Transactions

RFM 2009

Vienna, Austria
22 -25 March 2009

IAEA
International Atomic Energy Agency

ATOMINSTITUT
TU WIEN

European Nuclear Society
Rue Belliard 65
1040 Brussels, Belgium
Phone + 32 2 505 30 54
Fax +32 2 502 39 02
E-mail ens@euronuclear.org
Internet www.euronuclear.org

ISBN 978-92-95064-07-2

These transactions contain all contributions submitted by 20 March 2009.

Session I

International Topics

RESEARCH REACTOR COALITIONS – SECOND YEAR PROGRESS REPORT

IRA N. GOLDMAN[1], PABLO ADELFANG[1], AND DANAS RIDIKAS[2]
1. Department of Nuclear Energy, 2. Department of Nuclear Sciences and Applications
International Atomic Energy Agency (IAEA)

KEVIN ALLDRED AND NIGEL MOTE
International Nuclear Enterprise Group

ABSTRACT

The IAEA, in line with its statute and mandatory responsibilities to support its member states in the promotion of peaceful uses of nuclear energy, has an initiative to promote the formation of coalitions of research reactor operators and stakeholders. These networks of research reactors are conducting joint research or other shared activities, have the potential to increase research reactor utilization and thus to improve sustainability at the same time enhancing nuclear material security and non-proliferation objectives.

This effort builds upon existing IAEA efforts to enhance research reactor strategic planning, to encourage formation of research reactor networks, and to promote regional and international cooperation between research reactors.

The paper will describe the Agency's progress in the second year of activities to assist in the formation of research reactor coalitions. The paper will describe the Agency's efforts in serving a catalytic and "match-making" role for the formation of new the coalition relationships, and its activities in organizing various missions and meetings for exploratory and organizational discussions on possible coalitions and networks.

The paper presents the concrete progress that has been made during the past year, including new coalitions in Eastern Europe, the Caribbean, Latin America and Central Asia. These coalitions cover a wide range of activities, for example, enhancing the regional infrastructure and capabilities for neutron sciences, developing new supplies of medicinal radioisotopes, and expanding the reach of reactor physics training courses. The paper also outlines the path forward that has been established for 2009 to support these coalitions as they mature and develop toward self-sufficiency.

1. BACKGROUND

In order to continue to play a key role in the further development of peaceful uses of nuclear energy, research reactors need to be financially sound, with adequate income for safe, secure, and reliable facility operations. In a context of declining governmental financial support, and the need to improve physical security and convert to LEU fuel, research reactors are challenged to generate income to offset increasing operational costs.

Reactors operating at low utilization levels have difficulty providing the service availability and reliability demanded by many potential users and customers, which creates a significant obstacle to increasing utilization. Many research reactors have limited access to potential customers for their services and are not familiar with the business planning concepts needed to secure additional commercial revenues or international program funding. This not only results in reduced income for the facilities involved, but sometimes also in research reactor services priced below full cost, preventing recovery of back-end costs and creating unsustainable market norms.

The research reactor community possesses the expertise to address these concerns, although, this knowledge is not uniformly available. Parochial attitudes and competitive behavior restrict information sharing, dissemination of best practices, and mutual support that could otherwise result in a coordinated approach to market development, building upon strengths of facilities. These attitudes are based, in part, on the belief that the markets for research reactor products and services are "zero-sum," with market gains by one research reactor resulting in losses to another "competing" reactor. However, the success of user groups and organizations such as WANO in the nuclear power generation sector show that the benefits of cooperation can be obtained without sacrificing commercial interests.

Renewed interest in nuclear power and the worldwide expansion of diagnostic and therapeutic nuclear medicine presents new opportunities for research reactors – including by providing services to countries without such a facility. However, constructing a new national research reactor might not be necessarily the optimum strategy. A reactor constructed to meet a specific need may lack sufficient identified utilization to fully occupy the facility, or it might not be adequately available for its intended purpose. One answer to this dilemma would be the creation of a new, high-specification multi-national region-based facility rather than a national facility. Of course, this requires an increased level of coordination between current and prospective operators.

To address the complex of issues related to sustainability, security, and non-proliferation aspects of research reactors, and to promote international and regional cooperation, the Agency has undertaken new activities to promote Research Reactor Coalitions and Centres of Excellence, in the first instance supported by grants from the Nuclear Threat Initiative (NTI). The effort also integrates Agency's regular and extra-budgetary funded program activities related to research reactors, national and regional IAEA Technical Cooperation projects, in particular "Enhancement of the Sustainability of Research Reactors and their Safe Operation Through Regional Cooperation, Networking, and Coalitions" (formerly RER/4/029, now RER/4/032); Supporting a Sustainable Increase in the Use of Research Reactors in the Latin American and Caribbean Region through Networking, Exchange of Experiences, Knowledge Preservation and Training of Human Resources (RLA/0/037 - ARCAL CXIX), "Nutritional and Health-Related Studies Using Research Reactors" (RAF/4/020 - AFRA IV-12), and "Integral Use and Safety of the Nuclear Research Reactor IAN-R1 (COL/1/10)."

The goals, objectives, and generic types of "model" coalitions were described in detail in a paper presented to RRFM 2008 ("Research Reactor Coalitions - First Year Progress Report, RRFM 2008 Proceedings.")

These activities are also part of IAEA and international efforts to minimize the use of Highly Enriched Uranium in civil nuclear applications. They complement and support IAEA activities on conversion of research reactor fuel to LEU and return of fresh and irradiated RR fuel to the country of origin.

2. RESULTS/ACHIEVEMENTS IN 2008

The Agency's role is to serve as a catalyst and a facilitator of ideas and proposals. Activities during the second year have resulted in the successful formation of four research reactor coalitions. Three coalitions have been concluded on the basis of Memorandum of Understanding or Practical Arrangements as follows:

- Caribbean Research Reactor Coalition: research reactors in Colombia, Jamaica, Mexico, and Austria; and the IAEA;

http://www.iaea.org/OurWork/ST/NE/NEFW/rrg_CRRC.html

- Eastern European Research Reactor Initiative (EERRI): research reactors in Austria (1), Czech Republic (2), Hungary (2), Romania (1), Poland (1), and Slovenia (1).

http://www.iaea.org/OurWork/ST/NE/NEFW/rrg_EERRI.html

- Eurasian Research Reactor Coalition (EARRC): research reactors in Czech Republic, Kazakhstan, Ukraine, and Uzbekistan) and isotope and other organizations (isoSolutions, Canada; Institute of Isotopes Co, Hungary; Curative Technologies Corporation, USA; Eckert & Ziegler, USA; and International Nuclear Enterprise Group, USA).

http://www.iaea.org/OurWork/ST/NE/NEFW/rrg_EARRC.html

A fourth arrangement was established without a formal agreement, based on on-going IAEA efforts to promote regional networking and utilization between research reactors and their users:

- Mediterranean Research Reactor Utilization Network (MRR-U): Azerbaijan, Egypt, Greece, Montenegro, Syria, Tunisia, and IAEA.

http://ipta.demokritos.gr/mrrun

Finally, as a result of Agency-sponsored coalition discussions in 2007 between ININ (Mexico), Laguna Verde Nuclear Power Plant (Mexico), and Atominstitut (Austria), the ININ Triga research reactor and Laguna Verde have entered into an agreement to hold regular training courses at the ININ TRIGA for Laguna Verde personnel.

3. MAJOR ACTIVITIES

Activities during 2008 focused on building upon opportunities for establishing coalitions identified during the exploratory discussions and missions carried out in 2007. These resulted in the conclusion of the formal arrangements cited in the previous section of this report (2. Results/Achievements in 2008).

In addition, a number of meetings and missions were organized and conducted to explore the possibility of forming other potential coalitions and networks.

A variety of international conferences, meetings, and workshops were utilized to publicize the concept of research reactor coalitions, inform potential participants on the activities taking place, to develop ideas and proposals for coalitions, and to recruit participants. Such opportunities were also utilized as appropriate for side meetings related to development of coalition arrangements.

Project planning and coordination was a central part of all project activities, including:

- Weekly conference calls with the project team and regular revision of an Action Item list in order to monitor accomplishment of assigned and scheduled tasks, and

- Project planning meeting 26 to 27 May 2008 in Vienna.

A. Development of Proposed Coalitions

Efforts focused on bringing to fruition research reactor coalition concepts concerning the Caribbean, Central Asia/Eurasia, Eastern Europe, and Russia. Accordingly, the following meetings and missions took place:

- Exploratory Meeting on East Europe Research Reactor Initiative (EERRI), KFKI/AEKI, Budapest, Hungary, 28 to 29 January 2008,

- Second meeting of EERRI, Atominstitut/Vienna University of Technology, Vienna, Austria, 4 to 5 September 2008,

- Discussions on Russian nuclear education and training, Moscow, Russia, 12 to 14 March 2008,

- Workshop on commercial production of radioisotopes (UK BERR/CNCP organized) Almaty, Kazakhstan, 21 to 23 May 2008, and

- Workshop on establishment of a Eurasia Research Reactor Coalition, Vienna, 24 to 26 November 2008.

The January meeting in Budapest resulted in conclusion of a memorandum of understanding for EERRI, and as well as a series of action items. Consequently, the second EERRI meeting was held in September with financial support of TC project RER/4/029, and resulted in the designation of coordinators for each of the possible cooperation areas (e.g. education and training, beam tube applications, radioisotope production, fuel and material testing) and new action items in particular for standardized information collection on reactor operating schedules, beam tube instruments and radioisotope production capabilities, as well as material/fuel test facilities (loops/rigs) of the individual reactors. Regular monthly EERRI teleconferences began to be held in October 2008 and focused on follow-up on the action items, most of which have been accomplished, considerably enhancing available information on EERRI. A separate paper on the activities of EERRI is being presented at the RRFM 2009 meeting.

The first concrete utilization-related activity arising from EERRI is a training course that will be implemented, with funding support from the IAEA West Asia TC program, in May-June 2009. This training course is also the focus of a separate paper at the RRFM 2009 meeting by Tozsér et al.

The meeting that took place in October 2007 in Mexico on possible formation of a Caribbean research reactor coalition was followed by regular teleconferences (twice monthly) throughout the year in regard to implementation of the Caribbean RRC. These calls and complementary work focused on negotiation and signing of a Practical Arrangement to formally establish the coalition; mutual assistance supported by a Colombian national TC project for training and certification of Colombian research reactor operators and for re-establishing neutron activation analysis activities (NAA) in Colombia.

The Practical Arrangement has been signed by all participants and is in effect. Recent tasks have included the drafting of a pamphlet intended to advertise the capabilities of the coalition, and development of a three-year work plan focused on achieving ISO certification for NAA services, and on improving customer and business related services, with technical assistance from the Delft University research reactor. This work plan, including specific training courses, expert missions and human resource development to support the long-range objectives of the coalition is expected to be included in the activities of the new regional TC project, RLA/0/037.

At its founding meeting in November, the Eurasia Coalition agreed to form an isotope production and distribution venture involving the participating reactors and other organizations. The goal is to create a credible alternative isotope supply system, capable of supporting the U.S., European, and developing markets, and able to play a key role in resolving the worldwide shortages in certain key medical isotopes. This will require not just an incremental increase in existing radioisotope production from the participating reactors,

but a comprehensive alignment of technologies, logistics, and production schedules that can only be achieved through cooperation. It was also agreed that a primary focus would be production of Mo-99, and it was determined that it would be technically possible to meet a buyer requirement for generators containing at least 1 Ci of Mo-99. A work plan was adopted calling for various technology and business studies to be conducted prior to a firm decision on how to proceed.

The IAEA established and maintains both public and project (password-protected) web sites for the Caribbean, East European, and Eurasia coalitions, which are being used as information, communication, and public information platforms.

Preliminary agreement was reached with Russian authorities on possible assistance for cooperative activities related to nuclear education and training utilizing research reactors. However, the restructuring of the Russian nuclear research and industry has resulted in a change of focus on possible coalition-related cooperative activities. Following discussions with Russian authorities, a formal proposal was prepared by the IAEA in October 2008 to jointly organize a feasibility study on conversion of IRT research reactors in Russia to use LEU fuel. The Russian authorities have also formally requested IAEA assistance to arrange visits and missions of Russian research reactor experts to well-established European facilities, with a focus on effective utilization strategies, including strategic and business planning for research reactors. This is expected to be implemented in mid-2009.

B. Identification of Additional Potential Coalitions

Efforts continued to further develop specific ideas involving research reactors in different regions and related to various scientific, technical, and commercial topics which to serve as the basis for coalitions. This took place primarily through continued discussions with research reactor operators, national nuclear authorities, irradiation services users and other stakeholders to identify opportunities.

- Meeting on Investigating Formation of Neutron Scattering Research Reactor Coalition, 11 to 13 February 2008, Vienna,

- Technical Assessment Mission for Establishment of an Ultra-Cold Neutron Source at TRIGA Pitesti, Romania, 7 to 8 October 2008,

- East Asia Coalition: centered at the Bragg Institute at the OPAL reactor in Australia, building upon its role as an IAEA Collaborating Center for Neutron Sciences, with participants from the Asia and Pacific region,

- Meeting on North-South America Research Reactor Coalition, 10 October 2008, Columbia, Missouri, CCHEN (Chile), University of Missouri RR (USA), and McMaster University (Canada),

- Baltic Research Reactor Coalition: Poland agreed to host an initial exploratory meeting in June 2009 for a coalition focused on nuclear education and training needs for future nuclear power development in the region. There have been positive indications of participation in the initial meeting from Denmark, Estonia, Lithuania, Norway, Russia (Gatchina, St. Petersburg), Germany (Berlin), Sweden, and Finland.

In addition, during 2008 substantial effort was made regard design of three IAEA regional Technical Cooperation projects (2009-2011) for Africa, Europe, and Latin America to support research reactor coalition activities. Project design efforts were carried out in consultation with regional counterparts and resulted in the extension/follow-on of the Europe regional project from 2007-2008; a new project in Latin America on research reactor coalitions and a

reformulation of an existing AFRA project on research reactor utilization to bring it in line with efforts to form coalitions and networks.

C. Outreach

Extensive coordination was maintained with other external programs or partners with related objectives. This included establishment of a working relationship with UK Department for Business, Enterprise and Regulatory Reform (BERR), Closed Nuclear Cities Program (CNCP), and U.S Department of Energy Office of Science for relevant activities which are supportive of project specific coalition development activities.

Various international meetings, conferences and symposia were used to publicize the Research Reactor Coalitions initiative, to solicit additional prospective partners for proposed coalitions, and to organize side meetings related to specific coalition activities. In this regard, presentations were made by IAEA representatives at the following meetings:

- European Nuclear Society, Research Reactor Fuel Management Meeting (RRFM 2007), Hamburg, Germany, 3 to 5 March 2007.

- European Nuclear Society, Nuclear Education – Science and Technology (NEST 2007), Budapest, Hungary, 5 to 7 May 2007.

- DOE Isotope Workshop, Rockville, Maryland, 5 to 7 August 2008.

- TRIGA International Meeting, Lyon, France, 8 to 9 September 2008.

Side meetings related to EERRI and the Caribbean Coalition, as well as to several other potential coalitions, took place at the NEST and TRIGA meetings.

4. PLANNED 2009 ACTIVITIES

Plans for 2009 seek to consolidate the existing coalitions and to encourage maturation, self-reliance and sustainability, new business/utilization activities, and participation by countries without ready access to research reactors. Work will also continue on other prospective coalitions that have been under discussion.

In particular, the ongoing crisis and unreliability of supply in the international isotope market – especially relating to Mo-99 – seems to offer a potential opportunity for new irradiation and processing capability which could ideally be satisfied through a coalition of producers.

Caribbean RRC: Initial efforts in 2009 are focused on preparations for the initial workshop for TC project RLA/0/037 in Bariloche, Argentina, 2 to 6 March 2009. As noted in Section 3 A above, it is anticipated that the new regional TC project will support a activities to be proposed by the Caribbean partners for specific activities over a three-year period in marketing, customer/business planning, and NAA quality certification. Two NAA expert missions (from Jamaica and Argentina) to Colombia will take place in February and April/May as the Colombia reactor has received approval for regular operations beginning in 2009. It is expected that 1-2 workshops will be held in 2009 on NAA and customer services. It is planned that a sub-regional workshop will be held in late 2009 or early 2010 to introduce the coalition to potential users/customers from countries in the Caribbean region that do not have access to research reactors.

East European RRI: A third plenary meeting will be held in Vienna after RRFM 2009 to review follow-up and progress since the September 2008 meeting and to intensify work on cooperative activities for 2009. It is planned that this meeting will begin to develop a long-range plan for the development of EERRI and that an irradiation services market study for

EERRI will be launched. As noted above, EERRI has organized a theoretical and practical training course in May/June for countries considering research reactor projects in Asia; this course is expected to be offered again later in 2009. The 5th Central European Training School on Neutron Scattering is expected to be held in Budapest in May and will provide an opportunity for further discussion on EERRI cooperation in the field of beam instruments.

Eurasia RRC: The second meeting will be held in Vienna prior to RRFM 2009 and will review action items tasked at the first meeting, especially financial and technical information needed to make decisions on coalition-based (n,gamma) Mo-99 production and prepare a business plan. Further activities during the year will be aimed at addressing the technology, logistics, and quality assurance requirements necessary to expand existing Mo-99 production from the coalition facilities.

Others: An initial meeting is planned for June 2009 in Poland to explore the possibility of forming a Baltic RR coalition for nuclear education and training. Efforts will continue to strengthen the Mediterranean network, to formalize cooperation in radioisotope production between Latin America and North America institutions (likely under the framework of TC project RLA/0/037), and to initiate neutron science activities in East Asia.

5. CONCLUSIONS

The Research Reactor Coalitions initiative followed a promising start in 2007 with a highly successful encore in 2008 as three formal coalitions were formed, one other network came into being, and several others were under discussion. The IAEA has successfully played the role of “catalyst” and facilitator of ideas and a kind of engine to coordinate and manage the everyday activities of coalitions.

While two years has been sufficient to bring research reactor coalitions from concept to initial reality, it has not been sufficient to realize their full objectives and hopeful promise.

The existing coalitions have agreed on the scope of expected activities, have established regular modes of communication and coordination, they are exchanging information, and are introducing themselves to the public. However, substantial work needs to be accomplished in order to realize the objective of increased utilization of the individual research reactors through collective effort and on a self-sustaining and self-reliant basis.

The coalitions need to put into place medium-term plans so that at the close of the current IAEA projects (end of 2011) they will be in a position to continue forward independently. They need to pursue more detailed market analysis and business development to identify specific opportunities for revenue generation through sustainable commercial activities, through complementary marketing and delivery of irradiation products and services including education and training. They need to establish and implement common quality control and assurance standards for their services based on accepted international standards as well as they need to establish mechanisms to facilitate access by non-commercial users in countries without access to research reactor services.

The IAEA remains open to suggestions and proposals from other Member States and institutions.

AN OVERVIEW OF THE GLOBAL THREAT REDUCTION INITIATIVE ACCELERATING THREAT REDUCTION

ANDREW J. BIENIAWSKI
Assistant Deputy Administrator
Office of Global Threat Reduction
National Nuclear Security Administration
U.S. Department of Energy

ABSTRACT

The Global Threat Reduction Initiative (GTRI) mission is to reduce and protect vulnerable nuclear and radiological materials located at civilian sites worldwide. GTRI helps the Department of Energy achieve its Nuclear Security Goal to prevent the acquisition of nuclear and radiological materials that could be used in weapons of mass destruction or other terrorists' acts.

Three key GTRI subprograms provide a comprehensive approach to denying terrorists access to nuclear and radiological materials and provide permanent threat reduction.

- The Convert subprogram supports conversion of domestic and international civilian research reactors and isotope production facilities from the use of HEU to LEU, and eliminates civilian use of HEU. These efforts result in permanent threat reduction
- The Remove subprogram supports removal or disposal of excess nuclear and radiological materials from civilian sites worldwide, and eliminates potential targets for terrorists to acquire nuclear and radiological materials. These efforts result in permanent threat reduction
- The Protect subprogram supports security of high-priority nuclear and radiological materials worldwide from theft and sabotage, and increases physical security at vulnerable nuclear and radiological sites.

1. Introduction – President Obama's Vision for Nuclear Security

President Barack Obama has a comprehensive strategy for nuclear security that will reduce the danger of nuclear terrorism, prevent the spread of nuclear weapons capabilities, and strengthen the nuclear nonproliferation regime.[i]

The National Nuclear Security Administration's Global Threat Reduction Initiative (GTRI) within the Department of Energy's (DOE) contributes to this strategy and has the mission to reduce and protect vulnerable nuclear and radiological material located at civilian sites worldwide. GTRI supports the U.S. Department of Energy's nuclear security goal by preventing terrorists from acquiring nuclear and radiological materials that could be used in weapons of mass destruction or other acts of terrorism.

The primary weapons of concern for GTRI are nuclear materials that could be used by terrorists to fabricate a crude nuclear bomb and radiological materials that would be most effective for a radiological dispersal device (RDD, also referred to as a 'Dirty Bomb'). Of particular concern are the thousands of civilian sites where nuclear and radiological materials are used for legitimate and beneficial commercial, medical, and research purposes. This is because civilian sites generally have less protection than military stockpiles of nuclear materials.

Nuclear material in the form of highly enriched uranium (HEU) and plutonium is located at hundreds of facilities in dozens of countries. Experts agree that the most difficult step for terrorists seeking to make a crude nuclear bomb is the acquisition of HEU and plutonium. The most likely route for a terrorist to acquire a crude nuclear device is to purchase or steal this material.

Also, millions of radioactive sealed sources are used around the world for legitimate and beneficial commercial application such as, cancer treatment, food and blood sterilization, oil exploration, remote electricity generation, radiography and scientific research. Many of these radiological sources are no longer needed and have been abandoned or orphaned; others are lightly guarded, making the threat of theft or sabotage significant. Currently there are thousands of civilian locations worldwide with dangerous high-activity radioactive sources.

Immediately after taking office, President Obama issued the goal to lead a global effort to secure all nuclear weapons materials at vulnerable sites in four years. He will also convene a Summit on preventing nuclear terrorism in 2009 (and regularly thereafter) of leaders of Permanent Member of the UN Security Council and other key countries to agree on preventing nuclear terrorism.[ii]

President Obama's goal has resulted in an acceleration of the GTRI work. New, more aggressive metrics have been established to achieve his goals. This paper will outline the work underway.

2. The GTRI'S Accelerated Plan

GTRI has three goals–Convert, Remove, and Protect--that provide a comprehensive approach to achieving its mission and denying terrorists access to nuclear and radiological materials.

- CONVERT. This program converts research reactors from the use of highly enriched uranium (HEU) to low-enriched uranium (LEU). These effort result in permanent threat reduction by minimizing and, to the extent possible, eliminating the need for HEU in civilian applications. Each reactor converted or shut down eliminates a source of bomb maternal.
- REMOVE. This program removes and disposes of excess nuclear and radiological materials. These efforts result in permanent threat reduction by eliminating bomb material at civilian sites.
- PROTECT. This program protects high priority nuclear and radiological materials from theft and sabotage. These efforts result in threat reduction by improving security on bomb material remaining at civilian sites--building by building—until a permanent threat reduction solution can be implemented.

2.1 The Convert Program

GTRI's Convert Program, also known as the Reduced Enrichment for Research and Test Reactors (RERTR), supports the conversion of domestic and international civilian research reactors and isotope production facilities from HEU to LEU. This includes working with Mo-99 producers to convert their operations to LEU targets. The Convert Program is key to the GTRI mission because it removes the need for HEU at civilian sites. Once the need is eliminated, any remaining HEU fresh and spent fuel can be permanently disposed of by GTRI's Remove Program.

The Convert Program metric is to convert or verify shutdown prior to conversion of 129 HEU reactors by 2018. Since 2004, 18 reactors have been converted, 6 of which were completed in 2008. Six more reactors are scheduled for conversion in fiscal year 2009.

To enable the conversion of 27 high-performance research reactors (6 of which are located in the U.S.), GTRI is developing a new ultra-high density LEU fuel. The GTRI established the Fuel Fabrication Capability (FFC) project to work with industry, the Nuclear Regulatory Commission, the U.S. national laboratories, and other entities to accelerate efforts to create a commercial scale capability to fabricate and supply this new ultra-high density U-Mo LEU fuel. This capability will allow the U.S. to meet its international commitment to HEU reduction. It is hoped that this model will encourage other countries to meet their requirements.

Also as part of the HEU minimization program, DOE is required by the Energy Policy Act (EPAct) of 2005 to report to the U.S. Congress on the commitments from commercial producers to provide Molybdenum-99 (Mo-99) to meet current and projected critical needs in the United States without the use of HEU. As a first step, the National Academies was commissioned to perform a study on the technical and economic feasibility of procuring Mo-99 from commercial sources that do not use HEU. The completed study entitled "*Medical Isotope Production without Highly Enriched Uranium*", which did conclude it is technically and economically feasible, was released on January 14, 2009. The EPAct also requires DOE to issue a report to Congress about the findings of the National Academies study and the existence of any commitments from commercial producers to meet 100% of the U.S. need for Mo-99 without HEU consistent with the feasibility criteria in the EPAct, and not later than the date that is four years after the date of submission of the report (which is anticipated for October 2009)**.**

The National Academies study also recommends that GTRI review other important considerations that will strengthen the HEU minimization program. These include: development of a worldwide data base of research and test reactors in coordination with the IAEA to include large pulse reactors, critical facilities, and reactors with a defense orientated mission; investigation to determine if it is feasible to convert these reactors and include in the scope GTRI's Convert program; and focus on eliminating HEU wastes that result from Mo-99 production facilities using U.S.-origin HEU and examining options to down blend the waste or encourage its return to the United States.

2.2 The Remove Program

GTRI's Remove Program supports the removal and disposal of excess nuclear and radiological material from civilian sites worldwide. These efforts result in permanent threat reduction by eliminating nuclear and radiological materials that terrorists could acquire. The materials include U.S.-origin, Russian-origin and "gap" material that are not covered under the U.S. or Russian programs. Excellent cooperation with partner countries has enabled the removal of 47% of the targeted vulnerable material to date. The complex nature of the nuclear material shipments requires close coordination amongst host country, transporters, and final destinations. GTRI completed 4 shipments in 2008, but in its effort to accelerate material removal, it has scheduled 9 shipments in 2009, more that double the removals.

Removal of abandoned radiological materials in other countries include radioisotopic thermoelectiric generators (known as RTGs), with emphasis on recovery within Russia. An ambitious goal of removal or disposal of 851 Russian RTGs by 2017 has been established. The close cooperation with Russian partners has resulted in successful removal of 50% to date.

The GTRI domestic radiological material removal program is working in cooperation with Federal, state and local agencies, and private industry to recover and permanently dispose of excess radiological sources in the United States. A cumulative total of over 20,300 domestic sources have been recovered. GTRI is committed to remove at least 2,500 excess domestic radiological sources each year. While this is an impressive milestone, each year over 3,000 new sources are registered as excess creating a backlog of more that 8,000 sources to recover.

2.3 The Protect Program

GTRI's Protect Program involves both international and domestic material protection. Work is conducted to ensure material security building by building. Many of the buildings holding nuclear and radiological materials require a different approach since they are accessible to the public, such as hospitals, and university facilities. A systematic approach is applied to evaluate and implement security measures. The GTRI team works with international and domestic partners to: perform site protection assessments; design the security upgrades; obtain a site sustainability commitment; install the security upgrades; and conduct table top exercises with facility staff to ensure an understanding and proficiency with the protection upgrade.

Working with Federal, state and local agencies, GTRI has established a domestic goal of 2,191 high-priority U.S. buildings that require protection. This work has recently begun, and 17 have already been completed

International nuclear and radiological material protection has identified 1,759 buildings that require protection. To date, 30% are completed. Acceleration of this effort is particularly important because upgraded security is necessary until a permanent threat reduction solution can be implemented

3. Conclusion

Secretary of State Hillary Clinton stated in her confirmation hearing on January 13, 2009, "……The gravest threat .....is the danger that weapons of mass destruction will fall into the hands of terrorists. To ensure our future security, we must curb the biological, chemical, or cyber — while we take the lead in working with others to reduce current nuclear stockpiles and prevent the development and use of dangerous new weaponry…."[iii] This urgent warning has been echoed by many other WMD experts and expert committees. In order to meet President Obama's goal to secure nuclear weapons material in four years, the NNSA has developed a plan to accomplish this goal. Every facet of the GTRI mission will contribute to the success of this plan. International partners around the world must be involved and will be welcomed in these important efforts.

4. References

[i] White House website, www.white.house.gov/agenda/homeland_security/
[ii] Ibid.
[iii] Testimony of Senator Hillary Clinton before the Senate Foreign Relations Committee, January 13, 2009.

FRM II AND AREVA-CERCA COMMON EFFORT ON MONOLITHIC UMO PLATE PRODUCTION

C. JAROUSSE, L. HALLE
AREVA-CERCA ♦- Les Berauds, B.P. 1114, 26104 Romans Cedex – France

W. PETRY, R. JUNGWIRTH, A. RÖHRMOSER, W. SCHMID
Forschungsneutronenquelle Heinz Maier-Leibnitz (FRM II)
Technische Universität München, D-85747 Garching, Germany

ABSTRACT

Since 2003 the Technische Universität München (TUM) is strongly engaged on the UMo fuel development program. Dispersive UMo fuel type was firstly investigated through collaborative efforts involving CEA, TUM and AREVA-CERCA. IRIS TUM irradiation and associated PIEs were presented during international conferences. The European consortium also studied the fabrication of UMo monolithic fuel plates during 2005-2007 and tentatively manufactured full size plates for the irradiation program IRIS V. As the program unfolded technical information were obtained and gathered with the international community. UMo foils were produced at laboratory scale and different methods to clad the UMo foils within aluminium were investigated. Based on these first results TUM and AREVA-CERCA have decided to pursue a common effort on the development of monolithic fuel plates with the purpose of minimizing the enrichment in use. Emphasis will be more dedicated to the foil and cladding package preparation and further studies will be performed in order to investigate various processing techniques to join the fuel foil with the cladding. As a complete new approach for the manufacturing of monolithic foils and for foil cladding the DC magnetron sputtering technique is investigated. A full R&D program was defined between TUM and AREVA-CERCA. This paper aims at presenting the program, discusses the selected options and first results will be presented

1. Introduction

The monolithic fuel concept presented by INL early in 2002 allows the possible LEU conversion of high performance reactors which can't be converted using dispersed UMo fuel. The basic principle is to replace a dispersive fuel form with a density of 8 gU/cc by a solid form of UMo where a density close to 17 gU/cc can be reached.

The technical challenges for this kind of fuel are multiplex: Instead of having numerous interfaces between the matrix and each grain the interface is now reduced to two large interfaces on top and the bottom of the monolithic foil. Which kind of interlayer is best suited for an excellent adhesion of the cladding and for suppressing the interdiffusion layer? How to produce large foils with a typical thickness of 300 – 500 µm on an industrial scale to affordable costs? How to introduce gradients in thickness for the foils? Can the assembled fuel plate be shaped to geometries others than flat? And last but not least, how do large tests plates behave under in-pile irradiations which simulate the heat load and burn-up of high performance reactors?

2. State of the art

For fuel plate design, new processes technologies as friction welding (FW), AREVA-CERCAs welding and hot isostatic pressure (HIP) were proposed and evaluated during the last few years. Miniplates manufactured by INL through FW and HIP processing were irradiated in ATR according to the RERTR irradiation program. These irradiation results have brought a basic set of information regarding monolithic fuel concept behavior under irradiation. Mainly the weak interaction layer (IL) which is formed during irradiation at the UMo foil and Al cladding front edge interface should be eliminated to prevent any detrimental debounding in the course of the fuel burn-up during operation.

Anyhow at the same time a diffusion barrier in between the different materials to be bounded must be observed in order to withstand the mechanical stress in use as well as guaranty the heat dissipation required by the performance of the fuel during irradiation. A dedicated material which can simultaneously bring a good mechanical bound between the cladding and UMo foil and also reduces the formation of the IL should be defined.

A full R&D program is then required to down select such materials and to study a processing method to form this barrier. The beginning easy principle becomes more complex and a monolithic plate appears as a challenge to assemble a multilayer material: cover, barrier and UMo foil. AlSi or Zr layer have been proposed as a remedy for the bonding and diffusion problem and are still being investigated by INL [1].

More challenging, the fuel meat which is a solid UMo foil has to be produced at a large scale and throughout a cost effective processing way. UMo foils elaboration at a laboratory scale was described previously [2] and developments are ongoing at Y12 complex to scale up the UMo foils production [3].

The UMo foil production must include a step where the needed barrier that should wrap up the UMo foil surface has to be processed. Specifically shaped fuel plates including gradients in the thickness of the UMo foil have also to be considered as it may be necessary for research reactors with highest power density. This aspect needs to be further studied in order to integrate this demand early in the manufacturing foil development stage. Dedicated program was recently launched by INL to evaluate the production of shaped UMo foils [4]

As previously described a stainless material barrier can also be advantageously used to protect the UMo foils from oxidation prior or during monolithic plate processing [5].

Recently 4 full size monolithic plates composed of AlSi and Zr barrier processed by FW were irradiated through AFIP 2 experimental test with a peak power up 350 W/cm^2 (US/DOE program carried-out by INL). PIEs results are awaited for this year. The preliminary ultrasonic test inspection results performed by INL seem to indicate a relative good plate behavior under irradiation. The prototypes manufacturing results presented by INL state the difficulties to master routinely the monolithic plate fabrication.

Following a common effort started 5 years ago through a collaborative program carried out between CEA, AREVA-CERCA and TUM, TUM and AREVA-CERCA have commonly defined a new R&D program aiming at evaluating and testing the main challenges on the road of monolithic plate processing. In order to save time and taking into account to the ongoing development dedicated to UMo foil production our program will be conducted using depleted UMo foils coming from dedicated outside entities.

The monolithic program has been split down into sub items as shown in Table 1.

The program is scheduled to start this year and final results are forecasted at the latest beginning of 2011. As soon as successful results will be available an irradiation program encompassing the best down selection options will follow. In the following section each item is further detailed.

Items	Objective	Dedicated tools
Depleted UMo Foil	To have foils in order to carry-out the program (uncoated and coated foils)	
Material down selection & processing	To down select an appropriate material barrier between UMo foil and cladding To depose a selected material on UMo foils	Heavy ion irradiation of a set of UMo samples by FRM II Sputtering at FRM II
Monolithic plate processing	To investigate the material compatibility through the selected monolithic plate processing To further evaluate monolithic plate processing through AREVA-CERCAs processing options i.e. C2TWP and/or HIP	AREVA-CERCAs machineries FRM II characterization tool box

Table 1: FRM II & AREVA-CERCA monolithic program summary

3 Material down selection & processing

3.1 Options to reduce the formation of an unwanted IL at the UMo-Al interface

Regarding the UMo/Al system there are in principal three possibilities reported to avoid the formation of an undesired IL during pile irradiation. A literature inquiry in current and older publications revealed the most promising candidates:

a) Addition of a diffusion-limiting element to the Al-matrix which contains the UMo particles.

   It has been found that the addition of Silicon to the Aluminium matrix limits the formation of an interdiffusion layer at the interface UMo-Al. However, uncertainties remain for the optimal Silicon addition that should be added to the Aluminium matrix [6, 7, 8]. Moreover, diffusion coefficients given in older literature show, that the addition of Titan, Bismuth, Beryllium or Antimony to the Aluminium matrix is even more powerful in suppressing the formation of the interdiffusion layer than Silicon. Especially Bi has been found to be most promising due to its low neutron cross section and its high density [9].

b) Usage of ternary UMo alloys.

   Creation of ternary U-Mo-X alloys seems to have in some cases a suppressing effect on the formation of the interdiffusion layer. The main reason for this effect is the stabilization of the UMo γ-phase by the third element [10,11]. Consequently, the addition of some wt% of Ti, Nb, Pt, Si or Pd to the U-Mo has been considered [12,13].

c) Insertion of a diffusion barrier at the interface UMo-Al.

It is known since the early days of metallic fuel element development that an oxide layer at the interface Uranium-Aluminum prevents the formation of a diffusion layer very effectively [9]. Also in UMo-Al test fuel plates irradiated in-pile an oxide layer around the UMo particles has proven its effectiveness [14]. Furthermore, Nb, Ta, Ti or Zr has been proposed as a diffusion barrier [14, 15, 16].

d) Any combination of a), b) and c)

e) Usage of a completely different matrix material, such as Magnesium.

In 2006 it has been shown that it is possible to emulate the IL growth during in-pile irradiation of UMo/Al specimens by out-of-pile irradiation with ^{127}Iodine at 80 MeV. Typical burn-up fission densities are achieved within a few hours [17,18]. Because the energy of the ions usually is far below the Coulomb barrier the UMo samples are not activated during heavy ion irradiation. They are therefore easily accessible with normal laboratory equipment (SEM, EDX, and XRD).

Since then, considerable progress has been made to improve the reliability of this method. In collaboration with CEA-Cadarache, TUM has build up a complete new irradiation setup at the tandem accelerator in Garching (Maier-Leibnitz Laboratorium) which allows monitoring and controlling the irradiation conditions like flux, fluency, vacuum and sampling temperature automatically. The new setup allows the quick irradiation of different kinds of samples [19].

Consequently, we decided to start an irradiation campaign at the tandem accelerator in Garching to screen as many combinations of the materials mentioned above as reasonably possible. For the irradiations UMo powder dispersed in an Aluminum matrix has been chosen. The results are also valid for the here described monolithic fuel development program, because anyhow the interaction between Al matrix, IL and UMo will be the focus of our characterization.

We decided to divide the options mentioned into three parts (compare Tab. 2). For each option, one miniplate (in total 20) has been provided by AREVA-CERCA.

The first part consists of atomized U7wt%Mo powder dispersed in an Al matrix with and without addition of secondary elements. 2wt%Si, 5wt%Si and 7wt%Si have been chosen as addition to the Al matrix to find the best Silicon concentration. Furthermore, 2wt%Ti, 2wt%Bi and 5wt%Bi have been added to the Al matrix, respectively, to check the positive effect of these elements. It has been reported, that the addition of Magnesium to Aluminum accelerates the diffusion at the U-Al interface [8]. In consequence, 2wt%Mg has been added to the matrix to check this effect. In each case the samples have been prepared with and without an oxide layer (UO_2) of ~2µm thickness around the UMo particles, to check its effectiveness as a diffusion barrier – second part.

The third part consists of ternary U8wt%Mo-x ground powder dispersed in a pure Aluminum matrix. To study the principal effect of alloying the UMo, 1wt%Ti, 1,5wt%Nb, 3wt%Nb and 1wt%Pt have been added. One miniplate consisted of UMo dispersed in Magnesium, covered with AlFeNi. It was not possible to prepare a sample for irradiation from this miniplate. The meat was grayish and very brittle. It broke apart during cutting and polishing.

It was not possible to obtain UMo particles coated with metals like Ta or Ti to check the impact of those elements on the formation of the diffusion layer during bombardment with heavy ions. However, examinations on this issue are planned on monolithic UMo/Al layer systems that will be prepared with the newly installed sputtering device.

From the 20 miniplates provided by AREVA-CERCA 60 samples have been prepared for irradiation with ^{127}Iodine at 80 MeV. First irradiations have been performed since fall 2008 with an integral of $1x10^{17}$ ions/cm² for each sample. This corresponds to a full burn-up of a

high flux research reactor. The irradiation temperature was ~200°C. The irradiation campaign will be finished until summer 2009. Examinations on already irradiated samples are ongoing. The full set of data will be available until end of 2009.

Sample Nr.	Alloy	Matrix	Comment
MAFIA-I-1*	U-7Mo	Al-atomized	Reference specimen
MAFIA-I-2*	U-7Mo-ox	Al-atomized	
MAFIA-I-3*	U-7Mo	Al98-Si2	Different Si concentrations to find the best concentration
MAFIA-I-4*	U-7Mo-ox	Al98-Si2	
MAFIA-I-5*	U-7Mo	Al95-Si5	
MAFIA-I-6*	U-7Mo-ox	Al95-Si5	
MAFIA-I-7*	U-7Mo	Al93-Si7	
MAFIA-I-8	U-7Mo-ox	Al93-Si7	
MAFIA-I-9	U-7Mo	Al98-Mg2	Mg accelerates formation of IDL. Reproduction of this effect.
MAFIA-I-10	U-7Mo-ox	Al98-Mg2	
MAFIA-I-11	U-7Mo	Al98-Ti2	Study the effect of Ti on IDL formation
MAFIA-I-12	U-7Mo-ox	Al98-Ti2	
MAFIA-I-13	U-7Mo	Al98-Bi2	Different Bi concentrations to find the best concentration
MAFIA-I-14	U-7Mo-ox	Al98-Bi2	
MAFIA-I-15	U-7Mo	Al95-Bi5	
MAFIA-I-16	U-7Mo-ox	Al95-Bi5	
MAFIA-I-17	U-7Mo	Mg	Mg matrix, did not work, brittle matrix, matrix with no adhesion to cladding
MAFIA-I-18	U-7Mo-1Ti	Al-ground	To study principal effect of alloying the UMo on formation of IDL.
MAFIA-I-19	U-7Mo-1,5Nb	Al-ground	
MAFIA-I-20	U-7Mo-3Nb	Al-ground	
MAFIA-I-21	U-7Mo-1Pt	Al-ground	

Tab 2: List of mini-plates provided by AREVA-CERCA for heavy-ion irradiation. The irradiation has been completed for miniplates earmarked with (*).

3.2 Sputtering techniques

The sputtering process offers the advantage, that perfect layers from any material can be grown on any substrate in any size. This means to the first, that monolithic full size foils and blank sheets from any given UMo alloy can successfully be produced. It means to the second, that a given UMo foil or blank sheet can be surrounded with a layer of any desired material, be it as a diffusion barrier or as a cladding. It means to the third, that bonding between the different layers is not a problem at all, because the layers will have the maximum physically possible adhesion to each other.

On laboratory scale the sputtering technique provides the opportunity to quickly produce small numbers of high quality full size fuel plates for irradiation tests and further examinations, which is our primary aim. The applicability of sputtering to industrial scale seems promising but has to be further examined.

Over the last two years TUM has built up two DC magnetron sputtering assemblies: a small tabletop setup for the production of samples sized 100mm x 100mm as well as a large plant – see Fig. 1 - for the fabrication of plates sized 700mm x 65mm. Both assemblies have successfully been operated in the past with surrogate materials as copper or stainless steel and the process of full size foil production as well as the process of cladding respectively depositing a barrier layer on these foils have been shown successfully [20].

End of 2008 first DU-8wt.%Mo (depleted Uranium) foils have been sputtered successfully. DU-8wt.%Mo foils with 120mm x 50mm in size and 150 µm thickness were produced inside the tabletop assembly in 28 hours of sputtering. The surface quality of these foils is still poor due to thermal stress effects. A phase analysis of the deposited layer showed the DU-8wt.%Mo to be in the desired γ-phase after the sputtering process. These experiments are continued with the aim to reduce the thermal stresses.

In parallel the large sputtering plant was mounted inside a glove box to enable operation under inert atmosphere. The inert atmosphere guarantees, that the concentration of oxygen during the production of foils, barrier layers or cladding and even during handling of the material is always below 10ppm, which results in a nearly complete suppression of oxidation and oxide layer formation in all process steps.

The sputtering plant is currently installed in a radioisotope laboratory and will be ready for operation within the next weeks.

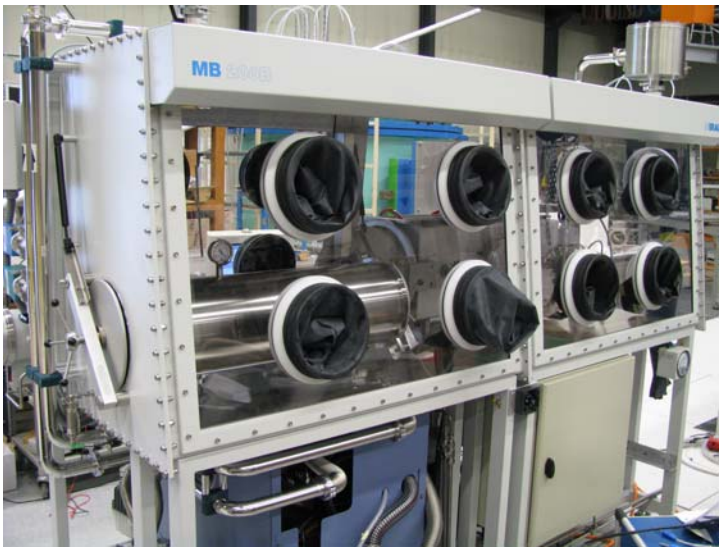


Fig. 1: DC-magnetron sputtering plant for foil production and barrier / cladding deposition in full size.

The main emphasis of the large sputtering device will be barrier coating and cladding deposition. Manufacturing of full size foils by sputtering DU-8wt.%Mo will be continued, too. But the perspectives to introduce this technique for an industrial production of UMo foils are still in the future.

4 Monolithic plates processing

As the main conclusion of the monolithic plate processing program carried-out in 2005-2007 we found that the difficulty encountered to clad an UMo foil with Al covers using an AREVA-CERCAs methodology was linked to the UMo foil oxidation during plate processing. This effect was especially observed during the production of UMo full size plates [2]. Replacing the UMo by a stainless steel foil of the same thickness a perfect bounding junction was always obtained. During these tests the UMo foil surface was free of any material barrier which is today strongly advised for preventing any wrong irradiation behaviour due to the IL growth formation at the Al/UMo interface. This barrier can also be advantageously used to prevent the oxidation of the UMo foil and then improved the efficiency of our processing method. Processing parameters can also be adjusted in order to reduce the temperature preventing oxidation.

The new program is structured into several phases where a material compatibility study shall be performed first. This stage should be performed to validate the compatibility of the barrier material we want to use with the Al covers and also the processing tool deployed. After this material down selection (heavy ions studies and compatibility aspects) miniplates and full size plates will be processed. In case of unexpected results recorded with this first manufacturing option a processing back-up program will be launched. The back-up option is based on HIP processing.

As previously did the quality of multilayer interfaces bounding will be checked through ultrasonic test (UT), metallographic inspection and dedicated mechanical test at TUM. The characterization tools available are presented in the following section. As proposed and already under evaluation in the US side the first evaluation will start with Zr coated depleted UMo foil.

5 Characterization

The here presented R&D program will benefit from the huge variety of methods and competences for the physical and chemical characterization, which are offered by the university campus at Garching. The nuclear operation licences of the Institute of Radiochemistry and the FRM II itself allow the handling of α-emitters like Uranium. Scanning electron microscope and EDX can be used with open α-emitters. XRD will serve for structural and phase analysis. FRM II itself is one of the leading neutron scattering facilities and offers among others non-destructive analysis of the internal strains introduced by the processing techniques (deposition, plate processing and bending). Micro focus beams of synchrotron radiation allow the spatial resolution of structures on a scale smaller than 1 µm [21]. Precise chemical element analysis serves as a gauging for locally resolved EDX analysis. Mechanical pulling tests will check the adhesion of the sputtered layers.

6. Conclusion

A new R&D monolithic program was defined in collaboration with AREVA-CERCA and TUM.

From barrier material down selection to monolithic plate processing our program encompasses the overall identified aspect which needs to be investigated to successfully produce a monolithic UMo plate.

The tools used will be adapted according to the maturity of the technology from a R&D lab scale to an industrial workshop. A part of the results will benefit to the ongoing UMo dispersive development program.

As soon as our program will answer to the main challenges, an irradiation of monolithic UMo LEU plates will be considered.

References

[1] D. AB. Robinson et al, "*US RERTR fuel development program post irradiation examination results"*, International meeting on Reduced Enrichment for Research and Test Reactors (RERTR 2008), Washington, DC, USA 5-9 October 2008

[2] C. Jarousse et al, "*Monolithic UMO full size prototype plate for IRIS V irradiation",* 11th International Conference on Research Reactor Fuel Management (RRFM 2007), Lyon, France, March. 11 – March 15, 2007.

[3] A.Demint et al, "*Update on Uranium-Molybdenium Fuel foil fabrication development activities at Y-12 National Security Complex"*, International meeting on Reduced Enrichment for Research and Test Reactors (RERTR 2007), Prague, Czech republic 23-27 September 2007

[4] N.E. Woolstenhulme et al, "*Fabrication of contoured monolithic foils*" International meeting on Reduced Enrichment for Research and Test Reactors (RERTR 2008), Washington, DC, USA 5-9 October 2008

[5] C. Jarousse et al, "*Last manufacturing results of monolithic UMo full size prototype plates",* RERTR 2006, Cape Town, South Africa, Oct. 29 - Nov. 3, 2006

[6] M. Ripert et al, "*Iris 3 Experiment – Status and Result of thickness increase"*, RRFM 2006, Sofia, Bulgaria, Apr. 30 – May 3, 2006

[7] G.L. Hofman et al, "*Results of Low-Enriched U-Mo Miniplates from RERTR-6 and -7A Irradiation"*, RERTR 2006, Cape Town, South Africa, Oct. 29 - Nov. 3, 2006

[8] M. Cornen et al, "*Physico-chemical aspects of modified UMo/Al interaction"*, RRFM 2007, Lyon, France, Mar. 11-15, 2007

[9] R.O. Williams, Terminal report on ORNL slug problem – causes and prevention, CF-50-7-160, 1950

[10] J.M. Park et al, "*Phase stability and diffusion characteristics of U-Mo-X (X=Si, Al, Zr) alloys"*, RERTR 2005, Boston, Massachusetts, USA, Nov. 6-11, 2005

[11] M. Rodier et al, "*Effects of Ti in the UMo/Al system: preliminary result"s, RRFM 2007*, Lyon, France, Mar. 11-15, 2007

[12] R.J. Van Thyne, "*Transformation kinetics of Uranium-Niobium and ternary Uranium-Molybdenum base alloys"*, Transactions American Society for Metals 49, pp 576-597, 1957

[13] G.L. Hofman et al, "*Design of high density γ-phase Uranium alloys for LEU dispersion fuel applications*", , RERTR 1998, Sao Paulo, Brazil, Oct. 18-23,1998

[14] G.A. Birzhevoy et al, "*Results of post-irradiation examination of the (U-Mo)-Aluminum matrix interaction rate"*, RRFM 2007, Lyon, France, Mar. 11-15, 2007

[15] H.A. Saller, *Uranium alloy newsletter 13*, WASH-295, USA, 1955

[16] D.D. Keiser et al, "*Characterization and Testing of Monolithic RERTR Fuel Plates"*, RRFM 2007, Lyon, France, Mar. 11-15, 2007

[17] N. Wieschalla et al, "*Heavy ion irradiation of U–Mo/Al dispersion fuel*, Journal of Nuclear Materials", 2006, 357, 191-197

[18] H. Palancher et al, "*Uranium–molybdenum nuclear fuel plates behaviour under heavy ion irradiation: An X-ray diffraction analysis"*, Journal of Nuclear Materials, 2009, in press

[19] E. Welcomme et al, "*Heavy ion irradiation of U7Mo/Al fuel: methodological approach"*, RRFM 2009, Vienna, Austria, Mar. 22-25, 2009

[20] W. Schmid et al, "*Manufacturing of thick monolithic layers by DC-magnetron sputtering"*, RRFM 2008, Hamburg, Germany, Mar. 2-5, 2008

[21] H. Palancher et al, "Characterization of the Interaction layer between decomposed UMo7 and aluminum using micro-focused XRD on a single particles " P. Lemoine, RERTR 2008, Washington, USA, Oct. 5-9, 2008

2009 CNEA REPORT ON PROGRESS ON THE DEVELOPMENT OF LEU FUELS and TARGETS in ARGENTINA

S. BALART, P. CRISTINI, C. FERNÁNDEZ, A. G. GONZALEZ, M. LÓPEZ, H. TABOADA

Comisión Nacional de Energía Atómica
Av. Del Libertador 8250 (1429) Ciudad de Buenos Aires, , República Argentina

ABSTRACT

Since RRFM 2008 meeting, held in Hamburg, Germany, LEU fuel and target R&D activities in CNEA were focused on: the commissioning for operation of the LEU core converted RA-6 reactor, applied research and development on dispersed U-Mo/Al(-Si) in Al cladding and monolithic U-Mo in Zry-4 cladding concepts to understand reaction mechanisms in interaction zone formation and developing promissory solutions for VHD monolithic and dispersed fuels and also CNEA continued deploying R&D on LEU target and radiochemical technology for radioisotope production, meeting international quality standards

1. Commissioning for operation for the RA-6 reactor with its new LEU core.

The RA-6 reactor is a pool-type 0.5MW one sited at San Carlos de Bariloche city, Province of Río Negro, Argentina. During 2008 following tasks took place:

- Transport of the new LEU core from the ECRI plant sited at the Constituyentes Atomic Center (close to Buenos Aires City) to the RA-6 reactor building. This new core is made of 25 regular and control fuel assemblies with a dispersion of U3Si2 and Al powders, 4.8 gU/cm3 density, Cd wires were employed as neutron absorbers.
- Preliminary tests for loading the new core started in September 18th, 2008.
- Criticality start-up operations, started on January 19th, 2009.
- Formal re-inauguration took place on March 16th, 2009

This last operation completes the successful conversion process started in October 30th, 2005 with the signature of two contracts between CNEA and NNSA-DoE and comprised swapping of HEU-LEU inventories, exportation of HEU SNF US-origin to USA, and fabrication of the conversion core and new graphite reflectors and improvements on primary and secondary loops.

2. Applied R&D on dispersed and monolithic U-Mo fuels.

- New results from out-of-pile tests using high intensity synchrotron X-rays diffraction techniques performed in the LNLS Campinas, Brazil are presented. Characterization of phases in the interaction layer, after 340ºC and 550ºC temperature treatment in diffusion couple samples of U-Mo(-Zr) / Al(-Si) were optimized.

3. Development and irradiation of promissory solutions to VHD monolithic and dispersed fuels technical problems

- Some improvements were done in the development of dispersed and monolithic fuels made with Uranium-molybdenum alloy. The purpose is to have additional alternatives to cover HEU-LEU conversion possibilities.
- The IAEA's Technical Cooperation, the project ARG/4/092 to irradiate and PIE in a high flux reactor a full scale fuel assembly prototype finished the tender process and a provider was selected.

- CNEA have been working in the fabrication of depleted 8 gU/cm^3 U-7Mo based miniplates. Once this process is stable and repetitive, the fabrication of a LEU prototypic fuel assembly will follow.
- In order to avoid an undesirable porosity in the aluminium side of the interaction zone with U-Mo due to the migration of gas fission products, and according to studies done on the convenience to add a proper component to matrix powder, Al-Si alloys were employed. Fabrication conditions of dispersed U-Mo miniplates and scaling-up to plate size related problems were achieved.
- Concerning very high density monolithic U-Mo both miniplates and plates, using MEU and LEU fuel meat with Zry-4 cladding to be irradiated in USDoE-ATR Reactor, are being developed
- In order to improve material performance and plate dimensioning fabrication conditions were studied and modified, like hot co-lamination of U-Mo and Zry-4 sheaths. Several depleted uranium prototypes were elaborated, characterized and tested to set up process variables and fabrication conditions.

4. **Improvement of the LEU target and radiochemical technology for Mo99 and other radioisotopes production:** It was already presented that CNEA has decided on 2001 to turn into LEU material for target fabrication, maintaining other characteristics of the production, i.e. the alkaline chemical digestion process. CNEA has been producing Mo-99 using LEU since 2002.CNEA produces Mo-99 primarily for its domestic market and secondarily for export to other South American countries. It began producing Mo-99 using HEU targets in 1985[i] and developed and converted to LEU-based production in 2002. CNEA manufactures its own uranium-aluminum alloy plate LEU targets[ii].
   - CNEA has developed and is using high-density LEU-aluminum dispersion targets to produce Mo-99 for its domestic market. The target meat has a density of 2.9 gU/cm3, which is obtained by increasing the ratio of uranium aluminide to aluminum in the target meat. The mass of U-235 in the target meat is about twice that of conventional uranium-aluminum alloy targets.
   - CNEA was able to convert to LEU-based production in the same set of hot cells that were being used for HEU-based production. Moreover, this conversion was made without interrupting Mo-99 production
   - Targets are irradiated in the RA-3 reactor at CNEA's Ezeiza Atomic Center near Buenos Aires. Target processing is carried out in a hot cell facility at the Ezeiza site. Process wastes are also managed at the site.
   - CNEA's development showed that there are no technical barriers to conversion of Mo-99 production from HEU targets to LEU targets. Production using LEU targets is technically feasible and is being carried out by CNEA in Argentina and will be shortly by the Australian National Nuclear Science and Technology Organisation (ANSTO) using CNEA technology using CNEA-developed LEU targets and target dissolution process to produce Mo-99.
   - This new LEU technology satisfies the most stringent requirements of quality for its use in nuclear medicine applications. Mo-99 purity has been consistently higher than that produced using HEU targets[iii]
   - Also in September 2005, CNEA began the regular production of high quality fission I-131, a by-product of Mo-99 production, meeting also international quality standards.
   - New results are that HEU-LEU production process comparison costs reveals that this new technology has no significant over cost. CNEA recently presented a comparison of its variable costs for producing Mo-99 using LEU and HEU targets[iv]. Variable costs for Mo-99 production for the three years prior to (i.e., 1998-2001) and three years following (2003-2007) conversion were compared. Costs were presented in three categories: (1) labor; (2) materials; and (3) services, maintenance, taxes, and miscellaneous. The costs were presented as present value estimates normalized on a per curie basis for the number of curies produced

in 2007. Overall costs for LEU-based production compared to HEU-based production increased by about 5 percent.

- Conclusions: no technical, quality or financial reasons make disadvantageous to change from HEU to LEU radiochemical technology for Mo99 and other radioisotopes production. CNEA becomes a leader in LEU based isotope production technology, and with INVAP built all-LEU production systems in Australia and Egypt
- Future plans: at present, CNEA is considering to expand Mo-99 production within its current facilities by increasing target throughputs. Such an expansion would put CNEA in the ranks of large-scale producers

.

5. Conclusions: CNEA continues deploying development activities on LEU technology for core reactor conversion and Mo99 and related radioisotope production. Future plans include prototypic fuel irradiation and optimization of LEU targets and alkaline digestion process.

---

[i] Cols, H. J., P. R. Cristini, and A. C. Manzini. 2000. Mo-99 from low-enriched uranium. 2000 International RERTR Meeting, Las Vegas, Nevada, USA, 1-6 October, 2000. Available at http://www.rertr.anl.gov/Web2000/PDF/Cristi00.pdf

[ii] Kohut, C., M. de la Fuente, P. Echenique, D. Podesta, and P. Adelfang. 2000. Target development of low enrichment for production of 99Mo for fission. 2000 International RERTR Meeting, Las Vegas, Nevada, USA, 1-6 October, 2000. Available at http://www.rertr.anl.gov/Web2000/PDF/Fuente00.pdf

[iii] Durán,A. 2005. Radionuclide Purity of Fission Mo-99 Produced from LEU And HEU. A Comparative Study. 2005 International RERTR Meeting, Boston, Massachusetts, USA, November 6-10, 2005. Available at http://www.rertr.anl.gov/RERTR27/PDF/S8-3_Duran.pdf.

[iv] Cestau D., A. Novello, P. Cristini, M. Bronca, R. Centurión, R. Bavaro, J. Cestau, E. Carranza. HEU and LEU cost comparison in the production of molybdenum-99. 2008 International RERTR Meeting, Washington, DC, USA, 5-9 October 2008, and Cestau D., A. Novello, P. Cristini, M. Bronca, R. Centurión, R. Bavaro, J. Cestau, E.Carranza. 2007. HEU and LEU comparison in the production of molybdenum-99. 2007 International RERTR Meeting, Prague, Czech Republic, Sep. 23-27, 2007. Available at http://www.rertr.anl.gov/RERTR29/PDF/6-4_Cestau.pdf

THE JULES HOROWITZ REACTOR PROJECT: STATUS AND FUEL LICENSING PROCESS

M.C. ANSELMET, G. BIGNAN, D. IRACANE, P. LEMOINE,
B. MAUGARD, P. TREMODEUX

,
Commissariat à l'Energie Atomique (CEA)
Direction de l'Energie Nucléaire
Centre d'études de Saclay et Cadarache
France

1 Context – Motivation for the development of the JHR

Sustainable and safe nuclear energy requires up-to-date experimental capacity to test fuel and material behaviour under irradiation with a high level of performances in order to meet needs of industry, research and public bodies:

- A constant improvement of the performances and safety of present and coming water cooled reactor technologies. Taking into account the lifetime extension and the progressive launch of generation III, NPPs using water coolant will be in operation through the entire century. They will require a continuous R&D support following a long-term trend driven by the plant life management, safety demonstration, flexibility and economics improvement.
- Fuel technology in present and future nuclear power plants is continuously upgraded to achieve better performances and to optimise the fuel cycle, still keeping the best level of safety. Fuel evolution for generation II and III is and will stay a key stake requiring developments, qualification tests and safety experiments to ensure the economical competitiveness and safety. Experimental tests, exploring the full range of fuel behaviour, contribute to determine fuel stability limits and safety margins, as a major input for the fuel reliability analysis.
- To meet nuclear energy sustainability, in terms of resources and waste management, fast neutron reactors are mandatory and require innovative materials and fuels which resist to high temperatures and/or fast neutron flux in different environments. These environments will be needed for demonstrating the technical, economical and safety performances of these technologies. The selection, optimisation and qualification of these innovative materials and fuels raise critical issues concerning their in-service behaviour; utilisation of high performance Material Testing Reactors and other facilities will be necessary to fix these issues.

These above stakes require a sustainable and secured access to an up-to-date high performance Material Testing Reactor. In addition, such a new research infrastructure will contribute to build up technical skills in the nuclear industry and to train a new generation of research scientists, engineers and, ultimately, executives.

2 JHR construction and associated experimental devices development

The Jules Horowitz Reactor project (JHR) copes with this context. JHR is designed as a user-facility reaching the needs of the international community. This means:

- flexibility with irradiation loops able to reproduce the operation condition of the different power reactor technologies,
- high flux capacity to address both Generations II, III, IV needs.

JHR is designed, build and will be operated as an international user-facility because:

- Given the maturity and globalisation of the industry, domestic tools have no more the required level of economic and technical efficiency. Meanwhile, countries with nuclear energy need an access to high performance irradiation experimental capabilities to support technical skill and guarantee the competitiveness and safety of nuclear energy.
- Many research items related to safety or public policy (waste management, etc.) require international cooperation to share costs and benefits of resulting consensus.

JHR design is optimised for offering high performance material and fuel irradiation capability for the coming decades.
This project is driven and funded by an international consortium gathering vendors, utilities and public stakeholders. This consortium has been set up in March 2007 where the construction began. On site excavation is now completed; the civil work will start in spring 2009 and the start of operation is scheduled for 2014.

At the present time, the members of the consortium are:

- Research European bodies: the European Commission for Europe, SCK/CEN (Belgium), NRI/UJV (Czech Republic), VTT (Finland), CEA (France), CIEMAT (on behalf of several public and industrial Spanish partners)
- International industrial companies: EDF, AREVA; VATTENFALL

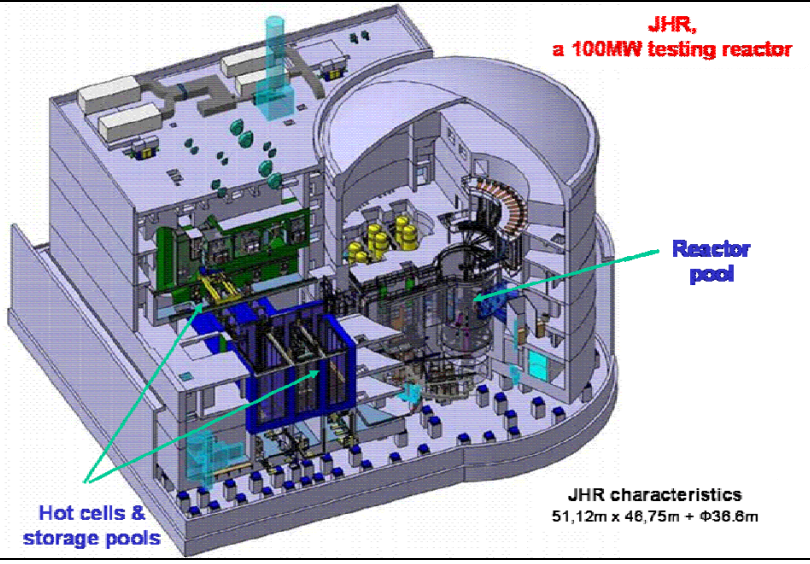
Moreover, two associated partners are committed in the JHR project: JAEA (Japan), DAE (India).
Discussions are on-going with research institutes and utilities to enlarge the JHR consortium.

The JHR experimental capacity

The JHR is a research infrastructure to perform screening, qualification and safety experiments on material and fuel behaviour under irradiation.
JHR is a water cooled reactor to provide the necessary flexibility and accessibility for managing several highly instrumented experiments, reproducing different reactor environments (water, gas or liquid metal loops), generating transient regimes (key for safety).
The JHR facility gathers (cf figure 1):

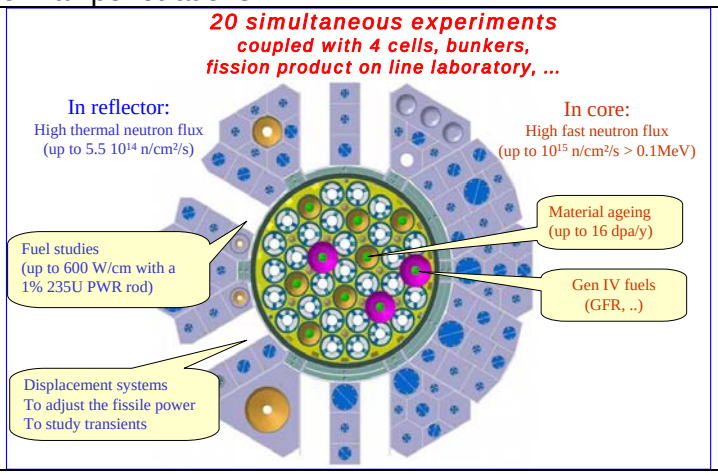
- the reactor building including the core, the cooling system and the experimental bunkers directly connected to the core through pool wall penetrations
- the auxiliary building including the pools and hot cells necessary for the experimental irradiation process.

Fig 1. JHR layout showing the Reactor Building and the Auxiliary Building, crossed together by a water block connecting the core, the poles and the hot cells



JHR core (cf figure 2) is optimised to produce high fast neutrons flux to study structural material ageing and high thermal neutrons flux for fuel experiments.

Fig 2. The JHR core, in his core pool, is a high power density fuel rack in a vessel slightly pressurised and surrounded by a Beryllium reflector. Experiments can be implemented in the centre of fuel elements, in place of fuel elements, in beryllium block or in water channels crossing the reflector. Experiments are connected to dedicated casemates in the reactor building through pool wall penetrations.



The JHR experimental capability is typically ~ 20 simultaneous experiences (in-core and in reflector) providing suited environments relevant for different reactor technologies and high neutron flux:

- Fast neutrons perturbed flux (taking into account a full experimental loading):
  - 10^{15} n/cm²/s >0.1MeV, resp. 5.10^{14} n/cm²/s >1MeV
  - Damage per year: 16 dpa/year (8 times the LWR flux on internal material allowing to accelerate ageing)
- Thermal neutrons flux: 5,5 10^{14} n/cm²/s, 600W/cm on 1% enriched fuel pin (to accelerate fuel BU and to simulate power transients)

The JHR experimental performance relies also on key out-of-core components:

- Loops for power reactors in normal or non-normal conditions

- Effective transient devices for safety studies, a major scientific challenge
- Hot cells for the current operation (preparing the experiment, non destructive exams) and alpha cell for Safety fuel experiments
- On line instrumentation and control (more data, better management, extrapolation capability with modelling)
- On line fission product measurement laboratory for gas and liquids

Last but not least, the JHR performance relies on the JHR experimental devices fleet. The starting of the development phases is related to the maturity of the demand and depends on the complexity of the device to set up[1].

3 Preliminary Safety Analysis of the reactor fuel

In order to comply with the evolution of safety requirements and to guarantee long term operations, the construction safety standards of JHR have been significantly improved compared to MTRs built in the 60s.

- The safety approach of JHR takes into account a systematic assessment (and the implementation of necessary design modification) of external or internal hazards on the nuclear buildings.
- Furthermore, the JHR confinement is designed to face severe accident conditions. The so-called “Borax accident” (hypothetic beyond design reactivity accident with explosion and core melt) is taken into account in the design of the containment and the water bloc.
- In addition, the JHR safety approach addresses irradiation devices as a potential aggressor of the facility. This problematic involves potentially energetic experiments (PWR loops, safety tests) and/or tests with significant radio isotopic content (eg. Tests on minor actinides).

Under the French laws and as part of the licensing procedure leading up to the decree authorising setting-up JHR large-scale nuclear facility, an examination of the Preliminary Safety Analysis Report (PSAR) by the French Nuclear Safety Authority (ASN) has led to the preliminary approval of the reactor fuel for the driven core.

3.1 Background and updated practices

In the past, the authorisation to use a fuel in a French research reactor was obtained on the basis of a qualification irradiation, which involved baking fuels samples under normal reactor conditions. This authorisation was generally dependent on the feedback provided by its use in the reactor or via the irradiation of several fuel elements prior to total core conversion. This phase helped to validate the expected fuel behaviour under operating conditions. However, only normal operating conditions were tested.

For non-normal conditions, numbers of tests were performed on MTR fuels in the 1960s. They involved aluminium-based fuel plates containing highly-enriched uranium. Reactivity insertion accidents (RIA) and their behaviour were especially studied in the USA using the SPERT reactor and in France using the CABRI reactor. These tests have not been updated since this time.

[1] For more information about these developments see the presentation of Daniel Parrat on RRFM 2008

The practice of baking samples of future fuels under normal operating conditions was questioned within the JHR project. A reassessment of past experiments relevance appeared to be necessary to support the safety demonstration for utilising aluminium-based fuels in JHR. Furthermore, French safety standards for research reactors have been upgraded to a level similar to that of power reactors. As early as the first discussions on the JHR safety options in 2003 in preparation of the licensing procedure, ASN and its technical support IRSN[2] asked the CEA to upgrade JHR safety to a level of performance comparable to the EPR project. ASN therefore issued a certain number of fuel recommendations, which are recalled below:

The CEA shall aim to design a driven core fuel that at least rules out cladding failure under Category 1(normal) and 2 (incidental) operating conditions.

The CEA shall specify the functional requirements that will be chosen for the first barrier (cladding) and for the fuel with regard to the different categories of operating conditions, including the related "operating limits", particularly in terms of temperature thresholds (fuel, cladding and primary water) and cladding failure rates applied in accident studies.

The fuel to be used in JHR must be qualified taking into account the operating conditions that are specific to this research reactor.
The operator shall submit its fuel qualification programme to ASN detailing its suitability in relation to the operating limits chosen for the different operating conditions and including accident conditions.
This programme must address the radionuclide release rate in the event of cladding failure.

Prior to examining the impact of these requests on the fuel development and qualification process, it is important to recall the general safety framework required for JHR so as to clearly understand the overall situation.

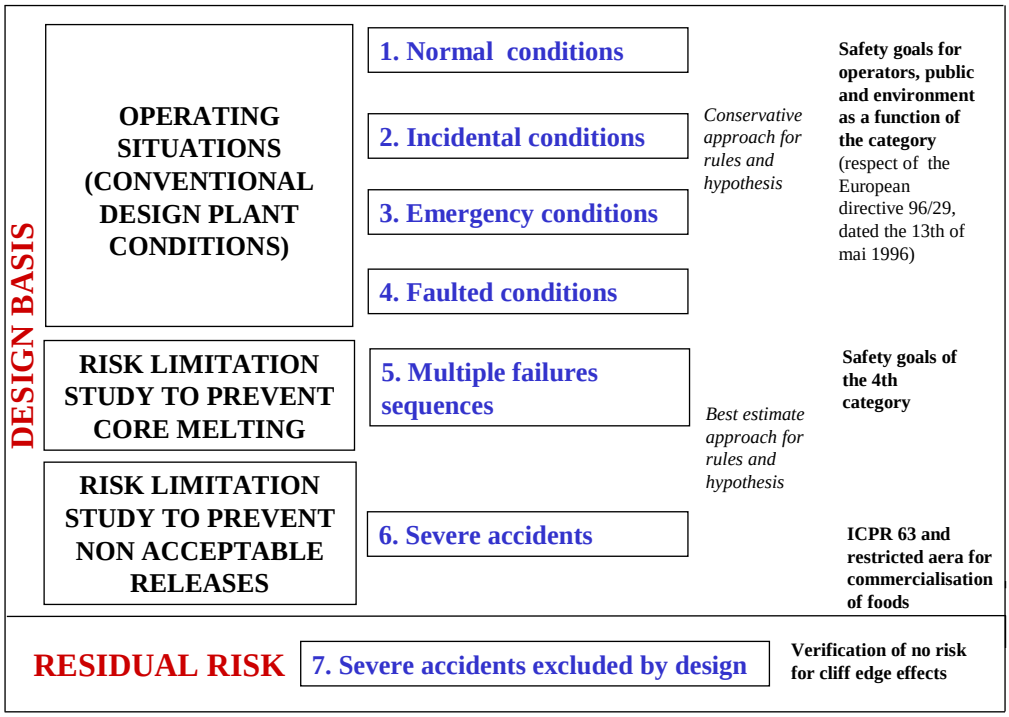
3.2 Overview of JHR PSAR

The safety approach developed for JHR and described in the JHR Preliminary Safety Analysis Report (PSAR) has been derived from power reactor methodology, taking advantage of decades of upgrades from early PWRs up to the EPR. This methodology has been adapted to the JHR experimental facility integrating feedback from the French experimental reactor.

The approach is deterministic and based on the principle of five levels of defence-in-depth. The JHR safety approach aims to show that the measures taken to apply these five levels are sufficient with regard to the General Safety Objectives (GSO) set for this facility. The purpose of the regulatory analysis is to demonstrate:

- Compliance with regulatory texts applicable to the design, implementation and operation of a nuclear facility, e.g. application of the ALARA principle.
- Compliance with the general safety objectives (GSO) defined by the nuclear operator and documented in a Safety Options File. This file was examined in 2003 by the Safety Nuclear Authority. These GSO define general rules for safety studies and make it possible to assess radiological risks expressed in terms of envelope operating conditions and radiological consequences (see below).

[2] Institute for Radiation Protection and Nuclear Safety



- Compliance with internal and external hazards (e.g. fire, explosion, flooding, aircraft crash, earthquake, etc.). It is checked that internal and external hazards do not lead to more severe operating conditions,
- Compliance with specific requirements on radioprotection, waste management, decommissioning, human factors, reliability (redundancy and the single failure criterion, diversity, etc.), physical protection, facility flexibility (core changes), etc.

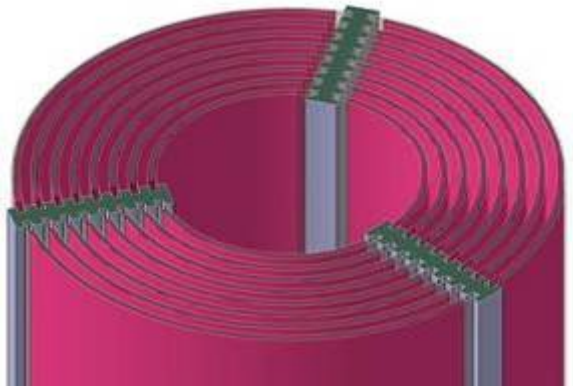
By applying the above-described methodology, the JHR design has the required quality to go through each stage of the licensing process, as any new modern nuclear power plant.

3.3 JHR fuel description

The reference fuel for JHR is UMo fuel (~8 g/cc UMo <20% ^{5}U) as a high-density, low-enriched reprocessable fuel. The CEA is deeply involved in international collaboration aiming to develop UMo. As this fuel is not yet available as an industrial product qualified for JHR operation, the CEA is qualifying a back-up fuel solution (4,8g/cc U_3Si_2, 27% ^{5}U) for the first power operations of JHR. Two fuel solutions are therefore described in the JHR PSAR.

Fuel assembly

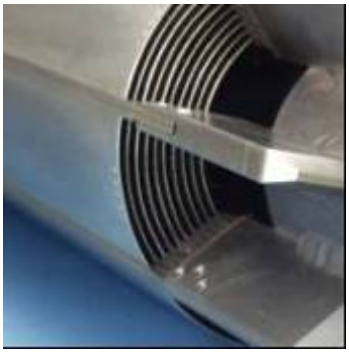
The fuel sub-assembly design was stabilised after the optimisation phase. This sub-assembly consists of an element with 8 concentric crowns; each crown results from the coupling of 3 bent U_3Si_2/Al fuel plates (see Figures 1 & 2) crimped inside 3 aluminium stiffeners. For neutron considerations, small boron (10% mass) /aluminium plates are set in the upper side of the fuel plates.

Fig 1. Fuel sub-assembly	Fig 2. Cross section
	

A manufacturing phase has started with AREVA/CERCA in order to qualify the industrial production of this fuel sub-assembly. Ongoing tasks are aiming to meet the following objectives:

- procure equipment specific to the manufacture of the JHR plates and elements,
- establish manufacturing parameters to meet the manufacturing and inspection specifications for the JHR assembly,

The manufacturing specification was drafted in 2008. Industrial-scale prototypes will be manufactured on the basis of this specification. These prototypes (12 assemblies) will be used in qualification programmes, i.e. the AMMON programme conducted in the EOLE reactor at CEA/CADARACHE, and the EVITA irradiation campaign in the BR2 reactor at SCK/CEN in Belgium.(**see presentations at this conference**)

Fig 3. Fuel assembly	
	

Design and operating limits

Safety options and functional requirements

On the basis of this description, the ASN recommendations in 2003 led the CEA to setting the following functional requirements for the fuel:

Operating category (OC)	Functional requirements
OC1- Normal conditions	Cladding integrity
OC2 - Incident conditions	Cladding integrity
OC3 - Emergency conditions	Several failures possible but no fusion
OC4 - Faulty conditions	Fusion possible though limited

Based on these requirements, the fuel (plates and elements) is designed to guarantee three fundamental reactor safety functions:

- Confinement of radio elements,
- Removal of decay heat,
- Control over reactivity.

The “radioelement confinement” function is directly related to the cladding integrity which acts as a containment barrier.

The “decay heat removal” and the "reactivity control" functions are related to the geometric integrity of the fuel sub-assembly and the stable core configuration, which ensure the hydraulic flow inside the cooling channels and the insertion of absorbers into the core.

Reactivity is controlled by maintaining the free movement of the absorbers, which means that any major changes in the geometry of the fuel sub-assemblies and the core must be avoided.

This analysis resulted in the definition of a set of functional fuel requirements in relation to the different operating categories.

Operating category	Mechanical integrity (leaktightness)	Geometric integrity	No fusion	No risk of explosion
OC1 & OC2	required	required	required	required
OC3			required	Required
OC4				Required

These requirements will need to be transposed into quantitative criteria for the facility safety analysis in order to demonstrate that the general safety objectives (GSO) are being met.

Operating limits, collapse modes and fuel criteria

The following validation criteria for normal operation have been investigated and are described in others papers in this conference (MC. Anselmet et al)
- Cladding mechanical integrity
- Local geometric integrity of plates
- Overall geometric integrity of plates
- No fusion
- No violent exothermal chemical reactions between components

Summary of transient criteria
The table below summarises all the criteria for U_3Si_2 which now correspond to standard practice in the field of experimental reactors, combined with the categories of operating conditions.

Objectives	Cladding mechanical integrity	Local geometric integrity	No cladding or meat fusion	No violent exothermal reaction
OC2	X	X	X	X
OC3			X	X
OC4				X
Criteria U_3Si_2	Mechanical criterion during definition phase	T_{meat} < 515 °C	$T_{cladding}$ < 616 °C T_{meat} < 645 °C	T_{meat}< 915°C

3.4 Experimental programme and fuel tests

U_3Si_2 fuel has been successfully used on an international level for several years now and benefits from a large amount of feedback. Within the JHR context however, this fuel will be subjected to higher loads than usual so CEA has developed a suitable qualification programme for this fuel under normal operating conditions.

The key objective of the analytical experiments is to validate a failure criterion, which must be defined without any reference materials. The method is mostly based on comparing calculations of the main mechanical quantities (e.g. stress or deformation) with threshold values measured on the cladding material.

Integral hydraulic or irradiation experiments will then aim to validate the design by demonstrating the satisfactory behaviour of the fuel sub-assembly.

Hydraulic tests
Hydro-mechanical tests were conducted in the BACCARA loop (CEA) with a full-scale aluminium mock-up in order to characterise the mechanical behaviour of the JHR element in representative flow conditions. The tests provided useful information and measurements on the physical parameters needed to model and consolidate the preliminary design calculations.

Tests showed no degradation in the plates or the element structures due to hydraulic or vibratory phenomena. Moreover, measurements were performed on the mock-up before and after the test campaign. They showed that the changes in the outer diameter were not significant compared with the manufacturing uncertainties and to the variations expected after irradiation.

U_3Si_2 fuel sub-assembly qualification
The correct behaviour of the fuel sub-assemblies under conditions representative of JHR normal operation (in terms of power and cooling rate) will be demonstrated by means of an experimental in-pile programme. For this reason, a dedicated loop (EVITA) has been designed to be implemented in the central channel of the BR2 reactor. The loop is currently under construction and irradiation of the first assembly is expected to start in 2009, with subsequent tests scheduled right up to 2013-2014. **(See paper from SCK-CEN MOL on this conference).**

3.5 Fuel evaluation by the French Nuclear Safety Authority (2006-2008)

The technical support organisation (TSO) IRSN has examined the JHR fuel file between 2006 and 2008 based on:

- The above-mentioned information,
- A review of scientific knowledge on the physicochemical behaviour of the fuel,
- Feedback on the utilisation of this fuel,
- Additional experimental programmes launched to demonstrate the feasibility of transposing such information to the JHR case.

Moreover, ASN intends to make sure there is no risk of fuel degradation by buckling. But available demonstrations for this risk in France are based on AG3 alloy cladding and loads relevant for existing reactors. At the opposite, JHR will make use of AlFeNi cladding with specific dynamics and thermal load conditions.

As a consequence, CEA launched characterisation analyses on AlFeNi in order to provide and justify the criteria defined to ensure cladding mechanical integrity.

JHR will use UMo fuel when this fuel will be industrially available. The JHR PSAR takes into account two different fuel options (UMo and U_3Si_2) based on the neutron studies, the thermohydraulic studies, and the radiological consequences. This JHR conversion to UMO could be carried out on the basis of a demonstration showing that UMo fuel fulfils the same safety criteria than U3Si2.

Conclusions and future prospects

The JHR safety standard for fuels is more comprehensive than historical standards used in French research reactors. The JHR standard is largely based on the framework recommended for power reactors. This standard nevertheless takes into account the specificities of the fuel.

In collaboration with partners such as AREVA-CERCA and the Belgian SCK-CEN research centre, the CEA has been implementing a significant effort for the JHR fuel licensing with the purpose to reach i) fuel high performances, as required by JHR operation, and ii) safety demonstration within stringent up-to-date safety standards.

The CEA is pursuing its efforts to improve its knowledge of:

- The operating limits conditioning the occurrence of cladding failure.
- Radioelement release rates with the presence of cracks or other damage (e.g. spalling, fusion, etc.) taking into account the kinetics governing the expected operating conditions.

Furthermore, CEA is deeply committed in the UMo fuel development within an international collaboration with the objective to demonstrate the same level of safety performance compared to U3Si2 fuel.

THE EAST EUROPEAN RESEARCH REACTOR INITIATIVE

S. TŐZSÉR
Reactor Department, KFKI Atomic Energy Research Institute
P.O.Box 49., 1525 Budapest – Hungary
I. N. GOLDMAN AND I. VIDENOVIC
International Atomic Energy Agency
P.O.Box 100., 1400 Vienna – Austria
H. BOECK
Vienna University of Technology/Atominstitut
Stadionallee 2, 1020 Vienna – Austria

ABSTRACT

In January 2008, representatives of research reactors (RRs) in Austria, Czech Republic, Hungary, Romania, and Poland, and a representative of the IAEA, met in Budapest, Hungary at the initiative of the KFKI Atomic Energy Research Institute to discuss co-operation among East European RRs in line with IAEA efforts to improve RR utilisation through formation of coalitions and networks. The participants agreed that co-operation between the participating reactors would be initiated under the name of East European Research Reactor Initiative (EERRI). The meeting agreed on a final report with action items as initial activities for the initiative.

Subsequently, a Memorandum of Understanding was signed by the institutions which participated in the meeting, as well by the three other research reactor institutions in the Czech Republic, Hungary, and Slovenia that were invited to join after the January 2008 meeting. A second meeting of the EERRI was held in September 2008 at the Atominstitut, Vienna with support of IAEA TC Project RER/4/029.

The paper will describe the formulation and initial implementation of EERRI in its first year of activities. This includes the results of the initial meetings, and subsequent follow-up activities, including the establishment of coordinators for different areas of potential activity, as well as agreement to share and post reactor operating schedules and available irradiation positions so that prospective customers including for isotope production will have more readily accessible knowledge in regard to the schedules of the reactors in this coalition. The future plans for EERRI will also be discussed.

1. Introduction

While research reactors (RRs) have been contributing to the development of nuclear science, power, and medicine for more than a half-century, the operating environment - especially financial, safety, and security requirements - as well as user and public demands, have significantly changed in recent years. Thus, nowadays there is a complex environment that encourages RRs to coordinate their activities and to pursue greater regional and international cooperation. A number of proposals have been made by the international RR community, including ideas already foreseen in a few IAEA studies [1], to form regional coalitions of RRs to collectively improve utilization and to strengthen best practices in management, operations, security, and safety.

The idea of forming a regional coalition among East European Research Reactors was outlined at the IAEA International Conference on Research Reactors: Safe Management and Effective Utilization (Sydney, Australia, 5-9 November 2007). Representatives of several East European Research Reactors started to discuss possible common initiatives that could help RRs in this region to accommodate to the changing world and requirements for nuclear training and irradiation services.

2. East European Research Reactor Initiative (EERRI)

Following the initial ad-hoc discussions mentioned above, an exploratory meeting was held in Budapest, Hungary on January 28-29, 2008 at the invitation of the KFKI Atomic Energy Research Institute (AEKI). Representatives of research reactors in Vienna, Austria; Rez, Czech Republic; Pitesti, Romania; and Swierk, Poland, and the IAEA met at KFKI/AEKI to discuss possibilities for enhanced co-operation among East European Research Reactors [2].

2.1. 1st Meeting – Budapest, January 28-29, 2008

Overview

At the meeting the representatives of each participating RR gave a summary of reactor operation and utilization. On the basis of the presented utilization histories and capabilities, it was noted that the East European RRs have been used for many decades mainly for:

- material testing by neutron physics methods, studying radiation damage, and the properties of irradiated materials,
- irradiation services for radioisotopes, and
- education and training.

It was noted that taking a strategic view, a considerable redeployment of the traditional roles for these RRs can be foreseen within about 10 years. Since Europe has decided to construct powerful new facilities intended to fulfil the needs of the whole continent for the above purposes[1] by 2015-2020, it is obvious that these leading edge facilities soon or later will take over the traditional roles of the present research reactors.

Due to this fact, the participants of the meeting agreed that in the coming 10-year period the existing East European research reactors will either decrease their services and will be shut down smoothly, or they will offer improved services for the user communities, which will also assist in preparing for the use of these new powerful European facilities (such as the Jules Horowitz Reactor, JHR; and the planned European Spallation Source). The participants agreed upon the desirability of the second course, and towards this end the East European research reactors should unite their efforts to maintain and improve their services, and use the synergies of the existing facilities.

Thus, the participants agreed that enhanced co-operation among the participating reactors would be desirable. In line with IAEA efforts to improve RR utilisation through formation of coalitions and networks, it was defined also that the co-operation would be initiated under the name of East European Research Reactor Initiative (EERRI).

In addition to the reactors present at the meeting in Budapest, it was decided to form a comprehensive coalition in the region and thus to also invite three smaller reactor institutions,

[1] European Spallation Source (ESS) for material testing by neutron physics methods; Jules Horovitz Reactor, Cadarache, France for studying radiation damage and the properties of irradiated materials; and High Flux Reactor, Petten, the Netherlands for preparing radioisotopes.

namely Jozef Stefan Institute TRIGA, Ljubljana; Czech Technical University SPARROW-1 (CTU, Prague); and Training Reactor of Budapest Technical University to participate in the future co-operation.

Subsequently, a Memorandum of Understanding was signed by the institutions which participated in the meeting, as well by the three other research reactor institutions in the Czech Republic, Hungary, and Slovenia that were invited to join after the January 2008 meeting.

Areas of Cooperation and Collaboration

The discussion at the Budapest meeting also agreed on the necessity of exchanging operational and other experiences in regard to spent fuel management and return of spent fuel to the country of origin. The participants thus focused on future enhanced cooperation to increase reactor utilization. The following main conclusions were reached:

In **education and training** field, it was noted that the European Nuclear Education Network (ENEN, www.enen-assoc.org) already plays a key role, as well as the Nuclear European Platform for Training and University Organizations (NEPTUNO, http://www.sckcen.be/neptuno/). Low-power research reactors are especially well-suited for many training activities, while such activities are not as easily accommodated in higher power research reactors. The smaller reactors may also serve as preparatory sites to develop advanced measurements and other methods development for larger reactors. On the other hand, the larger reactors are open for helping the education and training services offered by ENEN. However, interesting ideas on new types of training courses and on coordinating various existing courses were discussed, and it was agreed that there are interesting possibilities for coalition activity in this area.

Co-operation in the field of **neutron beam experiments** appeared to be the most promising area to be organised. Though neutron beam experiments are performed at most of the participating reactors, the user system applied at the Budapest Neutron Centre (BNC) was accepted as good practice. The participants agreed to develop similar systems at each reactor and then to merge them if practicable. The participants also discussed the possibility to co-operate in the marketing of such reactor services by using harmonised web-pages describing experimental equipment capabilities, reactor utilisation and also by other means. A related issue is **harmonisation of instrument developments** at the co-operating reactors. The harmonisation process should aim at improving services and also at jointly developing tools to be used as preparatory instruments of future ESS experiments.

Irradiation of materials and fuel/PIE. The parties pointed out that, since irradiation and PIE activities are generally organised as contract-based activities, co-operation in this field seems to be more complicated. Nevertheless, it was decided that information on the available and planned services should be collected for joint marketing purposes. It is desirable to find appropriate research targets for an open co-operation in this field that may be later continued at JHR.

Regarding isotope production, although the topic was raised, the participants did not discuss potential co-operation in this field during the initial meeting, instead they agreed to postpone this issue to the 2nd meeting of EERRI.

2.2. 2nd Meeting – Vienna, September 4-5, 2008

The purpose of the meeting held in at the Atominstitut in Vienna September 4-5, 2008, was to strengthen implementation of the EERRI coalition and to review action items defined at the

1st EERRI meeting. This meeting was organized with the financial support of IAEA Technical Cooperation Project RER/4/029, Enhancement of the Sustainability of Research Reactors and their Safe Operation Through Regional Cooperation, Networking and Coalitions. It was announced that the EERRI Memorandum of Understanding had been signed by all the participants of the January 2008 meeting and by the Budapest University, CTU Prague, and Josef Stefan Institute Ljubljana reactors. Thus, **the EERRI officially exists as a coalition** [3].

Representatives of the three additional reactors noted above participated in the second meeting. Thus, it can be declared that the EERRI coalition represents all together 8 reactors.

In addition to reviewing the action items outlined at the 1st meeting, tasks for the forthcoming period were defined. The tasks were agreed as follows, with coordinators assigned for each activity area:

The EERRI education and training activities were promptly addressed by responding to IAEA needs for training courses for countries considering new research or power reactor projects (nuclear candidate countries) with a near-term need for training in beginning-level to specific advanced topics. EERRI designated a coordinator (a representative of TU Vienna/Atominstitut, Austria), who was asked to summarise the education and training capabilities of EERRI on the one hand, and on the other hand to interact with IAEA to define the specific training activities, programmes and courses needed in the next 12-18 months.

Neutron scattering together with other beam applications (e.g. NR, NAA) was emphasized as a near-term EERRI activity. As the Budapest Neutron Centre (BNC)[2] will organize and host a “Central European Training School on Neutron Scattering” in May 2009, it was recommended that the organizers utilize lecturers from other EERRI members to introduce their relevant existing or planned neutron scattering facilities and capabilities. It was also recommended to invite young researchers from other EERRI institutions to participate in the school. A coordinator (a representative of AEKI, Hungary) was designated and was asked to establish a standard format (template) and to collect technical information on available neutron beam instruments and experiments, as well as relevant contact persons at the other EERRI facilities for disseminating on the EERRI web-site.

Isotope production received emphasized attention at the meeting. It was stressed that the isotope business in several EERRI members are run by organizations that are separated from the RRs and the RRs simply irradiate the targets for the isotope producer and in turn receive nominal income only for supplying neutrons. Economic and business problems with small-scale or occasional isotope production, especially Mo-99 were also highlighted. The responsible coordinator for this issue (a representative of IAE Swierk, Poland) was designated with the task to collect reactor operation schedules and display updated information on availability of irradiation spaces.

Irradiation of fuel and materials/PIE. Although the irradiation and PIE activities are generally organised on a contract-basis at each reactor, it would be desirable to collect information on the available and planned services for joint marketing purposes. Therefore a responsible coordinator was also designated and asked to prepare and circulate a template sheet for data collection on relevant irradiation loops/rigs and programmes in EERRI reactors. The coordinator (representative of NRI, Czech Republic) was also asked to prepare information on in-pile loops and irradiation programmes planned for the Jules Horowitz reactor for posting on EERRI web page in order to promote consideration of possible collaborations between EERRI facilities.

[2] BNC is a consortium, which manages the utilization issues of the Budapest Research Reactor (BRR).

Information and organizational issues were also considered. Establishment of a web-site for EERRI to ensure general information dissemination and a discussion board was decided (public platform for the information templates mentioned above). Regarding activity coordination and monitoring the task implementation between EERRI meetings, a monthly and/or bi-monthly conference call among the members was agreed. The IAEA representative was asked to manage these issues and to host a web-site for the EERRI coalition.

3. Summary of the first year activity of EERRI

Summarising the first year activities of the EERRI coalition, a formalized and regular dialogue among the RR operators can be mentioned as the most important achievement. Beyond the sharing of information, a kind of joint thinking and planning has been started since the first meeting. It is expected that this process will strengthen and mature.

The coalition defined the strategic goals and the fields of cooperation as follows:

- **Strategic goals:** based on the synergy effect of the existing facilities to offer improved services for the user communities which will prepare themselves for the time of using leading-edge European facilities.

- **Fields of cooperation:**
  - Education and training
  - Neutron beam application
  - Isotope production
  - Irradiation of fuel and materials/PIE

Reviewing the activities and the implementation status of the action items, the first year summary of the EERRI and results can be summarised in the following tables (see Tables 1 to 6):

Table 1. Meetings and communications

Meeting Place and date	Meeting subject	Participants countries/persons	Main achievements
Budapest, 28-29 January 2008	Exploratory meeting	6/13	Establishment of EERRI Define strategic goals and cooperation fields Formulate Memorandum of understanding Designate contact person Outline work plan with 7 action items
Vienna, 4-5 September 2008	Plenary meeting supported by IAEA TC RER/4/029	7/14	Official announcement of the existence of EERRI coalition Monitor implementation of action items Designate coordinators for each cooperation field Outline work plan with 9 action items

Table 2. Represented countries and organizations/reactors with the designated field coordinators

Country	Organization	Reactor	Person	Specific EERRI duties
Austria	TU Vienna	TRIGA 250 kW	Mr. H. Boeck/AUT	Coordinator on training activities
Czech Republic	NRI Rez	LWR-15 10 MW	Mr. V. Broz/CZE	Coordinator fuel and material irradiation and PIE
	CTU Prague	VR-1 (1 kW)	Mr. L. Sklenka/CZE	–
Hungary	AEKI	BRR 10 MW	Mr. S. Tozser/HUN	Primary contact person for general EERRI matters
			Ms. R. Baranyai/HUN	Coordinator on neutron scattering and beam application
	TU Budapest	100 kW	Mr. A. Aszodi	–
Poland	IAE Swierk	MARIA 30 MW	Mr. G. Krzysztoszek	–
			Mr. J. Jaroszewicz Mr. J. Milczarek/POL	Coordinator on isotope production
Romania	ICN Pitesti	TRIGA 14 MW	Mr. M. Ciocanescu/ROU	–
Slovenia	JSI Ljubjana	TRIGA 250 kW	Mr. M. Ravnik/SLO	–
Austria	IAEA	–	Mr. I. Goldman/IAEA	Organizational issues

Table 3. Action items and their implementation status

Action Item	Description (task)	Responsible	Status
Action items defined in the Budapest meeting (28-29 January 2008)			
A1/1	Inform and invite Ljubljana, Prague and Budapest TU reactors	Mr. S. Tozser/HUN	Done
A1/2	Collect ideas on new and coordinated training courses	Mr. H. Boeck/AUT	Done
A1/3	Provide information on the BNC system for other reactor managers	Ms. R. Baranyai/HUN	Done
A1/4	Organize a workshop on the harmonization of web-pages and user systems	Ms. R. Baranyai/HUN	Partly done and transferred to A2/1
A1/5	Collect information of planned instrument development and to present it on the next meeting	Mr. J. Jaroszewicz/POL	Done
A1/6	Collect information on available and planned irradiation and PIE services and to present it at the next meeting	Mr. J. Jaroszewicz/POL Mr. M. Ciocanescu/ROU	Done
A1/7	Provide information on the JHR meeting in March for the other participants	Mr. H. Boeck/AUT	Transferred to A2/8
Action items defined in the Vienna meeting (4-5 September 2008)			
A2/1	Establishment of a web site for EERRI, for general information, documents, and a discussion board	Mr. I. Goldman/IAEA	Done
A2/2	Hold a monthly or bi-monthly conference phone call of EERRI members to coordinate activities between major meetings and to review action items and implementation plans	Mr. S. Tozser/HUN Mr. I. Goldman/IAEA	Continues (4 conference calls were made by the end of January 2009)

Table 3. (cont.)

Action Item	Description (task)	Responsible	Status
A2/3	Define the specific training activates and courses needed in the next 12-18 months	Mr. H. Boeck/AUT	Done see Table 4
A2/4	Organization issues of Central European Training School on Neutron Scattering (Budapest, 11-14 May 2009)	Ms. R. Baranyai/HUN	In progress (final programme settled)
A2/5	Establishing a standard format and collect technical information on available neutron beam instruments and experiments and relevant contact persons	Ms. R. Baranyai/HUN	In progress
A2/6	Collect information on reactor operating and shutdown schedules	Mr. J. Jaroszewicz/POL	Done for first part of 2009 (see Table 5)
A2/7	Elaboration of a proposal to enhance cooperative mechanisms in order to strengthen coordinated isotope supply from EERRI organizations	Mr. J. Jaroszewicz/POL	Ongoing
A2/8	Prepare a template sheet for data collection on relevant irradiation loops/rigs and programmes	Mr. V. Broz/CZE	Done
A2/9	Preparing and submitting a paper for an EERRI presentation at the RRFM 2009 meeting	Mr. S. Tozser/HUN	Done

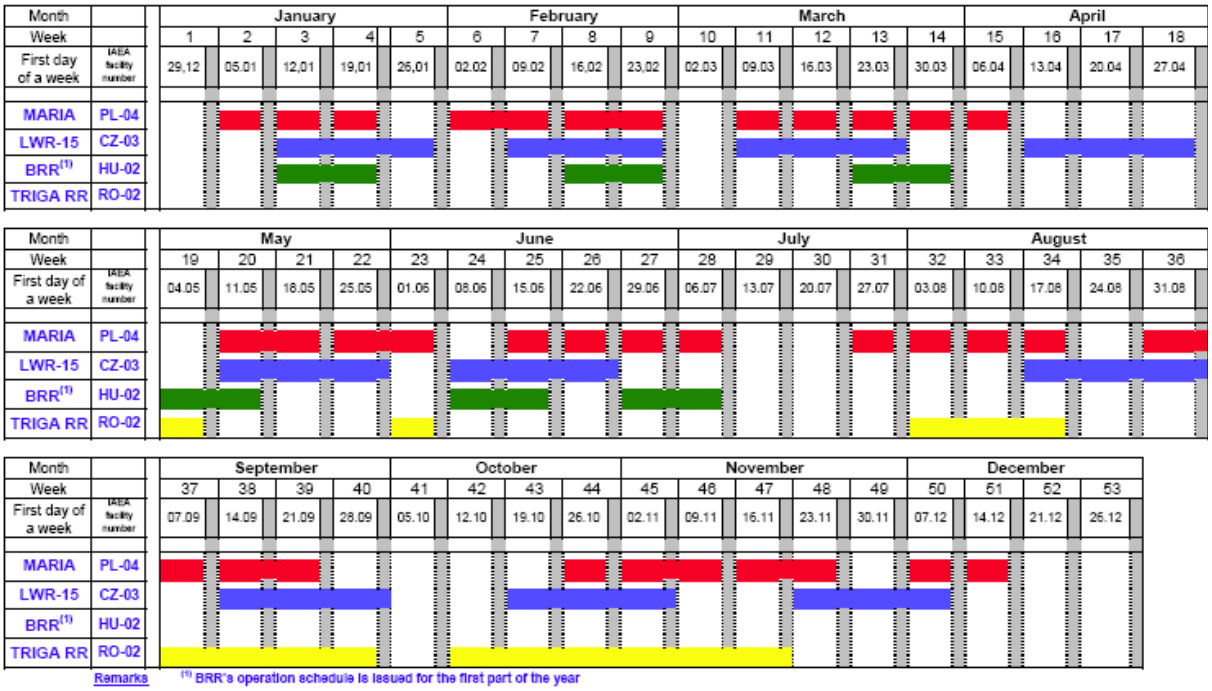
Table 4. Research Reactors of EERRI coalition (activity summary)

Country	AUSTRIA	CZECH REPUBLIC		HUNGARY		POLAND	ROMANIA	SLOVENIA
Reactor data								
Reactor	Triga II Vienna	LVR-15 REZ	VR-1 VRABEC	BRR	BME E&T Reactor	MARIA	Triga II Pitseti	Tigra-Mark II Ljubljana
Type	Triga Mark II	Tank WWR	Pool	Tank WWR	Pool	Pool	Triga Dual Core	Triga Mark II
Power	250 kW	10 MW	1/5 kW	10 MW	100 kW	30 MW	14 MW	250 kW
Education and Training abilities (maintained by Mr. H. Boeck/AUT)								
Disciplines	*The topic is discussed in a separate paper on RRFM2009 presented by Mr. H. Boeck/AUT.*							
Main Isotope Production (maintained by Mr. J. Jaroszewicz/POL)								
I-131	-	-	-	YES	-	YES	YES	-
Mo-99	-	-	-	-	-	-	YES	-
Ir-192	-	YES	-	-	-	YES	YES	-
Sm-153	-	YES	-	YES	-	YES	-	-
Yb-169	-	-	-	-	-	YES	-	-
Y-90	-	YES	-	YES	-	YES	-	-
Lu-177m	-	YES	-	YES	-	YES	-	-
Re-188	-	-	-	-	-	YES	-	-
Co-60	-	-	-	-	-	YES	YES	-
Au-198	-	-	-	-	-	YES	-	-
Σ A/year	-	192 TBq	-	56 TBq	-	380 TBq	41 TBq	-
Other vertical channel applications (maintained by Mr. J. Jaroszewicz/POL)								
Biological	-	YES	-	-	-	YES	YES	-
NAA	YES	YES	-	YES	YES	-	YES	YES
Other(1)	E&T	NTD	E&T	E&T, CI	E&T	NTD, CI		E&T
Materials/fuel test experiments (maintained by Mr. V. Broz/CZE)								
Loops/Rigs	*See Table 6.*							
Neutron Scattering and Beam Port Applications (maintained by Ms. R. Baranyai/HUN)								
Horizontal Facilities	*In progress*							

Remark (1): SE: Education&Training (student experiences), NTD: Neutron transmutation doping of silicon, CI: commercial irradiations.

In the education and training field, it should be mentioned that upon the request of IAEA the first course (to serve as a pilot or model course) will be delivered by the coalition with the involvement of four of EERRI's institutions and the active participation of the Agency, as well. The course is organised under the Group Fellowship Training Programme on Research Reactors (GFTPRR) for developing human resources. The first GFTPRR course is planned for a 6-week period in May-June 2009 and it is offered to participants from MS who have expressed interest in this subject to the IAEA. A separate presentation will be held on this subject in the RRFM 2009 [4].

Table 5. Operation Schedule of EERRI RR operation in 2009



As can be seen in the tables, considerable progress was achieved in the first year of the EERRI coalition. The EERRI in its first year of existence outlined the most important functional mechanisms (annual plenary meetings and monthly or bi-monthly conference calls), defined the fields of cooperation and shaped the forms of organisation and communication. Responsible persons were designated on each cooperation field who are expected to maintain the specific cooperation issues (collect, standardise and share information) within the coalition. Due to their efforts, some standard formats for listing the human and facility resources of the coalition have already been completed and are available in the web-site hosted by IAEA.

The concerned EERRI coalition web-sites are:
http://tc.iaea.org/tcweb/regionalsites/europe/news/newsstory/default.asp?newsid=356
http://www.iaea.org/OurWork/ST/NE/NEFW/rrg_EERRI.html

Regarding the IAEA contribution, although the IAEA representative participated on the first meeting as an observer "only", from the time of the second meeting the IAEA role became an integral part of the coalition. The IAEA not only ensures limited financial support at this time, but the IAEA hosts and maintains the EERRI web site and ensures significant coordination to manage the activity of the coalition. The technical conditions for conference calls were organised and supported by IAEA, and as well the coalition enjoys the support of the Agency.

Table 6. Materials/fuel test experiments (loop/rigs) operated in RRs of EERRI Coalition

Reactor (Country)	LVR 15 (CZECH REPUBLIC)						MARIA (POLAND)
Facility	*BWR 1*	*BWR 2*	*RVS 3*	RVS 4	CHOUCA	FLAT IRRADIATION RIG	NTD
Type	in-pile loop	in-pile loop	in-pile loop	in-pile loop	irradiation rig	irradiation rig	
Purpose	material behaviour and radioactivity transport under BWR conditions		material behaviour and radioactivity transport under PWR/VVER conditions		neutron irradiation of constructional materials used for reactor vessel construction		silicon doping
Parameters							
medium	water	water	water	water	He / N / Ar	He / N / Ar	
pressure	11 MPa	12 MPa	16.5 MPa	15.7 MPa	100 kPa	100 kPa	
temperature	310 °C	310°C	345°C	311-322°C	300 °C	300 °C	
volume	62 l	510 l	210 l	10 l	30 l	30 l	
flow	3000 kg/hr	3000 kg/hr	10000 kg/hr	2000 kg/hr			
heat flux / heating capacity	45 kW		100 kW	60 W/cm^2, heated length 560 mm	6 x 2 kW	8x800 W / 6x400 W	
Neutron flux:	~1x10^{18} n/m^{2}s	~1x10^{18} n/m^{2}s	~1x10^{18} n/m^{2}s	~1x10^{18} n/m^{2}s	~1x10^{18} n/m^{2}s	~1x10^{18} n/m^{2}s	
Specimen space		specimen strained			Æ 56 x 400 mm	50x120x500 mm, 20x60x260 mm	5 and 6 inches
Services	• Investigation of materials mechanical properties degradation and corrosion behaviour under irradiation and BWR water chemistry conditions • Investigation of radioactivity transport and behaviour under BWR conditions (eg. hydrogen water chemistry, zinc injection, etc.) • Testing of high-temperature, high pressure sensors for water chemistry monitoring		• Investigation of structural materials mechanical properties degradation and corrosion behaviour under irradiation and PWR/VVER water chemistry and thermal-hydraulic conditions • Investigation of behaviour (corrosion, hydriding) of fuel cladding materials under influence of irradiation, thermal flux and water chemistry conditions		• Tensile specimen, CT specimen, round Cts, up to 40 Charpy-V specimens	• Charpy-V specimens	

Table 6. (Cont.)

Reactor (Country)	BRR (HUNGARY)		Triga II Pitesti – SS Core (ROMANIA)				
Facility	*BAGIRA 1*	*BAGIRA 2*	*Loop A*	*C1&C2*	*C5*	*C6*	*C9*
Type	in-pile irradiation rig	in-pile irradiation rig	loop	In pile capsules	In pile capsule	In pile capsule	In pile capsule
Purpose	neutron irradiation of constructional materials used for reactor vessel construction	neutron irradiation of constructional materials used for reactor vessel construction	Irradiation tests of fuel elements and structural materials used in PHW reactors	Irradiation tests of fuel elements Two independent capsules for parametric testing	Structural materials irradiation tests in inactive environment	CANDU type fuel element tests in fast transient regimes in TRIGA ACPR reactor	Cycling tests on fuel elements
Parameters							
medium	He/Nitrogen		Demineralised water		Helium	Demineralised water	
pressure	300 kPa	300 kPa	13.5 MPa	12 MPa	0.6 MPa	0.4 MPa	10.7 MPa
temperature	150-500 °C	70-150 °C	3100 °C	3300 °C on fuel clad	2900 °C	500 °C	3250 °C on fuel clad
volume	5 l	5 l	252 l	30 l - convection		7.5 l	3 l
flow	-	-	3-7 m^3/h	4 m^3/h	-	Stagnant water	0 – 4 m^3/h
heat flux / heating capacity	80 W	-	100 kW	30 kW	10 kW	20.000 MW peak power pulse	21.5 kW
Neutron flux:	$4x10^9$ n/m^2s	$3x10^9$ n/m^2s	~3.2 $x10^{18}$ n/m^2s	~$2x10^{18}$ n/m^2s	~10^{17} n/m^2s	~$2x10^{18}$ n/m^2s	~$2x10^{18}$ n/m^2s
Specimen space	20x30x300	20x20x300					
Services	• Charpy-V specimens, tensile specimens, CT specimens	• Charpy-V specimens, tensile specimens, CT specimens	• Overpower type tests on fuel element • Power ramp type tests on fuel element • Corrosion and mechanical behavior studies on structural materials used in CANDU pressure tubes • LOCA type tests	• Fuel element dimensional measurement • Fission products pressure – on line • Power ramp • Short-time irradiation for residual deformation of the cladding determination	• Structural materials irradiation tests in inactive environment: Zircalloy-4, steel 403-M, Zr-2,5%Nb until 2,3X1022 nvt • Irradiation and tensile test of Chorpy standard minisamples – maximum 30 samples per irradiation campaign	• Thermomecanical behavior of CANDU type fuel element in fast power transients	• Cycling tests on fuel elements that should confirm the fuel capacity to support a wide range of power cycling that occurs in normal operation of a CANDU reactor during power load following.

In a recent development, it should be mentioned that the Vinca Institute (Serbia) expressed its willingness to participate in the coalition work. It was agreed that they could participate as an observer beginning with the first teleconference in 2009.

Regarding the future, it was decided that the 3rd plenary meeting will be held in Vienna at IAEA on 26-27 March after the RRFM meeting. Regarding achievements of the first year, the most attractive developments to date involve the education and training field, with plans to organise more specific and subject oriented courses based on the synergy of coalition represented by all participants.

Some progress may be expected in the isotope production area, as well as neutron scattering and beam application its first year the EERRI coalition has already begun to establish international visibility it is the intention of the coalition to gain broader awareness and international acceptance as well.

4. Conclusion

Although one-year is too short to make a real balance on the results and benefits of EERRI, it is long enough to draw some conclusions about the ongoing activities. It is obvious that the EERRI coalition has made considerable progress during its first year. On the basis of the one-year experience the following conclusions can be made in regard to “lessons learned” for others considering the formation of similar arrangements:

- The strategic goal(s) of the coalition as well as the cooperation fields should be clearly specified as early as possible. It is not feasible to start with full-scale cooperation with a large number of parties because there is too large a spectrum to manage, and it is better to start with a few items only (to have positive feedback and cooperation successes as soon as possible).
- The functional issues (how the coalition is working) have to be defined in the very beginning. In this field, information-sharing and regular communication (e.g. monthly or bi-monthly calls) should be mentioned as key factors to keep the coalition active. A web-site as a general information base and discussion forum can be recommended.
- As a working method, itemised task distributions with defined deadlines and responsibilities seem a manageable a questioning method.
- Personal responsibility and motivation is a key factor. Therefore it is important to designate one (and only one) responsible person for each cooperation field.
- The operation of the coalition has to be based on a questioning attitude. Progress must be monitored on a regular basis (e.g. quarterly).
- The IAEA plays an outstanding facilitator role. Beyond its financial support the Agency ensures regular management assistance for “operating” the coalition.

5. References

[1] Ira N. Goldman, Pablo Adelfang and Shriniwas K. Paranjpe, Kevin Alldred and Nigel Mote: Research Reactor Coalitions - First Year Progress Report. RRFM 2008 Conference, 2-5 March 2008, Hamburg, Germany.

[2] Meeting Report of the East European Reactor Initiative. Budapest, 28-29 January, 2008.

[3] Meeting Report of the East European Reactor Initiative (EERRI). Atominstitute/Technical University of Vienna. 4-5 September 2008.

[4] H. Böck, M. Villa, E. Bradley, A. Soares, P. Adelfang, I. Goldman, S. Tozser, A. Aszodi, M. Ravnik: A Multi-National Practical Training Course for Nuclear candidate Countries organized by EERRI. Paper submitted to the 13th RRFM 2009 conference. RRFM 2009, 22 -25 March 2009, Vienna, Austria.

STATUS AND PERSPECTIVES OF FUEL DEVELOPMENTS FOR FAST NEUTRON REACTORS OF 4TH GENERATION

C. RENAULT, J. ROUAULT, P. ANZIEU
CEA/Saclay, DEN/DDIN, 91191 Gif s/ Yvette, France

ABSTRACT

The R&D strategy in France on future reactors gives first priority to developing a new generation of fast neutron nuclear systems and recycling technologies so as to assure a sustainable and environment friendly electricity production in the second half of the 21st century. SFR is the reference option, not only in France but also in Europe. The European strategy considers both the GFR and LFR as alternatives to the SFR.

A common concern is to achieve a convincing demonstration of the capability of fuels to attain the ambitious goals set to 4th generation fast neutron systems, especially in terms of performance (uranium conversion, minimization of long life radioactive wastes) and safety.

Owing to the important and satisfactory feedback experience built upon oxide fuels, MOX is the reference fuel for the SFR, at least for the start-up of the prototype (ASTRID). The objectives followed for the 4th generation SFR for safety (for example sodium void worth reduction and limited core reactivity excess) and cycle performances (self-sustainable core with a near zero breeding gain, reasonable in-core Pu inventory, MA transmutation) are achievable with an oxide fuel in large power cores (3600 MWt) while implementing adequate design features.

Nevertheless, recent calculations show that the use of a dense and cold ceramic fuel might even improve the core performances. Carbide and nitride are candidate fuels to be seriously investigated for SFRs of 4th generation. For the GFR and the LFR, dense fuels are required to achieve self-generation because of the higher fraction of coolant in the core. Carbide and nitride are currently the reference fuels for the GFR and LFR, respectively.

Focused on some key design parameters (such as high breeding capability, safety, expected performances of the fuel cycle based on pyro-metallurgical processes), several countries (India, China, Korea, Japan, USA) are considering the metal fuel for the SFR either as a long term reference or as a challenger to oxide fuel. In such a context, the merits and drawbacks of the metal fuel option for large SFR cores must be re-assessed, and its performances compared with that of oxide and carbide/nitride fuels.

This paper summarizes the current status of fuel development and perspectives. Basic features of oxide, metal and other fast reactor fuels (carbide and nitride) are compared from the viewpoints of fuel cycle (fresh fuel fabrication and spent fuel treatment), in-pile behaviour, core performances, and safety. The paper also briefly reviews the potential offered by innovative structural materials developed for high temperature resistance (SiC, refractory metals) for the GFR, or low swelling behaviour under irradiation (ODS,…) for the SFR.

The role of experimental reactors is underlined for further assessment of the in-pile behaviour of fuels with representative materials and realistic conditions (burn-up, MA content, neutron flux…). An optimal use of existing irradiation reactors (Phenix, Joyo, Monju, BOR-60, BN-600) is necessary until new reactors, under construction (JHR, CEFR, PFBR) or planned (ALLEGRO, ASTRID) can be put in operation. The paper pleads for the implementation of multilateral collaboration at the European and broader international levels for a continuous capability of innovative fuel qualification.

1. Introduction: a renewed context for fast neutron reactor development

The safe operation of current power plants over the past 20 years, the increasing economic competitiveness of nuclear energy as fossil fuel prices escalate, as well as considerations of energy security pave the way for an active development of nuclear energy in Asia and a renaissance in the United States and Europe. This leads to anticipate an installed capacity of nuclear power of the order of 1000 to 1500 GWe by 2050, which is about four times the current installed capacity (370 GWe). Such a nuclear power capacity would require about 15 Mtons of natural uranium, if realized only with light water reactors which use less than 1% of the uranium (^{235}U mainly) over a lifetime of 60 years. This amount, which is comparable to the estimated assured plus speculative reserves at a price below 130 $/kg, incites to prepare the deployment by 2040 of fast neutron reactors with a closed fuel cycle that can burn more than 80% of natural uranium. Even if the situation around the middle of the century would not lead to a shortage of uranium because of additional reserves in phosphates or sea water, the rising cost of this resource, together with the accumulation of spent fuel, would drive the need to switch to fast neutron reactors to achieve a more efficient use of uranium and minimize the ultimate long lived radioactive waste [1].

The paper summarizes the current status of FNR fuel development and perspectives. After recalling the strategy for fast neutron reactor development in the world (section 2), the reference concepts of SFR and GFR proposed in France are briefly described (section 3). Then, reopening the scope, the basic features (mainly core performance and in-pile behaviour) of oxide, metal and other fast reactor fuels (carbide and nitride) are discussed, showing their potential and challenges (section 4). Section 5 addresses the fuel qualification procedure and underlines the role of experimental reactors for realistic assessment of the in-pile behaviour of fuels in the long term.

2. The strategy for fast neutron reactor development

2.1 Past experience on fast neutron reactors and trends for the short term

In parallel to similar efforts made in the United States, Russia and Japan, European laboratories and industries supported an active development of Sodium cooled Fast Reactors (SFR) from the 1960s to 1998. No less than seven experimental and prototype reactors were built and operated over this period: Rapsodie, Phenix and Superphenix in France, DFR and PFR in United Kingdom, and KNK-II and SNR-300 (which was never put in service) in Germany. However, the industrial development of SFRs stopped in Europe when the political decision was taken in February 1998 to abandon Superphenix. It had stopped earlier in the United States with the Non Proliferation Act promulgated in 1978. Russia proceeded with the development of SFRs in spite of budget constraints and is expected to put BN-800 (800 MWe) in service in 2012. Japan's efforts since 1995 are mainly devoted to putting Monju back into service. India and China, which both plan on nuclear power to supply part of the energy needed for their fast economic growth, have both aggressive agendas to develop light water reactors and SFRs with respective plans to start, respectively, a prototype fast reactor PFBR (500 MWe) and an experimental reactor CEFR (65 MWt) in 2010.

2.2 The strategy in France and in Europe

Prospective studies carried out by the CEA and industrial partners led to elaborate for France a R&D strategy on future nuclear energy systems for the medium and the longer terms (> 2040). This strategy, approved by the French Government in March 2005, gives clear priority on fast neutron nuclear systems with a closed fuel cycle, the Sodium-cooled Fast Reactor (SFR) and the Gas-cooled Fast Reactor (GFR), owing to the general recognition of their capability to meet sustainability goals. This has been confirmed

December 2006 and May 2008. The strategy sets the objective to build and start in 2020 a prototype of 4th generation reactor intended to proceed with demonstrations of sustainable fuel cycle (fuel fabrication and reprocessing).

In Europe, stakeholders acknowledge the need to not subordinate the development of sustainable energy to a single reactor technology and therefore to participate in the development of at least another type of fast reactor.

With this aim in view, European stakeholders prioritised the six systems considered in the Generation IV International Forum and identified three fast spectrum systems that were the most likely to meet Europe's energy needs in the long term in terms of security of supply, safety, sustainability and economic competitiveness:

- The Sodium Fast Reactor (SFR) as a first track aligned with prior experience of Europe, and
- An alternative fast neutron reactor technology to be determined between the Lead cooled Fast Reactor (LFR) and the Gas cooled Fast Reactor (GFR).

Technology breakthroughs and innovations must be achieved for all reactor types. Innovative design and technology features are needed to achieve safety and security standards anticipated at the time of their deployment, to minimize waste and enhance non-proliferation through advanced fuel cycles, as well as to improve economic competitiveness especially with a high availability factor. In particular, structural materials and innovative fuels must be developed to sustain high fast neutron fluxes and high temperatures, as well as to comply with innovative reactor coolants.

This research needs to be supported by the development of advanced fuel cycle technologies to possibly recycle minor actinides in fast reactors and afford progress on long term burden of radioactive waste to be ultimately disposed. Main milestones include selecting around 2012 technologies with greatest industrial perspectives and construction over the period 2012-2017 of an advanced MOX fuel manufacturing workshop to fuel the prototype of SFR, as well as a minor actinide bearing fuel production facility for advanced recycling demonstrations in this prototype and other fast reactors abroad. This will be followed by further developments of mature industrial designs for the most promising recycling processes.

The development of above fast spectrum experimental and prototype facilities will not only require materials testing reactors and hot cells, but also testing and qualification facilities for systems technologies and components (specific liquid metal loops, gas loops and hot cells), as well as code qualification and validation which are mandatory for safety analyses.

3. Innovative tracks for fast neutron systems and fuels: reference concepts

3.1 A common set of requirements (SFR, LFR, GFR)

Considering the general criteria assigned to innovative fast neutron systems in Generation IV, the requirements set to nuclear fuels are basically identical for SFR, GFR and LFR [2]:

- Competitiveness: core compactness (core power density), high fuel burn-up (> 100 GWj/tHM), optimization of fuel recycling techniques and specific fabrication process for fuels bearing minor actinides, efficiency of the techniques for fuel handling and in-service inspection;
- Safety: intrinsic core neutronic performance (internal and global breeding gains, reactivity coefficients), fuel element robustness (mechanical integrity at high temperature, fuel-clad

interaction, close retention of fission products), improved prevention and control of severe accidents involving core damage;
- Resource utilization: zero breeding gain without blankets, and positive breeding gain with radial blankets (designed for adequate resistance to proliferation), minimum fissile material inventory for deployment;
- Minimisation of highly radioactive and long-lived nuclear wastes: potential for minor actinides recycling, efficiency of transmutation in fast neutron reactors;
- Resistance to proliferation: integral recycling of fuel, without separation, in the homogeneous mode (co-management of all actinides together) or in the heterogeneous mode (recycling of minor actinides in proliferation resistant blankets).

Despite important similarities, the differentiated character of sodium, lead and gas coolants (physical, chemical and neutronic properties) induces significant deviations in the design of cores and fuels for SFR, LFR and GFR. Particularly, the choice of helium for the GFR opens the door for high temperature applications (up to 850℃ at the core outlet) and drives enhanced requirements on fuel robustness to assure its integrity in accidental conditions. As a result, a higher degree of innovation is required and the deployment is for longer term (2050).

3.2 Reference concept for the SFR (fuel, assembly, core)

A new generation of SFR

The Sodium cooled Fast Reactor (SFR) is the reference technology for fast reactors. It may be considered for industrial deployment around 2040 since Europe, in cooperation with Japan, Russia and the United States, has acquired important expertise in this reactor type. However, innovations are needed for a Generation IV sodium cooled fast reactor to compete with Generation III LWRs in economics and safety. This will require systems' simplification to reduce investment cost, enhanced safety with improved prevention and management of severe accidents, improved operability (fuel handling, maintenance and repair) to achieve high capacity factors, and advanced closed fuel cycles with multiple recycling of actinides offering appropriate resistance to proliferation and optimized waste forms.

Given the maturity of the technology, the prototype reactor planned in France for 2020 (ASTRID) will be in the range of 300 to 600 MWe to demonstrate the innovations selected in 2012 to upgrade sodium cooled fast reactor performance and to open the way to a "first of a kind" commercial reactor.

High performance SFR core with enhanced safety characteristics

The optimized design of the SFR core combines a number of criteria potentially hard to reconcile: prevention and control of accidents (favourable reactivity coefficients), limited reactivity excess (core with internal breeding gain close to zero), and eventual incorporation of minor actinides.

The optimization of core design leads to consider, besides MOX fuel, dense and high thermal conductivity fuels such as carbide in place of oxide fuels, as well as high performance cladding materials as ferritic-martensitic, oxide-dispersed strengthened steels (ODS).

Preliminary studies resulted in the definition of a 3600 MWt (~1500 MWe) reference core with improved characteristics compared with those of the European Fast Reactor project (terminated in 1998). The core has a low reactivity loss (self-sustainable core without fertile blankets) and improved safety parameters (voiding effect less than 5 $). This result was obtained by reducing the in-core sodium fraction and maximizing the fuel fraction. As a

consequence, the inter-pin space is minimized which implies to make use of a cladding material without swelling under irradiation.

Furthermore, the use of SFR reactor to recycle minor actinides (either in heterogeneous or homogeneous mode) has to be considered from the very beginning of the core design and assessment of potential MA impact on the SFR safety related parameters.

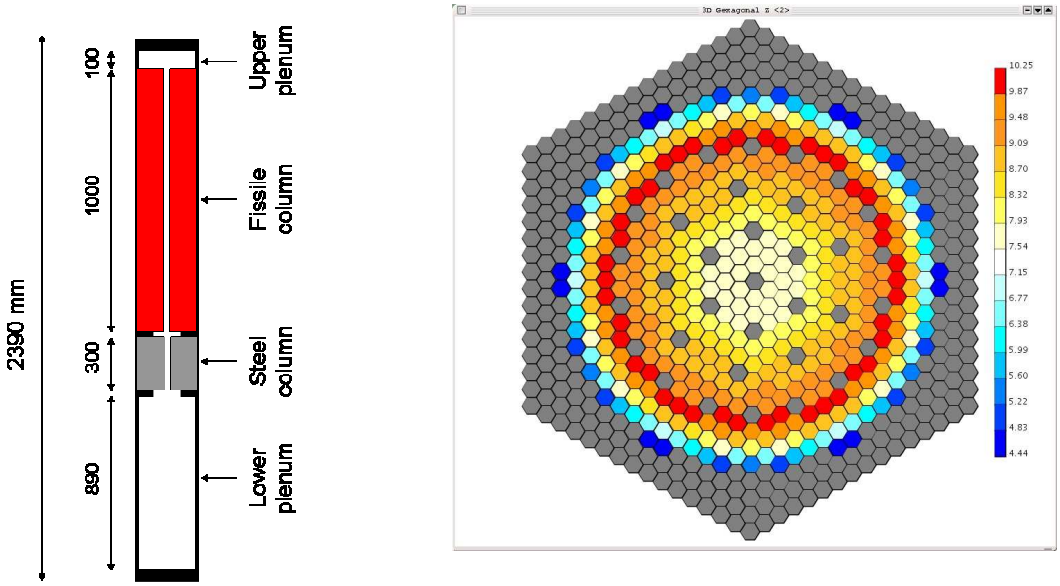


Figure 1: SFR fuel element and reference core design 3600 MWt (1500 MWe)

The selection of a reference SFR fuel

As far as fuel is concerned, two main options are investigated in France for the design of large SFR cores.

- Because of the rather short timescale for the prototype, and the important and satisfactory feedback experience built upon oxide fuels, MOX is considered as the reference fuel at least for the prototype start-up core. Oxide fuel is also a very serious fuel candidate for the longer term commercial SFR. Preliminary calculations are performed to assess the potential merit of a design options such as a Sub-Assembly (SA) with a tight lattice of large diameter pins, a sodium plenum at the top of the core, the use of moderator material to further optimize the efficiency of the Doppler reactivity effect.
- More challenging options (for the long term SFR) also are investigated: core concepts based on the use of denser and colder ceramic fuels (carbide is currently preferred to nitride) so as to assess the potential benefit of (1) reduced core volume (with therefore a higher power density), (2) higher safety margin between the fuel operating temperature and melting point and (3) better compatibility between coolant and fuel material in accidental conditions.

Focused on some other key design parameters (such as high breeding capability, ‘intrinsic’ safety, expected performances of the fuel cycle based on pyrometallurgical processes, non-proliferation issues), several countries (like India, China, Korea, Japan, USA) are considering the metal fuel for the SFR either as a long term reference or as a challenger to oxide fuel. In such a context, the merits and drawbacks of the metal fuel option for large SFR cores are re-assessed in France, and its performances are compared with that of oxide and carbide fuels.

3.3 Reference concept and alternative options for the GFR

GFR potential and deployment scenario

The helium cooled fast reactor is an innovative nuclear system with such attractive features as a chemically inert and optically transparent coolant, as well as a quasi-decoupling of the reactor physics from the state of the coolant. Other advantages of the GFR relate to its potential to operate at high temperature (at least 850℃), which enables in principle the production of hydrogen or synthetic hydrocarbon fuels in a sustainable manner. On the downside, since gas is a poorer coolant than liquid metals, key aspects demonstrating the viability of the GFR include development of a refractory and dense fuel, and robust management of accidental transients, especially cooling accidents.

A status on the GFR pre-viability has been made at the end of 2007, ending the pre-conceptual design phase. A reference set of design options has been proposed for a 2400 MWt GFR [3,4].

The feasibility of the GFR is essentially linked to two demonstrations: the mastery (fabrication, thermo-mechanical behaviour) of a high fissile content refractory fuel, and the implementation of appropriate safety systems for the prevention and a robust mitigation of accidental scenarios (especially depressurization). Because there is no experience available on the GFR, a first step for demonstrating its feasibility is the operation of a 50-100 MWt experimental reactor, ALLEGRO, to qualify its specific fuel, materials and operating principles. Ideally, R&D results expected by 2012-15 could support a decision to construct ALLEGRO, possibly as a European Joint Undertaking. The next step would be a prototype GFR that could come 10-15 years after.

A refractory fuel concept for the GFR: reference option and alternatives

The GFR fuel should comply with:

- an operating temperature of 1200℃ in normal conditions and 1600℃ in accidental conditions (to offset the gas poor efficiency as coolant);
- a high fissile atom density and high thermal conductivity, thus triggering a renewed interest for carbide or nitride fuels;
- a power density in the range of 100 MW/m^3 as a trade-off between minimizing the plutonium content (lower boundary) and safety (slow-down of adiabatic heat-up).

Attempts to transpose attractive features of HTR fuel particles to fast neutron cores (fission product confinement, very high temperature resistance, thermal conductivity…) remained unsuccessful. Two concepts are presently under study: (1) a macro-structured plate-type fuel and (2) a cylindrical pin-type fuel, similar to LWR and SFR fuel elements, with changes to enable it to meet GFR requirements Preliminary studies finally led to select a macro-structured plate-type fuel as the reference. However, the alternative design based on ceramics clad fuel pins is thoroughly investigated, too.

The reference fuel element consisting of fuel pellets arranged in cells within a ceramics clad plate is shown on Figure 2. Each cell contains a fuel pellet composed of mixed uranium, plutonium and minor actinides. The clad is made of composite silicon carbide reinforced with SiC fibres (SiC-SiCf) for an increased mechanical resistance. The sub-assembly is composed of a stack of such plates axially piled up in a triangular array and enclosed in a hexagonal wrapper.

Mixed carbide (U,Pu)C is considered the optimum choice for the actinide compound, combining excellent neutronic and physical properties (high melting temperature, satisfactory thermal conductivity).

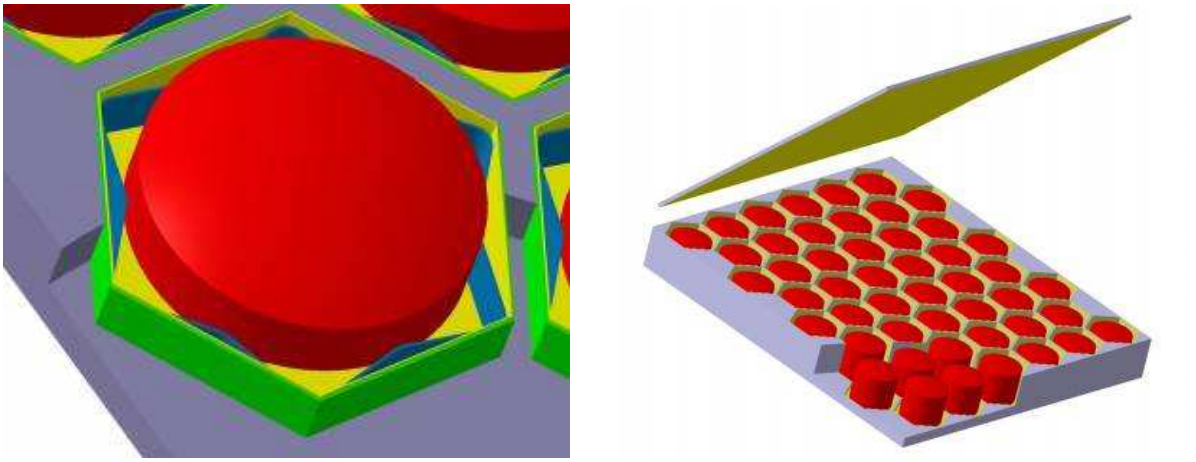


Figure 2: Cellular fuel sub-assembly with composite cladding material (SiC-SiC_f)

Significant progress has been made recently about the selection of constitutive materials (clad structure and liner) to ensure leak-tightness to fission products and comply with requirements of thermo-mechanical integrity and adequate chemical compatibility between materials. For the clad, candidate composites are Tyranno SA3 and Hi-Nicalon Type S fibers, both existing commercial-grade carbon composites. The internal liner is made of refractory metallic materials based on Mo, W, Nb, or Si based intermetallics. These metals are known to be more or less neutrons absorbers. The current design is a 50 µm layer of W-14Re.

4. Survey of fuel options for fast neutron reactors: potential and challenges

4.1 General fuel requirements for fast neutron reactors

Although the SFR, LFR, GFR fuel designs are quite different, their basic functional requirements are the same, summarized as follows [6]:

- to retain hazardous radionuclides in all but the most unlikely postulated conditions,
- to maintain a geometry that can be cooled,
- to maintain fissionable material in a controllled and predictable geometry, and
- to provide a convenient form for fuel handling.

Other mission-specific or system-specific requirements include reliable operation at high temperatures; compatibility with post-irradiation disposal or recycling technology; and technology-specific requirements for physical properties such as actinide element density, thermal conductivity, and melting temperature.

Neutron irradiation, high temperatures, and accumulation of fission products all work to degrade and stress the fuel's ability to meet these requirements. These factors limit the in-service lifetime or utilization of the fuel. In general terms, the degradation mechanisms that operate in current fuel designs include:

- chemical attack of the fuel cladding or fuel particle layers by fission products or fuel constituents, which weakens the barrier properties;
- stress of the cladding or fuel particle layers caused by increasing fission gas pressure and/or by volumetric swelling of the fuel material due to accumulating gaseous and solid fission products retained in the fuel material; and
- irradiation effects in the cladding or fuel particle layers, which can lead to embrittlement, enhanced creep damage, or dimensional changes. (Such dimensional

changes can be caused by void swelling in stainless steel cladding, irradiation shrinkage, and/or growth in materials with non-symmetric crystal structure).

Corrosion or attack of the cladding by the coolant remains an issue for most of the reactor fuels considered here, particularly for LFR fuels, where the oxygen content of the lead or lead–bismuth coolant must be controlled to prevent corrosion of stainless steel cladding and core components. Addressing these phenomena has a significant impact on burn-up limits for fuel utilization.

4.2 Fuel candidates for fast reactors

Four fuel systems have been investigated as candidates for fast reactor fuels [5,6]:

- mixed-oxide (MOX) fuel, historically (U,Pu)O2 or (U,TRU)Ox, where TRU represents transuranic elements, similar to the uranium oxide fuel used in commercial light water reactors (LWRs) and characterized by irradiation stability and relatively high melting temperature;
- metal alloy fuel, typically characterized by ease of fabrication, high thermal conductivity, and high uranium and plutonium densities;
- mixed carbides (MC), typically (U,Pu)C; and
- mixed nitrides (MN), typically (U,Pu)N.

For mixed carbides and mixed nitrides, the uranium and plutonium densities and thermal conductivities are closer to those of metal alloy fuels than to those of MOX fuels, and they exhibit good irradiation stability and relatively high melting temperatures (Table 1).

Table 1: Basic properties of oxide, nitride, carbide and metallic fuel

Pu/(U+Pu)=0.2	Carbide (U,Pu)C	Nitride (U,Pu)N	Oxide $(U,Pu)O_2$	Metallic fuel (U-Pu-Zr)
Heavy atom density (g/cm^3)	12.9	13.5	9.7	14.1
Melting point (°C)	2305	2720	2730	1070
Thermal conductivity (W/m.K)	12.8 (at 1000°C)	13.5 (at 1000°C)	2.1 (at 1000°C)	17.5 (at 500°C)

In the early days of the development of fast reactors, the key issue was to demonstrate the feasibility of breeder reactors (producing more fissile materials than they consume). In that respect, and among the various possible fuel candidates (Table 1), it was therefore natural to opt for a very dense fuel in heavy nuclides with metal fuel as a first choice. This is the reason why most of the very first liquid-metal cooled breeder reactors made use of metallic alloys.

However, the performances of these first metallic fuels were very limited in burn-up (a few GWd/tHM) because of important swelling under irradiation.

In the 50's, CEA also launched a R&D program on metallic fuels (basic physical properties, fabrication process, in-pile behaviour) looking for ternary compounds (such as U-Pu-Mo) to further improve the fuel performances. Again, swelling was much too high and, as a consequence, frequent spent fuel reprocessing was required to recover the fissile materials, thus inducing prohibitive fuel cycle costs.

This is the reason why France as most countries decided in the 60's to switch to oxide fuels with at that time some satisfactorily feedback experience of their use in PWRs. Despite its low thermal conductivity and theoretical density, oxide became the reference fuel for fast reactors because of its high melting temperature (above 2700℃ for 20% Pu enriched fuel) and its stability under irradiation.

4.3 MOX fuel

Despite low thermal conductivity and bad compatibility with sodium, oxide is still clearly the most mature and efficient fuel for SFR. Very high burn-up performances are proven resulting from considerable feedback from irradiation experience. Incidental and accidental behaviour has been assessed by a large number of tests, in particular the CABRI and SCARABEE international programs.

The demonstration of the mastering of fuel cycle is one of the key advantages of the oxide fuel. About 14 tons of Phenix fuel and fertile sub-assemblies were successfully reprocessed in representative conditions and the recovered Pu (about 4 tons) has been used to manufacture new Phenix fuels. Aqueous oxide fuel reprocessing has reached industrial maturity. The latest improvement is the COEX process which avoids pure Pu production while using available technologies. Pyrochemical process has been developed by Russia but the recovery yield and the used salts management are still to be improved.

MOX fuel can incorporate several percents of minor actinides as shown by the SUPERFACT pioneer experiment in Phenix (1986-1988). Several aqueous processes are under development for minor actinides recovery, either selectively or grouped with Pu, and have been successfully tested at lab scale.

4.4 Carbide fuel

Carbide offers both a high melting temperature (comparable to oxide) and a much larger, by a factor of 6, thermal conductivity associated to an increased heavy atom density. It is compatible with sodium (useful for clad failure management) but is more reactive with air than oxide with a pyrophoric character when provided through fine particles. In addition, although definitively not comparable to the oxide one, there exists a significant and globally positive experience on carbide behaviour under irradiation.

A key issue is the in-pile carbide swelling and fission gas retention. The behaviour is understood but must be mastered with respect to carbon content, oxygen impurity content and significant and stable porosity. In these conditions, it is possible to design a fuel element with reduced smear density but the advantage of the higher heavy atoms density over the oxide remains by a factor of 17%.

Linear heat rating can be increased (up to 750 W/cm for the He-bond concept) while keeping important thermal margins (Doppler reserve). This can be used to increase the core power density. Or, keeping the same level of power density, carbide fuel can significantly increase safety margins in comparison to MOX fuel.

However, undoubtedly linked to rather limited experience, many stakes remain for carbide fuel. A critical point is the mechanical interaction with a non-tensile clad (this presently limits the burn-up of carbide when cladded with ceramic). In the field of safety, the experimental database (transients, core accidents) is very limited and a complete evaluation is still necessary to properly balance advantages and disadvantages of carbide fuel.

For its manufacturing, an additional step is needed to produce the carbide (the oxide carbothermic reduction) and optimization is required to master pyrophoricity risks, Pu et Am losses, and to fulfil the specification required to properly manage swelling and gaseous products retention. Controlled inert atmosphere is necessary.

For reprocessing, despite very old fames claiming strong difficulties, carbide fuel is readily soluble in nitric acid and the aqueous processes is the attractive route to benefit from the experience gained with oxide fuels although the formation of organic soluble compounds may need for a decomposition step prior to the extraction cycles.

There is no available experience on carbide fuels incorporating minor actinides.

4.5 Metallic fuel

Metal fuel now is considered in many countries as the alternative to oxide or as the longer term option for SFR. Motivations for that are: high breeding capability, safety, pyroprocess fuel cycle,…

Metal has obviously the highest heavy metal density but the lowest melting point too. Metal is the “historical” SFR fuel. Compatible with sodium, its in-pile performances have been continuously increased. Together with oxide fuel, it can claim for proven high burn-up performances and significant safety database (US program in BRII and TREAT).

Experiments are underway on MAs beared fuels, in particular the METAPHIX experiment in Phenix.

As far as reprocessing is concerned, studies on aqueous processing did not generate very convincing results. UPuZr dissolution requires large amounts of fluoric acid (HF) or an anodic dissolution technique (BNFL). In addition, subsequent steps are needed to produce the alloy from the purified product (oxide). ANL studies led to promote the pyrochemical process. Three steps are involved: U recovery by electro−winning, transuranics recovery by electro−refining on a liquid cadmium cathode, and the separation of cadmium from TRU by distillation. The efficiency of U recovery is largely demonstrated (treatment of the EBR-II used fuel), additional work is still needed for TRUs.

Safety issues must be carefully addressed. Metallic fuel provides a large negative reactivity feedback due to fuel expansion. However, negative aspects are low temperature eutectic formation, an increase of sodium void worth and the degradation of Doppler effect.

4.6 Carbide or nitride?

Compared to the oxide experience, carbide and even more nitride fuels are far from mature and left unanswered a number of issues concerning their real potential as SFR fuel. Particularly, very limited experience exists in the areas of safety and of the closure of the fuel cycle.

Both carbide and nitride fuels are under consideration for GFR and SFR. For both, to achieve equivalent neutronic performances, nitride fuels require nitrogen enrichment to at least 50 at% ^{15}N to avoid ^{14}C production by (n,p) reactions on ^{14}N. Moreover, with nitride fuels, generation of helium, from (n,α) reactions in ^{14}N and additional tritium generation are concerns for fuel performance and for coolant radiological contamination. Recent irradiations in the Phenix (France) and Joyo (Japan) reactors have confirmed that, for certain closed-system operating conditions at high temperatures, nitride fuel can exhibit signs of dissociation of the (U, Pu)N phase.

Carbide and nitride are basically very similar in nature, in crystallographic structure, have equivalent physical properties, and quite similar in-pile behaviour. They exhibit fairly similar properties with regard to their chemical reactivity with air and/or water.

Nevertheless, recent results of experiments conducted in Phenix (NIMPHE) clearly showed a lesser stability of nitride fuel. In particular, in equivalent thermal regimes, nitride experienced a central hole formation (Figure 3).

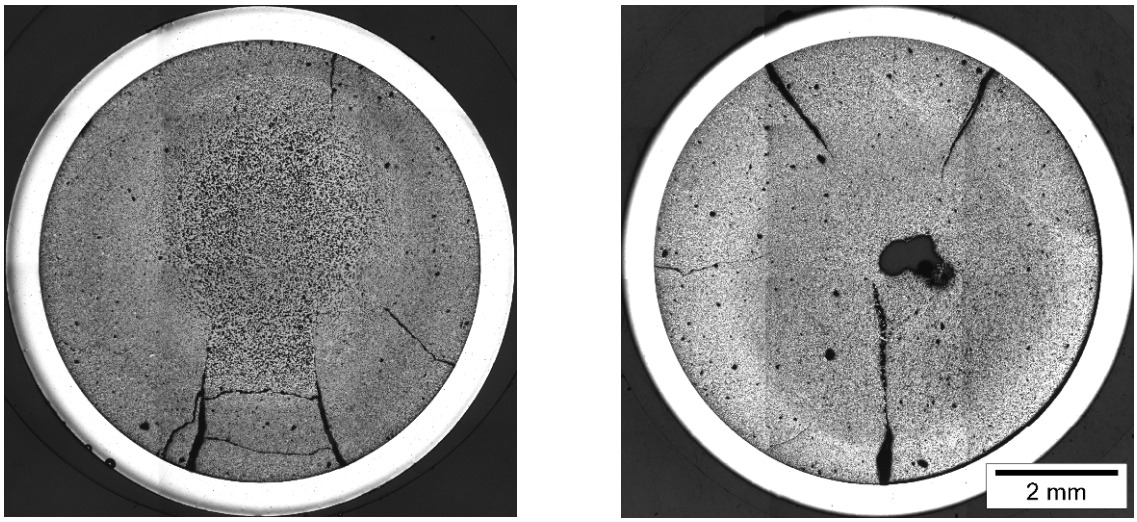


Figure 3: Carbide (UPuC) and nitride (UPuN) fuel behaviour, NIMPHE 2 irradiation in Phenix (~7at%). Carbide (left), Nitride (right)

More puzzling, examinations showed the presence of a metallic Pu-rich phase at the pellet-clad interface (Figure 4). This is considered to occur when the fuel maximum temperature exceeds 1600℃. The situation would be worsened if a concomitant clad failure occurred as it would not let the nitrogen partial pressure increase in the pin which is favourable to limit the extent of the metallic phase formation.

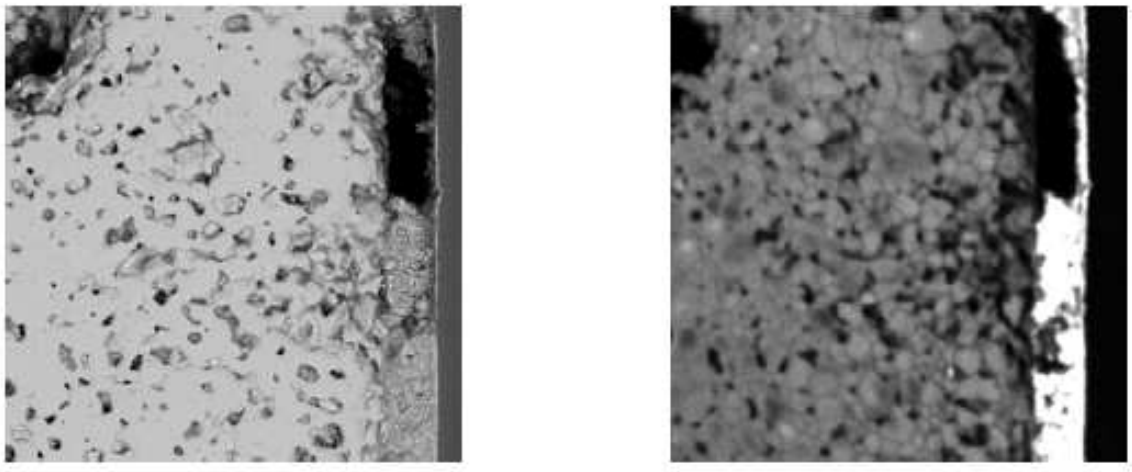


Figure 4: Microprobe pictures of nitride fuel irradiated in Phenix, showing metallic Pu in the fuel-clad gap. Electronic picture (left), Pu X picture (right)

4.7 Fuels for transmutation

The main restriction to introducing minor actinides into the core (homogeneous recycling mode) is linked to their impact via on the core reactivity and kinetic factors. The fractions of

minor actinides (MA) considered as acceptable are 3% and 5% for the SFR and GFR, respectively [7].

The incorporation of MA has some impact on the physico-chemical properties of fuel material. Some results are available for the incorporation of MA in MOX fuel (smaller melting temperature, influence of stoechiometry on thermal conductivity, redistribution of Am). But additional data are needed to guarantee the safe operation of the reactor and fuel cycle facilities (fuel fabrication and reprocessing).

The SUPERFACT irradiation in Phenix (1986-1988) represents the main body of existing knowledge on the in-pile behaviour of MOX fuel loaded with MA. This demonstrated the feasibility of MA incorporation up to 2% in (U-Pu-Am)O_2 et (U-Pu-Np)O_2 fuels (Figure 5).

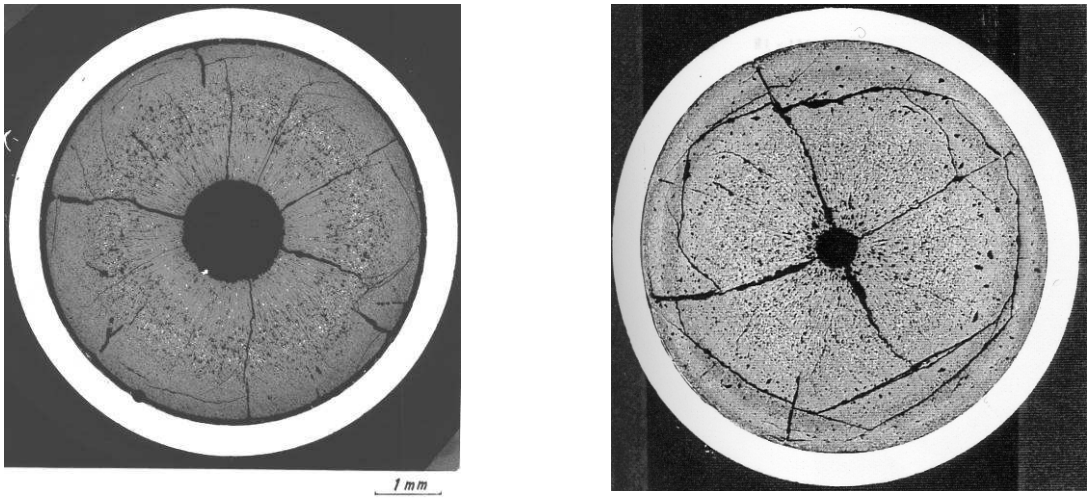


Figure 5: SUPERFACT fuels (MOX fuels with or without MA)

SUPERFACT also showed that the addition of MA in significant quantity leads to increased helium production which should be accommodated in the fuel design (fuel element free volumes).

5. The qualification of innovative fuels

5.1 Phenix feedback experience

The Phenix fast sodium reactor resumed operation in 2003 after 6 years of a safety re-evaluation process. Authorization was granted for an operating period of 720 EFPD, which is 6 cycles of approximately 180 days of operation at 2/3 power. The reactor has had good performance with availability factors at 74%, 85% and 78% in 2004, 2005 and 2006, respectively. Good reactor operation has enabled both electricity production of and the performance of irradiation programs according to the prescribed planning [8].

Phenix proved its excellent capability at performing experimental irradiations, owing to core characteristics, operation flexibility, availability of hot cells for capsule mounting and post-irradiation examinations. More than 200 experimental irradiations have been realized in the areas of MOX fuel and dense fuels (carbide, nitride) behaviour, clad and hexagonal tube materials, transmutation of minor actinides (homogeneous and heterogeneous modes) and of long-lived fission products, innovative fuel concepts and materials for 4th generation reactors,… Table 2 lists the experiments conducted in Phenix in the frame of irradiation programs on transmutation and innovative systems.

The feedback experience gained from Phenix is considerable and has been very helpful at identifying the areas to be further investigated for innovative fuels and reactor systems.

Table 2: Relevant irradiations in Phenix (transmutation and innovative systems)

Domain	Area	Irradiation	Topic
Generic	Neutronic data	PROFIL (R & M)	Individual isotopes (^{244}Cm, ^{243}Am, ^{241}Am and ^{242}Pu)
	Fuels	NIMPHE	Carbide and nitride fuels
	Inert matrices	MATINA 1A-2-3	Ceramic (MgO) and refractory metals as target support materials (MA simulated by fissile phases)
Transmutation (homogeneous mode)	Actinides (oxide fuel)	SUPERFACT 1	Np and Am oxide
	Actinides (metal fuel)	METAPHIX	Metallic fuel UPuZr with Am, Cm, Np and lanthanides
	Actinides (metal, nitride, CERMET, CERCER)	FUTURIX FTA	Metallic alloys (UPuAmNpZr, PuAmNpZr) and nitrides (PuAmZrN, UPuAmNpN)
			$PuAmO_2$ and $PuAmZrO_2$ macrodispersed in Mo (CERMET)
			$PuAmO_2$ microdispersed in MgO (CERCER)
Transmutation (heterogeneous mode)	Actinides	SUPERFACT 1	Np and Am oxide
		ECRIX B	AmO_2 dispersed in MgO (core conditions)
		ECRIX H	AmO_2 dispersed in MgO (blanket conditions)
		CAMIX	$AmZrYO_2$ in solid solution
		COCHIX	$AmZrYO_2$ micro- and macrodispersed in MgO
	Long-lived FP	ANTICORP 1	Tc
Innovative systems	SFR oxide fuel	CAPRIX	MOX fuel with high Pu content
		COPIX	Standard oxide fuel with austenitic clad
		MATRIX	ODS and other clad materials
	GFR (cladding materials)	FUTURIX-MI	Carbide and nitride ceramics, refractory Mo- and Nb- based metallic alloys
	GFR (fuel samples)	FUTURIX-Concepts	Carbide and nitride fuels in SiC and TiN matrices

5.2 Irradiation tools for the qualification of fuels in the 2008-2020 time period

Material testing reactors, hot laboratories and fast neutron reactors (with experimental capability) are essential R&D infrastructures to explore innovative research on fuels and fuel cycles, key technologies for future reactors.

All MTRs in operation in Europe (OSIRIS, HFR, Halden, BR2) and USA (ATR) are more than 40 years old. OSIRIS will be shutdown before 2015. The Jules Horowitz Reactor (JHR) is planned for start-up in 2014. JHR is a 100 MWt MTR designed to produce a fast neutron flux of 10^{15}n/cm^2, which is twice the capacity of classical MTRs and close to experimental fast neutron reactors. JHR should therefore provide a good answer to a number of irradiation needs.

However, fast neutron reactors will be needed, especially during the phase of concept qualification which requires irradiations in realistic conditions of fast neutron reactors.

In Russia, the BOR-60 experimental fast reactor may be shut down in 2010 and the BN600 (600 MWe) fast reactor continues to operate with an excellent load factor. Following the excellent results obtained by BN600, Russia has re-launched the BN800 project. China is currently in the process of building a 65 MWt research reactor (CEFR), scheduled for divergence in 2009. In Japan, Joyo has been shutdown recently and work is underway on Monju (250 MWe) for its re-divergence. In India, a 500 MWe power reactor (PFBR) is under construction, scheduled for divergence in 2010, the first out of a series of three sodium reactors. The potential of Monju (to be re-started), BN600 (under operation) and BN800 and PFBR (under construction) for experimental irradiations has to be confirmed.

In France, the decision to build a prototype of 4th generation fast neutron reactors (ASTRID) by 2020 is important for Europe to remain credibly involved in R&D on fast neutron systems after Phenix is shutdown in 2009 (Figure 6).

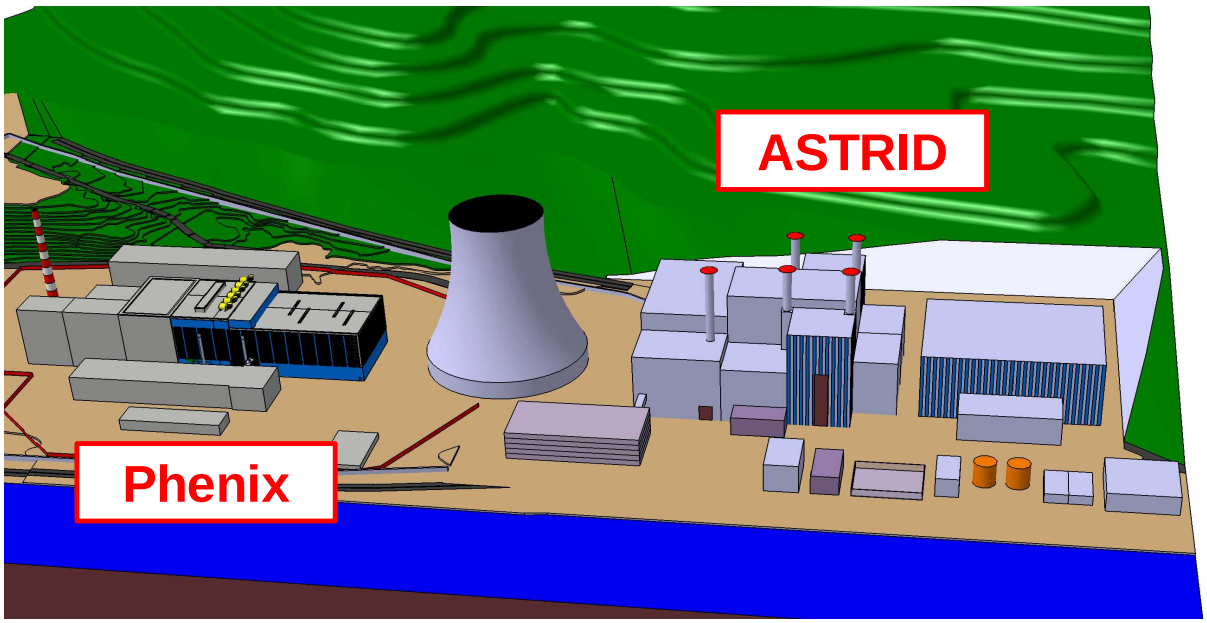


Figure 6: Phenix and ASTRID in the Marcoule site (artistic view)

Other European countries, and possibly other international partners, may decide to develop experimental or prototype reactors of other Generation IV systems of specific interest, with invited external participation. The prospect of such prototypes does not appear excessively ambitious in comparison with the number of experimental and prototype reactors operating in Europe in the 1980s.

5.3 Irradiation programmes

SFR fuel development

Three main domains are currently demanding for irradiation experiments: the start-up fuel for ASTRID (large diameter oxide fuel pins with a classical austenitic clad), the final driver fuel for ASTRID (large diameter oxide fuel pins with an ODS clad) and the development of transmutation fuel (MAs homogeneous recycling in which a little amount of actinides is diluted in the fuel, and MAs heterogeneous recycling for which the preferred option is currently to concentrate up to 20% of MAs in a UO_2 matrix in radial blanket S/As). At the current stage of development, SFR needs on carbide fuel are considered to be covered by the GFR program. These different options are obviously not at the same level of maturity, therefore requiring to developing different types of irradiation experiments.

In any case, after Phénix reactor shutdown in 2009, the need to realize prototypic irradiations on the most mature options will be a difficulty in the 2012-2020 period and France considers that discussions must be internationally opened to equip for example the Monju reactor with irradiation capsules. France is also considering that irradiation capsules are to be developed for the ASTRID prototype.

For less mature options, existing MTRs and, after 2014, the Jules Horowitz Reactor (JHR) will have a key role to play to provide analytical and instrumented experiments for acquiring basic knowledge, screening options and providing features on the fuel integral behaviour prior to the realization of prototypic experiments.

These general features of the SFR irradiation program are shown on Figure 7.

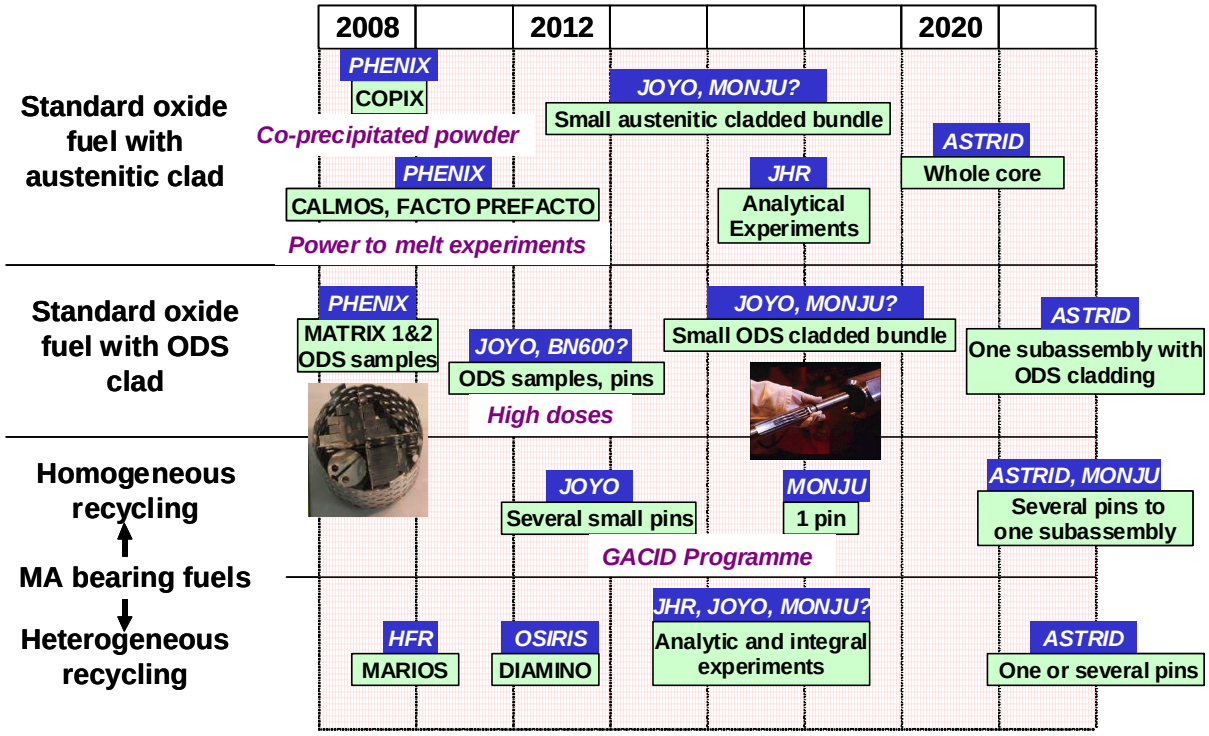


Figure 7: SFR irradiation programme

ASTRID start-up fuel and driver fuel

The objective is to start the ASTRID core with subassemblies made of large diameter oxide fuel pins cladded with a classical austenitic steel and to introduce, very soon after ASTRID start-up, one sub-assembly with ODS cladded pins.

The irradiation program takes benefit from the considerable experimental feedback from Rapsodie, Phenix (in which still relevant experiences have been made but are not already examined) and SPX. Major mid-term objective would be to irradiate in prototypic conditions small bundles of large diameter pins: this raises the question of fast neutron reactor availability with appropriate irradiation devices.

As far as ODS are concerned, the objective is to acquire quickly high fast neutron fluences on samples of the various candidate materials and then to irradiate the selected ODS cladded pins in SFR representative conditions.

The Phenix reactor last neutrons have been used in particular for material irradiation experiment (MATRIX 1 and 2) to start acquiring data on ODS and to check the behaviour of oxide fuel manufactures with (U,Pu) co-precipitated powder. A power-to-melt experiment on pre-irradiated oxide fuels will also be performed in order to enrich the corresponding database necessary for fuel codes qualification for an accurate evaluation in the design calculations of the margins to melting.

Transmutation fuels

For the homogeneous recycling which is a mature option after the SUPERFACT experiment performed in Phénix in the 80's that has shown a fuel behaviour similar to standard one, the program is performed within the GACID R&D project of the Gen IV collaboration. It has the objective to irradiate MAs bearing fuel in Monju, first at the pin level, then at a more significant scale (several pin or whole subassembly in Monju or GACID).

For the heterogeneous recycling and in particular the case of blankets loaded with MAs, the high MAs content raises the question of the management of the large quantity of He produced associated to very particular irradiation parameters evolution. Therefore, a specific transmutation fuel microstructure is to be developed which requires several steps in MTRs

reactors (HFR, OSIRIS and then JHR) before envisioning the irradiation of a whole pin in prototypic conditions. The objective is to irradiate one or several pins soon after ASTRID start-up.

GFR fuel development

The irradiation programme for GFR fuel was built in order to select materials and design for a first step (2003-2006) and then to develop some fuels in order to demonstrate their feasibility by the end of 2012. Figure 8 shows the main irradiations of the programme.

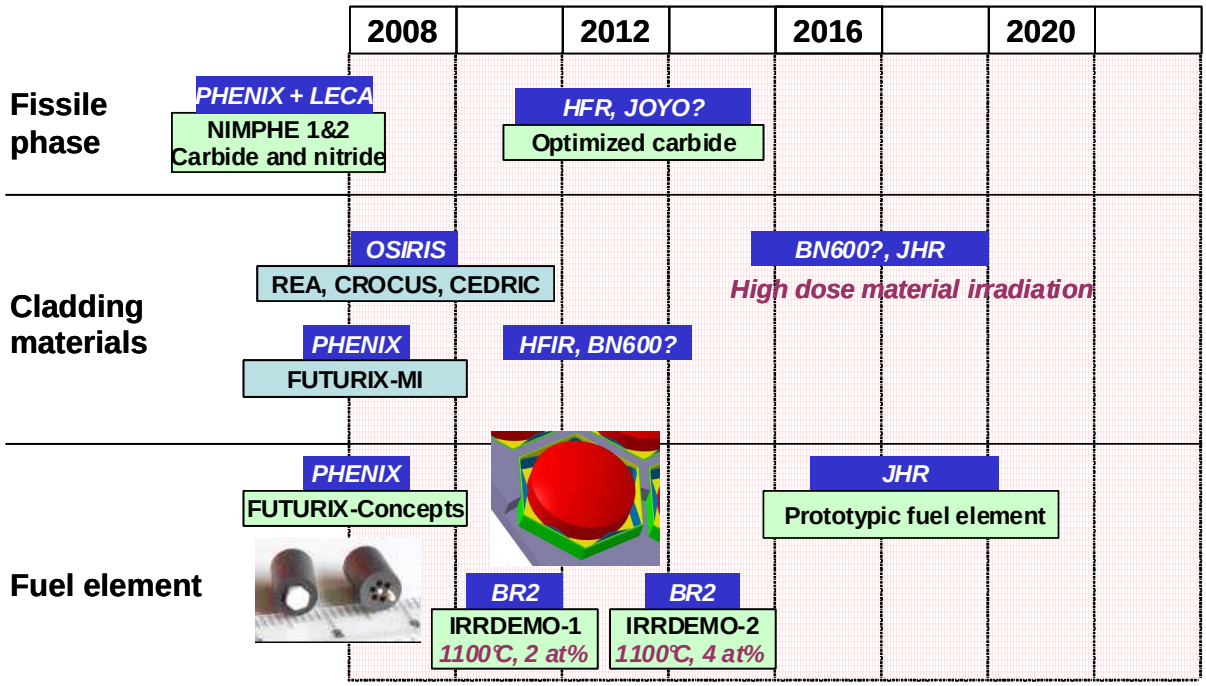


Figure 8: GFR irradiation programme

The FUTURIX-MI experiment aims at assessing the impact of irradiation in a fast neutron flux at high temperature (between 900°C and 1000°C) on inert materials considered as potential components for GFRs, essentially carbide (SiC, TiC and ZrC), nitride (TiN and ZrN) type ceramics and refractory Mo- and Nb- based metal alloys.

The objective of FUTURIX-Concepts irradiation in Phenix is to testing the behaviour of different concepts of fuels in a fast neutron flux (3.5 10^{15} n.cm^{-2}.s^{-1}). These concepts are considered as precursory fuels for GFR, carbide and nitride fuels embedded in silicon carbide and titanium nitride matrices, respectively. The IRRDEMO irradiation will be conducted for the qualification of fuel element designs and will be extended to higher burn-up, higher temperature, off-normal conditions and fresh fuel fabricated with spent fuel.

6. Summary and perspectives

The rising cost of uranium resources, together with the accumulation of spent fuel, drives the need to switch to fast neutron reactors to achieve a more efficient use of uranium and minimize the ultimate long-lived radioactive waste. SFR is the reference option, not only in France but also in Europe. The European strategy considers both the GFR and LFR as alternatives to the SFR.

A common concern is to achieve a convincing demonstration of the capability of fuels to attain the ambitious goals set to 4th generation fast neutron systems, especially in terms of performance (uranium conversion, minimization of long life radioactive wastes) and safety.

Owing to the important and satisfactory feedback experience built upon oxide fuels, MOX is the reference fuel for the SFR, at least for the start-up of the prototype (ASTRID). The objectives followed for the 4th generation SFR for safety (for example sodium void worth reduction and limited core reactivity excess) and cycle performances (self-sustainable core with a near zero breeding gain, reasonable in-core Pu inventory, MA transmutation) are achievable with an oxide fuel in large power cores (3600 MWt) while implementing adequate design features. Nevertheless, recent calculations show that the use of a dense and cold ceramic fuel might even improve the core performances. Carbide and nitride are candidate fuels to be investigated for SFRs of 4^{th} generation. Based on recent experimental results carbide is preferred to nitride by the CEA.

For the GFR and the LFR, dense fuels are required to achieve self-generation because of the higher fraction of coolant in the core. Carbide and nitride are currently the reference fuels for the GFR and LFR, respectively.

Experimental reactors are needed for further assessment of the in-pile behavior of fuels with representative materials and realistic conditions (burn-up, MA content, neutron flux…). An optimal use of existing irradiation reactors (Phenix, Joyo, Monju, BOR-60, BN-600) is necessary, until new reactors, under construction (JHR, CEFR, PFBR) or planned (ALLEGRO, ASTRID) can be put in operation. International collaboration is essential to assure the continuous availability of irradiation infrastructures.

7. References

[1] F. Carré *et al*, *Outlook To France's R&D Strategy On Future Nuclear Systems (From Gen II to Gen IV reactors and fuel cycle)*, RRFM IGORR 2007, Lyon, France, March 11-14, 2007.

[2] F. Carré, C. Renault *et al*, *Les réacteurs rapides de $4^{ème}$ generation et leurs combustibles*, RGN, 2006.

[3] J.Y Malo *et al*, *Gas Cooled Fast Reactor 2400 MWth, end of the preliminary viability phase,* Proceedings of ICAPP 2008, Anaheim, CA USA, June 8-12, 2008.

[4] P. Richard *et al*, *GFR Fuel and Core Pre-Conceptual Design Studies*, International Conference on the Physics of Reactors "Nuclear Power: A Sustainable Resource", PHYSOR 2008, Interlaken, Switzerland, September 14-19, 2008

[5] J. Rouault *et al*, *Survey of candidate fuels for Gen IV Sodium Fast Reactors with a closed fuel cycle,* Presentation at ICAPP 2007, Nice, France, May 13-18, 2007.

[6] D. Petti, D. Crawford and N. Chauvin, *Fuels for Advanced Nuclear Energy Systems*, MRS Bulletin Vol 34, January 2009.

[7] F. Varaine *et al*, *Review on transmutation studies at CEA: scientific feasibility according neutronic spectrum*, Proceedings of GLOBAL 2005, Tsukuba, Japan, October 9-13, 2005.

[8] J. Guidez, *Status of Phenix Operation and of Sodium Fast Reactors in the World*, Proceedings of ICAPP 2007, Nice, France, May 13-18, 2007.

DECOMMISSIONING PROGRESS OF THE DOUNREAY SITE.

ELIZABETH MACKENZIE
Fuel and Waste Strategy Manager,
Waste Services Unit, Dounreay, Caithness, Scotland, KW14 7TZ

ABSTRACT

Dounreay was at the centre of fast reactor research in the UK for over four decades however now it has moved into a new phase of its life and leading the way in reactor and nuclear facility decommissioning. The experimental nature of the many of the facilities means that clean-up and demolition requires innovation as well as great care. Over 180 facilities were built on the Dounreay site which covers over 140 acres.
DSRl are currently managing the decommissioning of the facilities and managing the waste and fuel leagacies on the site. This paper gives an overview of the work being carried out to deliver the work programme and how the fuel and waste is expected to be managed during its lifetime.

1 Introduction

Dounreay was at the centre of fast reactor research in the UK for over four decades however now it has moved into a new phase of its life and is leading the way in reactor and nuclear facility decommissioning. The experimental nature of the many of the facilities means that clean-up and demolition requires innovation as well as great care. Over 180 facilities were built on the Dounreay site which covers over 140 acres on the north coast of Scotland.

A number of facilities and buildings have already been demolished and work is moving forward at a significant pace with contaminated areas being removed and the resulting waste being conditioned and treated. This paper outlines the challenges of dealing with the variety of wastes created during the work.
The treatment of the fuels which remain on the site will also be discussed in this paper and the site is working with NDA (Nuclear Decommissioning Authority) to develop a United Kingdom strategy for the treatment and long term storage of the irradiated and un-irradiated fuels.

Dounreay began construction in 1954 and eventually stopped supplying power to the grid in 1994. It has had various milestones during its history including
1957 – First nuclear reaction in Scotland takes place criticality test cell at Dounreay,
1961 – Dounreay becomes first fast reactor in world to supply power to grid,
2004 – MTR fuel fabrication ceases.
Decommissioning commenced following the cessation of reprocessing in 1996 although it was quite limited at that time and was expected to take in excess of 70 years.

UKAEA formally published its first plan to return the site to a brown field site in 2000 and is now continually refining the timescales, the current completion date is 2025. At this point all that will remain on site are conditioned waste stores and a nuclear material store.
Dounreay has consulted the local stakeholders on the end state of the site and have an agreed brown field conditions for the site to be left in along with key facilities that will be required. The long term future and retention of the Dounreay Fast Reactor dome is still uncertain as discussion is ongoing with Scottish Heritage. It is after all a key historical building in Scottish history.
The current estimated cost to complete the decommissioning of the site is £2.5billion.

Just as Dounreay lead the world in research reactors it is now leading the world in decommissioning and removing some of the most challenging facilities. This paper gives a flavour of the work currently being undertaken at Dounreay and some of the challenges yet to be undertaken.

The site has recently completed full decommissioning of the criticality test cell and the fuel fabrication facility (D1202) maintaining our leading edge for successful decommissioning nuclear facilities.

2 General Decommissioning of Dounreay site

A number of facilities have already been fully decommissioned and taken back to floor slab level. This includes removal of all internals and demolition of the external building. These include a number of active and inactive facilities including the original fuel pond at the Dounreay Material Test Reactor (DMTR) and the cells where the first criticality experiments were completed.
It may sound easy to demolish a legacy facility but these building were not built like modern facilities standards which have decommissioning in mind. The operations that were carried out within them meant that the facilities contain both chemical and radiological hazards which required to be addressed.
A staged approach is undertaken to ensure that clear evidence of the waste type and activity can be assessed and agreed before any waste/materials are removed from the facility.

Stage 1 – Survey of facility– a) divide facility into discrete areas,
b) carryout physical inventory of areas,
c) carryout monitoring of area and
d) take appropriate sample using well defined techniques for characterisation purposes.
Stage 2 –Review of facility documents. This will include history of facility, work completed, any incidents, materials known to be used in the facility.
Stage 3 – Consideration appropriate adoption of the waste hierarchy.
Stage 4 - Review possible waste routes available to the site.
Stage 5 –Agree approach for removal and consignment of waste prior to removal.

Stage 1 and 2 are completed prior to any removal of redundant equipment and internals. This along with collation and review of the history of activities within the facility allows a better understanding of the waste types and volumes that will be generated during the decommissioning of the facility.
With this knowledge and careful planning undertaken then the waste hierarchy can be applied ie minimisation, reduction, potential reuse, etc.
A plan is then produced to demonstrate the proposed methodology for removal and for waste category assignment. This needs to be agreed prior to generation of the wastes.

2.1 MTR fuel fabrication facility.

The fuel fabrication facility, D1202 ceased operations in 2004 and went into a phase of shutdown prior to post operational clean out. Over the past few years the facility was internally stripped of it's equipment. The facility was a steel framed single storey building located within the Fuel Cycle Area (FCA). It was constructed in 1956 and measured 64m long by 26.5m wide by 4.73m high. It linked to the other facilities within the FCA via a small link corridor. The facility contained a number of small rooms and workshops which were used to fabricate various items prior to assembly of the elements.
The strip out and demolition of the facility followed the stages outlined above. Due to the nature of work that had been carried out in D1202 no RHILW was identified to be disposed of. However LLW, Exempt and clean waste was generated in various forms.
The removal of the waste was done in a specific order following a detailed plane so that segregation of the waste could be maintained. This allowed the team to fully utilise and maximise the amount of waste that could be demonstrated to be clean and hence disposed of to normal landfill sites.

An example of this was the soft strip of the internal building rooms which included removal of doors, windows, ceiling and plasterboard, plumbing and wiring. As may have been expected with a facility from this era, some asbestos was identified following detailed surveying and this was treated and followed the specific waste route for such materials.
The final demolition of the external framework was carried out over a short period of time, by an experienced local contractor who worked to the relevant codes of practice for such work.

This facility has now been completely demolished and all that is left is the concrete structural plinth. The demolition of D1202 was completed within 2years with no lost time accidents or safety events.

Photo 1 - D1202 before demolition commenced

Photo 2 – Only Floor slab left

3 Waste Routes

One of the biggest challenges for the Dounreay site is to have all the waste routes available for the waste expected to be generated during the decommissioning of the site. The includes all types and forms of waste from the Remote Handleable Intermediate Level Waste (RHILW) all the way through to the clean daily waste from the offices.
Due to limitations on removal and movement of the radioactive wastes from site imposed by Scottish Environment Protection Agency (SEPA), this has resulted in the radioactive waste requiring to be treated and stored on this site. Some treatment and conditioning facilities have been constructed on site but some are still due to be constructed to allow the decommissioning of the site to progress. Currently on site a conditioning (compaction) facility exists for Low Level Waste (LLW) and a cementation/grouting facility for MTR (Material Test Reactor) raffinate.

3.1 Waste conditioning and treatment facilities.

The new facilities which are currently in the final stages of planning and design and are expected to be constructed and ready for operation by approx 2014, they are,

1 - RHILW encapsulation facility (for solid and liquid streams)
2 - LLW repository which will hold all the LLW expected to be generated from the site.
3- Encapsulation facility for the shaft waste (due on line approx 2016)

Some smaller pre treatment or preparation facilities will also be constructed close to the reactor facilities. This will allow size reduction and in some cases encapsulation/grouting etc to be carried out close to the generation of the wastes. The resulting conditioned wastes will be stored either in the LLW repository or in RHILW conditioned waste store being constructed on site.

3.2 Waste Types

3.2.1 RHILW

The RHILW will be packaged into the various waste containers, chosen by the site as most suitable for its needs, these include 500l drum, 3m3 box and 4m box. The 4m box is currently only expected to be used for the solid graphite from decommissioning the DFR facility. The majority of the waste will be in the 500l drum, (current estimate is 8500 drums)
All RHILW generated will be subject to the RWMD Letter of Compliance process so that the resulting waste forms are suitable to be transferred to the national repository when it becomes available.

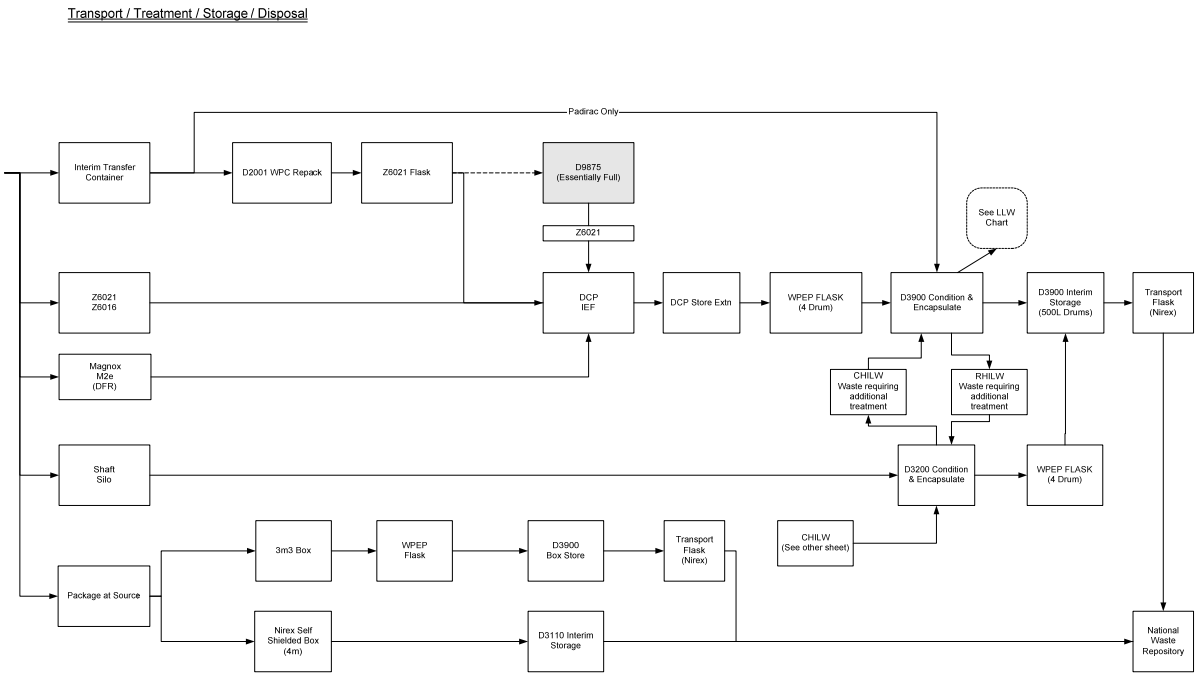


figure 1 - a flow chart for the movement of RHILW on the Dounreay site

3.2.2 LLW

The LLW will be put in 200l drums within the facilities and then these will be supercompacted and placed in Half Height ISO (HHISO) containers, these will be then be grouted prior to placing in the vaulted LLW repository.

Low Level Waste – *An article or substance that is radioactive or contaminated under the RSA 93. The activity of the waste must not exceed the following values: -*

(i) All alpha-emitting radionuclides 4 GBq/Tonne

(ii) All other radionuclides not including (i) above 12 GBq/Tonne.

3.2.3 Clean and Exempt

Radiologically Clean – *An article or substance that has never been contaminated or activated. This is usually declared based on provenance alone. An article or substance for which there is inadequate provenance to justify an immediate declaration as clean may still be declared clean if suitable measurements confirm the absence of activity above background for the article or substance in question. (Clearance and Exemption Principles, Processes and Practices for Use by the Nuclear Industry. A Nuclear Industry Code of Practice).*

Exempt – *An article or substance that is radioactive or contaminated because it:*

- *Contains levels of specified radionuclides above RSA 93 Schedule 1 exclusion limits*
- *It contains other radionuclides wholly or partly attributable to either an artificial process or because of the disposal of radioactive waste.*

However in both cases, at levels below relevant limits in Exemption Orders under the Act. (Clearance and Exemption Principles, Processes and Practices for Use by the Nuclear Industry - A Nuclear Industry Code of Practice).

The key to making the most of the waste routes available is good characterisation and planning.

4 Fuel Routes

All the research work undertaken at Dounreay has obviously left the site with a wide variety of fuels in all forms. This legacy of fuels now requires to be dealt with. In order to achieve this Dounreay will require tomake them safe and suitable for long term storage without foreclosing the option for some of the fuels to be used in nuclear new build.
New facilities will be built to repack the fuels into industry standard packages and this will allow long term storage on site.
The site has both irradiated and unirradiated fuels and Dounreay is working with NDA (Nuclear Decommissioning Authority) to develop a United Kingdom strategy for the treatment and long term storage of the both fuel types. The site currently has a reference treatment and stabilising option for each fuel type, these do not foreclose the option of the Plutonium and Uranium being re-used in the future.

4.1 Irradiated fuels (primarily from Prototype Fast Reactor - PFR)

The majority of this fuel is mixed Plutonium fuels in oxide form. Due to the specific nature and design of the fuel elements and the burn up, these will not be processed but will be conditioned and are currently undergoing evaluation to establish their suitability for direct disposal to a national spent fuel repository.

4.2 Unirradiated Plutonium and mixed Plutonium/Uranium oxides

The NDA are currently working on this particular fuel type and have published a high level draft of the options. At Dounreay a new facility is being designed to allow better characterisation of the material and then repackaging into containers which are used elsewhere in the UK to store Plutonium. This will allow all the Plutonium stocks to potentially be co-located and treated in the future once the UK strategy is finalised.

4.3 High enriched Uranium (HEU)

During its operational life Dounreay had a facility which dealt with a variety of enrichments of uranium (from LEU to 93% HEU). Some of this material is still on site in various forms from liquid to solid billet form. The material will be stabilised to allow continued storage, until another use can be identified. The HEU is an asset which has a value and with further treatment will be suitable for re-use.

4.4 Miscellaneous

The site also has carbides, natural/depleted uranium, metallic uranium (ex DFR breeder),thorium and these are in various forms. These fuels will also be stabilised and put in industry standard containers suitable for long term storage or transport if required.

5 Summary

Just as Dounreay lead the world in research reactors it is now leading the world in decommissioning through the removal of some of the most challenging facilities. This paper has just given a very brief outline of the work being undertaken on site.
There is still a lot of work and challenges to be undertaken at Dounreay with a number of new facilities to be constructed to allow the decommissioning of the site and many old facilities to be removed and demolished.

The waste routes need to be further developed. Maximising the clean and exempt exit routes by better segregation is a key area for the site as this will reduce the volume to be treated and stored on site for the long term. Business driver and cost reduction.

However over the next 15years Dounreay will be transformed to a site which is landscaped and ready to face the new challenges of the 21st century. As the NDA are looking at life after Dounreay and supporting new developments within the area maybe the site will possibly be the ideal place for new alternative energy developments or industries.

SECURITY OF SUPPLY FOR FISSION MEDICAL RADIO-ISOTOPES BASED ON OPTIMAL USE OF THE TEST REACTOR NETWORK

H. BONET
National Institute for Radioelements (IRE)
B-6220 FLEURUS Belgium

ABSTRACT

Nuclear Medicine relies to a large extend (80 % of the procedures) on radioisotopes produced by fission of uranium, on Mo^{99}/Tc^{99m} for 28 million diagnoses made annually all over the world for tracking diseases in cancerology, cardiology, neurology … and on $I^{131}$and Y^{90} for 3 million therapy procedures. The only four main producers (95 % of the world demand) are relying on 5 aging test reactors for irradiating HEU targets to be processed for extracting these short life isotopes before their conditioning as radiopharmaceuticals to be daily used in hospitals. Ensuring the security of supply has been a challenge for many years and if several shortages occurred in the past, the last crises in 2007 and 2008 revealed more than ever the weakness of the current situation despite the efforts and warning that have been devoted to facing many obstacles including possible technical failures, incidents, transport constraints and licensing issues, as well as political threat for the use of HEU. It is time for having all stakeholders drawing the lessons of the crisis and considering all possible serious and realistic improvements on technical and organisational issues without neglecting the resulting economical and safety aspects.

1. Introduction

Besides radioisotopes produced by activation of stable isotopes either in standard test reactors or in dedicated particle accelerators facilities, the isotopes used in nuclear medicine are mainly produced by fission of high enriched Uranium (HEU) in high flux reactors.
For diagnostics of many kind of diseases in cardiology, oncology, neurology, … nuclear medicine imaging is unique technique providing functional information and unique approach of physiological and biochemical process up to the cellular level, which complement other imaging methods focused on physical and structural information. More recently immuno-diagnostic agent combining monoclonal antibodies marketed with radioisotopes have been developed for diagnostic as well as therapy at the cellular level. Currently more than 100 different diagnostic procedures associating radioisotopes with cold molecules, are used for performing 35 millions exams per year in the world. They rely mainly on Tc^{99m} (70 %) when the growing use of PET isotopes (F_{18}DG) still does not exceed 5 %.
The uneven use of molecular imaging in more than 10.000 hospitals in the world (55 % in N. America, 25 % in Europe, 20 % in ROW) lead to an expectation of average growth between 5 and 10 % per year over the next decade.
Tc^{99m} ($6hT_{1/2}$) the daughter of Mo^{99} ($66hT_{1/2}$), is supplied in hospital as Mo/Tc generators useful for only 1 week because of the loss of 1 % of activity per hour. This approach allows availability of Tc^{99m} 365 d/year (20 % is to be used for emergency) on the basis of a weekly delivery of generators all over the world which present a logistic and cost advantage versus such isotopes as F_{18} ($2hT_{1/2}$).

2. Radioisotopes shortfall crisis

Although the supply of any medical isotopes could be disrupted by problems in the supply chain, the more significant world crises are related to the availability of the Mo^{99}/Tc^{99m} generator, because of its extensive use and the short half life precluding significant anticipation.

Since many years, the risk of a significant shortage of supply of Mo^{99} obsessed the leading producers but they generally succeeded to limit both the duration and level of the shortfall. The irradiation of HEU targets is limited by the availability of a very few aging test reactors not specifically dedicated to the radioisotopes production, having limited operating time due to maintenance and refurbishment requirements, even some of them being suddenly definitively shutdown. Then despite efforts to optimize the operating schedules which remain under the only control of the reactor operators, the theoretical excess capacity shown on the table 1 is not sufficient to face sudden interruption of operation considering the minimum time required for sufficient irradiation (100 to 160 hr) and circumstances of simultaneous shutdown. Technical incidents requiring repair or even simple reactor SCRAM followed by xenon poisoning, lead often to production shortages hardly compensated by the emergency supply of back up from other facilities. In some cases reactor performance was unexpectly affected by changes in test or irradiation conditions for other applications, by licensing requirements due to environmental impact or by change to reactor fuel design (e.g. conversion to LEU). Furthermore other events caused Tc^{99m} availability shortage in the past, among others, any supplier could be affected by impact of strike, restriction to shipment due to weather conditions or airline refusal to transport radioactive package, late obtaining of container agreement or shipment license and local transportation regulation (e.g. French prohibition of dangerous good transportation during long week end or end of the year period, constraints related to year 2000 or September 11 events in the US).

Reactor	Producer	Operation d/year	Production %		Capacity Mo^{99} production
			Mo^{99}	I^{131}	
NRU CAN	NORDION/AECL	± 280	40	-	70
HFR NL	COVIDIEN IRE	± 270	23 7	- 30	30 20
BR2 B	COVIDIEN IRE	± 115	5 4	- 20	15 20
OSIRIS F	IRE	± 190	4	15	20
SAFARI SA	NTP	± 305	14	35	30
OPAL AUS	ANSTO	(?)	-	-	15
OTHERS W	-	-	3	-	5
TOTAL (%)	-	-	100	100	225

Table 1: Tentative sharing of Mo^{99} and I^{131} fission radioisotope productions by reactor and producer in 2007
(Capacity Mo^{99} production corresponds to the irradiation of the maximum load of target for a full week).

Additional constraints result from the commercial aspect of the business, when customers adopting diversified procurement could order from a specific supplier very different quantities from week to week (e.g.: IRE at the end of 2006, had to process from 6 to 36 targets during successive weeks). This was at the time when Mallinckrodt US was forced to shutdown its generator production facility for several months imposed by radiopharmaceutical requirements enforced by the FDA.

Nuclear safety requirements, not only for transportation but also for processing of the targets or operation of the reactors might also lead to disruption of supply. This was in particular the case end of November 2007 when NRU reactor was not authorized to restart before having fulfilled specific licensing requirements. At that time the significant shortage was expected to late for more than one month and only a Canadian Parliament decision helped for a restart after 2 weeks just when the solidarity of the other producers succeeded to start providing a significant back up supply!

But the worst crisis developed by end of August 2008. At that time the scheduling of Reactor operation in Europe was such that after the definitive shutdown of FRJ-II in 2006 and the required maintenance shutdown of OSIRIS from end of June till 18th of September, BR2 was ending its summer cycle by 25 of August the day when the only reactor still available HFR was scheduled to restart for ensuring the continuity of irradiation. Unfortunately, on the 22 of August NRG officially indicated that HFR had to cancel the cycle, following the observation during the in service inspection of an unknown phenomenon: a very small intermittent stream of bubbles escaping from the wall of the primary cooling system. Later on, following a very impressive investigation it was foreseen to attempt performing a repair precluding the reactor restart before mid of February 2009. As a matter of fact the Authorities agreed to restart cautiously the reactor operation by end February while scheduling to have a few months shutdown for implementing a more definitive repair. Unfortunately at the same time an incident developed at IRE when an unforeseen chemical reaction took place during mixing of liquid waste streams, leading to the release over several days of about 1 Ci iodine 131 in the environment. Despite the fact that no health impact on workers and population could result from this incident, it was taken very seriously and the Licensing Authorities did not authorize the restart of IRE production before mid of November after a through full investigation of the causes and required preventive actions.
Therefore, beginning of September no reactor was available for both COVIDIEN and IRE. At the restart of OSIRIS by mid of September, IRE agreed to irradiate its targets and to have COVIDIEN processing them after the required transfer from one container to the other and adaptation of the target processing equipment, due to target design differences. In October COVIDIEN could use both IRE irradiation capacities in BR2 and OSIRIS for increasing its production, which minimized the shortage resulting from the IRE incident to the loss of only a few productions.
Although the impact of the crisis was reduced by a very significant contribution of NTP, boosting its production to a maximum for helping IRE to supply its customers in the frame of its consortium agreement and that NRU production was also significantly increased for providing IRE with some back up but also for Nordion contracting direct supply to some customers. During several months the world market was affected by significant shortage of Mo supply, less than 50 % in Europe and 80 % in ROW of the demand being satisfied during some weeks

3. Why such a crisis

The historical development of Mo^{99} production gives the root cause of the risk of such crisis despite several attempts to correct this development leading to the current situation.
In the 60th, Mo production started in America relying on a network of 2 private reactors in US and 2 public reactors in Canada.
By end of the 70th, IRE started Mo production in Europe by calling for irradiation services from test reactors available within a radius of 800 km (BR2, HFR, OSIRIS, ORPHEE, and HARWELL).
During the 80th, the 2 US reactors was definitively shutdown and the world production was relying only on NORDION (80 %) and IRE.
Beginning of the 90th, following political decision in Belgium, IRE sold its radiopharmaceutical business to Nordion and became mainly its subcontractor. Facing the risk of a monopoly situation Mallinckrodt decided to develop its Mo^{99} production facility in Petten and requested support from IRE which proposed to have a consortium with NTP (SA) for ensuring the best security of supply.
In the mean time one of the 2 Canadian reactors (NRX) was definitively shutdown as well as HARWELL and ORPHEE in Europe. Then NORDION succeeded to obtain from AECL the construction of the 2 MAPLE reactors which should have had by 2000, the capacity of producing twice the world demand. In front of this new threat of monopoly casted in Nordion contracts of 10 years exclusivity signed with most of big customers, IRE reacted by obtaining from European Commission and Japan Fair Trade Commission injunction for Nordion to

keep the market open. Progressively the Nordion share of market dropped from 80 % to 40 % and NTP became a major supplier together with IRE and Mallinckrodt (COVIDIEN)
At that time it was difficult for IRE to consider significant investment in a new reactor despite the fact that IRE collaborated to the investigation of different projects, namely with SIEMENS for a small dedicated classical reactor (MING), with B&W and TCI for a homogeneous reactor (MIPR) and with SCK/CEN-IBA for an Accelerator Driven System (ADONIS).
In the mean time IRE invested for having access to a 4th Reactor, FRJ-II in Jülich, but unexpectedly this reactor was definitively shutdown in 2006 and IRE had to investigate the feasibility of FRM-II in Munich for replacing it. In Canada the Maple reactors were never operational due to technical problems and the project is currently cancelled.
Therefore the greater risk of crisis is focused on the availability of 2 reactors NRU and HFR, when the other reactors also used by NTP (SAFARI), COVIDIEN (BR2) and IRE (BR2, OSIRIS) can only help to reduce the impact of the crisis. Definitively the world has been lucky up to now to succeed keeping IRE, NTP and COVIDIEN as significant producers without relying only on the Maple project.

4. Response to the crisis of 2008

4.1. On short time during the crisis, reactor operators considered the possibility to make last minute modification of their operating plans, and some of them succeeded to reduce outage time by postponing some works. COVIDIEN and IRE worked together for optimizing the use of the still available reactors and obtained a significant support of NTP.
The nuclear licensing authorities both from The Netherlands, Belgium and France contributed to manage the crisis consequences by issuing authorizations on short time notice both at the level of transportation, adaptation of process and restart of HFR.
The pharmaceutical authorities, working together with the nuclear medicine departments, optimized the use of Tc^{99m} giving priority to emergency cases (20 %), agreeing on an even split of the available generators to all hospitals, reducing the activity of each generator and optimizing its use and last but not least recommending alternative procedures (Tl^{201}, FDG) or other imaging techniques (MRI, CT, US, ….) even if quality and cost are affected.
AIPES (the Association of Imaging Producers and Equipment Suppliers) and its reactor and isotope coordination group, the Nuclear Safety Authorities from the world together with the Health Authorities and the NEA with the support of IAEA and EC, organized several meetings during the crisis for investigating possible short term solution to reduce the impact of the crisis and long term solution to reduce the risk of its occurrence.

4.2. On midterm, IRE and COVIDIEN are working today to increase the availability of the European reactor network, first by discussing with BR2 for installing 2 additional RIG's and possibly increasing the number of cycles. On the other side IRE has already, cost shared with TMU, a feasibility study for installing in FRM-II reactor an irradiation facility which could satisfy the requirements of its weekly current production when in operation. The results are quite promising as FRM-II provides high thermal flux and operates on very regular cycles of 60 days, 4 times per year. IRE also resumed discussions with REZ reactor for having additional irradiation support.
On its site, following on the incident, IRE decided to make important investments for upgrading the nuclear safety of its processing plant.
In Australia, ANSTO is currently starting again production of Mo99 by irradiating LEU targets in the new OPAL reactor and is expected to increase progressively its production for becoming within a few years a major supplier in the world.

4.3. On long term, IRE agreed with CEN/SCK and IBA to restart a new feasibility study for an improved ADONIS design with the expectation of having possibly this dedicated isotope production facility available within 5 years. Otherway the option of a small dedicated reactor (MING project) remains open.
IRE and COVIDIEN are also investigating the conditions for making use of JHR currently in construction in France and which should replace OSIRIS in 2014.

IRE and COVIDEN are following closely the projects of replacing in Petten HFR, with PALLAS a multi purpose reactor and in Mol; BR2, with MYRRHA, a large multi purpose ADS. In the US the adaptation of the MURR reactor and the installation of a Mo production facility have to be decided soon, and B&W is working again on the Homogeneous Reactor but with COVIDIEN this time.
In the rest of the world, different projects are also under consideration, in particular, in South Africa, Argentina, Korea, Japan, and Pakistan without forgetting China and the CRP sponsors by IAEA.

4.4. On very long term, i.e. 10 years and more, the major suppliers having secured a network of reliable reactors and processing facilities available for ensuring the security of supply of Mo^{99}, with significant over capacity the feasibility of conversion to use of LEU target might be considered without putting at risk the security of supply of Tc^{99m} . Today we have to state clearly that such option is still facing both significant technical and economical obstacles.
As a matter of fact the use of standard target design with LEU is feasible but will require irradiation and processing of five time more targets. Beside significant economical aspects this would require availability of 5 time more irradiation RIG's or reactors which is not foreseen with the current projects of extending the reactor network. Even having all producers adopting the CNEA dispersed LEU fuel target design, will be faced a similar obstacle as it requires still the availability of 3 time more irradiation RIG's and having the new process qualified and validated by all producers. Even if the ANL foil design would require only 10 % increase of number of targets to be irradiated it is a matter of fact that by to day, the design has not yet been proven acceptable for a large scale reliable industrial production which let CNEA and ANSTO adopting the dispersed fuel target design despite the capacity constraints involved. Within the next 10 years, the increase of the network of producers as encouraged by IAEA-CRP, might help, considering nevertheless that the Mo^{99} market size might have increased by 50 to 100 %. For small scale production, the use of the gel Mo^{98} (nγ) Mo^{99} proven process usually used in China as well as the possible validation of a new accelerator driven photo fission of U^{238} or possible positive results from feasibility studies for ADONIS with use of LEU targets as an option, might also contribute to improve the security of supply and reduce the use of HEU. Nevertheless we cannot today only relay on these technical options.

5. Conclusions

The current crisis resulting from the historical development of the Mo^{99} production industry is not really surprising and might have been more dramatic if on the basis of the MAPLE project only 1 or 2 producers might have survived.
The short time preventive actions, with in particular access to the FRM-II reactor as soon as possible should improve the reliability of supply. Whereas during the next few years the risk of a new crisis is real if the repair of HFR is facing problems or if NRU reactor has to be shutdown during several months for refurbishments and upgrading for getting extension of its operating license.
For mid term extending the reactor network or implementing successfully diversified options as ADONIS or MIPS might help but in any case all options will significantly increase the level of investments and the cost of productions which already today imply that the cost of Mo99 and Tc^{99m} generator production is significantly increased and has to be charged to Nuclear Medicines procedures.
The last crisis revealed the importance of the security of supply of Mo^{99} for the health of million of people in the world. This aspect has to be duly taken into consideration by both nuclear safety and drug control authorities, but this should never result as a threat to the safety of operations and workers or local population health.

QUALIFICATION PROGRAM FOR JHR FUEL ELEMENTS

M-C. ANSELMET[(1)], P. LEMOINE[(2)], D.IRACANE[(2)]
[(1)]CEA– Cadarache, 13108 St Paul lez Durance – Cedex – France
[(2)]CEA– Saclay, 91191 Gif sur Yvette – Cedex – France

E. KOONEN, P. BENOIT
SCK–CEN, Boeretang 200, B –2400 Mol - Belgique

I. CAILLIERE, P. COLOMB, T. PIN
AREVA-CERCA, les Bérauds, B.P. 1114, 26104 Romans – Cedex – France*

S. BRISSON, S. GUILLOT
AREVA–TA, 1100, av JR Guilibert de la Lauzière, 13593 Aix-en-Provence– France

ABSTRACT

The Jules Horowitz reactor (JHR) is the CEA new high performance material test reactor (MTR). Its startup is planned for the beginning of 2014. The reference fuel for the JHR is the UMo fuel as a high density, low enriched and reprocessable fuel, and CEA is deeply involved in the international collaboration for the development of this fuel. Moreover, the qualification and licensing of such a fuel is not expected to be ready for the startup of the JHR. Therefore, the CEA is qualifying a back-up solution, to ensure the first power operations of the reactor. This paper presents the status of the qualification program at the beginning of 2009, with a focus on manufacturing and qualification under irradiation.

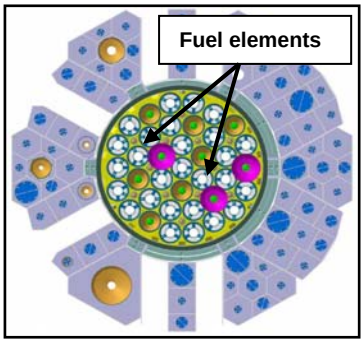
1. Introduction

The European material test reactors (MTR) are ageing and will reach more than 50 operating years in 2015. This situation cannot ensure the securing of experimental capability for the next decades. In this context and in the framework of an international partnership, the CEA has launched the JHR project in order to construct a new high performance MTR whose purpose will be to study material and fuel behaviour under irradiation with experimental capabilities relevant for different power reactor technologies and generation. It will also contribute to securing the production of radioisotopes for medical applications.

To meet these needs, the JHR has been designed for a maximum power core of 100MW with flexibility for operation at lower power levels in order to perform irradiations corresponding to with the demand. It will allow the performance of a significant number of simultaneous experiments in core and in reflector. Maximum performances are obtained at 100MW core operation with the reference core loaded with 34 fuels elements in a core rack with 37 cells (Fig 1).
Due to this performance level, the JHR requires a high density of fissile material and the reference fuel for the JHR is UMo fuel with uranium density of 8g/cm^3 and 20% ^{235}U enriched. It is for this reason that the CEA is deeply involved in the international collaboration on UMo fuel development. Moreover, UMo is not yet available as an industrial product qualified for JHR operation, and CEA is qualifying a back-up fuel solution for the first power operations of the JHR. This back-up fuel solution is U_3Si_2 particles dispersed into an aluminium matrix. UMo fuel dispersed into aluminium will replace U_3Si_2 as soon as it becomes available.

**AREVA-CERCA, a subsidiary of AREVA NP, an AREVA and SIEMENS Company*

Fig 1. Design of the core and reflector



2. The JHR fuel element

2.1 Design and characteristics

The three main parts of the element are (Fig 2):

- the lower and upper aluminium end pieces; they keep the element in position in its location in the core,
- the fuel assembly.

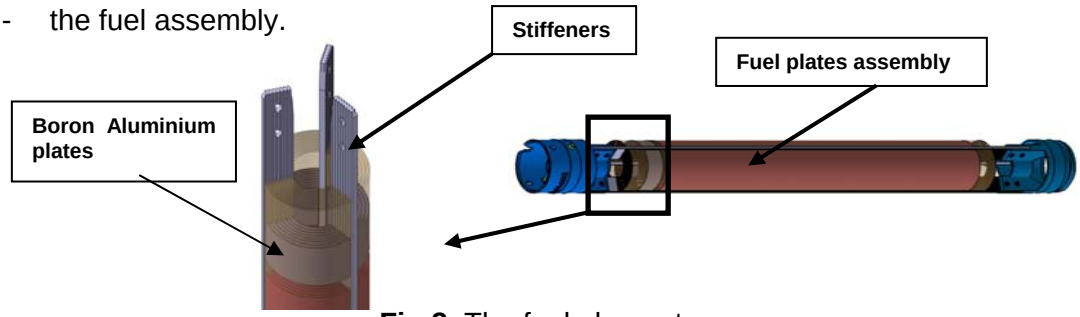
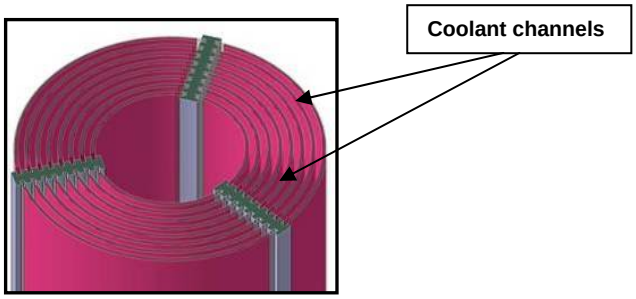


Fig 2. The fuel element

The fuel assembly consists of 8 concentric rings (Fig 3), each of them resulting from the coupling of three bended U_3Si_2/Al fuel plates crimped inside 3 aluminium stiffeners. For neutron consideration, small boron (10% mass) aluminium plates are set in the upper side of the plates (Fig 2). The cooling water channels defined by 2 rings are 1.95mm wide, and the velocity of the water through the channels is 15m/s. The fuel active length is 600mm. The inner and outer diameters of the element are 41mm and 96mm respectively; the inner hole has been designed to host an experimental device or a control rod.

Fig 3. Cross section of the plates assembly



The fuel meat is co-laminated with an aluminium cladding in order to obtain fuel plates (Fig 4.b); the fuel plate meat thickness is 0.61mm. The fuel plate cladding average thickness is 0.38mm; the cladding material is AlFeNi. The fuel meat is U_3Si_2 particles with MEUfuel uranium density of 4.8g/cm^3. These particles are dispersed in an aluminium matrix (Fig 4.b).

Fig 4: .a Fuel plate, .b Fuel meat

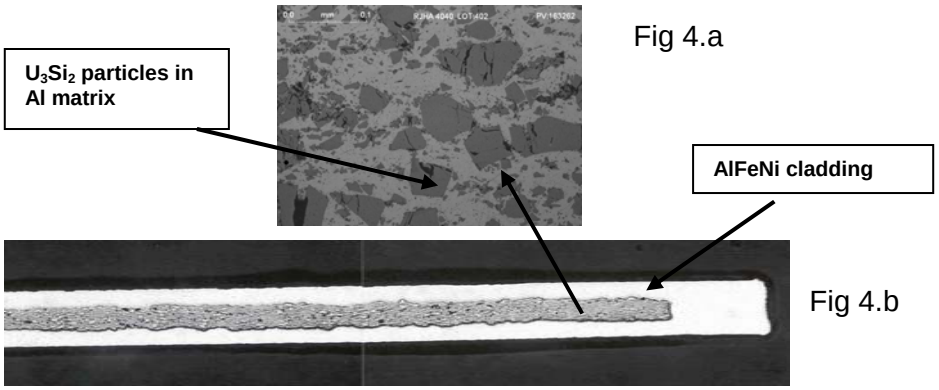


Fig 4.a

Fig 4.b

2.2 Fuel operating conditions; safety and functional requirements for the fuel

The reference core configuration is a core loaded with 34 fuel elements and a 100MW core operation. Table 1 gives fuel parameters and operating conditions :

Fuel type and ^{235}U enrichment	U_3Si_2 (first operation phase)
Clad material	AlFeNi
In core residence (cycle number x cycle length)(EFPD)	102,8 (4 X 25.7)
Hydraulic gap (mm)/ coolant velocity (m/s)	1.95 / 14.6
Average (1) / Max (2) / Peak (3) Heat flux (W/cm2)	152 / 420 / 516
Average (1) / Peak (3) wet temperature (°C)	65 / 165
Average (4) / Max(4) Burn-up (% ^{235}U)	51/53
Average (4) / (10^{21} f/cm^3)	2

(1) Core average; (2) local 3D; (3) All uncertainties included; (4) for discharged fuel elements, at End Of Life

Table 1: RJH fuel operating conditions

The general safety framework required for the JHR fuel underlies the objectives of the qualification program of the fuel element. More precisely stated, as the first discussions progressed on the JHR safety options, the ASN issued a certain number of fuel recommendations which led the CEA to set the following functional requirements for the fuel [1]:

Operating category	**Fuel functional requirements**
OC1- Normal conditions	Cladding integrity
OC2- Incidental conditions	Cladding integrity
OC3- Emergency conditions	Several fusion possible though no fusion
OC4- Faulty conditions	Fusion possible though limited

Table 2: Operating categories and associated functional requirements for the fuel

Based on these requirements, the fuel plates and elements are designed to guarantee three fundamental reactor safety functions:

- Confinement of radio-elements,
- Removal of decay heat,
- Control over reactivity

This led to a detailed description of the fuel functional requirements regarding the fuel and the plates, which need to be transposed into quantitative criteria for safety analysis in order to demonstrate that the safety objectives are being met (Table 3).

Table 3: Set of detailed fuel functional requirements and associated criteria

Operating category	Mechanical integrity (leaktightness)	Geometric integrity (local and general)	No fusion	No violent exothermal interaction
OC1, OC2	required	required	required	required
OC3			required	required
OC4				required
Criteria	*Experimental validation on full size elements*	*Local:Tmeat<515°C General: validation against buckling*	*Tcladding<616°C Tmeat<645°C*	*Tmeat<915°C*

To demonstrate that these functional requirements are observed for normal operating conditions, the CEA has developed an experimental program (see **3.3**).

Local geometric integrity of the plates

During a series of heat treatments- called blister tests- on irradiated U_3Si_2/Al plates involving several 30 minutes plateaux at increasing temperatures, blisters (several millimetres in diameter) tended to appear on the cladding [2]. These local blisters slightly pinched the hydraulic channel and thus may hindered decay heat removal. In compliance with the NRC report, the criterion thus defined was based on feedback from these tests an is express as T_{meat}< 515°C.

No melting

As in OC1 operating conditions, melting of the fuel plates is also prohibited in OC2 and OC3 operating conditions. The criteria are defined:

- by the solidus/liquidus temperature of AlFeNi for the cladding (616°C)
- and the solidus/liquidus temperature of the matrix Al (A5) (645°C)

Calculations of the temperatures evolutions in the matrix and in the hottest point in the cladding the more penalizing transient (among the OC2 situations and OC3 situations respectively) must show that these criteria are respected.

No violent exothermal reactions

This type of reaction may only have consequences when its kinetics becomes fast and creates more thermal flux than the coolant can extract. Based on this consideration, the NRC report [2] gives the following information regarding the interaction between particles (U_3Si_2) and matrix (Al):

- the kinetics of the interaction becomes significant only when Aluminium is molten,
- the heat produces by the interaction is progressively consumed by Aluminium melting,
- no other thermal phenomena is observed in the upper temperature domain (from aluminium melting temperature up to 1300°C)

Furthermore, many works have characterized the kinetics of the interaction between Al and water, and it can be concluded [3] that for T<915°C, there is no violent reaction.

The overall criterion of T_{meat}<915°C shall therefore be applied as a conservative one for the all above mentioned reactions.

3. The qualification program for the fuel elements

The fuel element is one of the major parts of the project because it has to ensure the performances required for reactor operation and to guarantee the functional requirements.

At the present time, elements with U_3Si_2 fuel are widely used in MTRs [2], but their behaviour has been qualified for a range of operating conditions which do not reach the high level of the specified JHR ones. Furthermore, those high level performances led to special design for the fuel element with very tight constraints on the geometry and the manufacturing margins. This position justifies the qualification of both design and behaviour of the different

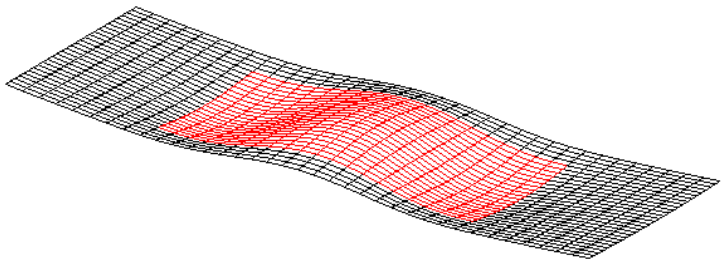
components of the fuel elements under reactor operating conditions. Therefore, the qualification program is aiming to meet the following objectives [4]:

- The validation of the fuel element design options to prove the required performances
- The qualification of the manufacturing routes for plates fabrication and assembly
- The qualification of the behaviour of the fuel element including hydraulic tests under irradiation in the JHR operating domain.
-

For OC1 operating conditions, the validation is led through an experimental demonstration (**3.3)**.

In OC2 operating conditions, the validation of the global geometric stability of the plates is checked through an analysis of the behaviour regarding the buckling. This methodology has been validated (fig 5)on the results of experimental tests performed on U3Si2 plane plates ([5], [6], [7]) then transposed to JHR plates. It shows that in OC2 and OC1 operating conditions, there is no buckling of the plates in a range of

Fig 5 : Results of an analysis on plane plates



For OC3 and OC4, the validation is based on thermal analyses taking into account the more penalizing transients (reactivity insertion, flow-rate reduction) among the relevant ones for theses categories of operating conditions.

3.1 Validation of the fuel element design

The objective is to demonstrate that the fuel element (shape and dimensions) is able to carry out the assigned functions (neutron flux supply, safety, interactions with other part of the reactor) and to insure all required performances are met for the whole duration of its location in-core.

Validations have been previously carried out on the neutron and thermal-hydraulic points of view and shows that the core performances meet the required objectives. The aim of this paper is focused on the mechanical part of the validation.

The final selection for the fuel and the preliminary design of the assembly have been validated at the end of the design phase. It is then required to validate that the design options which have been chosen (material, geometry,..) will lead fuel elements which will:

- respect the criteria associated to the different operating categories (see **2**),
- guarantee the required performances for the core, through their individual contribution to neutron flux supply,
- generate no disturbances on other reactor components induced by a damaging behaviour caused by the different thermal and mechanical constraints they undergo.

The validation of the fuel element design is performed by a set of theoretical demonstrations. This analysis is carried out using the same methodology for each type of operating conditions.

For the thermal-mechanical analysis, the study considers the JHR fuel element as an entire component with its different parts (upper end lower end pieces, fuel assembly) and their different joining technologies; it does not take into account the design and the qualification of the behaviour of the fuel plate, which is carried out elsewhere through in the program (**3.3).** The required output results (Fig 6) for each part of the element are:

- the definition of the behaviour (phenomena),
- the calculation of the constraints,
- the contribution of the validation of the choices for material,
- and the validation of the geometry.

It has to be demonstrated that the mechanical behaviour of the whole element is acceptable for each category of operating conditions, i.e. to the evolution of the significant parameters influencing the thermal and mechanical behaviour of the element, ensure the checking of the criteria.

The mechanical analysis takes into account the thermal-mechanical constraints which are undergone by the element for OC1, OC2 and OC3 operating conditions. For OC4 operating conditions, the respect of the criteria is check through thermal and thermal-hydraulic analyses which are the relevant ones for this type of conditions.

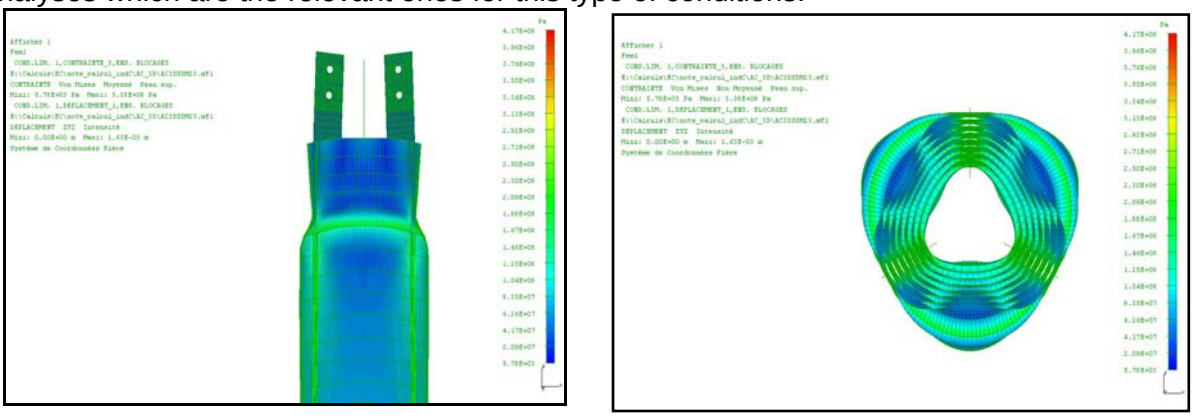


Fig 6: Example of results available from calculation of the mechanical behaviour of the fuel element (AREVA-TA)

.The final outcome of this program will consist of a general validation (**3.3**) of this design will be carried out on prototypes through an experimental program design to comprehend to the fuel behaviour under representative operating conditions (regarding flux and thermal-hydraulics), by verifying that changes occurring in the element are satisfactory with regard to the various criteria (evolution of the geometry, mechanics,…).

3.2 Qualification of the manufacturing

The qualification of the manufacturing must guarantee the capability:

- to complete the manufacturing of the elements with respect to the specifications to verify that an industrial rate for the manufacturing corresponding to the needs in terms of RJH core reloading can be ensure (about 80 elements/year),
- to deliver the first core and the following reloading s within the required times.

The manufacturing is performed for the CEA by AREVA-CERCA [8] which was early involved in the project for fuel design.

First of all, preliminary demonstrations have been carried out to validate the feasibility of the main stages of the manufacturing on U_3Si_2/Al plates. A very tight constrain is assigned on the depth of the coolant channels between the plates. As it was not in the usual range of the manufacturing parameters for CERCA, a test has been performed on an aluminium mock-up to prove the feasibility of crimping plates on 3 stiffeners at 120° for a reduced clearance.

Furthermore, the feasibility of the bending of the plates for the smallest and largest sizes (1 and 8 respectively) has been demonstrated with. It led to development of specific tools for the JHR plates manufacturing.
At the end of this preliminary demonstration phase, a preliminary version of the manufacturing was drawn up and sent to the manufacturer. It allowed us to start the following phase which is aimed at procuring equipment specific to manufacturing of the JHR plates and assembly, fixing the final tools, the accurate set of material and the final range for manufacturing parameters, and performing the pr-industrial qualification on a set of about 230 plates. This phase led to previous specifications for manufacturing and control to be applied to the manufacturing of 12 JHR prototypes elements.

The manufacturing is defined and supervised by the CEA, and validated for each key point with respect to the specifications:

- U_3Si_2/Al powder fabrication,
- Realisation of the plates,
- Bending of the plates (Fig 7),
- Assembly of the elements (fuel plates en end pieces)

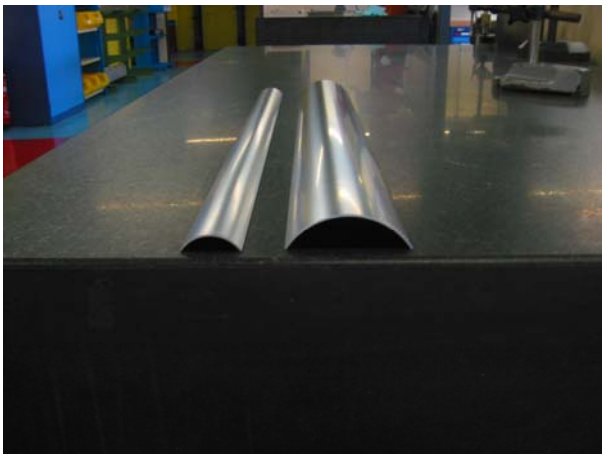


Fig 7: Bended plates size 1 and 8 (AREVA-CERCA)

Twelve elements (LTA; Lead Test Assemblies) have to be manufactured. Five among them are devoted to a qualification program under irradiation flux and operating conditions representative of the JHR normal operating domain (EVITA, see **3.3 ,**[9]). The other ones will be tested in the EOLE reactor (CEA, Cadarache) for neutron database acquisition**.**

The first fuel of these twelve elements has been delivered to BR2 reactor in January 2009, 4 last ones will be inspection for acceptance in March, then the seven last ones in June.

3.3 Qualification of the fuel behaviour in normal operating conditions

Behaviour under representative hydraulic conditions (AREVA-TA)

Hydro-mechanical tests have been conducted in the BACCARA loop (CEA), with a full-scale aluminium mock-up, in order to characterize the mechanical behaviour of the JHR element in representative flow conditions. The tests have delivered useful information and measurements on physical parameters for the modelling and the consolidation of the preliminary concept calculations. Tests showed no mechanical degradation of the plates or the structure of the element due to hydraulic or vibratory phenomena. Moreover, metrologies have been performed on the mock-up before and after the test campaigns. They indicate that the evolutions of the external diameter of the element (mock-up) are not significant in comparison to the manufacturing scattering and to the evolution expected after irradiation.

Behaviour under conditions representative of JHR normal operating conditions (thermal hydraulic and neutron flux)

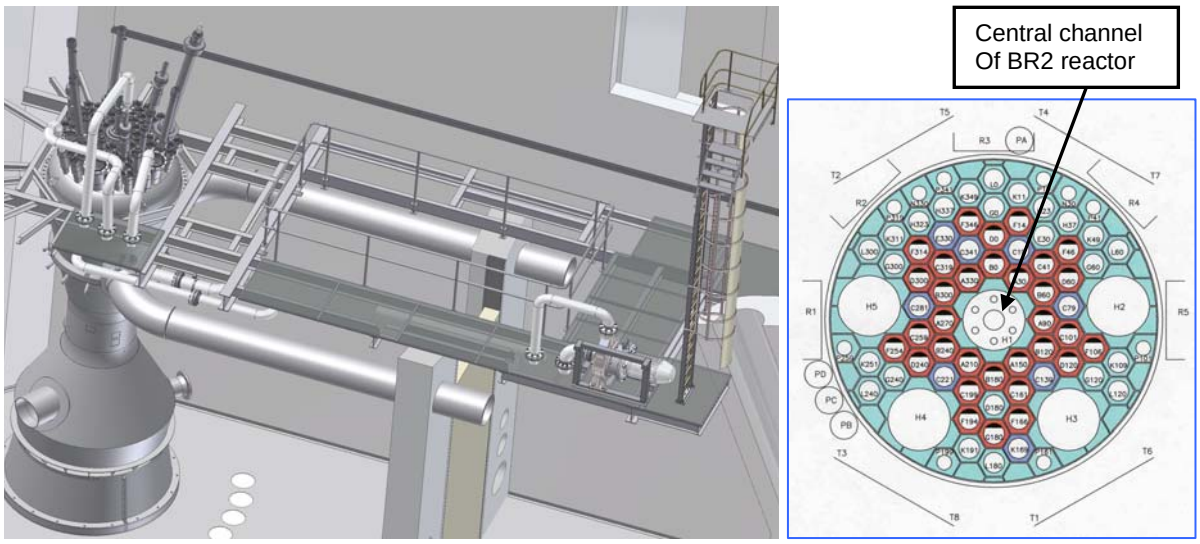
A preliminary test has been carried out in the BR2 reactor in 2005, to validate the behaviour of representative U_3Si_2 plates under conditions similar the JHR ones. It consists of a "mixed" BR2 fuel element, with its 5 inner usual rings (UAlx fuel and AG3N cladding) and an external ring representative of JHR fuel. The plates were tested during 3 cycles (21 EFPD), with a maximum power density of 435W/cm3 and BR2 cooling conditions (12m/s). The mean fuel burn-up in U_3Si_2 plates at the end of the experiment was 54%. Examinations carried out after irradiation allowed to conclude that the silicide fuel plates behaved correctly under the experimental conditions.

In order to complete this previous demonstration, it was decided to qualify the correct behaviour of the fuel elements under JHR normal operating conditions (OC1) by means of an experimental in-pile program. It will check that:

- the mechanical integrity of the fuel plates,
- the geometric integrity of the fuel plates and more generally of the whole assembly

are preserved in these conditions (in particular limited deformation under flux of the plates and of the coolant channels is expected). For this purpose, a dedicated loop has been designed to be implemented in the BR2 reactor of the SCK-CEN ([9], fig 8).

Fig 8: Artistic view of the EVITA loop in the BR2 reactor and location for JHR element during irradiation (SCK-CEN)



The in-core of the part of the loop is located in the central channel of the BR2 core (H1 channel). Adaptations have been done to host a JHR element for irradiation campaign at the specified irradiation conditions.

The program plan is the following:

- previous examinations at fuel element reception at BR2 (geometrical characterization) in order to obtain a dimension initial state,
- irradiation cycles under specified conditions with inter-cycles examinations of the plates (visual and fission product release tests),
- post-irradiation examinations (EPI) whose objective is the characterisation of the final state of the plates (swelling, corrosion of the cladding, levels of deformation).

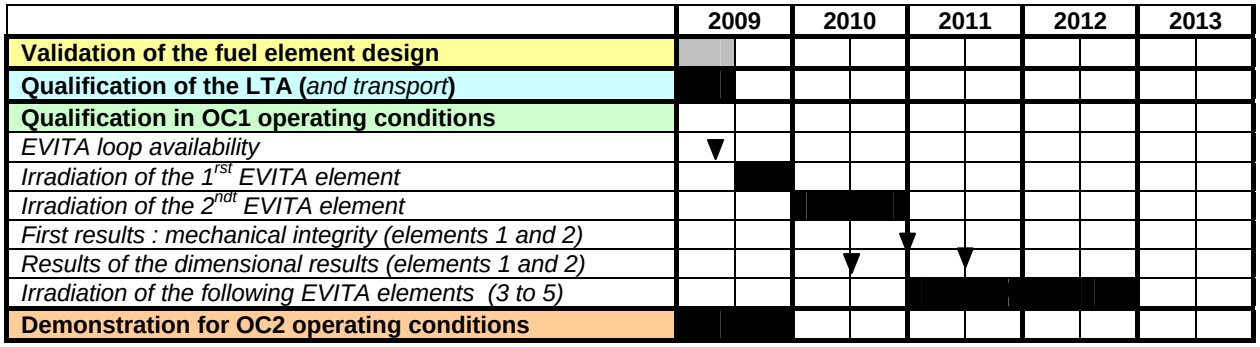
It will also allow to access to the thermal-hydraulic and mechanical behaviour of the element under normal operation and to consolidate some of the previous answers given by the test performed in the BACCARA loop.

Five elements will be available for successive tests. The five tests will aim at applying the JHR nominal operating conditions gradually (maximum surface density and cooling rate) and will achieve the target burn-up fraction, beginning with an initial test with an intermediate burn-up objective.

The leaktightness of the plates can be checked without recourse to destructive testing, so the information might be available in a short delay after the end of irradiation. The geometric integrity (plates and channels) of the assemblies will be verify by mean of three dimensional measurements, which should also give access to the comparison between initial state (after manufacturing) and final state (after irradiation). Then metallographic inspections on samples taken in the most stresses areas will give information on the behaviour of the uranium silicide particles.

Schedule for the availability of the loop and the details of the program can be found in **5**.

4. Schedule for the JHR fuel elements qualification



5. Conclusion

The CEA has launched a qualification program to demonstrate that the back-up fuel element will ensure the fundamental safety functions and the required performances for the core operation.

At the time being:

- theoretical studies are on going to validate the final fuel element design regarding its mechanical behaviour in the different category of operating conditions,
- the manufacturing of twelve JHR prototypes must demonstrate the capability to complete the manufacturing with respect to the specifications with the final acceptance of the last elements in June 2009.
- the demonstration of the correct behaviour of the fuel assembly under conditions representative of the JHR nominal ones will start in July with the first irradiation in the EVITA loop.

6. References

[1] – MC. Anselmet, G. Bignan, D. Iracane, P. Lemoine, B. Maugard, P. Tremodeux
Jules Horowitz Reactor Project: Status and fuel licensing project, this meeting
[2] -US Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. July 1988
Safety evaluation report related to the evaluation of low enriched uranium silicide-aluminium dispersion for use in no-power reactors, NUREG 1313.
[3] – H.M. Higgins, The reaction of molten Uranium and Zirconium alloys with water, AGC/AE/17, 1956
[4] - P. Lemoine, French experience in the scope of silicide fuel qualification for material testing reactors in “IAEA Good Practices for Qualification of Research reactor Fuels”
[5] K. Yanagisawa, T. Fujishiro, O. Horiki, K. Soyama, H. Ichikawa, T. Kodaira, Dimensional stability of low enriched uranium silicide plate-type fuel for research reactors at transient conditions, J. of Nuclear Science and Techno., 29 (3), pp 233-243, **1992**
[6] K. Yanagisawa, T. Fujishiro, O. Horiki, K. Soyama, H. Ichikawa, T. Kodaira, Transient behavior of low enriched uranium silicide plate-type fuel for research reactor during reactivity initiated accident conditions, J. of Nuclear Science and Techno., 30 (8), pp 741-751, **1993**
[7] K. Yanagisawa, T. Fujishiro, K. Soyama, H. Ichikawa, T. Kodaira, Transient behavior of silicide plate-type during reactivity initiated accident conditions, Jaeri-M 94-042, **1994**
[8] - I. Cailliere, P. Colomb, MC. Anselmet, P. Lemoine, S. Brisson, S. Guillot
LTA’s manufacturing for JHR fuel qualification program, this meeting
[9] – E. Koonen, P. Gouat, P. Benoit, SCK-CEN, EVITA : a semi-loop in BR2 for JHR Fuel Qualification, Session V, this meeting

INTERNATIONAL STATUS OF ACCELERATOR DRIVEN SYSTEMS FOR ADVANCED NUCLEAR FUEL CYCLES

A. STANCULESCU
Department of Nuclear Energy, International Atomic Energy Agency
Wagramerstrasse 5, 1400 Vienna – Austria

ABSTRACT

For nuclear energy to remain a long term option in the world's energy mix, nuclear power technology development must meet sustainability goals with regard to fissile resources and waste management. The utilization of breeding to secure long-term fuel supply remains the ultimate goal of fast reactor development. Plutonium recycle in fast reactors, as well as incineration/transmutation of minor actinides and long-lived fission products in hybrid reactor systems (e.g. accelerator driven systems) also offer promising waste management options. Several R&D programmes in various IAEA Member States are actively pursuing these options, along with the energy production and breeding mission of fast reactor systems. The paper presents an overview of the major national and international R&D programs in the area of accelerator driven systems for transmutation of long-lived nuclear waste. It also describes IAEA's ongoing and planned activities in this field.

1. Introduction

Based on a cumulative experience of almost 13'000 reactor-years, nuclear power is a mature technology that makes a large contribution to the energy supply worldwide. As of March 2009, there were 436 nuclear power plants operating in the world with a total net installed electrical capacity of 370 GW supplying slightly more than 15% of the world's electricity, and 44 nuclear power plants of 38 GW electrical capacity under construction [1]. In 2007, the nuclear share in electricity generation ranged from maxima of 76.9%, 64.4%, and 54.3% in France, Lithuania, and Slovakia, respectively, to minima of 2.8%, 2.5%, 2.3%, and 1.9% in Brazil, India, Pakistan, and China, respectively [1].

There are four major challenges facing the long-term development of nuclear energy as a part of the world's energy mix: improvement of the economic competitiveness, meeting increasingly stringent safety requirements, adhering to the criteria of sustainable development, and public acceptability. Meeting the sustainability criteria with regard to waste management is the driving force behind the topic of this paper. While not involving the large quantities of gaseous products and toxic solid wastes associated with fossil fuels, radioactive waste disposal is also today's dominant public acceptance issue. One of the primary reasons that are cited is the long life of many of the radioisotopes generated from fission. This concern has led to increased R&D efforts to develop a technology aimed at reducing the amount of long-lived radioactive waste through transmutation in fission reactors or accelerator driven systems (ADS). In recent years, in various countries and at an international level, more and more studies have been carried out on advanced and innovative waste management strategies (i.e. actinide separation and utilization/elimination).

This paper summarizes the major programs worldwide, as well as IAEA's activities on utilization of plutonium and transmutation of long-lived radioactive waste implemented within the framework of the "Project on Technology Advances in Fast Reactors and Accelerator Driven Systems" [2].

2. Advanced nuclear fuel cycles

The rationale for the research and technology development programs in the field of ADS must be seen in the context of the efforts to establish innovative nuclear fuel cycles including partitioning and transmutation (P&T).
Presently, industrial maturity can be claimed for two fuel cycle strategies, i.e. the "Once Through Fuel Cycle" (OTC), and the "Reprocessing Fuel Cycle" (RFC) in which plutonium and very limited uranium quantities are being recycled. It is helpful to recall some key data that set the stage for any fuel cycle (waste management) discussion: worldwide, the annual spent fuel discharge is in the range of 10'500 – 11'000 t heavy-metal (HM), while the industrial reprocessing capacity amounts to ~5'000 t HM [3]. Hence, less than ½ of the discharged spent fuel can be processed. Worldwide, the cumulative inventory of stored spent fuel is estimated to be ~130'000 t HM, and the amount of reprocessed spent fuel is estimated to be ~70'000 t HM. The latter inventory has been transformed into high-level waste (HLW) and spent light water reactor (LWR) mixed uranium-plutonium oxide (MOX) fuel. Considering the relatively low uranium ore prices, this situation is expected to continue over the next few decades (the cumulative inventory of stored spent fuel could reach 192'000 t HM by 2010). Therefore, it is likely that the need for repository space will increase accordingly. Taking Yucca Mountain (63'000 t HM capacity) as reference repository, the present worldwide inventories would require two repositories for the spent fuel, and one for the HLW. For the USA alone (OTC strategy), assuming a life time extension of the present nuclear reactors to 60 years, and no new reactors, the capacity of Yucca Mountain will be exceeded by ~2050. Given the strong public opposition to the construction of geologic repositories, it is understandable that over the last decade or so, in various countries and at an international level, more and more studies have been carried out on advanced and innovative waste management strategies aiming at reducing the amount of long-lived radioactive waste through transmutation in fission reactors or ADS. In several IAEA Member States, P&T is being revisited with the goal of reassessing its merits and investigating new approaches that could be followed in implementing this innovative fuel cycle and waste management option.

3. Status of ADS research and technology development

P&T is a complex technology that implies the availability of advanced reprocessing plants, facilities for fuel fabrication of transuranics (TRUs), and irradiation facilities beyond the presently existing nuclear reactors. For the major part, *partitioning* consists in extending the current reprocessing techniques: in addition to uranium, plutonium and ^{129}I, also minor actinides (neptunium, americium and curium), and, possibly, also long-lived fission products (^{99}Tc, ^{93}Zr, ^{135}Cs, ^{107}Pd and ^{79}Se) would be extracted from the liquid high level waste. This technology, could, to a certain extend, be extrapolated on the basis of decades-long industrial experience in Europe and Japan. *Transmutation*, on the other hand, requires fully new fuel fabrication plants and irradiation technologies and their implementation on an industrial scale. The use of existing nuclear reactors as transmutation devices results in modest incineration and transmutation yields, and, more importantly, is limited by both safety and operational considerations. Therefore, new concepts, i.e. dedicated fast reactors, and sub-critical systems (ADS and even fusion/fission hybrids) have been proposed as incineration/transmutation devices.
ADS rely on the availability of a hard neutron source produced by the spallation process induced by a high-energy proton beam impinging on a heavy nuclide target. While ADS present an attractive potential, various specific economics and technological issues remain to be solved (e.g. capital costs, additional energy consumption, accelerator reliability, various physics and material science issues linked, e.g. to nuclear data, fuel performance, heavy liquid thermal hydraulics, etc).
The assessment of the potential of the ADS technology must consider the fact that the spallation neutron source is much less effective than the fission one. Basic physics considerations comparing spallation and fission neutron sources lead to the conclusion that transforming the energy of one fission event (which yields roughly 200 MeV of energy and 3 hard neutrons) into spallation results in approximately 20 MeV of energy and, in an optimum

situation, 1.5 hard neutrons. In other words, compared to fission, the spallation source requires 180 MeV energy to produce half the number of neutrons. It is obvious that the introduction of ADS, which are depending on such an expensive neutron source, can be justified only with conspicuous advantages in other areas. These advantages, always compared to critical reactors, are claimed in two areas: firstly, the improvement of the dynamics behaviour (and hence safety characteristics), and, secondly, the enhanced flexibility linked to an improved neutron balance gained from the availability of an external neutron source. Both these advantages motivate R&D efforts aiming at substantiating the potential of ADS and studying their role in innovative reactor and fuel cycle strategies that include systems for large-scale utilization and transmutation of actinides and LLFP. The major ongoing programmes in Europe and Asia are summarized below.

3.1 European Union

In the European Union, research and technology development in the area of ADS is part of European Atomic Energy Community's 6^{th} (2002 – 2006) and 7^{th} (2007 – 2011) Framework Programmes (FPs) [4,5]. The objective of these activities is to investigate ways and means of reducing the amount and/or the hazard of the nuclear waste. The activities include research and technology development in partitioning and transmutation with the expressed goal of developing pilot facilities. In FP6 (to be continued in FP7), the focal point for ADS research and technology development was the EUROTRANS project (funded with a total of 45 million Euro, of which the EC contributed 23 million). EUROTRANS had two main objectives: firstly, perform preliminary design studies of experimental ADS (XT-ADS) in the thermal power range 50 – 100 MW, and, secondly, perform the conceptual design of a modular European Transmutation Demonstrator (ETD) in the thermal power range of several hundred MW. The mission of the XT-ADS is to demonstrate the feasibility of transmutation with ADS. Both the spallation target material and the coolant are lead-bismuth eutectic. The subcritical core is fuelled with mixed uranium-plutonium oxide, and the accelerator is a 600-MeV∗2.5 mA or 350 MeV∗5 mA LINAC. The Belgian SCK•CEN Research Centre at Mol has aligned its R&D activities aiming at the design of the full scale ADS demonstrator MYRRHA with the XT-ADS efforts. All major possible design cliff edges are being (will be in FP7) investigated, the key accelerator components demonstrated, the spallation target thermal hydraulics design completed, and, last but not least, the experimental coupling of a sub-critical core with a neutron source realised. The latter (the GUINEVERE experiment) will also provide an experimental facility allowing physics experiments and technological research under conditions representative for XT-ADS. GUINEVERE, currently being erected at SCK•CEN Mol, stands for "Generator of Uninterrupted Intense NEutrons at the lead VEnus Reactor". This facility couples a deuteron GENEPI-3C accelerator functioning in continuous, pulsed, and beam trip modes with a Ti^3H target producing 14.1 MeV neutrons (deuterium-tritium target), surrounded by a fast sub-critical metallic uranium fuelled core in a lead matrix.

3.2 Belarus

The Joint Institute for Power and Nuclear Research – SOSNY (JIPNR-SOSNY) of the National Academy of Sciences of Belarus is implementing the experimental program YALINA with the objective of studying the physics of ADS and investigating the feasibility of minor actinides and long-lived fission product transmutation in fast spectrum sub-critical facilities [6]. The research activities pursued by the YALINA group include the development of experimental techniques for monitoring sub-criticality levels (currently, the YALINA-BOOSTER configuration is the only installation where reactivity monitoring techniques used in a power ADS can be tested), the investigation of spatial kinetics of sub-critical systems driven by external neutron sources, the measurement of transmutation reaction rates, and investigations of maintenance and operation characteristics of sub-critical systems driven by external neutron sources. The YALINA experimental programme contributes, on the one hand side, to the EUROTRANS project, and, on the other, aims at converting the HEU loaded fuel zones of the sub-critical YALINA-BOOSTER configuration to LEU loaded zones without penalizing its functionality [7].

3.3 India

In addition to the rationale for ADS development mentioned in Section 2, India is pursuing an ADS programme in view of its potential to enhance breeding of fissile material in conjunction with the utilization of thorium fuel in nuclear power systems [8]. The programme includes research in the areas of high power proton accelerator technology, spallation target systems, heavy liquid metal thermal hydraulics, and sub-critical core design. In the field of accelerator technology, the Indian research programme addresses the challenges at both the low (due to high currents) and high (superconducting RF cavities) energy end. R&D related to spallation target systems includes thermal hydraulics experiments on a lead-bismuth eutectic loop with proton heating of the beam window simulated by a plasma torch and electron beam. Extensive validation and qualification of Computational Fluid Dynamics codes is being done based on these experiments. This loop is also being used for corrosion testing. The core design studies look at various sub-critical systems in view of thorium fuel utilization. The Indian programme includes also an experimental programme coupling a deuterium-tritium neutron source with a sub-critical water cooled, natural uranium fuelled reactor. Design studies for an experimental reactor that would offer the flexibility of being transformed into a sub-critical system driven by a spallation source are also underway.

3.4 Japan

R&D on ADS is pursued by the Japan Atomic Energy Agency (JAEA) with the objective of transmuting long-lived radioactive nuclides [9]. JAEA proposes an 800 MWth lead-bismuth eutectic cooled ADS. As in the case of India, various activities are conducted to investigate the feasibility of the ADS from the viewpoints of accelerator technology, lead-bismuth technology, and sub-critical core design. Within the framework of the J-PARC (Japan Proton Accelerator Research Complex) project, JAEA is pursuing the TEF (Transmutation Experimental Facility) design studies. Specifically, handling of minor actinides bearing fuel studies were performed, and a remote handling system was designed for the transportation of the fuel assemblies and for the core storage. Corrosion tests for various structural material candidates were performed in low oxygen concentration condition at 450°C and at 550°C. The sub-critical core design studies lead to a four-zone core concept, which was adopted to limit operating temperatures below 500°C and thus improve the compatibility with the lead-bismuth eutectic.

4. IAEA activities

IAEA activities include information exchange and collaborative R&D. Recently, the IAEA, in collaboration with the International Centre for Theoretical Physics (ICTP), convened the “Advanced Workshop on Model Codes for Spallation Reactions” [10], and the “Workshop on Nuclear Reaction Data for Advanced Reactor Technologies” [11]. Looking ahead, IAEA’s Department of Nuclear Energy and Department of Nuclear Sciences and Applications are organizing in collaboration with ICTP the School on “Physics, Technology and Applications of Innovative Fast Neutron Systems” in Trieste, from 9 to 20 November 2009. Two major international conferences related to the scope of this paper are organized by the IAEA in 2009: the “Topical Meeting on Nuclear Research Applications and Utilization of Accelerators, AccApp ’09”, organized in collaboration with the ANS by IAEA’s Department of Nuclear Energy and Department of Nuclear Sciences and Applications (Vienna, 4-8 May 2009) [12]; and the conference on “Fast Reactors and Associated Fuel Cycle – Challenges and Opportunities”, organized by IAEA’s Department of Nuclear Energy in cooperation with various Japanese and international organizations, and hosted by JAEA (Kyoto, 7-11 December 2009 [13].

With regard to collaborative R&D, the IAEA has completed the Coordinated Research Project (CRP) on “Studies of Advanced Reactor Technology Options for Effective Incineration of Radioactive Waste” [14], and has an ongoing CRP (2005-2009) on “Analytical and Experimental Benchmark Analyses of Accelerator Driven Systems (ADS)” [15].

Last but not least, the IAEA is maintaining the “ADS Research and Development Database”. It provides information about ADS related R&D programs, existing and planned experimental facilities as well as programs, methods and data development efforts, design studies, and so forth. While operational on the Web ((http://www-adsdb.iaea.org/index.cfm), the database must

rely on content contributed by the interested community. Data can be provided on-line, and contributions are solicited (the author will gladly grant access privileges upon request).

5. References

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Power Reactors Information System (PRIS) Database, http://www.iaea.org/programmes/a2/index.html

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Project on Technology Advances in Fast Reactors and Accelerator Driven Systems, http://www.iaea.org/inisnkm/nkm/aws/fnss/index.html

[3] OECD NUCLEAR ENERGY AGENCY, Accelerator-driven Systems (ADS) and Fast Reactors (FR) in Advanced Nuclear Fuel Cycles: a Comparative Study, Paris, 2002

[4] V.P. Bhatnagar, G. Van Goethem, Overview of EU Activities in Partitioning and Transmutation Research in the EURATOM 6th and 7th Framework Programmes, OECD/NEA Actinide and Fission Product Partitioning and Transmutation Ninth Information Exchange Meeting, Nîmes, France, 25-29 September 2006

[5] V.P. Bhatnagar, Overview of EU Research Activities in Transmutation and Innovative Reactor Systems within the EURATOM Framework Programmes, submitted to the International Topical Meeting on Nuclear Research Applications and Utilization of Accelerators, AccApp '09, Vienna, 4-8 May 2009

[6] A. Kiyavitskaya, V. Burnos, et al., Application of High Intensity Neutron Generator for the Investigations of ADS Neutronics, submitted to the International Topical Meeting on Nuclear Research Applications and Utilization of Accelerators, AccApp '09, Vienna, 4-8 May 2009

[7] Y. Gohar, I. Bolshinsky, et al., YALINA Booster Conversion Project, submitted to the International Topical Meeting on Nuclear Research Applications and Utilization of Accelerators, AccApp '09, Vienna, 4-8 May 2009

[8] P.K. Nema, P. Singh, et al., ADS Programme and Associated Developments in India; Roadmap and Status of Activities, submitted to the International Topical Meeting on Nuclear Research Applications and Utilization of Accelerators, AccApp '09, Vienna, 4-8 May 2009

[9] H. Takei, N. Ouchi, et al., Research and Development Programme on ADS in JAEA, submitted to the International Topical Meeting on Nuclear Research Applications and Utilization of Accelerators, AccApp '09, Vienna, 4-8 May 2009

[10] Joint ICTP-IAEA Advanced Workshop on Model Codes for Spallation Reactions, International Nuclear Data Committee Report, INDC(NDS)-0530, August 2008

[11] INTERNATIONAL CENTRE FOR THEORETICAL PHYSICS, Trieste, Italy, 19–30 May 2008, Lecture notes on http://cdsagenda5.ictp.trieste.it/full_display.php?smr=0&ida=a07153

[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Meetings Web Page http://www-pub.iaea.org/MTCD/Meetings/Announcements.asp?ConfID=173

[13] INTERNATIONAL ATOMIC ENERGY AGENCY, Meetings Web Page http://www-pub.iaea.org/MTCD/Meetings/Announcements.asp?ConfID=35426, and the conference Web Page at http://www.FR09.org

[14] W. Maschek, A. Stanculescu, et al., The IAEA CRP on "Studies of Advanced Reactor Technology Options for Effective Incineration of Radioactive Waste", GLOBAL 2007 Advanced Nuclear Fuel Cycle and Systems, Boise, Idaho, 9-13 September 2007, 1155

[15] A. Abánades, G. Aliberti, et al., IAEA Coordinated Research Project (CRP) on "Analytical and experimental benchmark analyses of accelerator driven systems". PHYSOR 2008, International Conference on the Physics of Reactors "Nuclear Power: A Sustainable Resource", Interlaken, Switzerland, 14–19 September 2008, FP305.pdf

European Nuclear Society

Rue Belliard 65
1040 Brussels
Belgium
Telephone +32 2 505 30 54
Fax + 32 2 502 39 02

rrfm2009@euronuclear.org

www.euronuclear.org