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Session I

International topics and overview on new projects and fuel developments

Making the Nuclear Renaissance a Certainty

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In the following AREVA's views and experience with the intensively discussed renaissance on the market for nuclear power for electricity generation will be briefly presented. For AREVA this renaissance started earlier than for anyone else by being awarded with the first contract for a nuclear power plant project of the new generation. Today we have two plants under construction, two more have been ordered and many more are under discussion.

But what are the drivers for this renaissance of nuclear power and what are the real challenges linked to it? It is expected that the world energy demand will grow for 15 000 TWh to 30 000 TWh by 2030. This is caused:

1. by an increasing demand of energy, due to several factors :

- Demography : there will be 2 billion more people on earth by 2030.
- The legitimate economic growth in fast developing countries, such as China, South Africa, Brazil, India etc.
- Growth in developed countries: in spite of improvements in energy efficiency, our modern way of life, with computers, air-conditioning and the like, is pulling demand

2. by security of supply, which comes in two components: reliable supply and at affordable cost

There is a consensus that prices of oil and gas will remain high. And reliable supply shall not be at the expense of affordability. Otherwise economical activities and jobs are threatened, development of poor countries may never happen!

And

3. last but not least, the environmental concern: climate change is a new and daunting global challenge.

In mitigating this increased demand caused by the three major drivers there is not a single solution for the world. All available means need to be developed and must play their role in a well balanced mix of energy sources for future electric power generation and nuclear has to be part of it.

For AREVA meeting the challenge means being an integrated supplier with a global infrastructure that is locally accessible with production and manufacturing in 41 countries and sales and marketing in over 100 countries. Our nuclear operations are supported by 38 000 nuclear experts.

The worldwide nuclear capacity will grow in the coming decades. It should be noted that in all scenarios, the nuclear stake remains constant in the mix at almost 15% of the electricity generation. But even in the minimum scenario more than 100 new reactors need to be built by 2030. Such a demand for new projects can only be managed by the industrial standardization of reactor models.

Already in the beginning of the 1990ties AREVA with its industrial partners from the electricity generating industry in Germany and France started the design of the EPR from the well proven basis of their existing reactor fleet. The project was closely monitored and supported by licensing authorities and independent inspection agencies in both countries to ensure the EPR's licensability in France and Germany. For the Finnish Olkiluoto 3 project, the EPR then underwent a complete design review for the first time. Following a positive overall assessment by the Finnish authorities the Government granted the construction license in February 2005. Before the customer takes over the power plant, he must first apply for an operating license as part of the second stage in licensing.

The EPR builds on proven technologies deployed in the two countries' most recently built nuclear power plants – the French N4-series units and the German KONVOI-series plants – and constitutes an evolutionary concept based on these designs. An evolutionary design was chosen in order to be able to make full use of all of the reactor construction and operating experience that has been gained not only in France and Germany – with their total of more than 2100 reactor operating years – but also worldwide. Guiding principles in the design process included the requirements elaborated by European and US electric utilities for future nuclear power plants, as well as joint recommendations of the French and German licensing authorities.

The EPR design as it is build now in Finland and France comprises and enhanced safety level as compared to the former reactor generations and assures competitive power generation cost with any kind of alternative power generation means, whether fossil or renewable. It is the basis of a standard design that can be realized on almost all available nuclear power plant sites around the world with only minor site specific adaptations.

Safety levels at nuclear power plants have been constantly enhanced in the past. The EPR, a nuclear reactor of the third generation, represents yet another step forward in terms of safety technology, offering in particular the following features:

- Improved accident prevention, to reduce the probability of core damage even further: This is provided by a larger water inventory in the reactor coolant system, a lower core power density, high safety-system reliability thanks to quadruple redundancy and strict physical separation of all four safety system trains, as well as digital instrumentation & control systems and an optimized man-machine interface.

- Improved accident control, to ensure that – in the extremely unlikely event of a core melt accident – the consequences of such an accident remain restricted to the plant itself: this is done by confining the radioactivity inside a robust double-walled containment, by allowing the postulated molten core material (corium) to stabilize and spread out underneath the reactor pressure vessel and by protecting the concrete against meltthrough.

- Improved protection against external hazards (such as aircraft crash, including large commercial jetliners) and internal risks (such as fire and flooding).

The EPR has a slightly higher reactor thermal output than other pressurized water reactors currently in operation. The deployment of steam generators with economizer sections along with an advanced steam turbine design lead to a higher efficiency. Safety systems directly connected to the reactor coolant system serve to inject coolant into the system and to remove residual heat in the event of a loss-of-coolant accident (LOCA) are designed with a four fold redundancy. The in-containment refueling water storage tank serves to store water for emergency core cooling and accommodates any leakage water discharged via a pipe break in the reactor coolant system.

In addition to the systems for residual heat removal that are connected directly to the reactor coolant system, a further system designed to assure heat removal in the event of loss of normal feedwater supply is connected to the secondary system. This consists of a four-train emergency feedwater system that supplies water to each steam generator. In the steam generators, the heat generated in the reactor is used to produce steam for driving the turbine. This steam is then condensed in the turbine condenser. If the condenser should be unavailable due to loss of the main heat sink, the excess steam can be directly discharged to the atmosphere from the steam generators. The emergency feedwater system on the secondary side is equipped with electric-motor-driven pumps that can be powered, if necessary, by the unit's four large emergency diesel generators.

Full four-fold redundancy is provided for all safety systems and all of their auxiliary systems. The risks associated with common mode failures – which can also affect redundant systems of technically identical design – have been reduced by systematically applying the principle of functional diversity. f one redundant system should completely fail, there is always another system of diverse design that can take over its tasks, thus enabling the EPR to be safely shut down and cooled. The redundant trains of the safety-related systems are installed with strict physical separation in four different buildings so that any interference between the redundant systems is ruled out.

Not only the probability of occurrence of core damage states has been drastically reduced, but the radiological consequences of severe accidents have additionally been limited by means of a new containment design. This new design ensures that the containment will retain its structural integrity under accident conditions. Any radioactive leakages from the primary containment are collected in the space between the two containment shells and can be directed through a filter system before being discharged to the outside atmosphere. This means that even in the hypothetical event of an accident causing melting of the core its consequences would be limited to the plant itself so that no emergency actions in the vicinity of the plant would become necessary.

Besides the mitigation of hypothetical severe accidents EPR features in addition a protection against the crash a commercial airliners. This protection is realized by thick reinforced concrete walls covering the reactor, the fuel and two of the four redundant safeguard buildings. In addition to the load effects, also induced vibrations need to be considered. This is realized by the double wall of the reactor building, so that the internal structures supporting safety related equipment are completely decoupled from the outer concrete structure. Due to this design induced vibrations cannot directly affect the component supports, but have to be routed via the basemat and being damped on that way. Another consideration to be made when addressing the protection against airplane crash is the effect of fuel fires caused by kerosene. Consequently all building openings and ventilation ducts need to be protected in order to avoid ingress of burning fuel into the building.

The high degree of redundancy does not only provide the required enhanced safety level, but opens as well the chance to maintain redundant systems even during power operation. This leads to shifting maintenance work from the shutdown period of the plant for routine refueling operations to the operation period. As a consequence the required annual shutdown time is reduced and the plant availability is increased, which contributes to lower operation cost and improves the economic advantage of the plant.

AREVA can claim today to be the first plant supplier with experience in constructing Generation III nuclear power plants with these design features. This experience is being gained through our projects in Finland and France. EPR is furthermore in advanced licensing processes in the US and the UK by applying for a design certification and by being subject to a generic design assessment, respectively.

In addition to these activities we are preparing for the projects in China for which the contracts were recently signed and for the Constellation Energy project at Calverts Cliff in the United States.

EPR is also under consideration for a number of emerging projects that are in an earlier status of preparation. For ESKOM in South Africa we have just submitted bids for two EPRs to be constructed as start of a fleet in this country. In the US EPR has been selected by a number of utilities other than Constellation for their nuclear programs to come in the short-term future. The GDA process of EPR in the UK is supported by more than ten utilities that plan to invest into projects, once a pre-licensing statement of the British authorities has been granted. Also for the project of the Baltic countries in Lithuania at the site of Ignalina, a plant with EPR technology is under consideration.

The above gives just a list of projects that are in an advanced planning state. There are many more countries and investors that started to reconsider nuclear power after the Finnish and French projects had been launched. Should all these projects that are under discussion to come on line by 2030 be realized, the nuclear industry will face a big challenge. Not only the recruitment and training of young engineers will be demanding, also the whole supply chain with its hundreds of subcontractors requires a reassessment. Thanks to the early start with EPR, AREVA can benefit from the advantages of an existing supply chain that had been established some years ago for Olkiluoto 3. AREVA has invested into its own manufacturing workshops in particular for upgrading its manufacturing capabilities for primary circuit equipment. In addition a number of strategic partnerships with experienced subsuppliers were concluded to ensure a reliable and timely delivery of components needed for all these projects.

The nuclear market is booming with a big number of new projects to be realized in the short-term future. AREVA has made a lot of valuable experiences in the early construction projects of EPR. We are well prepared and we continuing to adapt to the needs of the market in the years to come.

The Karlsruhe Institute of Technology (KIT): Research, Teaching and Innovation

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In the future, the Universität Karlsruhe (TH) and the Forschungszentrum Karlsruhe – an excellence university and a national Helmholtz center – will pursue their missions together at the Karlsruhe Institute of Technology (KIT). By consolidating their capacities in research, teaching, and innovation, the two partners are laying the foundations to become one of the internationally leading institutions for science and technology. Their integrated executive, management, and codetermination bodies will realize joint planning of strategy, structure, and development, following the principle that research, teaching, and innovation constitute a unified entity and introducing comprehensive lasting changes at both institutions. In Germany, the KIT will serve as a model and meet the recommendation repeatedly expressed by the Wissenschaftsrat *"to intensify networking between universities and extra-university research institutions "*[1].

Profile building and integration of the partners in the area of research will take place on two levels: on the one hand through the competencies[2], staff members of both partners will bring to KIT, and on the other hand through concrete research work conducted in projects of rather different scope and structure.

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[1] Wissenschaftsrat (Council for Science): Empfehlungen zur künftigen Rolle der Universitäten im Wissenschaftssystem vom Januar 2006; Wissenschaftsrat, Drucksache 7067-06. S. 31 [Recommendations on the future role of universities in the sciences, of January 2006; Council for Science, Print no. 7067-06, p. 31]

[2] **Competence** means individual topic-related skills and the expertise of the staff members, including methodological knowledge, to work on scientific and technological questions along generally valid quality criteria.

The expertise, skills, and research profiles of all KIT staff members will be organized into joint areas and fields of competence. The resulting competence portfolio will provide easy internal and external access to the scientific and technological competencies of KIT and make them transparent. The generation of new projects will be supported by seed money which is awarded to the best ideas emerging from internal competition. The joint competence portfolio will be the basis for all ongoing research at KIT and the breeding ground for new scientific ideas, projects, and networks either formed among staff members themselves ("bottom-up") or initiated strategically ("top-down").

Profiling of KIT research topics will take place at the institutional level through KIT Centers and KIT Focuses which will combine and provide strategic support to thematically related projects of different scope. KIT Centers stand out through their unique characteristics in terms of scientific approach, strategic objectives, and tasks. At the centers, national research objectives can be pursued in a better and more comprehensive way as the program-related research of the Helmholtz Association and the independent research of university groups will complement and strengthen each other. KIT Focuses differ from KIT Centers with respect to the nature of their socio-political mission, their size, and their duration. By consolidating research capacities at KIT Centers and KIT Focuses critical mass is being achieved, enabling KIT research to gain international competitiveness and visibility.

At KIT, excellent research is conducted outside of KIT Centers and KIT Focuses as well and plays an important role in developing new research topics. This is why KIT supports this research with measures laid down in the competence portfolio and described in the Concept of the Future[3].

Teaching and study at KIT are characterized by comprehensive supervision and care of students, promotion of their early independence, and the extensive inclusion of research. The integration of staff members from the Forschungszentrum into teaching will drastically improve the student/instructor ratio, which will help reach similar standards of international top-level universities in this respect as well.[4]

Early independence and inclusion into research activities will be supported by stronger integration of seminar, bachelor, diploma, and master's theses into research projects of different scopes throughout KIT, comprising even research projects of major social relevance. Feasibility studies carried out by students and supported in the context of the KIT Concept of the Future also serve this purpose. Establishing KIT Schools will considerably extend interdisciplinarity in teaching. Being closely related to and maintaining intense exchange with KIT Centers and KIT Focuses, they

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[3] Universität Karlsruhe (TH) (2006). A Concept for the Future of the Universität Karlsruhe (TH) – The Foundation of KIT (Karlsruhe Institute of Technology).

[4] Wissenschaftsrat (Council for Science): Empfehlungen zur künftigen Rolle der Universitäten im Wissenschaftssystem vom Januar 2006; Wissenschaftsrat, Drucksache 7067-06, S. 87. [Recommendations on the future role of universities in the sciences, of January 2006; Council for Science, Print no. 7067-06, p. 87]

incorporate research, its methods, and its findings into teaching.

At KIT, the promotion of young scientific talent is based on excellent scientific working conditions and aims at an adequate balance between early independence, individual supervision and care, and training during the doctoral phase. This support is provided by institutes and departments and is complemented by new interdisciplinary elements of the KIT Schools and by promotion measures in the context of the Concept of the Future[3]. The House of Competence (HoC) and the Karlsruhe House of Young Scientists (KHYS) - overall structures at KIT - provide support to young scientists in acquiring key qualifications and establishing international networks.

The Forschungszentrum Karlsruhe and the Universität Karlsruhe (TH) already rank among the leading innovative partners for business and industry in certain fields. With KIT, this position will be expanded strategically. For this purpose, KIT will introduce new instruments such as Shared Professorships and Shared Research Groups as well as the KIT BusinessClub and the Karlsruhe Foundation for Innovation.

KIT's central idea is the integration of university and non-university research[5], something that has been repeatedly demanded in the past. In implementing this idea, KIT will consistently surpass every other model, thus setting new standards for research, teaching, and innovation. In order for KIT to exploit its full potential, the internal and external conditions for all those participating in the research, teaching, and innovation process will need sustainable improvement.

Further specific information on KIT can be taken from the document 'Concept for the Karlsruhe Institute of Technology (KIT)' and from http://www.kit.edu.

On February 22 2008 the Founding Ceremony of KIT took place in Karlsruhe, with Federal Minister Dr. A. Schavan and Minister Prof. P. Frankenberg being present. „Now, an important step towards the real merger is done: KIT will be set up as a public body according to the Baden-Württemberg state law," announced BM A. Schavan. Thus, KIT will be one legal entity with two missions: the mission of a state research university and the mission of a national programmatic research centre within the Helmholtz association.

[5] Wissenschaftsrat: Empfehlungen zur künftigen Rolle der Universitäten im Wissenschaftssystem vom Januar 2006; Wissenschaftsrat, Drucksache 7067-06, S. 31. [Recommendations on the future role of universities in the sciences, of January 2006; Council for Science, Print no. 7067-06, p. 31]

Research Reactor Coalitions

- First Year Progress Report

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Abstract. The IAEA has initiated new activities with the objective of promoting formation of coalitions of research reactor operators and stakeholders. The aim of this effort is to promote concrete examples of enhanced regional cooperation, to form networks of research reactors conducting joint research or other shared activities, and to form a voluntary, subscription-based, self-financed coalition.

The objective is 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.

This paper will describe progress in the first year of IAEA activities to assist the formation of research reactor coalitions. This includes IAEA efforts to serve a catalytic and “match-making” role for the formation of new commercial and other relationships to increase research reactor utilization, including organizing various missions and meetings for exploratory and initial organizational discussions on possible coalitions and networks .This also includes activities to assist research reactors in carrying out strategic planning with a view to forming research reactor coalitions, training activities to assist in the development of nascent coalitions, and development of arrangements to facilitate access to stakeholders requiring irradiation services and for countries that are not operating a research reactor.

1. Background

Research reactors play a key role in developing the peaceful uses of nuclear energy. In order to continue in this role, they need to be financially sound, with adequate income for safe and secure facility operations and maintenance, including planning for eventual fuel removal and decommissioning. However, in a context of declining governmental financial support, many research reactors are increasingly challenged to generate additional income to offset their operational costs, without making any provision for the liabilities that will be incured when their facilities reach the end of their operating lives Reactors operating at low utilization levels have difficulty providing products and services with the reliability demanded by potential users and customers, and this creates a significant obstacle to increasing utilization.

These challenges are also occurring in the context of increased concerns about nuclear material safety and security and the threat of nuclear proliferation, due to which research reactor operators are compelled to substantially improve physical security and convert reactors from highly enriched uranium (HEU) to low enriched uranium (LEU) fuel. Thus, there is today a complex environment for

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research reactors, and one in which underutilized, and therefore likely poorly-funded, facilities invoke particular concern.

Many research reactors have limited access to potential customers for their products and 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 reactors contracting for services at prices below those required to cover their full costs, preventing recovery of back-end costs and creating unsustainable market conditions.

The research reactor community possesses the expertise to address these concerns. However, this knowledge is not uniformly available as parochial attitudes and competitive behaviour 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 by another “competing” reactor. However, the formation of coalitions will likely stimulate new demand for products and services, without reducing the demand from existing users.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 to expand the use of research reactors – including by countries without such a facility. However, a reactor constructed to meet a specific need might not have sufficient identified utilization to fully occupy the facility, or to be adequately available for its intended purpose. A potential solution to this dilemma would be the creation of one new multi-national facility rather than a number of national facilities, but 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. This integrates Agency regular and extra-budgetary funded program activities related to research reactors, national and regional IAEA Technical Cooperation projects, especially “Enhancement of the Sustainability of Research Reactors and their Safe Operation Through Regional Cooperation, Networking, and Coalitions” (RER/4/029) and “Nutritional and Health-Related Studies Using Research Reactors” (RAF/4/020; AFRA IV-12), and is also supported by a grant from the Nuclear Threat Initiative (NTI).

2. Concept outline

From the operational perspective, coalitions will facilitate peer group sharing of best practices, improve information availability to members, and both reinforce and develop the operating disciplines of safety, security and quality control. From the business perspective, coalitions will provide improved market analysis and support for strategic and business planning. Where appropriate, coalitions may jointly market services and increase contacts between research reactor operators and prospective customers. By so doing, they will help increase reactor utilization, improve the services provided to the communities they serve, generate additional revenues and thus justify additional investment in operational improvements.

From the public perspective, coalitions will have the opportunity to enhance the information available to help retain and build confidence in reactor operation.

There is not a “one size fits all” solution and coalitions can take several different forms according to the needs and capabilities of their members. Possible coalition variants include: bilateral sub-contracting, joint venture and other supply arrangements between pairs of, or larger groups of, research reactors; informal peer group networks that can share best practice information; and broader coalitions that are capable of effectively marketing their members' services and representing their interests in common, as well as setting standards for all members. It is expected that some coalitions will also offer access to members from non-reactor owning countries, with financial subscriptions paid in return for access to reactor services. This will result in increased utilization of existing, or purpose-built facilities, thus avoiding construction of new reactors that will not be fully utilized or continued operation of marginally supported reactors.

In most cases, it is envisaged that coalitions will not start with full scope implementation, but rather will develop from relatively modest starting points (e.g. involving two or three members coordinating a single activity), and will expand their scope of implementation as the confidence of the members, and their governments, increases. For example, a simple, bilateral backup supply arrangement may grow into an informal network, and eventually become a subscription-based coalition.

3. Concept benefits

A coalition is expected to have both general and specific benefits to participating research reactors. The general benefits include such items as standardization of operating practices and security procedures. The specific benefits of a coalition will derive from improved strategic and business planning (using IAEA-TECDOC-1212 "Strategic Planning for Research Reactors" as a guide) and joint marketing of the services of its participant reactors (commercial products and scientific/research activities), with the coalition thus able to:

- Optimize the services offered (possibly including education and training, production of isotopes, industrial irradiation services such as transmutation doping, neutron activation analysis and other analytical services for industry and government) on a geographical basis, and reduce operational costs.
- Maximize the use of specialized expertise or equipment at a particular facilities, and enable facilities to specialize in services in which they could have a "comparative advantage."
- Use the combined expertise of the participant facilities to best advise and serve their customers. This would help increase customer knowledge of, and access to, the services and products the coalition can provide, and support the customer with a more reliable and comprehensive customer service.
- Improve the utilization and sustainability of individual research reactors, and increase overall levels of demand to the mutual benefit of all market participants (suppliers and customers). Increasing reactor utilization would generate additional revenues, or help make the necessary justifications for additional local governmental support, thus improving sustainability. The additional funding could assist individual reactors to pay for operational, safety and security improvements.
- Develop a common methodology for calculating costs of reactor services to include spent fuel management and eventual decommissioning liabilities.
- Act as a coordinated entity in procuring new fuel and contracting for spent fuel management services, thus reducing the costs of these activities incurred by each reactor operator and benefiting from the economy of scale
- Provide assistance to reactors planning or undergoing conversion from HEU to LEU including sharing of experience and planning expertise.
- Address needs of user groups without access to a research reactor in their Member State(s).

4. IAEA Activities and Progress

The Agency's role is to serve as a catalyst and a facilitator of ideas and proposals. Meetings held by the IAEA in August and September 2006 resulted in preparation of a grant request on research reactor coalitions which was submitted to the Nuclear Threat Initiative (NTI) and approved in October 2006.

From October 2006 to January 2007, the IAEA conducted informal consultations with a wide number of research reactor operators, commercial entities, users of research reactor irradiation services, and other stakeholders. Approximately fifteen "notional proposals" for coalitions covering a range of subjects and virtually all geographic areas were initiated, which became the basis of the Agency's initial activities in 2007. Following initial discussions with potential participants, several of the notional proposals were further elaborated and then became the basis for exploratory meetings in fall 2007.

A. IAEA as "Matchmaker"

The IAEA identified several "matchmaker" opportunities. Two are described here as examples of how coalitions can benefit both reactor operators and their customers. In both cases, the Agency's initial contacts led to direct meetings and negotiations between the various partners without the Agency's participation.

The first was between a well-utilized research reactor and another, less-well utilized but state-of-the–art, research reactor in the same geographic region. In this case, the well-utilized reactor was seeking additional irradiation capacity for its commercial business. In this coalition, the well-utilized reactor will serve as the "lead reactor," sub-contracting work to the second reactor based on the first reactor's orderbook. It will ensure that quality control and quality assurance procedures and standards are adhered to by the sub-contracting reactor so that the products delivered to the lead reactor's customers meet the same standards as products irradiated in its own facility.

In the second example, the Agency brought together an existing research reactor supplier of industrial isotopes, which is planning for cessation of operations, a commercial user of industrial isotopes/tracers and an underutilized research reactor in a region where the commercial user had a growing demand for industrial isotopes. In this case, the reactor is projected to be a direct contractor/supplier to the commercial user, based on a non-exclusive contractual arrangement. The IAEA conducted a training workshop at Imperial College U.K. from May 14-16, 2007 to assist staff of the underutilized research reactor in understanding the management issues associated with supply of isotopes to a commercial customer.

Following consolidation of these contractual arrangements, the IAEA will encourage the respective partners to add additional members to the contractual arrangements, at a minimum to ensure back-up production arrangements and to expand the "menu" of technical capabilities offered by the coalition.

B. Strategic planning for coalitions

Strategic planning assists research reactors to better understand their strengths and weaknesses, and their stakeholders and stakeholder needs, and to adjust their activities to address national development priorities as well as the commercial marketplace. Strategic planning can also assist research reactors in developing ideas for alliances or coalitions based upon complementary strengths and weaknesses.

The IAEA organized an expert mission to Kazakhstan and Uzbekistan from 8-12 October 2007 to assist the staff at the respective Institutes of Nuclear Physics to further develop strategic plans and to consider formation of cooperative ties between the research reactors in the region. At an IAEA Workshop on Advanced Strategic Planning for Research Reactor Coalitions (Europe region), Vienna, 17-19 December 2007, representatives of the two countries proposed formation of a Central Asia Research Reactor Coalition, and a number of actions are contained in the meeting report with a view toward concluding such an arrangement.

The workshop cited above was also attended by representatives of user organizations and research reactor operators from Armenia, Austria, Azerbaijan, Czech Republic, Italy, Kazakhstan, Norway, Romania, and Russia. The research reactor operators made presentations relating to their utilization patterns and the development of strategic plans, based on a SWOT analysis (strengths, weaknesses, opportunities, and threats), including the example of a research reactor which made a successful transition from a state-supported institution to a fully commercial operation. Participants without research reactors made presentations regarding their nuclear science, irradiation, nuclear power plant support and training, and radiation protection needs for which access to, or services from, a research reactor are necessary. The participants also visited the TRIGA reactor at the Atominstitut (ATI) of the Vienna University of Technology for briefings on strategies and activities for the successful utilization of a low-power research reactor, particularly for education and training purposes.

The final report of the workshop contains suggestions from each of the participants regarding ideas for cooperation and collaboration with other research reactors and concrete proposals for research reactor coalitions, with specific action items. In addition to the Central Asia Research Reactor Coalition noted above, these include:

-Nuclear Education and Training Coalition (potentially involving Armenia, Azerbaijan, Austria/ATI, Czech Republic/CTU, and Italy)

-Innovative Reactor Systems and Fuel Cycles (potentially involving Czech Republic/Rez, Norway/Halden, Romania/INR, Russia/RIAR, and Ukraine.

-Central/Eastern Europe (via an external proposal from Hungary, and also involving Czech Republic, Romania, and Poland)

The IAEA is currently pursuing a number of activities relevant to the first two proposals through both regular budget and Technical Cooperation program mechanisms.

On the final proposal, the IAEA participated as an observer in an exploratory meeting organized by KFKI in Budapest, Hungary on 28-29 January 2008 concerning the formation of an Eastern Europe Research Reactor Coalition. The participants reached preliminary agreement to hold further discussions with the objective of initiating enhanced cooperation in the field of neutron beam experiments. .

C. Exploratory missions on forming research reactor coalitions

Missions and meetings were organized in fall 2007 to discuss forming specific coalitions:

1. Russian Federation experts and institutions, Dmitrovgrad, Russian Federation, 5-6 September 2007, and Vienna, Austria, December 13-14, 2007;
2. Instituto Peruano de Energia Nuclear (IPEN), Peru and Comision Chilena de Energia Nuclear (CCHEN), Chile, with Missouri University Research Reactor (MURR) and McMaster Nuclear Reactor (MNR), Lima, Peru and Santiago, Chile, 15-19 October, 2007;
3. CNEA (Argentina) and ATI, Buenos Aires, Argentina, 22-23 October, 2007;
4. ININ (Mexico) – Laguna Verde Nuclear Power Plant – ATI, Centro Nuclear ININ, 29 October 2007);
5. Caribbean region research reactor coalition (Jamaica-Mexico-Colombia), Centro Nuclear ININ, 30-31 October 2007.

The meetings with Russian experts in September and December resulted in conclusion of meeting protocols that cited a number of possible areas for coalitions among Russian research reactors and/or with research reactors outside Russia. These include Russian coalitions for i) education in nuclear science and engineering, and ii) industrial and medical radioisotopes; and international coalitions for a) nuclear science and materials testing and b) LEU fuel conversion. Follow-up meetings and facility visits to plan implementation steps are scheduled for March 12-14, 2008 in Russia.

The missions and facility visits that took place in October 2007 to Chile and Peru were led by the IAEA. The team included representatives from MURR and MNR for discussions on possible coalitions involving medical and industrial radioisotope research, development, and production. Protocols with action items were agreed for both missions, which included a number of concrete ideas for supply of radioisotopes between institutions and for transfer of production technology There has been an extensive exchange of information in the following months, as well as arrangements concluded for radioisotope supply. It envisaged that further meetings will be held in mid-2008 to further formalize the coalition arrangements and to plan next steps.

IAEA-led missions to Argentina and Mexico in October 2007 included a representative from the TRIGA reactor at ATI. These meetings focused on establishment of coalitions involving nuclear education and training activities, including with the Insituto Dan Beninson (CNEA/Argentina), ININ and the Laguna Verde Nuclear Power Plant (Mexico). Preliminary coalitions agreements were signed, with specific follow-up steps defined. As a result of the meeting in Mexico, ININ is developing a practical reactor operations training course for personnel from the Laguna Verde Nuclear Power Plant to be held at its TRIGA reactor in 2008.

Preliminary agreement was reached at a meeting at ININ on 31 October 2007 to form a Caribbean research reactor coalition between the three reactors in Colombia, Jamaica, and Mexico. It is envisaged that this coalition will serve as a regional resource for users of nuclear science and irradiation services in other countries in the Caribbean region that do not have research reactors. The focus of its activities will initially be on neutron activation analysis, especially for environmental applications, as well as training services. A draft Memorandum of Understanding for the coalition is under review by the parties, a reactor operator certification course is being formulated by ININ (for Colombia), and Jamaica is developing a course on neutron activation analysis.

Other proposals related to potential coalitions, including in Africa and East Asia and the Pacific are still in the formulation stage, with exploratory meetings to be held in 2008. Of particular note, the IAEA held a meeting in Vienna from 11 to 13 February, 2008, to explore the formation of a neutron sciences/neutron scattering coalition with representatives primarily from the Europe region but also from Australia and the U.S.

5. Conclusion

The Research Reactor Coalitions initiative has made considerable progress during its first year of full activity. The IAEA has successfully played the role of “catalyst” and facilitiator of ideas. As a result – and perhaps most importantly – the coalitions concept seems to be gaining international acceptance, with the term frequently used in international research reactor meetings and discussions.

As further evidence of this, a number of countries and institutions have formulated, and more are developing, their own proposals for coalitions.

The IAEA has also successfully identified a number of opportunities to act as “matchmaker” in introducing and facilitating discussions between partners that led to new commercial arrangements for increased utilization of specific research reactors. These arrangements are expected to form the basis for broader research reactor coalitions in the future.

In addition, a significant number of exploratory missions and discussions were held, resulting in initial or preliminary agreements for several coalitions. While these are still being developed, it is expected that one or more formal research reactor coalitions will come to fruition in 2008 as a result of these activities.

The IAEA invites suggestions and proposals for additional coalitions from other Member States and institutions.

OVERVIEW ON HIGH DENSITY UMo FUEL IN-PILE EXPERIMENTS IN OSIRIS

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ABSTRACT

This paper is an up date of the French IRIS program on high density UMo/Al dispersion fuel. Some PIEs performed on the recent IRIS-3 and IRIS-TUM experiments are presented and discussed. They confirm the good in-pile behaviour of full size ground powder based plates up to high power and burn-up. The positive effect of the Si addition to the Al matrix on the irradiation behaviour of full size plates is also evidenced, in particular for atomised powder based plates. Despite these good results and considering manufacturing and reprocessing aspects, an oxide coated atomised UMo fuel is consequently proposed as a promising solution.

1. Introduction

As alternatives to the very first fuel concept (dispersed atomised UMo in pure Al), the French IRIS program has tested two improvements: modification of the matrix composition and a change in the UMo powder characteristics [1]. Up to now, this program involves 4 full size fuel plate experiments performed in the OSIRIS reactor on high density UMo dispersion fuel, IRIS1 [2], IRIS2 [3], IRIS3 [4] and IRIS-TUM [5]. The FUTURE plates [6] irradiated in the BR2 reactor completed this program. The ground particle based fuels can show good in-pile behaviour, as the IRIS1 experiment demonstrated. This was now confirmed by IRIS-TUM plate tests, irradiated to higher equivalent burn-up at much higher load. The influence of the Si addition to the Al matrix has been studied on both atomised (IRIS3) and ground (IRIS-TUM) UMo powder. The Si benefit is obvious, especially for the plates made of atomised UMo powder. Post-irradiation examinations are in progress on the more recent irradiation tests. This paper gives a preliminary comprehensive overview on the in-pile behaviour of these different fuels. The predominant factors and their roles are discussed. In order to discriminate the different parameters influencing the conservative in-pile behaviour of ground powder, a new experiment, IRIS-4, with a fuel made of oxidised particles, is underway.

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2. Main features of the IRIS experiments

The IRIS 1 to 3 experiments have been performed by CEA, within a close collaboration with AREVA-CERCA for the manufacturing aspects. The IRIS-TUM experiment has been launched in the framework of a collaboration between TUM, CEA and AREVA-CERCA.
All irradiations have been performed in the OSIRIS MTR reactor with the IRIS irradiation and measuring device, originally developed to qualify the silicide fuel for the OSIRIS conversion and FRM II [7].
All plates are full size and manufactured by AREVA-CERCA through classical rolling process. The main manufacturing and irradiation features of the IRIS experiments are collected in Tab. 1.

	Experiment	IRIS-1	IRIS-2	IRIS-3	IRIS-TUM	*IRIS-4*
Manufacturing data	UMo powder type	ground	atomised	atomised	ground	*atomised*
	Mo in UMo (wt%)	7.6 or 8.7	7.6	7.2	8.1	*7*
	Enrichment (^{5}U wt%)	19.8	19.8	19.8	49.5	*19.8*
	Si in Al matrix (wt%)	0	0	0.3 2.1	0 2.1	*0 2.1*
	Matrix type	A5	A5	AlSi0.3 AlSi2.1	A5 AlSi2.1	*A5 AlSi2.1*
	Fuel loading (gU/cc)	7.9-8.3	8.2-8.3	7.8-8.0	7.3-8.4	*7.9*
	As fab meat porosity (%)	11-13	1-2	0.8-2.4	8-9	*1-2*
	Cladding material	AG3NE	AG3NE	AG3NE	AlFeNi	*AlFeNi*
Irradiation data	Year	2000-2001	2003	2005-2006	2005-2007	*2008-2009*
	Number of plates	3	4	4	4	*4*
	Status of experiment	completed	stopped	stopped completed	completed	*foreseen*
	OSIRIS core position	17	52	14	11 and 17	*52*
	Max heat flux at BOL (W/cm^2)	123-145	238	201	250-258	*290*
	Max clad surface temp. (°C)	68-73	93	83	97	*100*
	Number of cycles	10	4	7	8	*5-6*
	Duration (EFPD)	241	58	131	147	-
	Plate average BU (^{5}U %)	46.9	32.5	48.8	35.3-59.3 LEU_{eq}	*> 50*
	Average BU at MFP (^{5}U %)	54.0	39	56.5	43.4-69.8 LEU_{eq}	-
	Max BU at MFP (^{5}U %)	67.5	39.7	58.8	56.3-88.3 LEU_{eq}	-
	Average FD at MFP (f/cm^3_{UMo})	3.2 10^{21}	2.2 10^{21}	3.4 10^{21}	4.2 10^{21}	-
	Max FD at MFP (f/cm^3_{UMo})	4.6 10^{21}	2.7 10^{21}	4.1 10^{21}	5.6 10^{21}	-

Tab. 1: Main features of the IRIS experiments

The main differences are related to :

- the type of UMo powder, atomised or ground,
- the type of matrix, either pure Al or added with silicon up to 2.1 wt %,
- the 49.5% enrichment of the IRIS-TUM plates to reach higher irradiation conditions,
- the maximum heat flux of about 120 W/cm^2 for IRIS1 to 258 W/cm^2 for IRIS-TUM (cf. Fig. 2),
- the maximum clad surface temperature of 68°C for IRIS1 to 97°C for IRIS-TUM,
- the AG3NE or AlFeNi cladding.

3. Non destructive testing

The plate thicknesses have been measured before and after each cycle for all the IRIS plates. The results are plotted as a function of fission density in Fig. 1. They demonstrate:

- the better in-pile behaviour of the plates made of ground particles up to high burn up and heat flux, in comparison with the atomised UMo based fuel,
- the positive effect of Si addition to the Al matrix. This improvement is particularly visible in the case of atomised UMo based fuel plates (IRIS-3). For the plates made of

ground UMo (IRIS-TUM), the effect of Si is covered by the features of the ground UMo particles themselves (shape microstructure, defects, oxidised surface).

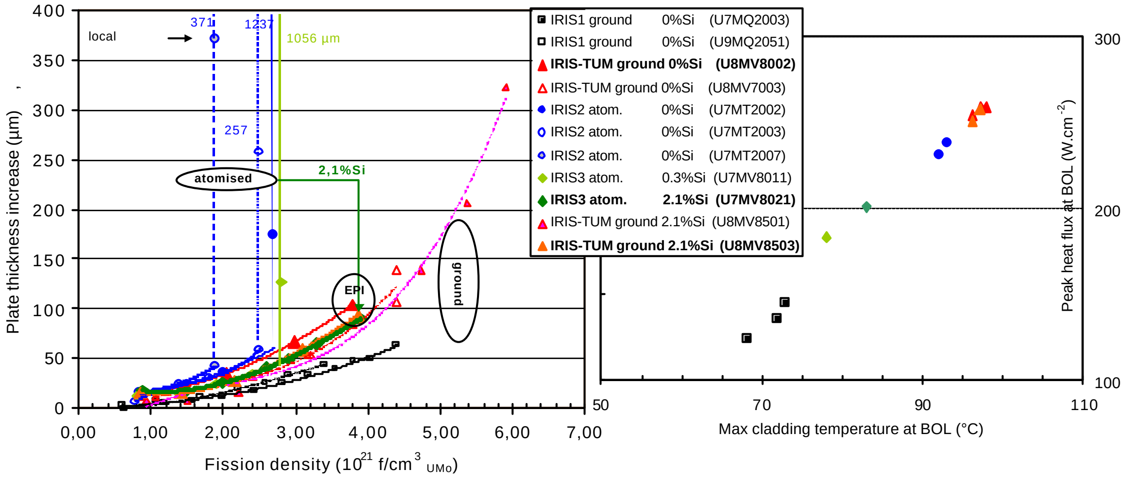

Fig. 1: Plate thickness increase with fission density in UMo particles

Fig. 2: Irradiation conditions of the IRIS experiments

4. Post Irradiation Examination

The IRIS-TUM plates U8MV8503 & U8MV8002 and the IRIS3 plate U7MV8021 (see Fig. 1 and Tab. 2) have been recently examined by optical and scanning electron microscopy at the hot laboratory (LHMA) of SCK•CEN in Mol, Belgium [8, 9].

IRIS-2 (0%Si) IRIS-3 (2.1%Si) IRIS-1(0%Si) IRIS-TUM (0%Si) IRIS-TUM (2.1%Si)

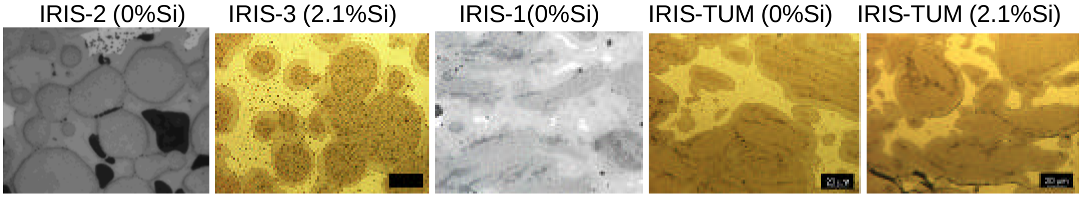

Fig. 3: Optical micrographs at MFP

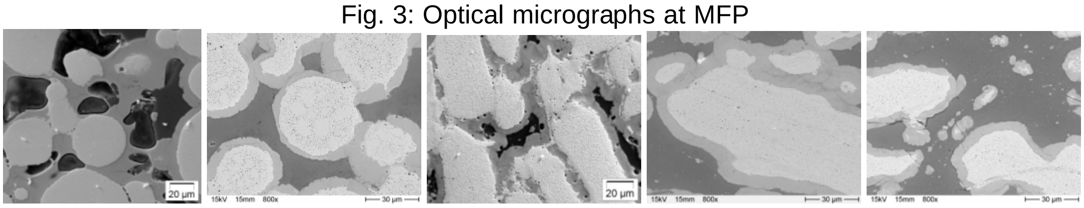

Fig. 4: SEM images at MFP

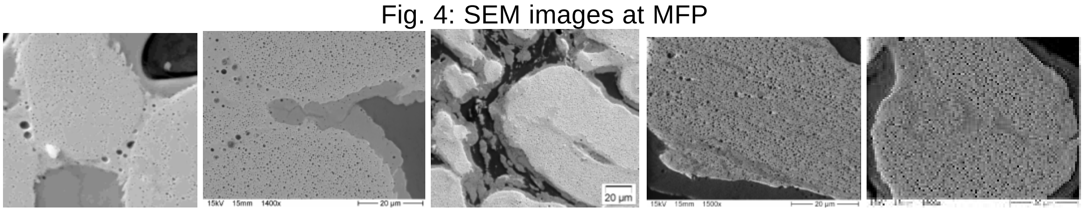

Fig. 5: Detailed SEM images

Experiment	IRIS-2	IRIS-3	IRIS-1	IRIS-TUM	IRIS-TUM
Plate number (Si content)	2002 (0%)	8021 (2.1%)	2003 (0%)	8002 (0%)	8503 (2.1%)
Powder type	**atomised**		**ground**		
As-fab porosity (%)	1.5	2.2	12.4	7.9	8.9
Max heat flux (W/cm^2)	**238**	**201**	**124**	**254**	**258**

T_{max} clad surface (°C)	93	83	68	96	97
Max FD at MFP (10^{21} f/cm^3_{UMo})	**2.7**	**4.1**	**4.4**	**3.8**	**3.8**
Max swelling (µm)	1237	90	77	104	93

Tab. 2: Main characteristics of the IRIS samples examined at the LECA and LHMA.

Some of the images collected are compared with those obtained at the LECA hot laboratory of Cadarache, France. Their characteristics are gathered in Tab. 2. The different phases existing in all the samples, determined by analysis of the SEM images, are plotted in Fig. 6.

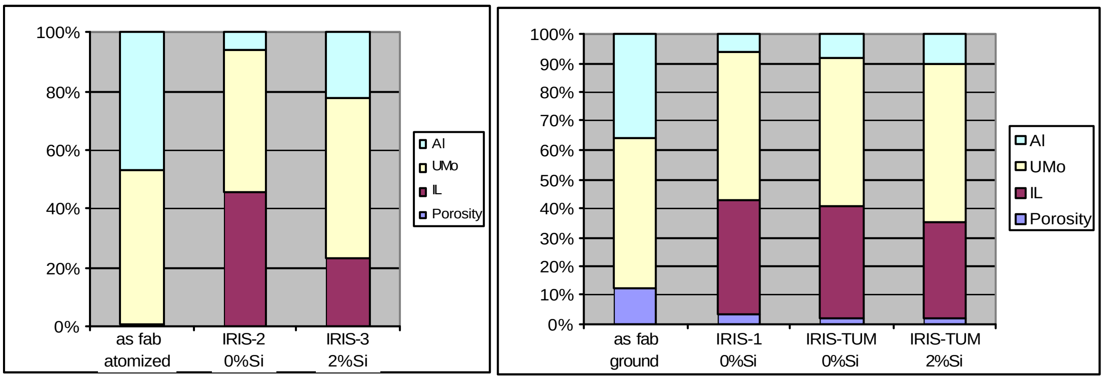

Fig. 6: Surface fractions of the different phases for atomised (left) and ground (right) UMo.

The main observations derived from those images and plots can be formulated as follows:

- In all the samples, an interaction layer (IL) is formed at the UMo/Al interface at the expense of the Al matrix and the UMo particles.
- The apparent volume of UMo particles is quite uniform. The UMo consumption is compensated by its swelling due to fission products (FP) and fission gas (FG) bubble formation.
- In the plates with few (0.3%) or no Si addition to the Al matrix, the IL is homogeneous around all the fuel particles, while in the plates containing 2.1%Si, the IL is thinner, irregular and jagged. In this latter case, the inter-diffusion Al/UMo seems to be partly hindered.
- In the plates with Si addition, Si particles are seen dispersed in the Al matrix except close to the fuel particles probably because of fission track enhanced dissolution.
- An oxide layer (dark in the OM images) is clearly observed around ground UMo particles.
- Fission gas bubbles, quite homogenous in size, are distributed in the fuel particles. In atomised samples, these bubbles seem to reveal the cell boundaries (Mo depleted zones).
- Some larger bubbles appear at UMo/IL interfaces and UMo/UMo inter-particle boundaries.
- No or only very few crescent moon shape pores due to FG are detected at the Al/IL interface. As these bubbles are the very start of the phenomenon leading to the large pillowing observed in the IRIS-2 and FUTURE plates, their absence is a hint for a more conservative behaviour of the IRIS-3 (2.1%Si) and IRIS-TUM plates.

5. Discussion

Recently, PIEs were performed on samples of IRIS-TUM plates U8MV8002 & U8MV8503. These irradiations at high heat flux and BU confirm the observations already made on the ground UMo based plates in the IRIS-1 PIEs. As discussed in our previous paper [1], several characteristics of the ground fuel play a key role and are certainly at the origin of its conservative in-pile behaviour. In random order, they can be listed as follows:

- Morphology/granularity:
    - The irregular shape and size of ground particles could strengthen the cohesion between the UMo particles and the Al matrix and increase plate mechanical properties.

- The initial residual porosity in Al matrix, of about 10 vol. %, (against 1-2 vol. % for spherical atomised powder), could act as a buffer for fission gases and compensate part of the swelling.
  - Another consequence lies in the amount of Al matrix available to react with UMo particles. In ground fuel, the Al surface fraction is about 35%, much lower than the 48% measured in atomised fuel plates.
- Microstructure:
  - The high concentration of "defects" introduced by the mechanical grinding process could also trap gas atoms.
  - The UMo raw material, prior to powder production, is heat treated at high temperature in order to avoid any Mo micro-segregation.
- Composition:
  - Influence of Mo, O, Si on the IL composition, properties and stability at severe irradiation conditions. Recent out-of-pile studies clearly showed the influence of Si on the IL nature [10].
  - The oxygen, introduced during grinding process as an irregular oxide layer (UO_2) around UMo particles, and the Si particles, added to the Al matrix, seem to act as a barrier to the inter-diffusion of Al/UMo, hindering the interaction between UMo particles and Al matrix.

This positive effect of Si is particularly visible in the atomised UMo based IRIS-3 plates. For the 0.3% Si containing plates, a pillowing occurred (cf. Fig. 1), as in the IRIS-2 and FUTURE experiments, while in the case of the 2.1% Si plate, no abnormal swelling is observed [4]. The PIEs performed on this 2.1% Si IRIS-3 plate U7MV8021 showed that 23% of the Al remains. The IL represents only 22% of the volume (cf. Fig. 6), which is not enough for pillowing to start. The Si particles close to the UMo fuel kernels act as obstacles to the inter-diffusion Al/UMo [11] and IL growth. Various out of-pile heavy ion irradiation [12, 13, 14, 15] and diffusion studies [10, 16, 17, 18] already showed this positive effect of Si in decreasing the interaction rate between UMo and Al.
For a better quantification of the fission products (mainly gases) and IL/Al volume fraction amounts and properties for breakaway swelling to occur, new measurements and examinations (SEM, EPMA, XRD) of the IRIS-3 and high burned IRIS-TUM plates U8MV8501 & U7MV7003 are planned in 2008-2009.

6. Conclusion - Perspectives

The recent PIEs performed on the IRIS-3 (2.1%Si) and IRIS-TUM samples confirmed the benefit of Si addition to the Al matrix. This effect is particularly visible in the case of plates made with atomised UMo particles. For the ground UMo based fuel plates, this positive effect is more difficult to evidence, because of the already good in-pile behaviour of ground UMo fuel even without Si, which is related to its composition, microstructure and morphology.

To better discriminate the role of those different parameters, a new experiment, IRIS-4, with a promising fuel made of oxidised particles has been launched. The objective is to test the influence of an oxide layer coating on the UMo particles on the in-pile plate behaviour [19, 20]. The main specifications of this experiment are given in Tab. 1. Considering manufacturing aspects and the difficulties to industrialise a grinding process, atomised particles have been selected. The thermochemically controlled oxidation of the atomised UMo powder has been done last autumn. The mean UO_2 thickness layer around UMo particles is 1.5±0.5 µm (cf. Fig. 7). The 4 full size plates, with or without Si addition to the Al matrix, have been already produced at AREVA-CERCA [21] and will be irradiated in OSIRIS reactor from the middle of 2008. The fabrication and irradiation of test samples similar to IRIS-4 are planned by TUM.

Here the objectives are atomised powder of an enrichment of 49,8%, oxidized, with and without Si addition and a heat load towards 400 W/cm^2.

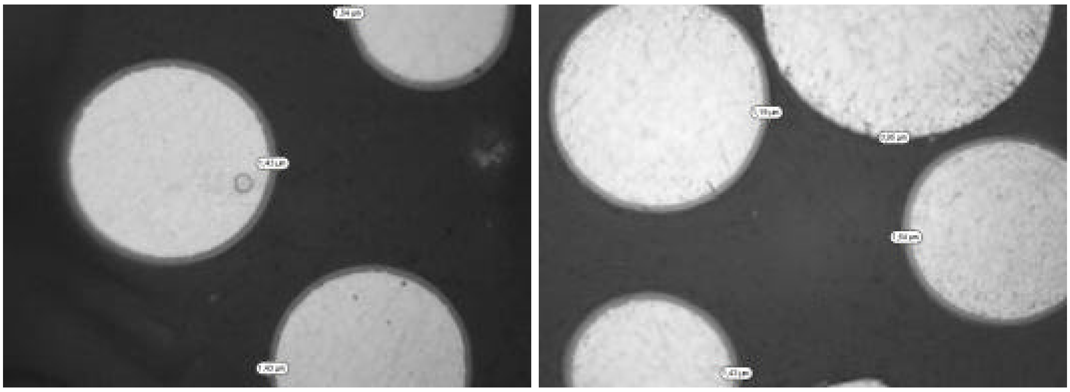

Fig. 7: Micrographs of the CEA/CERCA oxidised atomised UMo particles to be irradiated in IRIS-4 experiment

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PROGRESS IN US LEU FUEL DEVELOPMENT

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ABSTRACT

Very high uranium density nuclear fuels are currently under development in the U.S. to enable the conversion of many research reactors worldwide to LEU based fuels. Significant progress has been made in both the uranium-molybdenum based dispersion and monolithic fuel forms. The efficacy of silicon additions to the matrix of dispersion fuel meats has been demonstrated. Full size dispersion plates with loadings greater than 8.0 g-U/cc have been fabricated with silicon additions to the matrix and are ready for irradiation testing. Monolithic mini-plates with modified fuel/cladding interfaces (both silicon enhanced and zirconium diffusion barriers) have been fabricated by both friction bonding and hot isostatic pressing and have nearly completed irradiation to demonstrate their impact on fuel/clad interface chemistry. Full size monolithic plates have been fabricated with both types of interlayer by friction bonding and are currently under irradiation to evaluate mechanical response at prototypic scale. The plans for future development and qualification are discussed.

1. Introduction

The overall goal of the U.S. National Nuclear Security Administration's (NNSA) Global Threat Reduction Initiative is to minimize the use of highly enriched uranium worldwide. As part of this initiative, the Reduced Enrichment for Research and Test Reactors (RERTR) program has been charged with developing the nuclear fuels necessary to enable the conversion of civilian research and test reactors. The program began development of dispersion type uranium-molybdenum (U-Mo) based fuels in the early 1990's. Although early testing demonstrated very promising results, high power and burnup testing on U-Mo dispersed in aluminium revealed that the fuel/matrix interaction product was prone to the formation of large fission gas bubbles. Formation of these bubbles eventually lead to the onset of breakaway swelling.

Modifications to the fuel design were then sought to improve performance [1]. Adding silicon to the matrix material was proposed as a way to form interaction products more similar to the stable materials observed in U_3Si_2 based dispersion fuel. A second U-Mo based fuel type was also proposed at this time. The fuel meat was replaced by a solid (or 'monolithic') fuel foil that eliminated the matrix material altogether. This fuel design would substantially increase the net uranium density of the fuel and would consequently enable conversion of a new group of reactors. However, implementation of the monolithic fuel form required significant fabrication development before testing would be possible.

High density U-Mo based dispersion mini-plates (25 mm wide, 100 mm long, and 1.40 mm thick) were fabricated for testing using standard roll bonding techniques. Mini-plates with the silicon modified matrix material have been tested extensively at this scale in the RERTR-6 and RERTR-7 experiments. Several matrix materials were tested including Al-0.2% Si alloy, Al-2.0% Si alloy, Al-6061 (~0.9% Si), and Al-4043 (~4.8% Si). These tests showed that for silicon compositions greater than 2% a substantial reduction in interaction product thickness was achieved and that the interaction product was stable under irradiation to very high fuel phase burnups (>20% total uranium).

Fabrication techniques for very high density U-Mo based monolithic mini-plates were developed to enable performance testing on the mini-plate scale. Mini-plates were fabricated by friction bonding and were tested in the RERTR-6 and RERTR-7 experiments. These experiments showed that the fuel phase remained stable and that the overall fuel performance was good. However, behaviour similar to that observed in early dispersion tests was identified at the fuel/clad interface. Although the interaction layer was very thin, void formation was noted in regions of very high burnup. It was believed that formation of these structures might weaken the bond strength between the fuel and cladding. Two approaches to improving the bond behaviour were proposed including the application of a high silicon layer to the fuel/clad interface (to hopefully yield the same response as in dispersion fuels) and the insertion of a zirconium diffusion barrier between the fuel and cladding.

2. Recent Advances in Fuel Development

2.1 Fuel Fabrication

The implementation of monolithic fuel designs requires the development and demonstration of three key fabrication aspects, foil fabrication, interlayer application, and fuel/clad bonding. Significant advancements in all three areas were achieved in the last year.

In order to further strengthen the fuel/clad bond strength at the end of irradiation, the incorporation of an interlayer material was proposed to either alter the chemistry of the interaction product or minimize the amount of interaction. Adding silicon to the U-Mo/Al interface has been shown to improve the irradiation stability of the interaction product in both dispersion fuels and in monolithic fuel plates irradiated in the RERTR-7 experiment. A plasma spray technique was used to apply a thin uniform layer of Al-Si or Si to the cladding pocket prior to plate assembly thereby making it available in the fuel/clad interface region. The formation of a U-Mo/Al interaction product could also be prevented by the insertion of a diffusion barrier material between the fuel and cladding. A thin layer of zirconium has been applied to the fuel foil during coincident hot rolling of the fuel coupon with a top and bottom layer of zirconium [2].

Full size U-10Mo foils were successfully fabricated using two different processes [3]. Plate shaped U-Mo ingots were cast at the Y-12 National Security Complex to simultaneously dilute, alloy, and homogenize the fuel material. This plate was then hot rolled or machined to an intermediate thickness (5.08 mm down to 2.29 mm) that was suitable for final reduction. The plate was then sectioned into smaller coupons to simplify cold rolling into individual thin foils (nominally 0.25 mm to 0.38 mm thick). The second process demonstrated at INL started with the same Y-12 coupons and used a canned hot roll to enable interim annealing steps. The resulting product from each process showed distinct differences that impacted downstream processing. The grain structure of the cold rolled foils was equiaxed in nature and the foils behaved in a very 'soft' manner. Alternatively, the grain structure in the hot rolled foils was elongated in the rolling direction and the foils were stiffer and more brittle. These properties proved to be important during subsequent friction bonding [4].

Several full size fuel plates (roughly 600 mm x 50 mm x 1.27 mm) were fabricated for irradiation testing using the friction bonding process. Meaningful advances were made in the design of the friction bonding tool piece and in the definition of critical process parameters. These advances played a significant role in enabling the fabrication of two plates (without interlayers) for ATR-Critical facility tests and two plates for the AFIP-2 irradiation experiment in ATR. The AFIP-2 experiment consists of one fuel plate with a silicon enhanced fuel/clad interface and one plate with a zirconium diffusion barrier between the fuel and clad. It was observed during this fabrication campaign that hot rolled foils, which were more brittle, were more likely to fracture and flake during friction bonding while the cold rolled foils, which were softer, were more likely to deform and move in the cladding pocket during friction bonding. It is believed that an optimum condition may lie somewhere in between these extremes.

Although the program is currently focusing most of its resources on development of the monolithic fuel form, progress is still being made in the development of U-Mo based dispersion fuels. Mini-plates (25 mm x 100 mm x 1.4 mm) were fabricated at 8.5 g U/cc loadings with various high silicon matrix materials including Al-4043 (~4.8% Si), Al-2 Si alloy, and Al + 2 Si mixture for testing in the RERTR-9A/B irradiation experiment in the ATR. Several full size plates were also fabricated at BWXT following process development at ANL at >8.0 g U/cc with Al-4043 and Al-2 Si alloy matrix materials.

2.2. Fuel Performance

The second key area of the fuel development program is fuel performance testing and characterization. Three irradiation campaigns were completed in the last year, the RERTR-7A, RERTR-7B, and RERTR-8. These experiments have provided the opportunity to further assess the behaviour of U-Mo fuels under irradiation and to demonstrate the performance of other key aspects of fuel design and fabrication.

The first mini-plates fabricated by hot isostatic pressing were irradiated in the RERTR-8 experiment [5]. The irradiation behaviour of the mini-plates was generally good and was consistent with that of friction bonded fuel plates. The bond between the fuel and cladding appeared robust and remained intact throughout irradiation. Fuel/clad interface behaviour similar to that of the friction bonded fuel plates was observed (where small voids were seen in the interaction product that formed between the fuel and cladding). Surface corrosion on the cladding was comparable to that observed in both roll bonded dispersion fuels and friction bonded monolithic fuels.

Additional understanding of the fission product retention and swelling characteristics of U-Mo fuel was gathered through additional testing and modelling. Fuel plates were irradiated to peak burnups in excess of 22% total uranium (fission density of approximately 8×10^{21} f/cm^3) in the RERTR-8 experiment. These tests showed that the fuel swelling rates remain consistent with that of the recrystalization phase and that the threshold for the onset of breakaway swelling has still not been reached. A breakthrough was also achieved in the ability to model fission product swelling. Fracture surface specimens were examined by scanning electron microscopy and the intergranular fission gas bubble size distribution for U-Mo fuels was established. When coupled with recent transmission electron microscopy work [6] that established the size of intragranular fission gas bubbles, a model to predict fission product swelling [7] was developed and validated through the first stage of fission product swelling (up to roughly $3x10^{21}$ f/cm^3).

Additional analysis was also performed in order to evaluate the impact of silicon on the interaction products that form at U-Mo and aluminium interfaces. Small punchings (~1 mm in diameter) were removed from U-Mo dispersion fuel plates irradiated in the RERTR-6 campaign and examined using scanning electron microscopy [8]. The fuel plates sampled contained Al-0.2% Si and Al-4043 (4.8% Si) matrix materials. The examinations showed that the very thin interaction layers associated with the higher silicon matrix materials was comparable in thickness to the as-fabricated interaction layer thickness. It was also shown that the interaction layer observed through x-ray mapping contained an appreciable amount of silicon. It is believed that the presence of this silicon simultaneously limited the interaction product growth and increased its irradiation stability. These observations are expected to translate readily to the fuel/clad interface behaviour in monolithic fuels.

3. Results and Discussion

A significant amount of testing is necessary to achieve the goal of delivering a qualified fuel by the end of 2011. The results from three key irradiation tests in 2008 will be used to evaluate the readiness of U-Mo monolithic fuels for qualification testing. The RERTR-9A/B mini-plate experiment will be used to determine the efficacy of fuel/clad interlayers (both silicon enhanced and zirconium diffusion barriers) to control the formation of detrimental interaction products. The AFIP-2 and AFIP-3 experiments will be used to evaluate the dimensional stability of large plates under irradiation. At the conclusion of these tests, the performance of the fuel will be evaluated and a decision to proceed with element testing will be made. The first set of elements tested will consist of standard fuel designs (i.e. simple aluminium clad U-Mo foils with the selected interlayer) and will be the basis of the report submitted to the NRC for qualification. Additional development will continue in parallel to develop U-Mo based monolithic fuels with burnable poisons and graded fuel zones (complex fuels). This development will be reported in an addendum to the original qualification report to expand the utilization envelope of the U-Mo monolithic fuel.

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AQUEOUS HOMOGENEOUS SOLUTION NUCLEAR REACTORS FOR THE PRODUCTION OF ^{99}MO AND OTHER SHORT-LIVED RADIOISOTOPES

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ABSTRACT

In June 2007, the IAEA convened an international meeting of technical experts from organisations with experience in the design and operation of aqueous homogeneous reactors (AHRs), solution based fuel handling, radioisotope production management as well as the recovery of ^{99}Mo from ^{235}U fission. Participants discussed the current technology of AHRs and associated radiochemical processes for radioisotopes separation; the technical and economic feasibility of design, construction and operation of an AHR and radioisotope processing facilities; and identified and defined future lines of activity where the Agency's effort will most effectively support related activities in different member states.

This paper discusses the outcomes from the meeting. Specific detail is provided on the principal advantages of the technology, as well as the challenges associated with further development and deployment. The status of solution reactors for fission-based medical isotope production is presented. A summary of other areas of potential utilization is also included. Finally, future IAEA plans in support of further development are presented.

1. Introduction

The use of aqueous homogeneous reactors (AHRs), also called solution reactors, for the production of fission-based medical isotopes is potentially advantageous because of their relatively lower cost; small critical mass; inherent passive safety; and simplified fuel handling, processing and purification characteristics. These advantages stem partly from the fluid nature of the fuel and partly from the homogeneous mixture of the fuel and moderator in that an AHR combines the attributes of liquid-fuel heterogeneous reactors with those of water-moderated heterogeneous reactors. If practical methods for handling a radioactive aqueous fuel system are implemented, the inherent simplicity of this type of reactor should result in considerable economic gains in the production of fission-based medical isotopes. In June 2007, the IAEA convened a meeting of 10 technical experts from 7 institutions in 5 countries to review all the relevant issues and make recommendations for future work and this paper presents the output of this meeting.

2. Advantages of homogeneous aqueous reactors for the production of fission-based medical isotopes

2.1 Reactor design flexibility and inherent nuclear safety characteristics

The flexibility of solution reactor design parameters is an important feature of the AHR concept that allows customized design configurations to satisfy safety requirements and meet or exceed isotope-production targets. The greater flexibility afforded by solution reactors with respect to core operating power range is an important advantage with respect to ^{99}Mo production demand. Solution reactors for isotope production could range from 50 to 300 kW.

The choice of fuel base and solution composition is contingent on core design, operating and product isotope processing strategy. Traditionally, uranyl-sulfate fuel was preferred over uranyl-nitrate because of its greater radiation stability. However, the distribution coefficient for ^{99}Mo extraction is higher from irradiated uranyl-nitrate solutions than from irradiated uranyl-sulfate solutions; consequently a nitrate fuel base is clearly more advantageous from a processing yield point of view. The fuel concentration is selected to minimize core volume/fissile mass, optimize processing efficiency, or both. Solution reactors are typically operated at 80°C and slightly below atmospheric pressure. The low operating fuel-solution temperature, power density, and pressure provides thermodynamic stability, minimizes potential safety risks and yet allow for sufficient flexibility to optimize ^{99}Mo production demands.

The inherent nuclear-safety characteristics of solution reactors are associated with the large negative density coefficient of reactivity in such systems. The reactivity effect resulting from the operation of solution reactors at power may be thought of as the superposition of two effects, namely: (1) an overall uniform volumetric expansion of the fuel solution due to the increase in fuel temperature and the formation of gas bubbles due to radiolysis; and (2) a corresponding density redistribution within the expanding volume in which the introduction of an equivalent void volume displaces fuel from regions of higher reactivity worth to regions of lower reactivity worth. The resulting density reduction is manifested in a large negative coefficient of reactivity which provides a self-limiting mechanism to terminate a reactivity excursion and provides inherent nuclear safety. Relevant experiments in the French CRAC and SILENE facilities have demonstrated these phenomena.

2.2 Efficient neutron utilization, elimination of targets, less post-processing uranium generated per curie of ^{99}Mo produced, and overall simpler waste management

A unique feature of using the solution reactor for fission-based medical-isotope production compared to conventional production is that the reactor fuel and target are one, consequently a solution reactor can produce the same amount of ^{99}Mo at 1/100th the power consumption and waste generation. Thus the potential advantages of utilizing solution reactor technology are lower reactor power, less waste heat, and a reduction by a factor of about 100 in the generation of spent fuel when compared with ^{99}Mo production by target irradiation in heterogeneous reactors.

When one considers waste management in terms of both spent-reactor-fuel and spent-target disposition, waste management for the solution reactor is far simpler. A solution reactor has no need for targets and, therefore all processes related to the fabrication, irradiation, disassembly and dissolution of targets are eliminated. Because these target-related processes result in the generation of both chemical and radioactive wastes, ^{99}Mo production in solution reactors can significantly reduce waste generation. Since the recovery and purification of ^{99}Mo from conventional targets after dissolution will be quite similar (if not identical) to that of a solution reactor, the solid and liquid wastes produced will be similar, except for uranium disposition. Uranium from the solution reactor is recycled and only disposed at the end of the fuel solution's viability (up to twenty years).

2.3 Efficient processing of other isotopes using off-gas extraction

In addition to ^{99}Mo, other radioisotopes used by the medical community can be processed more efficiently from a solution reactor. Radiolytic boiling enhances the off-gassing of volatile fission products from the fuel solution into the upper gas plenum of the reactor. A number of valuable radioisotopes such as, ^{133}Xe and ^{131}I, can be recovered from the off-gas. There is a large demand for ^{131}I, as it continues to be widely used for therapy of thyroid disorders. Further, higher specific activity achievable in the off-gas recovery makes it much more effective for radiolabelling, compared to traditional uranium target irradiation technology. ^{89}Sr and ^{90}Y are two more products of interest for similar recovery due to their proven therapeutic utility and increasing demands, in particular for ^{90}Y. While the conventional source of ^{90}Y is from a radioisotope generator housing the long-lived ^{90}Sr separated from the waste stream of reprocessing plants, the AHR approach could be a potential new source for direct recovery from irradiated uranium salt solution..

2.4 Less capital cost and potential lower operating costs

The core cooling, gas management, and control systems and auxiliary equipment will be relatively small and simple compared to current research reactor target systems due to the lower power of solution reactors. Isotope separation, purification and packaging systems should be very similar to current target system facilities. The relatively smaller, less complex solution reactor will be less costly to design and construct than traditional research type reactors.

Operating costs may be reduced through many of the improvement mechanisms mentioned above. Specifically a target-free process eliminates all related costs, including the costs of target waste handling and disposition. Any resources involved in the transport of the irradiated target to a processing facility will be saved as will product losses due to any intermediate cooling periods. Reactor control and operation is expected to be simpler potentially resulting in reduced staffing requirements.

3. Design Challenges

Although AHR technology is well characterized in the research environment, the capability of a solution reactor to perform a medical-isotope production mission in a long-term continuous steady-state mode of operation in the 100 to 300 kW range is not guaranteed. Specifically, many technical challenges must be addressed in transitioning the technology to a commercial industrial environment.

3.1 Isotope separation technology

Solution reactor operation for medical isotope production could be challenged by the chemical stability of the fuel solution induced by a high radiation environment without introducing new undesired complex chemical structures in the product isotope and/or chemical reactions with the solution being processed. Furthermore, the potential problems caused by the build-up of adsorption and fission products, their effect on reactor operation, and the subsequent recovery system is another challenge which must be addressed. In addition, the effects of build-up of corrosion products, materials leached from the recovery system, and chemical additions must also be analysed and optimized. If the fission product build-up and/or corrosion product effects are important, a means to clean up the fuel solution in concert with waste-management and economic considerations must be devised.

Another important effect that has not been fully characterized is the effect of molybdenum redox chemistry of high radiation fields that will accompany fuel cooled for less time than current practices. Because recovery is based on maintaining Mo in the (VI) oxidation state, its reduction to lower oxidation states would diminish both its sorption in the loading phase and it's stripping from the column in alkaline solution, where the lower oxidation-state Mo species precipitate in the column as hydrous metal oxides. Limited studies have shown that four hours after irradiation, effects are seen by lowering of ^{99}Mo distribution ratios, especially in sulfate media. Much more experimental work is required to understand and design for this effect.

3.2 Design optimisation

Several design parameters must be optimised during any specific design process. Two fuel solutions are currently being considered for solution reactors dedicated to radioisotope production, namely, uranyl-sulfate and uranyl-nitrate. As described above, sulfates facilate easier reactor operation while nitrates tend to optimise ^{99}Mo recovery. Also the selected uranium concentration in the fuel solution is a compromise between reactor optimization and ^{99}Mo separation efficiency. A lower uranium salt concentration in the fuel solution results in a larger Kd for Mo(VI) and therefore a more effective and efficient recovery of ^{99}Mo. As a result, the size of the recovery column can be smaller making washing of impurities more effective and obtaining a more concentrated product solution of the raw molybdenum from the column. However, a higher concentration of uranium in the solution will minimize the reactor fuel solution volume leading to a more compact reactor.

3.3 Increasing power beyond current operating experience

Historically, solution reactors have been used either in a research capacity to: (1) study nuclear kinetics phenomena associated with nuclear excursions; (2) as a neutron generator to study the effects of irradiation on materials; or (3) to generate radioisotopes. As a result, most reactor operations were transient in nature, or limited with respect to steady-state operation. Physically, the radiolysis gas and vapour that form at high power densities create bubbles that migrate to the surface of the solution. The resulting perturbations at the liquid surface may cause reactivity variations, as well as waves and sloshing effects making it difficult for the automatic rod control system to maintain steady state power conditions. These phenomena are closely related to power density and need to be examined carefully to avoid potential power instabilities or uncontrolled power transients. The design of the core tank may also need to be reconsidered. These instabilities, while detrimental to predictable production operations, pose a relatively small potential hazard provided the reactor vessel design can accommodate pressure transients due to liquid perturbations. The use of Low Enriched Uranium (LEU) fuel requires a greater volume of fuel and thus results in an increase in core solution height which potentially diminishes the reactivity variations induced by perturbation of the solution surface. Furthermore, a non-cylindrical core tank design would probably attenuate the instability phenomena, thus further strengthening safety.

3.4 Licensing solution reactors

Since no operating license applications involving solution reactor facilities for isotope production have been submitted, world-wide nuclear regulatory bodies have not developed specific, relevant regulations. Hazard analyses for solution reactors have indicated significantly lower hazard to workers, surrounding populations and the environment than those reactors currently addressed by regulatory bodies. New regulations appropriately addressing specific hazards associated with solution reactors for commercial isotope production will be necessary. Until these regulations are formulated and issued, it may be feasible to address these facilities in a manner similar to current research reactor standards with appropriate modifications as needed.

4. Status of solution reactors for fission-based medical isotope production

Medical Isotope Production Reactors are under development in China, Russia and the United States. Two fundamental technologies have been patented in the US, Europe and Russia. These are solution reactors using LEU solutions of a) uranyl-nitrate salt and b) uranyl-sulfate salt as the fuel. The ARGUS reactor, a 20 kW(th), High Enriched Uranium (HEU) solution reactor has been operated as an experimental development activity by Kurchatov Institute in Russia. Irradiated solution from this unit was processed to separate and purify ^{99}Mo to European and US pharmacopoeia standards. It should be noted that meeting minimum pharmacopoeia purity requirements alone may not be sufficient for specific formulations used in the eventual medical imaging procedure.

Fundamental research on hydrated metal oxide sorbents continues both in the U.S. at Argonne National Laboratory, and at the Kurchatov Institute and Ural Technological University in the Russian Federation. Three sorbents have been considered for molybdenum recovery: alumina (the classical inorganic sorbent for Mo recovery from acidic solutions), and two sorbents specifically designed by Thermoxid (Thermoxid Scientific and Production Company, Zarechnyi, Russia) for recovering ^{99}Mo from homogeneous reactor fuel solutions. There could be scope for also exploring the use of a product called polyzirconium compound (PZC of Kaken Co., Ltd., Hori, Mito-shi 310-0903 Japan) developed for replacing alumina in ^{99m}Tc-generators for low-specific activity ^{99}Mo,

5. Conclusion

The current technology level is well established within the performed research tests. The next step is to confirm that this new technology can be used in a day-to-day reliable production environment. Active participation by both pharmaceutical and commercial nuclear reactor industries will be necessary in order to successfully develop viable commercial applications of this technology. While the advantages are numerous, commercial markets must be involved in the establishment of an evolving technology in place of an existing well developed alternative.

5.1 Principal recommendations

- Formulate a scheme to address R&D needs and launch an IAEA Coordinated Research Project (CRP) to share information on solution reactors and medical isotope processing systems,
- Complete identified research activities based on documented technical challenges associated with solution reactor technology, isotope separation technology, commercial utilisation, economic/market analyses,
- LEU should be considered for all solution reactors for fission-based medical isotope production,
- Consider a bilateral or multilateral project to develop a prototype solution reactor for the production of fission-based medical isotopes,
- Involve radioisotope technologists and regulatory and pharmaceutical agencies early in any design process,
- Consider an IAEA Safety Guide on solution reactors for medical isotope production.

6. References and acknowledgments

As mentioned above, this paper represents the output of an IAEA meeting. Each of the below participants presented papers during the meeting which will be included in an IAEA TECDOC report being developed on this topic. The authors wish to acknowledge the participants' input and express our appreciation for their support.

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TRIGA MARK II
FIRST MOROCCAN RESEARCH REACTOR FACILITY

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ABSTRACT

The research reactor facility is located at the Nuclear Research Centre of Maamora (CENM), located approximately 25 kilometres north of the city of Rabat. This facility will enable CNESTEN, as the operating organisation, to fulfil its missions for the promotion of nuclear Science and technology applications in various social and economic sectors in Morocco, to contribute to the implementation of a national nuclear power program, and to assist the National Nuclear Authorities in monitoring nuclear activities for the protection of the public and the environment.

The reactor building includes a TRIGA Mark II research reactor with a nominal power level of 2000 kW (t), and equipped for a planned future upgrade to 3,000-kilowatts. This facility is the keystone structure of the Research Centre, which contains, in addition to the TRIGA reactor, extensively equipped laboratories and all associated support systems, structures, and supply facilities. The construction of the Nuclear Centre was carried out in collaboration with AREVA-TECHNICATOME of France and US GENERAL ATOMICS, and with the support of the International Atomic Energy Agency.

The CENM with its TRIGA reactor and fully equipped laboratories will give the Kingdom of Morocco its first nuclear installation with extensive capabilities. These will include the production of radioisotopes for medical, industrial and environmental uses, implementation of nuclear analytical techniques such as neutron activation analysis and non-destructive examination techniques, as well as carrying out basic research programs in solid state and reactor physics.

The TRIGA Mark II research reactor at CENM achieved initial criticality on May 2nd, 2007 at 13:30 with 71 fuel elements and culminated with the successful completion of full power endurance testing on September 6th, 2007.

1. Introduction

A 2 MW type TRIGA Mark-II research reactor has been installed at Nuclear Research Centre of Maamora (CENM), located at approximately 25 kilometres north of the city of Rabat. This is the first nuclear reactor in the kingdom of Morocco. The reactor will be utilised for research, manpower training and production of radioisotopes for their uses in medicine, agriculture and industry. The fuel loading of the reactor started in May 1st, 2007 and the reactor went critical in the May 02, 2007 at 1330 hours with 71 fuel elements. The reactor achieved full power (2 MW) level and all the required reactor testing were completed in September 2007. A key feature of the reactor is that the design has been developed with the capability of being easily upgraded to a steady state power level of 3 MW.

2. Description of the reactor and design parameters

2.1 Reactor shield

The reactor shield is a reinforced concrete structure standing approximately 9.0 m above the reactor hall floor. The beamports are installed in the shield structure with tubular penetrations through the concrete shield and the reactor tank water and they terminate either at the

reflector assembly or at the edge of the reactor core. The reactor core and the reflector assembly are located at the bottom of a 2.5 m diameter aluminium tank, 8.84 m deep. Approximately 7.2 m of demineralised water above the core provides the vertical shield. The radial shielding of the core is provided by 2.5 m of concrete having a minimum density of 2.88 g/cm3, water, ˜ 21 cm of graphite and 6.3 cm of lead.
The reactor is equipped with a thermal column. The outer face of this thermal column is shielded by a track-mounted door approximately 1257 mm thick. The door is recessed into the reactor shield structure, and is flush with the shield structure outer surface when closed.

2.2 Reactor Core

The reactor core is at the bottom of the reactor tank, which has an inside diameter of 2.5 m and a depth of 8.84 m. The reactor core and reflector assembly is a cylinder approximately 1.092 m in diameter and 0.53 m high. The reactor core consists of a lattice of fuel- Moderator elements, graphite dummy elements and control rods. The core is surrounded by a graphite reflector and a 6.3 cm thick lead gamma shield. This entire assembly is bolted to a support stand that rests on the bottom of the reactor tank. The outer wall of the reflector housing extends 0.81 m above the top of the core to ensure retention of sufficient water for after-heat removal in the event of a tank drain accident. Cooling of the core is provided by natural circulation up to full power level. In case of loss of cooling water in the reactor tank there is a provision of emergency core cooling system.
The top grid plate is aluminium plate of 3.17 cm thick. There are 121 holes of 3.82 cm diameter in six hexagonal bands around a central hole for locating the fuel- moderator and graphite dummy elements, the control rods and the pneumatic transfer tube. There are 6 holes of 1.58 cm near the G-ring of the grid plate for locating and providing support for the neutron source holder at alternate positions.
A hexagonal section can be removed from the centre of the upper grid plate for inserting specimens up to 11.2 cm in diameter. Two other sections are cut out of the upper grid plate, for inserting specimens up to 6.1 cm in diameter.
The bottom grid plate is an aluminium plate 3.17 cm thick which supports the entire weight of the core and provides accurate spacing between the fuel-moderator elements. The safety plate of 2.5 cm thick aluminium is provided to preclude the possibility of control rods falling out of the core.

The active part of each fuel-moderator element is approximately 3.63 cm in diameter and 38.1 cm long. The fuel is solid, homogeneous mixture of U-ZrH alloy containing 8.5% by weight Uranium enriched to about 19.7% U-235. The H/Zr ratio is approximately 1.65. Each element is clad with 0.051 cm thick stainless steel can. Two sections of graphite are inserted in the can, one above and one below the fuel, to serve as top and bottom reflectors for the core.

2.3 Experimental and Irradiation Facilities

The reactor has extensive experimental facilities. It can be used to provide intense fluxes of neutron and gamma for research, training and radioisotope production. The experimental and isotope production facility of the reactor consists of the following:

(a) The rotary specimen rack assembly (Lazy Susan) located in the circular well in the reflector assembly.
(b) Production of very short-lived radioisotopes is accomplished by a pneumatic transfer system located in the G-ring of the core.
(c) One central experimental tube (Central Thimble) in the middle of the core (A-ring) for in-core irradiation at the region of maximum neutron flux.
(e) Three radial beamports, one of which pierces the graphite reflector and terminates adjacent to the fuel.
(f) One tangential beamport.

(g) Other in-core irradiation facilities, such as hexagonal and triangular cut-outs etc.

3. Commissioning of the Reactor

The commissioning Program (CP) was prepared using applicable guidance provided in IAEA Safety Series No. 35-S2 (Ref. 1) and the USNRC document NUREG 1537 (Ref. 2). The tests are organized in the following stages:

3.1 Preoperational and pre-fuel loading tests;

Facility systems, auxiliary systems, reactor systems, and physical parameters were tested for the appropriate operating conditions prior to fuel transfer into the reactor core.

Systems were tested according to designated specifications, when applicable, and acceptable operation was established before core loading. Facility systems tested include security, fire, communication, and ventilation systems. Auxiliary systems tested include radiation monitoring, pool coolant, alarm, and interlock systems. Reactor systems tested are the control system, and operation of reactor components.

The final preparation prior to loading fuel into the reactor for initial criticality was to inspect and make dimensional measurements on each UZrH fuel element. The dimensional data for each fuel element was recorded and will be retained for the life of the facility.

3.2 Fuel loading and low power tests

Certain verifications of instrumentation and control system functions were completed before initialization of an approach to critical experiment by standard reciprocal source multiplication factor measurements. The reactor achieved initial criticality on May 2nd, 2007 at 1330 hours with 71 fuel elements with a reactor just supercritical by an excessive reactivity margin of $0.042. Reactor configuration at criticality was as follows:

- Sixty four (64) standard fuel elements containing 8.5 wt% U,
- two (2) instrumented fuel elements containing 8.5 wt% U,
- five (5) fuel followed control rods elements containing 8.5 wt% U,
- eighteen (18) graphite reflector elements,
- fissile core mass of 2,653 kg U-235.

After criticality, fuel was safely added to the reactor core to achieve:

- An intermediate core loading of 86 fuel elements,
- calibration of control rods,
- verification of the required shutdown reactivity margin and other tests,
- final operational core loading of 101 fuel elements in preparation for conducting tests and calibrations at intermediate thermal reactor power levels during the next phase of the commissioning program,
- the reactivity control system was completely re-calibrated with the final, operational core loading and the availability of an adequate shutdown safety margin was verified.

3.3 Power ascension and tests at low and intermediate power (<750 kW)

The thermal power calibrations are the initial operations of the reactor at low and intermediate power levels. Specifically, the reactor power was increased for the first time, to indicated power levels of 400 kW and 750 kW for the calibration of reactor thermal power.

During this phase of the commissioning, calibrations to determine the thermal power of the reactor were successfully determined and the instrumentation adjusted to indicate the correct thermal reactor power enabling the reactor to be operated at powers up to full licensed power of 2.0 MWt and to perform all necessary tests at different power levels

3.4 Power ascension and tests at licensed power.

This phase presents the initial operation of the reactor at its full rated power level in the natural convection mode. Specifically,

- The reactor power was increased step wise to determine the relationships between reactivity and fuel temperature as a function of reactor power,
- the linearity check of detector current as a function of reactor power was performed,
- the determination of individual worth of fuel rods as a function of their location within the reactor core was performed,
- the scram function at 110% of nominal power was performed satisfactorily

During this phase of the commissioning, reactor operations at power levels up to full licensed power were accomplished successfully enabling the commissioning staff to proceed with the final phase of commissioning; the full power endurance runs.

3.5 Full power endurance runs

This part concerns the final phase of the commissioning, which is the extended reactor operation at full power. Specifically, the reactor was operated at full power, 2.0 MW, for eight consecutive hours per day, for four consecutive days while closely monitoring reactor operating characteristics such as: fuel temperatures, excess reactivity and xenon buildup, radiation levels, water chemistry and all available primary and secondary cooling parameters were performed. During these runs at 2.0 MW, radiological surveys of the shield structure and the entire facility were performed.
Further, a fifth day of continuous reactor operations was used to map primary coolant core outlet temperatures as they related to individual core locations as well as testing of the secondary cooling system in various line-ups other than the normal 2 cooling tower and 2 cooling pump alignment.
A sixth partial day of reactor operations was used to test the operability of the rotary specimen rack and the pneumatic transfer system at power levels up to 250 kW.

During this phase of the commissioning, all reactor systems operated as intended. The CNESTEN 2MW Triga Mark II reactor commissioning is therefore considered completed.

4. Reactor Utilization

The reactor will play a central role in the development of CENM and nuclear applications in the kingdom of Morocco. The highest thermal neutron flux will be extensively utilized for radioisotope production (I-131,…), neutron activation analysis and neutron beam experiments, i.e. neutron scattering, prompt gamma analysis and neutron radiography.

5. Conclusion

In meeting all of the objectives of the commissioning of the reactor, it has been demonstrated that the CNESTEN Triga Mark II 2.0 MW reactor, with natural convection flow, is safe to operate at all licensed powers.

Furthermore, this first research reactor will enable CNESTEN to fulfill its missions for promotion of nuclear technology in the Kingdom of Morocco, contribute to the implementation of a national nuclear power program, and assist the state in monitoring nuclear activities for protection of the public and environment.

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STATUS OF RESEARCH REACTORS IN INDIA

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ABSTRACT

India has formulated a country specific three stage nuclear power programme which is essentially based on the availability of uranium and thorium deposits in the country. At present three research reactors are in operation at BARC, India. These reactors provide a wide platform to the scientists for conducting research in basic and applied sciences & engineering. These reactors also meet the requirements of radioisotopes. A Fast Breeder Test Reactor in operation at Kalpakkam, India has provided overall insight into various aspects related to development of the first 500 MW Fast Breeder reactor. Kamini a 30 kW research reactor, at Kalpakkam, uses U-233 fuel. A thorium fuel cycle-based Advanced Heavy Water Reactor (AHWR) is being developed at BARC. Construction of a critical facility for experimental study of core physics parameters of the AHWR has been completed and will be operational soon. A programme for up-gradation and refurbishment of the 50 year old Apsara reactor is being undertaken. To meet the increasing needs of research & radioisotopes, construction of a 30 MW high flux research reactor is planned.

1 Introduction

The first Indian Nuclear Research Reactor went critical on August 4, 1956 and the event marked the beginning of the success story of Indian Nuclear Programme. India has formulated a country specific three stage nuclear power programme which consists of design, construction and operation of Pressurised Heavy Water Reactors (PHWR) in the first stage. During the second stage Fast breeder reactors would be developed and operated which will also produce U233 from thorium. In the third stage of the programme, reactors using U233 based fuel would be developed. At present, Apsara, Cirus, and Dhruva research reactors are in operation at Bhabha Atomic Research Centre (BARC). All three research reactors are utilised for basic & applied research in science and engineering. These reactors also meet the requirements of radioisotopes for applications in the fields of medicine, agriculture and industry. Extensive refurbishment and safety up-gradation of the Cirus reactor was carried out after more than 35 years of operation. For the ageing Apsara, detailed core design changes and system modifications have been worked out to convert it into a LEU fuelled core and to upgrade the reactor. Over 150 research reactor-years of operation has provided valuable experience in the areas related to design, operation, maintenance of nuclear reactors. These research reactors have provided valuable experience and inputs for technology developments for the PHWRs of the first stage of our power programme. Construction of a critical facility for experimental study of core physics parameters of the Advanced Heavy Water Reactor (AHWR) has been completed and preparations for its first criticality are underway.

2 Apsara

Apsara is a swimming pool type, light water cooled and light water moderated research reactor with a maximum thermal neutron flux of 1.0 x 10^{13} n/cm^2/s at the rated power of 1.0 MW. The fuel used is High Enriched Uranium (HEU). The core is suspended from a movable trolley in a pool 8.4 M long, 2.9 M wide and 8 M deep filled with de-mineralised light water. The reactor core is supported by an aluminium grid plate having 49 positions on a 7 x 7 lattice for fuel elements, control elements, reflectors, irradiation holes, neutron source and fission counter. Four cadmium rods function as control rods. Three of these rods serve as coarse control rods and are also used to shut down the reactor. The fourth one is used to regulate the reactor power.

3 Cirus

Cirus was the second research reactor built in India. Cirus uses natural metallic uranium as fuel, heavy water as moderator and light water as a coolant. Cirus has a maximum thermal neutron flux of 6.7×10^{13} $n/cm^2/s$. The fuel assemblies are placed in a vertical aluminium reactor vessel having 199 lattice positions. Demineralised light water circulated in a closed loop is used as the primary coolant. In case normal cooling circuit is not available shutdown core cooling is ensured by gravity flow of water from a water storage tank, called "ball tank". This ensures shutdown core cooling for about 72 hours. Sea water is used as the secondary coolant. The reactor is housed in a steel containment building.

4 Dhruva

Dhruva is a 100 MW (th) tank type high flux research reactor with natural metallic uranium as fuel and heavy water as coolant, moderator and reflector. The maximum thermal neutron flux is 1.8×10^{14} $n/cm^2/sec$. The reactor core is contained in a cylindrical stainless steel vessel which is placed vertically in a light water filled stainless steel lined vault. There are a total of 146 lattice positions in the reactor vessel, out of which normally 127 positions are used for loading the fuel assemblies and 9 positions contain the cadmium shut-off rod. The remaining positions are used for isotope production and experimental facilities. Heat generated in the fuel assemblies in the core during operation is removed by the heavy water coolant circulated by main coolant pumps. For shutdown core cooling three auxiliary coolant pumps are provided and each auxiliary pump has two prime movers (one an electric motor with uninterrupted power supply and other a turbine driven by gravity flow of water). Also an Emergency Core Cooling System is provided to take care of Loss of Coolant Accident. The reactor and associated systems are housed in a rectangular concrete containment building.

5 Utilisation of Apsara, Cirus and Dhruva

All the three research reactors have been well utilised for basic and applied research, neutron radiography, nuclear detectors testing, radioisotope production, material testing, shielding experiments and human resource training and development.

The National Facility for Neutron Beam Research (NFNBR) has been created at BARC to cater to the needs of the Indian scientific community. Scientists from universities and national laboratories also use these facilities in research reactors through collaborative research projects. Many of these collaborations are being supported by University Grant Commission-DAE Consortium for Scientific Research (UGC-DAE CSR), Board of Research in Nuclear Sciences (BRNS), and other agencies. These research reactors are also utilised for conducting various engineering experiments. Some of the important experiments performed in these research reactors are listed here:

- Experiments with different combinations of shield models were carried out at Apsara for optimising the in-core shielding of the intermediate sodium heat exchanger of Prototype Fast Breeder Reactor. Results obtained from these experiments have also been utilised for validation of various computer codes used for shielding calculations. A large number of shielding experiments were also carried out for radiation streaming studies through penetrations and ducts of various shapes and sizes for the proposed AHWR.
- Flow pattern transition instability studies were carried out in Apsara by constructing a loop similar to the geometry of AHWR coolant circuit. The neutron radiography facility

was utilised to visualise flow pattern and also to measure void fraction which is an important parameter causing the flow pattern transition.

- Irradiation of various biological samples like plants, seeds, etc. was carried out in Apsara. The experiments carried out at Apsara in the field of biosciences relate to studies on different biological crop plants and ornamentals. These experiments have helped in the development of high yielding varieties both in food crops and in ornamentals.
- Towards development of Mixed Oxide (MOX) fuel, UO_2-PuO_2 fuel pins were test irradiated for stipulated burn up in Pressurised Water Loop (PWL) of Cirus reactor. Various design and manufacturing parameters were assessed through these tests. Towards utilisation of Thoria based fuel in PHWRs, an experimental assembly containing ThO_2-PuO_2 fuel pellets was successfully irradiated to a burn up of more than 15000 MWD/Te in PWL. Irradiation of intentionally defected fuel pin was carried out for activity transport studies. These studies have contributed significantly to the development of Nat U oxide and Nat U-Pu MOX fuels for power reactors.
- Zircaloy calandria tubes manufactured by different routes were test irradiated in Dhruva reactor to study their comparative in-pile growth behaviour. Assessment of radiation induced creep of Zirconium materials has been carried out along with radiation embrittlement studies of various structural materials used in Indian PHWRs. These studies resulted in finalisation of manufacturing route for the PHWR pressure tubes and calandria tubes
- Towards assessing the adaptability of the neutron noise measurement technique for monitoring healthiness of the in-core components in PHWRs, an experimental assembly with number of self-powered detectors was irradiated in Dhruva.

6 Safety Management of Research Reactors in BARC

Principal aim of research reactor safety is to keep radiation exposure of plant personnel and members of public as low as reasonably achievable under all operational states and accident conditions. To achieve this, the design process incorporates defence in depth philosophy through multiple levels of protection. To ensure that the research reactors are operated within the design limits and provisions for safe operation made in the design, do not degrade during the life of the research reactor, a safety management system is established.
A well structured organisational set-up with clearly defined roles and responsibilities of its constituents is an important ingredient of safety management system. There exists a well defined hierarchical structure and line of communication, authority and regulation among operating organisation, regulatory agency, health and safety organisation, maintenance and services organisation and quality groups, and experimenters, to facilitate smooth and safe functioning of the research reactors at BARC.

Documentation forms a vital part of the operational safety management of our research reactors. The documents such as Design Basis reports, Safety Analysis Report, Technical Specification, Quality Assurance manual, In-Service Inspection Programme, Emergency Operating Procedures, Radiation Emergency Procedures, Plan for the regular emergency exercises and tests, Operation & maintenance procedures for normal operation, etc. form part of regular operating documents. Strict adherence to the technical specification for operation ensures operational safety of the research reactor.

Approved Emergency Operating Procedures for postulated off-normal conditions are kept available in the respective control rooms. The number of such procedures is kept to a bare minimum to avoid dilution of their significance.

A detailed radiation emergency preparedness plan is prepared bringing out the responsibilities of various agencies and the follow-up actions required are unambiguously laid down.

Occupational health and safety is given prime importance to ensure that the radiation exposure of plant personnel, members of public and environment is kept well within prescribed limits and as low as achievable.

Area radiation monitors are provided in each research reactor in various areas of the plant and the status is displayed in the control room. The areas around the plant are monitored by periodically collecting samples of air, soil and plantation. All radiation workers are monitored to keep their radiation exposure well within stipulated limits.

A comprehensive quality assurance programme covering all the operational and maintenance activities is implemented to strengthen the safety culture and for enhancement of safety by assessment of operational performance. Periodic Internal Regulatory Inspections are carried out by services agency which is not reporting to O&M.

7 Ageing Management and Safety Upgrades

Ageing management aims at identifying refurbishment requirements and retrofit upgrades that need to be implemented to qualify systems, structures and components to current safety standards. After over 30 years of service, signs of ageing started showing in Cirus resulting in its reduced availability and excessive efforts for maintenance. Detailed ageing studies were conducted in a systematic manner for all systems, structures and components. Based on these studies, refurbishment requirements were identified and refurbishing outage of the reactor was taken from end 1997. After unloading of fuel from the core, further inspections were carried out. Extensive refurbishing was then carried out and the reactor was made operational again in October 2002.

Cirus Refurbishment

Ageing assessment mainly consisted of Samples/Coupons testing for material degradation, Non Destructive Examination using various techniques such as Visual and Remote visual inspection, Ultrasonic Tests, Eddy current tests, Radiography, etc. and Integrity tests such as pressure testing, leak checking etc. This was carried out in two phases, in the first phase the assessment that could be done with reactor in normal operating condition was completed. In the second phase, the assessment that needed defuelling of core and/or draining of process fluids was taken up. Few of the important works carried out are described briefly in the following paragraphs.

Reactor Vessel: Visual inspection of the Reactor Vessel (RV) tubes and their eddy current testing for wall thickness monitoring and volumetric examination for flaw detection was done. Condition of the tubes had been found to be good and no unacceptable flaws were detected. The expansion bellows of RV joins the vessel shell to top tube sheet by helium tight lap weld. The fluctuating stresses in the bellows were assessed to be well below the endurance limit of the material. From these studies it was concluded that there was no necessity to replace the Reactor Vessel.

Graphite reflector: The air cooled graphite reflector around the RV is in two annular segments and undergoes concurrent annealing with reactor in operation dissipating heat to coolant air flowing between the inner and outer segments. A thermal safety analysis was carried out by developing a computational model for predicting steady state and transient temperatures of the graphite reflectors. Experiments were then carried out at different power levels and the predictions were found to be in excellent agreement with the experimental observations. Sample blocks were removed from the reflector and estimations of stored energy using Differential Scanning Calorimetery was carried out. These studies showed that the stored energy levels were within acceptable limits and there was no requirement of carrying out annealing operation.

Flanged joints between aluminium extension pipes of RV and system SS piping: There are nine flange joints with elastomer gaskets between the aluminium tubes extending from the top of the reactor vessel and SS helium system pipelines, located above the top of upper steel thermal shield. Leakages observed during reactor operation were arrested by installing sealing clamps using remote handling gadgets.

Primary Coolant System Piping: The condition of inside surface of the coolant system piping was assessed by metallurgical examination of a sample piece and was found satisfactory. A new protective coating was applied in two layers on external surface of the underground portion of piping. Couplings were separately coated with mastic compounds.

Civil Structures: All major civil structures were inspected. The tests comprised of visual examination, Core sample analysis, Ultrasonic pulse velocity tests, Corrosion potential tests and Rebound hammer tests. General condition of the Reactor building, Annulus building, Reactor structure block, wet storage block, Reactor ventilation exhaust stack and dump tanks of main coolant system was found to be satisfactory.

Water seepage noticed from ball tank was repaired by lining the inside of the tank with glass fibre and epoxy. The ball tank was also qualified to meet the present seismic requirements by incorporating additional reinforcements in the central shaft region.

Safety upgrades: A detailed seismic analysis of Reactor Containment Building, Ball tank, Dump tanks, etc was carried out and was found that these structures meet the current seismic standards. Physical separation of some of the safety related components was carried out to guard against common cause failures due to fire, flooding etc.
Emergency ventilation exhaust system of the reactor building was earlier provided with an alkali scrubber and silver coated copper mesh filters for trapping radioiodine under accident conditions. These were replaced by the more efficient and easy to maintain combination filters made of activated charcoal and high efficiency particulate air filters.

Desalination unit: A low temperature vacuum evaporation process based desalination unit of 30 Te/day capacity, developed by the Desalination Division of BARC, has been coupled to the reactor. This is done to serve as a demonstration of using low grade heat from a research reactor for the purpose of desalination. The product water is being used to meet the water requirements of the reactor.

8 Apsara Upgradation

Apsara has been in operation for the last five decades and an extensive refurbishment of the reactor is planned to extend its useful life and also upgrade the reactor systems in line with the current safety standards wherever possible.

As a part of up-gradation, it is planned to replace the existing HEU fuelled core by a LEU fuelled core designed to operate at a higher power of 2.0 MW (th). This will enhance the maximum available thermal neutron flux to 6×10^{13} n/cm^2/s. The refurbished reactor will thus provide enhanced facilities for studies related to material irradiation, shielding studies, isotope production, neutron detector testing etc. The core of the upgraded reactor will be mounted on a grid plate having 64 positions arranged in 8 x 8 square array. The reactor core consists of 11 standard fuel assemblies, 4 control fuel assemblies and one water hole for irradiation/experiments. The core is surrounded by BeO assemblies which act as the reflector. 8 irradiation positions are located in the reflector region. The upgraded reactor will use U_3Si_2-Al dispersion fuel with enrichment limited to 19.75 % w/w U^{235}.

9 Critical facility for AHWR

A low power Critical Facility is under construction as a part of the over-all technology development program to support the design effort essential for evolution of new nuclear reactor systems utilising the abundant reserves of thorium available in our country. As a step in this direction, conceptual design and technical feasibility of the thorium fuel cycle-based AHWR has been established and its detailed design is in progress.

The Critical Facility has been designed to facilitate study of different core lattices based on various fuel types, moderator materials and reactivity control devices. The reactor is designed for a nominal fission power of 100 W with an average flux of 10^8 n/cm^2/sec. The reactor can be operated at higher power levels of upto 400 W to obtain a neutron flux of 10^9 n/cm^2/sec for short durations. One of the main features of the reactor is variable lattice pitch which provides flexibility to arrange fuel inside the core in a precise geometry at the desired pitch. For the initial set of experiments heavy water is used as the moderator and the reflector. Reactor criticality is achieved by the manual control of moderator level in the core.

10 Multi Purpose Research Reactor

In order to meet the large requirement of high specific activity radioisotopes and to augment the research and irradiation facilities available in the country a new Multi Purpose Research Reactor (MPRR) with enhanced neutron flux is planned to be built.

MPRR is a 30 MW (th) research reactor with a maximum thermal neutron flux of 6.7 x 10^{14} n/cm^2/sec and fast neutron flux of 1.7 x 10^{14} n/cm^2/sec. The reactor is fuelled with Low Enriched Uranium dispersion type fuel and uses light water as coolant and moderator. An annular heavy water reflector tank surrounding the core provides highly thermalised neutron flux region over a large radial distance from the core. The maximum thermal neutron flux available in the reflector region is 3.5 x 10^{14} n/cm^2/sec. Most of the irradiation positions are accommodated in the heavy water reflector tank. The core is cooled by light water flowing from bottom to top across the core, which in turn is cooled by cooling tower water recirculated in a closed loop. Reactor heat will be ultimately rejected to the atmosphere through a cooling tower. A set of auxiliary pumps will be provided to remove the core decay heat in case of failure of the main power supply or unavailability of the main coolant pumps. Provision has been made for natural convection cooling of the core in the event of non-availability of both main and auxiliary pumps or during prolonged outage of the reactor.

There are five in-core irradiation positions and fifteen positions in the reflector region for radioisotope production and material irradiation studies. Two fuel test loops, one cold Neutron Source, five tangential beam tubes, two Pneumatic carrier facilities and two positions for Neutron Transmutation Doping are located in the reflector region.

11 Fast Breeder Test Reactor

The Fast Breeder Test Reactor (FBTR) is a 40 MW (th), loop type, sodium cooled fast reactor. FBTR uses a mixture of plutonium carbide and natural uranium carbide as fuel. Heat generated in the reactor is removed by two primary sodium loops, and transferred to the secondary sodium loops. Each secondary sodium loop is provided with two once-through steam generator modules. The principal material of construction used for the reactor and coolant circuits is Stainless steel (SS 316).

The reactor has been utilised for studying the irradiation creep behaviour of Zr-Nb. being used in the Indian Pressurised Heavy Water Reactors. The present mission of FBTR is to irradiate the MOX fuel (29 % $Pu0_2$) chosen for PFBR to the target burn-up of 100 GWd/t. In the coming years, FBTR will be deployed for irradiation of advanced structural materials contemplated for future fast reactors. The experience in construction, commissioning and operation of FBTR for 20 years has provided sufficient feedback to enable the launch of the Prototype Fast Breeder Reactor Project.

12 Kamini

Kamini reactor built in 1996 is probably the only reactor operating with Uranium-233 fuel. It is tank type reactor with a power of 30 kW and maximum thermal neutron flux of ~ 10^{13} n/cm^2/s. The reactor fuel is an alloy of Uranium-233 and aluminium in the form of flat plates. The plates are assembled in an aluminium casing to form the fuel subassemblies. Demineralised light water is used as moderator, coolant as well as shield. Cooling of the reactor core is by natural convection. Provision has been made for cooling this water to maintain the water temperature at a steady value when the reactor is operated for long durations at higher powers. Start up and regulation of the reactor is done by adjusting the positions of two safety control plates made of cadmium sandwiched in aluminium. The reactor is mainly used for neutron radiography for fast reactor fuel development.

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PALLAS, THE NEW PETTEN RESEARCH AND ISOTOPE REACTOR

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ABSTRACT

At present the High Flux Reactor, HFR, Petten, is involved in fission and fusion power plant research and development, and carries out isotope production for medical and technical applications. The power plant research and development will continue to be focussed on new materials and components for higher reliability and efficiency of Generation-4 fission power plants, and fusion power plants. The HFR will be more than half a century old in 2015. To address these future research requirements a replacement is inevitable.
The test reactor building industry is presently producing conceptual designs for PALLAS for a research reactor in combination with isotope production. The business plan has been drafted, supporting a design with the main parameters: a 30-80 MWth flexible, reliable reactor core with neutron fluxes up to $5*10^{18}$ $n.m^{-2}$ at all power levels.

1. Introduction

The world population increases, but the growth might reach zero in this century. Even if prediction materializes, the need for energy will multiply. The sources will have to change: fossil fuels will decline sharply before the mid of the century. Depending on the world development scenarios, [1], the contribution of fission and fusion energy must rise considerable to satisfy the demand for electricity in the first place. Also in the EU the predictions for the fission and fusion energy contribution are highly significant [2,3].
The HFR in Petten was designed and built in the fifties of the 20th century for the development of fission energy. In 1984 the reactor vessel replacement prepared the HFR for the next 30 years of operation. In the period between 2015 and 2020 the second HFR reactor vessel will near its end of its design life together with other major components that will have to be refurbished in that period. Therefore, HFR replacement will be more economical.
The preparations for the replacement of the HFR with a new research reactor, PALLAS, capable of isotope production in parallel, have started. This paper presents the major PALLAS project steps. First it presents a projection of the demands from the fission and fusion research areas, and the customers for isotopes. These culminate in the major requirements for the research reactor. The reactor building industry has started producing a conceptual design in 2008. The licensing situation for research reactors in The Netherlands will be shortly addressed, as it sets the boundaries for the design.

2. Prospects for research reactors in the EU

Presently the HFR contributes greatly to the development of fission reactors and since the latter decades of the previous century it delivers experimental results for the design and construction of ITER and following fusion power plants. The production of isotopes was always part of the production palette, but the production followed the increasing demand in recent decades.

The research and development of power plants in the first half of this century will consist of the development of generation-4 reactors in particular the high temperature reactor. The work will encompass both structural and functional materials testing en demonstration of endurance operation of (sub) components such as fuel elements of graphite and erosion and corrosion of lead based coolants. Table 1 sums the main items for investigation in the next decades for fission alongside the expected major items for fusion. Recently EFDA defined their missions to build DEMO a first fusion power plant. These missions include testing of materials and components in high flux fission research reactors. The 14 MeV neutron testing environment, provided by IFMIF in the early twenties of this century, will not produce sufficient volume for component testing. Therefore, component tests can be performed in Pallas to complement IFMIF irradiations.

Material	**Fission: Gen-4, HTR**	**Fusion: ITER, DEMO**
Structural	Pressure vessel steel Canning steel nano microstructure. Graphite, composite	Low activation steels ODS steels Tungsten SiC composites
Functional	Fuel particle element Inert Matrix Actinides	Lithium ceramics Beryllium pebbles
Process	Fuel element test in all conditions Pb-Bi compatibility Graphite creep	Tritium release Pb-Li behaviour Bolt relaxation First wall simulation

Tab 1: Fission and fusion power reactor research & development

Several studies, FEUNMARR and ESFRI [4,5] were recently completed to provide a roadmap for test reactor capacity in the EU. In those studies the major test and isotope production devices the EU needs are RJH, PALLAS, IFMIF and MYRRHA roadmap mission EFDA science/technology. The demand for isotopes now centers around many, but in medical isotopes the Mo production forms the majority. This need not be the case for the rest of the century, but new medical applications of isotopes, lutetium for example, show there is a future in medical isotope production. Tens of isotopes hold promise for new applications in treatment and diagnosis. For the technical isotopes the market shows similar movements.
The business case for PALLAS, stretching far into this century, of course holds uncertainties. At present the combination of research and development for power reactors and isotope production is the best basis for the sound operation of PALLAS, if the requirements set for the investment and the operation can be met.

3. Requirements for PALLAS

The main test reactor issues addressing the demands from the major R&D customers are high neutron fluxes to simulate life times double or three times more quickly than the HFR. In particular the end of life conditions in Generation-4 permanent structures and components near the plasma of a fusion power plant require radiation damage in the order of 70 to 150 dpa, accumulated in three to maximally five years. The production of isotopes has its own drive for swiftly delivery of the necessary isotopes benefiting from high fluxes. The role of automated production will have to be increased for higher productivity without compromising the reliability of the production streams.

A core that can be rapidly re-configured to adapt the irradiation volume needed for each new cycle satisfies the need for flexible operation. The core will have a nominal power of 40 MWth with a reconfigurable minimum and maximum geometry allowing reduction and increase in irradiation volume. These volumes should be made available through a 30 MWth minimum and 80 MWth power core. This flexible core should be well predictable with an appreciable irradiation volume providing neutron fluxes up to $5*10^{18}$ $n.m^{-2}$ at all power levels. The resulting flexibility of the core will allow a more economic use of the reactor fuel and limit the production of waste to a minimum. This core property is an important issue for the broad support of the utilization of PALLAS in the public domain.

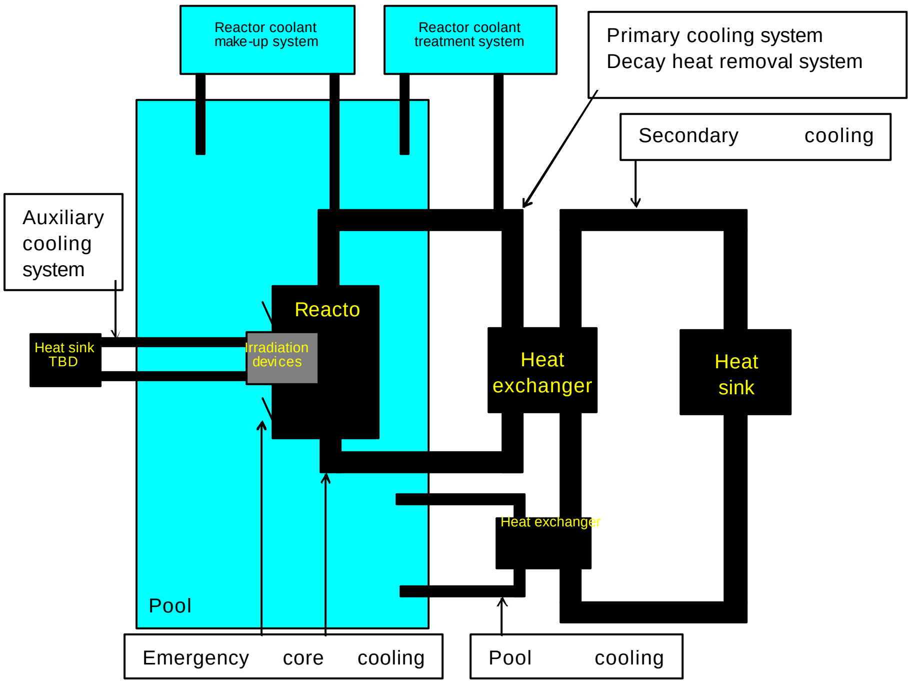

Fig 1. Lay-out of the cooling systems showing the dedicated secondary system

The secondary cooling system, Fig. 1, could address the requirement not to spoil low temperature heat generated by PALLAS. Different customers have an interest such as glass house farmers, fish and shrimp growers, and the mining industry to heat expanding gas from sources under the North Sea. In public information exchanges and hearings using the low temperature 30 to 80 MW heat can be an important asset.

Advantages	**Customer Satisfaction**	**Exploitation**	**Investment**	**Rating**
Double, threefold flux	2 to 3 times faster service	Neutral	Neutral	+++++
Flexible reflector	Neutral	Lower fuel cost Less waste	Design scope complex	+++++
Team workload	Neutral	Attractive schedules less night shifts	Automated handling	++++
Double hotcell	Increased availability	More efficient	1 more hotcell built	++++
Automated isotope production channels	Shorter term delivery	More efficient	Channel production	+++

Tank in Plus-pool	Smoother logistics	Smoother handling	Neutral	+++

Tab 2: PALLAS advantages

Tab. 2 summarizes the advantages of the PALLAS requirements over the present HFR. The high fluxes and flexible core arrangements will satisfy both the customers and the operator of PALLAS. Extra high fluxes might be generated using boosters, but this is only feasible for a limited number of experiments. Additional advantages, though of a lower overall impact, are the effect of isotope production automation improving the schedules for the staff. A four lobe pool concept (with the map of a plus sign) will improve the transfer and storage operations in the pool. The operation of two separate poolside hot-cells will strengthen the reliability of post-irradiation services. The cells are situated to the left and the right of the lobe for the reactor core. Each cell has its own lobe, with its own access.

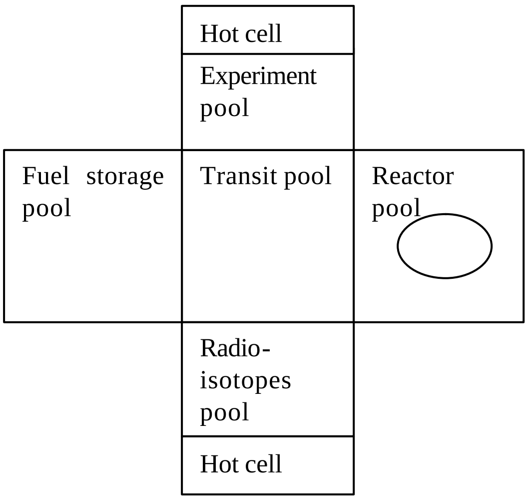

Fig 2. Schematic pool lay-out

4. The design and build project

JRC-IE, Petten, Mallinckrodt Medical, Technical University Delft, and NRG took the initiative to replace the HFR with a new research reactor: PALLAS. NRG has the lead in drawing up the requirements and initiate and manage the project leading to the conceptual design of PALLAS. The business plan has been drafted to support the design requirements. Special effort has been devoted to draw up the reactor requirements with optimal operation conditions, both from the technical, safety and the budget points of view.

The technical specification, including the general layout of the core, main buildings, and auxiliary equipment forms the basis for the conceptual designs to be provided by the research reactor builders. The conceptual design phase will prove whether the existing set of the requirements can be met, and will result in the optimal operation of PALLAS.

The tender approach is completely in line with the EU regulations for a restricted procedure (procedure with pre-selection) with 3 phases under fair competition. Selected test reactor builders are presently producing conceptual designs for PALLAS as a research reactor in combination with isotope production according to the requirements, revised during the consultation and dialogue phase of the tender procedure. After the delivery of three conceptual designs a selection will be made for one offer. The phase of the pre-design will reach from 2009 to early 2010, followed (after formal approval) by the detailed design phase lasting to

2011. The building of the reactor would lead to first criticality in 2015, starting full power regular operation, after a full year of test runs, in 2016.
It is expected that the financing of the detailed design and building will be firmly established in 2009. Prior to that the precise arrangements for reactor ownership, operator, and technical scientific acquisition will be established.
The licensing will follow the IAEA guidelines [6,7] interpreted for The Netherlands situation in the Nederlandse Veiligheids Regels, NVR. The present set of NVRs will be updated in the 2009, thus several issues will be discussed. Table 3 gives the five document clusters that must be treated together with the comments made by NRG. An example of an issue to be dealt with is the detailing of the requirements for the secondary shutdown system for PALLAS.

Document Cluster	**NRG comments**
Draft amendments to NS-R-4 by the Netherlands regulatory body	NS-R-4 is the principal guide for PALLAS. Special Netherlands requirements have been added.
Tentative list of standards applicable to PALLAS	Tentative selection by NRG of applicable (IAEA) standards.
Safety Codes and Guides in current regulation (2007)	Review of the current NVRs in Netherlands will yield a new set by the end of 2009.
Safety objectives in Netherlands legislation; dose and risk criteria	Criteria for doses to population, mortality risks, core meltdown frequency, and special safety requirements new reactors.
Legislative and regulatory framework	Description of framework including the licensing procedure and the governmental departments involved.

Tab 3: Research Reactor licensing issues in The Netherlands

5. Conclusion

The prospects for PALLAS, the contemporary replacement for the HFR, look bright. The development of the Generation-4 fission reactors and fusion power plants will provide many opportunities for PALLAS within the roadmap for research infrastructures in the EU.
The main requirements are neutron fluxes double to triple those of the HFR and a flexible core reducing fuel cost and waste production. Other improvements over the HFR are more automation in operation, double hot-cells alongside a plus shaped pool.
The tender process has reached the dialogue phase for fine tuning of the specifications followed by the end of the year the selection of the best conceptual design that must lead to first criticality of PALLAS in 2015.
The licensing procedure in The Netherlands has to be updated in 2009, that is a concern for the conceptual design but during final design and building the licence requirements will be stable.

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DEVELOPMENT STATUS OF IRRADIATION DEVICES FOR THE JULES HOROWITZ REACTOR

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ABSTRACT

After a brief description of the Jules Horowitz Reactor (JHR) facility building status, this paper will present in a first part the design work carried out on the irradiation devices.
For materials, pre-design studies concern mainly capsules containing NaK, with or without circulation. This type of device, inserted in the central hole of a JHR fuel element, ensures very good temperature homogeneity on a batch of samples and a high dpa rate. For LWR fuels, a set of loops and capsules adapted to PWR and BWR conditions will fulfil expected needs. As examples, the Adeline loop will be able to test a single experimental rod up to its operating limits. The Madison loop will be devoted to long-term testing of up to 4 instrumented fuel rods under normal conditions. The LORELEI-type capsule will implement LOCA tests.
In a second part, the paper describes the support facilities (laboratories, examination benches) also present in the JHR and enhancing the quality of the experiment. The conclusion underlines the international collaboration developed around the JHR project.

1. INTRODUCTION: CURRENT STATUS OF THE JHR PROJECT

The current development of nuclear energy will face in the first half of this century to a specific situation characterized by:

- Operation of the standard water reactors up to their end of life, facing to ageing process on irradiated materials and to maintenance of an expertise capability,
- Progressive commercial operation of new concepts of water reactors, using optimized fuels and plant cycle management,
- Development of innovative concepts, mainly based on fast neutron systems, either for energy production (electricity or heat) or for waste management (transmutation),
- Qualification of totally new components or materials for extreme conditions of use, such as under high neutronic flux of for fusion systems.

To fulfil the experimental knowledge needs coming from the large variety of materials and irradiation conditions to master, multipurpose research facilities are now key infrastructures (see ref.[1]), in complement of prediction capabilities gained thanks to progresses in the modelling. Within this frame, the JHR is designed to offer modern irradiation experimental capabilities for studying material & fuel behaviour under irradiation, mainly due to:

- High values of fast and thermal neutron fluxes in the core and high thermal neutron flux in the reflector+ (producing typically twice more material damages per year than available today in European MTRs),
- A large variety of experimental devices capable to reproduce environment conditions (pressure, temperature, flux, coolant chemistry…) of light water reactors (LWRs), of

gas cooled thermal or fast reactors, of sodium fast reactors, etc, including the development of new types of embarked components and instrumentation,

- The possibility i) to test highly instrumented samples under normal conditions and up to limits, in order to support advanced modelling for giving prediction on a broader range, ii) to manage degraded fuel samples after soliciting tests (e.g. safety tests), and iii) to perform a large variety of non destructive examinations on samples quickly after their irradiation and with a minimum of handling.

The reference power of the JHR is 100 MW. Presentations of this project and of its main features have already been done in several conferences [2], [3], [4].

The project is now at the development phase since beginning of 2006. After the construction permit delivery gained in the first half of 2007, excavation works started mid-2007 on the CEA Cadarache site in the southeast of France. Building construction is planned to start at the beginning of 2009. The first criticality is expected during the year 2014. The lifetime of the JHR will be at least of 50 years.

The safety assessment process, which is leading to the licensing of a reactor such as the JHR, is mainly composed by two phases. The first phase involving the issuing of the Preliminary Safety Report of the reactor and its analysis by the Safety Authority allows the beginning of the construction of the Reactor. The second phase, which is constituted by the issuing of the Provisory Safety Report of the reactor and its analysis by the Safety Authority, allows the start up of the reactor to reach the first criticality.

At this time the project is ending the first phase of the safety assessment process allowing the first concrete flooring at the beginning of 2009.

3. THE JULES HOROWITZ REACTOR IRRADIATION DEVICES UNDER STUDY

3.1 From the on-going device development to the JHR experimental capability

The design work of the JHR irradiation device park is driven by identified and expected future experimental needs. The starting of the feasibility and/or the development phases is related to the maturity of the demand and depends on the complexity of the device to set up. Consequently the device studies presented in this paper correspond to the current view of the long-term needs, which will be likely expressed during the coming decades. This development is a first initiative towards the set-up of the whole JHR experimental device park. It will also depend on the future irradiation market, and on the strategy applied by the JHR Consortium members or by the International Joint Program Committee.

3.2 Devices for material studies in Gen II – Gen III conditions

3.2.1 The CALIPSO integrated loop

Experimental needs in the nuclear material irradiation science concern mainly the characterization and the qualification of new cladding materials. They are characterized by i) the minimization of the temperature gradients between samples constituting the experimental batch, ii) the sample temperature stability versus time and iii) the possibility to apply a controlled stress to the specimen with in-situ measurement of the resulting strain. This last feature is a challenging technological issue, and is driven by both scientific knowledge (creep kinetics quantification) and operational stakes (minimization of budget and time to results by avoiding similar tests in hot cell).

To cope with this trend, the current design work is currently focused on the in-core CALIPSO NaK integrated loop. Placed in the central hole of the fuel element, this device shall be autonomous for long-term irradiations and embarks in a small volume all the components needed to ensure a forced convection in the test section:

- The technological feasibility of the electromagnetic pump for the nominal operating conditions (flow rate 2 m^3/h, ΔP = 1,25 bar up to 600°C, with an outer diameter of 80 mm maximum located out of the core region) is now confirmed. Pump characteristics include margins for sample-holders with a high pressure drop, and will ensure a maximum axial temperature difference between samples of 7,5°C.
- The heat exchanger, placed under the sample, is designed to remove the gamma heating deposited in the sample and the device structures. Different lengths will cover the standard LWR operation temperature range (250°C to 450°C), and the 600°C point will need a specific design.
- The head of the device holds the equipment box. It shall embark i) a connecting plate for instrumentation (about 50 signals), ii) the electric power supply of the pump and of the heater (to compensate if necessary an over-performance of the exchanger), iii) the control of the gas gap pressures (NaK blanket, external thermal barrier) and iv) the handling means. The objective is to design an upper head compatible with the maximum of in-core devices.

The sample–holder designed so far for the CALIPSO loop is based on the “ZO” concept used in the OSIRIS MTR. Designed firstly for LWR needs, it embarks 3 experimentation bases holding 3 pre-pressurized tubular samples placed at 120° on each. The device allows gaining quickly high fluencies thanks to a displacement per atom (dpa) rate up to 15 dpa/year. Strain measurements are performed after specimen unloading at the intercycle.

The design phase of the CALIPSO loop is now finished and some critical components (such as the electromagnetic pump or the embarked heat exchanger) have been studied more in details with the aim to launch very soon the manufacturing of prototypes. A contract for the detailed design and manufacturing of a CALIPSO prototype is under preparation and will be normally signed during the second quarter of 2008.

3.2.2 Other types of material experimental devices

A simpler design of the CALIPSO loop is also under study, operated with natural convection (no pump, but with an electrical heater). Called MICA, it will be able to hold a more sophisticated and instrumented sample-holder containing about 10 tubular samples placed vertically in the centre of the device. This sample-holder will allow applying a controlled biaxial stress on tubular samples (axial and circumferential) and to measure on-line the resulting strain. A first step towards this design is represented by the CEDRIC sample-holder, designed to apply a controlled uniaxial stress on specimens made of SiC fibres. Its operation in a CHOUCA device in Osiris is expected at mid-2008.

Stress corrosion cracking under irradiation in water coolant is taken into account through the conceptual design of a specific sample-holder allowing in-pile irradiation assisted cracking growth rate monitoring, thanks to the local electric potential drop measurement.

Other types of devices for material irradiation are planned, and mainly an in-reflector device capable to irradiate large specimens representative of power reactor pressure vessels.

One can also mention the on-going design of an out-of-pile NaK technological loop, which will be installed on the Cadarache site, as a common platform to test components belonging to future irradiation devices.

3.3 Devices for LWR fuel studies

Different types of devices are currently designed, driven by the type of experimental programme. As a first approach, one can classify the device design according to the solicitation applied to the fuel sample.

3.3.1 Water reactor fuel studies under nominal conditions

When the fuel rod failure is not an experimental objective or a risk, and when LWR conditions at the rod level are requested (temperature, pressure and coolant flow rate), the experiment will be set up preferably in the MADISON water loop. This loop will be put on a moving box in the JHR reflector, and will be capable to apply PWR or BWR conditions on the experimental load. This load will be constituted by a sample holder embarking up to 4 instrumented PWR or BWR-type geometry pre-irradiated fuel rods, with a fissile stack up to 600 mm and irradiated in an very homogeneous way. The target is to have less than 3% heterogeneity on the linear heat generation rate (LHGR) between any 2 rods. Of course 2 half-rods can replace each rod, if comparative or statistical results are a stake.

The standard instrumentation of each rod will be a thermocouple (e.g. for fuel central temperature measurement) and a Linear Variable Differential Transformer type (LVDT-type) sensor connected to one end of the rod and measuring on-line a given parameter (e.g. clad diameter, fission gas release…).

The feasibility study of this loop will be launched in Spring 2008 in collaboration with the Institute for Energy Technology (IFE), operator of the Halden research reactor (HBWR, Norway). Programs concerning fuel properties measurement versus burn-up or versus LHGR, fission gas release, or corrosion studies will be performed in the MADISON device. Some of these programmes will be long (several years), and the irradiation can be accelerated compare to power reactors conditions, however with respect of scientific constraints, in order to gain quickly high burn-ups and to gain knowledge on fuel end-of-life scenarii.

The MADISON-type concept will likely represent the standard performing and commercially attractive fuel irradiation service in JHR.

3.3.2 Water reactor fuel studies up to limits and under off-normal situations

Research of fuel product limits (e.g. class 2 ramps, internal over-pressurization, melting approach…), and post-failure behaviour studies under normal conditions (failed rod behaviour and fission product release studies), will be carried out in the ADELINE loop. The pre-design study of this PWR loop, also placed on a moving box in the JHR reflector, will be completed in Spring 2008. The in-pile part is based on the “jet-pump” flow-rate amplification system, to minimize the contaminated coolant quantity and flow-rate going to the loop components located in the experimental cubicle. The device head is designed for management of a degraded fuel rod, by tight connexion to the JHR alpha hot cell.

The out-of-pile part comprises in particular the fission product and fissile material purification system (resins, filters and degasser). Fission product concentration in the coolant can be measured either on-line (by gamma spectrometry or delayed neutron detection) or by sampling in the fission product laboratory, thanks to a specific line working at low flow-rate (a few l/h).

The device neutronic and thermal-hydraulical design will offer high performances and a large flexibility:

- The LHGR value of 500 W/cm with a 1% ^{235}U fresh UO_2 PWR fuel rod has been confirmed after pre-design studies.
- The standard power ramp rate will be up to 660 W/cm.min, with accuracy on the LHGR during the upper plateau, coming from the displacement system position, less than 5 %
- The inlet coolant temperature will be precisely controlled and will range from 280°C up to 320°C (from 150 W/cm). The outlet-inlet temperature difference will be of +5°C maximum, thanks to a high water speed (about 5 m/s).

The sample is one PWR rod (but including a possible diameter evolution up to 12,5 mm). The reference instrumentation is two sensors as for the MADISON sample. It can also be gas minitubes for internal free volumes sweeping and routing to the fission product laboratory.

3.3.3 *Water reactor fuel studies under accidental situations*

The safety experiments will constitute a key service offer by the JHR. For LOCA-type experiments, the feasibility study of the dedicated capsule LORELEI has been started from the end of 2007. The target is to be able to reproduce the typical temperature time history and the quenching phase of a LOCA sequence on a single instrumented fuel rod, based on a single-effect approach. The device itself will be heavily instrumented and capable to manage the post clad burst and the post quenching phases.

As the experimental needs are closely linked to the model prediction capabilities, and as these experiments are probably the most difficult ones to integrate in the JHR environment, there is a strategic way for defining, as soon as the device pre-design phase, the future experimental programmes. For this aim discussions or collaborations are being launched with utilities or institutes (e.g. EDF, OECD and IRSN).

The current LORELEI design will be able to fulfil a part of the LOCA demand. Other designs could be set up in a near future, depending on the physical mechanisms to explore. In particular, it is expected to adapt it for tests on a small bundle, in order to point out some fuel bundle effects. For these tests, the non-destructive examination benches (see § 4) will be a crucial support to gain quickly a first detailed status of the tested sample.

The design of a capsule for fast transient implementation is also planned. Based on a single-effect strategy, the target is to gain basis data on the activated phenomenology (e.g. fission gas release).

3.3.4 Other water reactor fuel device studies in progress or planned

The JHR reflector will also welcome simple boiling capsules for one instrumented LWR experimental fuel rod. Placed either in a fixed location or on a displacement system, it will be adapted to experiments, which don't necessitate representative LWR conditions outside the rod. The natural boiling conditions allow a large place around the rod, and this situation is favourable to fragile or cumbersome instrumentation (e.g. on-line axial and circumferential rod diameter measurement by metrology, as carried out by the DECOR sample-holder).

Small irradiation capsules with static gas gap around the sample are also foreseen. This type of device will be suitable for irradiations on small fuel samples with adapted geometry, for microstructure selection or material basis data obtaining.

Finally, it is worthwhile to point out that the MADISON-type concept (see § 3.3.1), could evolved and be adapted to other environment conditions, such for example a unit placed in a peripheral in-core position.

3.4 Fuel and material device studies for Gen IV power system conditions

Innovative development of a new generation of materials and fuels, which resist to high temperatures and fast neutron flux in different environments, is necessary for the development of these future reactors. There is a need to assess the behaviour under irradiation of a wide range of structural materials such as graphite (VHTR and MSR), austenitic and ferritic steels (VHTR, SFR, GFR, LFR), Ni based alloys (SCWR), ceramics (GFR)… These innovative structural materials are often common to fission and fusion applications. Experimental irradiations have to be carried out in order to study microstructural and dimensional evolution, but also the behaviour under stress. New fuels for the different Gen IV systems need also to be characterized or qualified in research reactors.

As the demand is less mature than for LWRs, the on-going studies address three topics:

- Materials behaviour under high temperature conditions: the conceptual design of an helium gas loop in the JHR core, at high temperature (700-1200°C) and high fast neutron flux (from 1 to 5 10^{14} n/cm²/), has started. This loop will be dedicated to separate effects experiments on selected materials, such as SiC/SiC or Oxide Dispersed.
- Gas thermal system fuels: This topic addresses high pressure and high temperature gas rig designed for the irradiation of compact stacks in the JHR reflector. The stack is swept by an inert gas at low flow rate to route the released fission gases to the fission product laboratory for quantitative measurements. A feasibility study has been performed in a European collaboration frame.
- Gas fast reactor fuels: The conceptual design of a gas rig or a gas loop in the JHR core has started. The chosen design has to cope with JHR constraints and will depend on the evolution of the demand. For this aim, the experimental feedback gained from the IRRDEMO experiment planned in BR2 will constitute a great added value.

4. NON DESTRUCTIVE EXAMINATION BENCHES

The JHR experimental process includes also non-destructive examination (NDE) stands which aim is to increase the experiment quality through NDE on full devices or sample holders by:

- Initial check of the experimental load state just before the beginning of irradiation (after transportation or insertion in the device),
- Adjustment of the experimental protocol after a first irradiation run (sample evolution, power tuning…),
- On the spot monitoring of the sample state after a test on the close-by stand located in the reactor pool and with limited handlings (e.g. geometrical changes after an off-normal transient, quantification of short half-life fission product distribution…).

The design phase of two underwater photonic imaging systems has just started end of 2007 in collaboration with VTT (FI). These systems will be respectively implanted in the reactor pool (for experiments with short decay or for quick measurements) and in the storage pool of nuclear auxiliary building (for longer examinations such as tomography). They should adapt for all sorts of experimental devices, even still lead-connected to ground-based experimental cubicles for some. Each system will accommodate on the same bench both quantitative gamma-emission and X-ray transmission scans which will allow performing detailed 3D images. Full device NDE can also be performed on an underwater neutron imaging bench installed on the reactor pool flooring.

After extraction from their carrier, samples will be also scrutinized in fuel and materials NDE hot cells, where one can find multipurpose test benches dealing with examination such as visual checks, sizing, corrosion thickness measurement, crack inspection, gamma and X-ray scans etc…

5. CONCLUSION: A FACILITY LARGELY OPEN TO THE INTERNATIONAL COLLABORATION

Besides the bilateral collaborations set-up for the development of some equipment already mentioned in this paper, the JHR facility design and operation is largely open to international collaboration. A first step was the signature in March 2007 of a Consortium Agreement for the reactor construction and operation. This consortium associates the European Atomic

Energy Community (Euratom/JRC), European fundamental and applied research institutes (CEA, CIEMAT, NRI, SCK•CEN, VTT), and two major companies: a utility (EDF) and a fuel vendor (Areva). India also joined the Consortium in January 2008.

As an important subsequent step, a new FP6 project (MTR+I3, "MTR plus" integrated infrastructure initiative) has been launched for 3 years from October 2006. This programme reinforces a major evolution toward the following key objectives:

- Building up the European MTR Community, including new facilities as well as existing ones (high performance MTRs as well as flexible small power facilities). Special attention is paid on complementarities between MTRs: operators training with staff exchanges, manufacturing practices, measurement best practices, opening accesses for testing experimental devices innovations.
- Establishment of the JHR as a new European MTR, because cross fertilization with existing European MTRs is important to take advantage of the available experience and of the impetus provided by the JHR project.
- Support state of the art design, fabrication and test of innovative irradiation devices or components with associated instrumentation. This addresses a comprehensive set of topics strategic for both present and future power reactors.

From 2008 is launched an International Joint Program in collaboration with OECD/NEA, for addressing issues of broad interest among the nuclear community, and gathering industry, academic institutions, safety bodies and research centres.

Discussions are also in progress with other countries, either through public research institutes or with industry, for joining the JHR Consortium or for defining a bilateral collaboration (Sweden, Germany…).

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