provisioning in Imperial College London annual accounts. The basis for this provision will be reportable to the external auditors and HSE (NII).
- Immediate closure in 2007 does not appear to be viable. There is a difficulty obtaining timely regulatory approval for such action (e.g. safety cases). Outstanding issues of intermediate level waste (ILW) management and spent fuel reprocessing will not be resolved in the short-term and these are in any case outside the control of Imperial College London. Early closure will not permit time to build up a reasonable decommissioning fund, nor to resolve possible funding from the NDA.
- A critical issue to any of the options studied is the ability of the business plan to deliver the revenues upon which the plan was predicated. Adoption of a mid-term option as foreseen will provide an opportunity to test if these revenues develop. A mid-course correction can be considered if revenues develop differently. The business plan projects operational surpluses which, if achieved, could make selection of a later date for decommissioning appropriate.
- A mid-term option also allows time to see if the potential benefits of funding liability externally by the Nuclear Decommissioning Authority (NDA) can be developed. The expected value of partial external funding distribution anticipated contribution of approximately £3 million. It should be noted that the distribution has a wide range, from zero to 100% of external funding. For Imperial College London accounting purposes no NDA assistance was assumed.
- The 2017 decommissioning strategy is based on planned closure. External insurance and/or self-insurance need to be considered to cover the risk of premature closure and enforced earlier decommissioning. At best, insurance will only cover fortuitous mechanical events.
• In addition, two alternative strategies for no refuelling with cooling were evaluated. These had decommissioning start dates of 2012 and 2028, with their own decommissioning costs and provisions.

8. Conclusions - How Imperial College interpreted the assessment

In January 2005, the Management Board (MB) concluded that immediate closure in 2005-2007 does not indeed appear to be viable due to factors outside of its control. The MB noted that the 2017 decommissioning strategy is based on planned closure, and represented lowest Net Present Value cost to IC. The closure date for the purposes of provisioning for the 2003/4 financial accounts was brought forward from 2028 to 2017. However, they also identified that the Reactor is not core to its current research and teaching activities, and accepted a recommendation that the most appropriate option was to develop an exit strategy – a pre-decommissioning planning project – over the next three years, so that IC could finally shutdown in 2008, if it chose to do so.

The model sensitivity to the business plan to deliver revenues has led to significant efforts to engage external users of the Reactor to secure additional funding in recognition of the strategic uses they make of the Reactor. In this regard stakeholders have asked to see an independent assessment of the business case before deciding to invest. The financial analysis described has been crucial in influencing stakeholders that the reactor will likely close if support is not given. A recommendation has now been made to support the development of a Supporters’ Club, albeit in parallel to development of an early exit strategy into decommissioning. Initial indications are that Supporters will provide a significant proportion of the £500K IC is seeking to realise for 2005/6.

Subject to the business case developing in line with the plans for the Supporters’ Club and improvements to the business portfolio, IC will undertake an annual review on the Reactor Centre’s financial performance and its progress on decommissioning preparations. It is now up to the Supporters to persuade IC to keep the Reactor open for the provision of the services they need. Users will note that failure to form a successful Supporters’ Club will mean the loss of a unique facility.

The bespoke financial model produced is flexible and has been delivered on CD-ROM. The model will therefore be re-run by IC to examine the impact of future change or data refinement. Review of the assumptions and outcomes of the model will be undertaken annually. The no-refuelling options recommended do not preclude decisions to close the reactor earlier than planned if the business plan does not deliver sufficient income. However, closure earlier than planned would leave a shortfall in provisioning which would need to be met in the four to five years following an early closure decision.

9. References

ABSTRACT

As a result of unstable swelling (pillowing) experienced during tests of U-Mo dispersed in aluminum under high-power conditions in 2003, the programs to develop, qualify, and license U-Mo dispersion fuels carried out in the United States, France, Argentina, and Russia have been reoriented. During the past year, a major effort has been carried out to determine the cause(s) of the failures and to find potential remedies for the problem. The past year’s activities are summarized and updated plans and schedules for the qualification of U-Mo fuels are presented.

1. Introduction

This paper is the third in a series summarizing the status of work in the U.S., France, Argentina, and Russia to develop, qualify, and license U-Mo fuels [1,2]. Two years ago, plans were in place to perform qualification irradiations of full-sized fuel assemblies containing U-Mo dispersion fuels with densities up to 8 gU/cm³ in plates and 5.4 gU/cm³ in tubes. Expectations of success were high. By the end of 2003, however, it had become clear that U-Mo dispersion fuels had definite limitations under some conditions owing to a number of cases of unstable swelling (pillowing) experienced during irradiation tests in several countries. Last year’s paper reported a revamping of the various development programs to search for a solution to the unstable swelling problem being experienced in the dispersion fuels and to accelerate work on the development of fuel plates and pins containing solid U-Mo fuel meat (monolithic U-Mo fuel). It was projected that, at best, U-Mo fuel qualification would be delayed until around the end of 2010.

This past year has seen a number of changes in each country’s development program, as well as some significant organizational changes. The advent of the U.S. Global Threat Reduction Initiative (GTRI) last May resulted in the incorporation of the U.S. RERTR program into the broader GTRI program, along with the U.S. and Russian fuel repatriation programs. The GTRI goal of converting all Western-designed research reactors by 2013 and Russian-designed research reactors by 2014 resulted in increased pressure (and funding) for the U.S. RERTR program to accelerate qualification of the necessary fuels. The French U-Mo program is now focused on qualification of a U-Mo fuel for the...
Jules Horowitz Reactor (JHR). The Russian program has experienced an apparent failure of one or more of the U-Mo dispersion fuel mini-pins being irradiated in the MIR reactor. As a result, the irradiation of full-sized pin-type and tube-type fuel assemblies in MIR is currently on hold. Irradiation of a U-Mo dispersion fuel element by the Argentine CNEA in its RA-3 reactor has been delayed.

Along with reorganization of program goals and schedules, the various programs have concentrated on searching for an understanding of the unstable swelling phenomenon and for a way to eliminate or, at least, mitigate the problem. This paper summarizes the current situation in the four programs.

2. U.S. program

The U.S. RERTR program has focused its efforts during the past year in three principal areas: (1) understanding the causes of the unstable swelling experienced in a number of irradiations of U-Mo dispersion fuels and ways to eliminate or mitigate the problem, (2) preparing for another set of miniplate irradiations in the Advanced Test Reactor (ATR) to test potential improvements to the dispersion fuel and to significantly increase the amount of irradiation data for monolithic U-Mo fuels, and (3) to improve and scale up the monolithic U-Mo fuel plate fabrication process. Even though significant effort is being expended on U-Mo dispersion fuel now, the long-term emphasis of the RERTR program is on monolithic U-Mo fuel because only it can provide the >8 gU/cm³ uranium density needed to convert a number of reactors in the U.S. and abroad. Development of irradiation-behavior models and codes is also being pursued both independently and in collaboration with the Argentine and French programs.

2.1. Investigations of unstable swelling in U-Mo dispersion fuel

As reported at the 2004 RERTR International Meeting in Vienna, a concerted effort has been made to understand the cause of the unstable swelling problem and find a way to fix it [3]. Although the fundamental mechanism is not fully understood, the most important factors influencing the unstable swelling have been determined from the results of examinations of irradiated fuel samples. For example, it is certain that the unstable swelling results from fission gas generated in the fuel/aluminum interaction product. Although the extent of the interaction does not appear to be a controlling factor, a certain amount of interaction product is required to allow the growth and interconnection of large fission gas bubbles. In addition, the composition of the interaction product is very important—fission gas bubbles appear to be associated with an interaction-product phase having a larger aluminum concentration than those of the familiar UAl₃ and UAl₄ compounds. This is not surprising, since UAlₓ dispersion fuel is extremely stable under irradiation. The onset of unstable swelling is correlated to the local burnup (amount of fission gas present) and very strongly to the local fission rate; in fact, there appears to be a rather sharp fission-rate threshold for the onset of unstable swelling. The onset of unstable swelling does not appear to be correlated to the fuel-meat temperature, however.

A variety of tools have been employed in the search for a way to eliminate or mitigate the unstable swelling problem, including study of postirradiation examination (PIE) and out-of-pile experiment results and theoretical investigations. One potential remedy is to include silicon in the aluminum of the matrix. As Hofman et al. reported [3], the first clue that addition of silicon to the matrix aluminum is a possible remedy came from the observation that the amount of fuel/cladding interaction was less for Al 6061 cladding, which contains silicon in solution, than for AG3NE cladding, which does not. Also, the literature revealed that adding silicon to uranium-aluminum alloy suppresses the conversion of UAl₃ to UAl₄. Other possible remedies include the addition of elements such as zirconium, niobium, or titanium to the fuel, or even more effectively, using them to coat the fuel particles so the added element is at the fuel/aluminum interface where the interaction is occurring. Theoretical studies have confirmed the stabilizing effect of such additions to the aluminum matrix and to the U-Mo fuel. Another remedy being pursued is to replace the aluminum matrix with a magnesium matrix; a number of fabrication problems must be solved, however. As reported below, these possible remedies are

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3 Owing to circumstances related to the reorganization of MINATOM into ROSATOM, a Russian author was not identified in time for ROSATOM approval to be obtained. The information on the Russian program was prepared by author J. Snelgrove from knowledge gained as the U.S. technical coordinator of the joint U.S./Russian fuel development work and was edited by one of the Russian collaborators.
being investigated by the other national programs as well. A number of bilateral and multilateral discussions have occurred during the past year, and the U.S. RERTR program is participating in a number of bilateral collaborations.

2.2. Irradiation tests
To investigate the effects of various additions to the fuel and matrix, to gain more information about the effect of fission rate, especially as a threshold parameter, and to provide the first data from ‘real’ monolithic U-Mo plates, a new set of miniplate irradiations will be performed in the ATR. The fuel plate size and the irradiation rig are based on those used for RERTR-4 and -5. RERTR-6 is a scoping test at moderate fission rate and burnup scheduled to start in April 2005. Only two different fuel alloys, U-7Mo and U-10Mo, are being used in this irradiation test. The dispersion fuel plates employ a number of different matrix aluminum alloys: 4043, 6061, and two binary compositions—0.5 and 2 wt.% silicon—in an attempt to counteract the unstable swelling detailed in the previous section. An unalloyed aluminum matrix is also being used as a known control. The monolithic fuel in the RERTR-6 test was fabricated using friction stir welding (FSW). Fuel foils of two different thicknesses, 250 and 500 μm, have been included in this experiment. RERTR-7 will be a similar scoping test at higher fission rate and burnup; it is expected to start several months after RERTR-6. In addition to the fuel types in RERTR-6, it will include dispersion fuel with a magnesium matrix. It will also contain monolithic fuel plates assembled by hot isostatic pressing (HIP) and transient liquid phase bonding (TLPB) if these techniques can be sufficiently developed in time. It is anticipated that RERTR-8 will be a short-duration, high-power test intended to provide data for peak, or off-normal, irradiation conditions.

In addition to the miniplate tests, which are intended to allow selection of the most-promising candidates for qualification, a series of full-sized plate tests are planned to confirm the results of the miniplate tests and to determine if there are any problems related to plate size, especially for the monolithic fuel. A series of full-sized assembly irradiations will follow. A major change in the U.S. program is its plan to make extensive use of the ATR for the full-sized plate and assembly irradiations instead of relying solely on foreign reactors.

2.3. Monolithic Fuel Fabrication Development
During the past year, considerable progress has been made in the fabrication of monolithic fuel plates by FSW [4]. While it is continuing to investigate other fabrication techniques, HIP and TLPB, the RERTR program’s major emphasis in the coming year will be on scale-up of the FSW process. Foil-production scale-up is an issue that also must be addressed no matter what plate bonding process is used.

2.4. Schedule
The RERTR program’s overall fuel development schedule is shown in Fig. 1.

3. French Program
As the reference fuel solution for the Jules Horowitz Reactor (JHR) project, the development of an optimized and reprocessable LEU fuel is an objective of first importance for the French CEA [5]. The schedule of the project requires development of a suitable LEU fuel by the end of 2007 and the qualification of this fuel before the end of 2009. So, in order to reach these goals, the French CEA has launched a large program on the development of dispersion and monolithic U-Mo fuels, including out-of-pile investigations, manufacturing, irradiation and postirradiation programs, and code development. Manufacturing aspects are being developed in close collaboration with CERCA. Figure 2 presents a schematic view of the RJH fuel element.
3.1. Out-of-pile program
The first development phase consists of out-of-pile investigations. A full scientific program has been created to increase our understanding of the U-Mo/Al interaction and to develop promising solutions to avoid inappropriate behavior during irradiation. Three items are being pursued in parallel:

- Thermodynamic studies to determine, through the U-Mo-Al ternary system, the phase equilibria at different temperatures;
- Out-of-pile metallurgical studies on diffusion couples to get a better understanding of the mechanisms of the U-Mo/Al interaction phenomena and to establish the parameters which could prevent this reaction; and
- Simulation studies to take into account the effects of irradiation using high-energy ion irradiation, typically 100-MeV xenon, in order to simulate the damage created by fission products. These experiments will be carried out on both the interdiffusion layer and the fuel meat from fuel plates.

The details of this program and the preliminary results of these investigations are presented in a later paper of this session [6].

3.2. Irradiation program
The irradiation program is dedicated to evaluate optimized irradiation behavior on both U-Mo dispersion fuels and U-Mo monolithic fuels. As was done in the frame of the French UMo Group, these experiments will be made using full sized plates for recording the whole information package on manufacturing and irradiation aspects.

The first irradiation experiment (IRIS-3 experiment) is planned for testing parametrically the effect of adding silicon to the aluminum matrix. Two sets of plates were produced by CERCA using, respectively, Al2%Si and Al0.3%Si as the matrix. These plates began their irradiation in the OSIRIS reactor at the beginning of January 2005.

The next irradiation (IRIS-4 experiment) will evaluate a solution which could avoid the U-Mo/Al reaction by means of a protective layer on the particles themselves. Out-of-pile studies are providing the basis for this approach by defining the protection layer and a way to produce it.

In parallel, the monolithic fuel solution will be tested in a U.S./French experiment on full-sized plates. Prototype U-Mo monolithic plates will be manufactured by Idaho National Laboratory (INL) and by CERCA with the technical support of CEA. They will be simultaneously introduced in the OSIRIS reactor early in 2006 (IRIS-5 irradiation). The first feasibility steps for manufacturing full-sized U-Mo foils are in progress in France and the U.S. Note also that CEA, CERCA, and TUM in Germany have signed a Memorandum of Understanding to support U-Mo fuel development.

The schematic schedule of this program is presented in Fig. 3, along with that of the RJH Fuel Project. Some more details of this program—manufacturing aspects and the irradiation schedule, for example—are presented in the next paper of this session [7].

3.3. Code development
In parallel, CEA is developing a 2D thermo-mechanical code, called MAIA, for modeling the behavior of U-Mo dispersion fuel. MAIA uses a finite-element method to solve the thermal and mechanical problems. Physical models from the ANL code PLATE are used to evaluate the fission product swelling and the volume fraction of the U-Mo/Al interaction product. The swelling produces strains in the meat, which are imposed as the loading for the mechanical calculation.

The fuel meat can be treated either elastically or plastically, and the Al alloy cladding can be either treated elastically or allowed to creep. Mechanical characterizations are in progress to improve the behavior modeling through a better knowledge of the material properties of the meat and the cladding. MAIA has been validated against the irradiations IRIS1, IRIS2, and RERTR-3, and rather good agreement with PIE results has been obtained. The results obtained are presented in later in this session [8].
4. Argentine program

During 2004, the CNEA program for the development and qualification of high-density fuels for research reactors was strongly influenced by the unexpected unstable swelling experienced in qualification experiments performed by the other three programs discussed in this paper. Those experiments included full-sized fuel plate and miniplate irradiations.

Because of the above-mentioned results, during this year the CNEA program was mainly focused on the investigation of the interaction between U-Mo and aluminum, the resultant interaction products, and the possible remedies to avoid the limiting phenomenon. The main activities were performed in the following fields: (1) chemical diffusion experiments using diffusion couples, (2) thermal compatibility studies using compacts, (3) development of technological solutions to reduce the amount of interaction, and (4) development of an alternative fabrication process to obtain very-high-density fuel plates.

4.1. Diffusion experiments

As discussed earlier, experimental results had shown that the addition of silicon or, perhaps, magnesium to the aluminum of the matrix could have a beneficial effect on the characteristics of the interaction layer in the U-Mo dispersion fuel. Out-of-pile diffusion experiments were performed to evaluate the effect of the above-mentioned additions [9]. These experiments were carried out at 550ºC and 580ºC, using aluminum alloys with magnesium and/or silicon as main alloying elements. Couples consisting of Al, U-Mo, and Al alloy were used to evaluate the influence of the alloying element and also to monitor the reproducibility of the experiments.

According to [9] the main results obtained up to now may be summarized in the following points:

- The addition of Mg to pure aluminum does not produce any important change in the characteristics of the U-Mo–(Al+Mg) interaction zone. The composition of the layer is similar to the one obtained with pure aluminum.
- Silicon addition presented an encouraging result, as a (U-Mo)(Al, Si)\(_3\) phase appeared in the interdiffusion layer close to the aluminum. According to [9], this phase is similar to the one reported in silicide dispersion fuels in an aluminum matrix, which has a satisfactory behavior under irradiation.
- The interaction layer of the Al+Si alloy is not affected by the decomposition of the \(\gamma\)U-Mo as happens when pure aluminum is used.

Figure 4 shows the result of the diffusion couple test with U-Mo/Al-Si. In this case, the interaction layer (IL) shows the above-mentioned (U-Mo)(Al, Si)\(_3\) zone. The next steps of the program in this field will include diffusion experiments under irradiation in the CNEA’s RA3 reactor.

4.2. Thermal compatibility studies

Compacts obtained by standard powder-metallurgical fabrication procedures were also used to evaluate the influence of different alloying elements for the aluminum matrix of U-Mo dispersion fuel plates. Thermal compatibility tests were performed, submitting the compacts to thermal treatments reproducing different conditions of the fuel plate fabrication process. Temperature and time-at-temperature were the parameters initially considered. Measurements with a dilatometer and X-ray diffraction are being considered, among other techniques, for evaluation of the tests. Initially the tests were performed at 425ºC and 485ºC. Although the results are not yet fully evaluated, preliminary observations seem to show that the influence of temperature is consistent with results of the irradiation experiments (growth kinetics of the interaction layer). Additional tests using other matrix materials and fabrication conditions are scheduled for this year.

4.3. Development of technological solutions

Two main technological alternatives to improve the feasibility of U-Mo utilization as a fuel material have been considered [10]. The first one is the use of particles coated with a multilayer diffusion barrier to reduce the amount of U-Mo/Al interaction in the dispersion fuel. Different materials, such as Al, Ge, Al-Si, and Mg-Al, are being considered for this purpose. The techniques that are being
tested are dip coating and chemical vapor deposition. Both techniques allow one to obtain layers a few microns thick. The coating process could be incorporated as one of the last steps of the powder production process.

The other technological development is an alternative for the so called U-Mo monolithic fuel. To eliminate the presence of the undesirable interaction zone, Zircaloy cladding is being considered. The first rolling tests are being performed. If the results of these tests, and also of thermal compatibility and corrosion tests, are satisfactory, the next step of the program will be the fabrication of miniplates to evaluate the performance of this solution under irradiation.

4.4. Monolithic fuel fabrication development
CNEA has been working also on the development of the FSW process to obtain monolithic U-Mo plates with aluminum cladding [11]. At the present stage of the development, the main task has been the optimization of operative parameters and the selection of the adequate tests to qualify the technique. It was possible to obtain satisfactory bonds with no defects or debonding at the interface for U-Mo foils. The method has already been used for the fabrication of diffusion couples. Considering the encouraging results obtained up to the present, one of the next steps of the program will be to evaluate the scale-up of the process to obtain full-sized fuel plates.

5. Russian program
The Russian RERTR program has continued to work on development and testing of LEU fuel based on U-Mo alloys for two fuel assembly designs—one with tubular-type and the other with pin-type fuel elements. Work has continued on development of fabrication techniques for monolithic U-Mo fuel pins, on irradiation in the MIR reactor and PIE of U-Mo dispersion mini-pins, on irradiation of full-sized U-Mo dispersion pin-type fuel assemblies in the WWR-M reactor, and on PIE of the failed U-Mo dispersion fuel tubes irradiated in the IVV-2M reactor to ~60% average burnup. The principal accomplishments in three of these areas are summarized below. A much-more-detailed description of the Russian program results is presented in another paper later in this meeting [12].

5.1. Irradiation and PIE of mini-pins at RIAR
The irradiation of one of the capsules containing mini-pins in the MIR reactor at the Research Institute of Atomic Reactors (RIAR) continued until late in 2004, when fission products were detected in the experimental channel containing the experimental device with the highest-burnup pins. The average U-235 burnup in the mini-pins ranged between ∼60 and ∼65%. This capsule has been cooling in the MIR pool, and disassembly and visual inspection is scheduled to take place in March 2005. The irradiation of the second capsule containing mini-pins was resumed and was completed early in February 2005 with average fuel burnup of the pins ranging between ~20 and ~60%. The PIE of the pins removed after 20% burnup has been completed, and the pins appear to have behaved normally in every way.

5.2. Irradiation of two full-sized pin-type fuel assemblies in the WWR-M reactor
The lifetime test of two full-sized pin-type fuel assemblies are in progress. Late in February 2005, the average burnup in the fuel assembly containing UO₂-Al dispersion fuel (being irradiated as a control) was ~35%, and that in the fuel assembly containing U-Mo dispersion fuel was 12%. The tests are planned to reach an average burnup of 60%.

5.3. PIE of failed fuel tubes at IRM
The basic PIE of the U-Mo dispersion fuel tubes that failed at around 60% burnup in the IVV-2M reactor has been completed at the Institute of Reactor Materials (IRM). The nature of the failure appears to be the same as that of the failures that have been experienced in plates. Gamma scans of the tubes indicate the presence of a rather high flux gradient across the failed tubes during the latter part of their irradiation. Calculations are underway at ANL to determine if the local environment of the test assembly would have produced such gradients.
5.4. Irradiations of full-sized pin-type and tube-type U-Mo dispersion fuel assemblies

The design and experimental and technological investigations for an IRT-type fuel assembly with pin-type fuel elements has been completed. The fabrication of experimental full-sized fuel assemblies of both tube-type and pin-type design is now planned for lifetime tests in the MIR reactor. These assemblies are prototypic of those which might be used to convert the WWR-SM reactor in Uzbekistan. Further work in this area is on hold, however, as the Russian program and its U.S. RERTR collaborators determine if changes in the planned irradiation program are needed as a result of the failures that have been experienced in pins and tubes.

6. Conclusion

The past year has seen significant changes in the four national programs as a result of failures of U-Mo dispersion fuels. On the one hand, tests are planned (and have even started in France) of possible ways to eliminate the unstable swelling problem, the threshold of which seems to be strongly related to the fission rate. On the other hand, greater emphasis is being placed on the development of monolithic U-Mo fuels, which may be immune to the unstable swelling problem and which are needed for the conversion of several reactors.

7. References


8. Figures

Fig. 1. Schedule of RERTR fuel development activities

Fig. 2. Schematic view of JHR fuel element

JHR fuel element

- Uranium load: 4.8 to 8 gU/cm³
- Active height: 600 mm
- Water channel: 1.84 mm
- Cladding: Aluminium alloy
- Structure: 3x8 plates
- Internal diameter: circa 37 mm
- External diameter: circa 95 mm
Fig. 3. Schematic schedule of the JHR Fuel Project

Fig. 4. Result of diffusion couple test with U-Mo/Al-Si; the interaction layer (IL) shows the above-mentioned (U-Mo)(Al, Si)₃ zone
THE FRENCH UMo GROUP CONTRIBUTION TO NEW LEU FUEL DEVELOPMENT

J.M. HAMY
Framatome ANP(*)
69456 Lyon Cedex 06, France

P. LEMOINE
CEA Saclay
91191 Gif-sur-Yvette Cedex, France

F. HUET
CEA, Cadarache
13108 St-Paul-Lez-Durance Cedex - France

C. JAROUSSE
CERCA (**) Les Berauds, B.P. 1114, 26104 Romans Cedex - France

J.L. EMIN
COGEMA (AREVA Group) BP 4, 78141 Vélizy Villacoublay Cedex - France

(*) an AREVA and Siemens company,
(**) a subsidiary of Framatome ANP, an AREVA an Siemens company

ABSTRACT
The French UMo Group was based on a close collaboration between CEA and AREVA's companies strongly involved in the MTR field. The aim of this program was to deliver industrially a high performance LEU UMo fuel able to be reprocessed, and suitable for a wide range of Research Reactor, covering the expected needs for MTR next generation. Since 1999, the program has been focused on industrial aspects with the intention to deal with the whole fuel cycle: manufacturing, irradiation behaviour, fuel characterisation, code development and reprocessing validation. It has been based on the fabrication of full-sized U-7\%Mo fuel plates with a density up to 8 \text{gU/cm}^3. The dedicated and advanced R\&D means provided by the CEA have been used intensively with the contribution of HFR and BR2 facilities in Europe. This paper presents a synthesis of the program and the corresponding significant results obtained. These results have played a major role as regards the UMo dispersion fuel qualification route by issuing, for the first time, evidence of severe performance limitations. Consequently, the global international effort to develop and qualify a high density LEU UMo fuel has been definitively rerouted and forced to overcome these discrepancies by exploring new technical solutions. A French extended program sustained by a CEA and CERCA collaboration has been launched in 2004 in order to develop a suitable UMo fuel solution. UMo dispersion and monolithic fuel are both investigated through three new full-sized plate irradiations planned in OSIRIS.

1. Introduction
During the last five years, the French UMo Development program has brought a significant contribution to the international effort to qualify a new LEU fuel that allows the conversion of
research reactors by achieving high performance and offering reprocessing capabilities. On the basis of the US RERTR program leaded by ANL, and the numerous parametric studies using small plate irradiation experiments and examinations, the LEU U-Mo dispersion fuel was identified as the more suitable candidate.

Taking into account this statement issued from the R&D side, the French UMo Group (FUMoG) has resolutely oriented its development program to a global approach covering the whole fuel cycle taking advantage of a strong synergy between CEA and AREVA. In that way, this program encompasses manufacturing aspects, irradiation of full-sized plates, characterization of irradiated fuel, code development for modelling the behaviour of UMo dispersion fuel and reprocessing validation [1].

Advanced R&D facilities has been dedicated to this program by the CEA : IRIS irradiation device for full-sized plates, OSIRIS for the irradiation experiments, LAMA and LECA hot cells laboratories for PIE, ATALANTE facility for reprocessing validation of spent fuel, with the additional contribution of HFR and BR2 facilities in Europe.

A synthesis of the different technical domains managed through this program is presented bellow, together with the corresponding significant results obtained. The unexpected performance limitations that have been revealed for the first time are also expressed with the direct consequence to necessarily reroute the international effort to develop and qualify high density LEU UMo fuel. The new French UMo extended program that will investigate new solutions is presented with the corresponding time schedule.

2. Manufacturing aspects

Since the beginning of the development program, CERCA has played a major role, acting intensively to supply diverse UMo full size plates for the different irradiation experiments implemented, producing the plates in a very efficient way that has allowed the continuation of the program in accordance with the expected milestones [2], [3]. Early in 1999, full-sized UMo plates have been manufactured for the first time by CERCA and were dedicated to IRIS-1 and UMUS irradiation experiments. Theses plates were based on UMo powder produced by grinding that induced a mean porosity higher than 10%.

In 2002, CERCA was able to produce a new set of full-sized plates for IRIS2 and FUTURE experiment in a very short time. Theses plates were produced using atomized UMo powder and the associated mean porosity was found at a level below 3 %.

Along the program, CERCA has spent a sustained R&D effort to comfort and adapt its proprietary advanced process for plate manufacturing, initially developed for highly loaded U3Si2 plates with densities up to 6 gU.cm\(^{-3}\). Progress has also been made regarding quality control methods available. CERCA has clearly demonstrated its ability to manufacture full-sized plates and elements for high density (up to 8 gU.cm\(^{-3}\)) UMo dispersion fuel. Due to this large experience, CERCA is ready to extend this know-how and master complementary techniques needed to overcome the present performance limitations encountered (see section 6).

3. Main results of irradiation experiments

The French UMo development program has performed four irradiation experiments on full-sized plates with increasing surface heat flux and clad temperature (see Table 1). The values tested, as regards the heat flux, have been ~140 W/cm\(^{2}\) for IRIS-1 experiment, ~240 W/cm\(^{2}\) for UMUS and IRIS-2, and ~340 W/cm\(^{2}\) for FUTURE. The extensive program of post irradiation examinations has confirmed the very good behaviour of the high uranium loading UMo7 and UMo9 plates of IRIS-1, with a peak burn-up of 67% that corresponds to 240 full power days and a cladding surface temperature below 75°C [4]. The UMUS experiment was stopped after two cycles due to the failure of the 35%-enriched plates related to uncontrolled oxide (boehmite) thickness on the cladding [5]. FUTURE experiment was stopped at the end of the
second cycle by the detection of an abnormal thickness increase in the hot spot region (local burn-up of 33%). It revealed for the first time severe performance limitation of UMo dispersion fuel for higher irradiation flux. Large porosities appear in the U-Mo/Al interaction phase leading to an excessive swelling that is amplified in an irreversible process by the increase of the local temperature.

IRIS-2 experiment results recently obtained are presented in [6]. In spite of less severe operating conditions than FUTURE ones, IRIS-2 experiment was definitively stopped at the end of the fourth cycle. The maximum swelling admissible value of 250 µm, defined for safety reason, was reached for one plate at the end of the third cycle and at the end of fourth cycle for the two remaining plates. PIE examinations are consistent with the FUTURE results. Large pore developments are observed in the interaction product, that induces the pillowing of the plates.

The origin of the phenomenon is not completely established and different hypothesis are still discussed [6]. The decrease of the aluminium content in the interaction compound versus the increase of the temperature is pointed out. The role of the fission products is also mentioned.

In addition the results show that for IRIS-2 the pillowing of the plates corresponds to a fission density of $2.10^{21} \text{f.cm}^{-3}_{\text{UMo}}$ (consistent with FUTURE), but for IRIS-1 with lower irradiation conditions and grinded particles no pillowing was observed up to $4.8 \times 10^{21} \text{f.cm}^{-3}_{\text{UMo}}$.

All these unique and crucial results, which correspond to unacceptable restrictive limitations for the UMo dispersion fuel, were revealed for the first time by the FUMoG to the MTR community at the end of 2003. Complementary results have been given during the first quarter of 2004. These results have definitively imposed to reroute the international effort to develop and qualify a high density LEU UMo fuel and have forced to overcome these discrepancies by exploring new technical solutions (see section 6).

<table>
<thead>
<tr>
<th>Experiment</th>
<th>IRIS-1</th>
<th>UMUS</th>
<th>IRIS-2</th>
<th>FUTURE</th>
</tr>
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<tbody>
<tr>
<td>Reactor</td>
<td>OSIRIS (Fr)</td>
<td>HFR (Neth)</td>
<td>OSIRIS (Fr)</td>
<td>BR2 (B)</td>
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<tr>
<td>Number of full-sized plates</td>
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<td>2 / 2</td>
<td>4</td>
<td>2</td>
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<td>Grinded</td>
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<tr>
<td>Enrichment (% $^{235}\text{U}$)</td>
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<td>19.75 / 35.00</td>
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<td>Max. heat flux at BOL (W/cm²)</td>
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<td>Status of experiment</td>
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<td>Max. local burn-up at EOL (% $^{235}\text{U}$)</td>
<td>67</td>
<td>20</td>
<td>40</td>
<td>33</td>
</tr>
</tbody>
</table>

Table 1: French program irradiations on full-sized plates.

4. Reprocessing qualification

The definition and the validation of the UMo fuel reprocessing process is a key point to confirm the advantage of UMo fuel as regards the back-end aspect. Although the reprocessing feasibility of UMo has been demonstrated since 1998, a research and development program has been performed by COGEMA and CEA, as part of the French UMo development program, to study and adjust more precisely the operating condition to reach the industrial feasibility.

The process is based on a specific dissolution in nitric acid media in order to obtain a dissolution solution that allows the dilution into standard UOx type fuel dissolution solution. The resulting solution has to be compatible with an adaptation of the Purex process for the U and Pu purification.
The main aspect to be investigated is the solubility of Mo as a function of the temperature and the U, Al, HNO₃ mass fraction. The first tests have been conducted on unirradiated UMo powder. Then pieces of fresh UMo plates have been used for the dissolution tests [7]. The last results obtained on irradiated materials coming from the IRIS-1 experiment made in OSIRIS are presented in [8]. These tests performed at the ATALANTE facility (CEA Marcoule) show that only the Al concentration can limit the Mo solubility that affects the fuel dissolution rate to guarantee a solution free of precipitate. The recommended parameters have been defined. The dissolution kinetics of unirradiated and irradiated materials have been precisely investigated. Very similar values have been observed for both types of material. The interaction product UMoAlₓ does not influence the dissolution kinetic.

The validation of the operating parameter for the reprocessing the irradiated UMo fuel plates is one of the most important result issued by the French UMo development program. This is the confirmation of the favourable properties of UMo fuels as regards the key point of the back-end solution. COGEMA will pursue the optimisation of the process and will expertise the new solutions that are envisaged to overcome the present performance limitation of UMo fuel (see section 6) in order to make sure that the reprocess criteria will still be satisfied.

5. Thermo-mechanical simulation of UMo behaviour

Characterization of fresh fuels, and post irradiation examinations based on IRIS-1, FUTURE and IRIS-2 have issued an important data base that allows the definition of the proper physical laws applicable to UMo dispersion fuel behaviour.

The thermo-mechanical MAIA, that is a 2D finite element computational code developed by CEA, is used to combine the physical laws and determine the global behaviour on the fuel [9]. The meat is assumed to be a homogeneous material. Thermal and thermo-mechanical calculations take into account swelling effects (minored by as-fabricated porosities), growth of the interaction compound and diffusion of the Al matrix. The improvements introduced in the last version of MAIA are presented in [10]. The model of the interaction compound evolution is issued from the PLATE code developed by DOE-ANL and the mechanical model have been extended and made more consistent with the expected results. IRIS-1 and RERTR-3 post-irradiation results have been used to validate MAIA with a good accordance as regards interaction layer thickness and volume fractions evaluation. MAIA has been applied to do analysis on one IRIS-2 UMo plate, in order to anticipate the final PIE results. Mechanical analyses have been done considering various constitutive laws to facilitate the understanding of the results [10].

MAIA is a useful tool that has been made consistent with the different results obtained as regards UMo dispersion fuel behaviour. It is an interesting feature of the French UMo development program that offers a synthesis of the physical results obtained and helps for the analysis and the understanding of the in-reactor fuel behaviour.

6. The French UMo Extended Program (CEA/CERCA)

As the reference fuel solution for the Jules Horowitz Reactor project –JHR project– the development of an optimised and reprocessing LEU fuel is an objective of first importance for the CEA. In order to strengthen and contribute actively to the international development of a suitable UMo fuel solution, CEA and CERCA has launched a large UMo development and qualification program [11]. After five years of UMo development CEA & CERCA agreed to pursue their investigation for contributing and challenging the nowadays difficulties encountered along the UMo fuel qualification way. Sharing fruitfully their motivation as well as skills CEA and CERCA signed in June 2004 a collaboration agreement for studying and developing a suitable UMo fuel solution. This section states the technical selected steps of the CEA program including the CERCA developments on manufacturing aspects, and gives the associated preliminary schedule.
6.1 Out of pile investigation
The first development phase has been defined as an out of pile phase investigation. This basic step will bring the fundamental knowledge to further understand (U,Mo)Al\textsubscript{x} interaction product formation which is led, at least, by the temperature. Two sub phases have been identified by CEA. Within the determination of the U-Mo-Al ternary diagram system at various temperatures, thermodynamic information will be obtained. Compounds identification associated with their characterization will determine how the U-Mo-Al reaction is led. In another hand, diffusion couples were produced for understanding the kinetics and the direction path of the interaction. With the help of such heat treated samples, various solutions could be easily investigated and the most promising idea selected for the next evaluation phase. The preliminary results are given in [12]. According to these results and prior publication as well [13] Si addition incorporated into the aluminium matrix could be of interest for reducing the (U,Mo)Al\textsubscript{x} interaction. Nevertheless promising "cold solution" is just track which has to be investigated up to the end for the final in pile verdict.

6.2 Irradiation experiments
Supported by the studies stated above, the irradiation program –See figure 1– is dedicated to evaluate optimized UMo dispersion fuels and UMo monolithic as well. As already done for recording the whole information package –Manufacturing and irradiation aspects– these experiments are going to be made using full size plates. The first experimentation is for testing parametrically the Si effect addition into the aluminium matrix. Two sets of plate were produced by CERCA using respectively Al\textsubscript{2}%Si and Al\textsubscript{10.3}%Si aluminium matrix. Plates were introduced in CEA OSIRIS reactor at the beginning of 2005 (IRIS-3 experiment). The following experimental irradiation (IRIS-4 experiment) will evaluate a solution which could avoid the (U,Mo)Al\textsubscript{x} reaction by means of a protective layer on the particles themselves. Out-of pile studies are sustaining this approach by defining the protection layer and the experimental way of production. Finally, monolithic fuel solution which seems to be a good candidate [14] will be tested with IRIS 5 irradiation. Our goal, at this step, is to produce monolithic full size plate prototypes for firstly learning the manufacture of such fuel concept and also for evaluating the fuel behaviour under irradiation. Monolithic UMo Plate prototypes manufactured by Argonne National Laboratory will be simultaneously introduced with the French CEA/CERCAs plates. If the manufacture of UMo dispersion fuel is industrially mastered by CERCA, industrial monolithic fuel production is not yet setting-up. Two main issues have been identified and the associated program already launched. In order to save time "fashionable" manufacturing techniques will be used: UMo foils are going to be produced by means of conventional rolling and plates -as proposed by ANL- by means of FSW –Friction Stir Welding–. Nevertheless, for long term consideration, some more industrial techniques will be simultaneously investigated. The first feasibility step for manufacturing UMo foils is in progress – The raw UMo material has been already manufactured by CEA- and the production will be over before mid 2005. A full FSW program is being launched for evaluating industrially this technique. The IRIS-5 experiment is scheduled early at the beginning of 2006. PIEs will be carried-out by CEA.
6.3 Time schedule

<table>
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<th>2004</th>
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<td>Microstructure investigation</td>
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<td>PIEs</td>
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</table>

Figure 1: French UMo (CEA/CERCA) extended program

7. Conclusion

After the French UMo Group development program completion and the issue of unacceptable results, and in the frame of developing a suitable fuel for the CEA RJH reactor project, CEA and CERCA signed in 2004 an agreement in order to find solution which could prevent the nowadays redhibitory UMo fuel behaviour. UMo dispersion and monolithic UMo fuel monolithic are both investigated through the program. For monolithic UMo form, a specific program has been launched for testing the manufacturing feasibility and finally for producing prototype full size plates which will be irradiated in OSIRIS reactor early in 2006. Irradiation of UMo dispersion plates are also scheduled for evaluating the proposed solutions.

8. References


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MAIN RESULTS AND STATUS OF THE DEVELOPMENT OF LEU FUEL FOR RUSSIAN RESEARCH REACTORS

A. VATULIN, A. MOROZOV, V. SUPRUN, I. DOBRIKOVA
Federal State Unitary Enterprise
A.A.Bochvar All-Russian Scientific Research Institute of Inorganic Materials (VNIINM)
123060 Moscow, P.B. 369, Russia

ABSTRACT

VNIINM develops low enrichment uranium (LEU) fuel on base U-Mo alloys and a novel design of pin-type fuel elements. The development is carried out both for existing reactors, and for new advanced designs of reactors.

The work is carried on the following main directions:

− irradiate LEU U-Mo dispersion fuel (the uranium density up to 6.0 g/cm³) in two Russian research reactors: MIR (RIAR, Dimitrovgrad) as pin type fuel mini-elements and in WWR-M (PINP, Gatchina) within full-scaled fuel assembly (FA) with pin type fuel elements;
− finalize development of design and fabrication process of IRT type FA with pin type fuel elements;
− develop methods of reducing of U-Mo fuel –Al matrix interaction under irradiation;
− develop fabricating methods of fuel elements on base of monolithic U-Mo fuel.

The paper generally reviews the results of calculation, design and technology investigations accomplished by now.

1. Introduction

The main direction of the Russian RERTR program is to develop fuel on high-density uranium alloys base. Currently, the main option is the dispersion U-Mo fuel. Two types of FA are under development to use this fuel:

1) standard design with tubular type fuel elements with up to 5.4 g/cm³ U density;
2) novel design with pin type fuel elements with up to 6.0 g/cm³ U density.

The paper gives a generalized review of the results of development of FA with pin type fuel elements on base U-Mo fuel [1-8]. During the recent years this direction is advanced. The work is being performed as assigned by RosAtom and "TVEL". Many organizations in Russia, i.e. RDIPE, RIAR, NPCC, PINP, RSC KI as well as ANL (USA) participate in the work in the frame of the international program RERTR.

2. Main results

During 2004 the developments of pin type fuel elements on base LEU U-Mo fuel were carried out in the following major directions:

− irradiate dispersion U-Mo fuel in two research reactors: MIR (RIAR, Dimitrovgrad) in pin type fuel mini-elements and WWR-M (PINP, Gatchina) within the full-scaled FA with pin type fuel elements;
− master the design and fabricating process the IRT type FA with pin type fuel elements;
− develop methods of reducing of U-Mo fuel –Al matrix interaction under irradiation;
− develop fabricating methods of fuel elements on base of monolithic U-Mo fuel.

To-day the following results are available on those work directions.

The irradiation of pin type fuel mini-elements with dispersion U-Mo LEU fuel has been completed in MIR reactor [2, 3, 5, 6]. The mean U²³⁵ burnup in the fuel mini-elements made up ~ 40-65%. The
The results of the post-irradiation experiments (PIE) are planned to be available in the autumn of 2005. This experiment will provide the first irradiation results of pin-type U-Mo dispersion fuel elements in characteristic operation conditions of pool-type research reactors. The design of the experimental device is illustrated in Fig.1. The main testing conditions are summarized in Tab.1 [3, 5, 6]. The experimental device have 13 types of fuel composition that are differed by the uranium density, the chemical composition of the alloy, the method used to fabricate fuel granules and the phase state of the alloy.

The process of testing included four stages (Tab.1). After the first stage (January, 2004) at the mean fuel burnup of ∼20% one capsules (N2) was discharged from the reactor to unload some mini-elements for preliminary investigations. At the second stage (during three months) only one capsule (N1) was under irradiation. In April 2004 mini-elements in capsule N2 were replaced by fresh fuel elements, and the capsule N2 was loaded into the reactor once again. At the third stage of the tests in the channel the indicators of nonhermeticity were fund out and so late in November capsule N1 was discharged at the mean fuel burnup up to ∼60-65%. The irradiation of capsule N2 was resumed and completed early in February 2005 at the average fuel burnup of ∼20-60%.

Currently, the primary investigations of the mini-elements of capsule N1 are in progress including: inspection of appearance, γ-scanning and measurements of the mini-element dimensions.
Tab. 1. Parameters of mini-element tests

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Stages of tests</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Stage 1</td>
</tr>
<tr>
<td></td>
<td>27.08.03-</td>
</tr>
<tr>
<td></td>
<td>12.01.04</td>
</tr>
<tr>
<td>Power experimental device, kW</td>
<td>75-95</td>
</tr>
<tr>
<td>Coolant rate, m/s</td>
<td>2,9-3,2</td>
</tr>
<tr>
<td>Inlet temperature of coolant, °C</td>
<td>≤ 35</td>
</tr>
<tr>
<td>Thermal flux density, kW/m²</td>
<td></td>
</tr>
<tr>
<td>Capsule N1</td>
<td>430-925</td>
</tr>
<tr>
<td>Capsule N2</td>
<td>320-690</td>
</tr>
<tr>
<td>Temperature of cladding, °C</td>
<td>88-105</td>
</tr>
<tr>
<td>Neutron flux density, E&gt;0.1 MeV, 10¹³ cm⁻² s⁻¹:</td>
<td></td>
</tr>
<tr>
<td>U density - 4 g/cm³</td>
<td>3,0-5,7</td>
</tr>
<tr>
<td>U density - 6 g/sm³</td>
<td>3,3-5,9</td>
</tr>
<tr>
<td>Fission rate per unit volume of fuel, 10¹³ cm⁻³:</td>
<td></td>
</tr>
<tr>
<td>U density - 4 g/cm³</td>
<td>3,5-7,4</td>
</tr>
<tr>
<td>U density - 6 g/sm³</td>
<td>4,8-11</td>
</tr>
<tr>
<td>Average U²³⁵ burnup, %:</td>
<td></td>
</tr>
<tr>
<td>Capsule N1</td>
<td>27,3</td>
</tr>
<tr>
<td>Capsule N2</td>
<td>20,5</td>
</tr>
</tbody>
</table>

The lifetime tests of two full-scaled pin type FA are in progress in the WWR-M reactor in Gatchina. One FA has LEU UO₂+Al dispersion fuel with uranium density ~ 2.7 g/cm³, other FA has U-9%Mo+Al dispersion fuel with uranium density ~ 5.3 g/cm³. The design of the experimental FA is given in Fig. 2 [4, 6]. Late in January 2005 the average burnup in FA with UO₂-Al fuel made up ~30% and that in FA with UMo-Al fuel was 8.3%. The tests are planned to the average burnup of 60%. Late in 2004 FA with UMo-Al fuel was loaded into a core cell where the same power density as that of the FA with UO₂-Al fuel is provided. At that the parameters q_v and q_s of FA increased by a factor of 1.6-1.8.

Fig. 2. Design of experimental FA [4]
Tab. 2. Test conditions of experimental FA [6]

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Stage 1 15.10.03-28.11.03</th>
<th>Stage 2 01.12.03-29.02.04</th>
<th>Stage 3 01.03.04-31.08.04</th>
<th>Stage 4 01.11.04-31.01.05</th>
</tr>
</thead>
<tbody>
<tr>
<td>Inlet temperature of coolant, °C</td>
<td>34-45</td>
<td>33-38</td>
<td>42-47</td>
<td>43-50</td>
</tr>
<tr>
<td>Average U^{235} burnup, %:</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>UO_2-Al</td>
<td>3.4</td>
<td>10.3</td>
<td>23.6</td>
<td>30.0</td>
</tr>
<tr>
<td>UMo-Al</td>
<td>1.52</td>
<td>3.6</td>
<td>6.8</td>
<td>8.3</td>
</tr>
<tr>
<td>Power density q_{Vmax}, kW/l:</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>UO_2-Al</td>
<td>140-210</td>
<td>200-220</td>
<td>200-250</td>
<td>223-250</td>
</tr>
<tr>
<td>UMo-Al</td>
<td>127-187</td>
<td>160-170</td>
<td>145-160</td>
<td>223-270</td>
</tr>
<tr>
<td>Temperature of cladding, °C:</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>UO_2-Al</td>
<td>68-86</td>
<td>80-90</td>
<td>86-102</td>
<td>90-95</td>
</tr>
<tr>
<td>UMo-Al</td>
<td>74-82</td>
<td>70-78</td>
<td>74-80</td>
<td>78-100</td>
</tr>
</tbody>
</table>

The other part of the activity in this direction is the development of the IRT type FA with pin type fuel elements. The lifetime tests of the experimental full-scaled fuel assemblies (tube and pin type designs) are planned to be carried out in the MIR reactor (Dimitrovgrad). This will allow to perform the post-irradiation examinations the result of which might be used to validate the serviceability of the fuel and the pin type FA.

Together with NPCC, ANL and RSC KI a large scope of the work on the design experimental and technologic investigations for the IRT type FA with pin type fuel elements has been completed [7, 8]. The hydraulic tests two full-scaled FA mockups have been carried out. The design of FA mockup is illustrated in fig. 3 [8]. The mockups are only differed by the design of their space grids (built-up construction and one-piece construction). As the results:

- the hydraulic characteristics of FA were determined (Fig. 4);
- the optimal design of the space grid was choose;
- calculation dependence was found, that allows the assessments of the hydraulic characteristics of FA with fuel elements having different dimensions (Fig. 5) [8].

Based on these results the calculations were carried out to optimize the main design characteristics of the fuel elements and FA. The calculations were performed as applied to the WWR-SM reactor in Uzbekistan where only six-tube fuel assemblies are used under the most heat operating conditions. As a result, the main fuel element parameters were established (Tab.3) to be used for fabricating an experimental fuel assembly to be lifetime tested.

The dynamic tests of the full-scaled mockup of a pin type FA are in progress. These tests include the vibration tests on a special stand and the test for the impact-resistance. The objective of the tests is to assess the reliability of the ITR type FA design during its transformation.

The extensive work is in progress to develop the devices and methods of controlling to ensure the fabrication quality of the pin type fuel elements under the commercial production conditions. The scope of the parameters to be controlled and their tolerable magnitudes were defined. The methods and the optimal conditions of controlling were selected. A mockup facility of a non-destructive control for pin type fuel elements (cladding thickness, fuel mass and its distribution uniformity) has been designed and fabricated.
Tab. 3. Main parameters of pin type fuel elements of IRT FA

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Dimension of fuel meat, mm</td>
<td>1.75 x 1.75</td>
</tr>
<tr>
<td>Active part length, mm</td>
<td>600</td>
</tr>
<tr>
<td>Cladding thickness, mm</td>
<td></td>
</tr>
<tr>
<td>nominal</td>
<td>0.4</td>
</tr>
<tr>
<td>minimal</td>
<td>0.3</td>
</tr>
<tr>
<td>Circumscribed diameter, mm</td>
<td>4.5</td>
</tr>
<tr>
<td>U-235 mass, g</td>
<td></td>
</tr>
<tr>
<td>in fuel element</td>
<td>2.2</td>
</tr>
<tr>
<td>in FA</td>
<td>380</td>
</tr>
<tr>
<td>U density, g/cm³</td>
<td>6.1</td>
</tr>
</tbody>
</table>

Fig. 3. Design of IRT pin type FA mockup
Currently a series of efforts are under way to determine ways of reducing of U-Mo fuel –Al matrix interaction under irradiation. The program of investigations has been worked out that include
- U-Mo alloys alloying;
- matrix alloying;
- creating protective coats barriers on U-Mo fuel particles;
- use of monolithic U-Mo fuel.

The activities in all the directions are being carried out at the stage of the pre-irradiation investigations. Under way is the work to choose alloying additives, develop methods of alloying, try the technology of coating, work out methods of fabricating monolithic fuel elements, manufacture the needed instruments and specimens.

3. Conclusion

At the present time the most important accomplishments can be summarized as follows.
- the irradiation of LEU U-Mo dispersion fuel (the uranium density equaled to 6,0 g/cm³) in pintoype fuel mini-elements has been completed in MIR reactor (RIAR, Dimitrovgrad). The average $^{235}\text{U}$ burnup made up 40-65%.
- the irradiation of two full-scaled pin type FA is in progress in WWR-M reactor (PINP, Gatchina).
- a large set of calculation and experimental research on substantiation of IRT-type FA design with pin type fuel elements has been completed. As result the optimum specification of dispersion pin-type FE and FA design has been developed.
- the development of fuel elements on the basis of monolithic U-Mo fuel is in progress.

The presently available results of the work to develop U-Mo alloy fuel and pin type fuel elements have demonstrated the expedience of the further evolution of this direction.

The authors are grateful TVEL, NPCC, PINP, RIAR, RSC KI as well as ANL (USA) for the large contribution and active participation in accomplishing the presented scope of the work.

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CURRENT STATUS OF THE CNEA QUALIFICATION PROGRAM FOR THE FABRICATION OF HIGH DENSITY MTR FUEL

L. ALVAREZ, N. BOERO and J. FABRO
Unidad de Actividad Combustibles Nucleares
Comisión Nacional de Energía Atómica (CNEA)
Avda. del Libertador 8250, C1429BNO - Buenos Aires, República Argentina
lalvarez@cnea.gov.ar

ABSTRACT
The Atomic Energy Commission of Argentina (CNEA) is actively working in a program to qualify the fabrication of high-density MTR nuclear fuels since the year 2000. The first part of the program, already completed, was the qualification of the silicide fuel fabrication. Irradiation, post-irradiation examination and hot cell destructive testing were the main activities carried out to complete this part. The second part deals with the qualification of the fabrication of fuel elements bearing the fissile material in a disperse U-Mo compound. In view of the results obtained by other irradiation programs part of the CNEA effort was put on the basic research to look for an improvement in the understanding of the interaction phenomena. Diffusion studies were performed to search for a suitable matrix material to reduce the U-Mo-Al interaction. The above mentioned results also indicate that a stable behaviour is likely to be expected at low power-low temperature conditions. Based on this consideration a full-scale irradiation program in the Argentine RA-3 Reactor is again being analyzed.

1 CNEA Qualification Program

The Atomic Energy Commission of Argentina (CNEA), as was reported in previous RRFM meetings [1-3], is actively working since the year 2000 in a program to qualify the fabrication of different high-density MTR nuclear fuels. The first part of the program, already completed, was the qualification of the silicide fuel fabrication. Fuel design, fabrication, irradiation, pool-side post-irradiation examination and hot cell destructive testing were the main activities developed to fulfil the objectives of this stage. The fuel assemblies involved in this part of the program were irradiated in the Argentine Reactor RA-3 reaching in both cases the foreseen target burnup without any signal of defects or abnormalities. The second stage of the program deals with the qualification of the fabrication of fuel elements bearing the fissile material as a U-Mo compound. This part of the program was delayed in view of the unexpected results obtained in another programs regarding the onset of interaction between U-Mo and Al. The third part, which is still in a development stage, will cover the fabrication of plates with U-Mo as a monolithic fuel.

2 Current activities

As a consequence of the results obtained in irradiation experiments performed by other organizations [4, 5] the full scale fabrication of U-Mo disperse fuel and its qualification was momentarily postponed during the last year to wait for more information and a better understanding of the U-Mo fuel behaviour under irradiation. CNEA activities were mainly concentrated in the study of fuel aluminum reaction rates and interaction products characterization, exothermic fuel-aluminum-reaction energy release and fuel-aluminum thermal compatibility. Part of the effort was applied to the development of
possible solutions to reduce the interaction phenomena. The modification of the material for the matrix of the compacts and the coating of the U-Mo particles are considered as the more suitable solutions. The development of fabrication techniques for monolithic fuel plate fabrication like Friction Stir Welding and the rolling of monolithic plates using zircaloy to replace de Al cladding are also new fields of development. Irradiations of miniplates in the RA-3 and in the ATR to test these solutions are also being planned.

3 Basic and applied research results

The recent results of CNEA research activities in the interaction field were presented elsewhere [6]. The main conclusions are summarized in the following points:

- The addition of Mg to the pure Al does not produce any important change in the characteristics of the U-Mo–(Al+Mg) interaction zone.
- Si addition produces the presence of a (U,Mo)(Al,Si)_3 phase in the Al side of the interdiffusion layer. This phase is similar to the one reported in silicide dispersion fuels, which has a satisfactory behaviour under irradiation.
- The interaction layer of the Al+Si alloy is not affected by the decomposition of the γU-Mo as happens when pure Al is used.

4 Continuation of U-Mo fuel fabrication developments and qualification

The results reported in the last RRFM and RERTR Meetings indicate that the irradiation behaviour of the U-Mo alloy is very stable and predictable [5]. At high power densities the interaction phase formed by interdiffusion of U-Mo and Al matrix develops large fission gas pores but this undesirable behaviour is not present at low power and low temperature conditions. Figure 4 of [5] shows a threshold delineating satisfactory and unacceptable fuel behavior of the U-Mo/Al plate type tests performed to date. Based on this threshold and considering that the RA-3 operating conditions are well below the mentioned curve, in the satisfactory region, a new irradiation of a U-Mo fuel in the main Argentine Research Reactor is again being analyzed.

In the mean time CNEA has continued working in setting-up the fuel plate fabrication techniques to reduce the amount of interaction present in the as-fabricated plate and to improve the homogeneity of the U distribution in the fuel plate. CNEA fabrication techniques are prepared to obtain sound plates suitable for irradiation.

5 The program

As mentioned in 4, a full-scale irradiation program in the Argentine RA-3 Reactor with the objective of qualifying the technology for the fabrication of full scale disperse U-Mo fuel plates is being analyzed.

The fuel fabrication will be performed in the ECRI facility at the Constituyentes Atomic Center and the target burnup will be >50 %.

The RA-3 reactor, located near Buenos Aires, is a pool type, refrigerated and moderated with light water. Cooling is provided by down going forced convection. Currently its nominal power is 10 MW, which is reached with an equilibrium configuration of 25 fuel assemblies. The feasibility of U-Mo fuels irradiation in this reactor has already been demonstrated [7].
5.1 Fuel design

The final details of the fuel assembly have not been completely defined yet. Tentatively the fuel will consist of 20 fuel plates bearing U-Mo as the fuel material disperse in Al. The final design might combine fuel plates fabricated from atomized powder and fuel plates with meats obtained from HMD powder although another alternative could be the fabrication of two fuel elements bearing different types of powders. The HMD (Hydriding-Milling-Dehydriding) is an alternative process developed in CNEA to obtain U-Mo powder.

Considering the results indicated in 3, CNEA is also analyzing the inclusion of plates fabricated with a modified Al matrix to improve the U-Mo - matrix interaction during the irradiation.

The U density of the plates will be 6 gU/cm3 and a U-7wt%Mo compound will be used. Meat and cladding thickness have not been defined yet.

The tentative characteristics of the fuel assembly are indicated in the following table:

<table>
<thead>
<tr>
<th>Fuel Material</th>
<th>U-7wt%Mo</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel Plates</td>
<td>20</td>
</tr>
<tr>
<td>Meat dimensions</td>
<td></td>
</tr>
<tr>
<td>Length</td>
<td>[mm]</td>
</tr>
<tr>
<td>Width</td>
<td>[mm]</td>
</tr>
<tr>
<td>Thickness</td>
<td></td>
</tr>
<tr>
<td>Outer</td>
<td>[mm]</td>
</tr>
<tr>
<td>Inner</td>
<td></td>
</tr>
<tr>
<td>Fuel Plate dimensions</td>
<td></td>
</tr>
<tr>
<td>Length</td>
<td>[mm]</td>
</tr>
<tr>
<td>Inner</td>
<td></td>
</tr>
<tr>
<td>Width</td>
<td>[mm]</td>
</tr>
<tr>
<td>Outer</td>
<td></td>
</tr>
<tr>
<td>Inner</td>
<td></td>
</tr>
<tr>
<td>Thickness</td>
<td></td>
</tr>
<tr>
<td>Outer</td>
<td>[mm]</td>
</tr>
<tr>
<td>Inner</td>
<td></td>
</tr>
<tr>
<td>Cladding Thickness</td>
<td></td>
</tr>
<tr>
<td>Outer</td>
<td>[mm]</td>
</tr>
<tr>
<td>Inner</td>
<td></td>
</tr>
<tr>
<td>Enrichment $^{235}U$</td>
<td></td>
</tr>
<tr>
<td></td>
<td>$19.78 \pm 0.012 %$ U weight</td>
</tr>
<tr>
<td>U Density</td>
<td>[g/cm3]</td>
</tr>
<tr>
<td>Gap between fuel plates</td>
<td>mm</td>
</tr>
</tbody>
</table>

(*) To be defined

5.2 Preliminary schedule

The following Figure shows schematically a tentative schedule for the proposed program. The target burnup could be reached during 2008 and the final results of the PIE examination might be available during the first half of 2009.
Intermediate visual inspections at the RA-3 pool-side station are also foreseen to evaluate the evolution of the irradiation. The inspections will also include the measurement of the gap between fuel plates. These inspections will allow the detection of any unexpected excessive swelling of the plates. Although this phenomenon is not likely to occur, the inspections will provide an early warning allowing an eventual interruption of the irradiation to avoid any undesirable effect on the operation of the reactor.

6. Final remarks

CNEA has completed the qualification program related with the fabrication of silicide plate type fuel for MTRs. The initial qualification program of the domestic U-Mo fuel fabrication had been postponed during 2003 but now, considering the information available about the behaviour of the U-Mo fuel at low power-low temperature conditions, a continuation of that program is been analyzed. An irradiation of a full scale U-Mo fuel element in the RA-3 reactor is being considered. CNEA has developed the technology to fabricate sound U-Mo plates suitable for irradiation and has the facilities to complete this program. Preliminary the first results from hot cell destructive examination of some of the fuel plates are expected for 2009.

7. References


7. Acknowledgments

We deeply recognize the collaboration and permanent support of J. Snelgrove, G. Hofman and T. Wienceck from Argonne National Laboratory.
ABSTRACT

IRIS2 irradiation was the last irradiation of 4 full sized plates launched by CEA for the French UMo group to test in which operating conditions the coarse porosity forms in the UMo/Al interaction product.

IRIS2 consists in four plates with high uranium loading and U-7wt%Mo atomised powder irradiated up to 60 days at OSIRIS reactor in IRIS device at a peak power of 238 W.cm⁻². The results show that in the tested conditions pillowing of the plate started from a fission density over 2 \times 10^{21} \text{fission.cm}^{-2}. Moreover, they show that the fission products and impurities have a key role in the origin of the excessive plate swelling.

1. Introduction

The French UMo Group program has been launched by 1999 in close collaboration with five partners: CEA, CERCA, COGEMA, FRAMATOME-ANP and TECHNICATOME. The initial goal of this program was to contribute to the high performances and reprocessable UMo fuel international development (RERTR). To reach this goal the French Group chose to irradiate full-sized experimental plates with low enriched uranium and high uranium loading up to 8 g.cm⁻³.

As a first step of this program, IRIS1 irradiation established the good behaviour of high uranium loading UMo plates, for heat flux up to 140 W.cm⁻² (\(-3.4 \times 10^{14} \text{f.cm}^{-2} \text{UMo.s}^{-1}\)), 67% BU (\(-4.6 \times 10^{21} \text{f.cm}^{-3} \text{UMo}\)) with a cladding surface temperature below 75°C. However, as an other step, FUTURE irradiation pointed out fuel performance limitations for higher irradiation flux, i.e. 340 W.cm⁻² (\(-6.3 \times 10^{14} \text{f.cm}^{-3} \text{UMo.s}^{-1}\)). Indeed this experiment indicated that a porosity can form and coarse in the U-Mo/Al interaction phase. This porosity leads to excessive swelling (pillowing) of the plates. This result was confirmed by others international programs.

In order to assess the irradiation limits for which the coarsening porosity forms in the UMo/Al interaction, IRIS2 experiment was initiated in OSIRIS reactor IRIS device. Peak heat flux and cladding wall temperature were 238 W.cm⁻² and 93°C. The results show that for these operating conditions coarse porosity and pillowing of the plates have also occurred from fission density over 2 \times 10^{21} \text{fission.cm}^{-3} \text{UMo}.

The aim of this paper is to present IRIS2 experiment results.

2. IRIS 2 as fabricated plates

For IRIS2 experiment, four experimental plates with high uranium loading (8.3 g U.cm⁻³) were fabricated by CERCA. The fuel plates consists of U-7wt%Mo atomised fuel powder dispersed in aluminium matrix. The cladding material is AG3-NE.

The main characteristics of the as-fabricated plate meat are summarized in Table 1: U-7wt%Mo loading is around 51vol% and the porosity lower than 2%.
Table 1: IRIS2 fuel plates main characteristics

3. Irradiation

3.1 Irradiation history
The four plates were irradiated in OSIRIS reactor IRIS irradiation test rig\(^J\). The IRIS device is associated to an in-pool plate thickness measurements device. It enables to assess plate thickness profiles at each reactor cycle stop.

Peak irradiation heat flux and cladding temperature of the four plates are given Table 2.

<table>
<thead>
<tr>
<th>Heat flux (W.cm(^{-2}))</th>
<th>Cladding surface temperature (°C)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cycle n°</td>
<td>FPD°</td>
</tr>
<tr>
<td>1</td>
<td>15.1</td>
</tr>
<tr>
<td>2</td>
<td>13.2</td>
</tr>
<tr>
<td>3</td>
<td>12.0</td>
</tr>
<tr>
<td>4</td>
<td>17.6</td>
</tr>
</tbody>
</table>

Table 2: Maximum Flux Plane (MFP) peak irradiation conditions ("Full Power Day")

The plate P#08 was removed after the second cycle, due to a slight warping of it during intercycle handling. The irradiation of the three other plates was further stop before reaching the 50% target burnup. Indeed, for safety reason the swelling admissible limit for the plate is 250 µm and this limit was reached at the end of the third cycle for P#07 and at the fourth cycle for the 2 other plates (P#02 and P#03):

3.2 In-pool plate thickness measurements
IRIS 2 plate thickness swelling is given Figure 1 as a function of the fission density. As a comparison IRIS 1\(^J\) and FUTURE\(^J\) experiments swelling are also plotted.

The results show that for IRIS2 irradiation conditions, plate pillowing occurs from around 2 \(10^{21}\) fission.cm\(^{-3}\)\(^{\text{UMo}}\). These results are consistent with FUTURE experiment one. For lower irradiation conditions and grounded particles (IRIS1 experiment), no pillowing was observed up to 4.8 \(10^{21}\) fission.cm\(^{-3}\)\(^{\text{UMo}}\).

![Figure 1: Plate swelling versus fission density (fission.cm\(^{-3}\)\(^{\text{UMo}}\)).](image-url)
4. Post-Irradiation Examinations (PIE)

A large set of PIE was performed on the most deformed plate, i.e. P#02 at the maximum flux plane. The sample is chosen in order to have the edge of the plate meat together with the large pillowing area. Burnup at this location evolves from $2.8 \times 10^{21}$ to $2.9 \times 10^{21} \text{ fission.cm}^{-3_{\text{UMo}}}$ and at the peak temperature in the meat is $112^\circ \text{C}$.  

4.1 Metallographic examinations

Metallographs of the sample, shown Figure 2, indicate a microstructure similar to FUTURE $^{[3]}$ experiment one. They confirm that the origin of the plate pillowing is coarse porosity at the (U-Mo)-Al interaction product interface with aluminium. All the stages of the porosity formation and coarsening are seen, from no porosity area (Figure 2a) to a big hole at the plate pillowing location (Figure 2d). In the pillowing area, it can be seen indications of interaction product plastic deformation but no signs of U-7wt%Mo alloy microstructure evolution.

![Image of metallographic examination](image_url)

**Figure 2**: Metallographic examinations on IRIS2 Plate P#02 at MFP

4.2 EPMA

X-ray mappings at the different locations identified in Figure 2 are presented Figure 4. The mappings underline:
A depletion of molybdenum content observed at the U-Mo alloy grain boundaries. It has already been observed in past experiment with atomised powders. A fission product accumulation at the interface between interaction product and aluminium matrix. However, for solid fission product, as for example Zr, no evolution of their concentration is visible, while at contrary for Xe, this accumulation disappears as the porosity increases, i.e. from location -a to -d of Figure 4.

Quantitative analyses were also done. They indicate that the aluminium content of the interaction layer decreases continuously from "U-MoAl₆" to "U-MoAl₄.₅", respectively from edge of the meat (Figure 4-a) to the pillowing area (Figure 4-d).

4.3 SEM examinations

![Figure 3 : SEM images of Plate P#02 meat.](image)

SEM examinations presented Figure 3 show:
- fission gas bubble precipitation at the grain boundaries of UMo alloy particles, where the Mo content is lower (-b & -c),
- no fission gas bubbles are observed in the "U-MoAl" interaction layer (-a, -b & -c),
- when 2 UMo particles are close together, some link, like sintering link are observed in between the particles, larger fission gas bubbles are observed in the linking compound (-c).
- the UMo particles interface with UMoAl interaction product is not smooth, it could indicate that interaction occurs at preferential location, as grain boundaries (-b & -c),
- dark lines are visible at the interfaces where two interaction layer meet together (-b).

5. Discussion

The results obtained indicate that, during irradiation, as the temperature increases, the aluminium content in interaction compound decreases from "(U-Mo)Al₆" at the coldest to "(U-Mo)Al₄.₅" at hottest position of IRIS2 sample.

These results are consistent with previous experiments. Indeed, for IRIS 1 low temperature experiment, "(U-Mo)Al₇" compound has been found while for higher temperature experiment (FUTURE) it was "(U-Mo)Al₃.₄".

The origin of the large pores development in the interaction product is still not clear, but from the results it seems that the fission products have a key-role in it. Indeed, it seems that while the interaction layer grows, the fission products (FP) and impurities (of aluminium matrix) are "pushed" at its periphery (see dark interaction borders Figure 3-b). One explanation of that could be a low solubilization of the FP in the interaction product. Then the reaction is stopped at some weak location, at the interaction product / aluminium interface when the amount of FP and impurities is high enough. It could be the porosity formation beginning. Moreover, when porosity started to form, gases could progressively fill it by diffusion process. This phenomenon is enhanced by the local temperature...
increase due to the low conductivity of the so-formed porosity. This process is irreversible and spiral out of control and leads to porosity coarsening. On the other hand, IRIS2 experiment indicates a good behaviour with temperature of the U-7wt% Mo alloy itself. Indeed, even at rather high temperature (pillowing area) no indications of excessive FG release or bubble growth are observed in the fuel particles.

6. References

Figure 4: X-ray mappings of IRIS2 plate P#2 at several locations.
MECHANICAL CALCULATIONS ON U-Mo DISPERSION FUEL PLATES WITH MAIA

V. MARELLE and F. HUET

CEA – Cadarache
F-13108 Saint-Paul-Lez-Durance Cedex, France

and

P. LEMOINE

CEA – Saclay
F-91191 Gif-sur-Yvette Cedex, France

ABSTRACT

CEA has developed a 2D thermo-mechanical code, called MAIA, for modelling the behaviour of U-Mo dispersion fuel. MAIA uses a finite element method for the resolution of the thermal and mechanical problems. Physical models, issued of the DOE-ANL code PLATE, evaluate the fission products swelling and the volume fraction of the interaction between U-Mo and Al. They allow establishing strains in the meat imposed as loading for the mechanical calculation. MAIA has been validated on the irradiations IRIS 1 and RERTR-3 and a rather good agreement is obtained with post irradiation examinations. MAIA is used to calculate the last irradiation of the French UMo group, IRIS 2. MAIA predicts a maximum temperature of 112°C and meat swelling of 16%. Mechanical calculations are finally performed to evaluate the sensitivity to some mechanical hypotheses such as constitutive laws and the way the meat swelling is applied.

1. Introduction

MAIA is a 2D thermo-mechanical code used for modelling the behaviour of U-Mo dispersion fuel plates. The meat is treated as a homogeneous material for the thermal and mechanical resolution with a Finite Element Method (FEM). The evolution of the meat composition is calculated throughout the irradiation taking into account the fading of as-fabricated porosities, the swelling due to fission products and the interaction between the U-Mo particles and the Al matrix. A model also evaluates the cladding oxidation.

2. MAIA Code evolutions

Several improvements have been made on MAIA since the last version of the code presented during RRFM 2004 [1]. A Contact ratio is now used to limit the increase of the interaction compound when the matrix volume fraction become lower and a shape factor on particles is applied to correctly evaluate the volume of the interaction compound for non-spherical particles. These evolutions are issued from PLATE, the code developed by the DOE-ANL [2]. Many improvements on the mechanical calculations have also been made to get more consistent results with more modelling options.
3. Validation

MAIA has first been qualified on analytical test cases to make sure of the good behaviour of the code. Then MAIA has been validated on the available irradiation data: IRIS 1 [3] and RERTR-3 [4]. The Post Irradiation Examinations (PIE) have been compared with MAIA results on the interaction layer thickness and the volume fractions. A rather good agreement is obtained (cf. Figure 5). A benchmark with PLATE has also been carried out in order to make sure of the consistency of the results and to identify improvements in the modelling that were necessary.

![Figure 5: MAIA/PIE comparison on interaction layer thickness (1a) and volume fractions (1b)](image)

4. IRIS 2 results

MAIA has been used to calculate the plate P#02 of the IRIS 2 irradiation (four cycles) [5]. Calculation has been made in the Maximum Flux Plane (MFP). The different results presented in this paper are in the middle of this plane where the flux is locally the highest. The maximum temperature in the meat calculated by MAIA is 112°C (cf. Figure 6). The external plate temperature in the middle of the MFP decreases from 94°C at beginning of life to 79°C at end of life.

![Figure 6: Temperatures in the middle of the plate at the MFP](image)
The results of the code will be compared with PIE as soon as they will be available (cf. Table 3 and Figure 7). It appears that meat swelling (16\%) is mainly due to fission product swelling in U-Mo. In MAIA, as-fabricated porosity is supposed to accommodate fission product and interaction product swelling. IRIS 2 plates are made with atomised powder so the as-fabricated porosity is low (1.5\%) and total meat swelling begins during first cycle. In IRIS 1, made with ground powder, as-fabricated porosity is much higher (12\%) and total meat swelling is delayed because of porosity accommodation. Above 1.8\times10^{27} fissions/m^3 of U-Mo, fission product swelling rate in U-Mo particles gets higher and becomes the dominating phenomenon for meat swelling. It also explains the increase of the U-Mo volume fraction above this threshold.

<table>
<thead>
<tr>
<th>Interaction layer thickness (m)</th>
<th>Oxide layer thickness (m)</th>
</tr>
</thead>
<tbody>
<tr>
<td>MAIA</td>
<td>3.9</td>
</tr>
<tr>
<td>11.6</td>
<td></td>
</tr>
</tbody>
</table>

Table 3: Interaction and oxide layer thickness calculated by MAIA at the MFP

![Graph showing volume fractions and meat swelling vs. fission density](image)

Figure 7: Volume fractions (3a) and meat swelling (3b) in the centre of the meat vs. fission density predicted by MAIA

5. Mechanical results

Mechanical calculations are performed on IRIS 2 with the hypothesis of generalized plane strains. The following results should be analysed as a parametric study to evaluate the sensitivity of some mechanical hypotheses (constitutive laws used and isotropy of meat swelling). The notation for directions of the plate are x (width), y (thickness) and z (length).

In case #1, the mechanical behaviour is elastic and the meat swelling, which is a loading for the mechanical calculation, is only applied in the plate thickness direction. The stresses calculated by MAIA (below 40 MPa) show that the cladding is in compression as the meat is in tension due to a
higher thermal expansion coefficient of the cladding (cf. Figure 8). The meat swelling doesn’t generate any significant evolutions on the stresses in the middle of the plate.

In case #2, the constitutive laws used are elasto-viscoplasticity for cladding and plasticity for meat. The meat swelling is still only applied in the plate thickness. The main differences with the previous results are stress relaxation and a sign change of the stresses during inter-cycles (cf. Figure 9).

Case #3 is like case #2 but with isotropic meat swelling. It gives very different results (cf. Figure 10). After the beginning of meat swelling, meat is in compression and cladding in tension. The stresses reached in the meat are much higher than in the previous cases (the yield stress is reached in the meat). In the cladding, because of the creep, the stress remains below 50 MPa. The highest values are reached during the stops of the reactor.
Figure 8: Stresses in meat (4a) and in cladding (4b) – Case #1

Figure 9: Stresses in meat (5a) and in cladding (5b) – Case #2

Figure 10: Stresses in meat (6a) and in cladding (6b) – Case #3
Plate thickness increase is similar for cases #1 and #2 (cf. Figure 11): viscoplastic strains are much lower than meat swelling. For IRIS 1, this increase is close to the value given by PIE. For IRIS 2, PIE cannot be compared straight with MAIA results because of the large porosities that appeared in this part of the plate and which are not modelled in MAIA.
In case #3, the plate thickness decreases because of the high strains in the width and the length of the cladding. This hypothesis does not seem very realistic.

6. Conclusion

MAIA can now be considered as a consistent code for U-Mo dispersion fuel plates thanks to its validation base. It can also be used for mechanical calculations. MAIA is therefore a useful tool to analyse and explain the behaviour of the U-Mo dispersion fuel plates. Nevertheless the mechanical hypotheses have to be studied further to fix the best options for a realistic modelling. The meat swelling for instance should be mainly applied in the plate thickness.
The mechanical characterizations that are in progress will also improve the modelling through a better knowledge on the material properties of the meat and the cladding.

7. References

OUT OF PILE FRENCH RESEARCH PROGRAM ON THE UMo/Al SYSTEM: FIRST RESULTS

S. DUBOIS, F. MAZAUDIER, J.P. PIRON, P. MARTIN, J.C. DUMAS, F. HUET
CEA-Cadarache, DEN/DEC, 13108 St Paul lez Durance – Cedex – France

H. NOËL, O. TOUGAIT
Université de Rennes1. CNRS-UMR6511 - Av. du Général Leclerc - 35042 Rennes – Cedex – France

C. JAROUSSE
CERCA – les Bérauds - BP 1114 - 26104 Romans – Cedex – France

P. LEMOINE
CEA-Saclay – 91191 Gif sur Yvette – Cedex – France

ABSTRACT
This paper focuses on a French scientific program built to gain further knowledge on the UMo/Al interaction and to discuss promising solutions to prevent this interaction under neutron flux. Three themes are thus developed. Thermodynamic investigations, through U-Mo-Al ternary system studies, may lead to determining the phase equilibria at different temperatures. Metallurgical out-of-pile diffusion couple studies are intended to improve the understanding of diffusion mechanisms and should underline the role of different parameters, which could prevent or decrease the UMo/Al reaction. Finally, the irradiation effect has been taken into account, either through a simulation program or an in-pile one. The results already achieved are described and discussed.

1. Introduction
Recent in-pile experiments have shown that dispersed UMo fuel in a pure Al matrix does not withstand high operating conditions [1], [2], [3]. Extensive porosity is formed in the reaction layer, at the fuel/matrix interface, resulting in an unacceptable pillowing and/or swelling of the fuel plate. The behaviour of this interaction seems to be responsible, directly or indirectly, for such a limit. Out-of-pile experiments have shown similar phenomena: UMo/Al interaction [4, 5] and, at times, a large amount of porosities [6]. The out of pile mechanisms, which are probably different from the in-pile ones, have to be considered in order to understand the UMo/Al system and then select solutions to prevent (or at least limit) this fuel/matrix interaction.
To this end, a French extended program has been carried out by the CEA, with a specific collaboration with CERCA, for the manufacturing aspects. This paper presents this program and the first results.

2. French research program on the UMo/Al system
In order to gain further knowledge on the UMo behaviour, and to propose solutions to its operating limit, the French scientific program is divided into three following closely linked themes.

2.1. Phase diagram studies
Thermodynamic studies with the determination of the phase equilibria within the U-Mo-Al ternary system are the basis of our program. These studies are performed within the framework of a collaboration with Rennes University.

Up to now, phase relations in the U-Mo-Al ternary system have not been fully determined. Previous investigations were limited to the partial isothermal sections at 500°C, 1050°C and 1250°C for the composition range U-Mo-Mo₃Al₈-UAl₂ [7]. The main reported feature is the formation of a rather extended homogeneous region with a cubic UAl₂₅Mo₆ structure, corresponding to the substitution of Al by Mo in the binary aluminide UAl₂, up to x = 0.66. It changes into a hexagonal phase at the higher substitution ratio. Two ternary compounds were subsequently discovered in the aluminium rich part of the system: UMo₂Al₂₀ which crystallizes in the cubic structure type CeCr₂Al₂₀ (space group Fd3m) [8], and U₆Mo₄Al₄₃ with crystallizes with the hexagonal Ho₆Mo₄Al₄₃ structure type (space group P6₃/mcm) [9]. In order to obtain more information on phase equilibria, we have started a program to determine isothermal sections of the U-Mo-Al phase diagram at 400°C and 800°C. The first results are presented in § 3.

This experimental part will be completed with thermochemistry calculations. These will be performed as a function of the temperature, with the SAGE (Solgasmix Advanced Gibbs Energy) code, in order to compare the stability of the ternary compounds which might form at the UMo/Al interface. The mathematical treatment is based on the minimisation of the Gibbs free energy. However, the input data, i.e. the thermodynamic functions of the components, are mostly unknown. The first step thus consists in calorimetric measurements on previously selected and well characterised phases in the U-Mo-Al system.

2.2. Metallurgical studies

In addition, metallurgical studies take into account the diffusion phenomenon, in relation with the material features (grain size, crystallography, defects, chemical homogeneity…). Thus, compatibility tests are intended to understand the influence of the different key parameters and select promising solutions (see § 4). Among the main parameters, we study the influence of crystallography (α or γ phase), the microstructure (grain size…), the Mo content and its distribution. Means for UMo/Al fuel improvement will be investigated, such as additive precipitates (Si), where a delaying effect on interdiffusion is expected.

The materials were investigated using classical techniques: optical and scanning electron microscopy, energy dispersive analysis (EDAX), and X-ray diffraction (XRD) techniques.

Furthermore, more specific techniques will be used. As interaction layers between the UMo and Al matrix might be heterogeneous [5, 10], micro-XRD characterisation will be performed, in order to identify the crystallographic phases. Besides, to determine the local environment of uranium and molybdenum elements, micro-XAS studies also appear to be well suited, even in case of poorly crystallized or amorphous materials. Both investigations require intense X-Ray beams and will be performed at the ESRF (European Synchrotron Radiation Facilities).

2.3. Influence of irradiation

The two previous parts of the program are necessary to improve the understanding of the UMo/Al interaction phenomenon. They may lead to select additives or other alternative solutions. However, the promising solutions have to meet the manufacturing process requirements and behave well under irradiating (irradiation damage, neutronic, thermal and mechanical issues).

To consolidate the out of pile program, we first intend to use high energy ion irradiation, typically 70 MeV Xenon (or Ag), in order to simulate (the most) damage created by fission products. These experiments will be carried out on both interdiffusion layers and fuel plates. The consequences of simulated irradiation on fuel/matrix interaction kinetics will be characterized by micro-XRD and micro-XAS as described previously.
At last, in-pile irradiation (see §5) will then be discussed and performed to determine fuel behaviour and to validate the pertinence of the chosen solution.

3. Phase diagram studies of the U-Mo-Al system

3.1. Experimental procedure

For each isothermal section, about forty samples were prepared; their composition is presented (as filled squares) in Fig. 1. High purity metals, uranium pieces (99.8 wt. %), molybdenum chips (99.99 wt %) and aluminium rods (99.999 wt %) were arc-melted in a water-cooled copper-hearth under an atmosphere of purified argon.

![Fig 1. Studied compositions in the U-Mo-Al system.](image)

The buttons were melted at least three times (weight losses < 1%) to ensure a proper homogeneity. A part of each as-cast alloys was annealed either at 400°C or 800°C for one to two months and for two weeks to one month respectively. After the heat-treatment, the samples were quenched in air. Samples taken from the as-cast and heat-treated buttons were ground and polished. Each average composition was characterized by optical microscopy, SEM analyses and X-ray powder diffraction.

3.2. First results

3.2.1. Binary and pseudo-binary UAlx phases

UAl4 has been confirmed to be a stoichiometric compound, as previously reported [11] and does not dissolve any amount of molybdenum. The solubility of Mo in UAl3 is also very slight, and limited to a maximum of 1-2 at. %.

Moreover, our results confirm that Mo has a large solubility in UAl2. The solid solution UAl2-xMox (MgCu2-type cubic structure) extends up to x = 0.5 both at T= 400°C and 800°C. But this study also confirms the formation, for higher Mo substitution, of a hexagonal phase with the MgZn2-structure type. Microprobe analysis indicate that this latter phase is stabilized within the limits 0.70<x<0.85 at T = 400°C and 0.6<x<0.7 at T = 800°C. However, it is not sure that full equilibrium was achieved at 400°C, even after two months of annealing.

Consequently, the substitution of Mo atoms to Al sites in the ternary extension of binary phases leads to U(Al2-xMox) type notation, rather than (U,Mo)Al2 commonly written in many papers, which is misleading concerning atomic site occupation.

3.2.2. UMo2Al30 and U6Mo4Al43 compounds

The preparation of ternary alloys with the stoichiometric compositions UMo2Al30 and U6Mo4Al43 systematically led to three phase mixtures. Both microprobe and X-ray powder diffraction analysis revealed non-stoichiometric phases showing large homogeneity ranges. U6Mo4Al43 is characterized by a positive deviation of the Mo content, whereas UMo2Al30 shows a positive deviation of Al content. Similar deviations from stoechiometry were also found in the analogous rare earth-based phases, like Ho6Mo4+xAl43-x [8] or CeMo2-xAl20+x [9] and can be thus considered as a prominent feature of these structural types.

Crystal structure refinements of U6Mo4Al43 and U6Mo4Al43 are being carried out using single crystal X-ray diffraction data, in order to more accurately determine crystallographic site occupations and to obtain further knowledge on the Mo/Al substitution process.

Our phase diagram studies clearly reveal the lines: UAl4-UMo2-xAl20+x at 400°C, and UAl4-UMo2-xAl20+x at 800 °C; thus UMo2-xAl20+x is directly formed as a result of the interaction between the
U-7Mo fuel and aluminium powder. The analytical data reported recently [10] for the “Al-rich (Al,U,Mo) phase” compares perfectly with the atomic composition of this ternary phase.

4. Metallurgical studies
4.1. Experimental procedure

The materials employed were aluminium (1050A, 99.5%) or aluminium alloy (e.g. 6061 with Si content) and arc-melted U–Mo alloys manufactured within the range of Mo content (0 to 10 wt.%). Part of the as-cast U-Mo alloys were heat treated at 900°C for 72 hrs to achieve a good homogenisation and then quenched (cooling rate > 2000°C/h) at room temperature, in order to retain the high temperature γ-phase.

The samples underwent a thermal treatment to modify both structure and microstructure and then evaluate their influence on interdiffusion layer. Heat treatments from 400°C to 600°C were performed in a high purity Ar atmosphere, under a controlled pressure (1-10 MPa), for several hours. These temperatures were chosen in order to be the upper and lower limits around the eutectoid temperature (565°C). Above 565°C, a metastable γ-phase decomposition is observed. Chemical diffusion couples were tightly mechanically maintained using a specific device during the whole thermal treatment.

4.2. First results

4.2.1 Mo content and structure effect

The Mo content must be sufficient (above 5 wt. % in UMo) to stabilize the metastable γ-UMo phase. The global and the local Mo concentrations are therefore both key parameters. Especially, Mo depleted areas may increase the interdiffusion and interaction zones. Since Mo is nearly insoluble in α-U, the intermetallics UAlx are thermodynamically more stable.

As expected, the Mo content is of major importance to stabilize the γ-phase and thus modify the U/Al reaction kinetic. An irregular interface is a constant feature of the as-cast alloy based samples (Fig 2.a). Such irregularities have been observed in [5]. Some of the interfacial areas did not react at all, probably due to local surface phenomena such as impurities or local strain variation.

4.2.2 Microstructure effect

In the case of a formerly heat treated UMo alloy, the thicknesses of the interaction layer are rather smaller, and even nil (Fig 2.c). The annealing at 900°C delays the decomposition of the metastable γ-phase, at a temperature in the range of 400-600°C. Actually, the former heat treatment results in two main changes in the UMo alloy:

- a microstructural one, with grains coarsening from ~ 20 µm to 400-1000 µm. Consequently, the grain boundary density decreases inversely, and thus limits the diffusion paths and amount of defects,
- a chemical one, with a better Mo distribution. The Mo homogenisation prevents local low Mo content which enhances the decomposition process of the γ phase and the UMo/Al interaction.

The interest of homogenisation has already been observed, on either U–7wt%Mo/Al diffusion couple [5] or dispersed atomized U–10 wt% Mo alloy particles [4]. The former homogeneity directly affects the type, morphology and interdiffusion layer growth.

4.2.3 Silicon effect

In the 6061 aluminium alloy, the Si content amounts to 0.4-0.8 wt. %. Its presence significantly reduces the interaction thickness (Fig 2.b and Fig 2.d).

Our first results are in agreement with previous experiments on UAl/Al, which have proven that a small amount of silicon in the Al matrix prevented UAl4 formation, stabilized UAl3 and reduced the diffusion rate at the UAl/Al interface [12, 13]. Recent studies [14] have not shown any change in the
growth kinetics with Si added aluminium, but in the interaction silicide component (U(Al,Si)$_3$ instead of U(Al,Si)$_4$).

When the interaction layer was observed, the global composition was measured near UAl$_3$. There is no noticeable concentration gradient between both sides (the Al side and the UMo side).

- a- U / Al
- b- U / 6061 alloy
- c- U7Mo homogenised / Al
- d- U7Mo / 6061 alloy

Fig 2. Influence of parameters on interdiffusion layer, after heat treatment at 550°C for 50 h.

5. **Influence of irradiation on the U-Mo-Al system**

At last, the influence of irradiation effects is studied to gain further knowledge on the UMo/Al interaction (composition, growth kinetics) and to validate the relevance of selected solutions.

5.1. **Out-of-pile irradiation**

Irradiation experiments, with a high energy ion beam (70 MeV Ag in ITIS facility in Münich), will be performed. Several types of samples have been prepared. These comprise diffusion couples or samples prepared from fuel plates. Some of them are pre-annealed in order to produce the reaction layer between the UMo and the Al matrix. For each sample, a 1 mm$^2$ area will be irradiated. The interdiffusion layer formation and its stability will therefore be tested, along with the UMo microstructure and the silicon added Al matrix.

The irradiated materials will be characterized by SEM, micro-XAS and micro-XRD, and compared to unirradiated sample (constituting a reference). Further thermal treatments will be performed on either irradiated or unirradiated samples in order to determine the influence of irradiation on the UMo/Al interaction. These experiments will begin in April 2005.

5.2. **In pile irradiation**

Based on the knowledge of the Si benefit in out-of-pile studies, a new in-pile irradiation (IRIS3) has been defined to evaluate the influence of adding silicon (0.3 or 2 wt% Si) to the aluminium matrix on the UMo dispersed fuel behaviour.

The low enrichment full-sized plates were supplied by CERCA, and manufactured using spherical and gamma phase particles obtained by atomisation process. The Al-0.3%Si powder was delivered to CERCA by ANL. The IRIS3 plate characteristics are similar to those of the IRIS2 experiment [15], but with Si addition. The molybdenum content in the UMo alloy is 7.3 wt%. The volume fraction of fuel particles was close to 50%, so that the uranium density was around 8 g.cm$^{-3}$. The results of the fuel plate inspections, by means of metallographic examinations, X-Ray diffraction, blister tests, ... indicated that the plates met the usual OSIRIS specifications.

The irradiation started in January 2005, in the OSIRIS reactor, in the IRIS device. The foreseen irradiation conditions are similar to those of IRIS2 [2], with a maximum BOL clad temperature of ~100°C and a maximum BOL heat flux of 200-240 W/cm$^2$. The expected burn-up is ~ 50 % at $^{235}$U.

The plate thickness measurements will be made plate by plate, at each intercycle, on five axial and one transverse profiles. The IRIS3 irradiation will end in November 2005. Post-irradiation examination (XRD, EPMA) are planned in 2006.
6. Conclusion

This paper focuses on a French extended research program on dispersed UMo fuel. An experimental determination of the U-Mo-Al phase diagram is in progress. The results reveal that the two previously reported ternary phases $U_3Mo_{4+x}Al_{43-x}$ and $UMo_{2.5}Al_{20.5}$ are non-stoechiometric, with significant homogeneity ranges. Up to 25% of Al can be substituted by Mo in $UAl_2$ all the while keeping the cubic $MgCu_2$-type structure, and a hexagonal $MgZn_2$-type $UAl_{2.5}Mo_x$ pseudo-ternary phase is stabilized for higher Mo contents. The Mo solubility is nil in $UAl_4$ and very low in $UAl_3$. Thermodynamical calculations will be further performed.

Couple diffusion studies are intended to improve our understanding of the phenomena and to select promising alternative solutions to UMo/Al problem. The first results underline the sensitivity of the UMo/Al interaction to, at least, the following parameters: Mo content, homogenisation, microstructure, structure and additives.

As an ultimate validation, irradiations have to be performed. High energy ion irradiation experiments are foreseen. Besides, 4 full-sized UMo LEU plate irradiation (IRIS3) is ongoing in OSIRIS. This experiment will test the effect of silicon added to aluminium matrix.
7. References

QUALIFICATION of HIGH DENSITY ALUMINIDE FUELS FOR THE BR2 REACTOR

André BEECKMANS de WEST-MEERBEECK
Pol GUBEL
Bernard PONSARD
SCK•CEN, BR2 reactor
Boeretang 200, 2400 Mol, Belgium

Thomas PIN
AREVA / CERCA
ZI Les Bérauds, 26104 Romans, France

Jean Louis FALGOUX
AREVA / CERCA
Tour FRAMATOME, 92084 Paris La Défense, France

ABSTRACT
The BR2 operation still relies on the use of 90..93% enriched HEU aluminide fuel. The availability of a limited batch of 73% enriched HEU from reprocessed BR2 uranium in Dounreay justified 10 years ago the qualification and use of this material. After some preliminary test irradiations, various batches of fuel elements were fabricated by the UKAEA-Dounreay and successfully irradiated. Due to their lower $^{235}$U content (0.050 g $^{235}$U/cm²), these elements were always irradiated together with standard 90..93% HEU fuel elements. A mixed-core strategy was developed at this occasion for an optimal utilization, and was reported during the 4th RRFM conference (March 19-21, 2000, Colmar, France).

The availability of a new batch of fresh 73% HEU material was the occasion, a few years ago, to initiate the development, fabrication and qualification of a new high density fuel element. An order was placed with CERCA to assess the optimal fabrication methods and tooling required to meet as far as possible the existing BR2 standard specifications and $^{235}$U content (0.060 g $^{235}$U/cm²). This development phase has been already reported during the 7th RRFM conference (March 9-12, 2003, Aix-en-Provence, France). Afterwards, six lead test fuel elements were ordered for qualification by irradiation. The neutronic properties of the fuel elements were adjusted and optimized.

After a short summary of the main results of the development program, this paper describes the nuclear characteristics of the high density fuel elements and comments on the nuclear follow-up of the lead test fuel elements during their irradiation for five cycles in the BR2 reactor and the return of experience for CERCA.

1. Introduction [1]

At the request of the SCK•CEN’s BR2 reactor, CERCA developed in 2002 a fuel element using high density plates with a 73% $^{235}$U enriched aluminide alloy instead of the standard 93% $^{235}$U enriched. The purpose of this development was mainly to maintain the $^{235}$U content per unit area at its 60 mg/cm² nominal value while keeping the same technical specifications for fabrication.

Due to the tolerances on enrichment, U content per plate, core thickness and others, the U content had to be increased from ~ 64,5 mg/cm² to a maximum of 86 mg/cm² (+ 33 %).
The CERCA specific development work, based on a Taguchi type test plan, demonstrated that, with some fabrication processes arrangement, it was possible to comply with the main constraints:

- Maintaining the quality of the meat distribution into the plate fissile area
- Maintaining the required cladding thickness, even in the so-called dog-bone area

In order to verify the general status of the plates, various inspection tests on non design-basis parameters were also performed. The absence of abnormal BT, UT or RT indications demonstrated that CERCA could guarantee the quality of the future products.

Designing the fuel plate is obviously essential in the qualification process of a new fuel element but it is only the first step. Indeed, the final issue is to verify that the fuel element can be normally utilized in the reactor without any detrimental effect. Therefore, six full scale lead test elements were manufactured and delivered to the BR2 in 2003, in order to check the perfect compliance of the new fuel elements in representative irradiation conditions.

As well, CERCA could validate whole of its fabrication methods. Among others, bending the plate was a process that had to be entirely reworked, because increasing the core load has direct consequences on the plate behavior during bending operation and then, on the element assembling conditions.

2. Nuclear characteristics of the high density fuel elements

The objective of the BR2 operation is to satisfy the irradiation conditions requested by the experimental load, and to do this by guarantying safe operation and by making optimal economical use of the available fuel elements.

The main nuclear characteristics of the BR2 reactor (a high flux MTR-type reactor) are the fast neutron flux available for experiments loaded in the axial position of a standard fuel element and the thermal neutron flux in the adjacent reflector channels. Those characteristics are limited by the maximum allowed heat flux at the hot spot. The safety criteria impose to keep a specified minimum negative reactivity worth available from shim rods at all times during a cycle (that may require a modification in the BR2 core arrangement).

BR2 standard fuel elements type 6n contain 6 concentric Aluminium fuel plates 0.127 cm thick 97 cm long; the side plates are 0.59 cm wide; fuel meat is a dispersion in Al powder 0.051 cm thick and 76.2 cm long; the inner available space has a diameter of 2.6 cm and the Be channel has a diameter 10.82 cm. The water gap is 0.30 ± 0.03 cm (see fig. 1).
The BR2 core configuration (fig. 2) reflects that all criteria are satisfied, e.g. the configuration 20G contains 34 fuel elements, 6 shim rods, 3 PWR loops, a flux trap in the central channel H1 and reflectors channels used for the regulating rod and for the production of radio-isotopes, most of them committed for the production of $^{99}$Mo.

The economical use of the fuel cycle is determined by the energy produced by a fuel element till it reaches the burn-up value at elimination. The residual fuel mass has been observed to be about 200 g $^{235}$U in an external channel during the last operation cycle. Routinely high reactive fuel at the 2nd batch are loaded in the central ring (channels A and B) and 6 to 8 fresh fuel are located in the second ring (channels C and D) in such a way the energy produced reaches at least 1200 MWd/cycle.

Repeating the reactivity calculation in an unitary cell as in previous communications [2,3,4], the adjoint of thermal neutron flux is the general shape of a configuration when the BR2 core contains an homogeneous fuel loading. The perturbation theory applied on the system: $B\Phi = \lambda \Phi$ gives the variation of the eigenvalue $\delta\lambda$ and the perturbed solution: $\Phi' = \Phi + \delta\Phi$; the variation of the eigenvalue is obtained with the bilinear product by weighting the neutron importance:

$$\delta\lambda = \lambda' - \lambda = \frac{<\Phi^+.(\delta B - \lambda \delta F)\Phi'>}{<\Phi^+.F\Phi'>} \quad \text{and} \quad (B' - \lambda' F')\delta\Phi = -(\delta B - \delta(\lambda F))\Phi$$

The comparison of the new fuel type "F" with the standard fuel type "G" and with other fuel compositions makes the main answer for the reactivity performance: reactivity when fresh, maximum reactivity and burn-up at elimination.

At the figure 3, the reactivity versus the equivalent burn-up for energy produced (taking into account the Pu production) shows the fuel element type "F" containing 400 g $^{235}$U enriched at 72.5% has the same reactivity curve as the standard BR2 fuel element type "G" containing 400 g $^{235}$U enriched at 72.5% meanwhile a modification in the content of burnable poisons. The number of fuel elements to produce 1000 MWd remains 6.2 elements permitting to maintain the routine schedule of BR2 operations. At a burn-up at elimination: 50%, the reactivity loss is calculated negligible: 0.006 $\times$ 34 el. = 0.20 $ for a whole configuration at the end of a cycle.
Table 1. Fuel Composition of BR2 Fuel Elements

<table>
<thead>
<tr>
<th>type 6 n</th>
<th>G</th>
<th>A</th>
<th>C</th>
<th>E</th>
<th>F</th>
<th>W</th>
<th>unit</th>
</tr>
</thead>
<tbody>
<tr>
<td>HEU</td>
<td>0.9</td>
<td>0.93</td>
<td>0.9</td>
<td>0.725</td>
<td>0.725</td>
<td>0.725</td>
<td>[-]</td>
</tr>
<tr>
<td>U-235</td>
<td>400.0</td>
<td>244.0</td>
<td>330.0</td>
<td>330.0</td>
<td>400.0</td>
<td>400.0</td>
<td>[g]</td>
</tr>
<tr>
<td>U-236</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>54.5</td>
<td>0</td>
<td>0</td>
<td>[g]</td>
</tr>
<tr>
<td>U-238</td>
<td>40.0</td>
<td>17.1</td>
<td>33.0</td>
<td>36.3</td>
<td>110.0</td>
<td>110.0</td>
<td>[g]</td>
</tr>
<tr>
<td>Sm2O3</td>
<td>1.4</td>
<td>-</td>
<td>1.3</td>
<td>1.3</td>
<td>1.4</td>
<td>-</td>
<td>[g Sm.nat]</td>
</tr>
<tr>
<td>B4C</td>
<td>3.8</td>
<td>-</td>
<td>2.8</td>
<td>1.8</td>
<td>3.2</td>
<td>-</td>
<td>[g B.nat]</td>
</tr>
<tr>
<td>Cd diameter</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>0.06</td>
<td>36*cm</td>
</tr>
<tr>
<td>density</td>
<td>0.060</td>
<td>0.037</td>
<td>0.050</td>
<td>0.050</td>
<td>0.060</td>
<td>0.060</td>
<td>g-235U/cm2</td>
</tr>
<tr>
<td></td>
<td>1.310</td>
<td>0.774</td>
<td>1.081</td>
<td>1.342</td>
<td>1.627</td>
<td>1.627</td>
<td>g-U/cm3</td>
</tr>
<tr>
<td>.mean</td>
<td>0.50</td>
<td>0.24</td>
<td>0.39</td>
<td>0.39</td>
<td>0.50</td>
<td>0.50</td>
<td>[-]</td>
</tr>
<tr>
<td>.max</td>
<td>0.62</td>
<td>0.32</td>
<td>0.50</td>
<td>0.50</td>
<td>0.62</td>
<td>0.62</td>
<td>[-]</td>
</tr>
<tr>
<td>max fission density</td>
<td>1.60</td>
<td>0.50</td>
<td>1.07</td>
<td>1.07</td>
<td>1.60</td>
<td>1.60</td>
<td>1e+21/cm3</td>
</tr>
<tr>
<td>number of batch</td>
<td>4</td>
<td>2</td>
<td>3</td>
<td>3</td>
<td>4</td>
<td>4</td>
<td>[-]</td>
</tr>
<tr>
<td>utilization</td>
<td>6.2</td>
<td>21.2</td>
<td>9.5</td>
<td>9.5</td>
<td>6.2</td>
<td>6.2</td>
<td>element/GWd</td>
</tr>
</tbody>
</table>

Figure 3. Calculated reactivity variation in unitary BR2 cell for various type of fuel elements

3. Experience gained with the irradiation of the lead test fuel elements [5]

The six test fuel elements were irradiated in the same circumstances that standard BR2 fuel elements are used. They were loaded firstly in the 2nd ring (channel C) when fresh and then in the central ring (channels A and B) and finally they were pushed at high burn-up rate successfully.
Table 2. Irradiation programme

<table>
<thead>
<tr>
<th>cycle</th>
<th>channel</th>
<th>burn-up EOC</th>
<th>days.eff</th>
</tr>
</thead>
<tbody>
<tr>
<td>2003-01</td>
<td>C</td>
<td>0.13</td>
<td>21</td>
</tr>
<tr>
<td>2003-02</td>
<td>examination</td>
<td>0.13</td>
<td></td>
</tr>
<tr>
<td>2003-03</td>
<td>A</td>
<td>0.32</td>
<td>28</td>
</tr>
<tr>
<td>2003-04</td>
<td>B</td>
<td>0.42</td>
<td>20</td>
</tr>
<tr>
<td>2004-05</td>
<td>D</td>
<td>0.50</td>
<td>21</td>
</tr>
<tr>
<td>2005-01</td>
<td>F</td>
<td>0.56</td>
<td>21</td>
</tr>
</tbody>
</table>

The reactivity effects of fuel elements type 6n "G" and 6n "F" loaded in the central ring (channels A or B, where the statistical weight is 2) have been measured – as a function of their mean burn-up – before the start-up of each cycle. Figure 4 shows that the measured reactivity curves are very similar and that there is no impact to expect on the fuel cycle.

Figure 4  Measured reactivity curves as a function of the mean burn-up of the fuel elements.

4. Experience feedback with the fabrication of the lead test fuel elements

After the successful irradiation of these 6 lead test elements, BR2 and CERCA joined together in order to define and agree on the final technical specifications for a large scale fabrication. The technical description was maintained at the highest level and CERCA integrated its R&D know-how in its fabrication process. New tools have been supplied in order to fit with the improved core compact design, rolling parameters have been fully rethought, bending tools and related processes have been adapted and assembling operation has been rechecked.

A fabrication order of 30 fuel elements is actually on-going in CERCA's plant in Romans. Plates are manufactured and inspected according to the above mentioned procedures and specifications. The fuel elements will be delivered in BR2 by August 2005 according to the agreed schedule.
The first industrial inspection results are conforming to the expectation. As an example the obtained cladding thickness (fig. 5) and U distribution (fig. 6) are as below:

Fig. 5a  Core center area
Fig. 5b  Dog bone area

Fig. 6  Plate U distribution

5.  Conclusions

CERCA proposed to develop, manufacture and deliver to the BR2 reactor a new design of fuel element based on a 73% $^{235}\text{U}$ enriched aluminide alloy as an alternative to the standard HEU element (93% $^{235}\text{U}$ enriched). The development challenge was to maintain the fuel element performances as high as possible while keeping the present acceptance criteria for the fabrication process. The results of the development program demonstrated it was possible to switch to the lower enriched alloy while maintaining the total amount of $^{235}\text{U}$ into the core.

Consequently, six full scale lead test elements were manufactured, delivered to the BR2, and irradiated for five cycles, reaching a final burn-up of more than 55%. The nuclear characteristics were chosen by design to meet those of standard HEU elements (93% $^{235}\text{U}$ enriched). The follow-up of these elements during their irradiation demonstrated that they completely met the requested performances and nuclear characteristics.

Final technical specifications were agreed between CERCA and SCK-CEN for a large scale fabrication taking account of the experience gained with the fabrication and irradiation of the six lead test fuel elements. A first batch of 30 elements is now under fabrication and should be delivered to the reactor site during the summer 2005. A second similar batch is foreseen for delivery in 2006.

The experience learned that the allocation of the necessary resources of a fabricator (CERCA) and an operator (BR2) was necessary to conduct successfully a development project in a timely manner.
6. References


