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Transactions

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Research Reactor Fuel Management
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TRANSACTIONS

7th International Topical Meeting

Research Reactor Fuel Management

European Nuclear Society (ENS)
in cooperation with the
International Atomic Energy Agency (IAEA)

Centre de Congrès
Aix-en-Provence, France

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International Topics and Education

U.S. DEPARTMENT OF ENERGY RESEARCH REACTOR INITIATIVES IN SUPPORT OF GLOBAL NUCLEAR NON-PROLIFERATION, NUCLEAR SECURITY AND NUCLEAR SAFETY

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The U.S. Department of Energy is engaged in a broad range of activities related to the safety and security of research reactors. Four principal activities being undertaken or planned by DOE support U.S. and global non-proliferation, security and nuclear safety goals.

The Foreign Research Reactor (FRR) Spent Nuclear Fuel (SNF) Acceptance Program was established in 1996 to help reduce stocks of highly enriched uranium (HEU) in research reactors around the world. To date, over 5,500 spent fuel assemblies from eligible research reactors in 27 countries have been accepted into the United States under this program. The spent fuel assemblies are stored at DOE sites, including those in South Carolina and Idaho. The Acceptance Policy will expire in May 2006, but some shipments will continue until May 2009.

A U.S.-Russian Summit initiative is being pursued to reduce inventories of highly enriched uranium (HEU) in each country. One of the key proposals includes acceleration of the U.S. and Russian Reduced Enrichment for Research and Test Reactors (RERTR) activities. This effort will accelerate development of reactor fuel designs to support conversion of research and test reactors from use of HEU to low enriched uranium fuel.

The DOE Integrated Research Reactor Safety Program (IRRSEP) is a five to seven year program initiated in 2002 that seeks to achieve and maintain a high level of nuclear safety in research reactors worldwide through the enhancement of national measures and international cooperation including, where appropriate, safety-related cooperation.

Finally, the changed global environment has highlighted the need to minimize the threat of use of radioactive sources for malevolent purposes. In this regard, the DOE is exploring opportunities to work with the research reactor community to create alternate forms of certain high-risk sources to make them less attractive for such purposes.

FUTURE NEEDS FOR MATERIAL TEST REACTORS IN EUROPE (FEUNMARR Findings)

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ABSTRACT

The European Commission agreed to a project to investigate the future European needs in Material Test Reactor (MTR) services. The project was carried out with the contribution of specialists in nuclear reactor materials and irradiation services for industrial, medical and basic research applications. The main conclusions are outlined in this paper. There will be a need for MTR experimental capabilities as long as nuclear power remains an appreciable contributor to the mix of energy production sources. Existing European MTRs have been very effective in providing valuable services both in Europe and worldwide. However, these reactors are ageing and measures should be taken for when the existing European MTRs will cease operation. While it is difficult to provide firm timetables, a new European MTR should be in place in about a decade from now.

1. Introduction

European research reactors have provided essential support for nuclear power programs over the last 30 years or more. More recently, they have also become an important service provider to the medical fraternity and isotope industry. However, by 2010 most of the Material Test Reactors (MTRs) in Europe will be at least 40 years old and many will require license renewal to remain operational, with some facing potential closure. In order to address the implications raised by the potential loss of several MTR facilities and associated services after 2010, the European Commission agreed to a project with the French Commissariat à l'Energie Atomique named FEUNMARR, to investigate the "Future E.U. Needs in Material Research Reactors". This report contains the main findings, conclusions and recommendations of the project.

The principal objectives for the work to be carried out in the FEUNMARR project were:

- To determine and make recommendations on the future irradiation needs for MTR facilities in Europe capable of carrying out experimental and safety programmes to support the continuation and further development of commercial nuclear energy.
- To determine the most likely types of problems, property measurements and examinations that will form the basis of the anticipated experimental and safety programmes.

Some MTRs also provide the facilities and irradiation conditions for branches of nuclear medicine, basic research and other industrial applications requiring neutron sources. These supporting MTR needs were also addressed in the FEUNMARR project, and are briefly covered in the conclusions of this report.

The FEUNMARR project was carried out in 2001 and 2002, with the support of a group constituted by representatives of the following industry and research organisations:

- CEA / Direction de l'Energie Nucléaire France
- JRC / IE Petten (HFR reactor) EU
- SCK•CEN Mol (BR2 reactor) Belgium
- Stoller Nuclear Fuel / NAC International UK
- NRI Rez, plc. Czech Republic
- Framatome ANP Germany
- Technicatome France
- Independent Consultant UK

In order to gather input on a broader basis, the FEUNMARR group organised an expert Workshop that was held in Cadarache in the period 27th February –1st March 2002. The detail reporting of that workshop, as well as the finding and conclusions of the FEUNMARR group were presented in a Final Report to the European Community at the end of 2002 [1]. This paper is an abridged version of the FEUNMARR Final Report.

2. Current status of European Material Test Reactors

With the current generation of nuclear power plants approaching maturity, the research conducted in material test reactors has gradually concentrated on issues related to improved operational economics, safety and reliability. In particular, nuclear operators have made operational changes and introduced measures to enhance safety, improve fuel utilisation and extend the lifetime of nuclear power plants. These initiatives have required and are still requiring specialised research, including experimental verifications and data from MTR simulations. At the same time, a certain rebirth of the industry is giving rise to new reactor concepts, which are likely to require innovative materials and technologies, as well as verification and testing in MTRs.

A few material test reactors in operation appear to have been successful, these being BR2 in Belgium, LVR15 in the Czech Republic, R2 in Sweden, HFR in the Netherlands, Osiris in France and the Halden reactor in Norway. The experience accumulated with these test reactors can be valuable in trying to determine what the main conditions to attain successful operation for an MTR can be also in the future.

Table 1. Main MTR Characteristics

Name	Age	Power[a]	Flux[b] (n/cm^{2}s)		Utilisation: % Share[d]		
		(MWth)	Thermal	Fast[c]	Nuclear	Medical	Other
BR2	39	60	$1\ 10^{15}$	$7\ 10^{14}$	51	28	21
HFR	41	45	$2.7\ 10^{14}$	$4.6\ 10^{14}$	45	45	10
LVR15	45	10	$1.5\ 10^{14}$	$3\ 10^{14}$	50	15	35
Halden	42	19	$1\ 10^{14}$	$2\ 10^{14}$ e)	100	0	0
Osiris	36	70	$2.7\ 10^{14}$	$4.5\ 10^{14}$	85	10	5
R2	42	50	$2.4\ 10^{14}$	$5\ 10^{14}$	45	25	35

Notes: a) typical operating, not maximum design power
b) neutron fluxes are indicative, they can vary significantly depending on local configuration
c) E > 0.1MeV.
d) generally shown as % of annual revenues, except for R2 which is share of total work-load.
e) maximum obtainable with booster rigs

The characteristics of the MTRs mentioned above (all having a power beyond 10 MWth) are shown in Table 1. These reactors continue to fulfil the requirements of the nuclear industry by providing sufficient thermal and fast neutron fluxes for representative irradiation of fuels and material samples.

Experience with European MTRs

All European MTRs have a variety of technical capabilities such as in-core instrumentation and controlled experimental conditions together with sufficient flexibility to cover a wide range of technical needs. These capabilities normally cover:

- Test rig design and fabrication
- In-reactor instrumentation
- Loops and coolant control
- Hot cells, including
- Re-fabrication techniques

Material test reactors can fulfil different tasks, such as for instance performing tests for nuclear power development concurrently with production of radioisotopes for medical applications. In Europe, conditions differ considerably from reactor to reactor. The Petten reactor is mostly dedicated to medical applications, although it also performs materials tests. The Halden reactor is on the other side of the spectrum, since its use is exclusively dedicated to experiments on nuclear fuels and materials. The other European test reactors tend to balance their use for various applications. In most cases, in Europe and elsewhere, MTRs have to rely on diverse customers and source of funding (see Table 1).

One way to achieve broader support and funding is by means of international projects, that is projects that are able to combine the interests of many participants and that are financed through cost-sharing arrangements among such participants. The OECD Halden Project, with about 100 organisations and companies supporting it in 20 countries, represents a very good example of how a joint international project should operate. Having a strong international dimension will be a decisive factor for the success of any new MTR initiatives.

Experience shows that more than one test reactor is needed to cover even the irradiations needed in the nuclear fuel and materials area alone. Each of the European MTRs mentioned earlier has its own special features and capabilities and customers may prefer one or another test reactor depending on schedule or on technical and financial conditions that are offered. As for the future, it is likely that up to ~2010 these MTRs will satisfy all current European needs (as well as many international) for neutron irradiation.

Ageing of European MTRs

By 2010, all of the European MTRs will be approaching or over 50 years of age and will all potentially be nearing the end of their operational life. Although it is likely that some applications will be made for continuation of operation beyond their anticipated lifetime, it is not at all certain that these will be granted, or would be cost-effective if plant improvements were required. It should also be noted that the risk of closure of some facilities on political and/or environmental grounds is at least as great as that due to age considerations. In summary, it is therefore not advisable to rely on the continuation of existing MTR facilities beyond ~2015, as demonstrated by the data in Figure 1 and the associated Table 2 below.

This figure and Table 2 show that all of the MTRs aged over 50 years and a large number (more than 50%) of the test reactors aged 40-50 have been shutdown permanently. This means that historically the lifetime of an MTR has generally been up to 40-50 years. The currently operating European MTRs that have a power beyond 10 MWth entered service in the early 1960’s. They are therefore approaching an age, which, from past experience, can be considered as their maximum expected lifetime.

Figure 1: MTRs (> 1MWth) Shut-Down According to Age

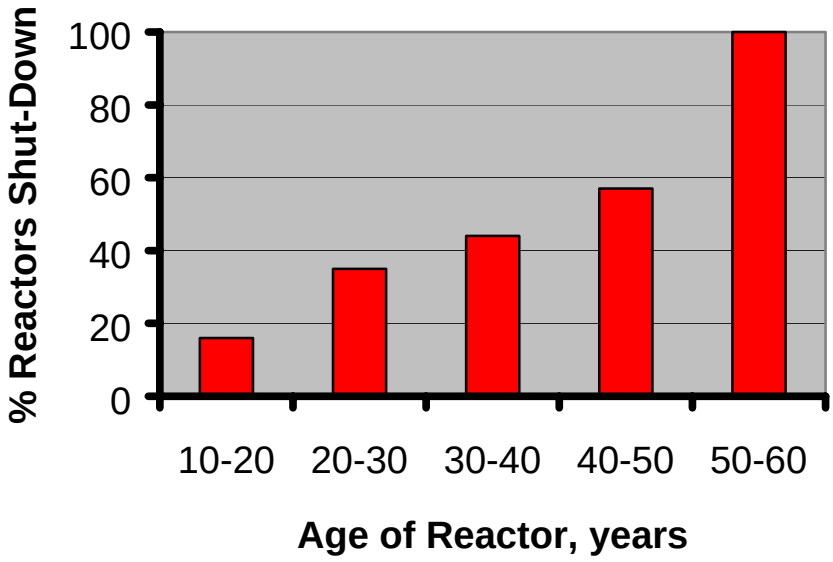


Table 2. Number of MTRs in Operation Versus their Age (Worldwide Statistic)

Period Commissioned	Age	Number MTRs (>1MWth) in Operation
1940-49	50-60	0
1950-59	40-50	14
1960-69	30-40	41
1970-79	20-30	15
1980-90	10-20	16

Proposals for new facilities in Europe

At European level, a proposal for a future MTR has been made by the French CEA. This MTR, denominated Jules Horowitz Reactor (JHR) is to be based at Cadarache, France. It is intended to study nuclear fuel materials in normal, off-normal and accident conditions. The pool-type MTR would have a maximum power of 100 MWth and an active core height of 0.6m. The design thermal flux is 5.2 10^{14} n/cm^{2}s and the fast flux (E > 0.9MeV) is 5 10^{14} n/cm^{2}s (fast flux above 0.1 MeV is about double as much). It is capable of producing 17 dpa/year.

Besides this MTR project, a proposal within Euratom is the Belgian MYRRHA concept for an irradiation facility around an Accelerator Driven System (ADS).

As a way to contribute to the build up of the European Research Area in the field of fission nuclear research, new facilities should optimise their interaction with the existing ones and benefit from the considerable experience that European-based test reactors have accumulated. In addition to irradiation and research needs, MTR facilities should also provide education and training in support to industry, nuclear research centres and universities, e.g. by offering courses in nuclear engineering and allied subjects. If nuclear power production is to continue in Europe, there is a clear need to maintain staff competence for the industry at large

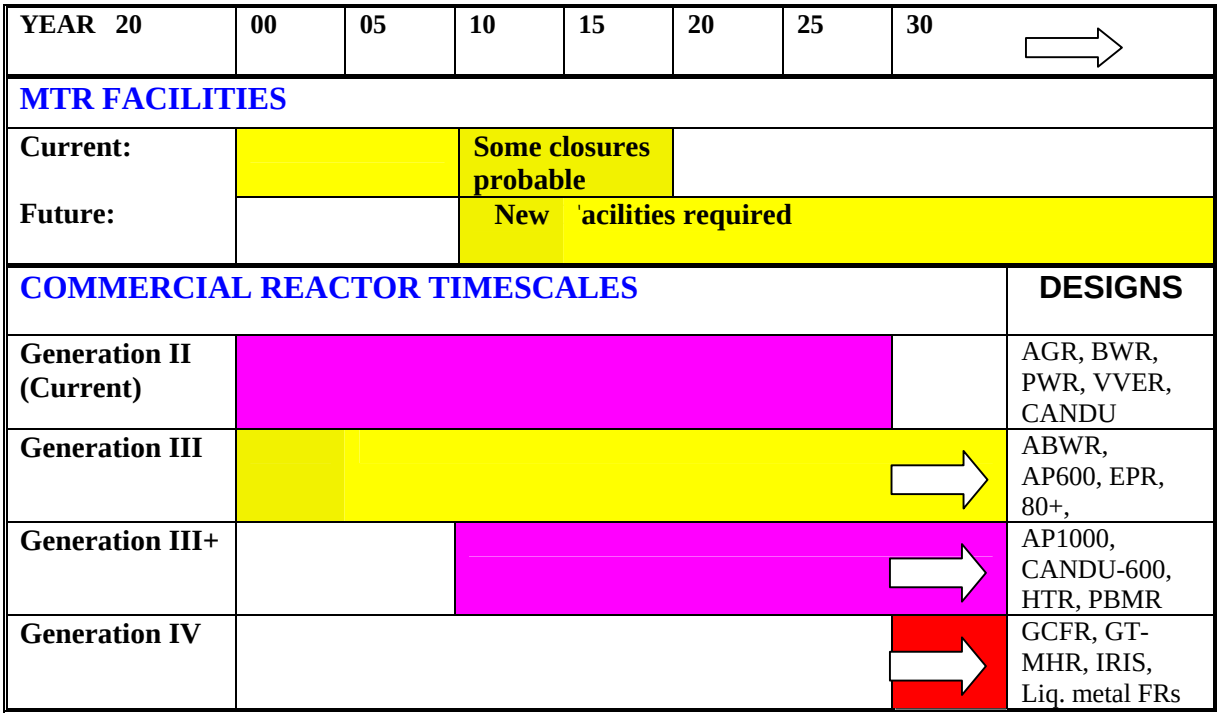
3. Current and future R&D requirements for nuclear power production

Commercial power reactors for electricity production operate in many of the countries comprising the EC and include a range of different designs (AGR, BWR, MAGNOX, PWR, and VVER). The majority of these Generation II plants are reaching the end of their design life (~30 years) but it is anticipated that some operators will acquire licenses to extend lifetimes up to 60 years.

At present, on a world-scale, there are increasing signs of a revival of nuclear energy brought about by several factors, such as environmental pollution by CO_2, price increase for fossil fuel and security of supply concerns. Plant lifetime extension programmes may delay the introduction of new reactors designs. However, studies are well-advanced for the European PWR (EPR), the ABWR and the high temperature reactor (HTR) as well as alternative international designs that may supersede the plants currently in operation in the EC.

Figure 2 provides a projected timescale for the introduction of various different candidate designs using the categories and timetable proposed by the US DOE. One interpretation of the present debate on energy needs, ranging alongside environmental factors raised first in the Kyoto Protocol, is the need to keep the nuclear option open. Europe is dominated by light water reactor technology and this is likely to remain the situation for a significant part of this century. A logical strategy is to replace current LWR reactors with an extension of existing, proven designs and technology. This is the basis for Generation III and III+ plant designs, incorporating passive safety features. It is envisaged that they could be required as early as 2010, with design optimisation and improvement in performance continuing over the next 20 years. Generation IV reactor designs, such as the gas-cooled fast reactor and others, are under discussion as a way of addressing the question of global sustainability. They also open the way for improvements in the management of nuclear materials to make them proliferation-resistant. In this regard Generation IV designs are characterized by long operating cycles (> 4 years) so that the containment vessel is opened infrequently. The latter has implications for new types of fuel assembly materials and the MTR testing that will be required to assess them, and then to qualify them.

Figure 2: Projected MTR and Commercial Reactor Timescales and Designs



For any particular reactor design, R&D requirements can be grouped into two main categories. The first can be referred to as *'start-up'* R&D, aimed to qualify both the reactor design and the materials to be used. This work has to be performed in advance of commercial operation. The second is *'on-going support'* which accompanies commercial operation up to the end of reactor life. The purpose of this is for developing new materials, qualifying new modes of operation and for addressing unanticipated issues that develop. Start-up R&D will be required for the Generation IV designs and, to some extent, for the Generation III and III+ designs, while ongoing R&D support will be needed for the reactors that will be in operation, regardless of design. In fact, history tells us that on-going R&D is needed and occurs regularly even for mature systems.

Current European MTRs may fulfil future R&D requirements only in the short and perhaps medium term. But for long term requirements to be met (at the beginning of the next decade), there is a clear need for new facilities to be built. Considering the lead-time before a new system can become operational, a decision to build a first new MTR in Europe is required in the very near future.

4. FEUNMARR conclusions and recommendations

Nuclear power production

Europe will strongly rely on the availability of technical and scientific infrastructure to continue providing the basis for the renewal of its nuclear electricity supply capability in the medium and long term. Given the age of current European MTRs, and anticipating continued R&D demand in the 21st century for material and fuel tests in support of nuclear energy production, there is a strategic need to renew material test reactors in Europe. Considering the lead-time before a new system can become operational, a decision to build a first new MTR in Europe should be taken in a very near future.

As first step in this direction, the CEA proposes to build the JHR reactor at the Cadarache site as an international initiative. The intent is to provide an effective tool that can help addressing needs of the international nuclear community in the medium and long term.

Following the example of for instance the OECD Halden Reactor Project structure, a new MTR should be operated as an user-oriented facility providing worldwide services and international R&D programmes for industry, regulators and R&D centres. However a single facility would be vulnerable to technical hold-ups such as routine or abnormal shutdowns. The availability of more than one facility also provides scope for competition to ensure economic irradiation services. For the coming decades, securing the required irradiation capability in Europe can be performed in an effective way by combining the use of a new reactor like the JHR with that of other test reactors. This is in accordance with the European Research Area concept.

Medical applications

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

Basic research and other techniques requiring neutrons

In the field of basic research, Europe can benefit from a strong situation with several dedicated reactors. The production of isotopes, the silicon doping and other applications, such as activation analysis or neutron-radiography, are considered as a side-application of neutron sources. Up to now these needs have been adequately met by existing reactors.

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RUSSIAN RESEARCH REACTOR FUEL RETURN TO THE RUSSIAN FEDERATION

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ABSTRACT

Beginning December 1999, and continuing to the present, representatives from the United States, the Russian Federation, and the International Atomic Energy Agency (IAEA) have been discussing a program to return to Russia Soviet- or Russian-supplied HEU fuel currently stored at foreign research reactors. Trilateral discussions among the United States, Russia, and the IAEA have identified more than 20 research reactors in 17 countries that have Soviet- or Russian-supplied fuel. Most of these reactors use at least some HEU fuel, and most have stocks of both fresh and irradiated fuel that must be carefully stored and managed for many years to come. The Russian Research Reactor Fuel Return Program is an important aspect of the US Government's commitment to cooperate with the other nations to prevent the proliferation of nuclear weapons and weapons-usable proliferation-attractive nuclear materials.

1. Background and History

Beginning December 1999, and continuing to the present, representatives from the United States, the Russian Federation, and the International Atomic Energy Agency (IAEA) have been discussing a program to return to Russia Soviet- or Russian-supplied HEU fuel currently stored at foreign research reactors. The primary goal of this Russian Research Reactor Fuel Return (RRRFR) Program is to advance U.S. and Russian nuclear nonproliferation objectives by eliminating stockpiles of high-enriched uranium (HEU) and encouraging eligible countries to convert their research reactors from HEU to low-enriched uranium (LEU) fuel upon availability, qualification, and licensing of suitable LEU fuel.

The goal of minimizing international commerce in HEU has been a pillar of U.S. nonproliferation policy since 1978. In that year, the Reduced Enrichment for Research and Test Reactors (RERTR) program was initiated to develop and qualify new LEU fuels that could replace HEU used in reactors of U.S. design, to aid reactor operators with the analyses required to optimize performance of LEU fuels, and to convert to LEU fuels. Russia has its own RERTR Program, under which it has significantly reduced enrichments on exported research reactor fuel and is working in cooperation with the US program to develop new LEU fuels suitable for use in Russian-designed research reactors.

To complement the RERTR program, the Department of Energy (DOE) established the Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) Acceptance Program in 1996. Under this program, the United States accepts specified types of U.S.-supplied spent and unused fresh fuel for management and disposition in the United States, on the condition that operators agree to convert their reactors to LEU as soon as practicable, and, in any event, to not use HEU fuel in the reactor after the program ends in 2006. In the years since the program began, the United States has received 23 (will be 24 by September) shipments of spent fuel from 27 countries containing more than 700 (update later) kilograms of HEU, enough for at least 30 crude nuclear weapons.

2. Research Reactor Fuel Return Program

Based on the success of the FRRSNF program, the US Department of Energy, supported by the Department of State, is working to bring about a similar effort in Russia. Trilateral discussions among the United States, Russia, and the IAEA in Vienna have identified more than 20 research reactors in 17 countries that have Soviet- or Russian-supplied fuel (Belarus, Bulgaria, China, Czech Republic, DPRK, Egypt, Germany, Hungary, Kazakhstan, Latvia, Libya, Poland, Romania, Ukraine, Uzbekistan, Vietnam, and Yugoslavia).Most of these reactors use at least some HEU fuel, and most have stocks of both fresh and irradiated fuel that must be carefully stored and managed for many years to come. The goal of DOE's/National Nuclear Security Administration (DOE/NNSA) is to help the Russian Federation develop a broad-based HEU minimization policy under which it would accept the return of spent and fresh HEU fuel from Soviet- or Russian-supplied foreign research reactors and develop new fuels that will allow conversion of such reactors to LEU. DOE/NNSA officials have led discussions with representatives from Russia's Ministry of Atomic Energy (MinAtom) and the IAEA on this issue, with the IAEA agreeing to provide technical and organizational support.

President Bush has committed his Administration to strong, effective cooperation with Russia and the other states of the FSU to reduce weapons of mass destruction and prevent their proliferation. To ensure that the promise of those programs is fully realized, the Administration has undertaken a detailed review of U.S. nonproliferation and threat reduction assistance to the Russian Federation. One conclusion of this review was the endorsement of the RRRFR Program as an important nonproliferation activity that should continue.

The RRRFR Program is an important aspect of the Administration's commitment to cooperate with the other nations to prevent the proliferation of nuclear weapons and weapons-usable proliferation-attractive nuclear materials.

The United States provides funding to the RRRFR program based on the following criteria:

- the fuel return program will include only existing FSU- or Russian Federation research/test reactors in eligible countries that possess nuclear fuel supplied by the FSU or the Russian Federation.
- any country desiring to return fuel to the Russian Federation must agree to (a) convert its operating research/test reactor(s) using Soviet- or Russian-supplied nuclear fuel to LEU as soon as (i) suitable LEU, licensed by the country's national regulatory authority, is available, and (ii) the reactor's existing inventory of HEU is exhausted; or (b) permanently shut down the reactor(s).
- whenever possible, all available HEU must be made available for return to the Russian Federation before any LEU is returned.
- all nuclear fuel to be delivered to the Russian Federation under the Program must be handled in accordance with IAEA documents INFCIRC/225/REV.4 and INFCIRC/153 (corrected), and subsequent revisions thereto**.**

The U.S. and Russian Governments and the IAEA will seek to encourage financial support from other IAEA Member States, where required, for the fuel return program to supplement any U.S. Government financial contributions.

3. Pilot Shipment

The first candidate for a pilot shipment of fuel to Russia is Uzbekistan, whose government has expressed a strong interest in participation in the RRRFR program. Uzbekistan possesses a VVR-SM research reactor at the Institute of Nuclear Physics, Uzbekistan Academy of Sciences, located in Ulugbek, about thirty kilometers northeast of Tashkent. It is a heavily used 10 Megawatt reactor of Soviet design that carries out an active program of research and isotope production. From its first criticality in 1959, it used 90 percent enriched HEU fuel, but was converted to 36 percent fuel in 1989. Over its lifetime, the reactor has generated a large amount of spent fuel and made a number of spent

fuel shipments to the reprocessing facility at Mayak between 1973 and 1992. Personnel who participated in the early shipments are still at the facility, so experience is maintained. The facility has the necessary room and hardware to accommodate the transportation cask.

DOE has provided assistance to improve the physical protection system of the VVR-SM reactor, but it is located in a politically volatile region of Central Asia. All parties agree that the spent HEU and any remaining fresh HEU should be relocated to a more secure environment, thus removing it as a potential proliferation risk. Similar nonproliferation, physical security, and safety concerns apply to other Soviet- or Russian-supplied research reactors.

On March 12, 2002, DOE and Uzbekistan's Ministry of Foreign Affairs signed an Agreement to facilitate cooperation between the parties for the return of Uzbekistan's Soviet- or Russian-supplied nuclear fuel to Russia. This Agreement also addresses conversion of the VVR-SM reactor from use of HEU to use of LEU; safety upgrades of the VVR-SM reactor control system as part of its HEU-to-LEU conversion; security enhancement of the VVR-SM reactor site and nuclear materials stored at the site; and the safe and secure storage of Uzbekistan's nuclear materials, including improving methods of physical protection, control, and accountability of nuclear materials to reduce the risk of theft or possible diversion. DOE contemplates entering into similar agreements with other countries that seek to return their Soviet- or Russian-supplied nuclear fuel to Russia and commit to convert from HEU to LEU use.

RESEARCH REACTOR EDUCATION AND TRAINING

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ABSTRACT

CORYS T.E.S.S. and TECHNICATOME present in this document some of the questions that can be rightfully raised concerning education and training of nuclear facilities' staffs. At first, some answers illustrate the tackled generic topics: importance of training, building of a training program, usable tools for training purposes. Afterwards, this paper deals more specifically with research reactors as an actual training tool. The pedagogical advantages they can bring are illustrated through an example consisting in the description of the AZUR facility training capabilities followed by the detailed experiences CORYS T.E.S.S. and TECHNICATOME have both gathered and keeps on gaining using research reactors for training means. The experience shows that this incomparable training material is not necessarily reserved to huge companies or organisations' numerous personnel. It offers enough flexibility to be adapted to the specific needs of a thinner audience. Thus research reactor staffs can also take advantages of this training method.

1. Introduction

CORYS T.E.S.S. and TECHNICATOME would like to express their interest in the set-up and organisation of a training centre for the energy industry. This consists in the delivery of an integrated training management system.
CORYS T.E.S.S. and TECHNICATOME propose products and services for energy industry training integrated in a unique concise training system, including distant learning software, simulation, CBT, training development, training support and training delivery on various tools including actual research reactors.

2. Importance Of Training For Nuclear Facilities

Training and retraining of operations personnel is essential for their acquiring the necessary knowledge, skills and qualification for operating a nuclear facility, and for effective feedback of experience including human based operating errors. However, training alone is not sufficient to ensure adequate performance on the workplace. Competence is only achieved by applying the skills and knowledge gained in training, with practice and experience to understand and successfully perform the work role.

3. Building A Training Program

To develop, assess, monitor and improve functional competence of nuclear facilities' operators, it is essential to use adequate tools, assessment methods, training support and management tools working

in close relation and warranting quality and traceability. The integrated training system proposed by CORYS T.E.S.S. and TECHNICATOME addresses this objective.
The Systematic Approach to Training (SAT) has been originally developed by the US DOE and is applied with success in the training process in nuclear power industry worldwide.
Five operational training engineering phases are applied to identify, develop, and adjust concrete demands in order to satisfy training objectives:
Analysing needs: working with designated managers, we identify needs to achieve the desired objectives; this phase consists of drafting of a report or specifications which become the reference document for subsequent phases.
Design: Based on conclusions to the previous analysis, appropriate solutions are sought and designed in this phase, which consists of specifying synthesised objectives in a document (i.e., the training specification).
Development : This phase consists of producing attention-retaining course materials and finding the relevant ways and means of achieving the defined objectives. All of these materials make up the core document known as the training file.
Implementation: The developed training course is put into application in compliance with the recommended programme and methods.
Assessment of the training course: This assessment is carried out at the end of the session, several months later with the trainees and their management. Feedback from the participants, management and instructors is analysed for proactive training management. This involves achieving a synthesis of all the assessment information produced during the operational progress of the training course by writing it up in an operational feedback report. The assessment measures the overall performance of the training course and can lead to changes in the different phases of training engineering.

4. Training Materials

Technically correct and educationally sound training materials are the backbone of a good training program. The materials should be amended, changed, or completely rewritten, as necessary.
Significant experience is required to develop or adapt materials that transmit clear, accurate, and easily understood information to trainees.
Written materials: Written materials should contain practical examples, exercises, and necessary numerical data. Facility specific information included in controlled documents and subject to change should be referenced rather that included in the written materials. Good written materials include illustrations and graphics to enhance the learning process. They should focus on the knowledge and skills required for the trainees' jobs and should not include unnecessary information.
Audio-visual media: A number of audio-visual media are used in training programs, including films, videotapes and videodiscs as well as slides. The advantage of these materials is the ability to transmit information which would otherwise be difficult to describe by oral presentation.
Models and mock-ups: Models are used to supplement classroom training, and mock-ups have proven valuable in developing some practical skills, scale models with cutaway sections are used in training for some complex components. The most suitable models are those corresponding to specific plant components. Training on mock-ups is very common. Mock-ups are especially suitable for teaching some skills to craftsmen.
Computer-based training: The purpose of CBT is to provide continual and consistent training for personnel. The combination of classroom lessons and self-paced, individualized instruction permits training at times convenient to the trainee's work schedules. A trainee can adjust the training duration to his ability to fully understand the presented subject. CBT can also be used in testing steps which include automatic scoring and record keeping.

Simulators: There are two different types of training materials relevant to simulator training :
- all manuals describing procedures and operating requirements which are used in the real plant operation,
- documents which have been developed especially for the purpose of simulator training.

In the case of simulator exercise, it is essential to define sufficient information on

– how to operate the simulator and the establishment of a pre programmed exercise scenario,
– the types of malfunctions and the sequence, method and timing of their insertion.

Research Reactors: They provide an essential practical approach, on the nuclear facilities, to the physical phenomena of reactors. Performing operations on an actual reactor is an irreplaceable experience in an education plan. It allows to federate complex theoretical knowledge with practical common sense. It also permits confirmed operators to ask and answer the questions that routine often hides.

5. Research Reactors For Better Training

Practical work performed on research reactors, to illustrate the theoretical lectures, provides trainees with a practical approach to the basic physical phenomena which govern nuclear reactor operation and reactor control by practical work concerning divergence, stepped operation and convergence.
The main advantages of research reactors are to:

– provide an understanding of real physical phenomena and be able to correlate these very easily with the changes in the reactor's main physical parameters,
– give trainees "hands-on" experience by allowing them to operate the reactor's controls, in the required nuclear safety conditions,
– be able to take part in core loading operations,
– be able to repeat operations thanks to the reactor's ease of operation, flexibility and easy access,
– carry out demonstration operations on equipment (radiological protection efficiency, nuclear detector response, etc.).

The practical work on research reactors can itself fulfil the pedagogical objectives specified for training; this practical work can also be completed by other training actions like, for example, practical work on simulators, which fully satisfies the pedagogical aims of training.

6. An Experimental Facility Used For Training: Example Of AZUR.

The AZUR pile was refurbished from mid 2001 to mid 2002, to extend its operating life beyond 2015. During this refurbishment the training interface was improved with:

– the integration, in the control console reorganisation, of the training objective to facilitate visualisation of the information displayed on the console and to facilitate operation of the facility by a trainee assisted by TECHNICATOME shift personnel (dual control),
– the facility of improved video facilities on the core video surveillance circuit, to allow the core to be seen from the control room during the divergence phase,
– modification of the control room by the facility of a dedicated trainee reception zone, for use outside the control phases, with general presentations (on nuclear physics and pile operation) and from which the control desk and core video surveillance can be seen (directly or via remote screen).

AZUR's main experimental possibilities are the following:

- finding critical parameters (absorber levels, water level, boron content, ...) for given loadings,
- measurement of the efficiency of different neutronic parameters (efficiency per ppm of boron, per mm of water, per mm of absorbents…),
- measurement of the temperature coefficient between 20° and 95° C,
- measurement of the integral efficiency of absorbents,
- measurement of the flux and power distribution, either by measurement of detector activation or by direct measurement on the fuel leaving the core (γ scanning),
- measurement of conversion factors after dissolution and appropriate chemical treatment,
- finding critical networks,
- control of reactor core loading and unloading under water,
- measurement of radiation protection,
- dynamic tests on nuclear instrumentation destined for other reactors,
- radiochemical measurement,
- irradiation of samples with short radioactive half-lives to study radioactivity and half-lives.

To these possibilities must be added those of carrying out programmed events (e.g.: incident simulations). These different possibilities are used according to the training objectives and to the corresponding pedagogical objectives fixed by those requesting the training.

7. CORYS T.E.S.S. And TECHNICATOME Experiences

CORYS T.E.S.S. and TECHNICATOME have both a huge experience in the field of training plans for nuclear facilities' staff.

For more than twenty years, CORYS T.E.S.S has used SILOETTE, a pool reactor located at CEA Grenoble for training power plant staff (EDF control operators and loading supervisors) as well as research reactor staff from various facilities (Marcoule's PHENIX, Saclay's OSIRIS, Cadarches's EOLE-MINERVE in France, Mol's BR2 in Belgium or Petten in the Netherlands).

TECHNICATOME has used AZUR at Cadarache for training Framatome and EDF operating and test engineers, for the start of the French electricity producing nuclear program, and to train operating teams for reactors operated by TECHNICATOME. On another hand, TECHNICATOME has been training more than 4000 seamen over the last 35 years in the management and maintenance of naval propulsion reactors. Up until 1990, this training course used the Land-based Prototype (PAT) which was, for training, followed by the New Generation Reactor (RNG).

An actual example of training session designed and performed by CORYS T.E.S.S. for the PHENIX operators and its computing department engineers follows:

In the specific case of operators, this session must be attempted every three years. So far, this one week session has been taking place in SILOETTE. It is divided into 5 practical exercises:

- Approach to criticality by loading fuel elements and extracting control rods.
- Thermal and fast neutron flux monitoring: axial and radial distribution in homogenous region and within different materials (water, aluminium, fuel, steel…)
- Control rod efficiency measurements: divergence method and reactivity-meter utilisation.
- Reactivity effect for different types of materials (neutral, absorber, reflector)
- Fission chamber calibration

The session is concluded with a written test insuring the correct understanding and assimilation.

After the recent decommissioning of SILOETTE and the unavailability of RNG for training purposes, CORYS T.E.S.S. and TECHNICATOME have chosen to join their efforts in finding a substitution way.
The retained solution that allows to follow up with the training on research reactors and to fulfil the same high level of requirements as formerly make use of AZUR, in Cadarache, and ISIS, in Saclay.
At the present time, several training sessions are designed for EDF staff as well as for seamen.
Two types of training sessions are suggested depending on the pedagogical objectives:

– Operator sessions:

This session takes place in AZUR and is suitable for power plant operators, seamen and eventually research reactor operators as well. In its most profitable version, this session is part of a larger scope that makes up the whole education either as initial instruction or as refresher course.
The session dedicated to EDF operators-to-be comprises practical works on the reactor as well as exercises achieved on a CORYS T.E.S.S designed basic principal power plant simulator which allow the trainer to bind the observed phenomena on the reactor with trainees' obvious reference that is to say their power plant.
This session taking place after a long theoretical course allows the participants to put in practice their knowledge in actual situations and to operate a reactor for real. The use of multifunctional and full scope simulator comes later on.

– Loading supervisor sessions:

This session can take place either in ISIS or in AZUR and is required by the nuclear safety organisation for EDF staff.
This session aims to point out the importance of the loading procedure and warn the trainees against the possible risks taken if this procedure is not fully respected.
To reach that objective, the trainees are made aware of the construction of a critical mass.

8. Conclusion

Research reactors have been an essential component in the development of Nuclear Energy right from its beginnings. At the start of the 21st century they remain an essential tool for maintaining and developing the skills of present and future nuclear reactor operator teams and designers.
Moreover, it has been recognized that full value and efficiency of a training system can only be achieved if all products and services are developed and integrated in a close relation. This is of particular importance if a clear and structured training framework is established in which all parts will fit seamlessly. For this reason, the practical work on research reactors is often a practical complement ("know-how" training) to theoretical training ("knowledge" training).
Research reactor personnel, for whom the educational approach is maybe not as systematic as the policies carried out by wider institutions such as electricity providers or navies, can however take great benefit of training on an experimental reactor. Moreover, this training mode should even be considered with greater attention in the case of research reactor operators, as they often can't make use of another type of practical training material, such as a relevant simulator.
A specifically designed session on research reactor taken into account the particular objectives to focus on is definitively achievable for any type of staff, from most kind of facility, and should be for a better efficiency integrated in a global training system.

THE SITUATION OF THE NUCLEAR ENGINEERING EDUCATION IN EUROPE

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ABSTRACT

The European Nuclear Engineering Network (ENEN) was initiated in mid 1999 by Belgium and accepted as an EU project by 1.1.2001 within the Fifth Framework Programme. Totally 17 European countries with 16 universities and 6 research organisations are participating with the aim towards closer co-operation on education in the nuclear engineering in Europe. It is developing a common "curriculum", identifying the most adequate organisations in Europe to perform nuclear education and training, and discussing financing schemes with stakeholders, including industry. Pilot education sessions are in the planning stage focusing on an international exchange programme of students and teachers. The goal is to create a EuroMasters degree using the instruments devised by the June 1999 Bologna European Council, e.g. the European Credit Transfer System. The outcome of the ENEN project should be a clear road map for the way ahead in nuclear engineering education in Europe. More detailed information on ENEN may also be taken from www3.sckcen.be/ENEN/.

1. Introduction

After the first initiatives by Belgium in mid 1999 a first meeting was held in Budapest in October 2000 with representatives from the following 17 European countries: Austria, Belgium, Czech Republic, Finland, France, Germany, Greece, Hungary, Italy, Netherlands, Romania, Slovenia, Slovak Republic, Spain, Sweden, Switzerland, United Kingdom.

During 2001 all necessary administrative procedures for an EU project were prepared under the co-ordination of Belgium, and the project was accepted and officially started on 1.1.2002 (project number FIR 1-CT-2001-80127) for a period of 2 years. The project covers different aspects, in fact generic, for a higher education network and includes the following tasks:

- summarise the state of the art,
- prerequisites to enter the education,
- education curricula, continuing education,
- teachers qualification,
- student and teachers mobility,
- identifying the most adequate organisations to perform the education and training,
- teaching practice such as distant learning, Euro-courses etc.,
- co-operation with research institutes that operate larger nuclear infrastructures,
- keep a finger on the pulse of industry,
- pilot education sessions are organised.

Due to the nature of the project, most participants are from universities, all with nuclear engineering education schemes. Other participants are from research centres with heavy nuclear infrastructure but also involved in education and training. Participants are both from EU countries, from "New Joiners", from countries without nuclear energy production as well as from countries with a major nuclear energy program. The work itself consists mainly in

- gathering and digesting the available information,
- formulate proposals for a global strategy,
- perform pilot education sessions,
- produce a clear roadmap for the way ahead in nuclear engineering education in Europe.

The two years period may be subdivided into two parts which are:

- During the first half of 2002 a survey was performed covering most of the European universities offering nuclear engineering education, the results were the basis for the following steps in ENEN and a major input to several of the 11 working packages described below.

- Towards the end of the second year a practical training course involving several member countries and student from several European countries should demonstrate the trans-border co-operation in Nuclear Engineering Education.

The project is a step towards farther reaching objectives e.g.

- the conservation of nuclear knowledge and expertise,
- the creation of a European higher education space,
- the integration of "New Joiners" in the EU.

2. ENEN participating institutions and description of the work packages (WP)

The following universities and research institutions are participating within the ENEN project:

Atominstitut der Österreichischen Universitäten (ATI)	A
K. U. Leuven Research and Development (KUL)	B
Universiteit Gent (RUG)	B
Université Catholique de Louvain (UCL)	B
Swiss Federal Institute of Technology Zürich (ETHZ)	CH
Czech Technical University (CTU)	CZ
Ustav jaderného vyzkumu REZ (NRI)	CZ
Technische Universität München (TUM)	D
Universidad Politecnica de Madrid (UPM)	E
CEA-INSTN	F
Helsinki University of Technology (HUT)	FIN
National Technical University of Athens (NTUA)	GR
Budapest University of Technology and Economics (BUTE)	HU
Consorzio Interuniversitario per la Ricerca Tecnologica Nucleare (CIRTEN)	I
Delft University of Technology (TUD)	NL
University "Politehnica" of Bucharest (UPB)	RO
Center of Technology and Engineering for Nuclear Projects (CITON)	RO
Kungl Tekniska Högskolan KTH)	S
Institut "Jozef Stefan" (IJS)	SLO
Slovak University of Technology (STUB)	SK
University of Birmingham (UOB)	UK

The ENEN project is subdivided into 11 working packages (WP), and each package is composed of one WP leader and two to four participating institutions. Below a short description of the individual WP`s and the participating institutions is presented. (The WP leading country symbol is in bold letters, with one country symbol several universities in the same country may be represented.)

Objectives of WP1 (**A**, RO, SK, S): The objective of WP1 is to survey the present state, strength and weaknesses, on nuclear education in Europe.

Objectives of WP2 (**F**, B, D, RO): WP2 aims at the design of the detailed organisation of the network, and the definition of its operation. These design and definitions should ensure that education and training provided by ENEN fulfil the academic standards and the professional requirements in order to efficiently contribute to maintain adequate competence at national and European levels in an attractive and economic way.

Objectives of WP3 (**S**, B, SLO, CZ): The objectives of this work package are to

- establish links with educational institutions in USA, Japan and other countries e.g. Korea, China etc.
- determine if the educational materials and methods they employ could be used in European educational institutions,

identify the position of international organisations e.g. IAEA, NEA on nuclear education.

Objectives of WP4 (**B**, SLO, SK): To establish the scientific and technical areas for education and training in nuclear sciences. Specify the different areas of (European) education: nuclear power engineering, radiation, nuclear science in medicine, nuclear material science, nuclear space science, nuclear science and societal issues (government, public, and ethics).

Objectives of WP5 (**SLO**, F, B, I, E): Develop a flexible and efficient curriculum for the European Master Degree in Nuclear Engineering taking into account the fact that nuclear engineering covers different professional disciplines and optimal use of available resources in Europe. The curriculum shall enable each individual graduate student to obtain high quality education tailored to her/his goals, needs and wishes, using potential and educational resources available at the European level as a whole.

Objectives of WP6 (**I**, B, GR): To offer a flexible, good standard and efficient basic education in Nuclear Engineering to graduates in other engineering fields, like Energy, Mechanical, Electrical, and Chemical Engineering, and to better satisfy the requirements of new employees by nuclear industry.

- To identify the prerequisites to attend postgraduate advanced courses at European level in Nuclear Engineering.
- To recommend on a learning methodology.
- To define the teachers profile.
- To offer a real European training tool.

Objectives of WP7 (**E**, H, S, I, NL, B): This WP aims to establish the feasibility of the organisation about Advanced Courses in the Nuclear Sciences and Engineering education for the benefit of post-graduate students in Europe.

Objectives of WP8 (**CZ**, H, CH): To establish the feasibility of organising practical training with respect to nuclear engineering education.

Objectives of WP9 (**SK**, H, CH): To achieve effective and continuous support for a safe and reliable operation of nuclear facilities through dedicated training courses for engineers of nuclear power plants (NPPs), other nuclear facilities and authorities. To provide transfer of knowledge on developments related to optimised operation of nuclear facilities.

Objectives of WP10a (**H**, CZ, SLO, SK, A, RO): To perform pilot sessions on nuclear education in order to ingrate the experience and the lessons learned in the handbook for a global network strategy. Because of the importance of the exercise, two groups will organise pilot sessions (WP10a and WP10b).

Objectives of WP 10b (**B**, S, CH, H,): To perform pilot sessions on nuclear education in order to ingrate the experience and the lessons learned in the handbook for a global network strategy. Because of the importance of the exercise, two groups will organise pilot sessions (WP10a and WP10b).

Objectives of WP11 (**B**, SLO, S): To reach the objectives of this concerted action.

3. Results of WP1:

As described above WP1 was performed and preliminary finished by the end of 2002. The main task was to collect the present state, strength and weaknesses of nuclear education in Europe. This was performed through questionnaires which were distributed through national co-ordinators to the respective universities, recollected and evaluated. It is important to note that research- and training-centers operated by utilities or industry are ***not*** included in this survey. To standardise the response "nuclear education" was subdivided into 14 categories as listed below:

1 Nuclear Energy: Introduction
2 Introduction to Nuclear Physics
3 Nuclear Reactors Theory
4 Experimental Reactor Physics
5 Nuclear Thermal-Hydraulics
6 Operation and Control
7 Reliability and Safety
8 Nuclear Fuel Cycle
9 Nuclear Materials
10 Radiochemistry
11 Radiation Protection and Nuclear Measurements
12 Advanced Courses
13 Others (includes also Fusion)
14 Waste Management and Decommissioning.

4. Conclusions and recommendations

Of the 23 countries that received ENEN questionnaires 79 universities and scientific institutions from 20 countries responded until the end of December 2002. Three countries have not responded and apparently offer no nuclear engineering education. Among the 20 countries which responded there is a significant difference in quality of the responses. It was observed that countries with less nuclear importance initially took much more care in completing the questionnaires than countries with a large NPP programme. Until June 2002 when about 80% of all the information was already available, some of these larger NPP countries made an effort to improve their survey. It seems that in some of these countries nuclear education is mainly in the hands of senior university staff with a strong dislike of questionnaires. This might also be connected to the age structure of university professors in nuclear engineering education.

In some countries universities offer specialised graduate and post-graduate programmes mostly in the field of medical and health physics. These courses were not detailed further with respect of titles and hours of lectures, and therefore cannot be definitely categorised into one of the 14 categories. For these reasons they were not considered in the evaluation and the diagrams. The same is valid for individual student’s projects.

In the diagrams, the total hours of lectures or courses are plotted instead of the European Credit Transfer System (ECTS) points due to the following reasons:

- Only a small fraction of universities quoted ECTS points for their courses.
- The calculation of ECTS points was generally inconsistent among the participating universities.

From all surveyed countries only two countries offer lectures and courses that can be assigned to all of the 14 thematic categories. In countries with a large NPP programme it is obvious that universities offer basic and advanced nuclear education but a large part of the nuclear education is performed by the NPP operators themselves in centralised training centres, these programmes are obviously not included in this survey. It is interesting to note that countries with no NPP programme apparently offer a large variety of nuclear education courses. A few countries feature almost exclusively theoretical lectures dominated by advanced courses, but provides very little practical courses or seminars but

most of the participating countries theory, experiments and seminars are combined in a well-balanced way. As far as it concerns the teaching language, 15 out of 20 countries offer a significant number of lectures and practical courses in English. Concerning the coverage of the 14 categories by European universities the following conclusions can be drawn:

- From all 14 categories only 2 categories show an overall small coverage which are
  Nuclear Fuel Cycle
  Waste Management and Decommissioning.
- All other categories are fairly well covered and evenly distributed among European universities.
- A very high coverage is in the category “Radiation Protection and Nuclear Measurements”. Astonishingly enough these categories are not offered in two major NPP countries.
- Another difficulty in comparing the nuclear educational system among European universities are the different educational systems mainly in France, Germany and the UK which presently cannot be placed under one umbrella. A positive step forward is the introduction of the European Credit Transfer System (ECTS), however the allocation of ECTS points needs a more strict and better definition among the European countries.
- Management of nuclear knowledge is presently an important issue among various institutions. The OECD/NEA recently has published an information booklet on nuclear education [1], at the IAEA a group is working on the management of nuclear education on an international scale [2,3], in Germany a centre of competence [4] has been established to improve the nuclear education in the coming years. The ENEN project has been also presented at a recent conference [5]. In summary it is highly recommended to co-operate closely with similar programs to avoid duplication and to optimise the output.

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Session 2

Fuel development, qualification, fabrication and licensing

QUALIFICATION AND LICENSING OF U-Mo FUEL

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ABSTRACT

The international effort to develop, qualify, and license U-Mo dispersion fuels is being carried out by programs in the United States, France, Argentina, and Russia. The first three programs are currently engaged in testing full-sized fuel plates and elements in the range of 6 to 8 gU/cm^3, fabricated using U-Mo powder produced by three commercially feasible methods. The Russian program is testing fuels with densities between 5 and 6 gU/cm^3, which are at the fabrication limits of the extrusion processes used there. Updated plans and schedules for these qualification tests are presented.

The common goal of the programs is to produce a fuel that can be licensed for use in research and test reactors. Initial contacts have been made with the licensing authorities in the USA, France, and Argentina. Plans for obtaining regulatory approval of plate-type U-Mo fuel and the process for obtaining regulatory approval in Russia are discussed.

1. Introduction

An international effort is underway to develop, qualify, and license a high-density fuel based on γ-phase U-Mo alloy dispersed in aluminum for application in plate-type reactors of Western design, tube-type reactors of Russian design, and pin-type reactors of the Canadian MAPLE design. This development work has been undertaken to provide fuels with the higher densities needed to extend the use of low-enriched uranium (LEU) to those reactors requiring higher densities than available in silicide dispersions and to provide a fuel that can be more easily reprocessed than the silicide.

The principal programs and/or organizations involved in qualifying plate-type fuels are the U.S. Reduced Enrichment for Research and Test Reactors (RERTR) program at Argonne National Laboratory (ANL), since March 1996 [1]; the French U-Mo Group, (CEA, CERCA, COGEMA, Framatome-ANP, and Technicatome), since 1999 [2]; and the Argentine Comisión Nacional de Energía Atómica (CNEA), since 2000 [3]. In January 2000, members of the RERTR program and the French U-Mo Group signed an agreement to collaborate in the areas of irradiation behavior, modeling, and licensing. At the beginning of February 2003, the French U-Mo Group, the RERTR program, and the Australian Nuclear Science and Technology Organization (ANSTO), which has early need for qualified U-Mo fuel, agreed to consider jointly funding an enhanced French program with the potential

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to obtain generic licensing approval early in 2006 by safety authorities such as the French DGSNR (Direction Générale de la Sûreté Nucléaire et de la Radioprotection) and/or the U.S. NRC. In the early stages of the U-Mo development work, the RERTR program collaborated with the Korea Atomic Energy Research Institute (KAERI) to assess the properties of dispersions made with atomized powder produced by a KAERI process and atomizer. However, as a result of a disputed patent obtained by KAERI in the U.S. and other countries, in which claims extended to all U-Mo dispersions containing spherical powder, whether made by the KAERI process or not, all collaboration between the RERTR program and KAERI was suspended in 2000.

In Russia, a parallel Russian RERTR program funded by the Russian Ministry of Atomic Energy (MINATOM) and the U.S. RERTR program has been working since 1999 to develop and qualify U-Mo fuel for use in Russian-designed research and test reactors. The principal organizations involved in the work are the All-Russia Research and Development Institute of Inorganic Materials (VNIINM), the Novosibirsk Chemical Concentrates Plant (NCCP), the Research and Development Institute of Power Engineering (RDIPE), the Institute of Physics and Power Engineering (IPPE), and the Joint Stock Company "TVEL." Uranium-molybdenum dispersion fuel is not new in Russia—U-Mo dispersed in magnesium was used for many years in the AM reactor of the first Russian Nuclear Power Plant in the Institute of Physics and Power Engineering (IPPE) in Obninsk. The alloy contained 9 wt.% molybdenum (U-9Mo), and, although the meat behavior was different than for U-Mo alloy-Al dispersions since U-Mo does not react with magnesium, extensive data are available in the Russian literature on the behavior of the U-Mo alloy itself.

Most Russian research and test reactors currently use tube-type fuel assemblies, for which the tubes are produced by extrusion [4]. Since extruding high-density dispersion fuel tubes is quite difficult, VNIINM proposed the use of dispersion fuel pins, which offer fewer technological challenges and, hence, offer the possibility of higher density [5]. Pins offer the added (economic) advantage that a single pin type, with perhaps small differences in pin diameter, can be used to replace many sizes and shapes of fuel tubes.

This paper reports progress on qualification and licensing of U-Mo fuel for the Western plate-type and Russian tube-type reactors.

2. Qualification of U-Mo Fuel

2.1 Definition and General Comments

We define a fuel to be qualified when all of the information required for a regulatory authority to approve its use has been acquired and documented in a qualification report. Licensing is the action of the regulatory authority to approve use of the fuel in a specific reactor, since the suitability of a fuel must be judged with respect to the conditions that the fuel may experience in that reactor. In order to facilitate the licensing of a number of reactors to use the same fuel, a regulatory authority may agree to review the qualification report on a generic basis and issue an approval for its use within an envelope of conditions covered by the qualification tests (as did NRC in issuing NUREG-1313 for U_3Si_2 [6]). Such a generic approval is not a license to use the fuel; each reactor must show that its intended use fits within the approved envelope. Generic approval of a fuel may be viewed both as the end of fuel qualification and the beginning of licensing. The RERTR program and the French U-Mo Group have both chosen to pursue generic approval as the final stage of qualification. In this paper, however, this subject will be discussed in conjunction with licensing.

In addition to qualifying the fuel itself, the fuel fabricator must be qualified. To be qualified, the fabricator must demonstrate its ability to manufacture, using a particular type of fuel powder, fuel plates, tubes, or pins and fuel elements (or assemblies)[†] which consistently meet specifications developed and demonstrated during the qualification process. This is normally accomplished through

[†] In Western terminology, a fuel element is an assembly of fuel plates or fuel pins, while in Russian terminology, the fuel tubes or fuel pins are called fuel elements, and an assembly of fuel elements is called a fuel assembly.

the manufacture and irradiation of full-sized elements as the last stage of the fuel qualification process. However, the option should be left open for a fabricator to show itself qualified through less direct means.

2.2 Qualification Process

The successful approach applied to qualification of the U_3Si_2 fuel is being applied again for the U-Mo fuel. This approach has been confirmed by international consensus [7], and was reaffirmed in November 2002 during a special panel of research reactor regulators attending the 2002 RERTR International Meeting in Bariloche, Argentina. The types of data to be included in the qualification report are listed in Ref. [7] and will not be repeated here. The Russian RERTR program will obtain similar types of data.

The qualification process for plate-type research and test reactors is based on developing industrial processes and adequate specifications for manufacturing U-Mo particles and fuels plates, getting basic data from ex-reactor and in-reactor tests on miniature or full-sized plates, confirming reprocessibility of U-Mo fuel, obtaining a comprehensive understanding of the behavior of U-Mo fuel under operational conditions, developing models to aid in understanding and predicting irradiation behavior, and confirming through full-sized element irradiations satisfactory behavior under normal operating conditions. The irradiation of full-sized elements both provides better statistics through the irradiation of many plates and demonstrates the ability of the fabricator to produce acceptable plates and elements by an industrial-scale process. Usually, one fuel element of each type tested is irradiated to a high burnup, typically 70-80% average, in order to demonstrate that adequate margin exists beyond the burnups normally reached in reactors that will use this fuel.

The qualification process in Russia is similar to that in the West, but with much less emphasis on experiments to determine the basic irradiation behavior of the fuel. The qualification process is defined by regulation, and qualification and licensing are more parallel than sequential processes, in contrast to Western custom. Fuel development and fuel fabrication are carried out by two separate organizations, with technology transfer between the two being part of the qualification process. Russian fuel irradiation qualification has traditionally been based on reactor-specific element and assembly tests, and the same practice is being followed now, albeit with the benefit of the large quantity of data already obtained by the U.S. RERTR program. Hence, although the emphases are somewhat different, the end points of qualification in the West and in Russia are the same—demonstrating manufacturability and quality control and demonstrating good irradiation performance through full-sized element/assembly irradiations.

Although the irradiation behavior of the U-9Mo alloy itself is well documented in Russia, irradiation tests are required to assess the behavior of U-9Mo dispersed in aluminum and the overall behavior of fuel tubes and pins. Full-sized U-Mo tubes are being tested by substituting them for one or two standard tubes in full-sized fuel assemblies (called combined assemblies). Such tests are analogous to the full-sized plate tests in the French program. Since aluminum-clad pins are new in Russia, short pins (so called mini-elements) are being tested in a special irradiation device to determine behavior of U-Mo fuel in this configuration. These tests are the analog of the RERTR program's miniplate tests. Irradiation tests of full-sized tube and pin assemblies will also be performed.

2.3 Manufacturing and Reprocessing Aspects

In France, CERCA has adapted its proprietary fabrication process to accommodate atomized U-Mo fuel powder and has used that process to produce full-sized fuel plates for irradiation tests in the OSIRIS reactor in France and the BR2 reactor in Belgium. BWX Technologies (BWXT) in the U.S. and CNEA are currently adjusting their processes as the first step in producing full-sized fuel elements for irradiation tests in the HFR in the Netherlands. In addition to atomized powder, powders produced by an industrial machining process and by a CNEA-developed hydriding/milling/dehydriding (HMD) process with commercial potential are being used. The very high density of U-Mo also has required adjustment of some quality control processes.

In Russia, fuel tubes and fuel pins have been fabricated using powders produced both by atomization and by a hydriding/dehydriding (HD) process. In the case of the pin-type fuel, hydraulic tests of fuel assemblies are an important component of the qualification procedure because the internal geometry is very different from that of tube-type assemblies.

Since the back end of the fuel cycle is critical for most reactor operators, it is necessary to consider reprocessing aspects during the fuel development and qualification process. This was done by the RERTR program early in the U_3Si_2 development effort at the Savannah River Laboratory [8] and is being done by COGEMA for the U-Mo development effort [9] and by the Russian RERTR program.

2.4 Qualification Data from Ex-Reactor Tests and Measurements

A number of ex-reactor tests and measurements have been performed or are planned by the RERTR program, the French U-Mo Group (CEA and CERCA), and CNEA. These include U-Mo powder characterization, fuel aluminum reaction rates and reaction-product characterization, fuel-meat thermal conductivity, and exothermic fuel-aluminum-reaction energy release. The Russian RERTR program is performing similar experiments. These experiments are expected to be concluded in the next two years and will not be discussed further in this paper.

2.5 Qualification Data from Irradiation Experiments

The RERTR program has performed a number of irradiation experiments with small plates (RERTR-1 through -5) to determine the basic irradiation behavior of U-Mo fuels, including the effects of different molybdenum contents, different powder production methods, different fission rates, and different temperatures. The main characteristics of these experiments are summarized in Table 1. The French program's plate irradiations, summarized in Table 2 are expected to confirm this understanding at full-scale. The failure of the 35%-enriched plates in the UMUS experiment has shown that operating limits related to oxide (boehmite) thickness must be imposed on U-Mo fuel plates [10]. The U-Mo powder for the RERTR-1 and –2 "microplates" and for the French IRIS1 and UMUS full-sized plates were produced by grinding processes that are not suitable for industrial-scale production. Subsequent tests have used powders produced by atomization or by an industrial-scale machining process. The maximum heat flux and cladding-temperature conditions tested are illustrated in Fig. 1.

Preparations are underway for the final stage of the qualification tests—the full-sized element/assembly irradiations listed in Table 3 and their postirradiation examinations. The overall schedule for the full-sized plate and element tests, as well as for the qualification report and licensing to be discussed in Sections 2.5 and 3, below, is shown in Fig. 2. It must be noted, however, that the inclusion of tests to advanced burnup in the French program, both of the IRIS2 plates and the

Experiment	RERTR-1	RERTR-2	RERTR-3	RERTR-4	RERTR-5
Reactor	ATR (US)	ATR (US)	ATR (US)	ATR (US)	ATR (US)
Number of small plates[(a)]	20 / 32	22 / 32	43 / 47	25 / 32	27 / 32
U-Mo powder[(b)]	Gr, At	Gr, At	At	Mach, At	Mach, At
Enrichment (% ^{235}U)	19.5	19.5	19.5	19.5	19.5
Fuel loading (g U/cm^3)	4.0	4.0	8.5	6.0, 8.0	6.0, 8.0
Heat flux at BOL (W/cm^2)	55	55	390	210	320
Cladding temperature (°C)	65	65	150	130	175
Coolant velocity (m/s)	7.6	7.6	6.0	4.0	4.0
Status of experiment	Complete	Complete	Complete	in PIE	in PIE
Number of irradiation cycles	2	8	2	7	3
Total duration (days)	94	232	48	230	116
Average burnup at EOL (% ^{235}U)	40	70	40	80	50

[(a)]Number of plates containing dispersions of U-Mo binary or ternary alloy powder / Total number of plates.
[(b)]Gr = Ground, At = Atomized, Mach = Machined and Milled.

Table 1. RERTR program small plate irradiations.

Experiment	IRIS1	UMUS	IRIS2	FUTURE
Reactor	OSIRIS (Fr)	HFR (Neth)	OSIRIS (Fr)	BR2 (B)
Number of full-sized plates	3	2 / 2	4	2
U-Mo powder	Ground	Ground	Atomized	Atomized
Enrichment (% ^{235}U)	19.75	19.75 / 35	19.75	19.75
Fuel loading (g U/cm^3)	8.0	8.0	8.0	8.0
Max. heat flux at BOL (W/cm^2)	120	170	230	340
Max. cladding temperature at BOL (°C)	75	90 / 110	105	130
Coolant velocity (m/s)	9	8	9	12
Status of experiment	In PIE	Stopped	Ready to start	In progress
Number of irradiation cycles	10	2	$0^{(a)}$ / $4^{(b)}$	$2^{(a)}$ / $4^{(b)}$
Total duration (full-power days)	240	48	$0^{(a)}$ / $100^{(b)}$	$35^{(a)}$ / $70^{(b)}$
Average burnup at EOL (% ^{235}U)	50	15	$0^{(a)}$ / $70^{(b)}$	$25^{(a)}$ / $50^{(b)}$
Max. local burnup at EOL (% ^{235}U)	67	20	$0^{(a)}$ / $85^{(b)}$	$33^{(a)}$ / $65^{(b)}$

$^{(a)}$Value reached by January 2003. $^{(b)}$Final goal.

Table 2. French program full-sized plate irradiations.

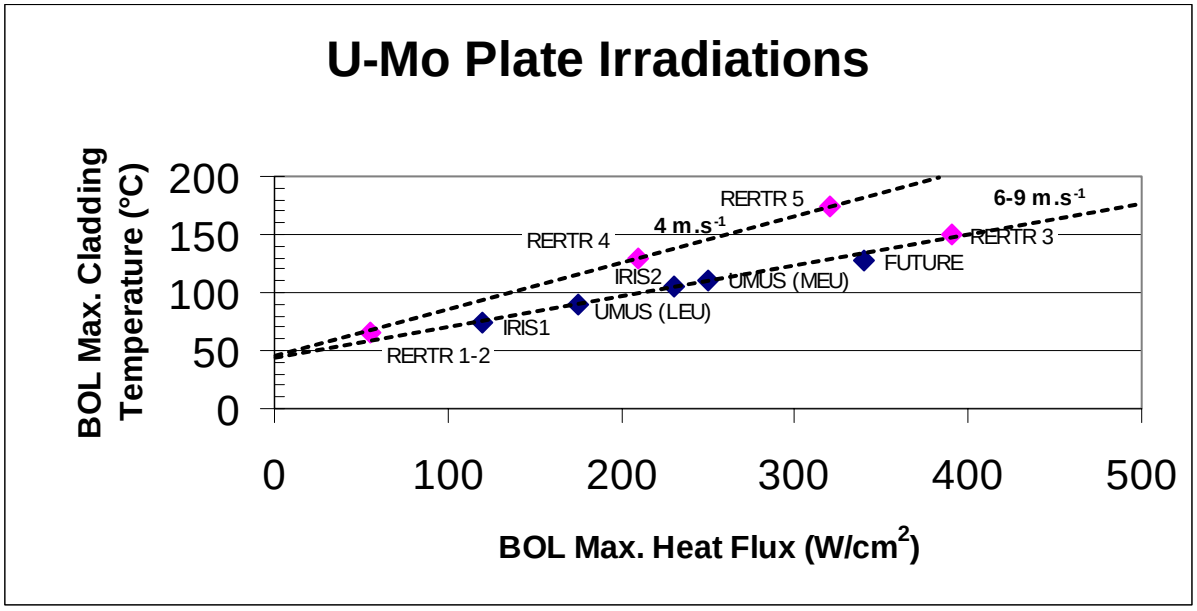

Figure 1. Maximum heat flux and cladding temperatures at BOL of small and full-sized plates tested.

elements in ORISIS, is contingent on agreements being reached with ANSTO and with the RERTR program to provide the supplementary funding needed to carry out the work.

As discussed above, the Russian RERTR program is developing and qualifying U-Mo for the traditional tubes and for the new aluminum-clad pins. The work on tubes is being carried out by RDIPE, NCCP, and IPPE; the work on pins is being carried out by VNIINM, in collaboration with the U.S. RERTR program, and by NCCP. Irradiation to 40-60% average burnup of two combined fuel assemblies, each containing two 5.4-gU/cm^3 U-Mo fuel tubes fabricated using atomized powder, has been completed in the IVV-2M reactor at RDIPE's Sverdlovsk Branch [4]. Postirradiation examinations will be conducted this year. The next step in the qualification of tube-type U-Mo fuel is the development and fabrication of IRT-3M fuel assemblies for testing in research reactors in Uzbekistan and the Czech Republic. Qualification of U-Mo tube-type fuel is scheduled to be completed during 2005. Irradiation of mini-fuel pins with fuel meat densities of 4 and 6 gU/cm^3 and containing both atomized and HD powders will begin this March in the MIR reactor at Dimitrovgrad. Following the end of the irradiation in August 2004 and a suitable cooling period, postirradiation examinations will be performed. In addition, irradiation of a full-sized U-Mo pin-type fuel assembly containing 5.3 gU/cm^3 is also scheduled to begin this March and be completed in June 2005 in the

Program	French	RERTR	RERTR	CNEA	Russian
Fabricator	CERCA	BWXT	CNEA	CNEA	VNIINM
Reactor	OSIRIS (Fr)	HFR (Neth)	HFR (Neth)	RA-3 (Arg)	WWR-M (Ru)
Number of elements	2	2	2	2	1
Number of plates per element	22	20	20	21	37
U-Mo powder[(a)]	At	At, Mach	At, HMD	HMD	At, HD
Enrichment (% ^{235}U)	19.75	19.75	19.75	19.75	19.7
Fuel loading (g U/cm^3)	8.0	6.0	7.0	7.0	5.3
Fuel loading (g ^{235}U)	650	625	625	425	93
Max. heat flux at BOL (W/cm^2)	170	140	140	70	67
Max. cladding temp. at BOL (°C)	95	<110	<110	<105	104
Number of irradiation cycles	5 / 10	13 / 17	13 / 17	64 / 90	Not defined
Total duration (full-power days)	150 / 300	325 / 425	325 / 425	320 / 450	600
Average burnup at EOL (% ^{235}U)	50 / 70	55 / 80	55 / 80	55 / 70	60

[(a)]At = Atomized, Mach = Machined and Milled, HMD = hydriding/milling/dehydriding, HD = hydriding/dehydriding.

Table 3. Full-sized element/assembly irradiations.

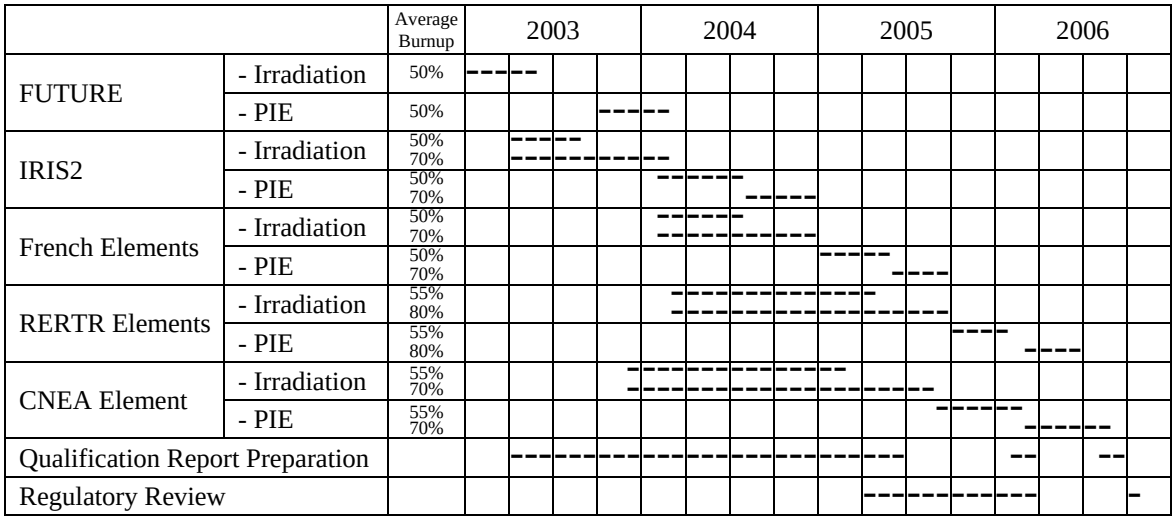

Figure 2. Schedule for qualification and generic approval of plate-type fuels.

WWR-M reactor at Gatchina. Work is currently underway to design and fabricate two assemblies each containing ~160 fuel pins for irradiation tests in Uzbekistan.

2.6 Modeling and Code Development

Following the agreement reached in January 2000, a close collaboration between CEA and ANL fuel experts has developed, starting with a detailed exchange of PIE observations performed on RERTR experiments and French experiments followed by exchange of different fuel behavior models and sharing the work of developing a common approach. The final objective is to have a common understanding of U-Mo dispersion fuel irradiation behavior. This collaboration has been instrumental in assessing the cause of the failure of the 35%-enriched U-Mo plates in the HFR.

During the past two years, ANL has put considerable effort into modeling the evolution of the temperature of dispersion fuel plates during irradiation [11]. CEA started the development of a thermal-mechanical code early in 2001. The major models for fuel-matrix interaction and fuel-meat thermal-conductivity change developed at ANL have been implemented in the French code. These codes were used to perform preliminary calculations for the plates of the French experiments. The comparison between ANL and CEA calculations shows very similar results.

ANL also is developing a mechanistic model of the evolution of fission gas bubbles in U-Mo fuel. Discussions are being held with the Russians to explore collaboration in this area. Such models, including the thermal model mentioned above, are the major components of a mechanistic code used to calculate fuel meat irradiation behavior. ANL and CNEA are collaborating in the development of this code [12].

2.7 Qualification Report for Plate-Type U-Mo Fuel

The RERTR program and the French program have agreed to begin as soon as possible the preparation of a qualification report, with the goal of completing an preliminary version by the end of 2005. This preliminary report will contain all qualification information except the results of the full-sized element irradiations and postirradiation examinations, which will still be in progress (see Fig. 1). The types of data discussed in Refs. [6,7] will be included; however, the format of the report is still under discussion.

3. Generic Approval and Licensing of U-Mo Fuel

Both the RERTR program and the French program have approached their national regulators, the NRC and the DGSNR, about the possibility of issuing a generic approval for U-Mo fuel similar to that issued in NUREG-1313 for U_3Si_2. Such a generic approval would be used in the U.S., France, and other countries as the basis for licensing the use of U-Mo fuel in specific reactors. NRC has agreed in principal to do this, and DGSNR is willing to consider the possibility. The French U-Mo Group will meet with DGSNR in mid-March 2003 to discuss the types of tests and data upon which qualification will be based, and DGSNR has agreed to comment on the adequacy of these tests and data. As shown in Fig. 1, NRC and DGSNR (if it has then agreed to review the qualification report on a generic basis) will be asked to begin their reviews using the preliminary report, since it will already contain most of the information that must be reviewed. Since the reviews are expected to take about one year, substantial, though not complete, information on the results of the full-sized element irradiations will be available near the end of the review period. Of course, the schedule shown in Fig. 1 is contingent on two things: 1) one or both regulators agreeing to start the review using the preliminary qualification report and 2) agreement being reached soon on funding the enhanced French program, which includes advanced burnup of IRIS2 plates, advanced burnup of one full-sized element to be irradiated in OSIRIS, and destructive postirradiation examinations of the plate and the element. Approximately a one-year delay in generic approval would result if the enhanced French program is not approved, and at least a one-year delay would result if the regulators do not agree to perform an expedited review.

The licensing procedure in Russia is somewhat different than in the West, and generic approval of the fuel is not anticipated. The qualification and licensing procedure in Russia begins with the development of the technical specifications of the fuel elements and fuel assemblies to be used in a specific reactor; the main technical decisions are confirmed by irradiation tests and postirradiation examinations. The number of mini-elements, full-sized elements, and full-sized assemblies required are determined by the existing database, among other criteria. Next, the technical documentation developed on the basis of experiments is presented to the national regulatory authority, Gosatomnadzor of Russia, for approval. Finally, a license must be obtained for the fabrication process, based on demonstration that the fabrication technology is adequate to meet the fuel element and assembly specifications and that the fabrication process is repeatable.

4. Conclusions

Significant progress is being made toward qualification and licensing of U-Mo fuels for both plate-type and Russian-built research reactors. Basic irradiation-behavior data have been obtained, tests of full-sized plates are underway in France, and tests of full-sized plate-type elements are planned in France, the Netherlands, and Argentina. The RERTR and French programs plan jointly to produce a preliminary qualification report by mid-2005 and hope to obtain generic approval of the fuel by NRC and, perhaps, by DGSNR during the first half of 2006. The Russian program, in collaboration with the

RERTR program, is pursuing a similar course, with the goal of qualifying during 2006 U-Mo fuel for use in Russian-built reactors.

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VALIDATION OF THE CONDITIONS OF THE IN-PILE TESTS OF HIGH-DENSITY LOW ENRICHMENT FUEL FOR RESEARCH REACTORS

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ABSTRACT

Several new types of a dispersion fuel composition based on high-density U-Mo alloys were chosen according to the analysis of literature data and pre-reactor material science investigations. High-density fuel (HDF) is designed for the use in the fuel elements of research reactors in order to reduce enrichment. The in-pile tests of HDF will be conducted in the MIR research reactor (FSUE "SSC RF RIAR", Dimitrovgrad). In the course of tests the operating conditions of fuel elements of pool-type research reactors will be simulated.
The program of in-pile tests and the design of an irradiating device (ID) were developed. The neutron-physical, thermal-hydraulic calculations were made. They validated the possibility of testing the developed ID design ensuring the required test conditions of the HDF samples with proper monitoring of test parameters.

1. Introduction

At present the Russian research reactors, including those built with participation of Russia abroad, use fuel elements with UO_2-Al dispersion type fuel (36% and 90% enrichment). The long experience of successful fuel element operation showed their high reliability. However, the complexity of fuel element fabrication technology restricts the possibility of increasing a volume fraction of nuclear fuel in the core and decreasing the fuel enrichment.
Therefore, according to the pre-reactor investigation results for radiation tests several alloys, mainly, uranium-molybdenum ones, were chosen. In-pile tests allow obtaining the data on irradiation effect on the new HDF samples-aluminum matrix compatibility, and the data on the cores swelling.
Some results of pre-reactor investigations of the test conditions were reported at the international conference [1]. By now some characteristics of fuel and test parameters are changed by the fuel element technologist. This necessitated additional calculated investigations. The calculated validation results of neutron-physical and thermal-hydraulic test conditions of HDF samples are presented in this paper.

2. Irradiating device

To test new types of the fuel composition in the MIR reactor (FSUE "SSC RF RIAR") the design of an irradiating device was developed (Fig.1, 2). The irradiating device consists of 2 assemblies (Fig.1, 2) with mini fuel elements. The assemblies inside the working channel (WC) are installed one above another as a "garland" (Fig.1) forming ID. An assembly comprises carcass and a cover. The carcass is made as a rod, on top of which there is a head gripped by a refueling tool. Mini fuel elements in the block are located in one row, on the diameter of 61 mm with a pitch of 5.3 mm. The number of mini fuel elements installed in one assembly is 36.
The main geometrical characteristics of mini fuel elements are presented in Fig.3, those of the fuel composition – in Tab.1.

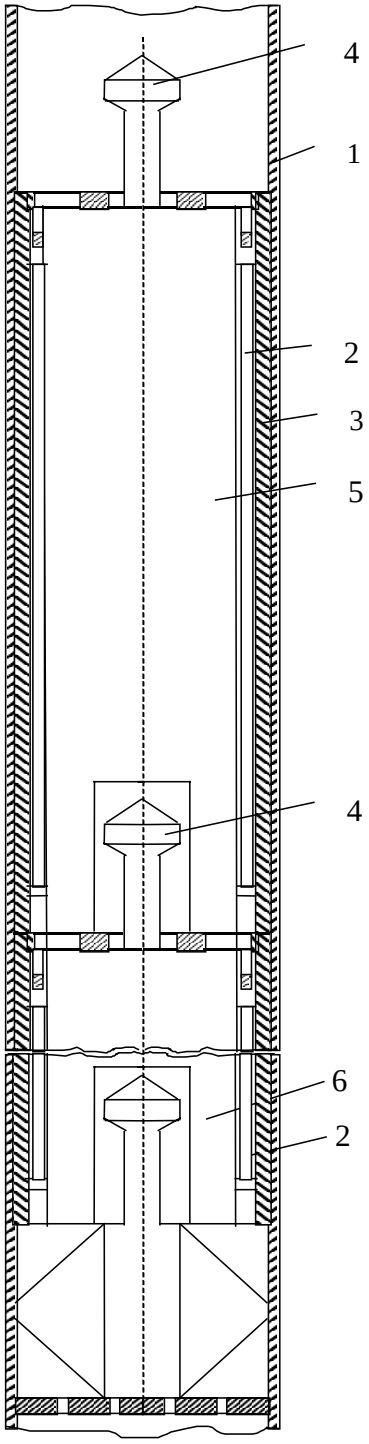

Fig.1. Sketch of the irradiating device.
1- channel case, 2- mini fuel elements,
3- cover, 4- carcass head, 5- assembly
No.1, 6- assembly No.2.

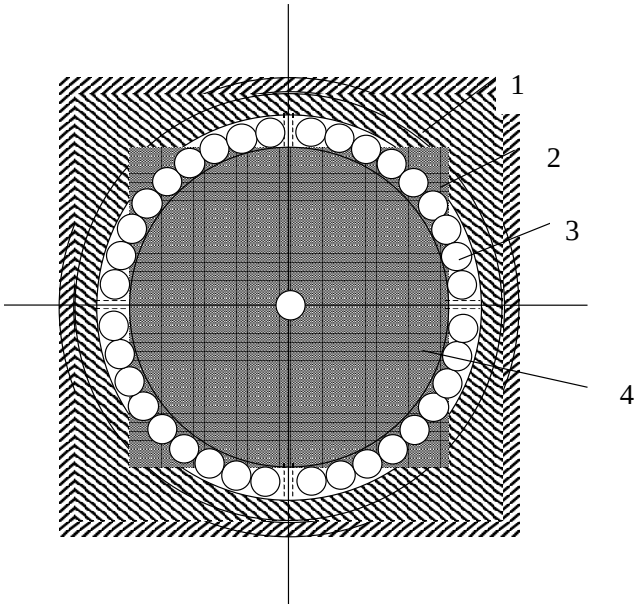

Fig.2. Cross-section of ID on the core level.
1- channel case, 2- cover, 3- mini fuel elements,
4- carcass.

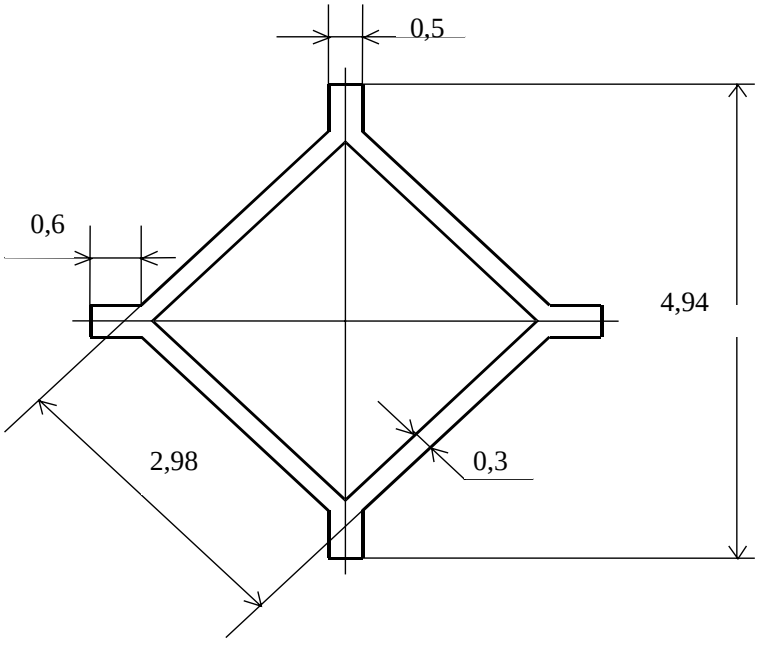

Fig.3. Cross-section of a fuel element.

Table 1 Main characteristics of a mini fuel element

Characteristic	Value	
Cladding material	Aluminum alloy CAB-1	
Fuel composition	UMo+Al powders PAZ/PA4	
Enrichment in U-235	19.7±0.3%	
U concentration, g/cm^3	4	6
U-235 loading	0.82±0.1	1.22±0.1
Volume fraction of fuel component, %	23-25	34-38

The main test parameters are presented in Tab.2.

Table 2 Test parameters of mini fuel elements

Parameter	Value
Power of mini fuel element, kW	1.3-1.9
Heat-flux density, MW/m^2 - average - maximum	 0.4-0.6 0.9
Inlet pressure, MPa	1.1-1.3
Inlet coolant temperature, C	<70
Coolant velocity, m/s	2.3-3.0
Maximum surface temperature of cladding, C	110
Maximum fuel burnup in unloaded mini fuel elements, %	20, 35, 50, 60, 70

3. Calculation results

The intermediate selections of mini fuel elements for investigations lead to a considerable increase of the experiment duration, since they can be performed only in the shielded chamber after keeping the irradiated assemblies with mini fuel elements in the pool. To reduce the experiment duration, irradiation of mini fuel elements is divided into 3 stages with 2 intermediate selections of mini fuel elements for investigations so as to obtain the required fuel burnups (refer to the scheme below, Fig.4).

Loading of 15 mini fuel elements with γ_u=4 g/cm^3, 15 mini fuel elements with γ_u=6 g/cm^3 and 6 simulators into each assembly.

⇩

Stage 1 68-day irradiation; maximum burnup in assembly No.2 - 20%

⇩

40-day keeping; unloading of 3 mini fuel elements with γ_u=4 g/cm^3 and γ_u=6 g/cm^3 from each assembly. Loading of fresh mini fuel elements. Re- arrangement of assemblies

⇩

Stage 2 185-day irradiation; maximum burnup in assembly No.2 – 50%

⇩

Unloading of assembly No.2

⇩

Stage 3 80-day irradiation of assembly No.1; maximum burnup – 75%

Fig.4. Irradiation algorithm

To create different test conditions and different degrees of fuel burnup for the same period, the axial energy-release distribution is simulated with maximum in the core lower half. Such an energy-release distribution is formed at the expense of intermediate position of shim rods near the channel with ID.

The calculated powers of mini fuel elements and the required power of working fuel assembly (FA) of ID surroundings are presented in Fig.5 and 6. At all irradiation stages the maximum heat-flux density is maintained equal to 0.9 MW/m^2. To maintain these irradiation conditions, there must be a gradual power rise of working FAs: about 1.2 times at the 1-st stage, 1.9 times at the 2-nd stage and 1.6 times at the 3-rd stage. The absolute power of working FA in the experiment increases from 238 kW to 697 kW (Fig.5).

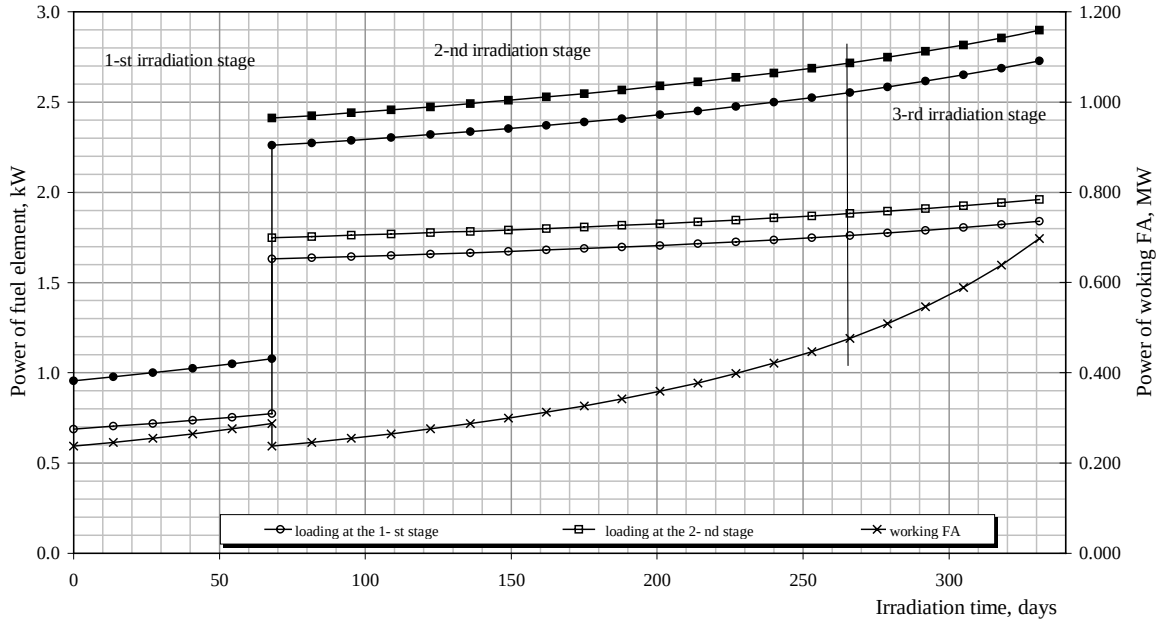

Fig.5. Power of mini fuel elements in assembly No.1 and the required power of working FA of ID surroundings versus the irradiation time (uranium density: unshaded markers – 4 g/cm^3; shaded markers – 6 g/cm^3).

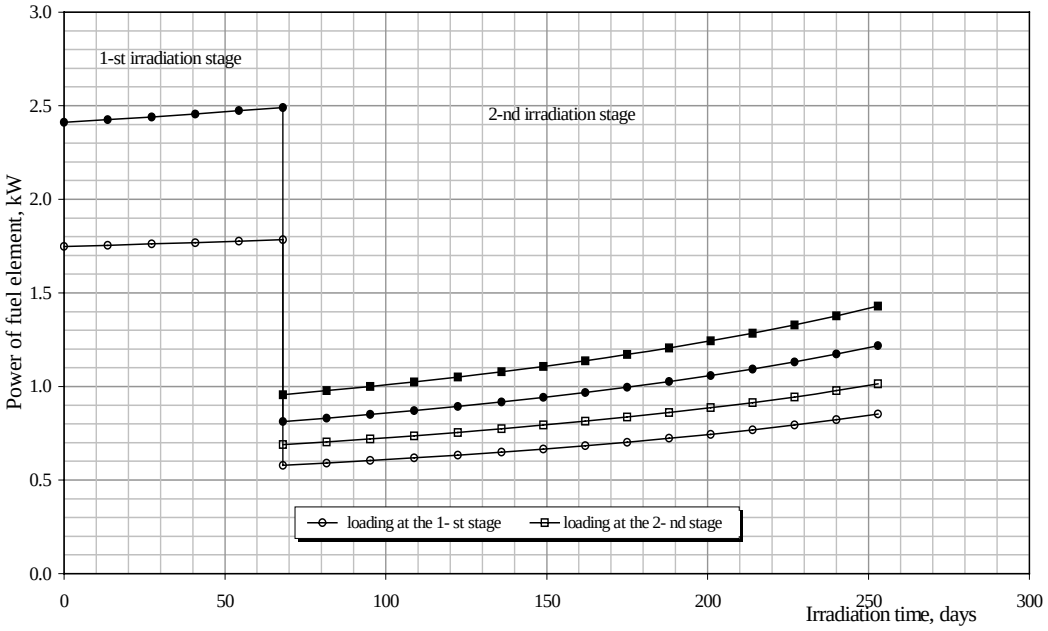

Fig.6. Power of mini fuel elements in assembly No.2 versus the irradiation time (uranium density: unshaded markers - 4 g/cm^3; shaded markers – 6 g/cm^3).

Fig.7 shows the temperature changes in claddings of energy-intense mini fuel elements along the active part length for each experiment stage.

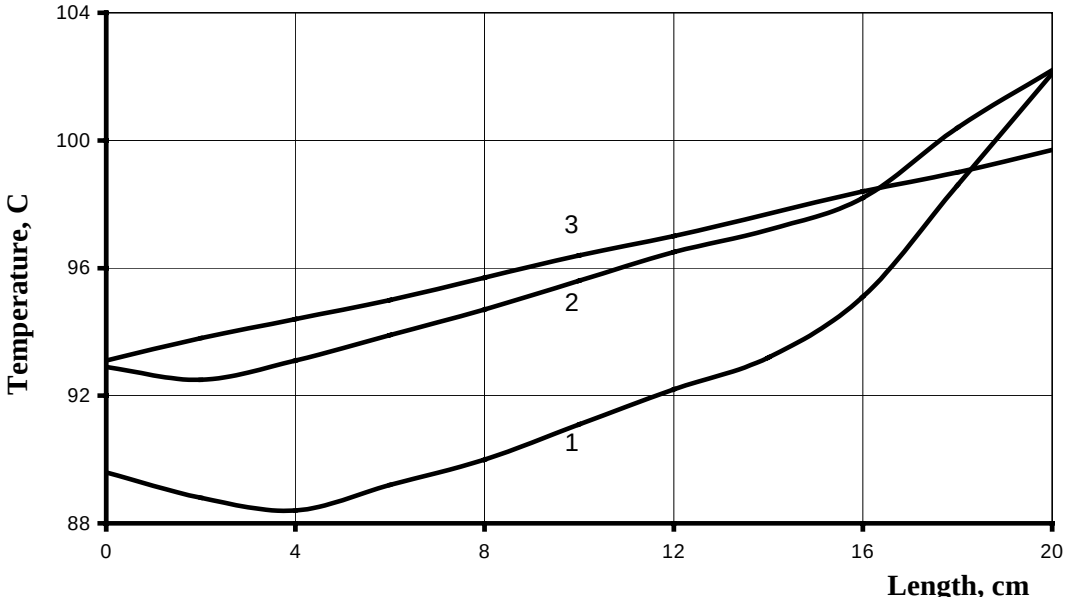

Fig.7. Temperature changes in claddings of energy-intense mini fuel elements along the active part length. 1- stage 1, 2- stage 2, 3- stage 3.

The calculated results of the temperature field of an energy-intense mini fuel element are shown in Fig.8, and it is obvious that the maximum cladding temperature does not exceed the permissible temperature of 110 C

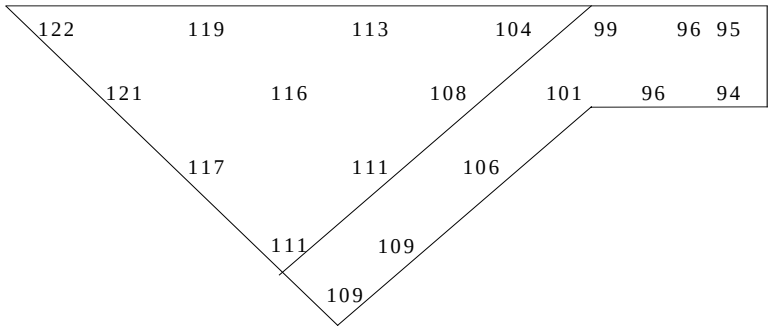

Fig.8. Temperature distribution over the cross-section of a mini fuel element fragment (1/8).

4. Conclusions

Several types of high-density nuclear fuel based on the U-Mo system alloys were chosen for tests in the MIR reactor.

The design of an irradiating device was developed and the tests algorithm was validated. This provides the irradiation conditions of mini fuel elements with HDF that are approximated to the operating conditions of fuel in pool-type research reactors as much as possible.

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EFFORTS TO OVERCOME THE FAILURE OCCURRING IN ROD TYPE U-Mo FUEL IRRADIATION TEST FOR HANARO IN KAERI

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ABSTRACT

The 1st PIE on the failed fuel rod of the 1st irradiation test of rod type U-Mo fuel in Korea showed that the failure would be influenced by the fuel meat swelling from severe interaction between U-Mo particles and Al matrix as well as a weak bonding in co-extrusion process. The further PIE on the other fuels in the 1st irradiation test revealed the completely interacted region in all three highly U-loaded fuels with 6.0 g-U/cc, some voids and cracks were found. The swellings calculated by dimension measurement were high in all high U-density fuels such as 16.1 %, 17.1 %, and 14.7. However, the fuel of low uranium loading density of 3.4 g-U/cc showed a small swelling of 7.8 % and the same small interaction layer thicknesses on the various regions. The most affecting parameter on fuel performance is considered to be uranium loading density. Therefore, U loading density change from 6.0 g-U/cc to 4.5 g-U/cc was decided for the next irradiation test. To avoid the severe interaction problem and take advantage of the good high temperature irradiation behavior of the U-Mo phase, a rod fuel consisting of monolithic U-Mo tubes was applied. Some other fuels, which aim for corrosion-resistance improvement of cladding, investigating the effect of particle size, poison material addition associated with life extension, the effect of higher Mo content, and a U_3Si fuel as a reference, were involved in the 2nd irradiation test. The 2nd irradiation fuel assembly was loaded in HANARO on January 9, 2003 and will be irradiated with the target burnup of 60 at.% until February 2004.

1. Introduction

In connection with the end of the FRRSNF Acceptance Program of spent research reactor fuel in May 2006 as well as the difficulty of uranium silicide dispersion fuel for the reprocessing process, a qualification program of U-Mo dispersion rod type fuel applicable to HANARO has been carried out since 1999. The 1st irradiation test was launched in June 2001, for a fuel assembly containing 10 fuel rods with two different Mo contents of 7 wt.% and 9 wt.% and two different uranium loadings of 3.4 and 6 g-U/cc for each Mo content. In order to reduce the temperature rise in the fuel meat of 6 g-U/cc the diameter of fuel meat was 5.49 mm, which is smaller than the standard diameter of 6.35 mm. It had been expected applicable for the high uranium loading density for HANARO in the aspect of converting some driving fuel sites to irradiation test holes. Due to the occurrence of a failure in the highly irradiated fuel, the fuel assembly was discharged from the reactor on August 27, 2001[1].

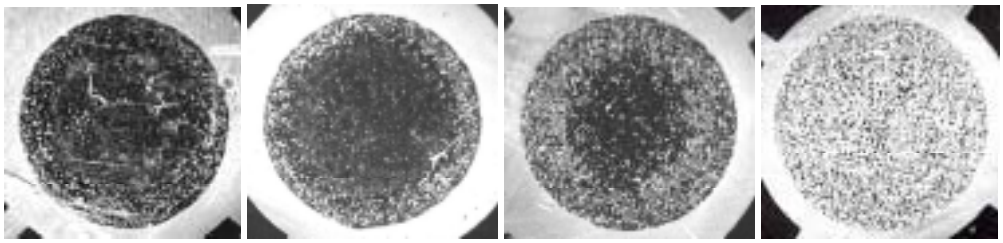
The failure of the 1st U-Mo dispersion rod fuel irradiation was observed to have occurred through the cladding cleavage. The very smooth cleavage surface implies that the defects in the cladding would act as a dominant role in the failure. The swelling calculations performed using the measured densities by the immersion method and the Al matrix fraction on micrographs of fuel meat cross sections revealed the maximum swelling to be about 15 Vol. %, which is not large enough to affect cladding failure because of the pure aluminum with ductile property. The estimated centerline temperature seems to be higher than 700 °C. On observation, the remaining U-Mo phase showed a very stable behavior without large bubble formation [2].

From the above back ground, efforts to overcome the failure occurrence problems have been made with consideration in preventing much interaction and improving fuel cladding properties such as ductility as well as anti-corrosion against cooling water. The obtained results were implanted in working out the 2nd irradiation test. In this paper, the measures to overcome the irradiation failure and the 2nd irradiation test progress are reported.

2. Further Investigation on the previously irradiated U-Mo rod fuels

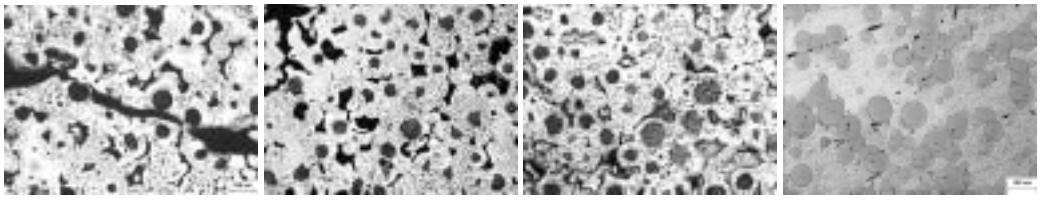
The post-irradiation examinations have been done on 4 U-Mo dispersion fuel rods, of which three fuel rods with 6.0 g-U/cc were chosen on 3 different power levels. The other fuel rod is chosen as the highest power level for a low uranium density of 3.4 g-U/cc. As shown in figure 1, the completely interacted region on the cross section appeared in all three highly U-loaded fuels with 6.0 g-U/cc. The fraction of the fully interacted region in fuel meat cross section seems to increase with the power level. Some observations have been done on the center regions of the fuels as shown in figure 2. It is shown that some voids exist among fuel particles in the other highly U-loaded fuels. In fact, the voids formed by breakups of the Al matrix phase. Presumably the loose Al matrix phase would come from the faster mobility of Al atoms diffusing into fuel particles than the mobility of uranium or Mo atoms in fuel particles. In addition, the high temperature of the center region due to the low thermal conductivity and high power-rate would accelerate the void growth.

Some cracks were found in the highly power-rated fuel at 104.9 kW/m. The crack would occur from the expansion difference between the center region and the outer region. The swelling rate of the center region would retard earlier than the outer region after the completion of the interaction of the center region because the swelling is strongly dependent on the interaction of U-Mo dispersion fuel.



(a) 6.0 g-U/cc (b) 6.0 g-U/cc (c) 6.0 g-U/cc (d) 3.4 g-U/cc

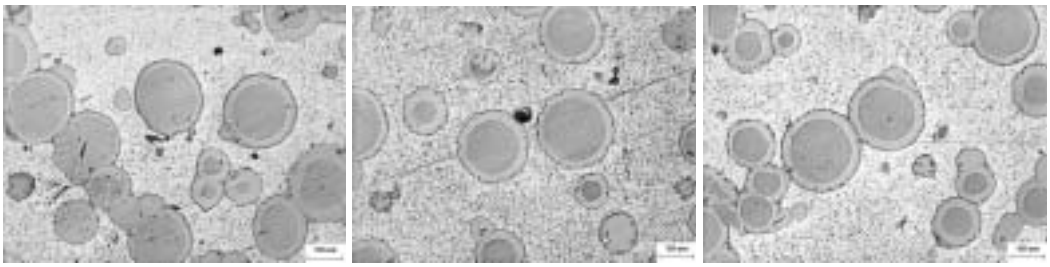
Fig. 1. Photographs of cross- sections of 4 differently power-rated fuel rods; (a) 104.9 kW/m, (b) 96.9 kW/m, (c) 86.0 kW/m, (9) 81.9 kW/m



(a) 6.0 g-U/cc (b) 6.0 g-U/cc (c) 6.0 g-U/cc (d) 3.4 g-U/cc

Fig. 2. Micrographs on the center regions of cross- sections of 4 different fuel rods

The interaction layer thicknesses on various regions in the fuel of low uranium loading density of 3.4 g-U/cc were observed to be almost the same as those shown in figure 3. The temperature of the center region was calculated to be 191 °C, which is a little lower than the center region temperature of the high U loaded fuel of 6.0 g-U/cc with the lowest power-rate. The temperature difference in the center and the periphery regions was calculated to be about 70 °C. In spite of such a big temperature difference the reason for the same interaction layer thickness is considered to be the fact that the interaction would be more dominantly induced by the irradiation.



(a) Center (b) Middle (c) Periphery

Fig. 3. Micrographs on the various regions of low U density fuel with 3.4 g-U/cc

The fuel meat swellings by irradiation were calculated by the measurements of fuel meat diameters on the photographs of the irradiated fuel cross-sections and obtained as Table 1. The swellings of the highly U-loaded fuels are shown to have little fluctuation depending on the BOL temperatures regardless of much BOL difference as shown in Table 1. The low U-loaded fuel appeared to have not much swelling. It is considered that the highly U-loaded fuels with densely dispersion would be deteriorated more quickly than low U-loaded fuel because of the smaller thickness of the Al matrix between fuel particles and the lack of the Al phase continuity. In addition, the larger interface area of the highly loaded fuel would be attributed to the swelling.

Table 1. Swelling calculation for various fuels by dimension measurements

Fuel	U-density	6.0 g-U/cc	6.0 g-U/cc	6.0 g-U/cc	3.4 g-U/cc
	Linear power	104.9 Kw/m	96.9 kW/m	86.0. kW/m	81.9 kW/m
Diameter	Before irradiation	5.49 mm	5.49 mm	5.49 mm	6.35 mm
	Post irradiation	5.92 mm	5.94 mm	5.88 mm	6.59 mm
Calculated center Temp.°C		265 °C	237 °C	205 °C	191°C
Swelling, %		16.2%	17.1%	14.7%	7.8%

3. Efforts to overcome the interaction problem of U-Mo dispersion fuel

As shown in the post-irradiation examination of U-Mo dispersion rod fuels, the failure in irradiation occurred in the highly U-loaded fuel under a high power rate. The most strongly affecting parameter has been concluded to be the fuel temperature. In planning the next irradiation test the maximum center temperature of BOL was limited below 200 °C. Accordingly the U-loading density was decided to be 4.5 g-U/cc in case of the reduced fuel meat diameter of 5.49 mm. On the other hand, the fuel meat diameter of the standard fuel with 6.35 mm was designed to be 4.0 g-U/cc. The lower uranium loading density in the U-Mo dispersion fuel would induce much more stable irradiation behavior because of the lower center temperature from higher thermal conductivity and the much smaller interface specific area associated with interaction swelling. In order for the center temperature to rise higher than 200 °C, a shorter fuel meat than half the length of the standard fuel meat of 700 mm was applied and positioned in the upper part, which has less neutron flux than the peak neutron flux.

As a method to avoid the severe interaction problem and to take advantage of the good high temperature irradiation behavior of the U-Mo phase, a fuel rod containing a monolithic U-Mo tube could be applied. A irradiation fuel rod containing a monolithic U-Mo tube was designed considering the interface temperature to be lower than 150 °C. So the outer and inner diameters and the length of the monolithic tube are OD 6.2 mm, ID 4.8 mm, and L 3.25 mm, respectively. The tube was fabricated by melting and casting.

It was reported that the failure in the previous U-Mo dispersion irradiation test occurred by not only the severe interaction swelling but also a fabrication defect of the fuel cladding. So in order to enforce the cladding integrity an effort has been made to establish the optimum co-extrusion parameters. It has been found that the higher preheating temperature of Al ingot for co-extrusion gives the better soundness of the cladding. Accordingly the preheating temperature was changed to be about 50 °C higher.

The corrosion layer thickness of U-Mo dispersion fuel with failure was measured to be 20 to 30 micrometer. It is known that the corrosion layer induces the fuel meat temperature to increase severely due to the very low thermal conductivity of the corrosion product. A corrosion test for a fuel rod coated thinly with Ni (less than 10 micrometers) and a fuel rod without any coating was performed at the elevated temperature of 350 °C under high pressure water at two different waters of pH 5.5 and pH 6.5. The cladding without coating was corroded up to a layer of more than 70 micrometers in both pH conditions. However, the cladding coated with Ni showed no corrosion. ANL in USA carried out a surface treatment of autoclaving in saturated water at 160 °C for 16 hours in order to pre-film 3 to 5 μm thick corrosion-resistant hoehmite layers[3]. In connection with technology development to prevent the temperature rise due to the corrosion, two fuel rods, which are treated with Ni-coating and pre-oxidation, were fabricated for the 2nd irradiation test.

4. The 2nd Irradiation Test for the U-Mo Dispersion Rod Fuels Applicable to HANARO

The 2nd irradiation test plan had been planned since the failure occurrence in the 1st irradiation test. Besides the above mentioned 1st aim to overcome the failure, some other purposes have been involved in the 2nd irradiation test. The U-Mo dispersion fuel with high U-density is mainly swelled by the interaction at the interface of fuel particles and Al matrix. Presumably the swelling behavior would be influenced by the particle size of the fuel powder because of the greater difference in specific surface area. Two different fuel rods were fabricated using larger particle power than 80 μm and smaller particle powder than 80 μm. The replacing of HANARO fuel with higher uranium loading density would allow for extended fuel life. In this case, an addition of poison material to fuel meat is required. A fuel rod added with poison material of Er_2O_3 was fabricated for irradiation. In general, it has been known that irradiation performance improves with Mo content. A fuel rod dispersed with U-9wt.%Mo powder has been fabricated to the same specifications as the U-7wt.%Mo fuel having high U density of 4.5 g-U/cc in order to compare it in the Mo content aspect. A U_3Si dispersion fuel rod was prepared as a reference. A comparison of U-Mo fuel with U_3Si fuel will be conducted. This irradiation test assembly consisting of 10 fuel rods is as shown in Table 5 and was fabricated and loaded in HAHARO on January 9, 2003. The target burnup is 60 at.% so that the discharging is expected to be in February 2004.

Table 2. Fuel rods to be loaded for the next irradiation test

Serial No.	Fuel Material	Loading (g-U/cc)	Diameter (mm)	Length (mm)	Number	Remarks
1	U-7Mo	4.5	5.49	360	1	High density
2	U-9Mo	4.5	5.49	360	1	High Mo
3	U-7Mo	4.0	6.35	360	1	Low density
4	U-7Mo	4.0	6.35	110	1	Larger particles
5	U-7Mo	4.0	6.35	110	1	Smaller Particles
6	U3Si	4.0	6.35	210	1	Reference
7	U-7Mo	4.5	5.49	210	1	Poison
8	U-7Mo	Pure	$^{OD}6.2\times^{t}0.7$	3.25	1	Tube
9	U-7Mo	4.0	6.35	360	1	Ni coating
10	U-7Mo	4.0	6.35	360	1	Pre-oxidation

5. Summary

The further PIE on the other fuels in the 1st irradiation test revealed the completely interacted region in all three highly U-loaded fuels with 6.0 g-U/cc, some voids and cracks were found. The swellings calculated by dimension measurement were high in all high U-density fuels such as 16.1 %, 17.1 %, and 14.7. However, the fuel of low uranium loading density of 3.4 g-U/cc showed a small swelling of 7.8 % and the same small interaction layer thicknesses on the various regions.

U loading density change from 6.0 g-U/cc to 4.5 g-U/cc was decided for the next irradiation test. To avoid the severe interaction problem and take advantage of the good high temperature irradiation be

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6. Acknowledgement

This work has been performed under the nuclear R&D support by the Ministry of Science and Technology, Republic of Korea

7. Reference

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PROGRESS IN POSTIRRADIATION EXAMINATION AND ANALYSIS OF LOW-ENRICHED U-Mo RESEARCH REACTOR FUELS

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ABSTRACT

High-density dispersion fuel experiment, RERTR-4, was removed from the Advanced Test Reactor (ATR) after reaching a peak U-235 burnup of ~80% and is presently undergoing postirradiation examination at the ANL Alpha-Gamma Hot Cell Facility. This test consists of 32 mini fuel plates of which 27 were fabricated with nominally 6 and 8 g cm^{-3} atomized and machined uranium alloy powders containing 6.5 wt% to 10 wt% molybdenum. In addition, two miniplates containing solid U-10wt%Mo foils. The results of the postirradiation examination and analysis of RERTR-4 in conjunction with data from a companion test performed to 50% burnup, RERTR-5, are presented.

1. Introduction

Presently five irradiation tests of U-Mo/Al dispersion fuel miniplates have been completed. Tests RERTR-1 and 2 were low-temperature scoping tests that contained a wide range of uranium alloy compositions. The main conclusion after postirradiation analysis of these first two tests was that uranium alloys with a molybdenum content between 6% and 10% by weight showed excellent irradiation behavior [1]. Test RERTR-3 was designed to explore the higher-temperature behavior of these U-Mo/Al dispersion fuels [2].

Postirradiation examination allowed the characterization of U-Mo/Al interaction. The formation of the aluminide interaction phase appeared to be the only aspect of fuel behavior that is significantly affected by temperature. The irradiation behavior of the U-Mo fuel alloy itself was deemed athermal over the temperature range tested. Extensive metallographic analysis of these first three tests was used to develop a fuel behavior model [3]. Based on the positive results of the first three tests, two further tests, RERTR-4 and 5, were designed with larger, so-called miniplates that are more prototypic of full-size test reactor fuel plates and also allow for more accurate postirradiation measurements.

The miniplates irradiated in RERTR-4 and 5 contained either atomized fuel particles, supplied by KAERI, or machined fuel particles, supplied by AECL, ranging in composition from, nominally, 6 wt.% Mo to 10 wt.% Mo. The fuel plates in these tests measured 100 mm x 25 mm x 1.40 mm; the meat was in a rectangular zone nominally 0.64 mm thick and contained, nominally, 6 and 8 g U cm^{-3} in the fuel meat. The RERTR -4 and -5 experiments were irradiated in the Advanced Test Reactor (ATR) for 204 EFPD (effective full power days) and 116 EFPD, respectively. In addition to 30 dispersion fuel miniplates, test RERTR-4 also included two miniplates with solid U-Mo alloy cores.

Test RERTR-5 was removed from the reactor at a peak burnup of ~50% U-235, whereas RERTR-4 terminated at ~80% burnup. Postirradiation examinations of the RERTR-5 plates have been largely completed, and a preliminary assessment was presented at the 2002 RRFM meeting in Ghent, Belgium

[3]. Examination of the miniplates from RERTR-4 has progressed through plate thickness measurements. In this paper the data obtained on RERTR-4 and -5 to date will be assessed.

2. Postirradiation Data

Postirradiation examinations of RERTR-5 have progressed through the destructive stage, and the pertinent observations obtained from these examinations have been reported previously [4, 5]. However, the emphasis in these earlier reports has been on the characterization of the irradiation behavior of U-10wt.% Mo fuel. In due course, detailed examination and analysis of more 6 to 8 wt.% Mo samples will be completed in order to characterize the effect of Mo content on the fundamental aspects of the swelling of U-Mo/Al dispersion fuel. The examinations of the high-burnup test RERTR-4 are in the early stages, and only fuel plate thickness measurements are available at this time. Plate thickness is measured at five axial locations on the fuel zone. The average thickness compared to a thickness measured outside the fuel zone and to preirradiation measurements to obtain an average change in the plate thickness. Although not as precise as immersion volume measurements, this does provide a preliminary measure of the amount of irradiation-induced meat swelling of the various miniplates at high burnup. In order to make comparison with the lower-burnup experiment RERTR-5, the plate volume measurement from this test were converted to average thickness assuming that the ratio of meat volume to total plate volume was the same for all plates.

The resulting data plotted against the beginning-of-life (BOL) peak meat temperatures are shown in Fig. 1. A linear normalization to 50% burnup and 80% burnup, respectively for tests RERTR -5 and -4 was applied to the swelling values to aid comparison. The BOL temperatures plotted in Fig. 1 only serve to illustrate the trend in swelling with increasing temperature. The actual fuel temperature during irradiation is a complex function of time and position in fuel meat. Not only are there temperature gradients in the fuel meat, but the temperature changes during irradiation as a result of the competing effects of the decreasing thermal conductivity and U-235 burnup. This issue was treated in detail previously [3] and remains the subject of continuing study.

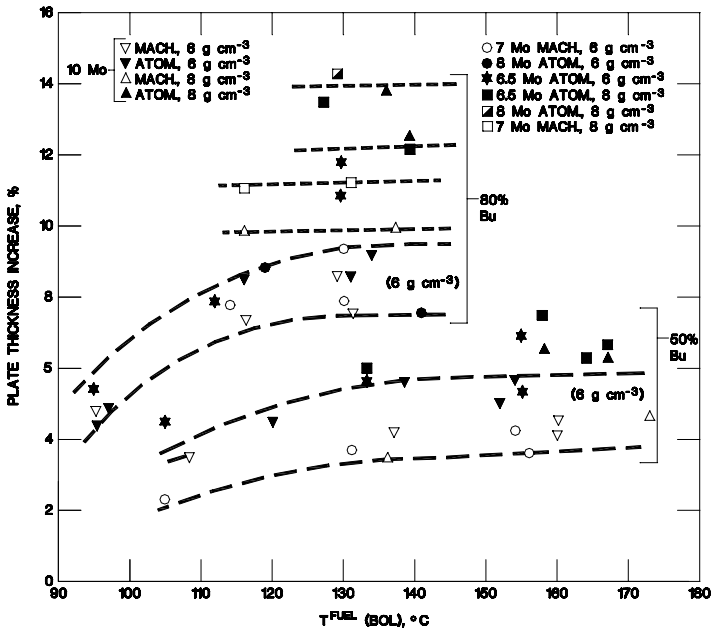

Fig. 1 Miniplate thickness increase versus BOL meat temperature for various Mo compositions and 6-8 g cm^{-3} loading of machined and atomized powders in tests RERTR-4 and 5.

3. Discussion

The plate thickness increases shown in Fig. 1 reflect the swelling behavior of each fuel type, fuel composition, and fuel loading of the individual miniplates. The swelling of plates made with machined powder (open symbols in Fig. 1) is generally lower than that of plates made with atomized powder, most prominently for the higher-loaded 8-g cm^{-3} plates. This can be attributed, primarily, to the higher volume fraction of as-fabricated porosity in the fuel meat made with machined powder, approximately 7% compared to ~2% for atomized powder at 8 g cm^{-3}. This porosity accommodates a roughly equivalent amount of fuel swelling that therefore does not translate to an increase of plate thickness. This effect is commonly observed in all dispersion fuels.

As expected, the amount of swelling in the higher-loaded 8-g cm^{-3} plates is larger than in the 6-g cm^{-3} plates particularly in the case of atomized powder where as-fabricated porosity is small and where, thus, the larger volume of fission products that is generated in the higher-loaded meat is more completely transferred to an increase in plate thickness. The general trend of the data shown in Fig. 1 suggests a rather weak temperature dependence of the meat swelling, tending to apparently athermal behavior at higher temperatures. This may be explained by considering the different components of the total swelling.

Swelling of the fuel meat is composed of several components:

1. Swelling of the U-Mo alloy particles by accumulation of solid and gaseous fission products.
2. Volume increase as the interaction phase forms by interdiffusion of U-Mo and Al at the fuel particle surface.
3. Accumulation of fission products in this interaction phase.
4. Partial accommodation of the above swelling by as-fabricated porosity in the meat.

Swelling of the U-Mo alloy was determined from immersion volume data and quantitative metallography of several plates from RERTR-5 [5]. The analysis covers a fuel temperature range of ~100-280ºC and shows no measurable temperature dependence. This is supported by scanning electron microcopy (SEM) of the fuel microstructure shown in Fig. 2. The fuel swelling rate is linear in burnup (fission density) both before and after fission-induced recrystallization takes place; however, the rate increases when the fuel recrystallizes. During recrystallization, the original grains in the U-Mo particles are subdivided in many much smaller grains as shown in Figs. 2, 3, and 4.

Fission gas bubbles preferentially nucleate and grow on grain boundaries; recrystallization increases the grain boundary area and thereby increases the nucleation and growth of fission gas bubbles. Recrystallization begins at a somewhat lower burnup (around 40%) in machined powder than in atomized powder, as shown in Fig. 3 for U-10 wt% Mo. This difference is presumably due to the deformation caused by machining of the powder. This effect is much more pronounced for lower Mo content. As shown in Fig. 4, the 7 wt% Mo machined fuel has completed recrystallized at ~45% Bu, whereas the atomized powder at 8% Mo is just beginning to show signs of incipient formation of small grains at some original grain boundaries. Swelling of machined powder should therefore be somewhat higher than atomized powder at burnup beyond ~40%. The reason this is not reflected in the thickness data is due to the high as-fabricated porosity in the plates made with machined powder.

The other contributing component of meat swelling is the formation of the U-Mo/Al interaction phase. Because the interaction-phase-formation rate is strongly temperature dependent, it will rapidly deplete the matrix Al during irradiation at higher BOL temperatures (see Fig. 5), and the volume fraction of interaction product, $(U\text{-}Mo)Al_x$, will reach its maximum value early. The amount of swelling associated with the interaction is proportional to its amount; however, in this higher temperature case, more U-Mo is consumed, and since U-Mo has a higher irradiation swelling rate than $(U\text{-}Mo)Al_x$, this compensates for the interaction swelling. At lower temperatures the U-Mo/Al interaction is only partially completed at 80% burnup, and its volume fraction and associated meat swelling contribution is consequently much smaller. However, because the amount of U-Mo consumed is proportionally

smaller as well, the original fuel volume fraction is only slightly diminished. The overall meat swelling at low temperature should be dominated by U-Mo swelling, showing no substantial effect of (U-Mo)Al_x formation.

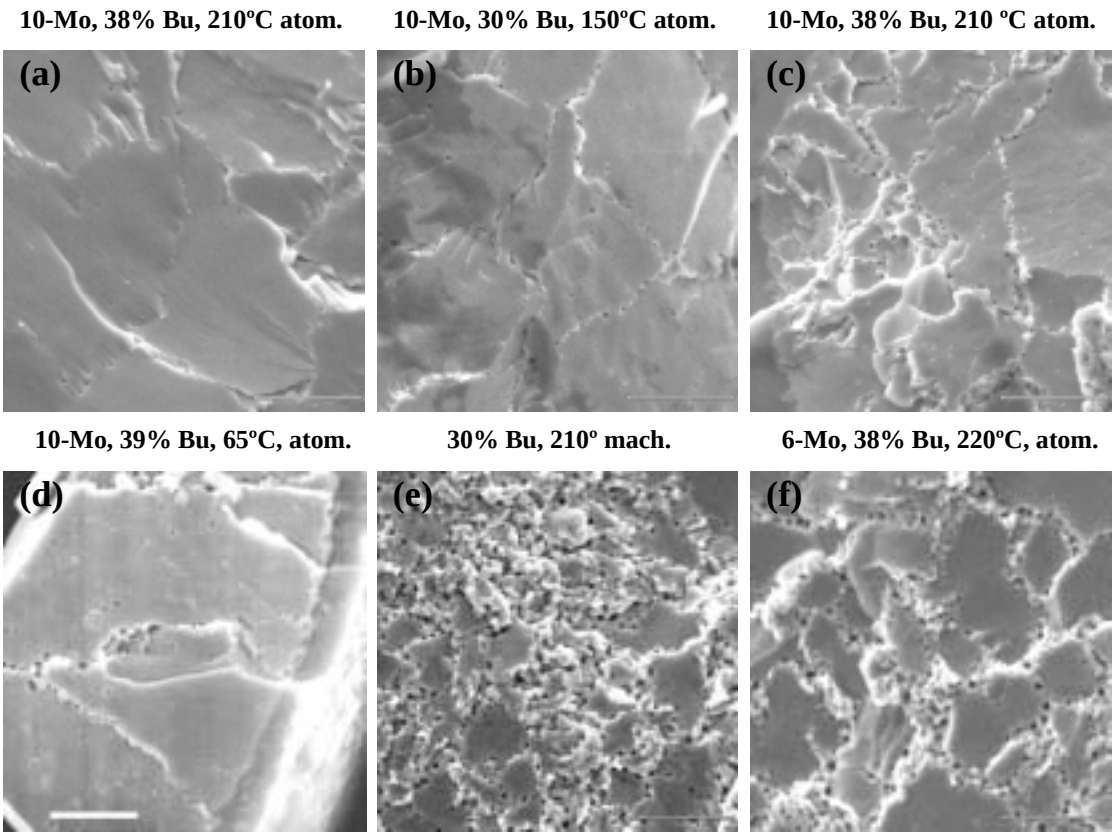

Fig. 2 SEM fractographs of U-10Mo (a-d) showing temperature independent fission gas bubble behavior, and beginning of recrystallization in machined power (e) and U-6Mo atomized powder (f).

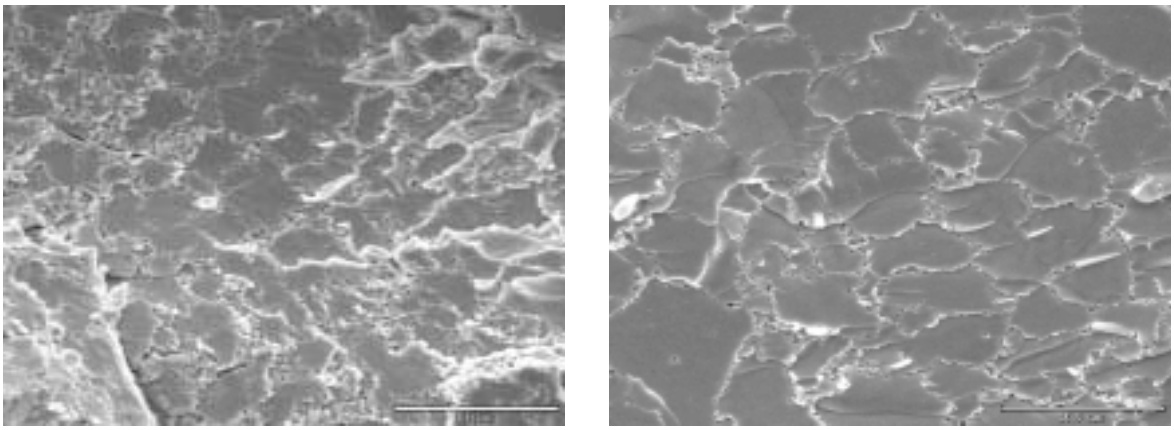

Fig. 3 Microstructure of U-10 Mo fuel particles at 40% burnup (left: machined powder; right: atomized powder) showing fission gas bubbles on grain boundaries and early stage of recrystallization in machined powder. SEM fractographs.

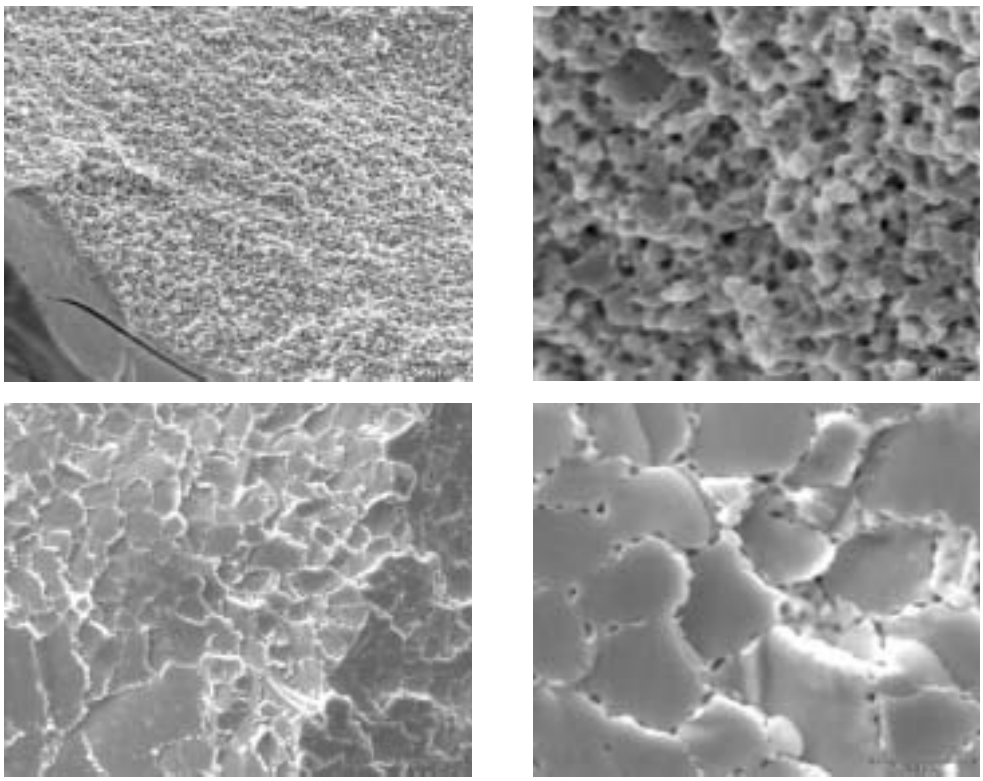

Fig. 4 Microstructure of U-7Mo fuel particles at ~50% burnup (upper: machined powder; lower: atomized powder) showing complete recrystallization of machined powder. SEM fractographs.

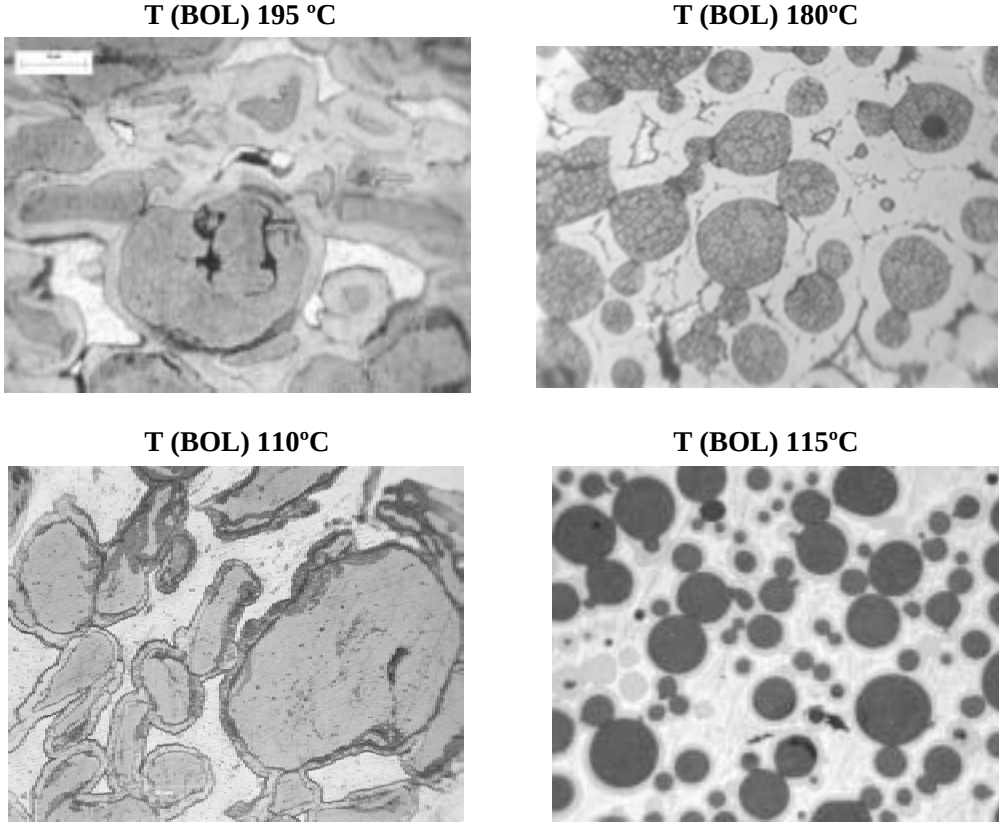

Fig. 5 Optical metallography of RERTR-5 miniplate meat microstructure showing extent of U-Mo/Al interaction at 50% burnup (left: machined powder, right: atomized powder at high and low BOL temperatures).

The net effect on the overall swelling and, therefore, on the plate thickness increase is that a temperature dependence is only apparent at lower BOL temperatures where the formation of the interaction phase is still progressing at 50 and 80% burnup. For the plates irradiated at higher temperatures, the interaction was completed at burnup below 50% when the Al matrix was consumed. This occurred at even lower burnup for 8-g cm^{-3} plates, which have less Al available for interaction and also operate at higher temperatures due to a higher ^{235}U loading compared to the 6 g cm^{-3} plates. Thus, after U-Mo/Al interaction has been completed, the meat swelling is determined by the athermal fission-induced swelling of the unreacted U-Mo fuel and of the (U-Mo)Al_x interaction phase, giving, the leveling off with increasing temperature of the thickness changes as seen in Fig. 1.

There is one set of plates that appears to behave, with respect to swelling, somewhat less predictably. The plate thickness changes in the lowest Mo alloy tested, i.e., 6.5% Mo, particularly the 6-g cm^{-3} set, are outside the range of their higher-Mo siblings (note the star shaped symbols in Fig. 1). Metallographic examinations are being performed to clarify this anomaly. As noted before, recrystallization occurs at lower burnup at lower Mo content, and previous tests [1] have shown that the rate of (U-Mo)/Al interaction is also higher at low Mo concentration. Whether these effects are the cause of the somewhat higher swelling in the 6.5% Mo fuel, or whether, perhaps, a fabrication variable plays a role as well, needs further study. The swelling of the 6.5% Mo plates, albeit less predictable, is nevertheless not excessive.

Finally, two unique miniplates were part of test RERTR-4. They contained each two thin discs of U-10 Mo of ~12-mm diameter and 0.3-mm thickness. The fuel meat density was 15.3 g U cm^{-3}. The meat was thinner than normal in these miniplates in order to remain within the U areal loading limits. These plates showed excellent irradiation behavior; the thickness increases and fuel swelling from these plates derived from the thickness measurements and from a metallographic section shown in Figure 6 are consistent with the extrapolation of the data derived from RERTR-5 miniplates. The bubble morphology shown in Fig. 7 indicates very stable swelling behavior.

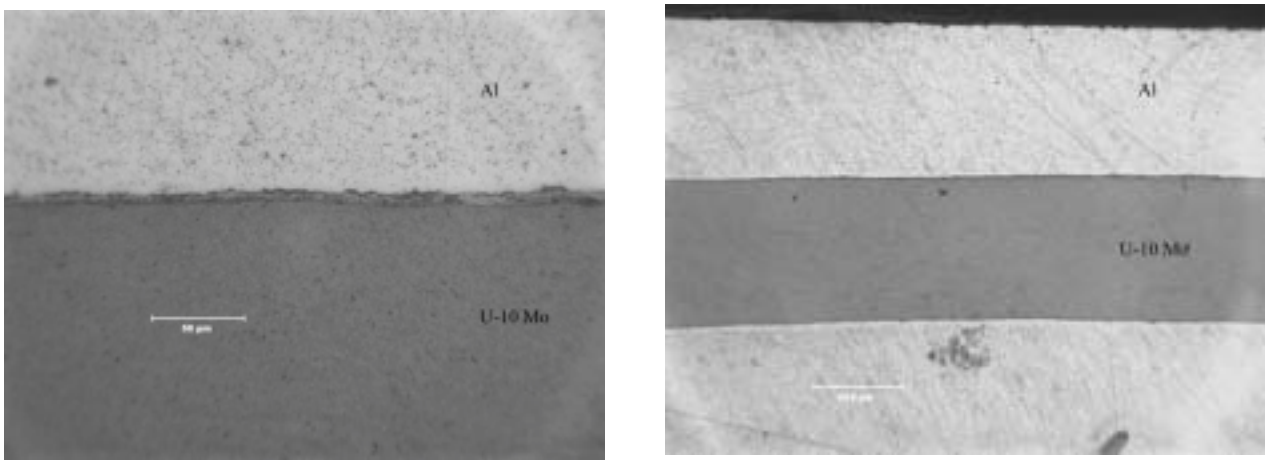

Fig. 6 Metallographic section of LEU U-10 Mo foil (monolithic) miniplate @ 80% U-235 Bu.

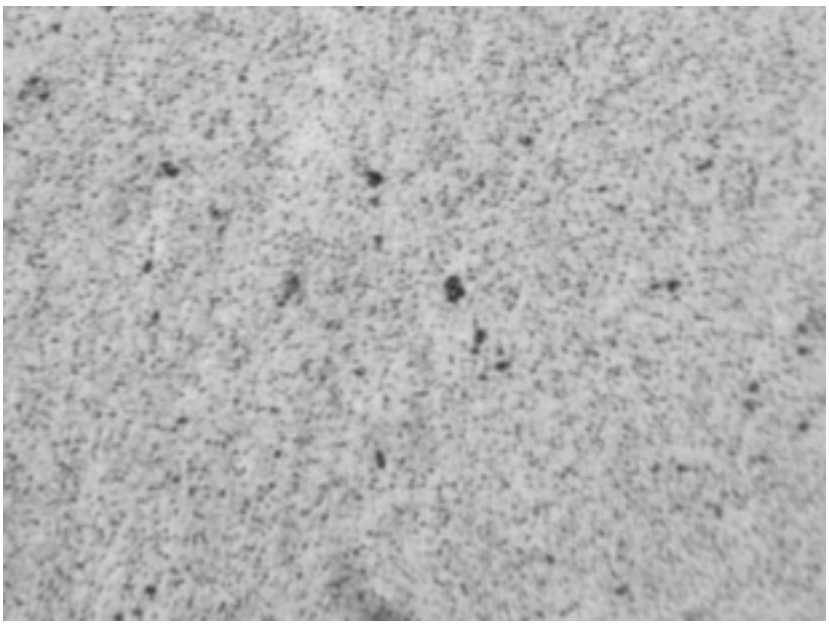

Fig. 7 Bubble morphology in monolithic U-10 Mo fuel at ~80% U-235 Bu.

4. Conclusion

Fuel swelling as represented by fuel plate thickness increases is stable and of acceptable magnitude for both machined and atomized LEU U-Mo powders tested to 80% burnup and beginning of life meat temperatures of 170ºC. The swelling rate increases with temperature up to a point where U-Mo/Al interaction has been completed; after this it is a function of burnup only. All miniplates from test RERTR-4 at 80% peak burnup showed flat and uniform thickness profiles. The results from two miniplates with small solid U-10Mo foils as meat showed them to have stable swelling behavior with values consistent with projections based on the dispersion fuel data.

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STATUS AS OF MARCH 2003 OF THE UMo DEVELOPMENT PROGRAM

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ABSTRACT

The French program for the development of UMo fuel takes benefit of the support of 5 partners acting intensively in the framework of MTR (CEA, CERCA, COGEMA, Framatome-ANP and Technicatom). The aim of the UMo Group program is to deliver a high performance and reprocessable UMo fuel qualified for a wide range of Research Reactor, covering the expected needs for MTR next generation. The program takes into account the milestone of 2006 induced by the US return policy of spent fuel.

The UMo Group program that has been launched in 1999 has already produced significant technical issues based on the fabrication of full sized fuel plates and various irradiation experiments.

The complementary PIE results of the IRIS I irradiation tests in OSIRIS are presented in this paper. Details of the preparation of the two new plates irradiation experiments, respectively the FUTURE experiment in BR2 and the IRIS II experiment in OSIRIS, are mentioned. This paper gives also some comments about the early results obtained from the FUTURE experiment in BR2.

A synthesis of the next steps of the UMo Group program is presented, with the corresponding time schedule.

CERCA UMo DEVELOPMENT STATUS AS OF MARCH 2003

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ABSTRACT

Considered as a suitable solution for non-proliferation and reprocessing purposes, UMo fuel has been chosen and studied by the RERTR program since 1996. Through the French working group whose aim is to qualify UMo before 2006, CERCA is in the process of qualifying the fabrication of U 7 % Mo with a density of up to 8 g Ucm^{-3}.
With a statistical process control (SPC) basis in the workshop and a wide experience of more than 20 years in the silicide and aluminide fuel element, CERCA has already shown its ability to manufacture UMo fuel. In 1999, for the first time in the world, seven full size plates of U-7Mo with a density of 8 g U cm^{-3} had been manufactured. Fabricated with machining powder, this first fabrication had allowed CERCA to define the adjustment needed to improve quality and delivery. Thanks to these first encouraging results, improvement axes have been set up for the second fabrication. Manufacture of U-7Mo full-scale plates were scheduled for 2002 using atomized powder.
This paper focuses first of all on the manufacturing results obtained on UMo full scale plates manufactured in 2002. Plates are planned to be irradiated in OSIRIS and an UMo lead test assembly in R2 reactor. Two plates are under irradiation in BR2 reactor (2^{nd} cycle completed). In a second section, we introduce the orientations and achievements which CERCA has already implemented so as to be able to produce in 2006 U-7Mo fuel through an industrial process which will lead to cost reduction, while retaining quality and on-time delivery.

1. Introduction

CERCA, a world-wide leader for civilian MTR fuel supplies, is continuously involved in the development of LEU fuel. Within the frame of the international RERTR program, the French UMo Working group (CEA, CERCA, COGEMA, TECHNICATOME and NOVATOME) aims at qualifying U-7Mo fuel with a density of 8 gUt/cm^3 before the end of 2006. In this way CERCA's goal is to succeed in the UMo fuel production on an industrial scale which covers quality, reliability and time delivery in order to satisfy the MTR community needs. Based on our own experience of high density U_3Si_2 fabrication, our contributions are to set up as well as to get a better understanding of specific processes involved in the manufacturing of high density UMo fuel.
For this and in coherence with the French UMo working group strategy and specific costumers needs, R&D full size plates were manufactured. This policy which has been carried out since 1998 has allowed us to define specifics actions and investigations in order to improve the knowledge linked to the UMo behaviour through the CERCA advanced process.
After the first UMo full size plates production presented in 1999 and 2000 [1][2] the results obtained according to our latest R&D development program are described below. These prototypes produced a few months ago, are, or will be irradiated in 2002/2003 according either to the French UMo working group or R2 reactor irradiation schedule.
The second part of this presentation introduces CERCA strategy and describes the main steps elaborated for the UMo workshop configuration set up.

2. R&D UMo fuel plates manufacturing and irradiation

After the first results obtained under irradiation in OSIRIS I (IRIS device) with surfacic power of 170 W/cm^2 and HFR (UMUS device) with 250 W/cm^2, two new full size plates irradiations in the frame of the French UMo group have been scheduled for 2002/2003.
This second set of irradiations will allow us firstly to test a new type of UMo powder. Secondly, limiting the cladding temperature while operating, a new set of data will be available in order to define the acceptable range for satisfactory UMo behaviour under irradiation. Moroever and dedicated to the costumer needs, CERCA, in close collaboration with R2 reactor, had manufactured in 2002 the first U-7Mo lead test assembly world-wide which will be irradiated in 2003 [3].
For these, U-7%Mo atomized powder with Low Enriched Uranium (LEU) were used and cladding temperature will be kept below 130 °C. The irradiation conditions associated with the UMo plates described in this paper are mentioned below in Table 1. The global UMo plan development and irradiations schedule is presented here after in Table 2. Concerning the french UMo working group project schedule, it can be pointed out that, the post irradiation examination (PIE) of OSIRIS and HFR presented during the 2002 RRFM [4] are still in progress. Explanation of the failure occured in the plate U-9Mo with an enrichment of 35% during the UMUS experiment is presented in this proceeding [5]
Furthermore, honoring our global schedule commitments, CERCA will be ready for industrial UMo delivery by the end of 2006. Concerning CERCA specifically, R&D UMo development program resources have been set up to optimize our advanced process through UMo fuel.
One is the numerical simulation of UMo plate rolling with a university partnership wherein significant improvement is expected for some plates characteristics. Secondly, we are evaluating internally and externally the different production mode of UMo powder. Technical, industrial and safety aspects are considered for this evaluation. Also encouraging results have been already obtained in the up-stream part of our advanced process and NDT studies have also been launched.

2.1 Powder manufacturing

UMo Powder production as a sensitive part of fuel manufacturing has to be evaluated.
Considering the powder production itself, machined UMo powder had been tested by CERCA three years ago. With a lot of difficulties due to UMo high ductility, this mode of powder manufacturing was questioned at the time with regard to production on an industrial scale.
Hydriding, milling and de-hydriding (HMD) is another way to produce UMo powder.
According to our own studies and the results obtained this production mode could be considered for the moment only for reduced quantities of powder. Further investigations are needed to reach industrial scale production.
Considering the mechanisms of hydriding and the sensitivity of each parameter, industrial scale production has to be demonstrated and R&D efforts have to be made to produce HMD UMo powder in a reproducible and safe manner. Atomization of UMo is the third way to produce UMo powder.
Industrial production scale is demonstrated externally, but safety concerns have to be well managed and could reduce productivity. Investment and cost development have also to be considered.
Considering the intrinsic principle of atomization, the UMo fusion in a crucible, as a crucial step for quality aspect, needs to be carefully controlled to avoid any segregation of Mo while operating.
On the HMD or machined powder production a pre-heating (homogeneous heat treatment) step could be used in order to obtain good UMo homogeneity.
Industrially, CERCA has to investigate the UMo powder production process further while at the same time evaluating powder compatibility throughout the CERCA manufacturing process.
From our point of view, final product quality and reliability is a function of UMo powder characteristics. Results from PIE examinations reflecting the behaviour of different types of UMo powder under irradiation will guide the final choice for an ad hoc UMo powder process production.

2.2 Fuel size plate manufacturing

The plates produced for BR2, OSIRIS II and R2 irradiations were manufactured with spherical and gamma phase particles obtained by atomization.

Figure 1 presents UMo particles in the fissile core after the CERCA manufacturing stage.
The homogeneity of U and Mo obtained with an energy dispersive spectrometer (EDS) from one of the UMo particles center to the Al matrix of the meat is presented in Figure 2. Interaction layer of UMo / Al in as-manufactured plates is not seen with CERCA's production parameters. The main characteristics of full sized plates are given in the table 3 below.

Reactor	**OSIRIS II (Plates)**	**BR2 (Plates)**	**R2 (Assembly)**
Max surfacic power (W/cm^2)	250	350	200
Max cladding temperature (°C)	< 130	< 130	<130
Max BU (%)	50 or 70	50	

Table 1: Irradiation conditions for U-7Mo full size plates or fuel assembly

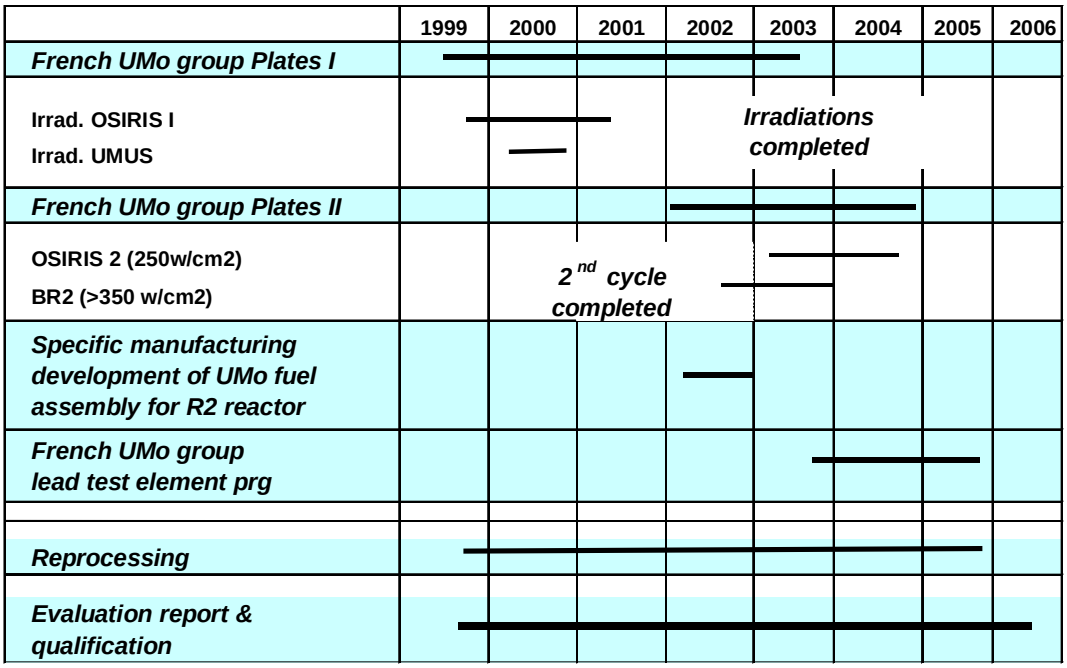

Table 2: French UMo working group and CERCA specific development time schedule

Identification	**Main plates caracteristics**		**1**	**2**	**3**	**4**	**5**	**6**
Reactors	R2		OSIRIS (II)				BR2	
% Mo in alloy	7.2		7.6				7.3	
Enrichment (%)	19.80		19.80				19.82	
^{235}U (g)	< 27.00		28.06	28.04	28.02	28.01	30.49	30.51
Meat density (gU cm^{-3})	7		8.3				8.5	
Porosity (%)	< 1		1.5	1.6	1.4	1.1	1.1	1.3
Cladding thickness (mm)	0.38 (int)	0.57 (ext)	0.36				0.38	
Meat thickness (mm)	0.51		0.51					
Cladding material	AG3NE		AG3NE					

Table 3: U-7Mo plates characteristics produced by CERCA in 2002

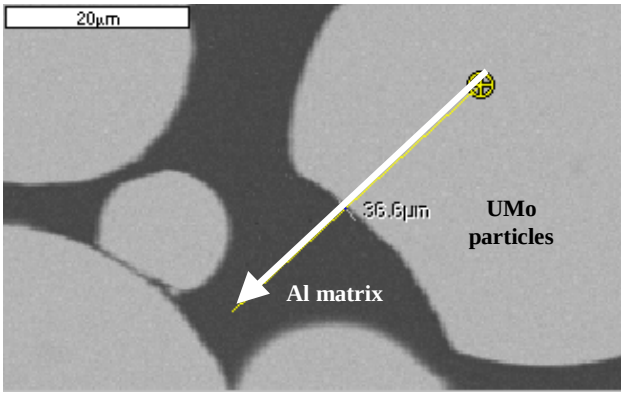

Figure 1: SEM picture of UMo particles in the fissile core after manufacturing

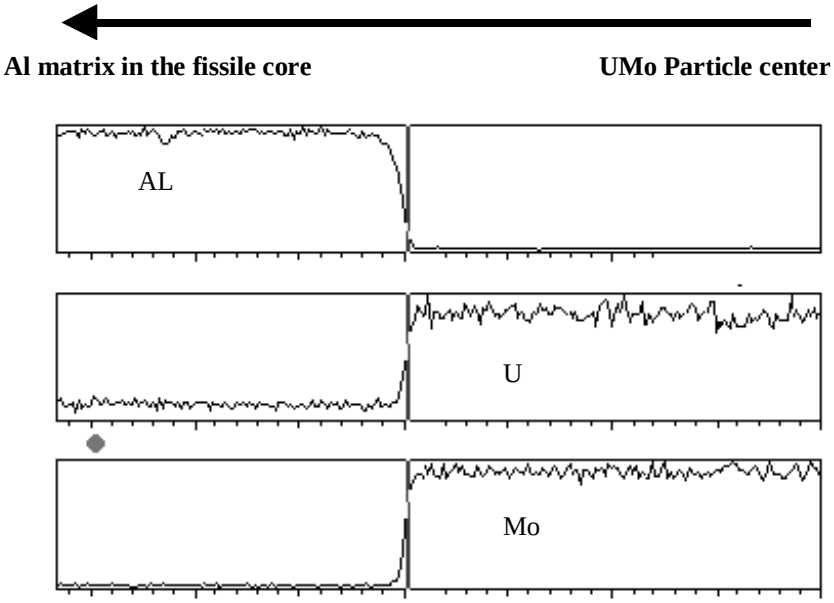

Figure 2 : EDS scanning of the U-7Mo fissile core after CERCA manufacturing stage.

2.3 Inspections results

2.3.1 Destructive examination

The meat and cladding thickness are measured by destructive examinations. The micrographs in Figure 3 show the meat shape in the main parts of the plates (longitudinal direction). UMo Spherical powder is compatible with our advanced process and confirms the good results already obtained.

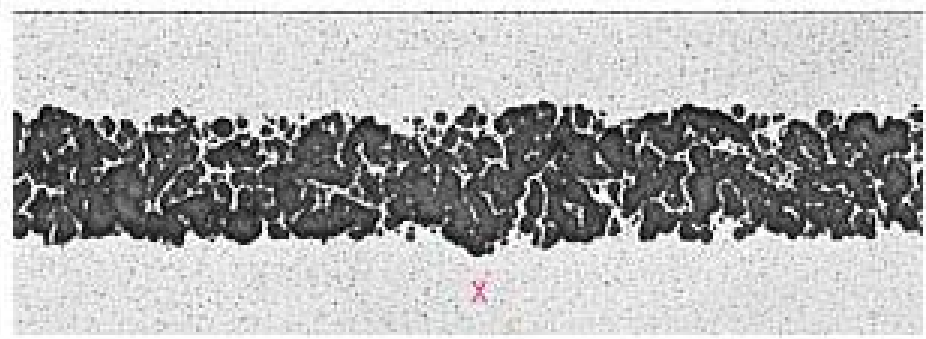

Figure 3: Micrographs of the U-7Mo plates with 8 gU/cm^3-Current part-

2.3.2 Homogeneity of U distribution

The homogeneity of Uranium distribution is quantitatively inspected by means of a radiographic system with X ray spot that scans the length of the plate. The uranium density is determined by measuring the absorption of the beam (Figure 4).A significant improvement in homogeneity results is obtained. This improvement is linked mainly to our last advanced process development. In parallel, the atomized UMo powder enables local lopping of some irregular interface shapes between the fissile core and the cladding frame. The homogeneity average is then controlled by the CERCA process when the local response is optimized with atomized UMo powder.
As a function of the manufacturing process itself and the material used, homogeneity results will be optimized. Some studies are under way in order to improve the UMo / Al powder mixing step as well as the rolling procedure. Improvements connected with the reproducibility of homogeneity are expected for our future production.

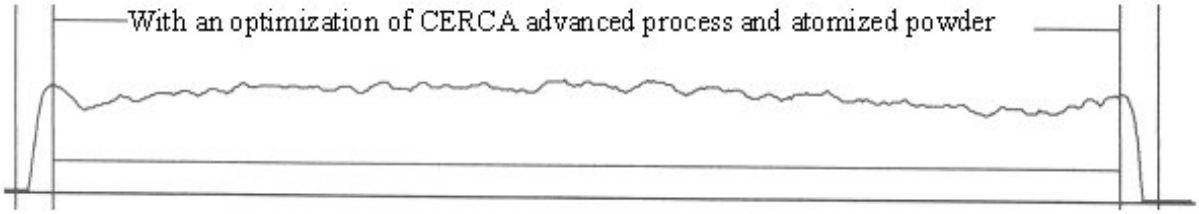

Figure 4: Homogeneity of Uranium distribution in U-7Mo plates with 8 gU/cm^3

2.3.3 Stray fuel particles inspection

To avoid local temperature increase and fission products release in some particular plate areas during irradiation, the stray particle specification has to be well defined. This is very sensitive with high density fuels. Then, depending on customer specifications, plates are acceptable or rejected. According to our long feed back experience, many manufacturing parameters impact the presence of free particles. From the compaction procedure to the rolling step, the stray fuel particles, which can be seen by examination of properly exposed X radiographs, are, statistically speaking, very difficult to avoid. Thanks to our considerable experience, CERCA has been able to produce plates that were acceptable to our customers without any problems.

2.4 Conclusion

After the first manufacturing experience in 1999 including Mo variability contents and different enrichment values, more than 25 new full size UMo plates with a density from 7 gU/cm^3 up to 8 gU/cm^3 were manufactured at CERCA in 2002. Thanks to the new R&D procedures implemented in the workshop, improvements had been seen on the U-7Mo full size plates. Significant improvements in homogeneity are obtained and confirmed the CERCA R&D developments previously defined. The atomized UMo powder tested is compatible with CERCA's advanced rolling process.
Concerning UMo powder production, CERCA is willing to test all types of UMo powder in order to have a global view of these topics. Further investigations on a specific device for UMo powder production will be carried out in 2003 so as to update and complement the preliminary evaluation file.

3. Workshop part

In order to be able to produce and sell a new generation of high performance fuels not only is it of utmost importance to master production technology, but it is necessary to be able to provide, at any time, the highest level of quality and service that customers expect.
Due to our experience with U3Si2 fuels, we know that the road from development on a laboratory scale to industrial scale is long and difficult. Consequently, a lot of things need to be anticipated.

In the following items, as one of the leaders in the fuel development, but, above all, as an industrialist that sells fuels all over the world, CERCA reviews the different actions already in hand or on standby to be ready when UMo fuel is qualified.

3.1 The production process

Generally speaking, since the beginning of UMo development in 1997, CERCA has chosen to produce only full size plates for irradiation tests in order to be as close as possible to industrial conditions. That has been very useful for producing files for statistical analysis on production and inspection processes.

The operating points with their stability areas are now well known.

3.2 Information process

As CERCA complies with all the Quality Standards, traceability and uranium accountancy is the main concern in a process with so many transformation steps.
To solve this difficult problem a computer system has been developed to monitor uranium throughout the process lines. At any time it is possible to know the exact amount of uranium contained in each ingot used in a plate. This new system has been parameterized to take into account UMo Alloys.

3.3 Criticality

If production is considered in a productive way, the question of economical batches quickly arises, but due to the specificity of our business, critical mass, with the different margin coefficients, provides the upper limit.
Preliminary neutron calculations are being carried out in FRAMATOME ANP and will go in depth in 2003 . These results are very useful for managing production fluxes through the process lines and preparing files for the safety authorities so as to get their approval of an industrial production.

3.4 Safety

As the wish of our shareolder (FRAMATOME ANP, a joint AREVA and SIEMENS company) is the everlasting quality of MTR fuel supply, a lot of investments have been made to comply with the new safety rules. Therefore CERCA can be considered a reliable supplier due to more than 30 years experience and the constant development and renewal of production means within the framework of one of the most severe safety references.

3.5 Services

Because we think that the future of a manufacturing plant is the associated services around the manufacturing of fuel elements, CERCA has developped knowleges in the following items:
Experience in the transport of Uranium: CERCA supplies Uranium for almost all its customers. CERCA has been the main logistic actor in the supply of Russian uranium for French and German reactors.
Experience in the delivery of fuel elements to reactors: Since 2001, CERCA has developed a new "CERCA 01" cask licensed, in accordance with all the laws and regulations actually in force in many countries. This cask is already used for the shipment, all around the world, of U3Si2 and UAlx fuel elements. CERCA is rather confident in the issue for an extension of this agreement for UMo fuels.
Development of a recovering process of scraps inherent to the fabrication process of elements. The Uranium which is in the scraps become usable.

4. Conclusion

To develop a new fuel requires a lot of technical resources, a clear vision of the development program and constant strength to reach the aim of qualification in 2006. That is the reason why, in the French

UMo Group, CERCA has constantly steered a course towards the development program since the early schedule presented in 1999. All valuable technical options are being considered to be sure that the right choice is made regarding the industrial scale.
In 2002, for the first time world-wide, an U-7Mo lead test assembly for R2 reactor was manufactured and delivered by CERCA.
Thanks to the plates which are under irradiation, the UMo specification will be finally adjusted in order to point out the main parameters and the associated range needed to be fixed.
However, to be ready from the technical point of view is necessary, but not sufficient. The fuel business has a long time constant. We have to submit our projects to safety authorities, and so, a lot of things must be anticipated as far as industrial production is concerned. From that point of view, CERCA has launched, in parallel, an industrialization program taking into account the whole production processes up to fuel shipment. The aim is to offer in 2005 a stabilized, qualified, licensed and cost effective fuel production process.
In the meantime, CERCA is willing to help any reactor that needs, either fuel assemblies for tests or contacts or information from the French UMo Group.

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STATUS OF CNEA QUALIFICATION PROGRAMS FOR THE FABRICATION OF HIGH DENSITY MTR PLATE TYPE FUEL ELEMENTS

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ABSTRACT

CNEA has endeavored to qualify as a manufacturer of MTR high-density fuels. Silicide and U-Mo based fuels are included in this qualification process. In both cases a qualification program was established and an irradiation test for the Silicide dispersion plate type fuels fabricated by CNEA was launched in RA3 in September 2000. These irradiations are expected to be completed by the middle of 2003. At the time of this paper preparation the burnup of both fuel elements is 40 % for P-06 and 28 % for P-07. The U density of the fuel plates in both cases is 4.8 g/cc and the material was prepared in accordance with NUREG 1313 recommendations. The main activities in the U-Mo qualification program were in the fields of the development of an alternative route for the fabrication of the U-Mo powder, the set up of the manufacturing of U-Mo fuel plates and the improvement and development of characterization techniques for high density fuel plates quality control. This techniques include US testing for the detection of unbonded areas and densitometric analysis of X-ray pictures for the evaluation of the U homogeneous distribution. The fabrication of two U-Mo fuel assemblies under the framework of the RERTR international qualification program is expected to be accomplished during 2003. The fuel plates for these fuel elements are to be manufactured with U-7%Mo and with 7 gU/cc using both powders, from atomization and from hydriding-milling-dehydriding (HMD). This paper describes the main activities performed during the last year and also provides information about the future steps of the qualification programs.

1 Introduction

CNEA, the National Atomic Energy Commission of Argentina, continues focusing its activities regarding fuel for research reactors on two programs related with the development and qualification of LEU high-density fuel manufacturing mainly based on uranium silicide and U-xMo alloy.
This paper describes the different activities performed within each program during the last year and outlines the main advances and achievements during this period.

2 Silicide fuel activities

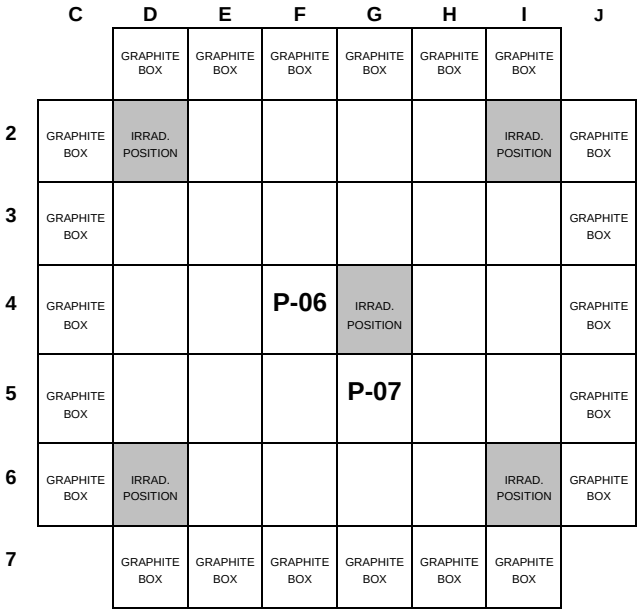
During 2002 the activities around the silicide qualification program were mainly around the continuation of the leading fuel elements P-06 and P-07 irradiation in the RA-3 reactor and in the set up of new quality control techniques for silicide fuel plates.

2.1 Irradiation of the silicide fuel elements P-06 and P-07

2.1.1 Background

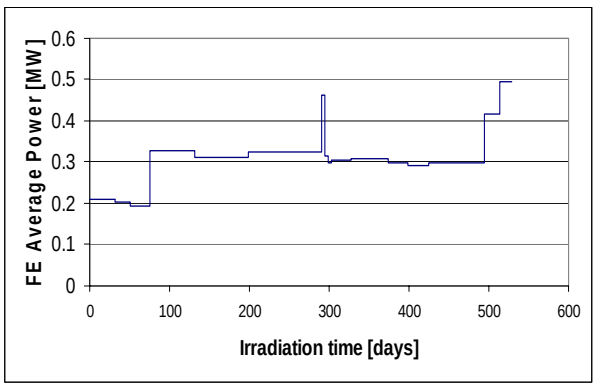
The irradiation of P-06 and P-07 started in September 2000 and September 2001 respectively.The following table and the Figure showing a schematic representation of the RA3 core summarize the positions where the two fuel elements have been irradiated.

Irradiation Time		Position	
From	To	P-06	P-07
30-Sep-00	22-Nov-00	D3	-
25-Nov-00	20-Dic-00	D4	-
10-Apr-01	15-Aug-01	G5	-
18-Aug-01	19-Sep-01	F4	-
22-Sep-01	19-Jan-03	F4	G5

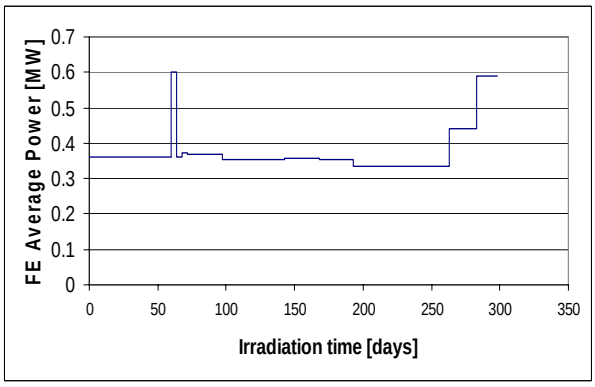


2.1.2 Burnup Evolution and Irradiation Parameters

At the end of the last irradiation cycle included in this report the burnup of both fuel elements was 40% for P-06 and 28 % for P-07. Burnup data was obtained from the RA3 fuel core management. The power histories of P-06 and P-07 are represented in the next Figures. At the end of the last cycle considered in this report there was no evidence of fuel failures or any other abnormalities and the operation of the reactor continued as it was programmed.



P-06



P-07

2.1.3 Pool side inspection program – Last examination of P-06 and P-07

As was defined in the Qualification Program, P-06 and P-07 Fuel Elements are inspected periodically in the reactor pool during the irradiation. The purpose of these inspections is to verify the structural integrity of the FE and the state of the fuel plates, structural components and cooling channels. The FE are visually inspected. The gap between fuel plates is visually evaluated with the aid of back lighting. The visual examination also includes the inspection for blisters, distortions or other phenomena that could affect the gap between fuel plates. The last inspection of both FE was performed during September. Burnup at that time was 32 % for P-06 and almost 20 % for P-07. Both fuel elements showed no evidence of deformations or distortions. The general appearance remains without modifications. No defects or changes of surface roughness were observed. The fuel plates show a parallel arrangement without any evidence of blistering, bowing or distortions of any type. Along the different visual inspections the gap between fuel plates remains unchanged in both cases. No oxide layer or crud deposition thickening of the fuel plates was detected. The removal and the introduction of both fuel elements from and in their positions in the core were performed without any inconvenience. There was no evidence of interaction with the other fuel elements of the core or with

the internals. The next two Figures are pictures of P-06 and P-07 obtained during this pool-side inspection showing the fuel plates from the top of the fuel.

2.2 Schedule for the remaining activities of the Qualification Program

Estimations based on the available information including the uprated power of RA3 indicate that P-06 and P-07 will be reaching their discharge burnup of 55 % before May 2003 and July 2003 respectively.

A schedule of the remaining activities of the Qualification Program was prepared consistently with those estimations. The main information of the schedule is summarized in the following table:

Fuel Element	End of Irradiation (55 % Burnup)	EOL In-Pool Visual Inspection	Hot-cell examination
P-06	May 2003	June 2003	2004
P-07	July 2003	August 2003	

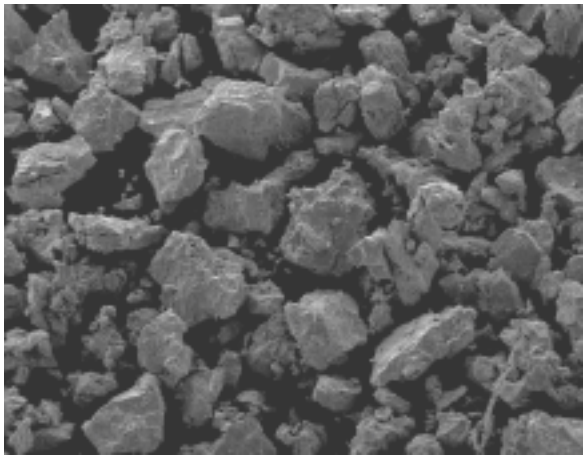
3 U-Mo Fuel Activities

CNEA is participating in the international program to qualify U-Mo as a nuclear fuel for research reactors and also is carrying out a domestic qualification program of the technology and facilities involved in the manufacturing of aluminum dispersed U-Mo nuclear fuel for research reactors. In the framework of the international program two fuel elements with UMo from the atomization process and from the HMD process will be fabricated. The main activities accomplished during the last 12 month are related with the procurement of the UMo powder and the set up of procedures and techniques for full scale fuel plate manufacturing with both powders.

3.1 The HMD Process to manufacture U-Mo powder

The HMD process is accomplished in four steps that have been fit in their characteristics for a scale production of approximately 50 kg/year of U-7Mo powder in one-kilogram batches. Temperature, atmosphere and time were specified for the initial hydrogen incorporation, hydriding and dehydriding steps. The milling of the hydride is performed in low impact mills. The hydriding and dehydriding have been fully automated. Conventional equipment that requires small manpower and low investment is used.

Actual development is focused in the optimization of the milling process of the hydride phase using low impact mills to obtain less than 30% fines (< 45 µm). Full optimization of the production process is in its way and a facility is being built up for performing initial practices with enriched uranium. This facility will be shortly in production. The following Figure shows the morphology (SEM) of the powder and the typical particle size distribution for the present stage of the development (large particles å 100 µm)



3.2 Set up of miniplates and full scale fuel plates manufacturing using U-Mo powder

Miniplates:

Eighteen UMo miniplates were fabricated. Four with atomized powder and fourteen with HMD powder. The experience was useful for a better understanding of the influence of parameters that affect the homogeneity of the U distribution.

Full scale fuel plates:

The development work was related mainly with the improvement of the U distribution in the fuel core. The activities were focused in the study of the influence at full plate scale of the blending of the powders, the techniques for filling the die and the modifications of process parameters like the characteristics of the Al powder.

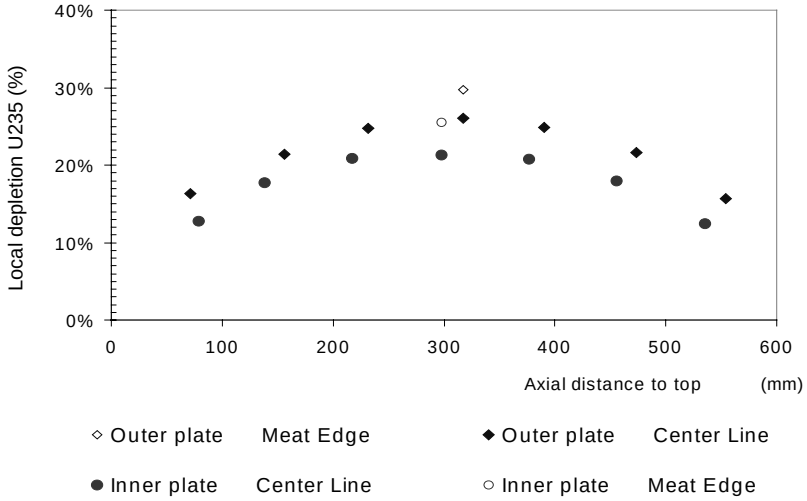
The experience and some results are summarized in the following table:

Powder	Quantity of Fuel Plates	U Density [g/cm3]	Al Vol. [%]	Meat Thickness [mm]	Porosity [%]
Spherical	16	6 – 7.5	48 - 59	0.4 – 0.6	2.4 – 7.5
HMD	5				

Recently four additional fuel plates were fabricated. Although the homogeneity of the fuel plates, evaluated by visual inspection of the radiographs, would be acceptable for irradiation, further improvements are still possible, mainly in relationship with the filling techniques and the parameters of the aluminium powder.

4 Silicide Fuel PIE destructive studies in CNEA Hot Cells

The works in the hot cells continued during year 2002 with the preparation of samples from irradiated fuel plates of the P-04 silicide fuel element (a prior stage of the CNEA qualification program) for metallographic examination and for accurate burnup determination through chemical analysis. Seven 4x4 mm samples from the axial centerline and also from the area close to the meat edge of an inner plate and an outer plate were obtained with this last purpose. The samples were processed by the acid dissolution and ion-chromatographic separation. Thermo-ionization mass spectrometric was used to determine the isotopic composition of U. The local depletion of ^{235}U as Bu indicator was evaluated. The local burnup distribution in both plates is shown in the next Figure.



5 Other activities

Works for the development of ultrasonic testing of fuel plates are also in progress. CNEA is developing the methodology to inspect the bonding between cladding and fuel core by means of ultrasonic techniques. To assess the sensitivity, reproducibility and the resolution of the equipment, a fuel plate with known bonding defects of 1.0 and 1.5 mm diameter is been used. This fuel plate was fabricated using spherical UMo fuel particles. At the present the main activities are related with:

- development of a rejection criteria according to specifications from experienced fuel users
- new design of the fuel plate with standard defects
- influence of parameters like the relative position of the transducer and speed of the fuel plate

6 Final Remarks

The fuel elements of the CNEA Qualification Program as manufacturer of High Density Uranium Silicide MTR Fuel Elements continue their irradiation in RA3 without evidences of damages or failures. The present burnups of P-06 and P-07 are 40 % and 28 % respectively The pool-side inspections showed that the general aspect of both fuel elements remain unchanged without any distortion or other abnormalities. P-06 and P-07 will expectedly complete their irradiation before May-2003 and July-2003 respectively. The HMD process to fabricate U-Mo powder has been fully automated. During the set up of the UMo fuel plate manufacturing a reasonable quality in U distribution suitable for irradiation has been achieved. Despite that is still possible to obtain furthers improvements. With this objective several alternatives are been analyzed and more fuel plates will be manufactured in the near future.

7 References

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8 Acknowledgments

We deeply recognize the collaboration and assistance of M. Markiewicz, H. Taboada, A. Piazza, A. Pérez and many other contributors to the CNEA Qualification Program. We also thank to Ross Finlay for the pictures of P-06 and P-07 visual inspection.

MANUFACTURING OF URANIUM HIGH DENSITY FUEL ELEMENTS USING URANIUM 73% U^{235} ENRICHED, IN AN ALUMINIDE ALLOY FOR THE BR2 REACTOR

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ABSTRACT

By end of 2001, CERCA proposed to develop, manufacture and deliver to the BR2 reactor sited in Mol (Belgium), a new design of fuel element based on a 73% U^{235} enriched aluminide alloy as an alternative to the common HEU element (93% U^{235} enriched).
The development step challenge was to maintain the fuel element performances as high as possible while keeping the present acceptance criteria for the fabrication process. Specifically, the minimum cladding thickness of the fuel plate had to be maintained.
Various parameters, possible process improvements & alloy density were scanned in order to satisfy the reactor needs & constraints.
The results of the development program demonstrated it was possible to switch to the lower enriched alloy while maintaining the total amount of U^{235} into the core.
We propose to present how this specific development program was conducted and discuss the main results.

1. Introduction

The BR2 reactor sited in MOL (Belgium) is commonly running using HEU type fuel element manufactured by CERCA.

The fuel elements are made of 15 or 18 fuel plates properly roll swaged into 3 solid radial webs so as to form concentric cylinders.

The nominal U^{235} content is 364 g per each 5 tube fuel element (15 plates) and is 400 g per each 6 tube fuel element (18 plates).

The actual specification is valid for an aluminide alloy, U^{235} enriched from 89 % to 93.5 %.

In 2001 CERCA located a batch of 73 % enriched U metal. It was immediately anticipated that this material could be used as an alternative to the common 93% enriched HEU. The main issue was then to check whether or not this material could be burnt with maintaining equivalent performances of the reactor.

2. Scope of CERCA's proposal

By end of 2001, CERCA proposed to develop, manufacture and deliver to the BR2 reactor a new fuel element design based on a high density aluminide alloy using uranium 72 % to 74 % U^{235} enriched as an alternative to the standard 90-93% enriched HEU alloy.

This new element should be designed to approach the nuclear performances of a standard HEU fuel element as close as possible without any negative consequence for the operational safety of the reactor.

The overall scope of CERCA's services includes:
- Material purchase;
- Material transfer to CERCA's Romans plant and storage;
- Conducting the development of the high density fuel plates;
- Manufacturing, shipping and on site delivery of 6 lead test elements;
- Addressing experience feedback as necessary;
- Manufacturing, shipping and on site delivery of 2 batches of 30 high density elements.

3. Main technical issues

The requested nominal U^{235} content per unit area is 60 mg/cm^2. So, switching from a [93.5% - 89%] to a [74% - 72%] enriched alloy means that the nominal U^{tot} content may be increased by up to 29 %.
- 60 mgU^{235}/cm^2 → [64.2 – 67.4] mgU^{tot}/cm^2 for [93.5% - 89%] enrichment
- 60 mgU^{235}/cm^2 → [81.1 – 83.3] mgU^{tot}/cm^2 for [74% - 72%] enrichment

In fact, due to the tolerances on enrichment, U content per plate, core thickness and others, a maximum U content of 86 mg/cm2 should be reached.

For high density aluminide fuels, the present CERCA experience was limited to a maximum 80 mg/cm2 U content and another fuel element design. The purpose of the development program was therefore to demonstrate the CERCA's ability to manufacture a fuel element with a higher U content to maintain the nuclear performances, while keeping the technical specifications for fabrication within the present acceptance criteria.

4. Development program and results

The development program conducted by CERCA was based on a statistic Taguchi type test plan. This test plan scanned various alternatives of the CERCA's advanced fabrication process and various U contents into the plate, from a known configuration up to another with the highest U content.

The tests were performed on full scale plates, using the same base materials (except depleted uranium) :

- aluminum powder
- poison powders
- cladding material

so that the design-basis parameters could be easily compared with those of standard plates. More particularly, it was verified that the obtained results for cladding thickness and homogeneity of the U distribution were similar to those gained from standard 90 % or 93 % enriched plates.

The two following figures are cross-sections of the plate center area. They show clearly the higher U^{tot} content in order to maintain the same U^{235} content. Also, it can be seen that the cladding thickness values are very close.

Standard 90 % enriched plate

New high density 73 % enriched plate

Here below are the associated U distribution test results for both 90 % and high density (HD) 73 % enriched alloy plates. The conducted development program demonstrated it was possible to maintain a very homogeneous U distribution in the core with a much larger U content.

Present 90 %

New HD 73 %

Nevertheless, one issue had to be specifically looked at.
Indeed, the transient area that was perfectly controlled in the previous classic configuration needed to be addressed in order to get an acceptable cladding thickness in the so called dog-bone area.

HD 73 %
before working on the dog-bone

The two following figures are cross-sections of the dog bone area on HD 73 % enriched plates during the development program, before and after working on the cladding thickness issue.

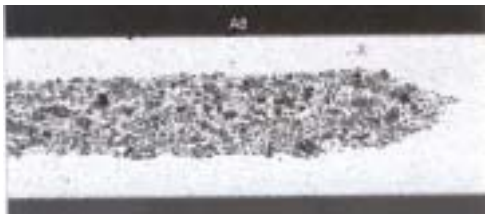
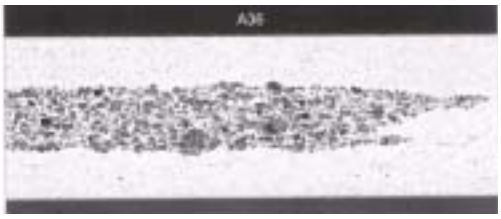


Plate with dog bone



Dog bone issue addressed

Clearly the deployment of the test plan allowed improvements of the fabrication process, notably for the transition zones. Most probably this improvement will avoid many concerns during later fabrications.

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

5. Conclusion of the development program

Finally, the amount of analyzed data was significant enough to propose a draft of technical specifications for fabrication of high density plates.

CERCA and BR2 closely cooperated for jointly finalizing the specification draft by selecting among the available results, the most adapted answers to the reactor needs & constraints.

6. General conclusion

In order to answer to its customer needs, CERCA intends to propose fully integrated services as material procurement, fuel element development / manufacturing and logistics.

The example of the BR2 HD 73 % enriched fuel development shows that when a situation has been correctly anticipated, CERCA is able to deploy all the necessary competencies in order to find adequate solutions. In the present case the development of a new high density fuel was only possible because CERCA and BR2 decided to allocate the necessary resources and to work closely together.

MANUFACTURING AND LICENSING OF A LEAD TEST ASSEMBLY FOR THE R2 REACTOR

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ABSTRACT

In Sweden there is a law stating that a reactor operator must have a final solution for the back-end of the fuel cycle. The R2 reactor, operated by Studsvik Nuclear outside Nyköping in Sweden, has such a solution at hand until May 2006, through the US policy regarding Foreign Research Reactor Spent Nuclear Fuel, FRRSNF. For the period after that date the RERTR program has been working on a new fuel type the UMo fuel. However, this program is lagging behind and the full qualification program will not be finalised in time for the R2 needs. Based on this Studsvik Nuclear decided to go along on its own with an LTA-program aiming for the full qualification of an UMo fuel design in 2005.
In 2002, CERCA proposed to develop, manufacture and deliver to the R2 reactor, a new design of fuel element based on the alloy (UMo 7% alloy) as an alternative to the common Low Enriched Uranium silicide element. The development challenge was to keep the fuel element performances as high as possible without going beyond the actual inspection criteria. The specification agreed between STUDSVIK and CERCA showed the possibility to manufacture this high-density fuel element in CERCA's facility in Romans-sur-Isère (France).
We propose to present how the fabrication was conducted and what were the main obtained results and how the licensing of and the introduction into the R2 reactor are planned.

1. Introduction

For a long time, CERCA, as a fuel manufacturer, and Studsvik Nuclear, as a reactor operator with two MTR reactors, R2 and R2-0, have maintained a close relationship regarding LEU fuel development. In fact, after the ORR test with a full core loaded with Silicide fuel, R2 became in 1984 one of the first reactors in the world to activate, and in 1987, to start irradiation and reactor testing several U_3Si_2 fuel assemblies. One of these first U_3Si_2 assemblies was manufactured by CERCA. These first assemblies were manufactured to demonstrate the acceptable range of U densities that was feasible, both from a manufacturing as well as from an operating point of view. The assembly manufactured by CERCA had a U-density of 4.8 gU/cm^3, corresponding to the maximum density licensed by the US NRC in 1988 [1].
Since then, almost one thousand U3Si2 fuel assemblies have been manufactured by CERCA whereof about 500 assemblies have been delivered to the R2 reactor without any reactor operating problems. The R2 fuel assemblies have all had a lower fuel meat density of 3.9 gU/cm^3.

Recently, the U_3Si_2 fuel design for R2 has been slightly changed. In order to increase the cycle length and consequently decrease the cycle cost (and in the same time the amount of spent fuel to manage), the meat density has been changed from 3.9 gU/cm^3 back to the test assembly density of 4.8 gU/cm^3.
Due to the Swedish legislation no reactors are allowed to operate if there is no final solution of the back-end of the fuel cycle. In order to have a solution ready before the present solution is running out at the end of the US FRRSNF Acceptance Program in May 2006 Studsvik Nuclear has decided to try to speed up the process. A further reason for this is that the R2 operating license is running out in 2004 and Studsvik Nuclear has to demonstrate in the renewed license application what it has done to handle the backend situation for the next license period, i. e. 10 years.
In parallel, Studsvik Nuclear is still careful to plan reactor activity on a long-term basis and to anticipate technical changes. In 2001, they contacted the French UMo group to offer the R2 facility as a test facility for a prototype UMo fuel assembly. The aim of the R2 team concerning the UMo fuel is to convert the reactor to a fuel type suitable for conditioning/passivisation, (that is, reprocessable).
Due to thermohydraulic, neutronic, operating, manufacturing as well as economic and fuel reliability considerations, the R2 management wouldn't accept a fuel loaded to 8 gU/cm^3 which is still the qualification aim of the French UMo Group. As the French group did not want to disperse its development effort, it was decided that the R2 management had to deal directly with CERCA for the manufacture of a 7 gU/cm^3 fuel (which was the preferred option by the reactor operator). Of course it was also stated that the R2 team and the French UMo Group could pool their resources in order to be more effective regarding the analysis**.**
This paper is divided into two parts. On one hand, CERCA will deal with the UMo R2 fuel manufacturing while on the other hand; Studsvik Nuclear will discuss on how the introduction of the lead test assembly is planned.

2. The R2 UMo fuel manufacturing

2.1 The main characteristics

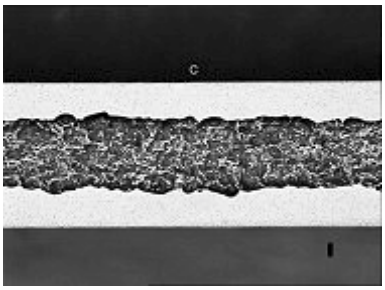
The main characteristics are summarised as follows. A comparison is made between standard U_3Si_2 fuel of the first and second generation and the UMo fuel. As explained previously, R2 has ordered U_3Si_2 standard fuel loaded to 4.8 gU/cm^3 instead of 3.9 gU/cm^3.

Table 1. Characteristics of CERCA fuel manufactured for R2

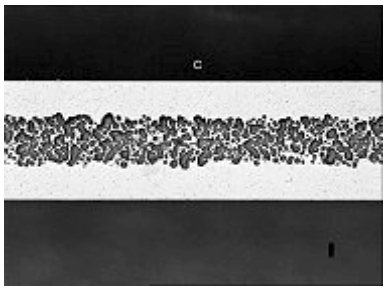
Main characteristics	**U_3Si_2 Fuel first generation**	**U_3Si_2 Fuel second generation**	**UMo Fuel**
Density g Ut / cm^3	3,9	4,8	7,00
Type of Powder	Comminuted	Comminuted	Atomised
U alloy	U_3Si_2	U_3Si_2	U- 7 wt% Mo
Meat length (cm)	59.7	59.7	59,7
Meat width (cm)	6.38	6.38	6.38
Meat thickness (cm)	0.76	0.76	0.51
Cladding thickness	0.38	0.38	0.38
Number of plates	18	18	19
^{235}U / Fuel assembly	410	489	510

It must be pointed out that there is a limited increase in the U 235 content due to the use of dense U Mo compound because in the same time the meat volume has decreased. The design of the fuel element (number of plates, geometry of the meat, geometry of the element, water gap) has been chosen to minimise the thermohydraulic changes, and the geometry of the fuel has been kept as close as possible to the current fuel element. Nevertheless, the geometry configuration of the plate is the same as for the other irradiation planned within the framework of the UMo fuel qualification program. For the same conventional cladding thickness of 0.38 mm, the meat thickness was 0,51 mm.
Here below is shown, for comparison purposes, micrographs of U_3Si_2 (4.8 gU/cm^3) and of UMo (7 gU/cm^3) R2 plates and a micrograph of a UMo plate from the French UMo group (8 g U/cm^3)

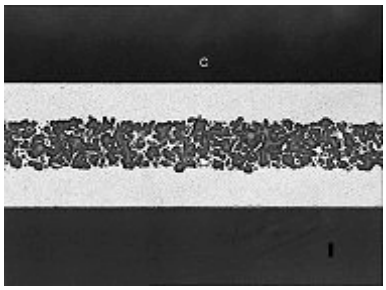
Those pictures emphasise the change in meat thickness, the impact of the density and the powder geometry.



R2 U_3Si_2 4.75g Ut/cm^3



R2 U-7Mo 7g Ut /cm^3



French group U-7Mo 8g/cm^3Ut

2.2 The R2 UMo Fuel manufacturing at CERCA

From the plate manufacturing point of view, the main recent developments were applied especially concerning the mixing procedure, the core preparation and the rolling parameters. Then, one can state that the homogeneity was quite good (under the limits of +/- 10 %) and stood easily within the specification limits of +/- 20 %.
Consequently, the mechanical deformation of the materials, during rolling, was smoother leading to a more regular cladding thickness. The minimum cladding thickness measured by destructive test was 0.29 mm (specification 0.25 mm) for the internal plates and 0.48 mm (specification 0.44 mm) for the external ones.
Furthermore, a special care was taken in order to prevent stray particles of UMo that is very sensitive with such high-density fuel.
For more details, see the paper "CERCA UMo Development - Status as of March 2003" by Ch. JAROUSSE from CERCA in Session 2 of this conference.
The manufactured plates were inspected and approved before assembling by Studsvik Nuclear at the fuel factory in Romans in the beginning of December 2002
CERCA strongly recommends its specification approach. As UMo fuel manufacture is completely different from the one for U_3Si_2 and UAlx, the specifications have been focused on the parameters that are sensitive for reactor safety. Those are, the homogeneity of the fuel meat, the bonding of the plate and the cladding thickness. Other parameters were carefully measured. The aim of this approach is to avoid adopting U_3Si_2 specification parameters that were based on UAlx specifications that are, is some cases, out of scope.
Afterwards, the irradiation results will also be used to validate a new UMo specification based on the fabricated plates used in the irradiation test, rather than being reusing old UAlx specifications.
From the assembly point of view, as the geometry was very close to the actual U_3Si_2 fuel, no manufacturing problems were noted. Nevertheless, some criticality adaptations were needed due to the fact that UMo assemblies were not described in the factory safety reports. Thanks to the engineering department of CERCA's shareholder, FRAMATOME, the safety case has been very quickly demonstrated, allowing CERCA to manufacture the fuel assembly before the end of 2002 as promised to R2.
Owing to CERCA’s background in the fuel manufacturing business, and especially in the UMo fuel development, its industrial approach combined with the support of FRAMATOME, CERCA has been able to manufacture **the first UMo fuel assembly in the world** within a short time period. It must be emphasised that this was possible also thanks to the French UMo group that constantly maintains the same aim and the same schedule since the beginning of the UMo fuel development.

3. Introduction of the UMo Test Assembly into the R2 reactor in Sweden

In order to be allowed to introduce a new fuel type into the reactor, several conditions have to be fulfilled before the safety authorities will give their consent. However, some constraints may be relaxed due to the very nature of a Lead Test Assembly, LTA, program. Some aspects of the new fuel type behaviour are already very well examined by test irradiation of mini fuel plates, full-scale plates and all other development efforts that have been the subject of many RERTR and RRFM conferences. However, the final proof of principle before the full scale introduction of reloads is the irradiation of an LTA assembly into the reactor to demonstrate the behaviour at site, such as the response to the operating and chemical conditions of the reactor.
To minimise the licensing efforts and to facilitate the LTA introduction, Studsvik Nuclear has chosen to try to apply as many features as possible from the earlier fuel experience, while at the same time adhering to the specific requirements of the new fuel type.
The fuel types that mainly have been in use in the R2 reactor are described in Table 2 together with the new UMo LTA assembly.

Table 2. Comparison of the main fuel types in the R2 reactor and the LTA UMo fuel assembly

Assembly type		LEU 400	LEU 490	HEU 250	UMo 510
U alloy		U_3Si_2	U_3Si_2	UALx	UMo
No. of fuel plates		18	18	19	19
Plate thickness*	mm	1.90/1.52	1.90/1.52	1.65/1.27	1.65/1.27
Cladding thickness*	mm	0.57/0.38	0.57/0.38	0.57/0.38	0.57/0.38
Fuel meat thickness	mm	0.76	0.76	0.51	0.51
Uranium density	gU/cm^3	3.90	4.80	0.73	7.00
Total Uranium weight	g	2025	2481	269	2580
Total U-235	g	399	490	250	510
Typical enrichment	wt.%	19.75	19.75	93.00	19.75

* Outer and inner plates, respectively

As can be seen from Table 2, there are in reality only two fuel types from a geometric point of view. The two U_3Si_2 LEU assembly types differs only in Uranium density, which means that their thermo-hydraulic properties are the same, while they differ from the HEU fuel type.
When Studsvik Nuclear started the conversion from HEU to LEU in the late 1980'ies the LEU design was at the time chosen based on the manufacturing capabilities for the new U_3Si_2 fuel and the need for as small changes as possible of the fuel geometry. The outer fuel dimensions where kept the same and so was the water channel dimensions. Due to the need to increase the U-235 content of the meat and thus the total Uranium content even more, notwithstanding the five-folded increase in the Uranium density, the number of fuel plates was reduced to 18. This resulted in a 5 % reduction of the assembly flow area and a resulting decrease in total coolant flow but an increase in flow velocity. The safety implications of these changes were a decreased margin to flow instability, an increase in power peaking factors. The safety consequences were an increased scram level for minimum core flow.
The same considerations were at hand when contemplating an UMo LTA introduction. Even more so as the experience of UMo fuel is smaller compared to the situation when introducing U_3Si_2. Thanks to the large efforts already made in the RERTR program, by the French UMo group and by all other stake holders Studsvik Nuclear considers the UMo fuel development program being mature enough to motivate a full LTA insertion. However, the UMUS failure demonstrates that problems can occur and that a cautious approach is preferable, thus the somewhat lower fuel density.
The benefit of this decision was that the UMo LTA could be designed with exactly the same geometrical dimensions as the old HEU 19-plate fuel, as is shown in Table 2. This means a lower surface heat flux and a better cooling capacity than for the LEU 490 assembly. Thus, the T/H safety case for UMo is already demonstrated.
The UMo assembly has a slightly higher U content than the LEU 490 assembly. This is due to the fact that Mo has a somewhat high neutron absorption cross section, higher than the one for Si. With this

increase in U content we match very well the unirradiated k-inf of the two fuel types, see Figure 2. There is an almost perfect match between the UMo and the LEU 490 assemblies with the UMo fuel slightly lower. The fuel types are undermoderated, but the LEU-490 assembly has a 5% lower flow area and has thus a higher potential for a reactivity increase if the assembly geometry is changed. This means that also the criticality analysis is contained by the design.

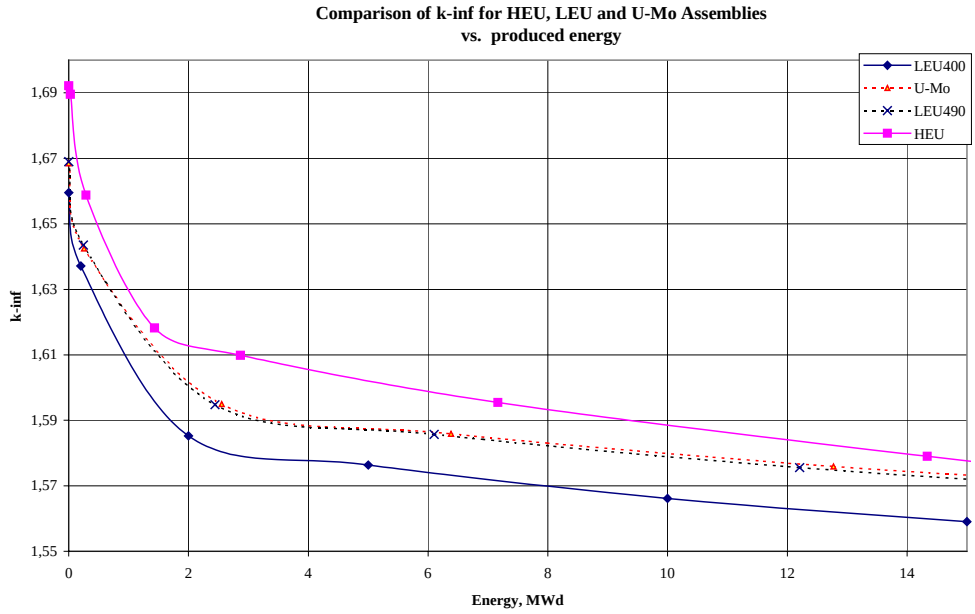


Figure 2. K-inf at low burnup for the R2 fuel types

By maintaining the same fuel cladding as for the U_3Si_2 fuel, there is no reason to expect any different corrosion behaviour with the UMo LTA assembly compared to the well-known and proven LEU experience.

In summary, all aspects of the LTA assembly are already proven and demonstrated in the R2 reactor except for the meat type. The UMo LTA is thus compatible with the other fuel of the core, it is contained by the existing safety analysis and the existing experience of UMo from irradiation experiments demonstrates that the introduction of an LTA UMo assembly into the R2 reactor can be safely done.

The LTA assembly will although be cautious introduced into the reactor in low power positions in the first cycles and then slowly be moved to higher demanding positions.

It is expected that the assembly will have reached its final burnup of up to 80 % after about 17 operating 3-week periods, that is one and a half-year after the introduction into the reactor which is planned to take place during the spring.

If everything goes as planned, the UMo fuel would then be qualified for reload insertions in the beginning of 2005.

4. References

[1] NUREG 1313 Safety Evaluation Report related to the Evaluation of Low-Enriched Uranium Silicide-Aluminium Dispersion fuel for Use in Non-Power Reactors

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THE UMUS FAILURE : FACTS AND ANALYSES

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ABSTRACT

Four Plates were irradiated in HFR Petten at a maximum surface heat fluxes of 140 and 250 W.cm^{-2} for enrichments of 20% and 35%, respectively. The experiment was stopped after only two cycles after failure of the two 35%-enriched plates. Analysis of postirradiation examinations and the evaluation of the experiment operating conditions have established high cladding temperature as the root-cause of the failure.

1. Introduction

Four UMo full-sized plates have been irradiated in the High Flux Reactor, in a special irradiation device. The experiment, named UMUS, was stopped at the end of the 2nd cycle, after 48.4 full-power days (FPD), due to the failure of the 35%-enriched plates and fission-product release from one of them [1], [2].

This paper summarizes the main facts of this experiment and presents the analyses made to explain the failure of the 35%-enriched plates.

2. Main facts relative to the UMUS experiment

2.1. Manufacturing aspects

The manufacturing of the UMo plates at CERCA was described two years ago [3]. Two ^{235}U enrichments were tested: 19.75% for Low Enriched Uranium plates (LEU) plates, and 35% for Medium Enriched Uranium (MEU) plates. It is important to note that the UMo powders were produced by grinding. This process leads to large surface-to-volume ratio, due to a high ratio of fines (40% of particles with diameter less than 40 μm), and high content of oxygen in the UMo powder, in range of 0.8-1% in weight. In the plates, the volume fraction of fuel particles was 50%, so that the uranium densities in the fuel meat were greater than 8.0 g cm^{-3} for the two types of UMo alloys used (7 and 9% Mo mass content). The results of fuel-plate inspection (based on micrographs, X-ray, blister test…) indicated that the plates conformed to the specifications.

Plate	Reference	% Mo in alloy (wt %)	Enrichment (%^{235}U)	Mass of total U (g)	Mass of ^{235}U (g)	Meat density (gU.cm^{-3})	Mean plate thickness (mm)	Porosity (%)
#1	U7MQ2005	7.6	19.65	150.0	29.5	8.2	1.34	13.5
#2	U7MQ3505	6.6	34.54	150.3	51.9	8.1	1.35	14.7
#3	U9MQ2055	8.7	19.66	145.3	28.6	8.0	1.34	11.5
#4	U9MQ3552	9.6	34.86	144.4	50.4	8.2	1.34	10.8

Table 1 : Characteristics of UMo plates

The main characteristics are given in Table 1, and Fig. 1 shows a section of fuel meat (longitudinal direction view) in an unirradiated plate.

Fig. 1 : Metallographic longitudinal section of as-fabricated UMo sibling fuel plate (U7MQ3504)

The mean values were 8.1±0.1g.cm^{-3} for uranium density and 0.51 mm for meat thickness, with local variations of ±20% for a sampling area of 1 cm^2.

2.2. Irradiation requirements

The objective of this experiment was to test two enrichments of UMo fuel up to a ^{235}U burn-up of at least 50%, with requirements on maximum temperature limits during irradiation: 150°C for the cladding temperature, and 250°C for the fuel meat. The four plates were positioned in a special device and irradiated in HFR core position D2.

2.3. Postirradiation examinations

A summary of the postirradiation examinations performed is given in the Table 2 below.

	Plate #1 UMo7 - LEU	Plate #2 UMo7 - MEU	Plate #3 UMo9 - LEU	Plate #4 UMo9 - MEU
Non destructive examinations [4]				
Visual inspection	X	X	X	X
Plate thickness measurements	X	X	X	
Gamma spectrometry	X	X	X	X
Destructive examinations [5]				
Optical microscopy	X	X		
SEM	X	X		
EPMA		X		
Hardness measurements		X		

Table 2 : Postirradiation examinations summary

The most significant results were already given in a previous paper [1]:

- MEU plates surfaces show large dark areas on their two sides, whereas LEU plates

exhibit much-smaller dark spots only on their sides facing the MEU plates (Fig. 2a).

- Fission-product release from MEU plate #4 is due to a 17 mm length crack in the centre of the clad and at the maximum flux plane of the plate. Its position seems located at the top of a large bulge (Fig. 2b).
- Plate transverse thickness measurements at the maximum heat flux plane (MFP) show a slight increase (20-25 µm) for LEU plates, and a more significant increase for MEU plate #2, up to 100 µm at mid-width (Fig. 3),
- Two bulges can be noticed on plate #1. They are located at the same position as the two dark spots observed the plate surface (Fig. 2a).
- As for plate #1, plate #2 shows a smaller increase on the cooling entrance side than at the maximum flux plane. In contrast, the longitudinal profile of plate #3 seems very flat and shows a uniform increase of about 20 µm.
- Rather porous and multi-layered thick cladding oxide layer, up to 85 µm, is observed on plate #2, while for plate #1 the layer is dense and rather homogenous and does not exceed 25 µm
- Other optical observations for plates #1 and #2 (Fig. 4) are summarized in Table 3:

	Plate #1 (UMo7/LEU)	Plate #2 (UMo7/MEU)
Meat Aluminium remaining	Few percent	Less than 1 %
Meat/cladding interface	Good cohesion	Gap on both side(1)
UMo/Al interaction layer : - Meat centre - Meat/cladding interface	4-6 µm	 10-12 µm 4-6 µm
Brittle rupture/cracks	Several in the meat	Several in the meat

(1) Measured values are not representative of the in-reactor plates values due to the samples preparation

Table 3 : Most significant optical metallographic results at MFP

3. Post experiment analyses of UMUS experiment

3.1. Beginning of Life (BOL) conditions

The thermal powers of the four plates in the actual HFR core loading configuration have been evaluated by using the TRIPOLI code [6]. The results are presented in the Table 4. Thermal powers of the LEU plates are close to those acceptable for standard HFR element whereas they are roughly 50% higher for the MEU plates.

Plate	Enrichment ($\%^{235}U$)	BOL conditions of UMUS experiment		
		Thermal power (kW)	Surface heat flux ($W.cm^{-2}$)	Volumetric power ($kW.cm^{-3}$)
1	19.65	94.5	174	7.3
2	34.54	134.4	248	10.8
3	19.66	76.0	140	6.1
4	34.86	131.5	243	10.4

Table 4 : Beginning of life conditions of the UMUS experiment

The coolant velocity was also evaluated using the "as-built" device geometry and the FLICA code. The computed value was $8.0\pm0.1 m.s^{-1}$ [7], but some uncertainty remains on this value, since for the same pressure drop and the same width of channel, the coolant velocity of the HFR standard fuel element is only $6,5\ m.s^{-1}$.

Fig. 5 gives the longitudinal cladding surface temperatures obtained for each plate side, with these input "best-estimated" data. The maximum temperatures are 110°C for the MEU plates, and 90°C for

the LEU plates. Note that with a velocity of 6,5 $m.s^{-1}$, the maximum value for the UMUS MEU plates increases to 125°C.

3.2. End of Life (EOL) conditions

3.2.1. Surface temperatures at the coolant / plate interface

Due to the failure of the plate #4, the experiment was stopped after 48.4 FPD. The 15% decrease of the power owing to burnup causes the slight decrease of the surface temperatures (less than 7°C for the MEU plates and 4°C for the LEU plates).

3.2.2. Clad temperatures deduced from oxide thickness measurements

Due to the formation of a boehmite layer on the cladding surface, the clad temperatures were considerably increased during irradiation. Indeed, for a oxide thickness of 85 µm, as measured on plate #2, and a thermal conductivity value of 2.25 $W.m^{-1}.K^{-1}$ of the boehmite [8], [9], the clad temperature is as high as 200°C at the MFP. Moreover, taking into account that the value used for the thermal conductivity is largely overestimated (probably more than 15%) for the porous and multi-layered oxide observed on the MEU plate #2, and that the maximum oxide thickness is higher (~100 µm) than the highest value measured on one PIE sample, it can be assumed that the maximum level of the MEU clad temperature at the end of the UMUS experiment was definitely above 220°C.

3.2.3. Clad temperatures deduced from micro hardness values

It is known that the strength of aluminium generally increases with accumulating neutron dose, due to various irradiation effects, such as generation of Si from (n,γ) reaction with Al, irradiation precipitation of small Mg_2Si particles, and accumulation of dislocations. However, all these processes are temperature dependent, and at temperatures above ½ the absolute melting point (180-190°C for Al alloys), thermally activated diffusion causes drastic softening of the alloy. It is thus possible to determine whether a particular fuel element cladding operated above a certain temperature by measuring its mechanical properties after irradiation. This is most conveniently done by micro hardness tests.

Fig. 6 presents the change in cladding hardness as a function of fast fluence. The ANL data from miniplates irradiated in ORR at temperatures below 100°C show hardening for both cold worked and annealed 6061 Al cladding, as do the HFIR data at 160°C. A German experiment [10] of miniplates clad with 1100 Al showed no hardness change after irradiation at 185°C. In contrast, the cladding on UMUS plate #2 (U7MQ3505) appears to have drastically softened.

Since no preirradiation data are available on the UMUS cladding, a value from CERCA plates fabricated for the ORR-U_3Si_2 demonstration irradiation was used. The hardness of the UMUS cladding after irradiation was approximately equal to that of fully annealed 1100 Al, indicating that its temperature was sufficiently high to anneal out irradiation hardening as well as any solution hardening and cold work existing after fabrication. As mentioned before, this annealing happens at temperatures higher than ~ ½ the melting temperature (~190ºC); however, the time required to soften is very temperature dependent. This is shown for aging of 6061 Al and for annealing of irradiation hardening of 6063 Al in Figs. 7 and 8.

Considering the relatively short UMUS irradiation time of 48.4 FPD, the cladding temperature must have been substantially higher than 190ºC during the later part of the irradiation for its hardness to reach the practically fully annealed value. Taking all these, albeit, indirect observations into account, it seems reasonable to estimate a maximum cladding temperature of ~250ºC for the 35%-enriched UMUS plates.

4. Discussion

4.1. Oxidation and spalling of UMUS plates

As considered in a previous section, the rise in cladding (and fuel) temperature during irradiation is due to the formation of a hydrated aluminum layer ($Al_2O_3 \cdot H_2O$) on the cladding surface. This "boehmite" layer has a much-lower thermal conductivity than Al, and, depending on its thickness and the surface heat flux, may have a significant ΔT across it. The rate of formation of boehmite was

recognized as a fuel performance issue for the HFIR and ATR reactors and more recently for the ANS project. Extensive ex-reactor studies were performed, resulting in the so-called Griess and ANS correlations, respectively. Both correlations contain a strong dependence on the boehmite-coolant interface temperature and are valid for coolant with pH of 5 or less. The more recent ANS correlation includes weak heat flux dependence as well. Both studies found that the rate of formation increased drastically (by a factor of 3) at a pH over 5, resulting in unacceptably thick boehmite layers for fuel in high-power reactors. Consequently, these reactors operate with acidified cooling water controlled at maximum pH of <5. In addition, the HFIR study derived a thermal conductivity for the ex-reactor boehmite of 2.25 ± 15% $W.m^{-1}K^{-1}$.

Boehmite thicknesses measured on metallographic samples taken from peak thickness locations of UMUS plates #1 and #2 are, respectively, 25 μm and 85 μm. The irregular plate-thickness profile shown in Fig. 3 for UMUS plate #2 indicates that the maximum boehmite thickness may be as much as 100 μm and that it has been spalling off. This is supported by the dark (rough) surface condition of MEU plates #2 and #4 (see Fig. 2). The spalling or flaking of the boehmite was also observed on HFIR fuel plates where the ΔT across the boehmite layer had exceeded ~120ºC. Spalling would therefore first occur at a position of maximum heat flux, which is consistent with the UMUS PIE observations. The spalling also indicates that the maximum boehmite thickness was reached sometime before the end of the irradiation. Insufficient data are available to determine the time at which spalling began. However, the extensive darkening (roughness) of the cladding, shown in Fig. 2, indicates that spalling began well before the end of the irradiation.

Note that if the minimum ΔT of 120ºC for spalling is assumed, the minimum cladding temperature during cycle 2 would be 110+120=230ºC from perhaps the end of cycle 1 on. Based on these arguments and the hardness data, it seems probable that the cladding temperature of the MEU plates was in the range of 230-250ºC from the end of cycle 1 on.

4.2. Proposed failure mechanism

What then, may be the consequences of operating the UMUS test plates at such high cladding temperatures? As shown in Fig. 9, the immediate effect is a drastic decrease in mechanical strength of the cladding. However, this in itself is not likely to result in cladding rupture unless there was some locally severe cladding thinning resulting in locally high cladding stress. Although these highly loaded experimental plates do have a rather irregular meat thickness (see Fig.1), the fabrication records do not show that these irregularities are of sufficient magnitude to expect them to be the primary cause of cladding failure. Other possible sources of cladding stress include high fuel swelling rate and development of large fission gas bubbles. Although the plate with the cladding rupture (#4) was not sectioned, metallography of sibling plate (#2) showed no indication of either.

However, as discussed in Section 3.2.3, the metallography of this sibling proved that the MEU plates operated at rather high temperatures. In addition to the aforementioned low microhardness of the cladding, the matrix aluminum has been almost completely consumed by UMo-Al inter diffusion as a result of high meat temperature. The meat in the MEU plates now consists of a dispersion of residual UMo particles in a matrix of a very hard and brittle UMo-Al compound (DPH ≈ 750) rather than UMo in ductile aluminum that was ultimately bonded to the cladding. Another significant observation is that the cladding and meat have separated. We believe that this separation set up the conditions for cladding rupture. This separation (explained in more detail below) is due to high thermal stresses occurring most likely during shutdown or startup of the reactor. This cladding separation could have occurred after cycle 1 in the failed plate #4 and at the end of cycle 2 in sibling plate #2.

If indeed cladding separation in plate #4 occurred after cycle 1, fission gas would have slowly accumulated in the meat-cladding gap by diffusion from the ever-hotter meat, eventually leading to rupture of the mechanically weak cladding. Thermal fluctuation caused by boehmite spalling may have contributed as well. Although the meat-cladding gap in sibling plate #2 appears to be rather wide on the metallographic samples, comparing the sum of measured meat, cladding, and boehmite thicknesses with the plate thickness measurements leads to the conclusion that the gap was very narrow prior to cutting of the metallographic samples, hence the conclusion that the separation in plate #2 occurred at the end of cycle 2. Presumably this plate would have ruptured as well had the irradiation continued.

The source of the stress that leads to meat-cladding separation could stem from thermal differential expansions between the meat and the clad or thermal gradients along the fuel plate. This latest is shown schematically, and simplified, in Fig. 10. The entire fuel-containing part of the plate is surrounded by a "rail" of aluminum that is maintained at coolant inlet or outlet temperature (on average ~65ºC), while the cladding over the fuel at beginning of life (BOL) is moderately hotter; however, as we have argued in section 3.2.2, it increases significantly as irradiation proceeds. The resulting thermal expansion difference between the cladding and surrounding rail (this may include some of the cooler fuel section at the periphery as well) would result in very high compressive lateral stress in the cladding if no deformation occurred. As the temperature rises gradually with the growth of the boehmite layer, these thermal expansion stresses are readily relieved by creep of the Al—more readily when the temperature rises since the creep rate increases exponentially with temperature.

The cladding stress histories for an MEU and an LEU plate are shown schematically in Fig. 11 for the first cycle of irradiation. During the cycle, the compressive stress is maintained at a very low level by creep relaxation. At shutdown, the cladding rapidly cools, reversing the thermal expansion stress to tension. There is now no time for creep, the stress is relaxed by yielding of the softened cladding, and the stress is maintained at approximately the low yield point value.

Incidentally, a similar process occurs in the meat. If, sometime during the cycle, the meat has converted to primarily fuel and interaction phase, the thermal stresses (and any swelling-induced stresses) are relaxed by, in this case, irradiation-enhanced creep. However, at shutdown no yielding is possible in this hard meat microstructure; instead this brittle material may respond to the tensile stress by cracking. Cracks are indeed observed in the metallographic sections.

Upon restart of the reactor for cycle 2, the stress again reverses to compression and the possibility of local buckling now exists along the thin Al cladding sheath. Exactly how the fracture between the brittle meat and ductile cladding is initiated is difficult to predict. It may occur either during shutdown or startup at a favorable location such as a large variation in meat thickness, the location of a meat shutdown crack, or a combination of these, possibly aggravated by oxide contamination at the meat surface (the fuel was reported to contain relatively large amounts of oxygen). The fact that U-9Mo plate #4 presumably experienced this separation earlier than U-7Mo plate #2 may be due to the somewhat higher loading, and perhaps to a more-irregular meat thickness, of plate #4.

The main point, however, is that high thermal stresses developed in the MEU plates compared with the LEU plates (see Fig. 11) as a result of much higher cladding temperatures in the former.

5. Conclusions

The root-causes of cladding rupture of one of the two MEU plates in the UMUS experiment was high operating temperature.

The high cladding temperature was caused by the formation of a thick, low-thermal-conductivity boehmite layer on the cladding surface. Under these high-temperature conditions, the aluminum in the fuel meat rapidly converted to a brittle aluminide and the mechanical strength of the cladding decreased. High stresses in the fuel plates due to a large thermal expansion difference along the plate could have caused cracking at the meat-cladding interface initiating cladding rupture.

This thick boehmite thickness is linked to the cladding surface temperature and the chemistry of the coolant water. Aluminum-clad, highly loaded fuel, operating at temperatures such as existed in the MEU UMUS test requires careful evaluation of boehmite formation based on the prevailing coolant conditions for the reactor in question to prevent uncontrolled boehmite buildup resulting in excessively high cladding temperatures.

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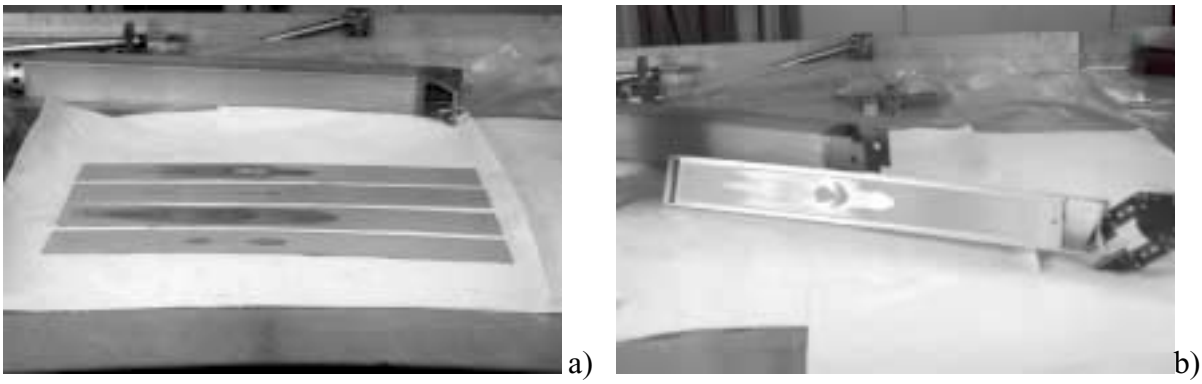
7. Figures



a) b)

Fig. 2 : UMUS plates (a) and Failed Plate n°4 (b) visual inspection

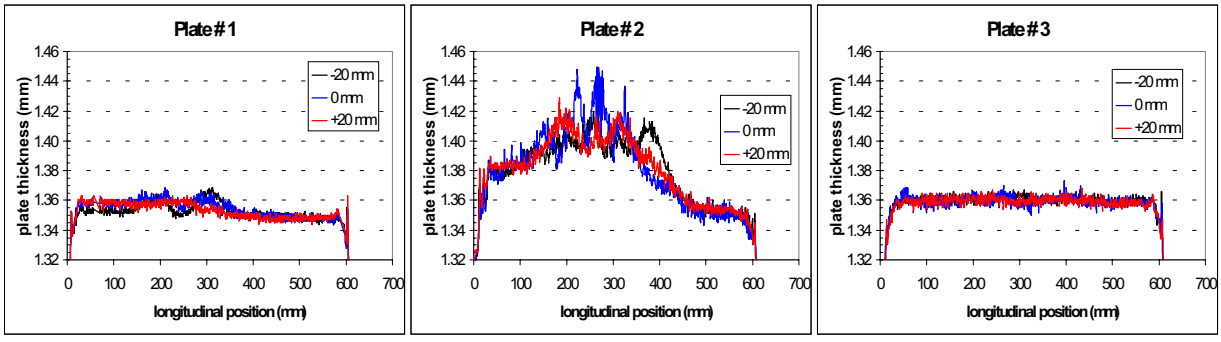


Fig. 3 : Longitudinal plate thickness measurements

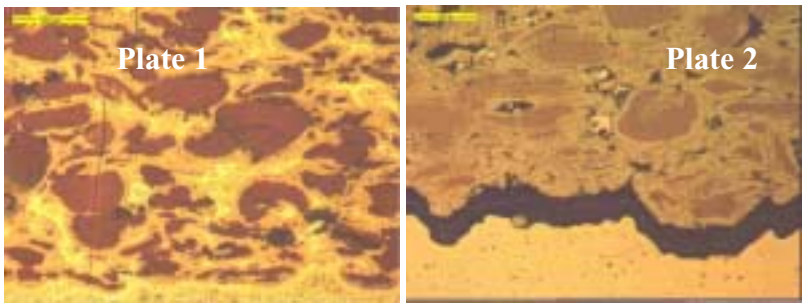


Fig.4 : Plate 1 and 2 meat optical microscopy

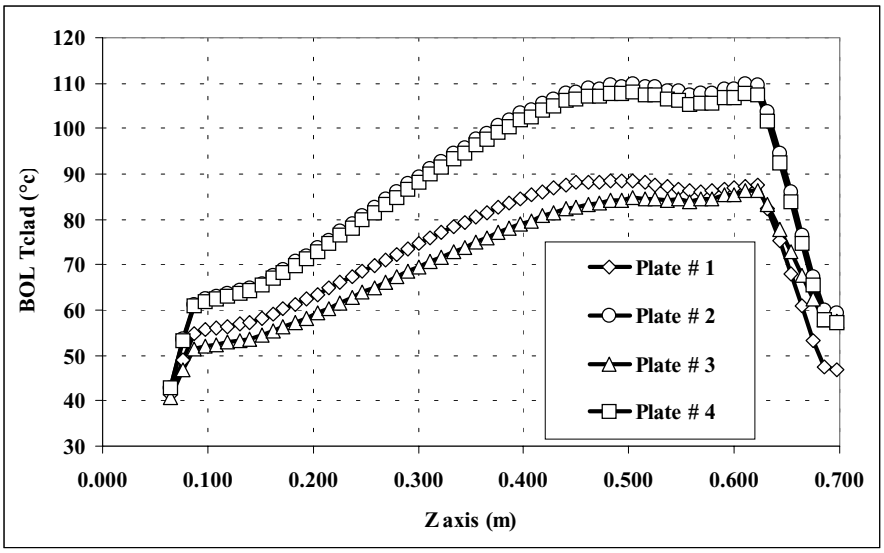


Fig. 5 : UMUS plates longitudinal temperature profiles.

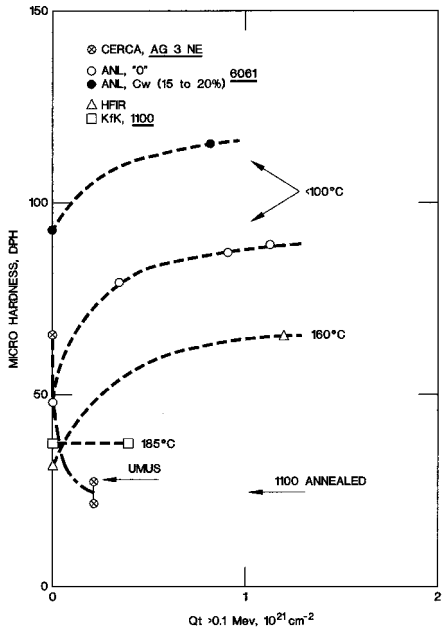


Fig. 6 : Microhardness of various Al claddings irradiated as a function of fast fluence

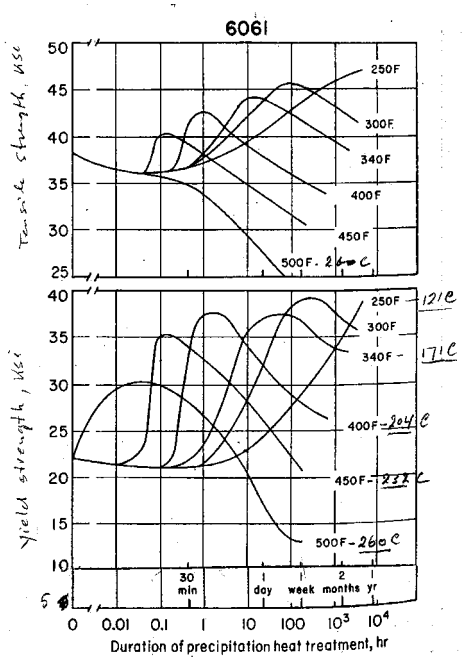


Fig. 7 : Room-temperature strength of 6061 Al after aging treatment.

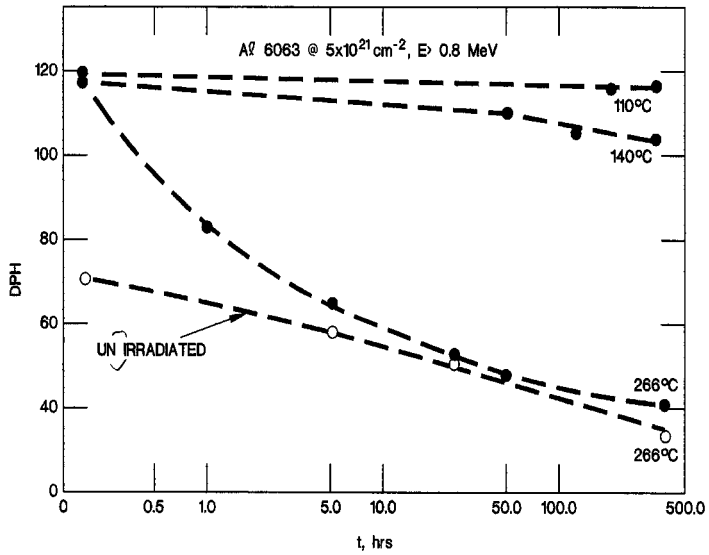


Fig. 8 : Annealing of irradiation-induced hardness for 6063 Al

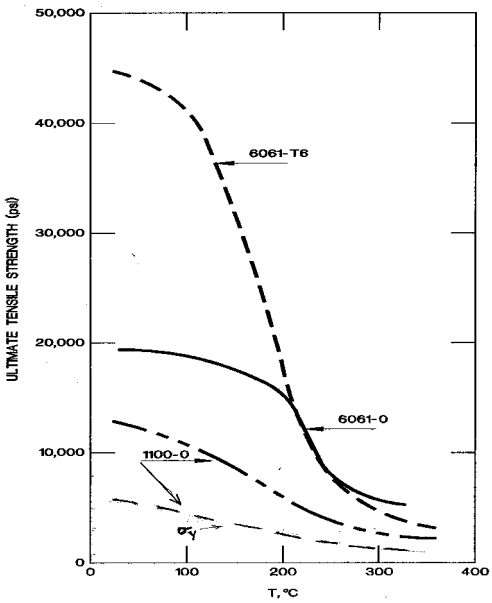


Fig 9 : Tensile strength of Al alloys as a function of temperature.

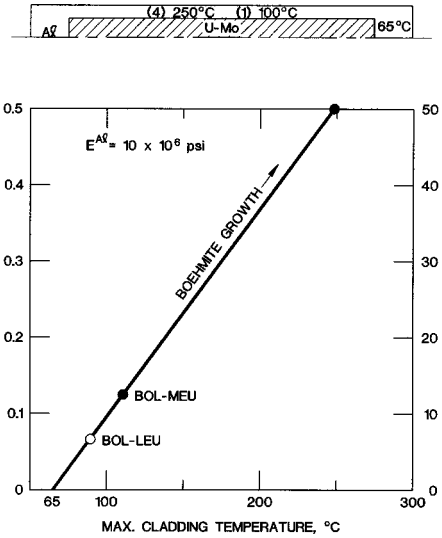


Fig. 10 : Thermal strains between fueled cladding and unfueled part of the plate

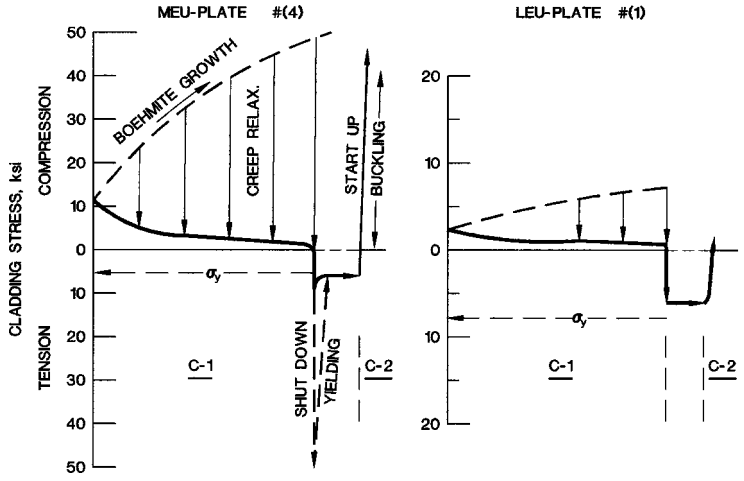


Fig. 11 : Stress history in hottest sections of MEU and LEU plates up to the start of the 2nd cycle

Session 3

Reactor operation, fuel safety and core conversion

ACCELERATION OF THE RERTR PROGRAM: SCOPE, STATUS, AND PLANS

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ABSTRACT

The terrorist threats demonstrated by the September 11, 2001 attacks, the promising results of the RERTR program in testing monolithic U-Mo fuel, and the advances of the Russian RERTR program in developing new LEU fuels for Russian-designed research reactors have led to a reevaluation of the speed and focus with which RERTR program goals should be pursued.
In their May 2002 Summit in Moscow, President Bush and President Putin agreed to establish a joint expert group to work out proposals to reduce Russian inventories of HEU and plutonium. During currently ongoing negotiations between the United States and Russia, one important activity identified by the expert group is to "work on accelerated development of LEU fuel for both Soviet-designed and United States-designed research reactors." This paper addresses the scope, status and plans for activities of the RERTR program consistent with this goal.

1. Background

Since 1978, the United States has pursued a policy aimed at reducing the threats posed by civil commerce in HEU. Since HEU-fueled research reactors provide essential services to society, it was recognized that new and advanced LEU fuel technology would be required to replace existing HEU fuels and to accomplish our nonproliferation goals while maintaining the flow of products and services provided by these reactors. Later, in 1992, the United States amended U.S. law to clarify the civil HEU minimization policy with regards to exports of HEU from the United States, and in 1996 implemented the Foreign Research Reactor Spent Nuclear Fuel Acceptance (FRRSNFA) program to ship certain spent HEU and LEU fuel back to the United States for disposition.
To date, nearly a quarter century of implementation of the civil HEU minimization policy has resulted in more than three metric tons of avoided commerce in civil HEU. This amount of material is sufficient to make well over 100 nuclear weapons. Moreover, the FRRSNFA program has resulted in acceptance of over 700 kg of HEU in spent fuel from twenty-seven countries to the United States for secure disposition. In addition, eleven of the seventeen U.S. university research reactors for which a suitable LEU fuel is available have been converted to date. This positive long-term effect of our policies has been achieved both through direct avoidance of the use of HEU through the conversion of existing research reactors to use LEU fuel, and by the avoidance of HEU fuel in new reactor designs and construction.
In the post September 11 world, it is no longer debatable whether it is useful to pursue an HEU minimization policy. Rather, the concern has shifted to how quickly we can proceed toward the goal of reducing the civil use of HEU to the minimum level possible.

There should be no doubt in anyone's mind that a significant quantity of HEU in the hands of a terrorist group would constitute an unacceptable threat to peace-loving nations. The disaster that might ensue following the theft or diversion of such material would be catastrophic, changing the course of human history and fundamentally altering international security relationships. Of course that is a view as a historian might express it, after the fact. The personal loss and tragedy suffered by the victims of such an attack would be immeasurable, and the collective grief suffered could never be adequately captured by historical analysis.

It is with these sober thoughts in mind that we move forward with the International RERTR Program. Furthermore, this meeting is convened following reports of recent and brutal terrorist attacks such as those carried out in Bali, Moscow, and elsewhere. Many nations have now joined together with resolve to fight a war against terrorism. We should all reflect on the importance and high purpose of what our joint efforts aim to accomplish in minimizing the civil use of HEU, on the role of the RERTR program in this endeavor, and on the immediacy of this mission.

Research reactors produce products that will continue to save lives and improve the quality of life well into the future. Life-saving medical isotopes and procedures, agricultural products, scientific and engineering endeavors all rely on the international research reactor community. We must press forward with renewed purpose to continue providing these essential products, but also to work diligently toward the goal of minimizing the civil use of HEU in the process. Although we may encounter practical technical and political difficulties when implementing reactor conversion plans, we must press forward, keeping in mind the importance of our goals - understanding that our efforts will improve the lives and security of the entire international community.

It was in response to similar concerns, that Presidents Bush and Putin signed a Joint Declaration at the May 2002 summit in Moscow directing technical experts to find ways to accelerate elimination of weapons-useable fissile materials that have been declared excess to defense needs. One outcome of the resulting talks by technical experts was an agreement that the United States and Russia would accelerate our respective RERTR programs by quickening the pace of development of new LEU fuels to replace HEU fuels in both U.S. and Russian-designed research reactors.

2. Scope, Status, and Plans

As part of these expert talks, the United States has agreed to accelerate ongoing efforts to develop new LEU fuel that can be used to convert the five civil HEU-fueled U.S. research reactors with the highest power. It is our goal to develop and qualify new LEU fuel technologies and to complete the conversion documentation within the next decade, so that these reactors can be converted as soon as possible. Similarly, the MinAtom-sponsored Russian RERTR program has also agreed to accelerate its efforts in cooperation with the United States. The Russian RERTR program will focus on developing fuel that could be used to convert twenty-one Russian-designed research reactors engaged in civil programs. Eleven of these reactors are located in Russia and ten are located outside Russia (Table I). To date, the Russian RERTR program has made substantial progress toward this goal and we look forward to accelerating our cooperation in this joint endeavor.

Table I. US-designed and Russian-designed HEU Research Reactors

US Reactors			RF Reactors in Russia			RF reactors outside Russia		
Name	**MW**	**Location**	**Name**	**MW**	**Location**	**Name**	**MW**	**Location**
MITR	5	Massachusetts	IRT-MEPhI	2.5	Moscow	WWR-K	6	Kazakhstan
MURR	10	Missouri	IR-8	8	Moscow	WWR-M	10	Ukraine
NBSR	20	Maryland	IRT-T	6	Tomsk	WWR-CM	8	Uzbekistan
HIFR	100	Tennessee	WWR-TS	15	Obninsk	LWR-15	10	Czech Rep.
ATR	250	Idaho	WWR-M	18	St. Petersburg	VR-1	0	Czech Rep.
			IVV-2M	15	Sverdlovsk	IRT-Sofia	0.2	Bulgaria
			MIR-M1	100	Dmitrovgrad	WWR-SZM	10	Hungary
			SM-3	100	Dmitrovgrad	MARIA	17	Poland
			RBT-6	6	Dmitrovgrad	IRT-DPRK	8	N. Korea
			RBT-10/2	10	Dmitrovgrad	IVV-7	10	Libya
			PIK	100	St. Petersburg			

The U.S. and Russian RERTR program teams will be meeting in the near future to plan the detailed path forward to meet our mutual objectives. Those plans will cover acceleration of ongoing work on development of new LEU fuels, to be manufactured in Russia, to supply Russian-designed reactors both in foreign countries and in Russia. We will also discuss conversion feasibility studies and other technical assistance required for conversion of Russian-designed reactors. We have accelerated our efforts to engage Russian-designed reactor owners and operators and to introduce them to the RERTR program. We hope to interest them in partnering with us to help increase international security through conversion of their HEU-fueled reactors to use LEU fuel.

These efforts, taken in combination with the ongoing cooperation between the United States and Russia to implement a broad program to repatriate HEU fuel from Russian-designed research reactors operating outside Russia, promises to make a substantial reduction in HEU inventories held at Russian-designed research reactors worldwide. The Russian Research Reactor Fuel Return program is the complement to the U.S. FRRSNFA program.

During the past few years, much of the effort of the US RERTR program has been dedicated to the development of dispersion fuels based on particles containing a uranium-molybdenum alloy, with molybdenum weight between 7% and 10%. The goal is to qualify this fuel type with a uranium density of 6-7 g/cm^3 before the May 2006 expiration date of the FRRSNFA program and to provide an alternative LEU fuel for reactors currently using or planning to use LEU silicide fuel

This past year, a new monolithic LEU fuel concept developed by Argonne National Laboratory surpassed all performance expectations in post irradiation examination. If monolithic LEU fuel is successfully qualified it could raise the maximum uranium density in research reactor fuel as high as 16 g/cm^3. At such high densities, nearly every research reactor in the world would be able to convert. Moreover, there has been recent progress and success at the facilities and institutes of our international colleagues. Some of this progress will be reviewed during the course of this conference.

As part of our efforts to accelerate the RERTR program, we are currently preparing to begin conversion feasibility studies on two U.S. Government-owned HEU-fueled reactors: the NBSR and the ATR. A conversion feasibility study has previously been completed on the HFIR reactor at Oak Ridge. Moreover, two university-owned reactors, the MITR and MURR, which will require monolithic fuel in order to convert, will be studied in the near future. Once the LEU U-Mo monolithic and dispersion fuel technologies are developed, suitable LEU fuels will be available for conversion of all civil research reactors in the United States. It is our goal to convert all U.S. civil research reactors to use LEU fuel consistent with the U.S. civil HEU minimization policy.

3. Conclusion

It is through the collective technical efforts of all the international RERTR program participants that a significant and positive difference is currently being made in global security. Without the continuing support and efforts made by the international research reactor community, directed toward the goals of providing exceptional products and services while cooperating to reduce the civil use of HEU, it would not be possible to move decisively toward our mutually held objectives. To date, substantial progress has been made worldwide toward conversion of civil research reactors to use LEU fuel. However, in view of new and increased dangers, now it is time to focus and move forward rapidly with renewed resolve to further reduce and minimize the risk.

PERFORMANCE OF THE WWR-M RESEARCH REACTOR IN UKRAINE WITH A MIXED-FUEL CORE

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ABSTRACT

The in-core fuel management optimization problem was studied for the WWR-M research reactor in Ukraine. Two available types of fuel assemblies were considered: WWR-M2 with 36% enrichment and WWR-M5 with 90% enrichment. It has been demonstrated that a mixed core consisting of WWR-M2 and WWR-M5 fuel provides higher neutron flux and even less fuel expenditures in comparison with the core consisting of only WWR-M2 fuel. Utilization of mixed-fuel is especially effective for high "neutron prices". The optimum core configuration, as well as the number and location of fuel assemblies of different types in the core strongly depend on "neutron price". With growth of the "neutron price", the total number of fuel assemblies in the core should be decreased, while the number of WWR-M5 fuel assemblies should be increased. For a not too low "neutron price", WWR-M5 fuel assemblies should be located near the irradiation channels to make power density in these areas, limited by the maximum allowed temperature of fuel surface, as high as possible, thus increasing neutron flux.

1. Introduction

The WWR-M research reactor in Ukraine is used for various purposes including neutron physics and materials research, radioisotope production and neutron transmutation doping of silicon. The core of the reactor can be built using two available types of fuel assemblies: WWR-M2 with 36% enrichment and WWR-M5 with 90% enrichment, containing 32 and 66 grams of ^{235}U, respectively. WWR-M5 fuel assemblies provide more intensive heat transfer and can be used under 1.5 more power density than WWR-M2. However, WWR-M5 discharge burnup has to be less that 50% of initial mass of ^{235}U, while WWR-M2 fuel assemblies have operated successfully up to 80% burnup.

The total power of the WWR-M reactor in Ukraine is restricted by 10 MW. The number of channels for experiments, isotope production and neutron transmutation doping of silicon is not too large. Moreover, the current design and requirements to the control rod system do not allow the channels to be near the center of the core. Thus, a mixed core consisting of WWR-M2 and WWR-M5 fuel can provide higher neutron flux and, hence, higher performance of the reactor. This problem is studied using the PORT code developed to optimize core loading patterns and improve fuel management for the WWR-M research reactor [1].

2. The Optimization Problem

The optimization problem is to maximize $p_n\Phi_{th}$-C_F, where p_n is the "neutron price", Φ_{th} is the average thermal neutron flux in the large irradiation channels in the core, and C_F is the feed fuel cost. The optimal core loading pattern, fuel types used, number of fuel assemblies of each type and their discharged burnups are determined under the constraints on the maximum allowed burnup for each fuel type and temperature of fuel surface.

The feed fuel cost can be written as follows: $C_F = \sum_{i=1}^{M} \frac{c_i P_i}{\gamma_i m_i B_i^D}$, where γ_i, m_i and c_i are the thermal energy released per a gram of the ^{235}U burned, initial mass of ^{235}U in grams and cost for fuel assembly of type i, respectively, M is the number of fuel types, and P_i and B_i^D are the total power of all fuel assemblies of type i in the core and their average discharge burnup, respectively.

The maximum allowed burnup is assumed to be 0.8 for WWR-M2 and 0.5 for WWR-M5 fuel assemblies, respectively. The total power of the reactor is 10 MW. The irradiation channels and control rods can not be moved, while fuel assemblies in the periphery of the core can be replaced by beryllium blocks. Φ_{th} is assumed to be $\frac{1}{3}(\Phi_{51/58} + 2\Phi_{80/32})$, where $\Phi_{51/58}$ and $\Phi_{80/32}$ are the average thermal neutron fluxes in the irradiation channels 51/58 and 80/32, respectively.

The core loading pattern is optimized using simulated annealing [2]. For neutronics calculation, the iterational hybrid method combining diffusion model with higher-order approximations of neutron transport equation is applied [3]. Thermal calculation of the core is performed using the approach described in [4].

3. Numerical Results

For the calculations, a Pentium III/733 – based personal computer was used. CPU time was about 3 hours for each optimization problem. The results are shown in Figs.1-4. For convenience, fuel expenditures are presented in US dollars ($) per effective full power day (EFPD).

As seen in Figs. 1-2, though WWR-M5 fuel is more expensive than WWR-M2 fuel, it can provide higher neutron flux and even less fuel expenditures when it is used in a mixed core. Since the WWR-M5 discharge burnup has to be less that 50%, the core consisting on only WWR-M5 fuel is possible only if the number of fuel assemblies in the core is very small, thus it can not be optimal if the "neutron price" is not very high.

As seen in Figs. 3-4, with growth of the "neutron price", the total number of fuel assemblies in the core (N) should be decreased to increase neutron flux, though this lowers discharge burnup and, hence, increases fuel expenditures. The optimal total number of fuel assemblies in the core is 218 for $p_n \leq 10^{-16}$ \$/(n/cm^2) and 151 for $p_n = 2 \cdot 10^{-16}$ \$/(n/cm^2).

Meanwhile, the number of WWR-M5 fuel assemblies (N_5) should be increased with growth of the "neutron price". The optimal number of WWR-M5 fuel assemblies is 20 for $p_n = 0$, 40 for $p_n = 10^{-16}$ \$/(n/cm^2), and 60 for $p_n = 2 \cdot 10^{-16}$ \$/(n/cm^2).

As seen in Figs. 3-4, since WWR-M5 fuel assemblies contain more ^{235}U and provide more intensive heat transfer than WWR-M2 fuel, they should be located near the irradiation channels to make power density in these areas, limited by the maximum allowed temperature of fuel surface, as high as possible, thus increasing neutron flux. Rise of the number of WWR-M5 fuel assemblies results in increasing the discharge burnup of WWR-M2 fuel and, hence, the power near the irradiation channels.

As seen in Fig.5, utilization of mixed-fuel is especially effective for high "neutron prices".

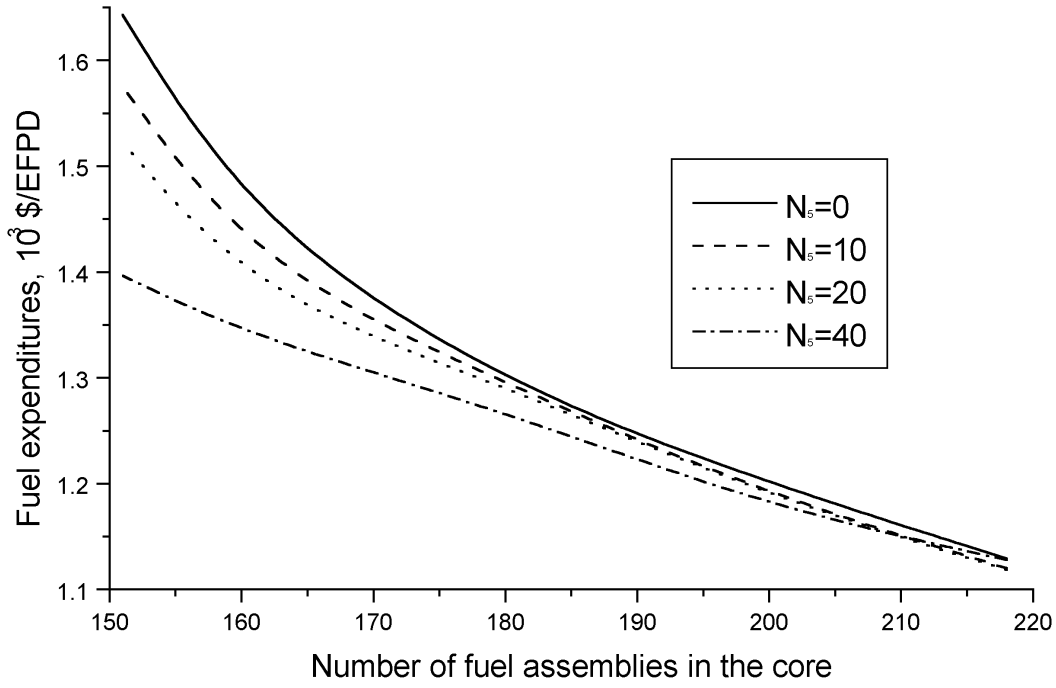

Fig.1. Fuel expenditures (C_F) as a function of total number of fuel assemblies in the core (N) for various numbers of WWR-M5 fuel assemblies (N_5)

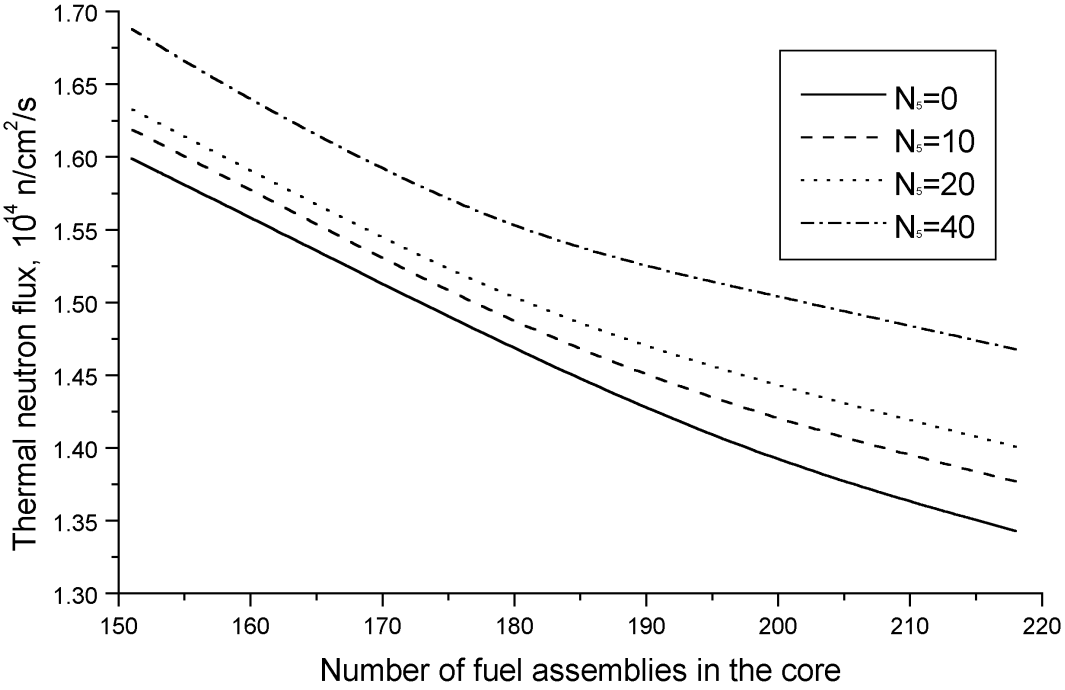

Fig.2. Average thermal neutron flux (Φ_{th}) as a function of total number of fuel assemblies in the core (N) for various numbers of WWR-M5 fuel assemblies (N_5)

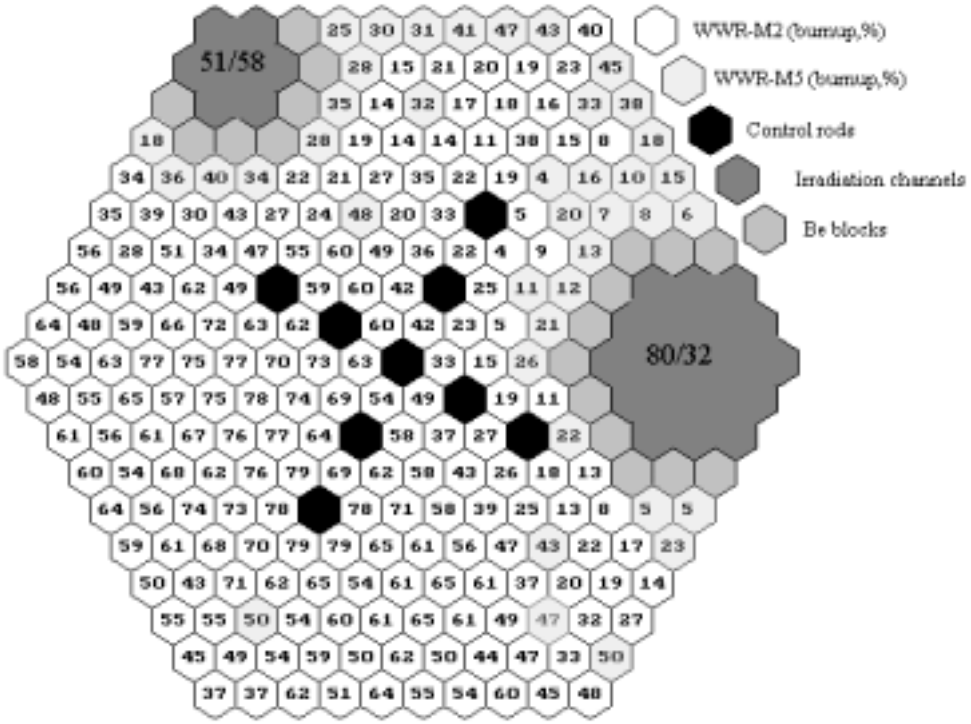

Fig.3. Optimal core loading pattern for $p_n=10^{-16}$ \$/(n/cm^2)

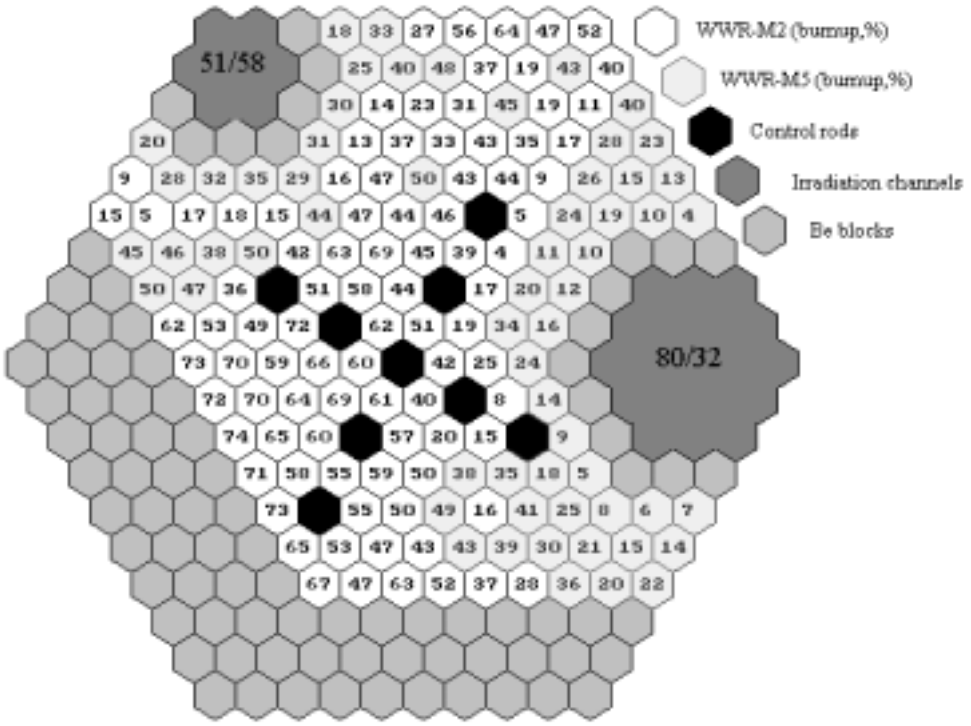

Fig.4. Optimal core loading pattern for $p_n=2\cdot10^{-16}$ \$/(n/cm^2)

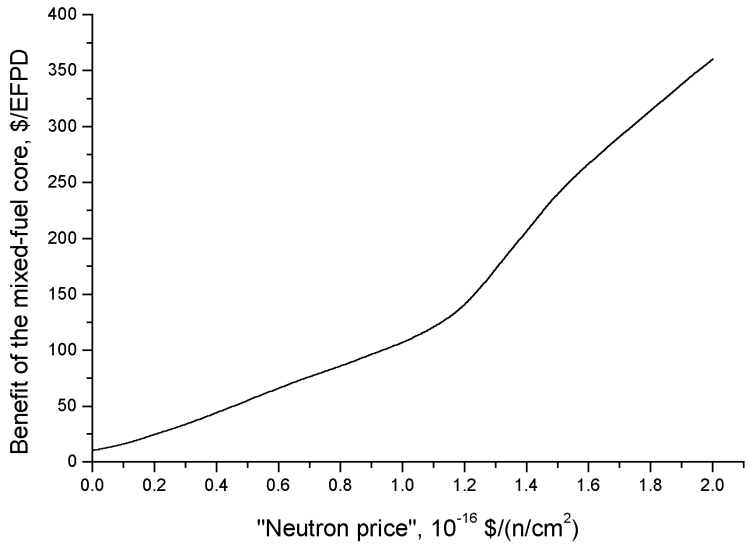

Fig.5. Benefit of the mixed-fuel core in comparison with the core consisting of only WWR-M2 fuel

4. Conclusions

The in-core fuel management optimization problem was studied for the WWR-M research reactor in Ukraine. In this study, the core loading pattern was optimized using simulated annealing. For neutronics calculation, the iterational hybrid method combining diffusion model with higher-order approximations of neutron transport equation was applied. Two available types of fuel assemblies were considered: WWR-M2 with 36% enrichment and WWR-M5 with 90% enrichment, containing 32 and 66 grams of ^{235}U, respectively.

It has been demonstrated that a mixed core consisting of WWR-M2 and WWR-M5 fuel provides higher neutron flux and even less fuel expenditures in comparison with the core consisting of only WWR-M2 fuel. Utilization of mixed-fuel is especially effective for high "neutron prices". The optimum core configuration, as well as the number and location of fuel assemblies of different types in the core strongly depend on the "neutron price". With growth of the "neutron price", the total number of fuel assemblies in the core should be decreased from 218 for $p_n \leq 10^{-16}$ \$/(n/cm^2) to 151 for $p_n = 2 \cdot 10^{-16}$ \$/(n/cm^2). Meanwhile, the number of WWR-M5 fuel assemblies should be increased with growth of the "neutron price" from 20 for $p_n = 0$ to 40 for $p_n = 10^{-16}$ \$/(n/cm^2) and then 60 for $p_n = 2 \cdot 10^{-16}$ \$/(n/cm^2).

For a not too low "neutron price", WWR-M5 fuel assemblies should be located near the irradiation channels to make power density in these areas, limited by the maximum allowed temperature of fuel surface, as high as possible, thus increasing neutron flux.

5. References

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EXPERIMENTAL CAPABILITIES OF THE MIR REACTOR FOR TESTING AND QUALIFICATION OF RESEARCH REACTORS FUEL

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ABSTRACT

To test fuel element fragments, fuel elements and full-scale fuel assemblies (FA) of research reactors, lately there have been developed, manufactured and approved several types of irradiation devices. This paper presents design features of the MIR reactor and technical characteristics of irradiation devices for plate type fuel elements testing.
The design of the irradiation devices for fuel elements' fragments (miniplates) testing is dismountable that enables to carry out intermediate inspections and to replace miniplates in the course of the test.
During the test coolant inlet and outlet temperature, outlet flow rate and pressure are measured in the experimental channel. The thermal power of the tested fuel elements is calculated by the measured thermodynamic coolant parameters. The coolant pressure ~1,1 MPa, the coolant temperature at the inlet of the test channel ~(40-60)°C, coolant flow rate can be adjusted within the range between 6 to 130 t/h, non-disturbed thermal neutron flux ÒÅ·10^{18} m^{-2}. Neutron flux or heat power in the test channel can be maintained at Ðpreset level. The maximum heat power in the test channel is up to ~3,5 MW. The height of the fuel elements' core is up to 1000 mm, the irradiation device maximum diameter is up to ~148 mm.

1. Introduction

The MIR reactor is mainly designed to test fuel element fragments and fuel assemblies (FA) of different type nuclear power reactors under normal (stationary and transient) operating conditions as well as in certain project emergency situations. For these tests there are loop installations. At present there are six test loop installations being in service (table 1).

Table 1. Principle parameters of loops

Item	Property, units	Loop facilities						
		PV-1	**PVK-1**	**PV -2**	**PVK -2**	**PVP-1**	**PVP-2**	**PG**
1.	Coolant	water	boiling water	water	boiling water	water, steam	water, steam	nitrogen, helium
2.	Number of test channels	2	2	2	2	1	1	1
3.	Maximum channel power, kW	1500	1500	1500	1500	100	2000	160
4.	Maximum coolant temperature, C°	350	350	350	365	500	550	500
5.	Maximum pressure, MPa	17,0	17,0	18,0	18,0	8,5	15,0	20,0
6.	Maximum coolant flow rate through the channel, m³/h	16,0	16,0	13,0	13,0	0,6	10,0	

Furthermore, the MIR reactor makes it possible to test fuel of water-cooled research reactors in test channels not connected with loop installations and being directly cooled by reactor primary circuit coolant. Such tests are the most representative ones for research reactors having water pressure and

temperature in primary circuit similar to the MIR reactor, i.e. ~1 MPa and ~ 50°C correspondingly.
Lately there have been performed a series of qualification tests of the low-enriched fuel with ring MR type fuel elements within the framework of the Russia RERTR program, a validation test on serviceability of MTR type plate fuel under commercial projects, and there are tests of rod-type fuel elements with low-enriched fuel within the framework of cooperation with ANL under RERTR program that are being planned.
The present paper treats experimental potential and test procedure features of plate-type research reactor fuel in the MIR reactor.

2. Design features of the MIR reactor core for fuel testing

The MIR reactor is a heterogeneous thermal reactor with a moderator and a reflector made of metal beryllium [1]. The cross-section of the reactor with the basic elements of the core is shown in figure 1.1. It has a channel-type design and is placed in the water pool. The frame of the core is made up of hexagonal beryllium blocks with width across flats of 148,5 mm on the triangle grid with 1,5 mm gaps between them. In the central axis holes of the blocks channel bodies are installed for operating FAs (37 pcs); combined operating FA with absorber (12 pcs); experimental channels (11 pcs). The channels to locate control rods are placed in the holes between the adjacent beryllium blocks.

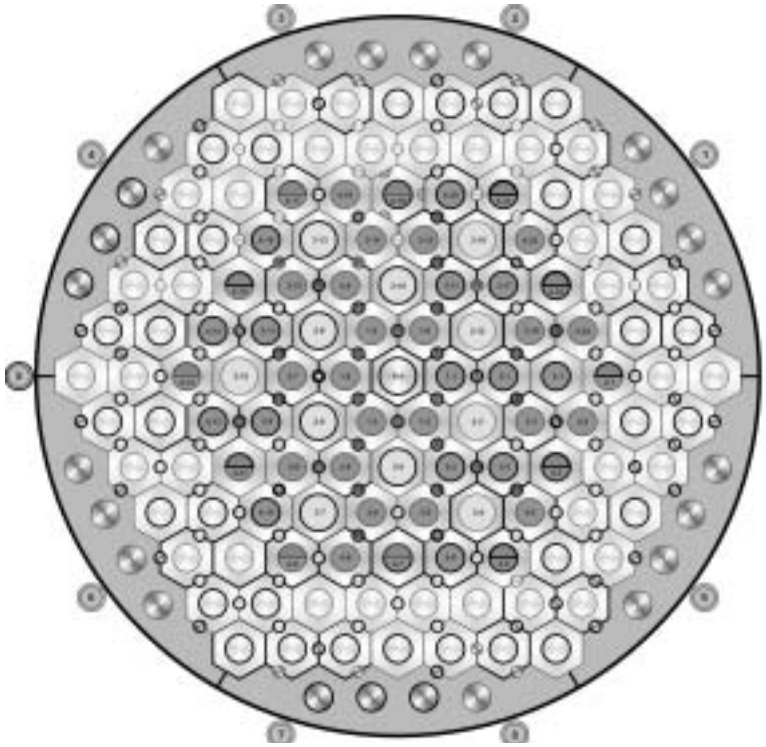


Fig 1.1. The MIR Reactor Core

– operating FA channel
– experimental channel
– combined hanger with the operating FA and absorber
– control rod channel

Each experimental channel is surrounded by six channels with operating FAs and (3÷5) control rods. By varying operating FAs burn-up at overloads and control units location around test cells it becomes possible to ensure simultaneous maintenance of test conditions practically in all the experimental channels. The tests were performed in the reactor simultaneously in 10 experimental channels with neutron flux maintenance on a preset level in each channel, and with that the thermal neutron flux value in different channels varied up to 5 times. The maximum thermal neutron flux in experimental channels is $5 \cdot 10^{18}$ m^{-2}.
Another important feature of the MIR reactor is the location of operating and test FAs in separate channels and the coolant supply independently in each channel. At the channel outlet pipelines there are mounted valves for remote control of the coolant supply, flowmeters and temperature sensitive elements. There is also envisaged the sampling to the fuel element cladding leak test system.
The unique reactor design makes it easy enough to mount test devices or standard operating channels instead of loop installations, and to connect them to the reactor cooling system. In such experimental devices it is possible to perform fuel tests at the parameters of the MIR reactor primary circuit coolant.

3. Design of irradiation devices for plate-type fuel testing

In order to qualify the MTR type plate fuel there have been designed two irradiation devices: one for reactor testing of plate fuel element fragments (miniplates) and one for full-scale FA testing.

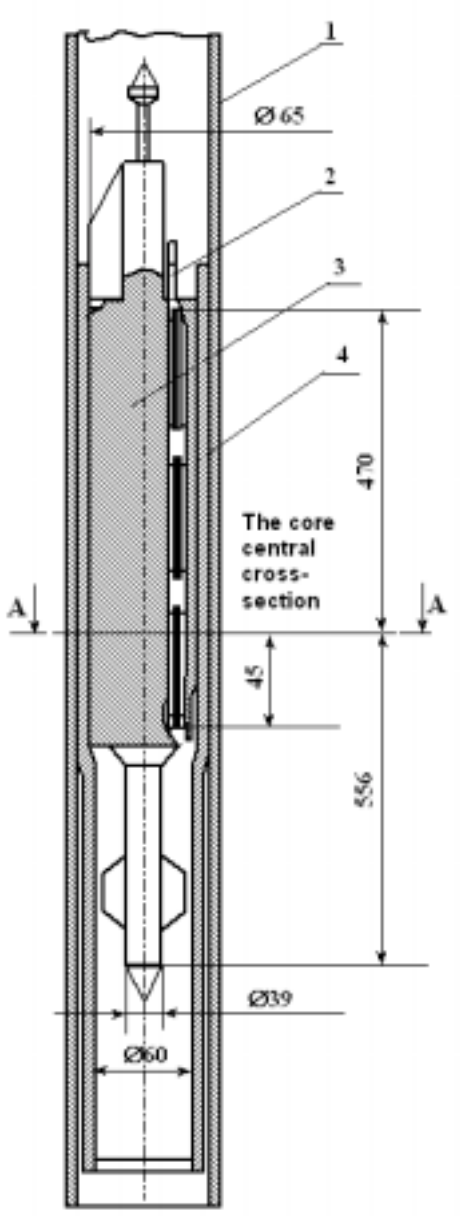


Figure 3.1 shows the cross-section and the longitudinal section of the irradiation device for miniplates testing. This irradiation device is mounted into a standard Ø78x1,5 mm operating channel of the MIR reactor with a special tool. In the irradiation device there are three dismountable cassettes, containing three miniplates each. Miniplates are located on different levels of the core height. The miniplates are tested at different heat powers, which is of interest for determining the influence of this parameter on the fuel serviceability.

Figure 3.2 shows relative distribution of miniplates heat power for three different mean fuel burns-up in the irradiation device.

In the course of tests during the reactor shut down the irradiation device can be removed from the channel and placed into the reactor pool. In its turn any of the cassettes can be removed from the irradiation device and the inspection of one of the miniplates' surfaces can be performed with a video camera.

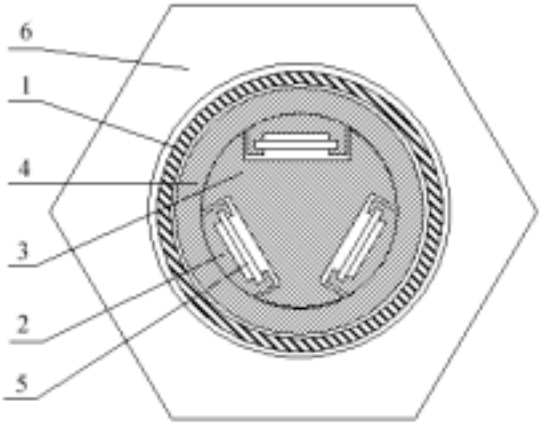


Fig 3.1. Irradiation device for miniplates testing

1 – channel vessel; 2 – cassette with three miniplates;
3 – central displacer; 4 – aluminium tube;
5 – miniplate; 6 – beryllium block.

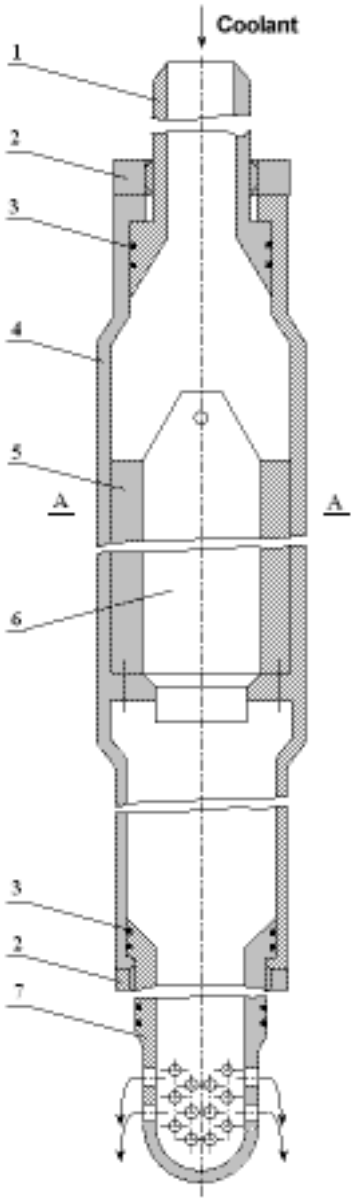
Then the cassette can be reinstalled into the irradiation device to continue the tests or can be sent for cooling with further transfer to the post-irradiation examination. To replace the removed cassette a new cassette or a displacer is put into the irradiation device. It is also worth mentioning that the cassette design makes it possible to carry out the measurement of the oxide film thickness using the superimposed eddy current sensor and gamma scanning without removing the miniplates from the cassette. After the intermediate inspection there is a possibility to return the cassette with the miniplates for further testing.

The miniplates geometry, number, location along the channel height and the cross-section can be changed at the customer's request. The maximum diameter of the experimental channel is up to 148 mm. In order to install the channels with more than 120 mm in diameter in the reactor, beryllium block is to be removed from the test cell.

Figure 3.3 shows the irradiation device scheme for FA testing. The irradiation device consists of a cylindrical body, displacers, ensuring the passage of the coolant through the fuel plates, end plugs for remote placement into the core and connection to the primary reactor circuit.

The irradiation device has a one-piece design. The FA is installed into the irradiation device at the mounting. The FA removal is carried out in the hot cell during the post-irradiation examination after the tests and cooling.

In the MIR reactor the 74,0×79,2 mm FAs were tested in the Ø120 mm channel. There exists a technical possibility to increase the diameter up to 148 mm while the FA dimensions can reach ~100×100 mm.



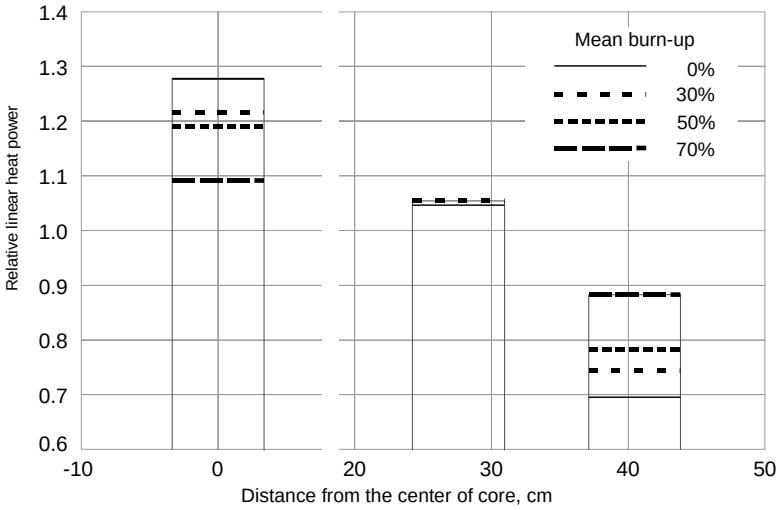


Fig 3.2. Relative heat power distribution in miniplates.

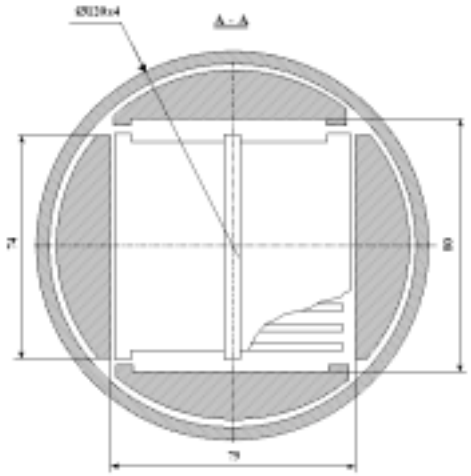


Fig 3.3. Irradiation device design for FA testing

1 – upper tube; 2 – nut; 3 – O-ring; 4 – aluminium vessel;
5 – displacer; 6 – FA; 7 – end plug.

4. Means and methods of test parameters control

Figure 4.1 shows the layout of the instrumentation to measure coolant parameters at the test channel inlet and outlet. The coolant pressure and temperature at the test channel inlet are determined by the results of the measurement of the cited parameters at the distributing manifold input. The coolant temperature and flow rate are measured directly in the channel outlet pipeline. The corrective amendments to stipulate the coolant temperature and pressure at the inlet and outlet of the irradiation device are attained by means of calculation and test while processing the results of the specially prepared experiments. After each channel the coolant is sent to cladding leakage detector. The method of determination is based on measurement of fission products' delayed neutrons.

In order to determine the total energy release power in the miniplates and FAs the procedure as follows is used:

$$N = k_1 \cdot C_P \cdot G\,(T_{C1} - T_{C2}) + k_2 (T_{C2} - T_{P1}) - k_3 \overline{N}_{OFA},$$

where: G, T_{C1}, T_{C2}, T_{P1} are parameters to be measured;

k_1 is the energy carry-over coefficient of n-, ɔ- irradiation beyond the channel ,

k_2 is the heat-transfer coefficient from the channel pipeline into the reactor pool water,

k_3 is the heating coefficient of the experimental channel by the reactor irradiation,

$\overline{N}_{OFA}$ is mean heat power of 6 surrounding operating FAs.

Coefficients k_1, k_2, k_3 are determined in special verification experiments using irradiation device fragments.

The distribution of the neutron flux, energy release and burn-up in the mini-plates and FAs are determined by calculations using a three-dimensional code MCU by "Monte-Carlo" method [2]. As a rule, in the course of preparation of experiments with new types of irradiation devices the investigation of flux distribution and reactivity effects are carried out on the physical reactor mockup. The attained results are used for verification of the MCU calculation model.

The usage of the calculation procedure to determine heat and neutron fluxes, fuel burn-up based on the measured test parameters makes it possible to increase the precision of the results. The experiments carried out in the reactor and post-irradiation examination of the miniplates showed the agreement between the Uranium-235 burn-up depth as per the above procedure and that as per the mass-spectrometry procedure in the limit of 7% within the range of burn-up depth values (30-70)%.

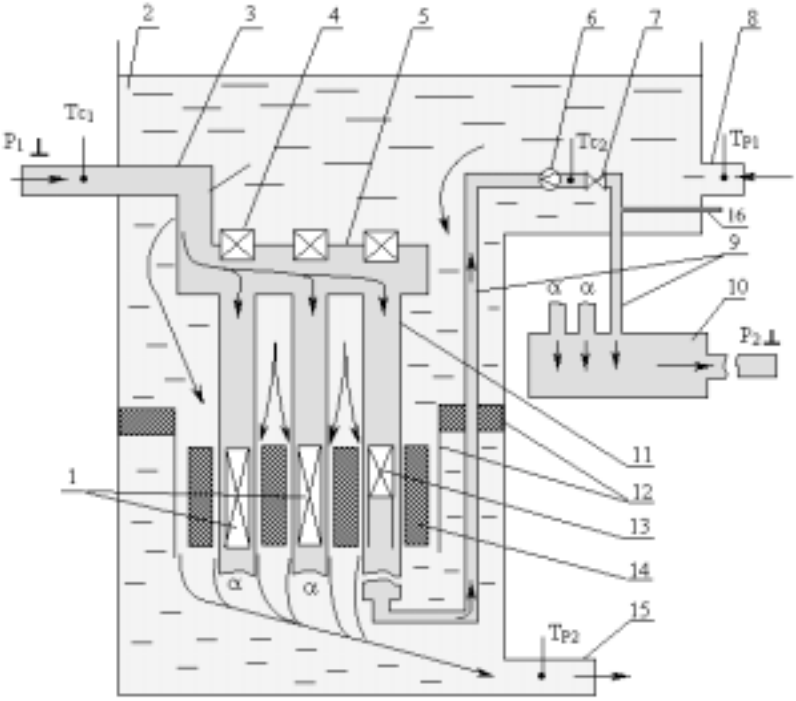


Fig 4.1. Layout of the instrumentation to measure coolant parameters.

σ_{u1}, σ_{u2}; σ_{t1}, σ_{t2} – thermometers; t_1; t_2 – pressure transducer.

1 – operating FA; 2 – reactor pool; 3 – primary coolant inlet; 4 – channel plug; 5 – inlet collector; 6 – flowmeter; 7 – adjustable valve; 8 – coolant inlet to the pool; 9 – RC outlet pipe; 10 – outlet collector; 11 – reactor channel; 12 – reactor casing; 13 – irradiation rig; 14 – beryllium block; 15 – coolant outlet from the pool; 16 – coolant sampling to cladding leakage detector.

5. Conclusion

Lately there have been performed several tests in the MIR reactor aimed at qualification of MR and MTR type research reactors' fuel. Extensive experience has been accumulated in regard to development and making of irradiation devices as well as test procedures, which is an evidence of an ample scope for using the reactor to test any type research reactor fuel. The fuel test at low parameters (P~1 MPa, ~50˚C) can be performed with test channels connected to the primary reactor circuit, and for testings at higher parameters loop installations are available.

The MIR reactor is an integral part of the RIAR material testing complex, i.e. after irradiation tests the fuel is transferred for post-irradiation examination through an organized transport-technological chain to the material testing department where all modern destructive and non-destructive methods are available. Thus, in the SSC RIAR all the conditions for test and qualification of research reactors fuel are available, and the institute is ready for cooperation with all the organizations interested.

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UPGRADING THE HOR:
AN UPGRADING STUDY FOR THE HOR

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ABSTRACT

Results of feasibility studies are presented for upgrading the HOR, a 2 MW pool-type research reactor at the Delft University of Technology. The primary utilisation of the HOR focuses on beam research applications with neutrons and positrons. The aim of being scientifically competitive in that research area requires a thermal neutron flux level of at least $1x10^{14}$ n/cm^2/s. The feasibility of upgrading the present core to a super compact core for reaching this goal has been investigated at large from a safety and operational point of view. For the upgraded core, a 3x3 fuel assembly arrangement and beryllium reflected at all sides was chosen. Figures on the system performance, including the merits of a cold neutron source application feeding the neutron guide system, are presented.

1. Introduction

The HOR is a pool-type research reactor at the Interfaculty Reactor Institute (IRI) of the Delft University of Technology using MTR-type fuel assemblies. It has been in operation since 1963, maintaining good performance by upgrading instruments and beam tube facilities as well as keeping good maintenance records over the years for the reactor equipment and ancillary facilities. The reactor core set-up itself has been modified several times, the latest major change being the core conversion form HEU to LEU fuel with core compaction from 32 to 20 fuel assemblies. At the moment, the HOR is operated on a regular basis at a thermal power level of 2 MW, attaining a modest thermal flux of about $2x10^{13}$ n/cm^2/s.

The primary utilisation of the HOR nowadays focuses on neutron-beam experiments plus positron beam research. However, an analysis of the performance of operating research reactors and their utilisation for neutron beam research clearly shows that to be scientifically competitive in that research area, such facilities should at least exceed an unperturbed thermal neutron flux level of $1x10^{14}$ n/cm^2/s at the thermal flux peak. The maximum flux should occur at the position of the beam port entrance coupled with a cold neutron source. Feasibility studies have been performed for modifying and upgrading the HOR with the aim to improve the utilisation performance in such a way that the above mentioned goal can be reached, without exceeding the maximum (for the time being) licensed thermal power level of 3 MW. These studies follow an integrated approach and are pertaining to the core itself and the beam port facilities, including a cold neutron source, neutron guides and coupling of core and facilities. Further studies are ongoing to improve the relatively short operation cycle and

the relatively low burn-up figures, which are disadvantageous economically. Thus, preliminary results on improvement of the operation cycle and fuel management when using higher fuel loadings are reported as well.

Fig. 1 gives a general view of the proposed core and beam arrangement showing the core only partially loaded with fuel and reflector assemblies. The assumed HOR core is a 3x3-arrangement, using LEU silicide fuel at the highest licensed and industrially available U-density of 4.8 g/cm^3. The core is composed of 5 standard fuel assemblies with an initial ^{235}U-loading of 420 g each and 4 control fuel assemblies with an initial ^{235}U-loading of 310 g each. The core is beryllium reflected at all sides, except at top and bottom. Moreover, at one core side the beam ports, including a cold neutron source facility, are embedded in a beryllium reflector block. A core shroud to have clear coolant flow conditions surrounds the entire region of core and reflector, which consists of single assemblies. This shroud may make a new grid plate necessary.

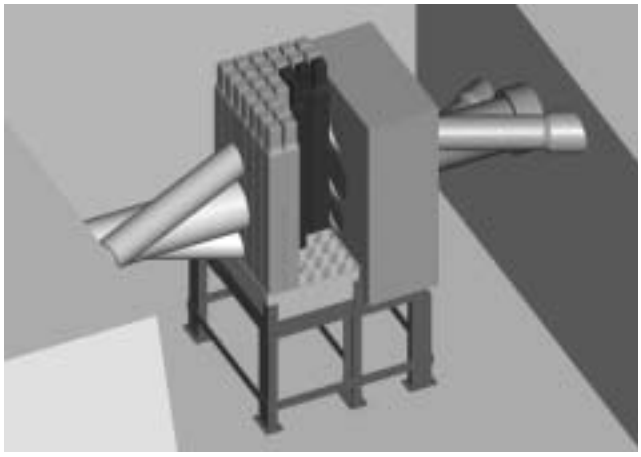

Fig. 1. General view with partially loaded core

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In Section 2, a comprehensive screening study [1] for a very substantial core size reduction and its consequences are discussed. In Section 3, the application of an elaborate MCNP model for calculating HOR configurations and fuel cycle issues is highlighted. Section 4 discusses the performance gain at the experimental facilities for different sets of modifications. In Section 5 some conclusions are drawn.

2. Screening study

In the screening study, the feasibility of a super compact core with only 9 fuel assemblies was investigated by performing neutronic and thermal-hydraulic calculations for 3 MW power level operation. The operational conditions were evaluated at the same time. The screening study was split into two parts, a nuclear and a related thermal-hydraulic analysis of the compact core plus of some accidental conditions, which may limit the compacting.

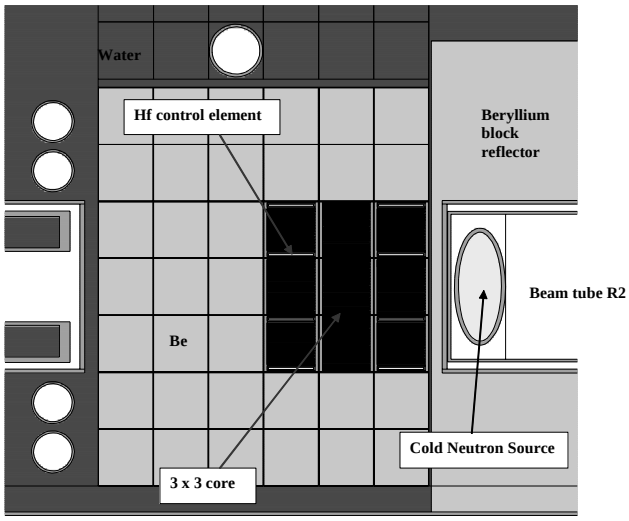

Fig. 2. Layout of 9-element core

2.1 Nuclear analysis

Step 1 of the nuclear analysis was performed by applying a burn-up chain with xy-diffusion code for each burn-up step of the core set-up (Fig. 2) on basis of the ENDF/B-IV cross section library. Group cross sections were established by transport code generated flux weighing.

Step 2 of this analysis applied MC-code MORSE-K to check and to adjust the results of step 1 at BOC by these high accuracy calculations. These calculations were performed in 60 energy groups based on the JEF-1 library. Special emphasis was put on the exact modelling of the geometry of core, reflector, supporting structures and the rather

wide beam tubes. Shut down margins were also received from MC-calculations. The actual HOR-license requires the assumption of 2 out of 4 shut down rods stuck as worst case.
Two approaches were made for the fuel management: Exchange of 1 fuel assembly or of 2 per cycle. Although the exchange of 2 per cycle need higher excess reactivity at BOC, the stuck rod case of 2 units simultaneously could be met. A higher U-235 loading however than what is possible today (4.8 gU/cm^3) may need to limit the management to the exchange of 1 assembly per cycle. With the actual foreseen fuel using the currently available U-density the two cycles under consideration lasted 26 FPD (1 assembly exchange) and 48 FPD (2 assemblies exchange), respectively. These cycles which are relatively short for a 3 MW research reactor, are partially also a consequence of the loading scheme, which is entirely directed to enhance the flux in front of the beam tubes, especially at the later position of a cold neutron source (see Fig. 2). The calculations of the 3x3 core demonstrated an adequate safety margin to exist for shutting down the core at any state during the operation cycle.

2.2 Thermal-hydraulics analysis

The enhancement of the performance of the HOR is linked to a higher value of the average power density in the core, which is mostly originating from the compacting of the core. That compacting cannot be recommended without a check of the heat removal capacity of the HOR-plant and of the thermal-hydraulics design of the new HOR-core.

As the HOR is a low-pressure system, the power density of the reactor core is limited by the requirement that hydro-dynamically stable coolant flow must be ensured in all cooling channels of the core. A violation of this criterion would lead to a drastic deterioration of the heat transfer accompanied by a sudden rise of the fuel plate temperature to an intolerable level.

The screening of the thermal-hydraulics design deals with the determination of a) the coolant flow rates through the various coolant channels of the core based on an overall flow rate at HOR of 80 kg/s, b) the corresponding pressure losses, c) the temperatures in the coolant and in the fuel plates, and d) the safety margins against the above mentioned flow instabilities.

The applied code system for thermal-hydraulic design of research reactors basically consists of a survey code for parametric studies, a network code to determine flow distributions within a network of hydraulic resistances and a sub-channel code to investigate the thermal-hydraulic details within the sub-channels and the fuel plates of the fuel assemblies.

The maximum permissible power, which can be extracted from a cooling channel at low system pressure under forced convection conditions, is limited by the excursive flow instability (FI) phenomenon. The numeric quantity suitable to comprehend the FI-phenomenon is the flow instability power ratio FIPR describing the factor by which the reactor power can be increased without violation of the criterion of the minimum acceptable safety margin.

In case of the HOR compact core, about 82 % of the total flow rate is used to cool the fuel plates, which is quite a good value expressing a sound design. The irreversible pressure loss across the core amounts to 0.138 bar, larger than the value of 0.026 bar of the present core. Having however in mind that the pump head at nominal flow rate of 80 kg/s is 3.2 bar, this increase of the flow resistance of 0.112 bar seems to be of minor importance.

For the HOR compact core the analysis of the coolant channels (10 different types) resulted in a maximum meat temperature in the centre of the plate of the hottest channel of 91.2 °C. The minimum FIPR of 2.42 for the related sub-channel indicates that the HOR 3x3 core is well protected against instabilities; the well-known minimum bubble detachment parameter η linked hereto is 179.1 cm^3 K/J.

In the limited transient analysis as performed for the screening, the Loss of Flow-accident (LOF) turned out to be the limiting case. Thus, it was decided to consider only the LOF-accident within the screening. The main result of the LOF-analysis is documented in Fig. 3.

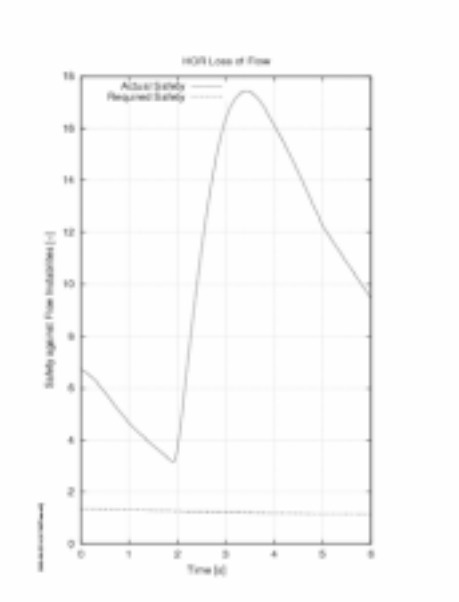

Fig. 3. Safety at Loss-of-Flow

During LOF the forced convection mode with top to bottom flow reverses into the natural convection. During that reversal the axially averaged heat flux of the hottest plate is about 4 W/cm^2. This value is well below 6 W/cm^2, which is the limiting value, which guarantees that pressure pulses during the chugging of the coolant are less than ± 0.1 bar in order to avoid additional mechanical loads to the plates. The value of 6 W/cm^2 has been determined experimentally in a test section with flow reversal simulation. The LOF-analysis resulted in a safety against flow instabilities of 3.17.

It is obvious from the results that there is quite a large flexibility with respect to the pump half time, as the safety against flow instabilities gained from the screening is larger for all considered cases than that minimum acceptable safety of 1.27. From the analysis, it follows that the thermal-hydraulic safety margin for operation as well as for the design basis accidents to be assumed was demonstrated to be quite sufficient.

3. MCNP model calculations and fuel cycle issues

As mentioned before the screening study resulted in comfortable safety performance figures for the adopted fuel design and power level. On the other hand, the operation cycle length turned out to be relatively short for a 3 MW reactor, also resulting in relatively low burn-up figures for End-of-Life-fuel (EOL). To improve the situation, one could try to go to higher U-densities in the meat. However, higher density fuel than the present limit of 4.8 gU/cm^3 is not expected to be licensed and commercially available shortly.

Another possibility for increasing the assembly loading is by changing the fuel geometry using thicker fuel plates with thicker meat [2]. Of course, at the same time the number of fuel plates of the FA decreases, but if the thermal-hydraulic safety margins are sufficient still, this could be a good option. As has been demonstrated by the screening study, this is a valid proposition for the HOR case studied.

A general investigation has been performed concerning the merits of such a choice to get an impression of possible operation cycle improvements, without going into optimisation and additional safety assessment considerations. The technical details in terms of the calculation approach for the present HOR configuration and for two possible fuel options, indicated as the HOR-1 and HOR-2 cases are discussed below. Furthermore, the calculation model includes a Cold Neutron Source option for considering the utilisation aspects for different fuel assemblies and core set-ups.

3.1 The MCNP model of the HOR

Independent of the work under 2.1 above, a MCNP model was established to calculate present HOR configurations and to study the upgrading alternatives. The model describes the fuel assemblies by individual fuel plates and uses 15 axial regions along the height of the fuel. Each movable beryllium reflector element is modelled separately. For the upgrading studies an optional beryllium block reflector of 70×30×100 cm is also included. All the existing beam tubes are represented in the model. The most important beam tube R2 is modelled including the concept cold neutron source: an ellipsoidal

region of hydrogen in aluminium cladding. In x-y-z directions the overall 3D model extends to 180×155×180 cm. The insertion of the control absorbers can be changed individually.
The material composition of the fuel is given per fuel assembly and per axial region. The burn-up dependent fuel composition is determined separately. A modified scale sequence called SAS6 is used to generate on cell level detailed nuclide densities for different burn-up levels. The generated tables can be used in different codes like Bold-Venture, KENO and MCNP. In this case, the excess reactivity (xenon-free, cold condition) for fresh fuel and at different stages of average burn-up (uniform for all assemblies) has been determined. However, an axial burn-up shape is used for each assembly, validated for the HOR configuration and determined from nodal 3-D calculations. The depletion of beryllium and the photo neutron production in beryllium have been neglected. For the calculations the ENDF/B-VI data library was used as distributed with the MCNP4C-2 package. The model has been validated on a recent HOR configuration by comparing reactivity and the measured axial distribution of ^{59}Co activation reaction rates. The average value of the ^{59}Co activation measurement was used for power normalization, the ratio of the total power to fission power is assumed to be the same at all systems. A horizontal cross section of the 9-element core as plotted by MCNP is shown in Fig. 2.

3.2 Calculations with MCNP

Calculations have been performed with two types of (conceptual) fuel: HOR-1 and HOR-2. The HOR-1 fuel type is nearly identical to the fuel characteristics used in the screening study discussed in Section 2, whereas the HOR-2 fuel loading is taken from one of the cases considered in [2]. The relevant data characterizing these fuels are given in Table 1. Just for reference, the fuel characteristics of the present HOR configuration are also listed in Table 1.

	HOR	HOR-1	HOR-2
Fuel meat material	U_3Si_2-Al	U_3Si_2-Al	U_3Si_2-Al
Uranium density [g/cm^3]	4.3	4.8	4.8
^{235}U loading per standard FA [g]	300	410	546
Enrichment [%]	19.75	19.75	19.75
Meat thickness [mm]	0.5	0.51	0.76
Cladding thickness [mm]	0.35	0.39	0.38
Fuel plate thickness [mm]	1.20	1.29	1.52
Moderator channel thickness [mm]	3.0	2.23	2.46
Hf absorber blade thickness [mm]	n.a.	3.1	3.1
Number of plates in standard FA	19	23	20
Number of plates in control FA	10	17	15

Table 1. Characteristic data of two potential fuels compared to the present HOR fuel

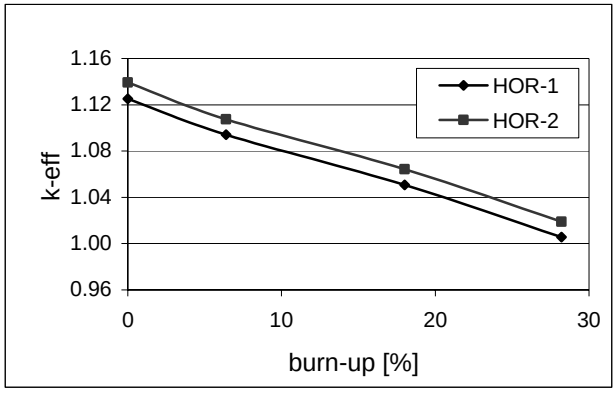

Fig. 4. k_{eff} as function of uniform burn-up

For each configuration the critical rod position, the shutdown reactivity and the excess reactivity were calculated. In addition, the reactivity when any two control rods are in the fully withdrawn position while the other two are fully inserted, was also determined (see para 2.1). The shutdown conditions have been met for in all configurations.

A set of calculations was performed with uniform burn-up, i.e. where all fuel assemblies in the core have the same burn-up. The dependence of k_{eff} on uniform burn-up for the two types of fuel is shown in Fig. 4. For purposes of comparison BOC- and EOC-excess reactivity

values were defined. Calculations with a realistic burn-up distribution and with all control rods fully withdrawn, gave k_{eff} (BOC) = 1.068 and k_{eff} (EOC) = 1.043 for cold, clean core conditions. The same values were also achieved by applying uniform (average) burn-up levels. For HOR-1 the average burn-up at BOC was 13.4%, the EOC value was 19.8%. For HOR-2 17.0% at BOC and 22.8% at EOC are the corresponding figures. These values, together with the resulting cycle data are displayed in Table 2. It can be seen that although the difference in burn-up at BOC and EOC is less for HOR-2 than for HOR-1, still, due to the much higher ^{235}U loading of the HOR-2 fuel, the achievable cycle is substantially longer. Due to the simplification with uniform burn-up, in practice the achievable operation cycle will be somewhat shorter than the figures displayed in Table 2. For actual burn-up distributions, the figures are estimated to be lower by about 15 %.

		Burn-up [%]		^{235}U consumed [g]		Cycle length [MWd]	
	k_{eff}	HOR-1	HOR-2	HOR-1	HOR-2	HOR-1	HOR-2
BOC	1.068	13.4	17.0	208	254	166	203
EOC	1.043	19.8	22.8				

Table 2. Cycle length calculation results for HOR-1 and HOR-2 fuel options

4. Performance

To gain an insight into the improvements of the performance by upgrading the HOR the neutron spectra at the experimental facilities were calculated. In the summer of 2000, at the HOR a multi-beam neutron guide system (NGS) was installed in beam tube R2 [3]. This NGS consists of four neutron guides, delivering neutron beams with high signal-to-noise ratio at the instruments in the experimental hall. Further the NGS is designed in such a way as to facilitate, in a later stage, the incorporation of a CNS. Beam tube R2 is considered to be the most important beam tube with respect to neutron beam research at IRI. Thus, performance calculations were restricted to this beam tube.

The calculation of the neutron flux at the experimental facilities was performed in two steps. First the wavelength-dependent flux for neutrons travelling in the direction of the neutron guides was calculated, both at the surface of the reactor core and at the surface of the CNS. A 42-point energy distribution and four direction values were used. Secondly the wavelength-dependent neutron fluxes at the exits of the neutron guides were determined.

The wavelength-dependent transmission of the NGS was determined by means of a Monte-Carlo code. This code treats the horizontal and vertical phase space separately, so the round surfaces of the CNS and the experimental tube R2 were approximated by square surfaces of the same area. As the outer diameter of the CNS is smaller than the inner diameter of R2, part of the neutrons entering the NGS do not come from the CNS but directly from the surface of the reactor core. Therefore three calculations were made: the transmission of a) neutrons coming from the CNS (A), b) neutrons coming directly from the core for an area equal to the inner area of R2 (B), and c) neutrons from the core but for an area equal to that of the CNS (C). The transmission of the NGS was then given by A+B-C.

This method was used before and has proven to yield neutron fluxes, which are in good agreement with, measured values [3,4]. Three different cases have been considered, involving differences in system configuration and operating power level:

1) present compact core configuration with 20 fuel assemblies and beryllium reflected on three sides at 2 MW power;
2) present compact core configuration, beryllium reflected at all sides and CNS installed in beam tube R2 and power level of 2 MW (partial upgrade case);
3) super compact core configuration with 9 fuel assemblies, beryllium reflected at all sides, CNS and increased power level of 3 MW (full upgrade case).

Fig. 5 and Fig. 6 show the results of the performance calculations. For the fully upgraded HOR, the cold neutron flux at the neutron guides exit is calculated to an integral value of 4.4×10^8 n/cm^2/s, which is comparable to the figures achieved at the Swiss continuous spallation source SINQ and a research reactor like the FRG-1. Another common figure-of-merit in connection with neutron beam research is given by the neutron flux per unit wavelength at a wavelength of 0.45 nm. With a value of 6.8×10^8 n/cm^2/s/nm for the upgraded HOR, this is about a factor of 40 higher than the present performance.

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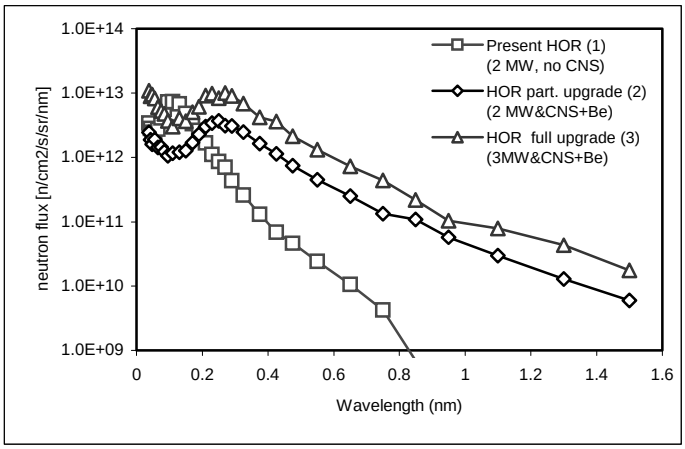

Fig. 5. Cold Neutron Source exit spectra

Table 3 lists both performance figures for the three cases considered.

case number	1	2	3
integral cold neutron flux [n/cm^2/s]	2.9×10^7	1.6×10^8	4.4×10^8
neutron flux at 0.45 nm [n/cm^2/s/nm]	1.6×10^7	2.2×10^8	6.8×10^8

Table 3. Performance figures for different core design options

Naturally, the partial upgrade option results in lower performance figures than the fully upgraded case, but it may be still an interesting proposition from a cost-benefit point of view.

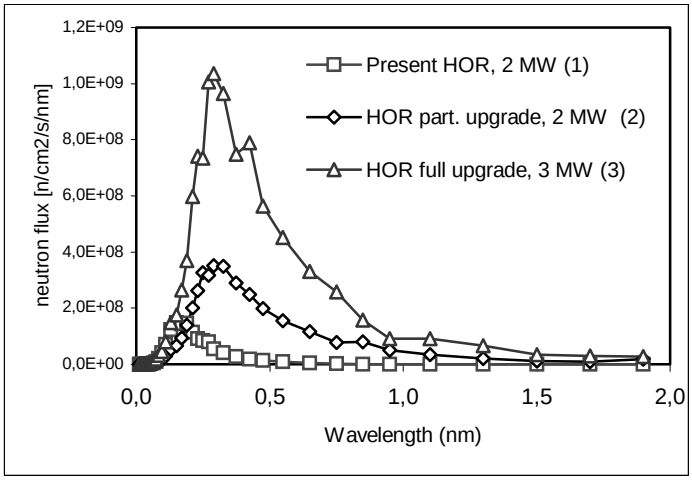

Fig. 6. Cold neutron flux at neutron guide system exit

5. Conclusions

The feasibility of upgrading the HOR to a scientifically competitive level of performance has been investigated at large. The work scope included a screening study, considering the nuclear and thermal-hydraulic aspects of a super compact core design of a 3x3 fuel assembly arrangement. Independent of the screening study, an MCNP model was set up for studying the upgrading alternatives. The studies included fuel cycle issues and the performance of a Cold Neutron Source option. Also, calculations were performed concerning the wavelength-dependent neutron intensity at the exit of the multi-beam neutron guide system.

The screening study indicated an adequate safety margin to exist for shutting down the core at any state during the operation cycle under the assumption of 2 out of 4 shut down rods stuck as worst case. The operating cycle was investigated on the basis of the exchange of either one fuel assembly per cycle or two assemblies per cycle. The resulting cycle length for both options turned out to be rather short for a 3 MW research reactor. This is partially due to the loading scheme, which is entirely directed to enhance the flux in front of the beam tubes and at the cold neutron source position.

In a separate study the merits of a different fuel design with higher fuel loading of 546 g ^{235}U per fuel assembly (density 4.8 gU/cm^3) for increasing the operation cycle length and discharge burn-ups were investigated. It seems feasibly to increase the cycle length by about 20 % in comparison to the fuel design adopted earlier, resulting in substantially higher discharge burn-up figures. Further optimisation and safety margins for higher loadings have not yet been considered by now.

The screening of the thermal-hydraulics of the shrouded core design resulted in adequate margins against flow instability for 3 MW operation and for the investigated LOF-accident for the super compact core.

In connection with the utilisation of neutron beam equipment, performance figures were calculated for the present core configuration at 2 MW power and for two upgrade options: A partial upgrade, keeping the present core configuration at 2 MW power, but extension with a cold neutron source and the beam tubes embedded in a beryllium block, and a full upgrade to a super compact core with 9 fuel assemblies, beryllium reflected at all sides, CNS and increased power level of 3 MW. A common figure-of-merit for the beam performance is the flux per unit wavelength at a wavelength of 0.45 nm (4.5 Ù). With a value of 6.8×10^8 n/cm^2/s/nm at the neutron guide system exit for the fully upgraded HOR, it improves the present performance by a factor of 40. For the partial upgrade this factor amounts to about 14. The integral cold neutron flux at the neutron guide system exit amounts to 4.4×10^8 n/cm^2/s for the fully upgraded system, and 1.6×10^8 n/cm^2/s for the partial upgrade. From the results mentioned above, it can be concluded that a full upgrade of the HOR is feasible. It can be considered to be a viable option to increase the performance to the desired level. Also, depending on cost-benefit criteria, a partial upgrade may remain an attractive proposition.

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ON-LINE MONITORING OF LOCAL IRRADIATION CONDITIONS IN BR2

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ABSTRACT

The thermal neutron sensitivities of a set of self-powered neutron detectors (SPND) with Co, Ag and Rh emitters were investigated extensively by detailed MCNP-based Monte Carlo calculations and by experimental comparisons of the SPND signals with activation dosimetry results in BR2 fuel channels as well as in reflector channels. The device containing these SPNDs was subsequently loaded in an appropriate BR2 channel to monitor continuously the local irradiation conditions in the neighboring FUTURE device, dedicated to MTR fuel plate testing. The SPND data can be used in a relative way to detect any changes in irradiation conditions. Moreover, they also yield reliable on-line information on absolute local neutron fluxes, which can be linked to other irradiation parameters (e.g. the local fuel element power) by advanced reactor core calculations (using the MCNP Monte Carlo code combined with SCALE/ORIGEN-S for burn-up determination) or be used to validate this type of calculations.

1. Introduction

Any irradiation in a materials testing reactor (e.g. in-reactor qualification tests of innovative fuel) can benefit from a continuous monitoring and control of the local irradiation conditions. Self-Powered Neutron Detectors (SPNDs) and miniaturized Gamma Thermometers can be used in a relative way to detect changes in irradiation conditions due to reactor operation actions or due to effects in the irradiated specimens itself. However, after a careful qualification of these detectors, they can also yield reliable on-line information on absolute local neutron fluxes and gamma heating rates, which can be linked to other irradiation parameters (e.g. the fuel element power) by advanced reactor core calculations.

In order to test MTR fuel plates in the BR2 reactor (SCK•CEN, Mol, Belgium), a dedicated device, called FUTURE (FUel Test Utility for REsearch reactors) was built and qualified [1]. For real qualification irradiations, usually stringent irradiation specifications are defined in terms of fuel surface temperature, fuel power and fuel burn-up. In order to minimize any undesired changes in fuel power level, the BR2 reactor conditions can be varied within a reactor cycle, on the basis of detailed time-dependent reactor core calculations using the MCNP Monte Carlo code in combination with SCALE / ORIGEN-S for burn-up determination. A posteriori experimental data on the integrated fuel power can be obtained from activation dosimetry using samples at various positions in the vicinity of the fuel plates and from gamma spectrometry and radiochemical analyses of the fuel plates after the end of the irradiation campaign. However, the quality of the irradiation can be assured in a much more reliable way by using on-line information on the fuel power, which can be obtained using the data from a set of SPNDs loaded in a BR2 channel next to the irradiation channel. These detectors have been qualified extensively by an experimental comparison with activation dosimetry data and by detailed Monte Carlo sensitivity calculations, thus yielding reliable instantaneous values of the local thermal neutron fluxes. Experimental neutron flux data can be used to validate the reactor core calculations; alternatively, using these calculation results to convert the flux data to fuel power data,

the SPNDs can serve as an on-line fuel power monitor. In this contribution, we will discuss recorded experimental SPND data and compare them with core calculation results.

2. Monte Carlo model for the calculation of induced currents in SPNDs

A self-powered neutron detector consists of a cylindrical metallic emitter surrounded by an insulator (MgO or Al_2O_3) and a sheath, usually made of Inconel or stainless steel. After neutron impact, several possible interactions lead to the creation of energetic electrons (mainly in the emitter). A fraction of these electrons is sufficiently energetic to cross the insulator. These electrons constitute a current which can be measured by connecting a current meter between the emitter and the sheath. SPNDs can be classified in two groups. In the delayed SPNDs the main process is the production of β rays upon decay of unstable isotopes formed after neutron capture; typical response times are of the order of a few minutes. In prompt SPNDs γ rays instantaneously produced after neutron capture create energetic electrons (by photoelectric effect, Compton scattering, or pair formation). Also γ rays from outside the detector can contribute to the detector current via similar processes, so the SPNDs have also some γ sensitivity. In many cases however, opposite currents due to electrons originating from the emitter and the sheath cancel to a large extent, leading to a small detector signal contribution.

For the calculation of neutron (and gamma) induced detector currents, we developed a model [2][3] on the basis of the Los Alamos MCNP-4B code [4]. A neutron (or γ) source is defined with an origin uniformly sampled on the outer surface of the detector and with an initial direction distribution proportional to cos(θ), θ being the angle between the direction and the surface normal. As initial energy 40 energy bins in the range from 1 meV to 10 MeV were taken for the neutron source and 20 energy bins from 0.02 MeV to 14 MeV for the γ source. MCNP calculates the time evolution of the ensemble of electrons (directly from the source or produced via γ interactions), γ rays (from the source or produced after neutron capture) and neutrons. MCNP does not follow the history of the unstable activated nuclei, produced by neutron capture. These will decay with their characteristic time and emit β and/or γ rays. To include this effect, which is crucial for the calculation of the sensitivity of delayed SPNDs, the calculation has to be performed in two steps: first the production rate and the spatial distribution of the activated nuclei is calculated from a specified neutron source, and next this information is used to define the spatial distribution of an electron (β) source (or a γ source) for the calculation of the electron currents. For the second step calculation a random distribution is assumed for the initial velocity direction. The energy distribution for the electrons is calculated according to the Fermi β energy distribution with the appropriate β end-point energy; for the γ ray energy distribution and intensity, experimental data are needed. For some isotopes (e.g. ^{103}Rh), the libraries associated with MCNP-4B do not include the generation of γ rays upon neutron capture. In this case a similar procedure as described above has to be applied to take into account the effect of prompt γ rays.

As output of the code we obtain (directly or in two steps as explained above) the net charge deposit in the respective parts of the detector (emitter, insulator, sheath). Usually the major contribution to the resulting current is the net charge deposit in the emitter (usually positive since negatively charged electrons are leaving). If all electrons stopped in the insulator would diffuse further to the sheath, it would be the only contribution. In order to determine the fraction of electrons stopped in the insulator that return to the emitter, the insulator is divided into a number of concentric segments and the charge deposition in each of these segments is calculated (by MCNP). This information is used to solve the Poisson equation for the potential profile in the insulator, assuming a zero potential at the inner and the outer insulator radius. Assuming that all electrons that are stopped closer to the emitter than the potential extremum position are driven back to the emitter by the space charge field, the fraction f of electrons ultimately returning to the emitter can be calculated. The net current is finally calculated as the net charge deposit in the emitter plus f times the net charge deposit in the insulator (which is negative if more electrons are stopped than emitted). To convert the obtained 'current per incoming particle' into a sensitivity ('current per flux unit'), MCNP also calculates the neutron flux in the SPND sheath per incoming particle. All fluxes referred to in this paper are (conventional) thermal neutron

fluxes defined as the product of the neutron density integrated up to an energy of 0.414 eV and the most probable velocity in a Maxwell-Boltzmann distribution with T = 293 K (v_0=2200 m/s).

3. Qualification of SPNDs in BR2

The DOLMEN (Device for On-Line MEasurement of Neutron fluxes) experimental rig contains a set of SPNDs and 2 gamma thermometers, distributed over 3 vertical levels with mutual distances of 270 mm [3]. Six of these SPNDs (with the characteristics specified in Table 1) were selected for the on-line monitoring of BR2 irradiation conditions. The complete rig could be moved in a controlled way within a range of 1000 mm in order to obtain axial neutron and gamma flux profiles. An extra calibration of the neutron field was foreseen by the introduction of Co, Ag and Fe activation wires. DOLMEN was inserted in BR2 reflector channels (K349, L180, G300) as well as in fuel channels (F166, A270).

Table 1. Characteristics and positioning (average vertical position of the SPND relative to mid-plane, in the lowest possible DOLMEN position) of the selected SPNDs in DOLMEN. All sensors were positioned about 7 mm off axis.

	Detector vertical position (mm) and orientation	Emitter material	Isolator material	Sheath material	Supplier	Diameters (emitter, insulator, sheath) (mm)	Emitter length (mm)	Cable diameter (mm) - number of cores
NFE1	-270; 180°	Co	Al_2O_3	Inconel	Studsvik	2/3/3.5	210	1 (2c)
NFE3	-270; 60°	Ag	Al_2O_3	St. steel	Thermocoax	0.5/1.1/1.4	85	1 (1c)
NFE4	-270; 300°	Rh	Al_2O_3	Inconel	Studsvik	0.7/2/2.5	50	1 (2c)
NFE7	0; 60°	Ag	Al_2O_3	St. steel	Thermocoax	0.5/1.1/1.4	85	1 (1c)
NFE9	0; 300°	Ag	Al_2O_3	St. steel	Thermocoax	0.5/1.1/1.4	85	1 (1c)
NFE11	+270; 60°	Ag	Al_2O_3	St. steel	Thermocoax	0.5/1.1/1.4	85	1 (1c)

A detailed comparison between calculation data, SPND signals and calibration data can be found in [3], focussing on compensation wire currents, on gamma-induced currents, on parasitic delayed current contributions from activation and on techniques to improve the temporal and spatial resolution of the neutron flux measurements. Here we only recall the main result: the consistency of all thermal neutron sensitivity data. By combining the (gamma-corrected) SPND signals with the global sensitivities calculated using the model described above, 'experimental' values for the thermal neutron fluxes from the SPNDs were obtained. In Figure 1, these were plotted against the calculated relative local thermal neutron fluxes; also activation dosimetry data were included. All data, for various DOLMEN levels in channel K349 and for various SPNDs, are quite consistent. Similar results were obtained for the other BR2 channels, including the fuel channels A270 and F166, which proves the validity - both in relative and absolute terms - of the developed model and also of the neutronic calculations.

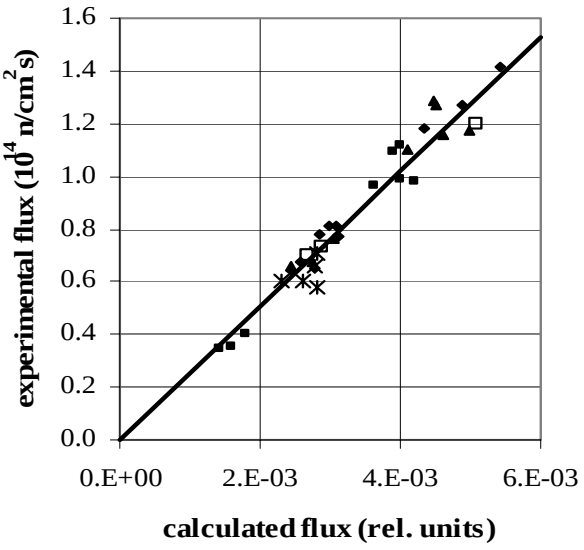
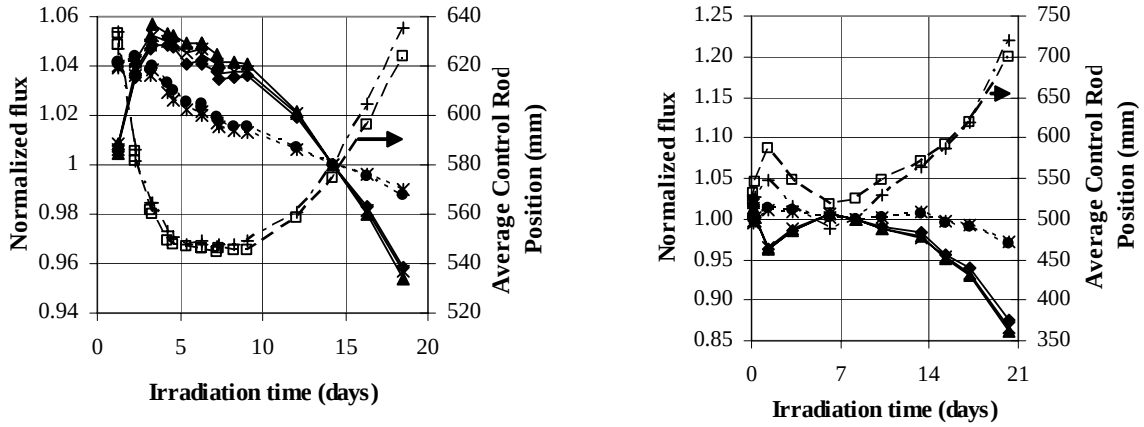


Figure 1. Experimental vs. calculated perturbed local thermal neutron fluxes. The three open squares refer to results from activation dosimetry and the other symbols to SPND results at four different vertical DOLMEN positions

4. Monitoring of the axial flux profile

The DOLMEN device can not be introduced in the same channel as FUTURE, so it was mounted in a neighbouring BR2 driver fuel channel. Several monitoring modes can be chosen: (1) irradiation at a fixed DOLMEN vertical position to obtain continuous flux data at a few selected locations, (2) continuous vertical scanning of DOLMEN to get quasi-continuous full axial flux profile measurements or (3) a number of short measurements by inserting DOLMEN and moving it out of the reactor core again after a certain dwelling time. In order to limit the burn-up of the SPND emitter material and to limit the influence of the presence of DOLMEN on the fuel irradiation parameters, the third option was chosen: every 2 days and also before and after every major change of BR2 conditions (e.g. power adjustment, load modification) a flux profile measurement was performed, with a duration of about 45 minutes (15' insertion, 15' dwelling at the lowest position, and 15' retrieval time).
In principle, the data of each measurement allow the extraction of complete flux profile data. Here we limit the discussion to the data obtained during two BR2 reactor cycles at the lowest DOLMEN position, i.e. flux data from SPNDs with emitter centres located at the reactor mid-plane, and 270 mm below and above mid-plane. The flux data from six SPNDs were first normalized to the BR2 power (ranging from 52 to 56 MW) and then normalized to unity at a selected date; the normalized flux data are shown in Figure 2, along with the average vertical position of the four nearest BR2 control rods.



All relative data for SPNDs at the same level are mutually consistent and the time behaviour of the corresponding axial profile can be perfectly correlated with the average control rod position evolution.

Figure 2. Normalized thermal neutron fluxes during the BR2 cycles 04/2002A (left) and 05/2002A (right) from NFE1-3-4 (solid lines – 270 mm below mid-plane), NFE7-9 (dotted lines – at mid-plane) and NFE11 (dash-dot line – 270 mm above mid-plane) together with the average control rod positions (open squares – dashed line).

5. Experimental validation of core calculations

Detailed MCNP neutron calculations have been executed to predict simultaneously the irradiation conditions in the fuel test channel and the thermal neutron fluxes at the positions of the SPNDs in the neighbouring channel. A 3D full scale Monte Carlo model of the BR2 reactor was used, including a detailed geometrical description of the inclined reactor channels and of all irradiation devices with their experimental contents (for cycle 04/2002A). The burn-up of the BR2 fuel elements and the poisoning of the beryllium matrix are also represented in the model. The calculations were performed for six irradiation times ranging from 0 to 17 days, taking into account the time evolution of the test and driver fuel compositions. The MCNP results in terms of generated fission power and flux per fission neutron were all normalized to the BR2 power (P_{BR2}) using a common multiplication factor for the intensity of fission neutrons I_n :

$$I_n = \frac{\nu P_{BR2}}{k_{eff} E_F^{BR2}},$$

where $\nu = 2.43$ neutrons/fission, k_{eff} the effective neutron multiplication factor and $E_F^{BR2} = 192$ MeV.

Table 2 summarizes the normalized experimental and calculated conventional thermal neutron fluxes at the SPND sheaths, averaged over the SPND length and over the BR2 cycle 04/2002A. The mean difference between both data sets amounts to -7.8 % with a standard deviation σ_n on the difference of 6.2 %. The relatively large spread in the individual deviations might be attributed partly to the assumption in the model that the BR2 driver fuel burn-up at the beginning of the cycle is cylindrically symmetric, while in reality for non-fresh fuel there will be an azimuthal dependence due to the influence of neighbouring channels (especially if containing control rods) during the operation history of the fuel element. The observed deviations are similar to the deviations between activation dosimetry results in the FUTURE device itself (during an earlier qualification experiment) and the corresponding calculated data [5]: in that case, the deviations ranged from +4 % to -11 %, leading to an average deviation of -3.5 % with a standard deviation of 4.7 %. Both experiments indicate a slight underestimation of the calculated data.

Table 2. Calculated and experimental thermal neutron fluxes in the SPND sheaths normalized to BR2 power and their percentage difference. The errors quoted for the calculated fluxes are purely statistical (from the Monte Carlo calculation). The experimental errors are entirely due to the uncertainties on the absolute neutron sensitivities, based on the results of the SPND sensitivity validation procedure; the noise on the observed currents is less than 0.2 %. The error quoted for NFE1 is higher than the others because of the higher relative contribution of gamma-induced currents to the detector signal (about 18 % compared to 3 % for NFE4 and 5.5 % for the other SPNDs).

	ϕ_{calc} ($cm^{-2}s^{-1}MW^{-1}$)	ϕ_{exp} ($cm^{-2}s^{-1}MW^{-1}$)	$\phi_{calc}/\phi_{exp} - 1$ (%)
NFE1	$1.00\ 10^{12}$ (±3.0%)	$1.20\ 10^{12}$ (± 10 %)	-16.1
NFE3	$1.14\ 10^{12}$ (±5.2%)	$1.13\ 10^{12}$ (± 5 %)	0.1
NFE4	$1.02\ 10^{12}$ (±4.5%)	$1.21\ 10^{12}$ (± 5 %)	-15.8
NFE7	$2.30\ 10^{12}$ (±3.9%)	$2.44\ 10^{12}$ (± 5 %)	-5.7
NFE9	$2.18\ 10^{12}$ (±3.6%)	$2.35\ 10^{12}$ (± 5 %)	-7.2
NFE11	$1.39\ 10^{12}$ (±5.2%)	$1.42\ 10^{12}$ (± 5 %)	-2.1

6. Conclusion

A procedure for on-line monitoring of irradiation conditions in the BR2 reactor was developed: a device with a set of SPNDs is installed in a channel close to the irradiation channel and the SPND data can be continuously recorded to monitor the irradiation process. The absolute thermal neutron flux data from the SPND data can also be used to validate detailed MCNP calculations for the BR2 core, so as to obtain more reliable information on crucial irradiation parameters like fission power. Data for 6 SPNDs during the BR2 cycle 04/2002A yield an average difference between experimental and calculated flux data of -7.8 % with a standard deviation of 6.2 %.

7. References

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BEHAVIOR OF URANIUM-ZIRCONIUM-HYDRIDE FUEL UNDER REACTIVITY INITIATED ACCIDENT CONDITIONS

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ABSTRACT

Pulse irradiation tests were performed with uranium-zirconium-hydride ($U\text{-}ZrH_{1.6}$) fuel to examine performance under reactivity initiated accident (RIA) conditions. During the pulse irradiations $U\text{-}ZrH_{1.6}$ temperature escalated up to 1240 K in pellet average. Departure from nucleate boiling (DNB) occurred in all the present tests, and cladding temperature reached 820 K at the maximum. Plastic radial deformation of the cladding was observed due to pellet-cladding mechanical interaction (PCMI). The deformation was less than 1.2% at the maximum. Rod internal pressure increase was observed during the pulse irradiation, which was probably caused by hydrogen dissociation in zirconium hydride. However, post test analyses of rod internal gas suggested that hydrogen was fully re-absorbed into zirconium in the pellet after fuel temperature decreased. Cladding ballooning or burst did not occur in the present tests.

1. Introduction

Uranium-zirconium-hydride ($U\text{-}ZrH_{1.6}$) fuel is widely used in TRIGA reactors over the world. The TRIGA fuel has a strong negative feedback nature to the reactor reactivity transient. This passive safety feature against reactivity initiated accidents (RIAs) has made the TRIGA reactors one of the easiest research reactors to use. For the TRIGA reactors, fuel temperature limits are applied for safe operation in order to avoid fuel rod rupture due to hydrogen dissociation. Characteristics of the TRIGA fuel, such as equilibrium hydrogen pressure, thermal and mechanical properties, have been reasonably well investigated [1][2]. However, integral rod behavior under severe power transient conditions has not been examined well. The current operational limits are based on conservative assumptions. In order to examine TRIGA fuel performance under power transient conditions, pulse irradiation tests of TRIGA fuel segments have been conducted in the Nuclear Safety Research Reactor (NSRR) at the Japan Atomic Energy Research Institute (JAERI). The results would provide basic data to extend operational limits of the TRIGA fuel and to enhance its reliability under the power transients. This paper presents results of first five tests with peak fuel temperatures up to 1240 K.

2. Test fuel rod and instrumentation

The fuel rods, short-segmented as illustrated in **Fig. 1**, were used in the tests. The fuel rod contained one $U\text{-}ZrH_{1.6}$ pellet of about 140 mm length. The uranium content was 45 wt% and U-235 enrichment was 19.7 wt%. Hafnium disks were located at both ends of the pellet to avoid undesirable axial power peaking during pulse irradiation. The fuel pellet was sheathed with an Incoloy-800H cladding tube of 13.8 mm in outer diameter and 0.41 mm in wall thickness. The radial

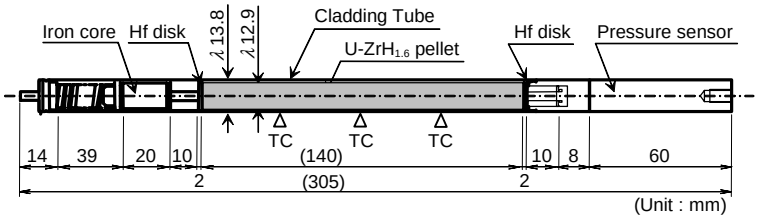

Fig.1 Schematic drawing of test fuel rod

Table 1 Specification of test fuel rod

Fuel Pellet	U-$ZrH_{1.6}$
Density (g/cm^3)	8.23
Uranium (wt%)	45
Enrichment (wt%)	19.7
O.D. (mm)	12.9
Height (mm)	140
Shape	Dished, chamfered
Cladding	Incoloy-800H
O.D. (mm)	13.8
Thickness (mm)	0.41
Initial Radial Gap (μm)	25

gap between the pellet and cladding inner surface (P/C gap) was 25 μm, which is much smaller than those of light water reactor fuel rods. The fuel rod was filled with argon of 0.1 MPa. Specifications of the test fuel rod are listed in **Table 1**.

The test fuel rod is instrumented with three thermocouples welded on the cladding surface, two axial elongation sensors for pellet and cladding, and a pressure sensor to measure the rod internal pressure.

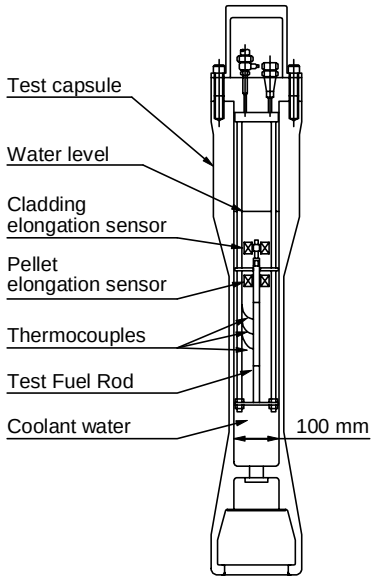

Fig. 2 Schematic of test capsule

3. Pulse irradiation and post test examinations

The test fuel rod was pulse-irradiated in stagnant coolant water of around 293 K and 0.1 MPa, which was contained in a stainless steel capsule of 100 mm in inner diameter. **Figure 2** shows a schematic of the test capsule. Five pulse irradiations were performed using the NSRR, which is a TRIGA reactor with capability to produce a power transient as shown in **Fig. 3**. In the time range shown in the figure, nuclear heat generation in the test fuel is so large compared to heat removal by the coolant water that fuel enthalpy can be estimated adiabatically from the energy deposition in the test fuel. The peak fuel enthalpies were estimated from 180 to 460 J/g for the present tests, and the corresponding peak fuel temperatures were from 770 to 1240 K. Note that these are radially and axially averaged values in the pellet when they reached the maximum. The peak enthalpies and temperatures during the pulse irradiations are listed in **Table 2**. According to neutronic analyses, the test rod has a radial power peaking factor of 1.25 in the NSRR core [3]. Therefore, the local fuel enthalpy could be 25% higher in the pellet peripheral region than the average. Even in the pellet peripheral region, the fuel temperature is much lower than the U-$ZrH_{1.6}$ melting temperature of 1950 K.

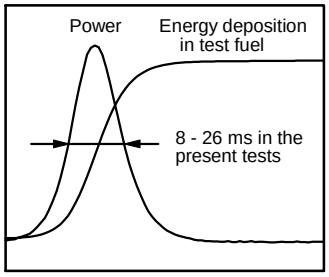

Fig. 3 Power history during pulse irradiation

Table 2 Pulse irradiation conditions

Test ID	Peak Fuel Entahlpy (J/g)	Peak Fuel Temperature (K)
TR-1	180	770
TR-2	250	910
TR-3	320	1030
TR-4	390	1150
TR-5	460	1240

After the pulse irradiation, comprehensive post test examinations (PTEs), including dimensional measurements, metallurgical examinations, rod puncture and gas analysis, were performed on the test fuel rods.

4. Results and discussion

4.1 Transient data during the pulse irradiation

Figure 4 shows transient data from test TR-3, including cladding surface temperature, axial elongations of both cladding and fuel pellet, and rod internal pressure. The origin of the time scale is adjusted to the onset of power

transient. The cladding surface temperature reached the peak of 735 K within 0.35 s. The escalation indicates that departure from nucleate boiling (DNB) occurred on the cladding outer surface. The pellet and cladding axial elongations simultaneously reached their peaks of 2.1 and 1.8 mm. The cladding elongation initially arises from cladding thermal expansion, but the cladding elongation history did not follow the cladding temperature drop at around 0.5 s. These observations suggest that strong pellet-cladding mechanical interaction (PCMI) sustained the cladding elongation. The rod internal pressure rose up to 0.5 MPa and then slowly decreased. It should be noted that the pressure was measured at the rod plenum region below the pellet as shown in **Fig. 1**. The measured pressure could be local pressure, not the average pressure of the rod interior; because the pellet-cladding gap was closed under the strong PCMI and axial gas transport could not occur. The pressure reached 0.5 MPa during test TR-3.

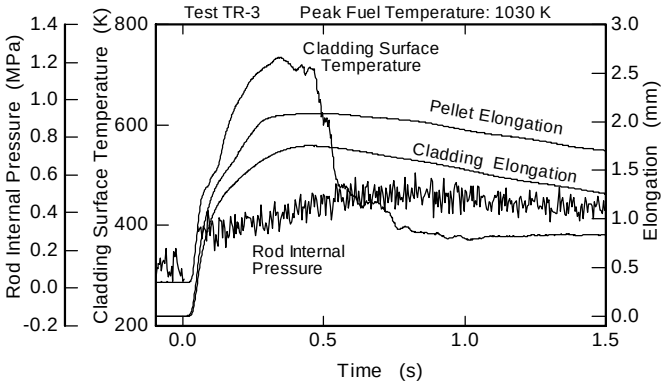

Fig.4 Transient data in test TR-3

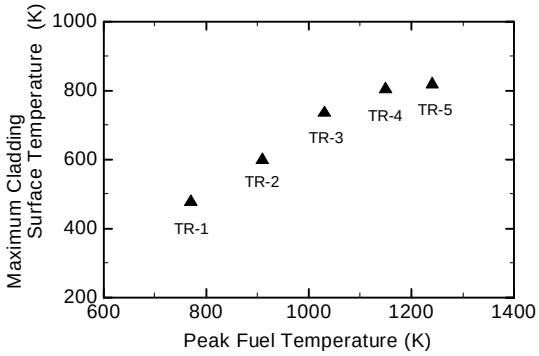

Fig. 5 Maximum cladding surface temperature

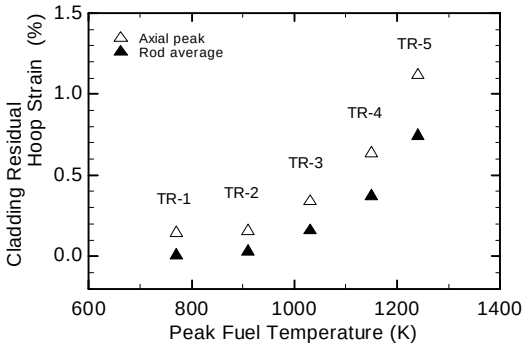

Fig. 6 Average cladding hoop strain

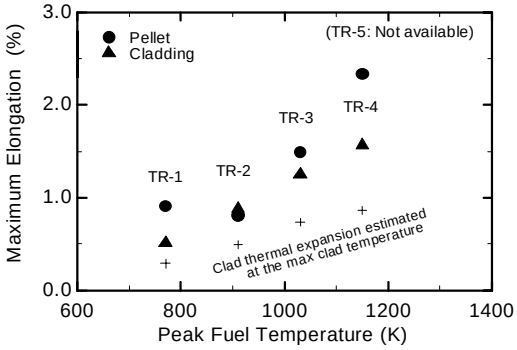

Fig. 7 Maximum elongations of pellet and cladding

4.2 Cladding surface temperature and fuel deformation

Figure 5 shows the maximum cladding surface temperatures measured in the pulse irradiation as a function of peak fuel temperature. Even in the test TR-1 with the lowest peak fuel temperature, DNB occurred and the cladding surface temperature reached 480 K. The maximum cladding surface temperature increased with the peak fuel temperature, and it reached 820 K in test TR-5. Each rod had three thermocouples at three elevations, but the present tests showed no regularity for the elevation of the maximum temperature.

Plastic radial deformation of the cladding in fuel active region was precisely measured after the pulse irradiation. The cladding residual hoop strains are shown in **Fig. 6** as a function of the peak fuel temperature. Both the rod averaged and axial peak strains are indicated in the plot. The strain was less than 1.2% at the axial peak in test TR-5 with a peak fuel temperature of 1240 K. One can consider that the plastic radial deformation was generated due to PCMI, since the rod internal pressure was too low to cause ballooning of the cladding.

Figure 7 shows maximum elongations of both pellet and cladding in the axial direction. Cladding thermal expansions, which were evaluated at the maximum cladding surface temperatures measured in the tests, are also plotted. Except in test TR-2, cladding elongation is larger than the thermal expansion, but smaller than the pellet elongation. This relationship indicates that the cladding was constrained by strong PCMI and mechanically strained by the pellet elongation beyond the cladding

thermal expansion level. As previously noted, the rod internal pressure was too low to cause the cladding ballooning.

4.3 Pellet metallography

Metallographs of pellets from tests TR-3, -4 and -5 are shown in **Photo 1**. The pellet microstructure of rod TR-3 seems unchanged from the fresh state. Rods TR-1 and -2, irradiated with lower peak fuel temperature conditions than that of test TR-3, showed similar appearances to TR-3. On the other hand, the pellet of rod TR-4 has radial cracks, and rod TR-5 shows fine and circumferential cracks in addition to the radial ones. These cracks were probably generated by pellet thermal stress or strong PCMI. These metallographs show that P/C gaps are not significantly changed. This observation supports that cladding ballooning did not occur in any tests, which is consistent with transient data presented above.

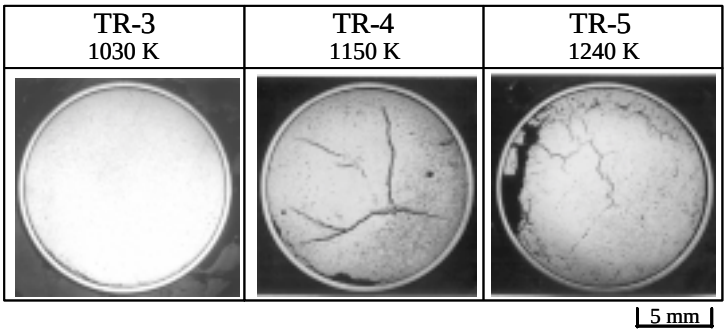

Photo 1 Pellet metallographs after pulse irradiation

4.4 Hydrogen dissociation

The rod internal pressure increase was observed in tests TR-3 and -4. It is considered that these pressure increases were caused by release of dissociated hydrogen from heated $ZrH_{1.6}$. The hydrogen dissociation level was roughly estimated, assuming that rod internal pressure increase was equivalent to the hydrogen partial pressure in the cladding. **Figure 8** shows the maximum increases of the rod internal pressure in tests TR-3 and -4 together with the equilibrium hydrogen pressure curve for $ZrH_{1.6}$ as a function of fuel temperature [1]. The results show that all hydrogen could be released from the fuel pellet to rod plenum. However, the results seem somewhat strange, because the partial pressures are larger than the equilibrium hydrogen pressures corresponding to their peak fuel temperatures. Some uncertainties could cause errors; (1) an assumption that the rod internal gas temperature is unchanged during the pulse irradiation, (2) possible higher temperature at pellet local region due to the radial power peaking, and (3) carbon impurity in the fuel coming in the production process, which can raise the equilibrium hydrogen pressure by forming ZrC [1]. The results shown in **Fig. 8** suggest that a significant amount of hydrogen could be released in the power transient. However, the maximum rod internal pressure did not reach the threshold for the cladding ballooning in the present tests.

After the pulse irradiation test, rod internal gas was sampled and analyzed. Residual hydrogen was detected as H_2 or CH_4. It was clarified that the fraction of the residual hydrogen to the total hydrogen initially contained in the fuel was of a magnitude of 10^{-5}. This suggests that dissociated hydrogen was fully re-absorbed into zirconium in the pellet after fuel temperature decreased. Regarding the carbon impurity, CH_4 was detected in the rod internal gas. This supports the raised equilibrium pressure by carbon impurity.

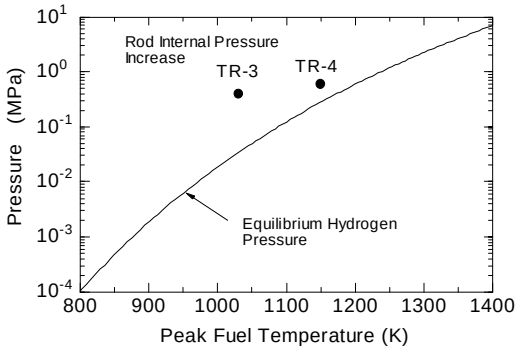

Fig. 8 Rod internal pressure increase and equilibrium hydrogen pressure

5. Conclusions

Five pulse irradiation tests were conducted with U-$ZrH_{1.6}$ fuel with peak fuel temperature conditions from 770 to 1240 K, in order to examine performance under RIA conditions. The present experiments have provided the following conclusions.

(1) DNB occurred in all the tests. The maximum cladding surface temperature increased with the peak fuel temperature, and it reached 820 K in test TR-5 with a peak fuel temperature condition of 1240.

(2) Plastic radial deformation of the cladding was measured. The maximum deformation was estimated as below 1.2% for the highest peak fuel temperature of 1240 K in the present tests. It is considered that the deformation was caused by pellet-cladding mechanical interaction (PCMI).

(3) Rod internal pressure increase was observed during the pulse irradiation probably due to hydrogen dissociation from zirconium hydride, but post test analyses of the rod internal gas suggested that the dissociated hydrogen was fully re-absorbed into the zirconium in the pellet after fuel temperature decreased. The rod internal pressure levels were too low to cause cladding ballooning or burst.

6. Acknowledgments

The authors would like to acknowledge the staffs in the NSRR Operation Division and Department of Reactor Fuel Examination Division for their support of pulse irradiation and gas analysis.

7. References

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[2] Simnad, M. T., et al., Nuclear Technology, Vol. 28, No. 1, p. 31, January 1976.

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FUEL IN-CORE MANAGEMENT AND SAFETY AT SAFARI-1

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ABSTRACT

The demand for reliability, availability, economy, efficiency and quality assurance have changed dramatically since the operation of the SAFARI-1 Research Reactor has evolved towards commercial isotope production rather than pure research. Optimising the routine refuelling shutdowns of the reactor is one area in which these demands have received attention. By innovation and careful planning, it is possible to complete the refuelling of the reactor, including a comprehensive analysis of the new core cycle, within 3½ days. It is shown how each step in the refuelling process is conducted while meeting the highest levels of excellence in safety, fuel management and quality control.

1. Introduction

SAFARI-1 is an MTR type tank-in-pool research reactor with a licensed operating power of 20 MW. It is a major producer of medical and industrial isotopes for both domestic and international consumption, the production of which is organised on a commercial and contractual basis. Consequently, the operation of the reactor is characterised by a high degree of flexibility and variability according to client demand. This has a number of interesting consequences on in-core fuel management and safety: firstly, equilibrium core cycles (which, by definition, require constant powers and cycle lengths) are impossible to attain since every core loading and cycle is different; while secondly, an annual utilisation factor approaching 90% means that refuelling shut downs are too short for a conventional theoretical approach to the refuelling process.

It has therefore been necessary to develop an approach to routine refuelling (i.e. refuelling without a change in core configuration) that takes account of all the safety requirements, including a full 3-D theoretical analysis, yet can be accomplished in the limited time available during a refuelling shutdown. This paper presents the approach that has been found to work very well for SAFARI-1.

2. Operating Cycle

The nominal operating cycle of the reactor is shown in Figure 1. It shows the end-of-cycle (EOC) shut down occurring on a Sunday morning at 07:00, a beginning-of-cycle (BOC) start up on Wednesday afternoon at 16:00 and power operation for the following 24 ½ days to the next EOC shut down.

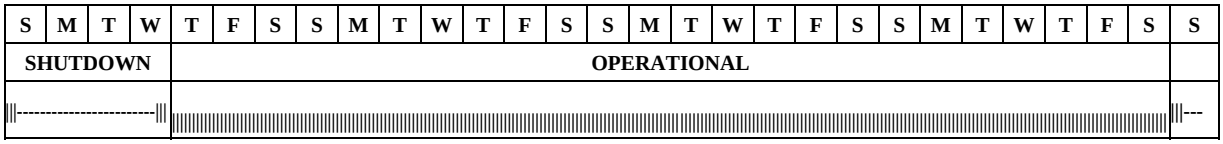

Figure 1: SAFARI-1 Nominal Operating Cycle

Variations that perturb the nominal cycle occur routinely as well as non-routinely. For example, once per quarter, the EOC shut down takes place on the Saturday preceding the first Sunday shown in Figure 1 in order to accommodate 3-monthly, 6-monthly, yearly and 2-yearly maintenance and in-service inspection activities. On a more irregular basis, various factors arise that lengthen, shorten or relocate shut downs, some planned and some unplanned.

3. Core Layout

A typical core configuration is shown in Figure 2, giving control rods, fuel assemblies and other devices as loaded at present. The core box geometry and layout are similar to that of many research reactors, having a 9 x 8 matrix of rectangular cells. Nine dedicated control rod positions are located at the intersections of columns 3, 5 and 7 and rows C, E and G (see Figure 2). Where control rod positions are not used for control, they can be filled with any of the other devices mentioned above. All the hollow devices in the reflector region represent one or other form of irradiation facility, e.g. hydraulic, pneumatic and long term irradiation facilities. The "flux traps" are devices called "thimbles", used mostly for short-term irradiation for isotope production. The numbers in the fuel assembly fields are discussed later.

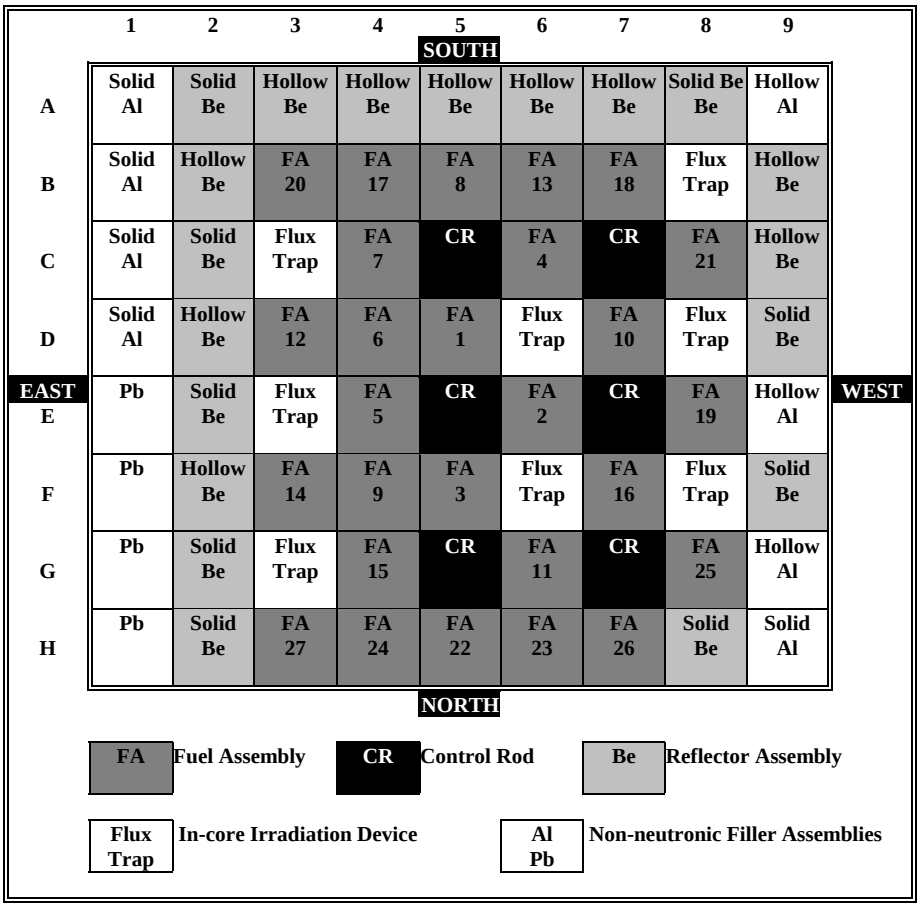

Figure 2: Typical Core Layout

4. Procedures and Codes

4.1 Core Management and Refuelling Procedures

There are several written procedures at SAFARI-1 that control the refuelling and management of the reactor core in accordance with the Operating Technical Specifications (OTS). Principal among these are:

A. Procedure for Core Refuelling Calculations;
B. Procedure for Fuel Transfers;
C. Procedure for Fuel Handling and Core Loading;
D. Procedure for Preparation and Measurement of Flux Wires;
E. Procedure for Irradiation of Flux Swords in the Reactor Core;

These procedures are subject to strict configuration management within the SAFARI-1 Quality Management System (QMS). They are applied by various authorised competent persons at one or

other stage during the refuelling process. The overall flow of the refuelling process in accordance with these procedures is followed in section 5 below.

4.2 Core Management Codes

The two main codes used in the refuelling process are both locally developed and are also subject to revision and version control within the QMS. Core management calculations may only be performed by fully tested, frozen versions of these codes. The two codes are:

SAFI-2000: A fully functional fuel and core management code based on a spreadsheet. While the code does not perform neutronic calculations it is the main instrument for determining the EOC condition of the previous cycle and the loading requirements, mass distribution map and BOC condition for the next cycle. It also computes the peak fuel temperatures based on experimental data and on the results of the neutronic calculations performed with the OSCAR-3 code (see below). It generates most of the quality records required by the QMS for the new core loading. In addition, SAFI-2000 maintains a complete database of all the fuel in the fresh fuel vault, the reactor core, the spent fuel storage pool and other designated areas for safeguards reporting.

OSCAR-3: An integrated nodal diffusion code system for performing full 3-dimensional neutronic calculations for the SAFARI-1 core. This code is the principal innovation that helps to complete the refuelling of the core in the short time available, due to the fact that employing the nodal calculation methods yields results more rapidly than the more conventional finite difference methods. Extensive benchmark testing of the code, against measurements and detailed heterogeneous Monte Carlo models, has shown that it produces accurate results in a fraction of the time required for the more conventional code systems.

5. Refuelling Activities

5.1 Preparation

The Reactor Safety division (known as the ST division) initiates the refuelling process prior to the shut down, i.e. on the Friday before the first Sunday in Figure 1. An estimate of the final MWh that will be produced by the depleted core at shut down is made and applied to the SAFI-2000 code to predict the ^{235}U mass in each fuel assembly and control rod in the core at EOC. The power requirements for the next cycle are then obtained from the Reactor Utilisation (RU) division, detailing the initial power after start-up and the timing of any planned power changes during the cycle. From this information the MWh to be produced over the next core cycle can be determined.

The initial ^{235}U mass of the new core is determined with SAFI-2000, and the mass is increased from the EOC value to the new BOC value by transferring spent fuel assemblies from the core database to the spent fuel database and transferring the same number of fuel assemblies from the fresh fuel vault database to the core database. Partially depleted fuel in the spent fuel database can also be sourced for transfer to the core in order to refine the mass distribution and final mass in the new loading. The code also determines the fuel assembly map for the new core by arranging the assembly ^{235}U masses in reverse order to a target neutron flux map based on a theoretical equilibrium core situation. For the present core configuration, the equilibrium flux map is given by the numbers in the fuel assembly locations in Figure 2, where 1 is the position with the highest flux and 27 the position with the lowest flux.

A hardcopy of the new core map is handed over to the Reactor Theory (RT) division for later reference and also to the Reactor Operations (RO) division, who then make out the requisite fuel transfer permits and determine the optimum sequence of fuel moves to make the transition from the old core to the new core, in readiness for the physical refuelling of the core on the Sunday. The optimisation of the fuel moves uses operations research techniques and is an important aspect for reducing the risk of human error during the physical refuelling activities and reducing the time it takes.

5.2 Physical Refuelling

The physical refuelling to change the core to the new loading is conducted by the RO division on the Sunday of the shut down and commences directly after the mandatory cool down time has elapsed (one hour). The planned fuel moves between the fresh fuel vault, the storage facility and the core take 6-8 hours, depending on the number of fuel moves involved. As each fuel assembly is moved, its own identification number and the unique identification numbers of its source and destination locations are confirmed by at least two competent persons and checked against the fuel move check lists. A physical record is also maintained of each move as it takes place by relocating tags on a tally board system. Furthermore, as each move is completed it is signed-off by the responsible supervisor on the fuel transfer permit and fully recorded in the control room logbook. All the completed documentation becomes part of the quality records of the new core cycle.

In the mean time, the RO division also calculates the actual total MWh delivered by the recently depleted core and completes a cycle history report containing the details, timing and duration of every change of power, change of moderator temperature, etc.

5.3 Theoretical Analysis

On the Monday of the shut down the RT division, equipped now with the actual EOC MWh and the cycle history report, can set this information up as input to the OSCAR-3 code and complete the full three-dimensional neutronic core follow calculations to obtain the detailed EOC conditions for the depleted core. They then shuffle the OSCAR-3 core model fuel assembly loading according to the core map received earlier (i.e. according to the way the core has already been physically loaded) and proceed to determine the following:

- The three-dimensional thermal flux and fission rate distribution in the core, identifying the location of the power density peak, for full 20 MW (irrespective of the actual start-up power) directly after start-up. This is the time of highest flux/power peaking due to the low control rod bank;
- The control rod worths, excess reactivity and shut down reactivity margin;
- A prediction of the control rod bank withdrawal required for criticality, for full power directly after start-up with the ^{135}Xe free core and after Xe equilibrium has been established;
- The expected cycle length of the core;
- Flux characteristics of all irradiation positions;
- Heating and power generation in all irradiation facilities that are subject to OTS limitations for heat transfer or other safety reasons.

The results are fully presented in a written report that becomes a quality record of the new core cycle. In addition, the thermal flux distribution data is transmitted to the ST division in electronic format for importing directly into the SAFI-2000 code in order to determine the fuel temperature distribution and peaking. An engineering hot spot factor, based on the effect of the fuel assembly manufacturing tolerances on the heat generation within and heat removal from the fuel plates, is applied by the SAFI-2000 code in the calculation of the fuel plate clad temperatures. These are calculated with the core power, coolant flow and coolant inlet temperature each set to its respective Scram trip setting and then checked for compliance with specified limits. The OTS states that no part of the fuel plate cladding shall have a temperature in excess of 130° C under the most adverse operating conditions, based on the consideration of incipient boiling. A 10° C margin is applied to this limit and clad temperatures in excess of 120° C are not accepted.

The theoretical analysis is usually completed by the Monday afternoon. If the results yield any non-compliance, e.g. a fuel plate hot spot temperature in excess of 120° C, a cycle length that is too short or too long, excessive heating in irradiation facilities etc, an adjustment can be made to the core loading and the whole theoretical analysis repeated. This seldom happens, but the shut down programme allows time until the Tuesday afternoon to deal with any such problems and to provide the quality documentation required in order to make the core critical.

5.4 Flux Measurements

By the Tuesday evening the ^{135}Xe in the fuel assemblies that remained in the core from the previous cycle will have decayed to a level that will not have any serious effect on flux measurements in the new core. A length of copper wire is inserted into each fuel assembly and control rod follower along its axial centreline and over the full height of the assembly. The wire is sealed inside an aluminium "sword", designed to fit freely between the fuel plates and to support the copper wire in a hole along its length. The reactor is made critical (at very low power) and the copper wires are activated for about ten minutes. On the Wednesday morning the swords and their contents are transferred to a counting facility where the axial activity profile of each copper wire is obtained on a dedicated counting device. This data is then input to the SAFI-2000 code and used to determine a second set of fuel-clad temperatures, based on the assumption that the relative thermal neutron-copper reaction rates in the core are equivalent to the relative thermal fission rates.

5.5 Evaluation of Theoretical and Experimental Results

A comparison between the theoretically and experimentally determined fuel clad temperatures and various other parameters provides confidence that there were no errors in either the analysis or the physical core loading. Having established this fact to the satisfaction of all parties, a Core Authorisation Permit is generated, providing a brief summary of the results, and signed by the responsible representatives of each division involved in the refuelling process. This document is the final addition to the set of quality records for the cycle and is required by the reactor start-up procedure before the reactor can be taken beyond critical to power operation.

6. Start Up

The start-up starts early on the Wednesday afternoon so that the reactor will be at the designated power by 16:00. During the start-up procedure there are several hold points (e.g. at critical, at first thermal balance, at 50% of full power and at full power) during which visual core inspections are carried out. Visual access to the core for this purpose is provided through a Perspex porthole in the reactor vessel top cover. The visual inspections ensure that:

- No tools or other objects were inadvertently left inside the reactor vessel when the top cover was replaced;
- There are no obstructions to coolant flow through the fuel assemblies;
- No fuel assemblies were damaged during handling (sometimes evident by a noticeable distortion of the top edges of some of the fuel plates);
- No unexpected behaviour is evident (e.g. fuel assemblies or other devices moving or vibrating excessively);
- All the fuel assemblies appear to be fully seated in their positions and all the control rods appear to be withdrawn to the same height and as indicated in the control room.

7. Conclusion

Refuelling of the SAFARI-1 core is achieved in an exceptionally short time, so that the availability of the reactor is maximised for commercial isotope production. Each step in the refuelling process is characterised by checks and balances, under strict quality management, to ensure that the highest levels of excellence in safety, fuel management and quality are maintained.

CONSEQUENCE OF USING LEU FUEL ON THE EFFECTIVENESS OF THE ABSORBER SYSTEM OF THE GERMAN FRJ-2 RESEARCH REACTOR

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ABSTRACT

Effectiveness and behavior of the absorber system are of particular significance for the optimum and safe operation of the FRJ-2 research reactor. Due to safety considerations high amount of shutdown reactivity is required at any operating state of the reactor. In case of the use of LEU fuel elements, the effectiveness of the absorber system is considerably influenced depending on loading and mass of U 235, respectively. The effect of the core loading on the behavior of the shutdown system of FRJ-2 was investigated in the framework of the core conversion study using the coupled code system MCNP and BURN. The analysis revealed that in the case of a LEU core with the same thermal power, the loading of the fissile uranium has to be increased by 27 %. Due to the high amount of U-238, the thermal flux in the LEU core is reduced and the neutron spectrum becomes harder than in a HEU core. The consequence of changes in neutron flux and spectrum is a reduction of the effectiveness of the absorber systems by an amount of about 15 %. Parallel to the loss of effectiveness, the amount of reactivity being needed for the shutdown of the reactor as well as the excess reactivity which is required for the whole operating cycle is reduced so that significant consequences are not expected on the safety of the reactor at any operating state.

1. Introduction

The change from the HEU to the LEU core of the German FRJ-2 research reactor has consequences to the effectiveness of the shutdown system. Due to the higher amount of U-238 in the LEU core, the absorption rate in the core is considerably increased in comparison with HEU fuel elements. For the LEU core it must be assured that sufficient reactivity is available from any operating state to shut down the reactor in case of any accident.
One of the design basis accidents at FRJ-2 is the rupture of the supporting mechanism of one of the **C**oarse **C**ontrol **A**rms (CCA) during power operation. In spite of very low likelihood of such an accident, the protection system of the reactor must be capable of controlling the reactivity and safe shut down of the reactor by the residual CCAs without consideration of the second effective CCA and the **R**apid **S**hutdown **R**od**s** (RSRs).
To demonstrate that the safety system of the reactor is capable of controlling the design basis accident, the shutdown behavior of the absorber system was investigated by a sophisticated and reliable calculation method. For this aim, a detailed 3-dimensional model of the FRJ-2 was developed using the MCNP code which was coupled with the depletion code BURN, to consider the change of material composition in the fuel elements during an operating cycle due to power operation and burnup, respectively/1-4/.

2. Description of FRJ-2

The FRJ-2 is a DIDO-class tank-type research reactor cooled and moderated by heavy water. The core consists of 25 tubular MTR fuel elements arranged in five rows of 4, 6, 5, 6, and 4 fuel elements. It is accommodated within an aluminum tank 2 m in diameter and 3.2 m in height (see Fig. 1). The tank is surrounded by a graphite reflector of 0.6 m thickness enclosed within a double-walled steel tank.

The active part of the tubular fuel elements is formed by four concentric tubes having a wall thickness of 1.5 mm and a length of 0.61 m. Each tube is formed by three standard fuel plates containing fuel meat and aluminum cladding. The tubes are encased by a shroud tube of 103 mm diameter, to which they are attached by four combs at either end. To set up the new LEU core of the FRJ-2, the HEU elements with an initial mass of 150 g and 170 g (U-235) are replaced by 200 g fuel elements with an enrichment of 19.75 %. The amount of U-235 in the LEU core is 27 % higher than in the HEU core. Table 1 gives an overview over the characteristics of the corresponding HEU and LEU fuel elements.

The FRJ-2 has two independent and diverse shutdown systems the CCAs and the RSRs. The CCAs are raised during the power operation in order to compensate the reactivity loss due to burnup, whereas the RSRs are permanently in their upper position.

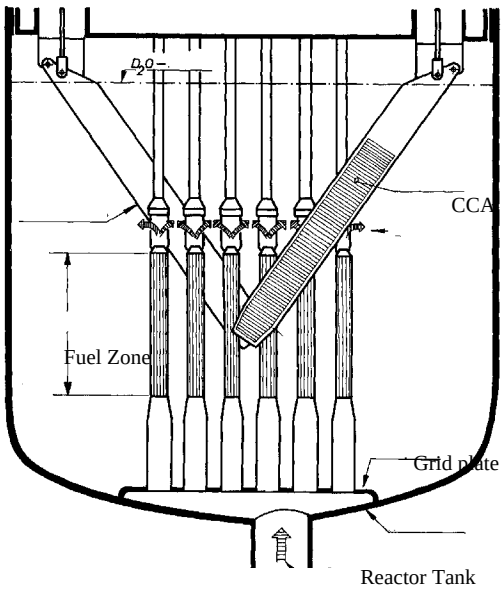

Fig. 1: Vertical Cross section of the FRJ-2 tank

Fuel type	HEU	LEU
Fuel specification	UAl_x	U_3Si_2-Al
U-235 mass (g)	150/170	200
Density (g/cm^3)	0.50/0.57	2.90
Enrichment (% w)	80	19,75

Table 1: Main characteristic data of the HEU and LEU fuel elements

In case of demand, the six CCAs are released from their electromagnets and drop into the shutdown position by gravity, whereas the three RSRs are shot in by pneumatic actuators.

The six CCAs are raised and lowered by angular movement around a pivot expressed in angular degrees. A failure of the supporting mechanism would allow an absorber arm to fall down through the core in a non-effective vertical position causing an accident with the insertion of a high amount of reactivity. Such an accident is identified by the protection system and the reactor is shut down by dropping the residual CCAs after being released by different trip signals.

3. Modeling of the FRJ-2

The MCNP model of FRJ-2 is a complete 3-dimensional full-scale model with a very high level of geometric fidelity. It comprises the reactor core, CCAs, RSRs, core structure, beam tubes, the graphite reflector and the biological shield. The core region consisting of 25 fuel elements was modeled as a cylinder containing square lattice with an array of cells representing the individual fuel elements, part of the absorber arms and cooling gaps. For the detailed modeling each individual fuel element was divided into 15 axial, 35 radial and 6 azimuthal material zones. Due the continuous change of the material composition in the fuel meat resulting from the fuel consumption, it was necessary to couple the MCNP code with a depletion code. In this way the variation of the neutronic state of the core could be simulated by multiple linked burnup and MCNP calculations. The detailed segmentation of the core and the surroundings for the MCNP and depletion calculation resulted in a model with 11250 material zones /2/. Due to differences in the material composition of HEU and LEU fuel elements many simulation steps were performed to set up a typical LEU core. For this aim 5 operation cycles each 21 days were calculated. At the end of each cycle 5 spent fuel elements with the maximum burnup were replaced by LEU fuel elements. The burnup distribution of the fuel elements for the first LEU core at the **B**egin **o**f **C**ycle (BOC) is given in Fig. 2. Accordingly the mass of U-235 in the LEU core is 27 % higher than in the HEU core.

4. Results of Simulations

The use of LEU fuel results in a considerable change of the neutron-physical and safety related parameters of the reactor, respectively. To analyze the behavior of the LEU core, the neutron spectrum of the HEU and the LEU case were determined. For this aim, the neutron flux was calculated as a function of the neutron energy ranging from 0 to 20 MeV. The result of the calculation has been depicted in Fig. 3. Accordingly the LEU core shows a distinguished reduction in the thermal range and an increase in the epithermal groups.

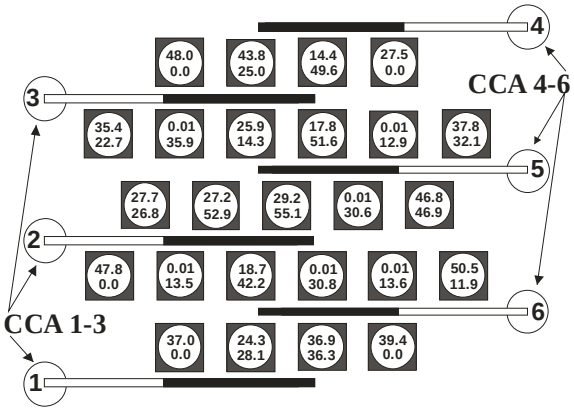

Fig. 2: Burnup distribution for the HEU core (upper row) and for the LEU core (lower row)

The reason for this behavior in the LEU case is the high amount of uranium isotopes causing an increase of neutron absorption due to high amount of U-238. The distribution shows a reduction of 15 % for the thermal range and an increase in the same amount for higher neutron energies, representing a hard neutron spectrum for the LEU core.

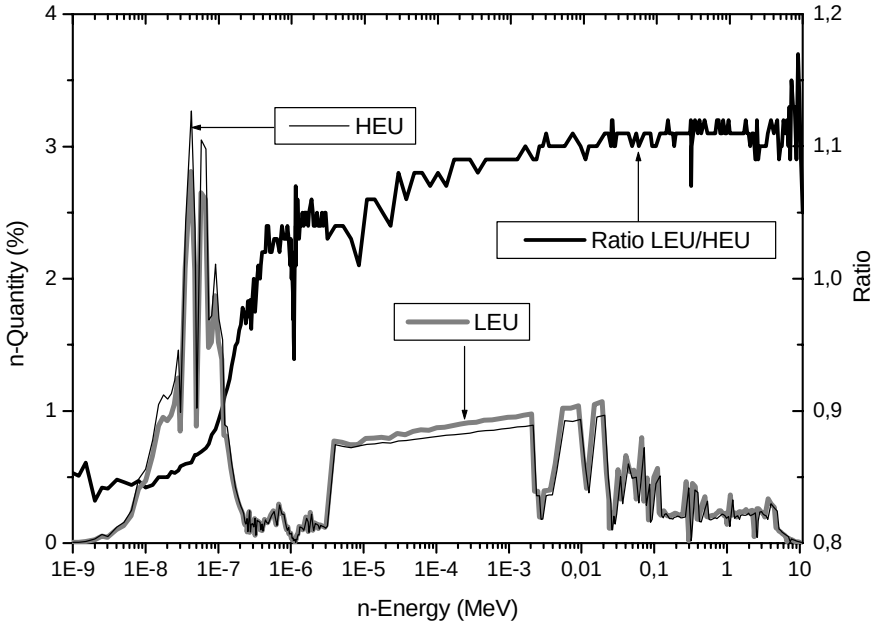

Fig. 3: Neutron spectra for the HEU and the LEU core in comparison

As a consequence of the reduced neutron absorption rate in the absorber materials the effectiveness of the shutdown system is influenced in a considerable amount. The effect was studied for the HEU and LEU core by calculating the effectiveness of the CCAs as a function of the CCA angle. For this aim the multiplication factor (k_{eff}) was calculated for different angles of the CCAs which result during an operating cycle.

The effectiveness was determined as a difference of k_{eff} between the operating angle and the shutdown position (0°). The results are given in Fig. 4. Accordingly the effectiveness of the shutdown system is reduced by 2.81 % dk/k corresponding to a loss of 15 %. The consequence is a reduction of shutdown and excess reactivity needed at BOC for the whole length of the operating cycle. At BOC sufficient

shutdown reactivity is needed in the case of a reactivity accident due to the loss of the most effective CCA. The CCA-2 is the most effective one with the highest weighing factor of 1.38 in the case of a total effectiveness of 20.88 % dk/k (Fig. 2, HEU). As a further safety requirement it is assumed that in the case of such a reactivity accident the CCA with the second-highest effectiveness is blocked at the critical operating position. For the second effective CCA the same effectiveness and weighing factor are assumed in order to consider the conservative conditions. Due to the dependency of the weighing factor on the total effectiveness of CCAs, 1.60 results for the weighing factor in the case of the LEU core. Under these conditions the four residual CCAs must be able to shut down the reactor with a sufficient level of subcriticality.

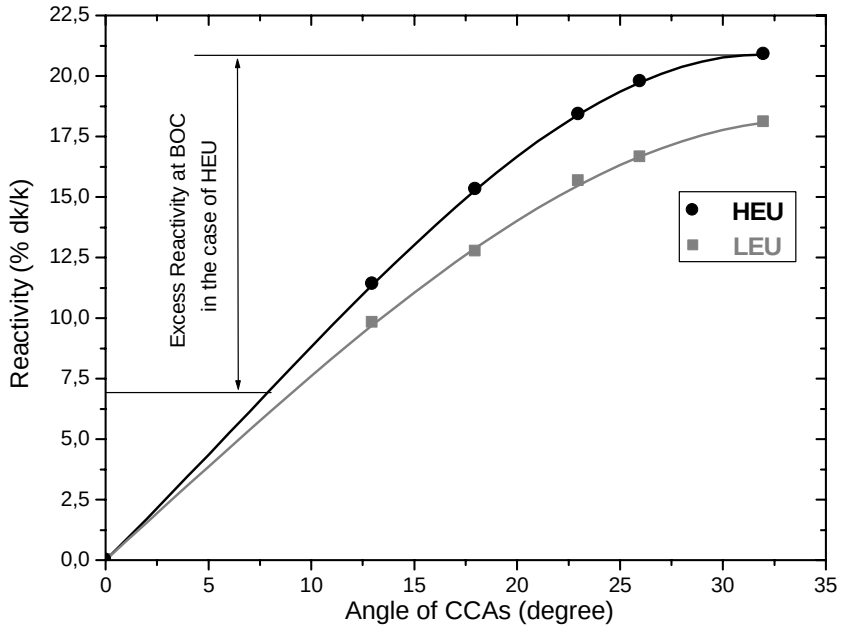

Fig. 4: Curve of effectiveness of the CCAs for a HEU core and a LEU core in comparison

According to the operating manual, a minimum reactivity of 6,90 % dk/k is required to cope with this type of reactivity accident at BOC. This value (Fig. 4) corresponds to an angle of 7.82° in case of HEU core and 9.08° for a LEU core. Under these assumptions the positive reactivity released in the course of the accident in the HEU case amounts to 3.22 % dk/k. On the other hand, the four residual CCAs will be capable of providing 3.73 % dk/k for the compensation.
In the case of the LEU core and for an initial shutdown reactivity of 6,90 % dk/k, the loss of the most effective absorber arm (CCA-2) would result in a reactivity increase by 2.98 % dk/k at BOC. By comparison, the insertion of the four residual CCAs causes a reactivity absorption by 3.22 % dk/k.
For the operating mode with a three weeks cycle, the angle of the CCAs at BOC is considerably higher than in the case discussed here, so that a criticality can be definitely excluded. In the case of demand for an extended operating cycle the loading of fuel is limited in such a way that the shutdown reactivity remains sufficient for the compensation of the positive reactivity at any stage of the operation.

5. Conclusions

The conversion of FRJ-2 to use LEU fuel results in a reduction of the effectiveness of the absorber systems including shutdown and excess reactivity. The loss of the effectiveness has however no significant influence on the safety behavior of the reactor because the fraction of excess reactivity being released in case of a reactivity transient is reduced in the same manner. Apart from the change of the

effectiveness of the CCAs, the loading of the reactor core is managed in such a way that a minimum shutdown reactivity remains available for the control of transients and safe shut down of the reactor from any operating state.

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A NEW RESEARCH REACTOR FOR AUSTRALIA

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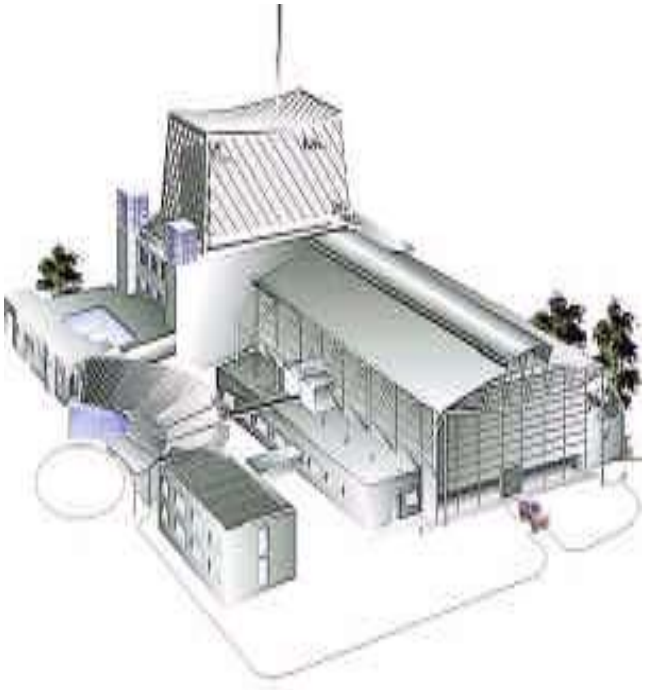
ABSTRACT

On the 3rd of September 1997, the Australian Government Minister for Science and Technology announced that a new research reactor would be built at ANSTO's Lucas Heights site near Sydney. The A$286 million (1997 dollars) facility would replace the existing High Flux Australian Reactor (HIFAR), a 10 MW DIDO type reactor that had been successfully operating since 1958.
The new state-of-the-art facility will produce the volume and comprehensive range of diagnostic and therapeutic radiopharmaceuticals needed to satisfy Australia's requirements in the coming decades. The replacement reactor will support ANSTO's nationally important work in such areas as environmental studies, agriculture and assisting industry. It will also support scientific research and higher education through improved access to a modern, versatile neutron source.



1. Brief Technical Data

20 MW thermal, open pool reactor,
Pool diameter 4.5m internal, 13m deep.
Light water moderator and coolant, heavy water reflector
Liquid Deuterium type cold neutron source
Thermal neutron source
5 control plates operated from below the reactor
First Shutdown System - inserts 5 control plates
Second Shutdown System - dumps heavy water from the Reflector vessel

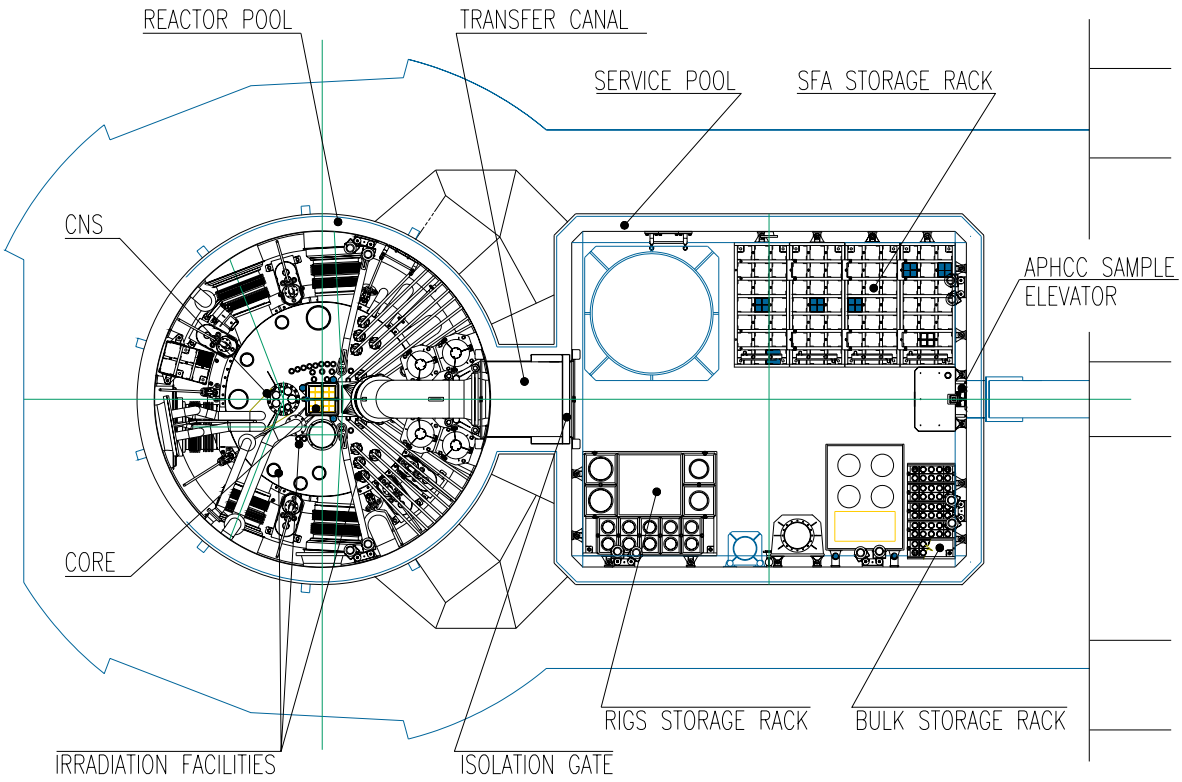
2. Projekt progress

3 September 1997	Government announcement of new reactor.
December 1998	Prequalification of reactor vendors - AECL (Canada), INVAP (Argentina), Siemens (Germany), Technicatome (France).
September 1997 - March 1999	Environmental Impact Assessment (EIS)
October 1997 - September 1999	Senate Economic References Committee Report on a New Reactor at Lucas Heights
February 1999 - August 1999	Parliamentary Joint Standing Committee on Public Works. Unanimous report recommended the construction of the reactor.
August 1999	Request for tender issued
22 September 1999	Facility licence to prepare site issued by ARPANSA
3 January 2000	Tenders closed
6 June 2000	INVAP preferred tenderer
13 July 2000	Contract signed with INVAP SE and its Australian alliance partners John Holland Construction and Engineering Pty Ltd and Evans Deakin Industries Ltd (JHEDI)
August 2000 - May 2001	Senate Select Committee for an inquiry into the Contract for a New Reactor at Lucas Heights
21 May 2001	Preliminary Safety Analysis Report (PSAR) and construction licence application to ARPANSA. 11,500 public submissions (11,000 on standard form letters). ARPANSA required answers from ANSTO to 1,159 questions.
12 November 2001	Site handed over to contractor
4 April 2002	Facility licence for construction issued (allows construction up to, but not including the loading of nuclear fuel)
May 2002	Greenpeace Australia Pacific Ltd seeks a court order declaring that the construction licence is invalid as the CEO of ARPANSA failed to take into account "international best practice in relation to radiation protection and nuclear safety in relation to the management of spent nuclear fuel and radioactive waste".
June 2002	Geological anomaly found during site excavations. The Institute of Geological and Nuclear Sciences (IGNS) of New Zealand assessing significance. Main excavation work halted.
13 September 2002	Federal court dismissed with costs the Greenpeace application
22 October 2002	ARPANSA decision to allow construction to proceed. ARPANSA subjected the results of expert geological studies, undertaken by ANSTO, to independent review by Geoscience Australia and by an international expert in seismic analysis nominated by the IAEA. It was demonstrated conclusively that the absolute minimum age of last movement of the fault is 5 million years and probably significantly older.
13 December 2002	Major concrete pour (1,000m^3) for the -5m level floor and the -7m to -5m walls. 1.5m thick slab under the reactor block.
February 2003	Concreting continues. It is expected that the 0m level slab will be completed by the end of March 2003.

It is expected that commissioning of the new reactor will begin in 2005.

3. Fuel Specification

Core dimensions	35 x35 x 61.5 cm
Number of fuel assemblies	16 (4 x 4)
Number of control rods	5
Core standard fuel load	7.7 kg U-235
Fuel element type	Plate
Overall fuel assembly dimensions	8.05 x 8.05 x 104.5 cm
Number of plates per fuel assembly	21
Initial fuel type	19.70% enriched, U_3Si_2 dispersion fuel
Uranium density	4.8 gcm^{-1}
Meat thickness	0.61 mm
Active fuel length	615 mm
Active fuel width	65 mm
Aluminium cladding thickness outer plate	0.445 mm
Aluminium cladding thickness inner plate	0.37 mm
Cladding surface temperature limit	110^{o}C
Coolant velocity in core coolant channel (internal)	8.1 ms^{-1}
Burnable poison	Cadmium wires in side plate grooves



COMPUTATIONAL ANALYSIS OF STRESSED-STRAINED STATE AND PERFORMANCE OF RESEARCH REACTOR FUEL PINS IN CASE OF FUEL DETACHMENT FROM CLADDING

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The paper presents the technique and results of the computational analysis of stressed-strained state and performance of research reactor fuel pins having detachments of fuel from cladding. As a result of computing the temperature field in a fuel pin, the fuel swelling rate and gas release from the fuel in the detachment area have being evaluated. The computations were carried out using the observed experimental data on the in-reactor creep of the aluminium-based alloy. The gas pressure has been shown to be able to cause the localization of the creep strain at the fuel-cladding detachment boundary and the possible loss of tightness by the cladding of a fuel pin.

1. Introduction

The conversion of research reactors to low-enriched fuels (not higher than 19.7% by ^{235}U) requires that the volume fraction of a fissile material in a dispersion-type fuel should be increased even in the case of employing high-density UMo fuel particles. So French fuel development engineers tested fuel pins whose volume fraction of the fissile component particles (U-9% Mo) was as high as 50% [1]. A volume fraction of the fissile component at such a high level can cause both some worsening of the adhesion between the cladding and the dispersive-type fuel while fabricating fuel pins and the possible appearance of detachments of their fuel from their cladding in the course of bringing the reactor to its full power due to a higher thermal expansion coefficient of cladding material than that of the fuel.
The objective of the present work is the analysis of the effects that the possible appearance of those detachments will have on the kinetics of the stressed-strained state (SSS) of a fuel pin and on its performance.

2. Technique of computational analysis

The object of the analytical consideration was a research reactor fuel pin whose cladding is made of an aluminium-based alloy and whose fuel is a dispersive-type composition consisting of the (U-9% Mo) alloy particles uniformly distributed throughout the aluminium matrix. The fuel pin has got a detachment between its outer cladding and its fuel kernel, see Fig. 1.
The fuel pin SSS was assumed to be determined by its temperature field, the fuel swelling, the amount of the gaseous fission products (GFPs) releasing out of the fuel under the cladding in the detachment area, and physicomechanical properties of the cladding and the fuel changed in the course of irradiation.
The computation of the fuel pin SSS was conducted by means of a finite element program complex named FEMINA [2]. The corresponding design diagram is presented in Fig. 2.

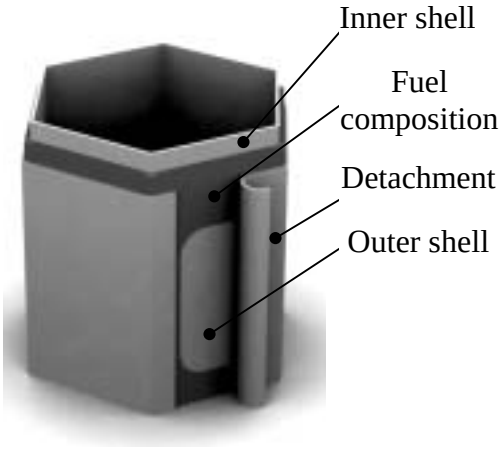

Fig. 1. Location of a detachment in a fuel pin.

3. Input data for computation

The parameters of the fuel pin that have been adopted for computations are presented in Table 1.

Table 1 Fuel pin parameters adopted for computation

Parameter, unit	Value/material
Fuel pin cladding diameter, mm: outer inner	 50 45.4
Thickness of cladding, mm	0.45
Cladding material	Aluminium-based alloy
Matrix material	Aluminium
Thickness of the dispersive fuel layer, mm	0.9
Volume fraction of fissile particles, %	40
Enrichment by ^{235}U, %	19.7
Burnup, % ^{235}U	50
Duration of operation, h	12000
Maximum temperature of fuel pin cladding, ºC	90
Maximum power density in a fuel pin, $W \cdot m^{-3}$	$3 \cdot 10^9$
Maximum flux (with E>0.1 MeV), $n \cdot cm^{-2}s^{-1}$	$2 \cdot 10^{14}$

The dimensions of the detachment have been adopted as follows: width l_1 = 15 mm; length l_2 = 50 mm, maximum radial depth h = 0.01 mm.

The effects of the irradiation dose on the yield stress of the cladding and matrix materials were assumed in accordance with the results of Ref. [3].

The in-reactor creep rate of the aluminium-based alloy was adopted in the form presented in [4], see Fig. 3.

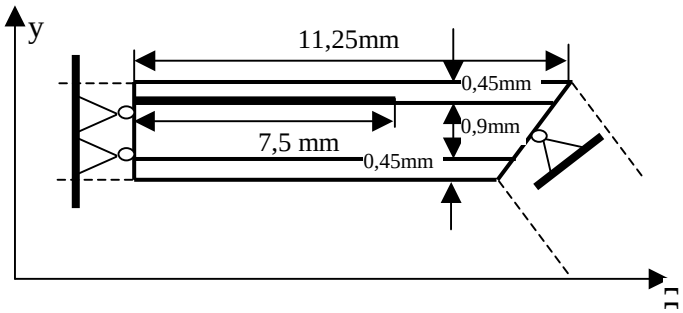

Fig. 2. Design diagram for computation.

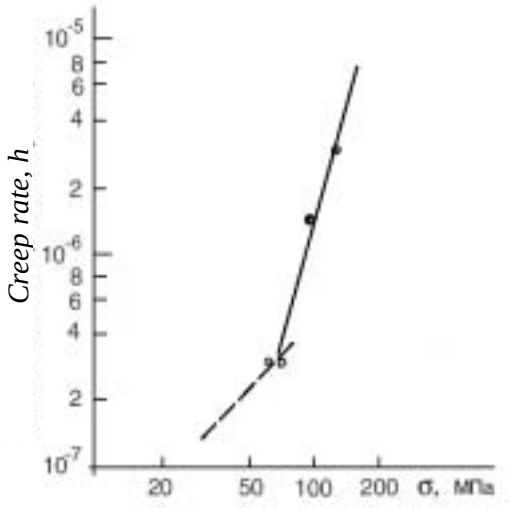

Fig. 3. "In-reactor creep rate – stress" dependence in the 80-90°C temperature range for the aluminium-based alloy SAV-1 (Mg+Si+Al < 5%).

The swelling, S_i^{max}, of (U-9% Mo) particles was adopted, according to [5], to be of ≈ 3.54% at a burnup of 1% h.a. The pressure of the fission gases releasing out of the fuel under the cladding in the detachment area, $P(\tau)$, was computed in accordance with a relationship as follows:

$$P(\tau) = (0.1\ T_V/273) \times [(V_i \cdot \tau)/(V_0 + \Delta V_{cd} + \Delta V_{fs} + \Delta V_i)] - P_c\ , \text{MPa},$$

where T_V is the temperature in the detachment area, K;

ΔV_{cd} – the volume change of the detachment in the course of the creep of cladding under the action of gas fission products (FGPs);

ΔV_{fs} – the volume change of the detachment due to the fuel swelling;

ΔV_i – the volumes of individual fission gas products having released into the detachment area;

P_T – the pressure of the coolant.

While computing, it was assumed that the release of gaseous fission products is determined by the kinetic energies of their direct escape out of a 0.01 mm thick layer of the fuel. As the gaseous fission products whose V_i values were to be taken into account, stable isotopes of Xe and Kr have been considered.

The computation has shown that in the end of fuel pin life, under the assumptions made, the resultant volume of these GFPs, V_i, equalled ≈ 16.7 mm^3 at NTP.

4. Analysis of computation results

The results of the computation are presented in Figs. 4 – 7.

The results of computing temperatures in a fuel pin with a fuel-cladding detachment are shown in Fig. 4. The presence of the detachment caused in the fuel an increase of temperature up to ≈ 148°C. The temperature change in the detachment area was insignificant (the cladding temperature in the region of the detachment boundary, T_{clad}, remained equal to ≈ 92°C).

The results of computing the thermal stresses at the detachment boundary at the beginning of fuel pin operation are presented in Fig. 5. The maximum level of thermal stresses, σ_{max}, is equal to ≈ 60 MPa.

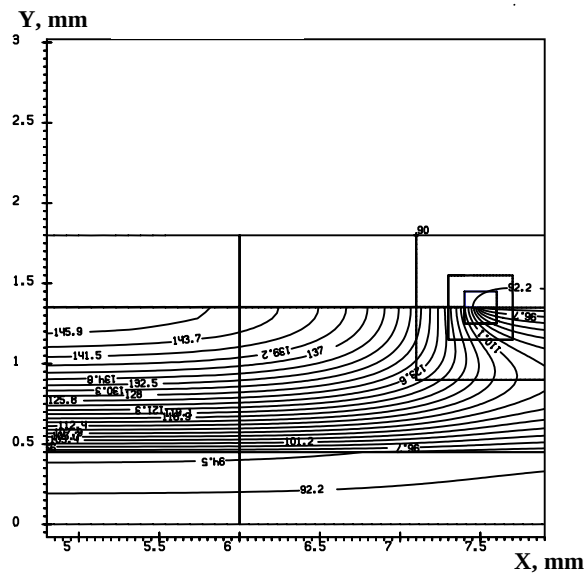

Fig. 4. Temperature field of a fuel pin in the detachment area.

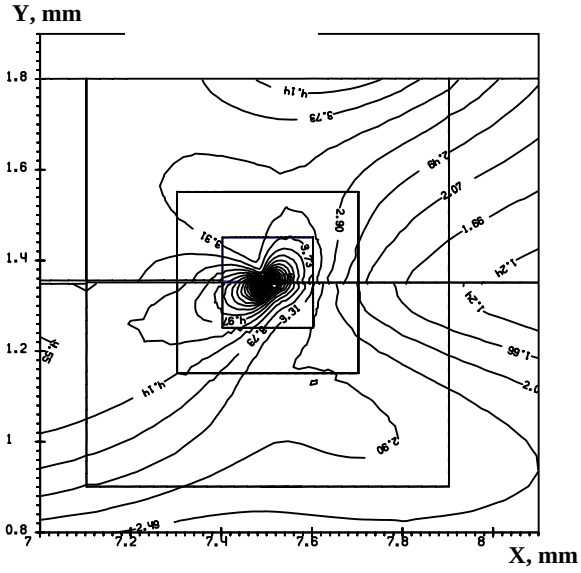

Fig. 5. Thermal stresses in the detachment area (stresses are expressed in kg·mm^{-2}).

The results of computing the stresses in the boundary region of the detachment caused by the temperature field and fuel swelling taking into account the creep of cladding and fuel are presented in Fig. 6. As can be seen in this figure, the thermal stresses have relaxed under creep conditions, and the fuel swelling has created the stresses in cladding at a level of σ ≈ 55 MPa. This level of stresses is ensured by the presence of the equilibrium between the rate of fuel swelling and the creep rate of the cladding material (an aluminium-based alloy). The stresses caused by the swelling of the fuel are far from the yield stress of the cladding material, but they can become summarised with those caused by the pressure of gaseous fission products.

Fig. 7 shows the results of stress computing in the area of the detachment caused only by the action of the gas pressure at the end of fuel pin life. As can be seen, the maximum stress at the detachment boundary reaches approximately 70 MPa.

Thus, an overall level of stresses caused by the common action of the fuel swelling and the gas pressure at the end of fuel pin life equalled to $\sigma_u \approx 125$ MPa. As is shown in Fig. 3, the creep rate of the cladding material significantly increases at a stress to $\sigma \approx 70$ MPa and becomes then stress-dependent in a non-linear way.

As a result, practically during 2/3 of the fuel pin operation time a strain localisation is going on at the detachment boundary, which can lead to the crack formation as soon as the creep strain reaches its limiting value.

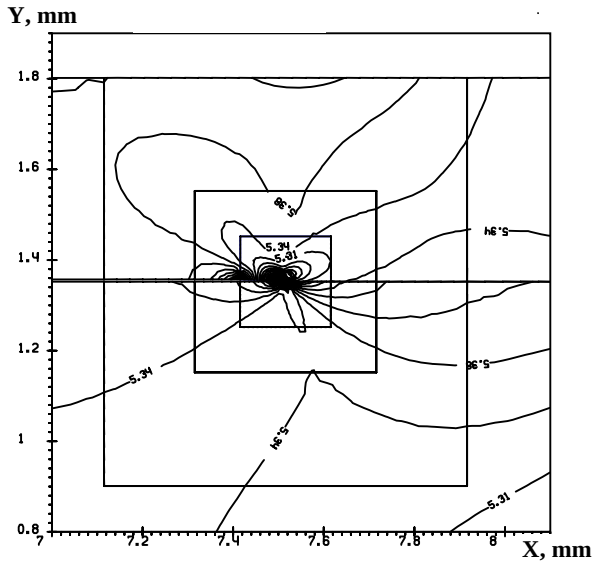

Fig.6. Stresses at the boundary of the detachment area caused by the fuel swelling (stresses are expressed in kg·mm^{-2}).

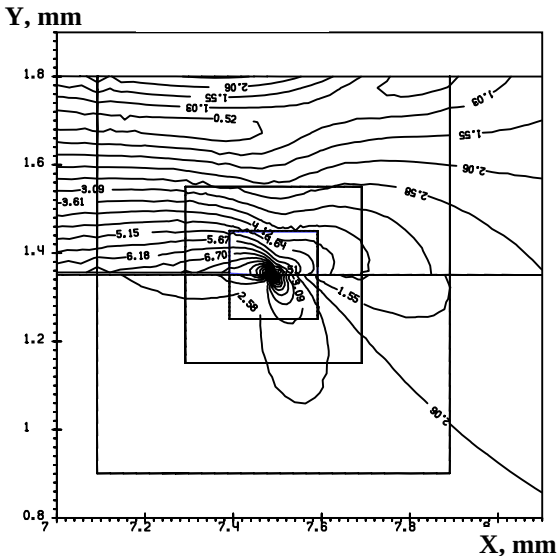

Fig.7. Stresses at the boundary of the detachment area caused by the gas pressure (stresses are expressed in kg·mm^{-2}).

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LOCALIZED FAST NEUTRON FLUX ENHANCEMENT FOR DAMAGE EXPERIMENTS IN A RESEARCH REACTOR

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ABSTRACT

In irradiation experiments on materials in the core of the OSIRIS reactor (CEA-Saclay) we seek to increase damage in irradiated samples and to reduce the duration of their stay in the core. Damage is essentially caused by fast neutrons (E>1MeV); we have therefore pursued the possibility of a localized increase of their level in an irradiation experiment by using a flux converter device. We have studied several parameters that are influential in the increase of fast neutron flux within the converter, such as dimensions, geometry and nature of the fuel. We have also considered the problem of the converter's cooling in the core and its effect on the operation of the reactor. Experimental validation of the flux calculation scheme was carried out in the ISIS reactor, the mock-up of OSIRIS, by optimizing the loading of fuel elements in the core.

1. Introduction

The materials used in the reactors of nuclear power plants are subject to mechanical and thermodynamic strains, causing their ageing over the course of years. In addition, neutron irradiation induces changes in the structure of the materials and accelerates the degradation of their mechanical properties. Studies of the behaviour of materials under irradiation aim to qualify the materials, as well as to understand, model and anticipate the process of their ageing. Such studies therefore play an active role in the safety of current facilities, in the improvement of their performance, and in the preparation of future plants. Research and materials-testing reactors allow us to expose materials within the course of a few years to an irradiation equivalent to several decades in a power reactor, thus accelerating their ageing.
OSIRIS is an MTR reactor located at the CEA Saclay. It is a 70 MW reactor of the open-core ordinary water pool type. In irradiation experiments on materials in the core of OSIRIS we seek to accelerate further the process of ageing in order to increase damage in irradiated samples and reduce the duration of their stay in the core. The rate of damage is closely tied to the level of neutron flux, notably, the fast neutrons emitted by fission (E > 1 MeV) [1]. One possible solution involves a localized modification of fast neutron flux in the core by using a converter flux device so as to increase locally the fast neutron flux in irradiation experiments. We have studied the feasibility of designing such a device in the OSIRIS reactor context.
In this paper, we will first define the specifications of a flux converter device for the OSIRIS reactor. Then, we will present the main results of neutron and thermal-hydraulic studies of the converter. Finally, we will give certain results of experimental validation of the flux calculation scheme.

2. Specifications of a flux converter for the OSIRIS reactor

OSIRIS uses a fuel made of a U_3Si_2-Al silicide alloy. A standard fuel element is composed of 22 fuel plates, each containing 0.51 mm of silicide and clad by 0.38 mm of aluminium. The cooling channel

width is 2.46 mm. The Uranium is enriched to 19.75% ^{235}U. A control element consists of two parts : the lower part is fissile; the upper part is an absorber made of hafnium.
The core of OSIRIS is a compact unit ($57 \times 57 \times 60$ cm^3). The core tank contains a centrally located rack containing 56 cells (8.74×8.74 cm^2). These cells are loaded with 38 standard fuel elements, 6 control elements and 9 beryllium elements. The 5 remaining core cells are occupied by water boxes having the same outer dimensions as the fuel elements (Fig. 1).
In the case of damage experiments in the core, samples of the materials to be irradiated are usually placed in devices called rigs (32 mm in diameter) designed to reproduce the conditions of a power reactor's operation. An experiment rig is placed in a hole (37 mm in diameter) of a water box. A water box can contain up to 4 experiment rigs (Fig. 1).

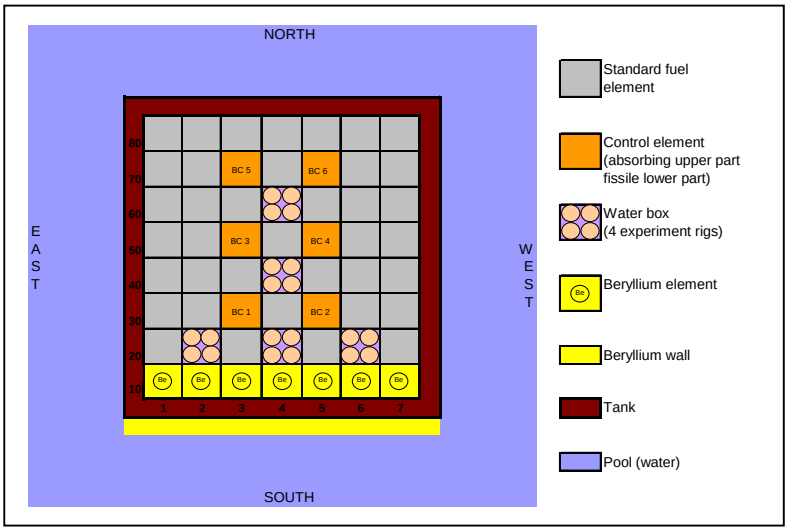

Fig. 1. Horizontal cross-section of the core of the OSIRIS reactor

To increase locally the level of fast neutrons in an experiment rig, we propose the use of a flux converter instead of the water box. A flux converter is a device made up of fissile material arranged according to a suitable geometry that allows the device to receive experiments. Thermal neutrons coming from the reactor's core cause fission in the fissile material of the converter, thereby generating fast neutrons. In addition, it is foreseen that the flux converter design should hardly increase the gamma heating level, in order to keep present rig design operable.
The flux converter must occupy a cell (8.74×8.74 cm^2) in the core rack and must also be able to receive at least one experiment rig (32 mm in diameter) in a hole that is equal in diameter to the hole of a water box (37 mm). The cooling has to be assured by the primary system loop.
We define the fast flux yield (E > 1 MeV) as the ratio of fast flux in the center of the rigs irradiated in the converter to fast flux in the center of the rigs irradiated in the water box. The fast flux yield depends on the position of the converter in the core. We have chosen the experimental cell 64 (Fig. 1) as the reference point for our studies, precisely because the level of fast flux is higher in this centrally located position. We expect that the yield would be lower than in the other cells.

3. Neutron studies

We have studied several neutron parameters that are influential in the fast flux yield within the converter, such as dimensions, geometry and nature of the fuel. The main results of these studies follow.

3.1 Flux calculation scheme

For flux calculation, we used the modular 2D-transport code APOLLO2 developed by CEA [2]. This code solves the multi-group Boltzmann transport equation using collision probability or S_N methods. It also supports a self-shielding model and a predictor/corrector for isotropic depletion of the fuel. In

addition, it includes modules for critical buckling computations, transport/transport equivalence and space and energy collapses.
In order to evaluate the fast flux yield of a converter, we need to calculate the fluxes in experiment rigs within the converter and within the water box; each of these is placed in the experimental cell 64 in the core of OSIRIS and subject to the same reactor power. We describe also the total environment surrounding the experimental cell : the core, the tank and the pool. In addition, we seek to obtain the flux spectra, especially in the experiment rigs. Thus, we calculate a multi-group flux (99-group energy meshes ranging from 0 eV to 10 MeV).
We used the collision probability (or Pij) method. It solves the transport equation in its integral form. This method makes it possible to accurately describe the geometry of the converter and of the water box (fuel plates, claddings, water canals, rigs, etc…). The main disadvantage of the Pij method is the flat flux hypothesis in each calculation mesh. It is therefore essential to use a sufficient number of calculation meshes.

3.2 Flux calculation results

The following results correspond to the use of a cylindrical fuel plate converter. The width of the fuel plates and water channels are chosen to be identical to those of a standard element. In an experimental cell, we can put up to 5 cylindrical fuel plates around a rig placed in the center (Fig. 2). We used three fuels : UAl enriched to 93%, U_3Si_2 enriched to 19.75% and UMo enriched to 19.8%. Both UAl and U_3Si_2 are already used in research reactors, whereas the utilization of UMo is still under study.

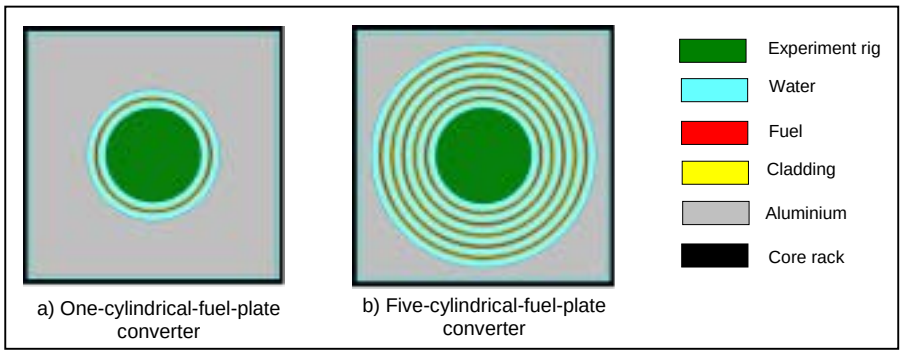

Fig. 2. Horizontal cross-sections of cylindrical fuel plate converters

The fast flux yield increases with the number of plates. In the case of the five-cylindrical-plate converter, the yield obtained is equal to 1.36 for UAl, 1.38 for U_3Si_2 and 1.45 for UMo. If we plot the yield as a function of the converter power (Fig 3.), we remark that the yields corresponding to the UAl and U_3Si_2 have similar dependencies. These fuels have nearly identical densities of the fissile isotope ^{235}U. The yield obtained by the UMo for any given value of the converter power is higher than that obtained by the other fuels. Actually, UMo is denser in ^{235}U. Therefore, for a given power, fission sources are closer to the ring than in the case of UAl or U_3Si_2.

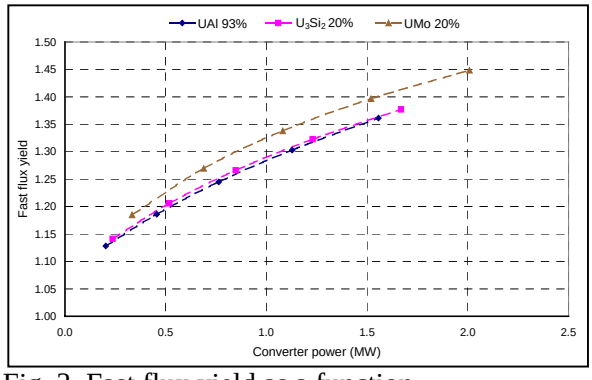

Fig. 3. Fast flux yield as a function

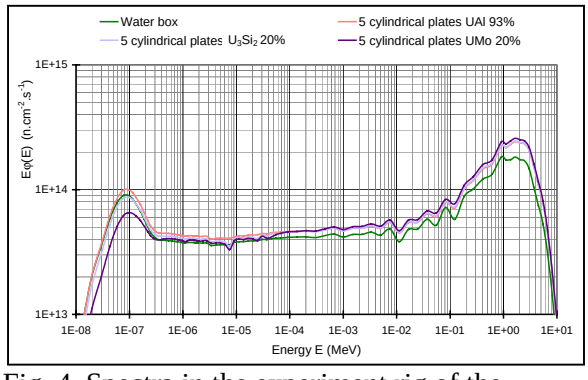

Fig. 4. Spectra in the experiment rig of the converter power

Figure 4 shows the 99-group neutron spectra in the experiment rig irradiated in the center of a five-cylindrical-plate converter. The spectrum obtained by UMo converter seems slightly harder than the spectra obtained by UAl or U_3Si_2 converters. The same effect has been established from in irradiation on UMo plate in OSIRIS.

3.3 Burn-up calculations

We have studied the evolution of the yield during irradiation in the case of the five-cylindrical-fuel-plate converter (Fig. 5). The yield decreases with the consumption of the fissile isotope ^{235}U. The rate of this decrease is more significant with a high enriched fuel (UAl). With a low enriched fuel (U_3Si_2 and UMo), plutonium produces up to 30% of the power produced by the fuel at the end of irradiation.

If we plot the fast flux yield as a function of the power of the five-cylindrical-fuel-plate converter (Fig. 6), we remark that the yields obtained by the three fuels have similarly proportional dependencies on the power.

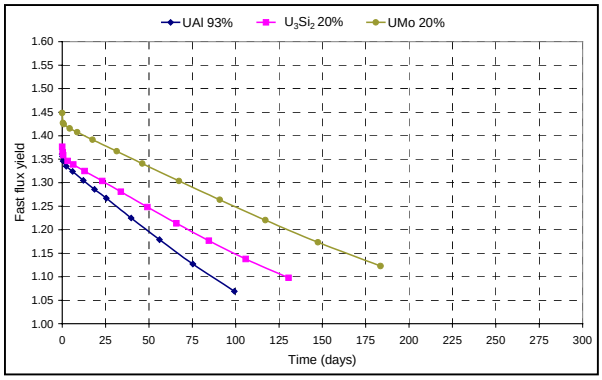

Fig. 5. Fast flux yield as a function of time

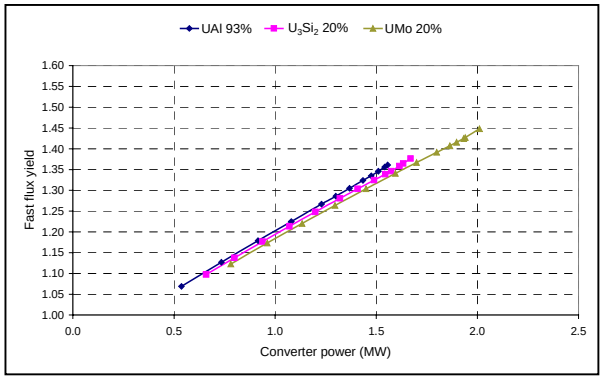

Fig. 6. Fast flux yield as a function of the power of a five-cylindrical-plate converter

4. Thermal-hydraulic studies

For the study of the converter's cooling by the primary system loop, we have used the thermal-hydraulics code FLICA3 developed by CEA [3]. This code is based on a 4-equation two-phase flow calculation with slip ratio model.

We have determined for each studied converter geometry the margins of heat flux so as to respect the criteria corresponding to nominal and safety-limit conditions of the reactor operation. For example, in the case of a five-cylindrical-plate converter, these criteria are only respected by UAl and U_3Si_2 converters. High power generating by UMo converter makes its cooling impossible by using only the primary system loop of OSIRIS. We have also remarked that heat flux margins depend strongly on the width of water channels and weakly on the type of converter geometry.

5. Experimental results

Experimental validation of the neutron calculation scheme was carried out in the ISIS reactor, the mock-up of OSIRIS, by optimizing the loading of fuel elements in the core. The objective was to surround the experimental cell 52 (Fig. 7) with fresh fuel elements in order to increase the local density of fission sources. We have carried out fast and thermal flux measurements in the four holes of a water box placed in the experimental cell 52. For fast flux measurements, we used threshold foils: $^{115}I(n,n')^{115m}I$ and $^{58}Ni(n,p)^{58}Co$. For thermal flux measurements, we used resonant foils: $^{59}Co(n,\gamma)^{60}Co$ and $^{197}Au(n,\gamma)^{198}Au$. We have compared values of measured fluxes to those of fluxes calculated by using the neutron calculation scheme described in §3.1. For fast fluxes, the ratio C/M (calculation/measurements) obtained is between 0.97 and 1.01. The ratio C/M obtained for thermal fluxes is between 1.05 and 1.15. The neutron scheme estimate the fast flux in close agreement with the measurements, but it overestimates the thermal flux.. The thermal flux is more perturbed in the core than the fast flux, especially by control rods. The axial effect is neglected in calculations flux.

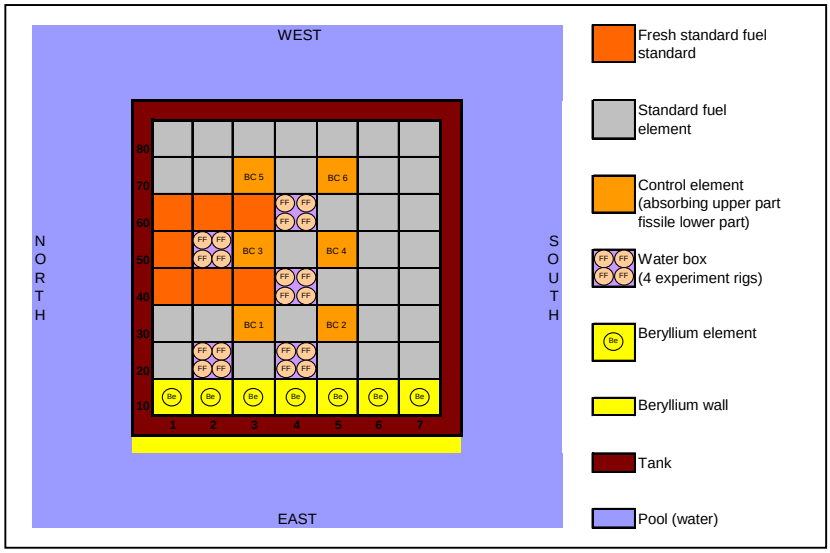

Fig. 7. Horizontal cross-section of the core of ISIS reactor with experimental loading

6. Conclusion

We have studied the feasibility of designing a flux converter in the context of OSIRIS reactor in order to increase fast flux in experiment rigs. We have carried out a neutron calculation scheme based on APOLLO2. This scheme calculates the fast flux in close agreement with the measurements.
With a five-cylindrical-fuel-plate converter, we could obtain a fast flux yield up to about 1.4. The fast flux yield depends essentially on the converter power. Therefore, the converter should be sufficiently dense in fissile isotopes (^{235}U). The use of a lowly enriched fuel, however, allow the converter to have a significant yield over a longer duration of irradiation. Plutonium produced by ^{238}U conversion compensate the consumption of ^{235}U.
Because of the relatively high converter power, the cooling of this device represents the main problem in its design. Studies have to be extended to a broader context than the OSIRIS reactor. The effects of other fuels and coolants should likewise be explored.

7. References

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FUEL PERFORMANCE EVALUATION FOR ADS MYRRHA

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ABSTRACT

MYRRHA is an Accelerator Driven System (ADS) under development at Mol in Belgium and aiming to serve as a basis for the European experimental ADS to provide protons and neutrons for various applications. It consists of a proton accelerator delivering a 350 MeV∗5 mA proton beam to a liquid Pb-Bi spallation target that in turn couples to a Pb-Bi cooled, subcritical fast core. A preliminary design of the experimental fuel rods and analysis of their performance under typical ADS conditions is a first step of the fuel development program. Different designs are to be studied and different kinds of fuels are expected to be tested. In the current design, $(U,Pu)O_2$ MOX fuel with the enrichment of 20-30 wt.% of Pu is considered to be a basis of the subcritical core, whereas different sorts of experimental fuel rods should be tested in the experimental channels to study minor actinides and low-lived fission products transmutation. In a later phase, a part of the MOX fuel in the subcritical core will be replaced with non-uranium fuels with transuranium elements. In the present report some result of modelling of the behaviour of two experimental rods (one with MOX and another with IMF) in the hottest fuel assembly of the ADS MYRRHA - are presented.

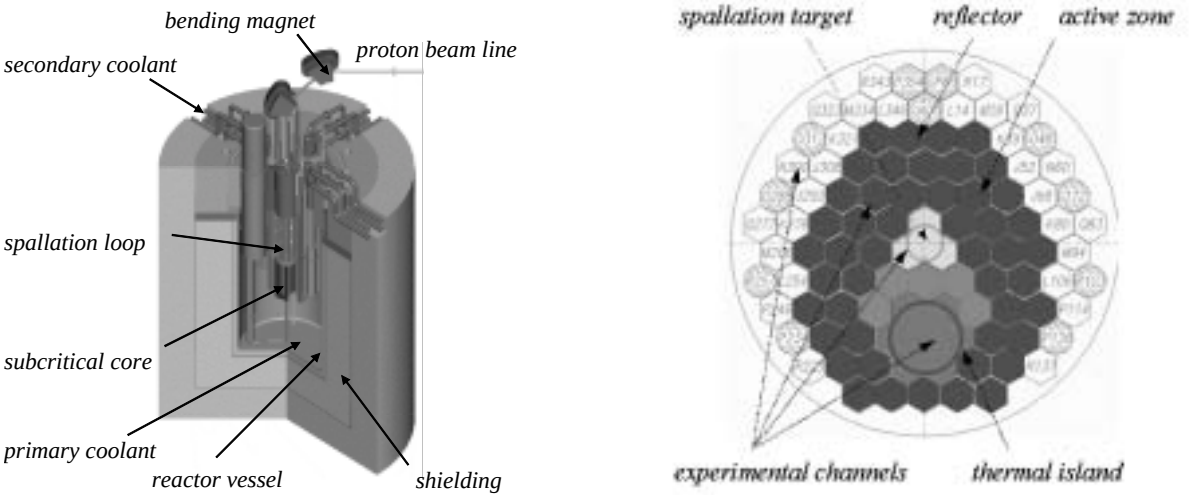
1. Introduction

The Belgian Nuclear Research Centre, SCK·CEN, is currently working on the pre-design of a multipurpose experimental ADS which is named MYRRHA (Multipurpose hYbrid Research Reactor for High-tech Application) [1]. MYRRHA can serve as the new European fast–neutron irradiation facility complement to the Jules Horowitz thermal-neutron research reactor [2]. The R&D program for the ADS fuel development is an important part of the MYRRHA project. In the current design $(U,Pu)O_2$ MOX fuel is considered as a basis of the subcritical core, whereas different sorts of experimental fuel rods will be tested in the testing channels. Some targets with minor actinides (MA) will be irradiated close to the spallation target, to study the MA-incineration in the ADS. The fuel of commercial nuclear reactors could be tested in the reflector zone and thermal neutron traps. In a later phase, a part of the MOX fuel in the subcritical core could be replaced with non-uranium and MA containing fuels. A preliminary design of the experimental fuel rods and analysis of their performance under the typical ADS conditions is a first step of the fuel development program. In the present paper two variants of the fuel rod design are considered and the results of simulation of their behaviour during operation are summarised. The analysis was performed by using a thermo-mechanical fuel performance code MACROS, which is under further development and validation at SCK·CEN [3]. The irradiation conditions of the hottest fuel assembly of the ADS MYRRHA were used in calculations. Thermomechanical behaviour of the fuel rods was analysed during three irradiation cycles of 90 EFPD with shut-down periods of 30 days as presently foreseen for the MYRRHA facility maintenance.

2. MYRRHA - a multipurpose experimental ADS

The applications considered presently in the MYRRHA ADS are focused primarily on the ADS concept demonstration, studies of MA and LLFP transmutation in representative ADS conditions, radioisotope production, material and fuel studies and safety research on sub-critical systems. The MYRRHA concept, as it is today, is based on the coupling of a 350 MeV*5 mA proton accelerator with a liquid Pb-Bi spallation target surrounded by a subcritical neutron-multiplying medium. The protons hit

the spallation target and produce the primary neutrons needed to sustain the chain reaction in the subcritical core-blanket surrounding the spallation target (Fig. 1). Maximising the fast neutron flux and minimising the core power at the fixed neutron multiplication factor of $k_{eff} \sim 0.95$ were chosen as the primary performance objectives in the MYRRHA core design. After neutron modelling and optimisation of some design parameters the total thermal power level of about 40 MW and the peak fast neutron of $\sim 10^{15}$ n cm^{-1} s^{-1} were reached in pre-design calculations. In the framework of the MYRRHA project, the pre-design stage is currently well advanced and the associated R&D programs, to assess the most risky points of the present design, are running since 1998.



(a) (b)

Fig. 1. General schematic view and a special core configuration of the MYRRHA ADS.

To meet the aimed research goals and to assure a large experimental flexibility, four different neutron spectrum zones – thermal, resonant, fast and quasi-spallation – were foreseen in the different configurations of the current core design. In the basic design, the central annular fast spectrum zone is composed of hexagonal fuel assemblies of fuel pins with $(Pu,U)O_2$ MOX containing 20-30 wt.% of the plutonium dioxide. This zone is very suitable for reactor material testing and transmutation studies due to the high fast (E_n>0.75 MeV) neutron flux ($\sim 10^{15}$ $cm^{-2}s^{-1}$) attainable there. The thermal spectrum zone (called "thermal island") represents an inserted in-pile section at the core periphery and can be realised by using a special core configuration (Fig 1b). This thermal island will enable LLFP transmutation experiments, irradiation experiments with fuel of LWR, the production of radioisotopes. Whereas $(Pu,U)O_2$ MOX fuel is expected to be used as driver fuel in the subcritical core at the first stage of the MYRRHA operation, the different sorts of experimental fuel rods containing TRU and LLFP will be tested in the experimental channels disposed nearby the spallation target, in the fast core, in the thermal island and in the reflector. At a later stage, a part of the driver fuel in the core can be replaced with fuel containing TRU.

3. Fuel pin design

Two important requirements in fuel pin design are fuel non-melting and non-damage of the cladding by inner or outer pressures and thermo-mechanical loads during the total fuel life [4,5]. Given the maximum desired power density in the core, the first criterion determines the pellet dimensions. The second criterion determines the clad diameter and thickness. Taking into account that peak power density in the MYRRHA fuel (~1.3 kW cm^{-3}) is close to that in SuperPhenix (SPX) [5], we started design with the SPX type fuel rods, except that a low-swelling martensitic stainless steel T91 was chosen as the cladding material, which has good mechanical parameters and corrosion resistance in the liquid Pb-Bi environment [6]. Although SPX type fuel has been chosen as starting point in the preliminary fuel design, continuous performance assessments are being performed to optimise all parameters of the fuel pin (fuel type, pellet density and dimensions, cladding diameter and thickness, gas plenum dimensions, etc.), aiming at better performances in normal operation and resistance to off-

normal transients. Taking into account that thermal conductivity of the advanced non-uranium fuels containing MA is usually lower than that of MOX, thinner pellets have been chosen for experimental fuel rods. The axial sizes were kept the same as in the reference design of the driver fuel rod in order to facilitate its integration in the basic fuel assembly. Axial schematic of the experimental fuel rod is presented in Fig. 2. Two experimental fuel rods were analysed: one with the same fuel composition as the driver fuel (i.e. $Pu_{0.3}U_{0.7}O_{2-x}$) named below MOX and another with the inert matrix oxide fuel containing plutonium and americium ($Am_{0.25}Pu_{0.25}Zr_{0.4}Y_{0.1}O_{2-x}$) named IMF.

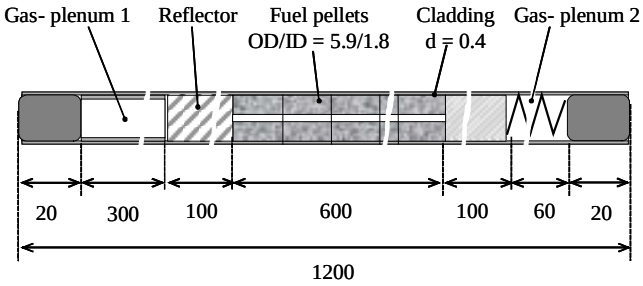


Fig. 2. Axial schematic of an experimental fuel rod

4. Results of modelling

The modelling was performed by using the fuel performance code MACROS [3], which is under further development and validation at the SCK·CEN. The data base of the material properties was modified to include new materials, the lead-bismuth eutectic coolant, and thermo-hydraulic models in order to simulate in-pile conditions of the MYRRHA ADS. In view of the important role that helium may play during a long-term irradiation or storage of spent fuels, the burnup model has also been modified to include the production of helium due to alpha decay of actinides.

The typical irradiation conditions of the hottest fuel assembly of the ADS MYRRHA were used in calculations. It was assumed that the total neutron flux in this position is kept about constant. The initial axial distribution of power in the rods was calculated with MCNPX code [7].

The thermomechanical behaviour of the two experimental fuel rods - MOX and IMF - have been analysed during three irradiation cycles of 90 EFPD with shut-down periods of 30 days in between.

Figure 3a illustrates evolution of the peak linear power. Its values are very close (350-380 W cm^{-1}) in the both rods during all operation period and show a tendency to a slow decrease with time, mainly due to plutonium burnup. After three operation cycles this reduction reaches about 7 %.

The central and surface temperatures of the peak pellets in function of operation time are presented in Fig. 3b. The levels of the central temperatures are rather high: 1750-1950 K for MOX and 1900-2350 K for IMF. The highest temperature (2350 K) is observed in the IMF fuel at start, because of a relatively low thermal conductivity of the yttria stabilised zirconia matrix. The IMF maximum temperature decreases with irradiation due to decreasing the pellet surface temperature related to the pellet-cladding gap reduction. The centre-surface temperature difference in the IMF pellet remains about constant during operation, indicating that irradiation defects give only a small contribution to the initial thermal resistance of this fuel. The radial temperature difference in the MOX peak pellet increases with irradiation and compensate the effect of the gap closing. As a result, it's central temperature changes slowly during the first and second irradiation cycles. In the third cycle, where the pellet surface temperature is about constant, it increases and becomes even slightly higher than the central temperature in the IMF pellet. The pellet surface temperatures are about the same in the both rods at the start, but decreases more rapidly with irradiation in IMF, mainly because of a more rapid closing of the gap (Fig. 3c), caused by a higher initial swelling rate (Fig. 3d). It is interesting to note that the pellet-cladding gap remains still open in the both rods after 270 EFPD of operation indicating the absence of the fuel cladding mechanical interaction and possibility to extend the operation period (however, in the IMF rod it is close to zero by the end of the third cycle).

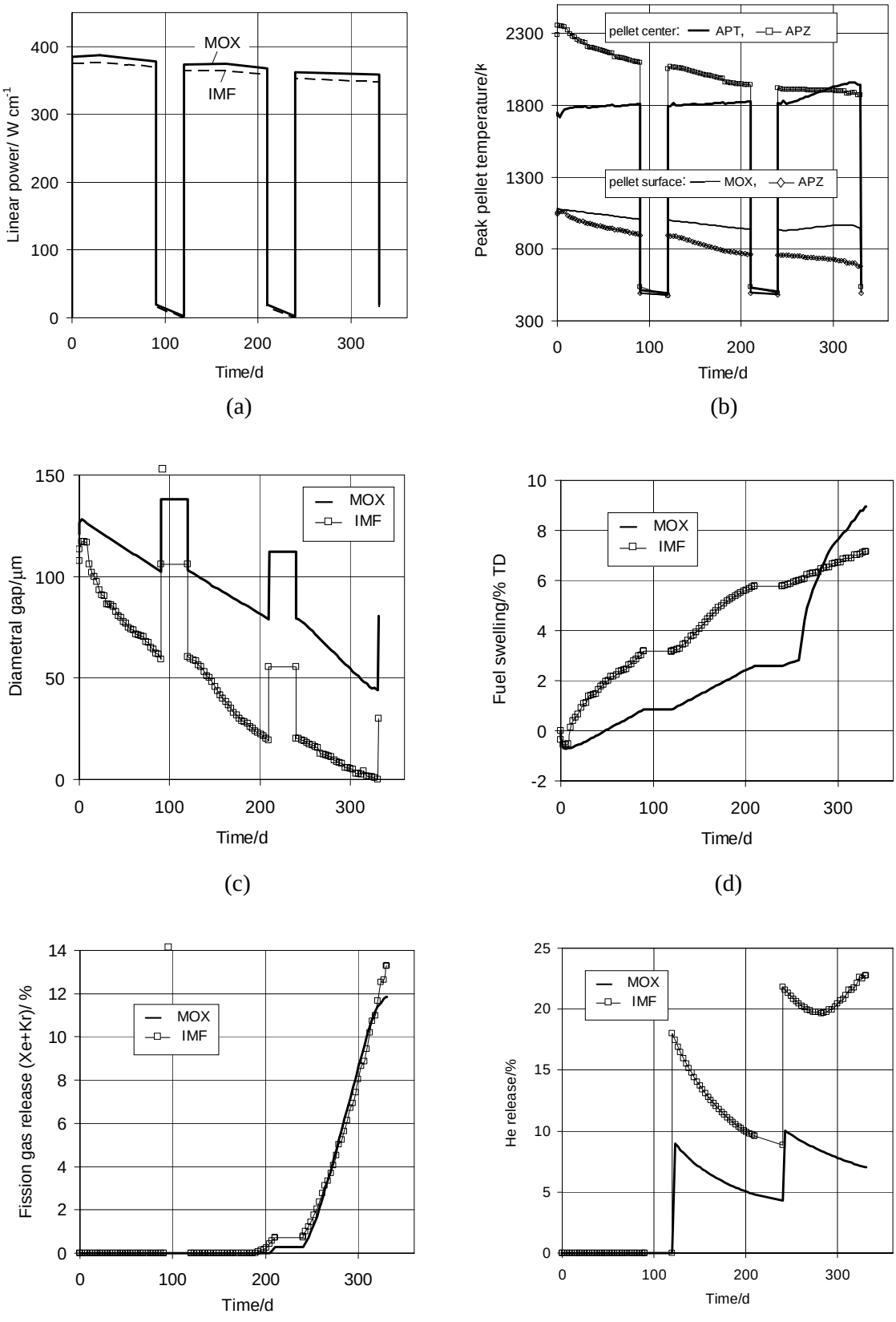


Fig. 3. Linear power (a), central and surface temperature (b), fuel swelling (c), pellet-cladding gap evolution (d), release of Xe+Kr (e) and release of He (f) in the peak pellets of the MYRRHA experimental rods with MOX and IMF.

The calculation show that amount of helium produced alpha decay of some nuclides of curium and americium is very important due in the IMF. In one hour after the reactor start it is already about 8 % of the summary production of xenon, krypton and helium; at the end of the third cycle, more than half of the produced noble gases consists of helium. At the beginning of the reactor operation the generated helium, krypton, and xenon diffuse and are captured by the intra-granular and inter-granular traps (grain boundaries, micro-bubbles and pores) which are still not saturated. They produce an increase in the internal pellet stress which, in its turn, causes swelling. The swelling rate in IMF fuel decreases in the third cycle when intensive release of the trapped Xe and Kr atoms starts (Fig. 3e) despite of a slight temperature decrease. The effective release of helium starts at the beginning of the second cycle (Fig. 3f). At this time the produced helium fraction is about 40-45 %, and the helium traps are almost saturated with helium generated before the reactor start and during the first cycle followed by the maintenance period. A rapid temperature increase at the second cycle start-up causes a peak release of helium that was accumulated during the "cold" period (Fig. 3f). The same behaviour is also observed at the beginning of the third cycle. At the end of the third cycle, more than 20 % of the produced helium is released from IMF and about 7 % from MOX fuels. Pressure in the rods plenums follows the gas release evolution and attains 1.5 MPa in the reference MOX rod and 2.2 MPa in the IMF rod at the end of the third cycle.

5. Conclusions

The preliminary modelling of the thermomechanical behaviour of two experimental fuel rods – one with 30 % MOX and another containing americium and plutonium in zirconia inert matrix – were performed under typical irradiation conditions of the research ADS MYRRHA with the new fuel performance code MACROS allowing to analyse various oxide fuel compounds under a large range of irradiation conditions.

The first results showed that both rods can survive without fuel melting or cladding damage during at least 270 EFPD of irradiation in the hottest channel of the MYRRHA subcritical core and suggest that irradiation can be extended.

It has also been revealed that helium production in fuels containing transuranium elements can contribute significantly in their irradiation behaviour and internal pressure build-up.

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MCNP/HELIOS SYSTEM FOR THE HANARO RESEARCH REACTOR USING THE TABLE LOOKUP METHOD

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ABSTRACT

For the detailed neutronic analysis of HANARO, a new Monte Carlo depletion system, the MCNP/HELIOS system, is set up. Static calculation is done by MCNP, and the table lookup method is employed for the depletion calculation. The table required in this system has the burnup dependent number densities of all isotopes considered, which are prepared by a state-of-the-art lattice code, HELIOS.
The table lookup method for the Monte Carlo depletion system is validated for two dimensional calculation of the HANARO fuel assembly, and the calculated eigenvalues and power distributions as a function of burnup show good agreement with the standards of verification.
The old MCNP model of HANARO is revised for the new MCNP/HELIOS system. In the new model a fuel rod is axially segmented every 5 cm, then the total 13,104 fuel segments are traced individually in the depletion calculation. The calculated results for each burnup step of the equilibrium core of HANARO are satisfactory compared with the current HANARO fuel management system.
In the case of HANARO, the calculation by the table lookup method is twice as fast as the calculation by the microscopic depletion calculation usually used in the Monte Carlo depletion calculation and the output file size of MCNP is reduced dramatically.

1. Introduction

Since the Monte Carlo method is, in principle, a virtual analog to the real world, one can obtain the exact solutions to the neutron/photon transport problems as long as one can model the desired system. Generally the Monte Carlo method requires longer computing time compared to other deterministic methods and the usage of the Monte Carlo method has been limited. Due to the rapid advances of computing power in recent decades, Monte Carlo based codes are becoming popular and applied to some burnup problems.

From the design stage of the HANARO research rector, the MCNP[1] code has been widely used in nuclear engineering and has been used to analysis the core characteristics. Since MCNP does not have a capability to do the depletion calculation, MCNP was limited to the analysis of the fresh core. The analysis at the burned core has been performed with HANAFMS (HANARO fuel management system)[2], which basically consists of WIMS/D-4[3] and Bold VENTURE[4]. The lattice code, HELIOS[5], can describe the complicated geometry, HELIOS instead of WIMS/D-4 is used for the more accurate analysis of the fuel/material irradiation targets, which usually have complex geometric shapes. Since HELIOS is a 2-dimensional code and Bold VENTURE uses diffusion theory method, MCNP calculations are still required to compensate for the geometric effects. To improve the accuracy and complicated procedure, a Monte Carlo depletion system is required.

2. Methodology

There are many Monte Carlo depletion systems, which couple the Monte Carlo code and the depletion code. MCNP and ORIGEN2[6] are the most popular constituents of the system. The first implementation of combining MCNP and ORIGEN2, known as MOCUP[7], dates back to the 1990s. Since then, other coupling programs such as MONTEBURNS[8], MCODE[9] etc have appeared. The principle function of these programs is to transfer one group cross-sections and flux values from MCNP to ORIGEN2, and then transfer the resulting material compositions from ORIGEN2 back to MCNP in a repeated, cyclic fashion. Since both codes cannot consider all nuclides, the accuracies of the systems are dependent on the nuclides considered and burnup chain adapted by the users. The eigenvalues as a function of burnup are calculated using MONTEBURNS, and Fig. 1 shows that the results from several selected nuclides are very different.

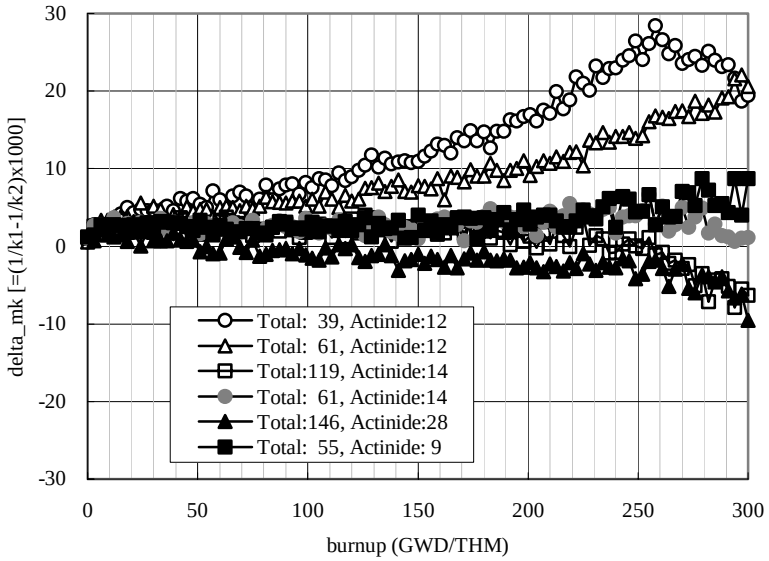

Fig. 1 Eigenvalue differencies from HELIOS

Carefully one must select nuclides within the boundary of the prepared nuclear library. It is convenient and guarantees the accuracy of the calculation to use well validated burnup chains of the conventional lattice codes such as WIMS, CASMO[10], HELIOS.

Depletion calculation can be done by microscopic or macroscopic depletion methods. Depending on the depletion methods, the efficiency of calculation is very different and the implementation of some large systems may be impossible. The Monte Carlo depletion system for HANARO uses MCNP and HELIOS. Static calculation is done by MCNP, well validated during the nuclear commissioning test of HANARO, and the table lookup method, which is a kind of macroscopic depletion calculation, is employed for the depletion calculation. The necessary tables for the macroscopic depletion calculation are prepared from HELIOS-1.6. The depletion calculation is done by a simple program, in which the total power density at each fuel segment is required instead of the cross section of each nuclide and flux at each segment. The table required in this system has the burnup dependent number densities of all the isotopes considered, which are generated by HELIOS. Total 142 nuclides for the HANARO fuel are considered in the table of each burnup step. Because HELIOS considers sufficient nuclides in the burnup chain and does not have any lumped fission products, the MCNP calculation would be more accurate than other cases such as MCNP/WIMS, MCNP/CASMO etc. for neutron and gamma transport problems. Since HELIOS can model the whole core, the number densities of isotopes are more accurate and the analysis by HELIOS can be used as the preliminary one in a very short time.

3. Calculation

3.1 A fuel assembly

To confirm the accuracy of the table lookup method, a comparative study is performed for the two dimensional HANARO fuel assembly. HELIOS is selected as a standard code for comparisons. The MCNP geometric model for the MCNP/HELIOS depletion calculation is same as that of HELIOS.

The calculated reactivity rundown curves are shown in Fig. 2. It can be found from the figure that the results agree well with each other even though the time interval is longer. The errors associated with the power distribution are investigated from 0 up to 50 %U-235 burnup level. Fuel rods in an assembly are classified into 5 categories due to their symmetrical arrangement. Fig. 3 shows the

relative errors of rod power. The power distribution also seems to agree well taking into account of a 0.5 % statistical error, therefore it is interpreted that the MCNP/HELIOS depletion calculation for the HANARO fuel assembly is adequate.

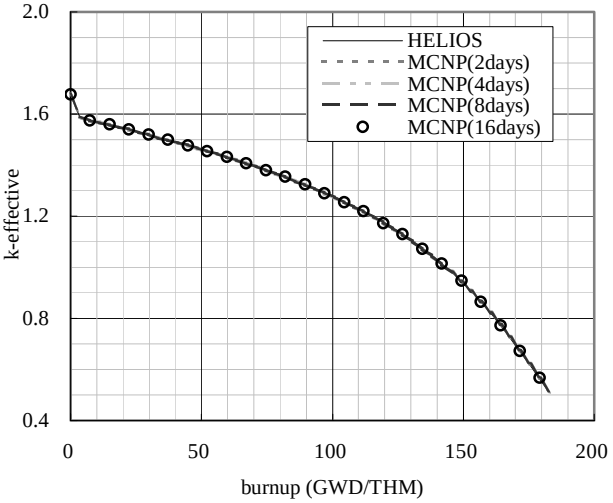

Fig. 2 Reactivity rundown curves

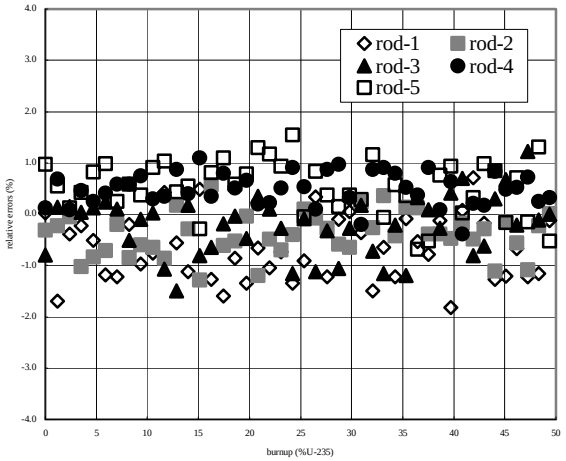

Fig. 3 Relative error distributions

3.2 The whole core

It was necessary to make a new MCNP model in order to establish the MCNP/HELIOS system. The old model used a repeated structure geometry, i.e. only one fuel assembly is modelled and then other assemblies in the core are transferred to their position by copying it using the 'LIKE m BUT' card of MCNP. However the position dependant material compositions are not allowed for in such a model. In case of fresh core the material compositions are identical for the same kind of fuel assemblies, but it is no longer true for the depleted core. Therefore the old MCNP model has to be modified so as to assign the materials separately to each position. In the new MCNP model each fuel rod is axially segmented every 5 cm for a compatibility with HANAFMS using the same geometric segmentation.

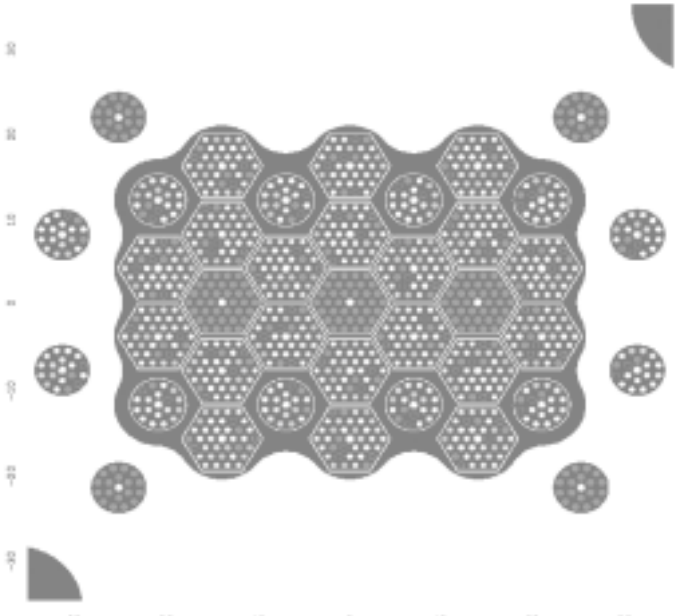

Fig. 4 Cross sectional view of the new MCNP model

For the full core of HANARO, 20 of the 36-element assemblies and 12 of the 18-element assemblies are loaded, and then a total 13,104 segments are present for the all fuel regions. Fig. 4 shows the cross sectional view of the new MCNP model for HANARO. The different grey levels in the figure indicate the different materials.

The MCNP/HELIOS system is applied to the depletion calculation of simulated equilibrium cores, and then compared with the results from HANAFMS. The fist core of the MCNP/HELIOS system is prepared using a burnup table of HANAFMS, in which the nuclei number densities calculated by the table lookup method are inserted as the material inputs of MCNP. MCNP was executed in KCODE mode with 10 million neutron sources. The reaction rate of each segment extracted from the MCNP output is used to calculate a new burnup table for the next core of the MCNP/HELIOS system. After

the first core, the burnup calculations of the MCNP/HELIOS system are progressed independent of HANAFMS. The first core of the equilibrium core (B), in which several fresh fuel assemblies are added, is prepared using a burnup table of the last core of the equilibrium core (A) and its loading plan. After that, the burnup calculations are continuously progressed.
The calculated eigenvalues of each equilibrium core is compared in Table 1. The differences between MCNP/HELIOS and HANAFMS are within 5 mk. For the equilibrium core (A) with the control rod positions of 200 mm, the statistical error from the MCNP/HELIOS calculation is 2.3~10.0 %, and RMS error is around 3.6 %. The linear power values are also compared for the same core condition, the difference between MCNP/HELIOS and HANAFMS are within 3 sigma of the MCNP statistical error. There is not any error propagation in eigenvalues and power distribution in the continuous run.

CORE (A)	calculated k-effective HANAFMS	MCNP/HELIOS	delta mk	CORE (B)	calculated k-effective HANAFMS	MCNP/HELIOS	delta mk
200 mm	0.99576	0.99603	**0.27**	200 mm	0.99596	0.99616	**0.20**
250 mm	1.01227	1.01141	**-0.84**	250 mm	0.97318	0.97863	**5.72**
300 mm	0.98843	0.99294	**4.59**	300 mm	0.98867	0.99293	**4.34**
350 mm	1.00033	0.99952	**-0.81**	350 mm	1.00055	0.99987	**-0.68**
400 mm	1.00403	1.00243	**-1.59**	400 mm	1.00421	1.00220	**-1.99**
450 mm	1.00555	1.00366	**-1.87**	450 mm	1.00567	1.00416	**-1.49**
500 mm	1.00679	1.00560	**-1.18**	500 mm	1.00686	1.00547	**-1.37**
550 mm	1.00690	1.00597	**-0.92**	550 mm	1.00690	1.00612	**-0.77**

Table 1. Comparison of eigenvalues calculated from VENTURE and MCNP

3.3 The efficiency of calculation

The table lookup method is compared with the microscopic depletion method from the aspects of execution time, necessary memories, and the size of output file. Because the calculation load is concentrated on MCNP, the comparison may be restricted to MCNP calculation. The results for a fuel assembly and the whole core are listed in Table 2. In both burnup calculations, reaction types are the same as MONTEBURNS' and the number of nuclei considered is same as HELIOS'.
The macroscopic depletion method can be executed more than 2 times faster than the microscopic depletion method, further more it makes the size of the output file much smaller.

comparison		running time (min)	memory (MB)	output size (MB)
HANARO fuel ass.	micro burnup	235.5	61	20
	macro burnup	67.15	41	12
	ratio	**3.5**	**1.5**	**1.7**
HANARO whole core	micro burnup	2177.09	701	2,122
	macro burnup	846.38	322	67
	ratio	**2.6**	**2.2**	**31.7**

Table 2. Comparison of the efficiency of calculation

4. Conclusions

The Monte Carlo depletion system is set up to give more accurate neutronic calculation in the HANARO core analysis. Static calculations are done by MCNP code, which has been widely validated in HANARO. The table lookup method for the depletion calculation is used for more efficient calculation.
To establish the MCNP/HELIOS system for HANARO, a MCNP model designed. The new model has independent fuel rods, which axially segmented every 5 cm, and a total of 13,104 fuel segments are traced individually during the depletion calculation. Since the geometric regions of MCNP and HANAFMS are the same, any core state can be simulated using a burnup table of an other system with

perfect compatibility. The calculated results of this system are in good agreement with the results from HANAFMS. Further validation of this system must be with experimental results.

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HIGH ACCURACY SENSOR FOR ONLINE MEASUREMENT OF THE FUEL ROD INTERNAL PRESSURE DURING IRRADIATION EXPERIMENTS

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ABSTRACT

A specific sensor for online measurement of the fuel rod internal pressure during irradiation experiments is developed by the Department of reactors and nuclear services of the C.E.A. (French atomic energy commission). This sensor is designed to be set up on a pre-irradiated PWR fuel rod, and to work under the heavy environments of experimental nuclear reactors. Its conception, based on the counter-pressure principle, avoids any drift phenomenon due to nuclear radiation. This sensor is qualified out-pile and in-pile in the OSIRIS reactor, located in Saclay, near Paris. The results validate the very high performances of the sensor, reaching an accuracy about ± 0.32 bar on the whole pressure range (up to 120 bar).

1. Introduction

OSIRIS is a materials testing reactor managed by the C.E.A. and located in the Nuclear Research Centre of Saclay. This open-core light water pool reactor is dedicated to experiments on nuclear fuels and materials for irradiation programs. This multi-purpose facility is designed for :

- technological development irradiation for the needs of the nuclear industry or the fundamental research,
- production of radioisotopes and doped silicon,
- activation analysis.

The 70 MW OSIRIS reactor can provide a thermal neutron flux up to 3×10^{18} $n.m^{-2}.s^{-1}$, and a fast neutron flux (energy > 0.1 MeV) up to 4.5×10^{18} $n.m^{-2}.s^{-1}$.

Experiments are either managed inside or outside the core tank. The neutron spectrum on irradiation stations is very similar to that of pressurised water reactors.

Among the experiment devices used for technological irradiations, some pressurised water loops are designed for power ramps or constant power irradiation of new or pre-irradiated PWR fuel rods.

In order to improve the knowledge of the physical parameters in these experimental devices during irradiation, specific measurement instruments are on continual studies.

Fuel rods internal pressure is one of these parameters of major interest, especially for fission gas release studies. But until now, the nuclear environment and the strong safety criteria were serious obstacles to such an online accurate pressure measurement. A specific sensor has thus been developed by the Department of reactors and nuclear services of the C.E.A., in order to take up this challenge.

2. Description of the sensor

The sensor is designed to be set up on a pre-irradiated PWR fuel rod. In order to be less intrusive as possible in the experiment irradiation devices, the sensor has the same diameter as the fuel rod (approximately 10 mm), and is roughly 250 mm long. Its design allows to work on severe irradiation environments, which means very high neutron flux and gamma radiation, high temperature (up to 350°C) and heavy pressure (up to 160 bar).

The sensor is based on the reliable and drift-less counter-pressure principle. It consists of two gas cavities, separated by a double expanding wall (see fig. 1). The first cavity communicates with the internal fuel rod pressure. The second cavity is connected to an external helium circuit, which is called “counter-pressure” circuit.

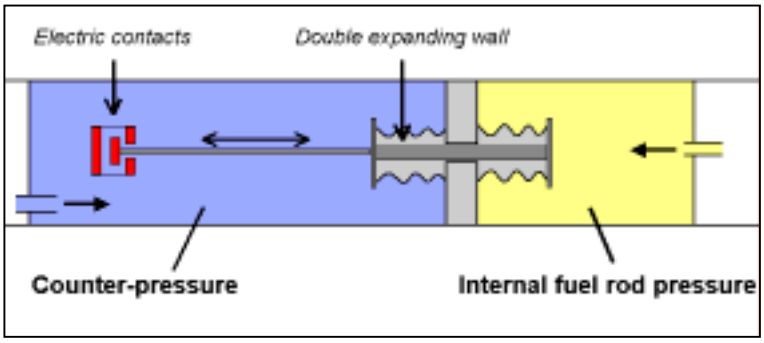

Fig1. Illustration of the counter-pressure sensor principle

The imbalance between the internal rod pressure and the counter-pressure is accurately detected by two electric contacts, activated by the motion of the expanding wall. This imbalance can be automatically compensated by inflating or deflating the counter-pressure.

This system provides a very accurate online measurement of the internal fuel rod, through the simultaneous knowledge of the imbalance detection signals on the one hand, and the direct measurement of the counter-pressure on the other hand.

The physical principle of the counter-pressure is expected to avoid any drift phenomenon due to nuclear radiation. This sensor, alike all in-pile facilities, has to deal with strong safety criteria. For example, integrity of any of its two expanding walls is continuously checked by a specific detection system.

Making transfer operations and transport easier has been a priority at each step of the design of this instrument. The lower part of the measurement system, including the two cavities and the fuel rod, is linked to the upper part of the device through a specific connector, providing both the electric and the gas transmissions. This connector is also designed to be easily handled by telemanipulators. Once it has been set up on the pre-irradiated fuel rod, the sensor can thus be linked to the upper part of the device, then inserted into the experiment device, before being transferred into its irradiation station.

3. Qualification of the sensor

This sensor is tested for its qualification, in out-of-pile and in-pile experiments. For those qualification phases, the sensor is set up on a dummy fuel rod, that is similar to a real one, except it is not filled with nuclear fuel pellet. A special gas circuit is connected to the dummy fuel rod, in order to precisely and directly adjust and monitor its internal pressure. The comparison between this online control of the internal rod pressure and the corresponding measurement given by the sensor is the key parameter of the test experiments.

After the preliminary out-of-pile tests (working in real conditions except nuclear radiation), in-pile qualifications are performed in the OSIRIS reactor, working on severe irradiation experiment conditions.

For these tests, data acquisition and processing are handled by a specific system including a real-time device. A local network transmits continuously data to the global OSIRIS database which is dedicated to experiments.

These in-pile tests are performed in the GRIFFONOS pressurised water loop experiment device.

The physical parameters in this loop are specified to be very similar to those of pressurized water reactors. This experimental device is also equipped with a dedicated motion system, which makes the pressure tube translate toward or backward the reactor core, in order to adjust the power released by a fuel rod in the experiment.

The goals of the in-pile tests are :
- to validate the good working conditions of the sensor in a real irradiation environment,
- to measure its in-pile performances,
- to detect eventual effects of nuclear radiation on the sensor signals.

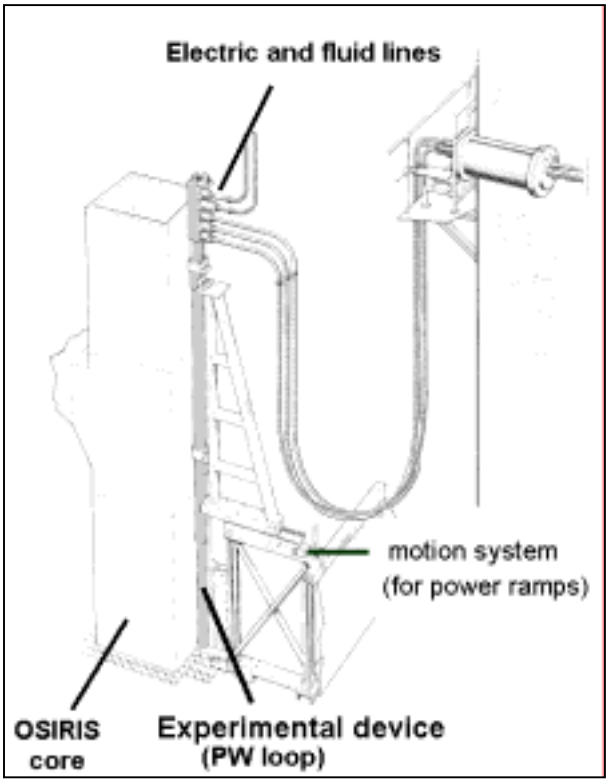

Fig 2. External view of an experimental pressurized water loop rig in the OSIRIS reactor

4. Results of the qualification tests

>From the beginning of these qualification experiments, approximately 900 measurement sets have been done, by manually adjusting the internal rod pressure within the scale from 10 bar to 120 bar.

During the first irradiation program, the thermal flux in the device was maintained mainly constant during 300 hours, at about 1.4×10^{18} $n.m^{-2}.s^{-1}$ (at the rod mid-plane). At the end of this phase, the integrated thermal fluence at the rod mid-plane was approximately 1.5×10^{24} $n.m^{-2}$.

Further in-pile experiments are carried out in order to run more parametric tests on the sensor performances.

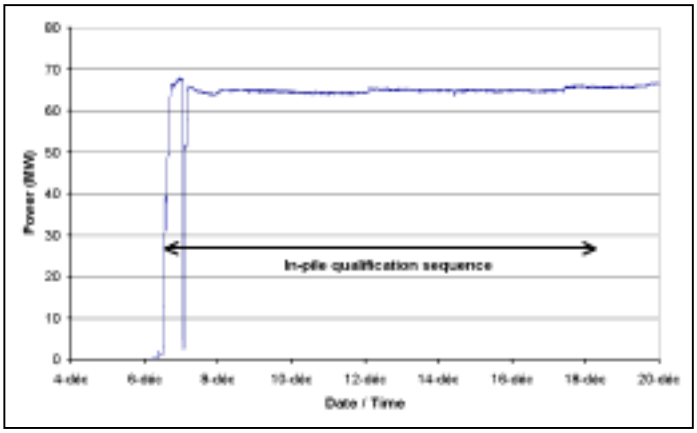

Fig 3. Power of the OSIRIS reactor during the in-pile qualification sequence

The sensor has been working without failure during the whole irradiation sequence, proving its complete capacity to work in the environment of severe irradiation experiments.

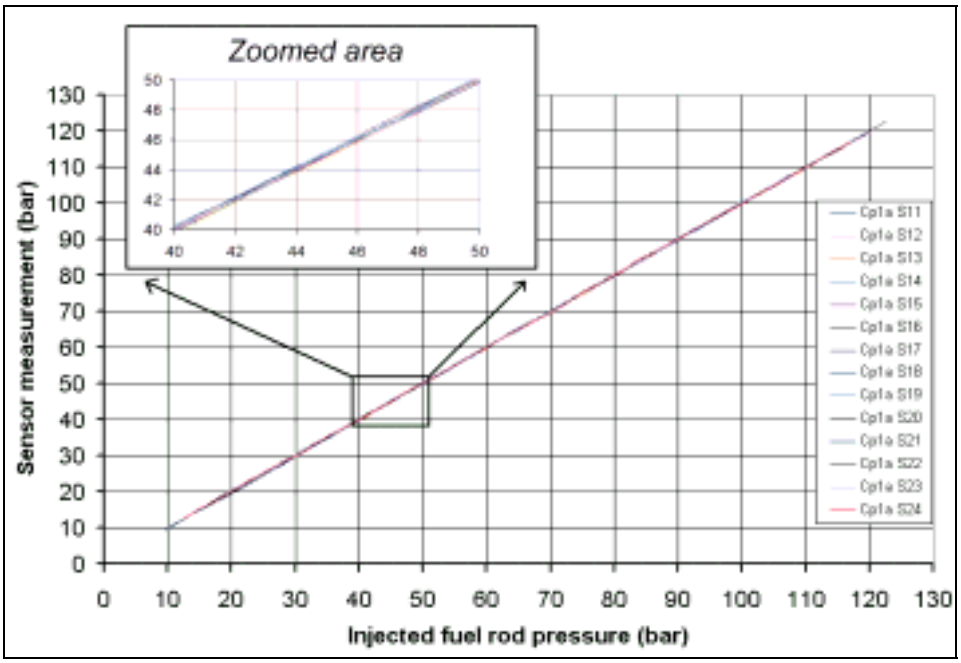

Fig 4. In-pile qualification results, showing sensor measurements for the successive tests within the 300 hours irradiation sequence

The sensor has totally reached its objectives toward performances (see fig 4.). The accuracy of the sensor for the internal pressure measurement during the in-pile tests is ± 0.325 bar (with a confidence of 95%), on the whole pressure range up to 120 bar (see fig 5.).

This accuracy appears to stay at a constant level during the entire irradiation period. This observation proves that the sensor exhibits no drift due to neutron flux or gamma radiation.

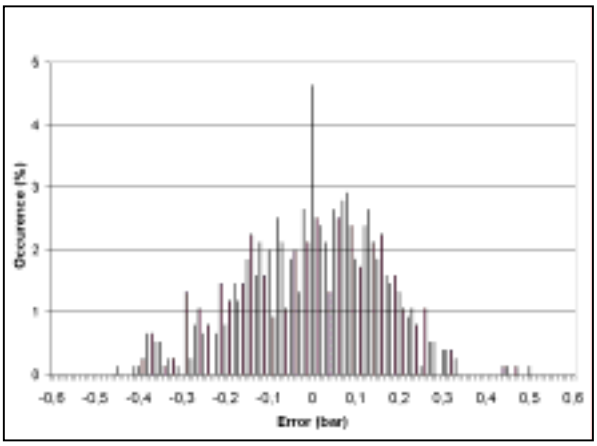

Fig 5. Errors between the sensor measurements and the injected fuel rod pressure during the in-pile qualification

Finally, the sensor gave a complete satisfaction during these qualification experiments, and those results validate the high performances of the device.

This new accurate tool now allows the achievement of advanced programs regarding fission gas release studies. Irradiation experiments using this sensor with pre-irradiated fuel rods are in preparation and will shortly be carried out in the OSIRIS reactor.

DEVELOPMENT OF A COST-EFFECTIVE, RELIABLE BUT DISPOSABLE CAMERA FOR FUEL ELEMENT INSPECTION UNDERWATER AT A LOW POWER RESEARCH REACTOR

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ABSTRACT

This paper will describe a successful project whose aim was to develop a cost-effective, reliable but disposable camera for fuel element inspection underwater, that could also allow inspection of confined spaces in air.
The Objectives for the camera were:

- easy to load, easy to unload
- links to normal VHS/TV/PC, real colour image
- integrally mounted low voltage lighting
- view core at 100kW/elements after 2 weeks decay

This was achieved using the Waterproof Colour Camera SK-2120 [1], which provided excellent underwater images and out of water images, some 1cm from irradiated fuel during a summer outage. The camera system has been employed to operate at depths of over 10 m. To allow for accurate control of the imaging individual polypropylene screwed rods were assembled to the correct operating depth. The camera was able to be pre-set to any angle prior to use. All fixings were individually secured for security. The camera system incorporates 12 white LEDs for underwater illumination negating the need for additional lighting. Water damping provided sufficient stability, although impact protection was incorporated at the camera to (a) protect the camera & lights (b) to protect the fuel cells from direct & indirect damage. The paper will present the images taken during a core inspection which demonstrated fuel clad integrity in fuel some 37 years old. It did also reveal problems in collocation of fuel elements of different design, and the successful resolution of that problem.

1. Introduction

For many years it has become increasingly evident that there is a need to establish a capability for the provision of clear and unambiguous pictorial evidence in support of validation of maintenance regimes and the inspection of safety related items of plant. Earlier work has taken place with the Atominstitute of Vienna and its endoscope, which a group from Vienna brought over to the UK to operate in 1999, and which has been described fully elsewhere [2]. An alternative underwater camera system has now been developed for use in more routine inspection of the Imperial College CONSORT reactor [3]. Both systems have the capability to examine the fuel elements located at a depth of 5 metres under the surface of the cooling/moderating light-water within the reactor tank. This new system has provided a larger full colour image of the fuel elements, and is cheaper and more disposable than a system based upon an endoscope, and is ideal for reactors where the fuel presents a lower radiation dose, and therefore damage, to a camera system. The camera has also proved invaluable in examining other associated parts of the reactor tank internal structure. The system was commissioned for use during the in-tank fuel movements that accompanied the Christmas shutdown in 2002, and has since been in regular use as one of the key inspection tools used on the plant. This poster describes the elements of the system and also gives results and discussion related to the operational experience gained at CONSORT using this system.

2. System description

The camera was selected on the basis of its size, manoeuvrability, and provision of integrated low-voltage illumination. Also the comparatively low-cost and high quality of image were primary considerations. It was established at the outset that the chosen camera would need to be costed to allow for the camera to be effectively sacrificed and replaced should contamination or radiation induced damage become significant. Other important considerations were that the system should be as simple as possible and had to involve using minimal connections between the camera and the Reactor Top. A schematic showing the camera is included as Figure 1 with the camera specification set out in Table 1. Notwithstanding the camera's fixed focal length lens, good quality images can be obtained of most accessible structures within the reactor tank. The camera can easily be used to record the numbers engraved on the sides of the fuel elements and this has proved a useful feature.

Camera Model:	SK 2120LED
Dimensions:	Overall length: 63.6mm Overall diameter: 34.5mm (excluding cable and mounting bracket connection.
Integral Illumination:	12 white light LEDs
Power Supply:	12 volt DC regulated
Consumption Current:	120mA (at 12V DC Max)
Max submergible depth:	30 metres
Temperature range:	–10 °C to + 50 °C
Image sensor:	¼ inch Colour CCD
Light sensitivity:	3 Lux (without LEDs)
Video Output	1.0 Vp-p (Sync. Negative). Termination 75Ω
Umbilical cable length	18 metres
Camera weight:	280g approx.

Table 1. Specification for the Underwater Camera

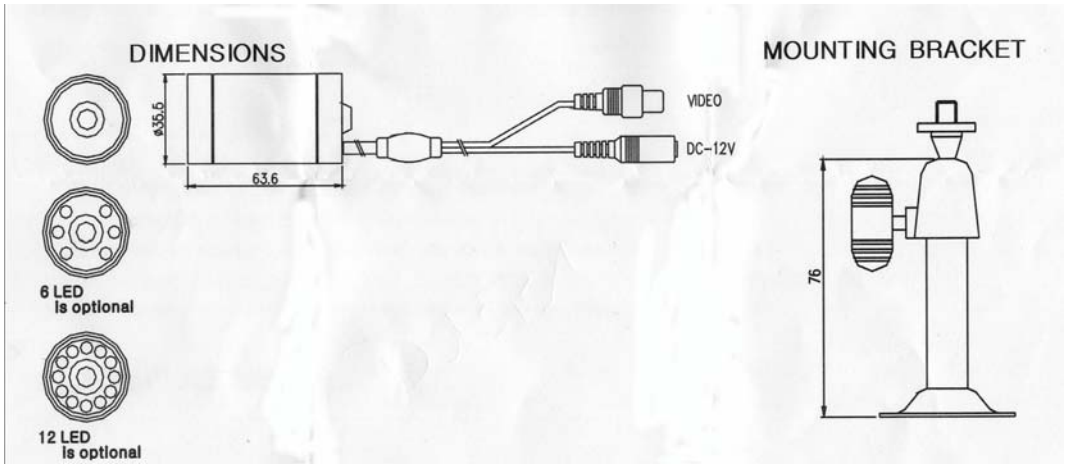

Fig. 1: Schematic of Underwater Camera (12 LED option chosen).

Pre-commissioning tests of underwater image quality were carried out utilising a series of objects (nuts, bolts and washers) dropped to the bottom of a standard 200 litre drum filled with water. These tests gave good results clearly indicating that the camera and its associated LED array could distinguish these objects in a dark drum even in the presence of no external lighting. These tests also verified the cameras submersible integrity.

Fig. 2. The camera assembly showing the array of 12 LEDs, the tip of the deployment tool and camera safety cable (note buffer rods not shown).

A series of polypropylene demountable rods were obtained to form the basis of the deployment tool and in order to ensure that the camera could be placed at the desired angle, a standard adjustable mounting bracket was modified to allow it to fit securely onto the first of the deployment rods. As the camera system is immersed into the reactor tank, the demountable rods are progressively coupled together to build up the desired length and the associated cables are secured to the deployment tool using conventional nylon cable ties. This arrangement allows the camera to be inserted into the tank without extraneous unnecessary rod lengths protruding from the top of the tank. This is important due to the limited headroom available above the CONSORT reactor tank. A pair of nylon 'buffer rods' were mounted axially along the side of the camera to minimise the risk of damage to the lens caused by impact with objects. Electrical connections to the camera were by means of a single cable carrying the 12 volt DC supply and 75 Ω coax for the conventional video output signal. The cable is terminated at the camera via a waterproof sealed gland in the back of the camera assembly. The resulting signal is then fed via a video recorder to a conventional colour monitor for the display of the image. The required images can be stored and downloaded using PC based conventional video capture software. A schematic showing the arrangement is included as Fig. 3.

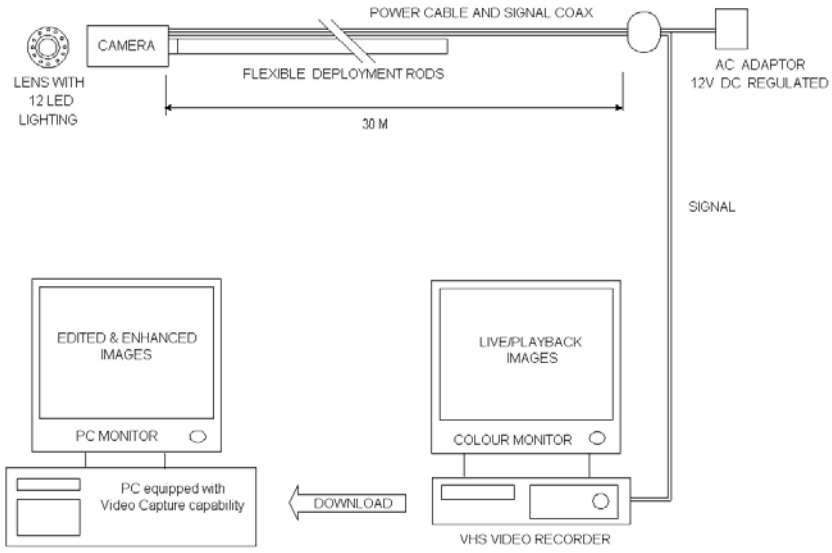

Fig. 3: Schematic showing the underwater camera connected to it's peripheral devices.

3. Operational experience

The system was used for first time during the inspection of fuel elements accompanying the routine in-tank fuel movements associated with the Christmas 2002 shutdown. As a result of the images produced by the system, it was possible to readily identify a number of cases where the fuel elements had not seated down fully into the core-support plate. This situation had evidently been the case for a number of years as it had previously only been possible to view the fuel movements from directly above the open tank through some 5-6 metres of water. From this viewing angle the fuel elements had in the past looked to be seated correctly. The introduction of the camera system allowed the problem to be readily identified. Standard procedures were applied to reconfiguring the elements within the affected channels, and the use of the camera system to survey the core structure allowed complete confidence that all the fuel elements were completely located in the core support plate on completion of the re-fuelling operation.

This system is now routinely used as part of the plant maintenance programme for in-tank inspection and has allowed additional supporting information to be gathered and recorded. Visual information has been previously unobtainable and the additional detail now obtainable during inspections carried out following events with potential for in-tank disturbances such as the recent minor earth-quake that occurred on 23rd September 2002 considerably adds to confidence in the integrity of the components.

Fig. 4. shows the nature and quality of images that can be obtained, and shows the insertion of a Mark I fuel element into the core support plate next to a later design Mark III element during routine fuel reconfiguration. There is always some degree of colour variation resulting from the underwater nature of the image, however, it is clearly possible to gain substantial improvements in loading operations from such images. Fig. 5 shows a view of a partially de-fuelled section of the CONSORT core support plate with a view vertically downwards showing the array of stored elements that are present in the in tank storage pockets below and to the side of the core structure.

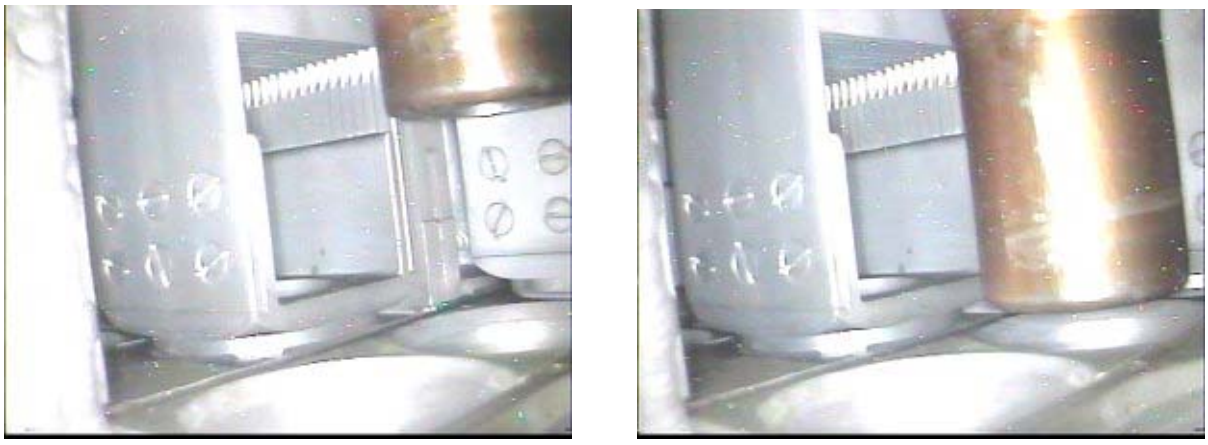

Fig 4. The loading of a Mark I fuel element into the core-support plate.

4. Additional experience

The camera has also proved a useful tool in surveying constricted and confined spaces of uncertain radiological history. The system was used to good effect during an information gathering exercise within the CONSORT Reactor Fast neutron facility, NISUS. Removal of a central graphite plug allowed the insertion of the camera deep into the graphite pile allowing invaluable information to be gathered on the exact location and structure of a series of Uranium items. The camera also allowed a visual survey within the cave to be carried out prior to committing Health Physics personnel for monitoring and survey activities. It has been noted that the camera will have a prominent role to play in any decommissioning operations carried out on the plant. In addition it allows dose-uptake to be minimised by collecting information for planning and Risk Assessment purposes. This is an increasingly important and valuable application.

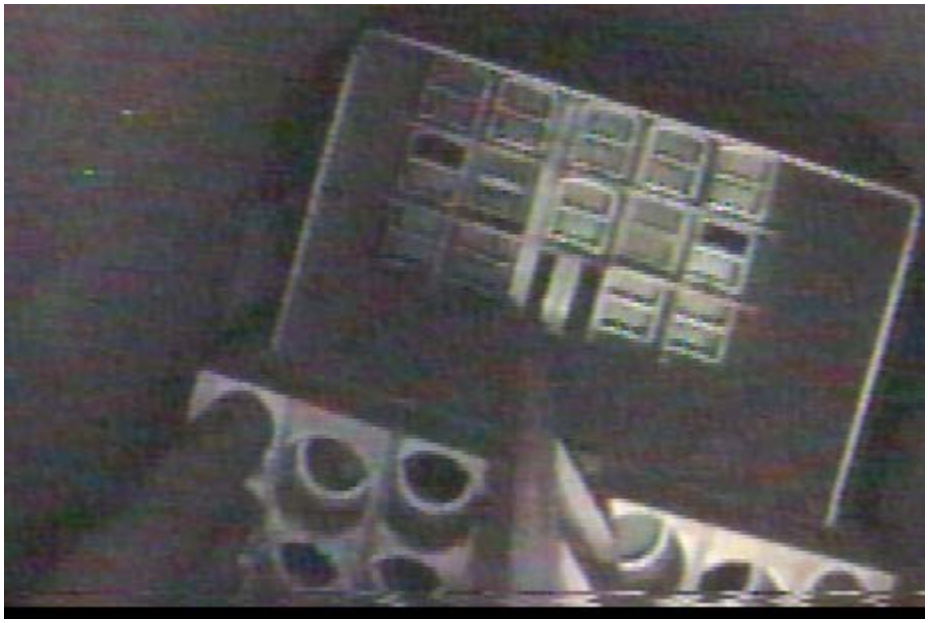

Fig. 5: View of a partially de-fuelled section of the CONSORT core support plate with unused elements visible in the storage pockets.

5. Recommendations for further development

Clearly the camera used in this system has a lifespan primarily determined by the dose uptake of the CCD during use. The first camera used on CONSORT lasted for 12 months of inspections before it was sacrificed in an experiment to establish whether it is possible to provide an in-tank capability for viewing the Cerenkov radiation from the core with the reactor operating at 10kW. An assessment is currently planned to attempt to estimate the failure dose for the device, however, in all extended inspections the reactor will have been in a Full Shutdown state for 14 days prior to the survey. And any additional surveys are normally carried out after an overnight decay. It is clear that the conditions of use for the camera with the reactor at power will dramatically reduce the operational life of the camera. For this reason it may be necessary to mount an in-tank camera only when required at low powers and remove the unit when not in use.

6. References

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ENHANCEMENT OF THE FAST FLUX IN THE AXIS OF A STANDARD BR2 FUEL ELEMENT

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ABSTRACT

A model of a fast neutron flux booster inside a standard BR2 fuel element is proposed. Calculations are performed by MCNP4C code for a full-scale 3–D heterogeneous model of BR2. The geometry of the standard BR2 fuel element allows to insert a fast neutron flux booster around the axis of FE. The booster – uranium fuel ring or basket with fuel pins – converts the high thermal neutron flux in the water around the axis into fast flux. The basic geometry of the srandard BR2 fuel element (FE) is not changed. The poisons are located outside the fuel, in the Al frame of the fuel element. For the booster different types of fuel are considered: LEU (UMo 8.0 gU_{tot}/cm^3) and HEU (UAl_x 1.3÷1.7 gU_{tot}/cm^3) . The achieved fast flux (E_n>1 MeV) in an Al sample surrounded by booster is 60% higher than for a standard BR2 FE without a booster. A comparison between fast fluxes obtained by the Monte Carlo method (MCNP4C) and estimated by diffusion equation approximation is presented.

1. Introduction

The behaviour of structural materials under neutron irradiation is important for prediction and modeling of their ageing and mechanical degradation. The Belgian high flux material testing reactor has an intensive experimental programme on various structural and fuel materials. The specific geometry of BR2 fuel elements with 6 concentric fuel plate rings and a big light water gap around the axis of the FE is appropriate for irradiation of materials. The insertion of some additional uranium volume close to the sample in the axis of the FE allows to convert the high thermal neutron flux in H_2O around the sample into fast neutron flux.

This paper presents a few models of fast neutron flux uranium boosters, located inside a standard BR2 fuel element. The calculations are performed by the Monte Carlo code MCNP4C for a typical reactor core loading. The purpose of the present work was to obtain a maximum fast neutron flux in axis of a FE essentially without modification of its geometry. The only changes made have been the addition of a fuel ring or a basket with fuel pins around the sample. As candidate booster materials UMo LEU (20% enr., 8 U_{tot} g/cm^3) and UAl_x HEU (93÷90% enr.) with different densities (1.3 to 1.7 U_{tot} g/cm^3) were considered. The poisons, normally mixed with the fuel meat in the standard fuel plates, were introduced in the Al frame of the modified FE.

2. Calculational Model by MCNP4C

A full-scale 3-D heterogeneous model of BR2 with exact description of all fuel elements, loaded experimental devices and the inclination angles of the different channels has been developed with MCNP4C. Detailed axial and radial fuel burn-up in all standard BR2 fuel elements, as well poisoning in every channel of Be-matrix are also taken into account. The calculations are performed for a typical BR2 operating cycle (see Fig.1).

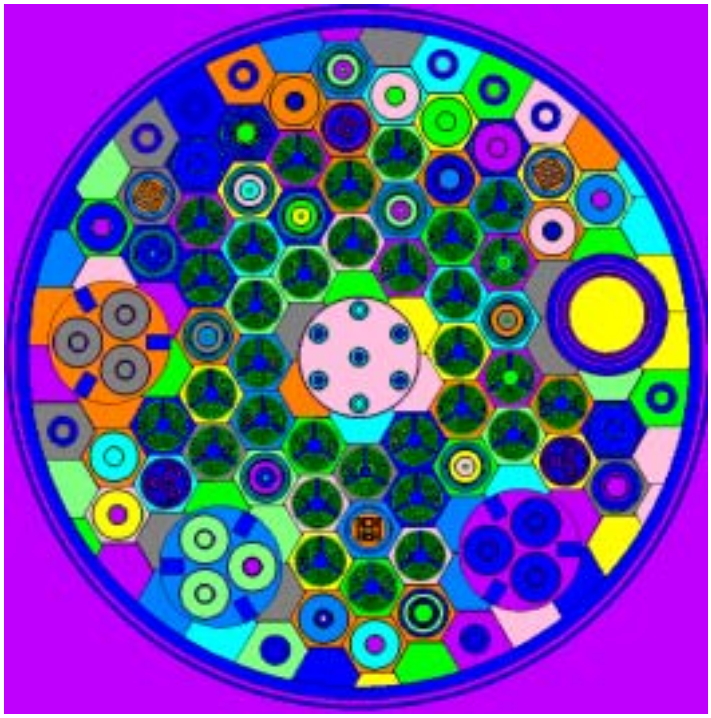


Figure 1. Full-scale 3-D heterogeneous model by MCNP4C for typical BR2 operating cycle.

3. Fast Neutron Flux Booster in a Standard BR2 Fuel Element

Any channel in BR2 containing a standard FE is a potential candidate for application of a fast neutron flux booster. Standard fuel for BR2 reactor operation contains HEU (93%) under the form of UAlx-Al with a density of 1.3 U_{tot} g/cm^3. The mean burn-up in all 32 standard BR2 FE varies from 0% to 50%. In the present paper we consider a FE, located in a channel of the central crown (Fig.2). The booster – an additional uranium fuel ring (Fig.3) or a basket with uranium fuel pins, inserted between the sample and the inner fuel plate (Fig.4)- converts the high thermal flux in H_2O around the sample into fast flux.

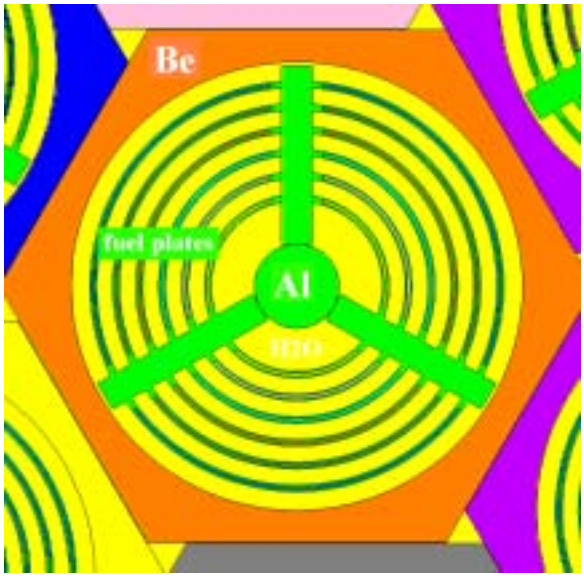


Figure 2. Calculational model of standard BR2 fuel element with Al sample in axis.

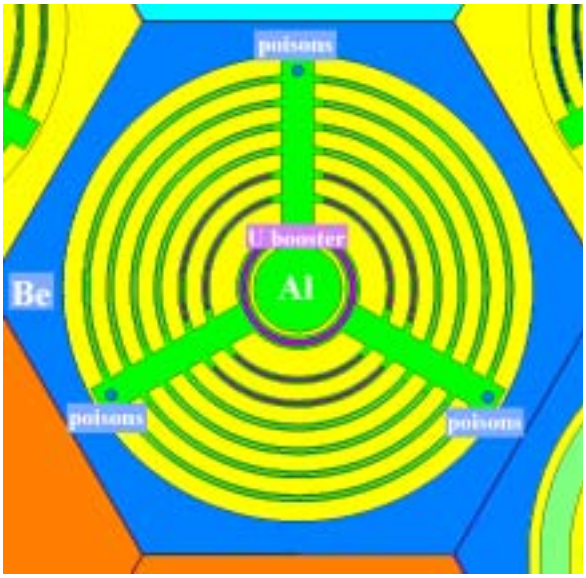


Figure 3. Calculational model of FE with uranium booster - fuel ring around Al sample.

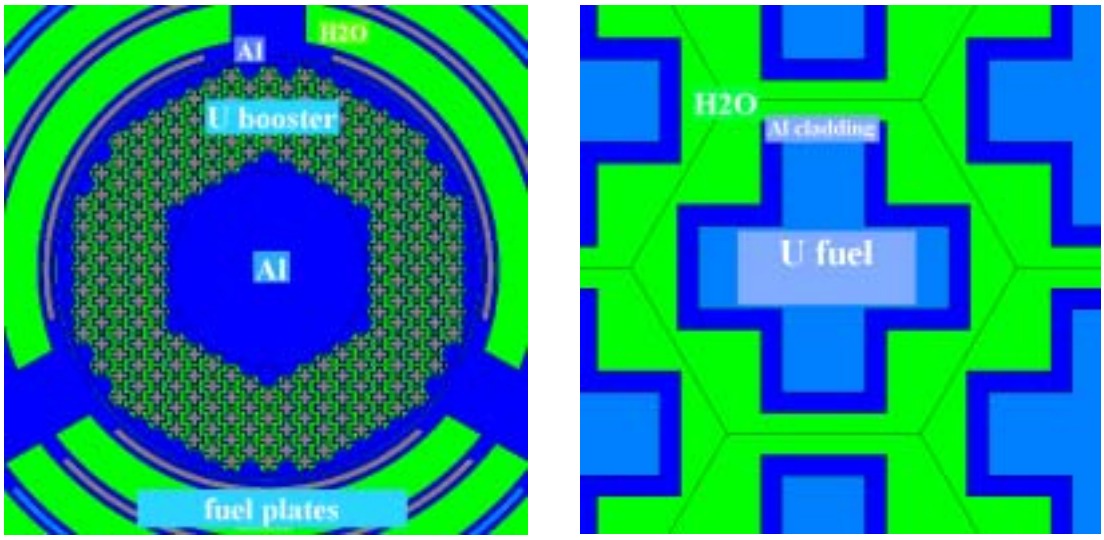


Figure 4. Calculational model of modified FE with uranium booster - basket with cross shaped fuel pins around Al sample in axis.

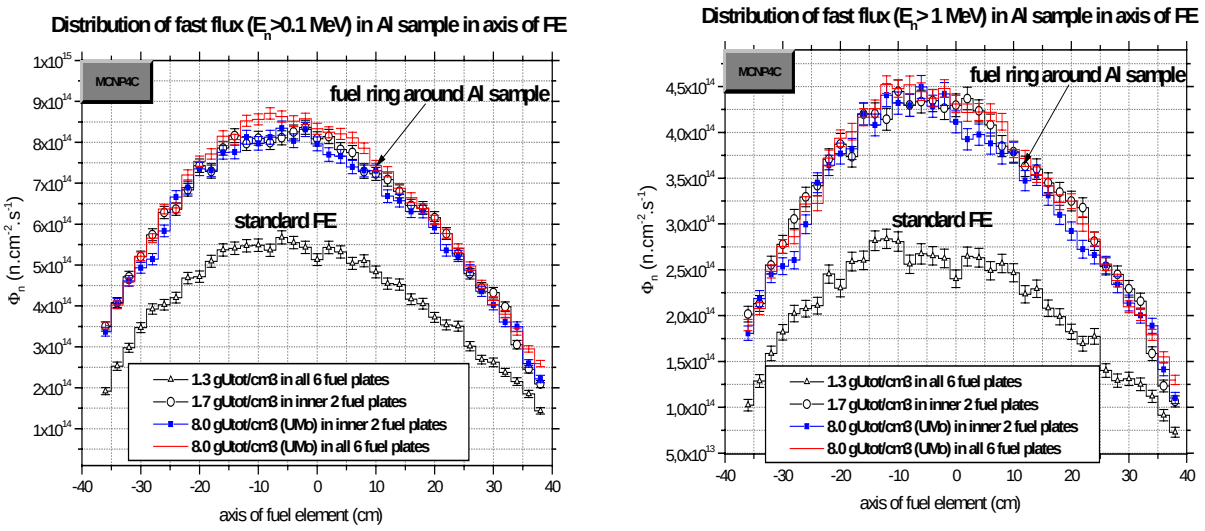
The standard geometry of a BR2 FE has not been changed. The poisons have been separated from the fuel meat and placed in the Al frame.

4. Monte Carlo calculations of Neutron Fluxes in Fuel Element with a Booster

Detailed calculations of axial distributions of Φ_n (E_n>0.1 MeV), Φ_n (E_n>1 MeV) and Φ_n (E_n<0.5 eV) have been performed by the MCNP4C code for a standard FE and for a modified FE with fast flux uranium booster – fuel ring or basket with cross shaped fuel pins around the Al sample (see Fig.5a and Fig.5b). Different fuel types have been considered for fuel plates and booster – LEU (20% enr. UMo 8 U_{tot} g/cm^3), HEU (93% enr. 1.3÷1.7 U_{tot} g/cm^3). The axial averaged fluxes in Al sample are summarized in Table 1 for a normalized reactor nominal power P_r=56 MW. The value Q^{FE}_{hot} (W/cm^2) of the specific fission power (heat flux) in the hot plane of the outer fuel plate (averaged over 360°) of the FE is also presented in Table 1.

Table 1. Comparison of neutron fluxes Φ_n (n.cm^{-2}.s^{-1}) in Al sample in axis of a standard BR2 FE and in modified FE with fast flux uranium booster around Al sample (at reactor power P_r=56 MW). Calculations are performed by the Monte Carlo code MCNP4C.

	Standard FE, Fig2	Modified FE with booster, Fig.3 (Uranium ring around Al sample)			Modified FE with booster, Fig.4 (Basket uranium pins around Al)		
Fuel for Uranium Booster	–	90% enr. UAl_x 1.7 g/cm^3	20% enr. UMo 8.0 g/cm^3	20% enr. UMo 8.0 g/cm^3	90% enr. UAl_x 1.7 g/cm^3	20% enr. UMo 8.0 g/cm^3	20% enr. UMo 8.0 g/cm^3
Fuel for inner 2 fuel plates	93% enr. UAl_x 1.3 g/cm^3	90% enr. UAl_x 1.7 g/cm^3	20% enr. UMo 8.0 g/cm^3	20% enr. UMo 8.0 g/cm^3	90% enr. UAl_x 1.7 g/cm^3	20% enr. UMo 8.0 g/cm^3	20% enr. UMo 8.0 g/cm^3
Fuel for remaining 4 fuel plates	93% enr. UAl_x 1.3 g/cm^3	93% enr. UAl_x 1.3 g/cm^3	93% enr. UAl_x 1.3 g/cm^3	20% enr. UMo 8.0 g/cm^3	93% enr. UAl_x 1.3 g/cm^3	93% enr. UAl_x 1.3 g/cm^3	20% enr. UMo 8.0 g/cm^3
Φ_n E_n>0.1MeV	4.1×10^{14}	6.2×10^{14} (+51%)	6.1×10^{14} (+49%)	6.4×10^{14} (+56%)	6.8×10^{14} (+66%)	6.6×10^{14} (+61%)	6.9×10^{14} (+68%)
Φ_n E_n>1MeV	2.1×10^{14}	3.3×10^{14} (+57%)	3.2×10^{14} (+52%)	3.3×10^{14} (+57%)	3.6×10^{14} (+71%)	3.5×10^{14} (+67%)	3.6×10^{14} (+71%)
Φ_n,E_n<0.5 eV	2.4×10^{14}	1.8×10^{14}	1.7×10^{14}	1.7×10^{14}	1.0×10^{14}	0.9×10^{14}	0.9×10^{14}
Q^{FE}_{hot},W/cm^2	276	345	331	406	325	321	378



a) b)

Figure 5. Axial distributions of fast neutron fluxes in Al sample of standard and modified FE with uranium booster (fuel ring) around Al sample in axis for E_n>0.1MeV (a) and E_n>1MeV (b).

5. Estimation of Fast Neutron Flux in axis of FE by Diffusion Equation Approximation

The fast flux in the axis of a FE can be estimated by using the diffusion equation approximation which in general is valid for the following assumptions: 1/ The neutron flux (in considered case – fast flux) is not changing significantly in FE; 2/ The scattering of neutrons is assumed to be isotropic ($\overline{\cos\theta}=0$); 3/ The absorption of fast neutrons is negligible, i.e. $\Sigma_a << \Sigma_s$. We consider the diffusion equation for monoenergetic fast neutrons (E_n=0.1÷20 MeV):

$$D\Delta\Phi(r) - \Sigma_a\Phi(r) + S(r) = 0 \qquad (1)$$

The solution for the flux in the axis of FE for a point neutron source is:

$$\Phi(r) = \frac{S(r)\exp(-\kappa r)}{4\pi D r} \qquad (2)$$

where: $D = \frac{1}{3\Sigma_{tr}} = \frac{1}{3\left[\Sigma_a + \Sigma_s\left(1-\overline{\cos\theta}\right)\right]}$ is the diffusion parameter, $\kappa = \sqrt{\frac{\Sigma_a}{D}}$ - characteristic of the properties of the medium; Σ_a and Σ_s are the macroscopic absorption and scattering cross sections; $S(r)$ - intensity of neutron sources, generated from a point at distance r from the axis (n.s^{-1}).

The fast flux in the axis of the FE can be calculated as a sum of fluxes, generated by neutron sources distributed in all 6 fuel plates and the U-booster for the modified FE, and the neutron flux resulting from scattering by the light water into axis of FE. In this case $S(r)$ in (2) must be integrated over the volume of the corresponding fuel plate, booster or water layer.

The neutron sources S include neutrons from fission reactions $S^{fis} = \frac{\nu_f (Q/V)}{E_f}$ and from all scattered neutrons with E_n>0.1MeV: $S^{scatt} = <\Sigma_s^{fast}>\Phi^{fast}dr$ (ν_f = 2.43 fission neutrons per 1 fission event; Q, V – the fission power and the volume of the fuel plate or U-ring).

Detailed MCNP4C calculations showed that the absorption of fast neutrons in FE is negligible (Σ_a^{fast}≈0.001 cm^{-1}, κ=0.037 cm^{-1}), so we can write (2) as:

$$\Phi(r) \approx \frac{S(r)}{4\pi D r} \;; \quad D \approx D_s^{fast} = \frac{1}{3<\Sigma_s^{fast}>} \qquad (3)$$

where: $<\Sigma_s^{fast}>$ is the effective macroscopic scattering cross section for fast neutrons in H_2O.

It was calculated by MCNP4C: $<\Sigma_s^{fast}>$=0.47 cm^{-1}; D_s^{fast}=0.709 cm (for E_n> 0.1 MeV).

The various components used for the calculation of fast flux in axis are summarized in Table 2.
The sum of all components Φ_{axis} (from fuel plates, U-fuel ring and H_2O layers) gives the total fast neutron flux in axis of FE: $\Phi_{axis}(DIFF) \approx 6.0\times10^{14}\, n.cm^{-2}.s^{-1}$

In this value is included also the flux scattered from the Al frame: ~4.6×10^{13}n.cm^{-2}.s^{-1}.
The Monte Carlo calculated fast flux in the axis of FE (Table 1, 3rd column) is:

$$\Phi_{axis}(MCNP) = 6.2\times10^{14}\, n.cm^{-2}.s^{-1}.$$

The contribution to this fast flux in axis from the uranium booster (first fuel ring around Al sample) is about 20% from the total fast flux. The contribution from the remaining fuel plates is equal to ~47%. The contribution from fast neutrons scattered by H_2O is ~ 25% and by Al frame about ~ 8%.

Table 2. Estimation of fast flux (E_n>0.1 MeV) in axis of modified FE with uranium booster around axis using diffusion approximation (3). (for comparison with Monte Carlo, see 3rd column in Table 1).

Fuel plates, booster	Booster (U-ring)	1st, inner fuel pl.	2nd fuel plate	3rd fuel plate	4th fuel plate	5th fuel plate	6th, outer fuel pl.
r (cm)	0.99	1.661	2.088	2.515	2.941	3.368	3.795
Q(MW), MCNP*)	0.44	0.28	0.40	0.42	0.53	0.66	0.83
$\Phi^{fiss}\approx\int S^{fiss}dr/4\pi Dr$ $=\int\nu_f(Q/V)dr/(E_f4\pi Dr)$	8.3×10^{13}	5.1×10^{13}	4.3×10^{13}	3.0×10^{13}	2.7×10^{13}	2.5×10^{13}	2.4×10^{13}
Φ(fuel), MCNP*)	7.6×10^{14}	6.6×10^{14}	6.6×10^{14}	6.5×10^{14}	6.4×10^{14}	6.3×10^{14}	6.0×10^{14}
$\Phi^{scatt}\approx\int S^{scatt}dr/4\pi Dr$ $=\int<\Sigma_s>\Phi(fuel)dr/4\pi Dr$	4.0×10^{13}	2.1×10^{13}	1.7×10^{13}	1.3×10^{13}	1.1×10^{13}	9.7×10^{12}	8.3×10^{12}
$\Phi_{axis}(fuel)=\Phi^{fiss}+\Phi^{scatt}$	1.2×10^{14}	7.2×10^{13}	6.0×10^{13}	4.3×10^{13}	3.8×10^{13}	3.4×10^{13}	3.2×10^{13}
H_2O layers	1st,inner	2nd	3rd	4th	5th	6th	7th, outer
r (cm)	1.194	1.875	2.301	2.728	3.155	3.581	4.034
$\Phi(H_2O)$, MCNP*)	1.6×10^{15}	6.1×10^{14}	6.0×10^{14}	5.9×10^{14}	5.8×10^{14}	5.6×10^{14}	5.2×10^{14}
$\Phi_{axis}(H_2O)\approx\int S^{scat}dr/4\pi Dr$ $=\int<\Sigma_s>\Phi(H_2O)dr/4\pi Dr$	8.5×10^{13}	1.7×10^{13}	1.4×10^{13}	1.1×10^{13}	9.7×10^{12}	8.2×10^{12}	6.7×10^{12}

*) These values have been calculated by MCNP4C. Φ(fuel) and $\Phi(H_2O)$ are the averaged fast fluxes (E_n>0.1 MeV) in fuel plates, U-ring or water layers, located at a distance r from the axis of FE.

6. Conclusion

The geometry of a standard BR2 fuel element allows the introduction of a fast neutron flux booster inside this FE. The purpose is to convert the thermal neutron flux in the water inside the fuel element into a fast neutron flux available for experimental samples loaded in the axis of the FE. Various possibilities have been investigated: an uranium fuel ring or a basket filled with uranium fuel pins around the experimental samples located in the axis of the FE; LEU (UMo, 8 U_{tot}g/cm^3) and HEU (UAl_x 1.3÷1.7 U_{tot} g/cm^3) are considered.

The calculations are performed with the Monte Carlo code MCNP-4C for a full-scale 3-D heterogeneous model of BR2; axial and radial fuel burn-up and poisoning of Be matrix are taken into account. The basic geometry of a standard BR2 FE is not changed but the burnable poisons are removed from the fuel meat and placed in the aluminium frame of the fuel element.

The main conclusions of this investigation are:

– the considered types of boosters allow to enhance the fast neutron flux about 60% compared to a standard BR2 FE;

– the fast flux in axis is inversely proportional to the distance between axis and the corresponding source of fast neutrons in FE. This is in accordance with the diffusion equation approximation.

INVENTORY REPORTING USING OSCAR-3

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ABSTRACT

The reporting of the fission material inventory to the regulatory authorities and the IAEA is common practice for all nuclear installations including research reactors. The input values to these reports are often based on approximate methods which typically make use of measured radial power profiles, simplified relationships between uranium burnup and thermal power delivered or generic depletion calculations. When making use of the OSCAR-3 core neutronic code system, not only are the actual histories of the fuel elements taken into account, but spectral effects introduced by neighbouring elements, reflector or irradiation rigs are also included in the microscopic depletion calculations. This yields more accurate element inventories and source terms. The importance of spectral and burnup history effects is illustrated and supported by practical examples comparing OSCAR-3 results with the approximate methods.

1. Introduction

Historically research reactors were operated without much calculational support. Safety margins were very large and measurement was the main method of determining most parameters. The changing role of research and specifically MTR reactors has necessitated more advanced support. Improvements to the calculational tools available and improved computer power have resulted in generally much better theoretical support to these reactors. SAFARI-1 is a good example where the OSCAR-3 calculational code system [1] plays a very important role in supporting day to day operations but also the commercial development of new irradiation products [2],[3].

One aspect where the use of OSCAR-3 has not yet been implemented is in making use of the isotopic compositions calculated in OSCAR-3 to perform inventory reporting. This paper will show and explain the differences between the historical approach and the OSCAR-3 results. In particular, the sensitivity of the mass reduction of the element U-235 during burnup will be shown for different fuel elements as well as the differences obtained for a selection of SAFARI-1 fuel element histories.

The availability of element cycle history and transuranic isotope concentrations in OSCAR-3 is another advantage. The transuranics and in particular the plutonium isotopes can easily be added to the inventory reports in this way. The data are also valuable when source terms need to be calculated for spent fuel. The planned inclusion of a reporting feature to OSCAR-3 will also simplify the reporting of the fission material inventory with a quality assurance benefit.

2. The historical approach to inventory reporting

The approximate historical approach is currently used to update the inventory reports at SAFARI-1. The method is based on two quantities, namely a relationship between U-235 burnup and power delivered and, secondly, the power delivered by an element of interest during a cycle. In the current approach used at SAFARI-1, the change in the U-235 content of a fuel or control-fuel element is based on the assumption that a constant amount of U-235 is depleted to deliver a given amount of thermal power. In the case of SAFARI-1 a value of 1.26g/MWd is used. The power delivered by the element of interest in the core was previously determined from flux measurements. In SAFARI-1 copper wires were irradiated at very low power in all fuel element positions and a relative power distribution was then derived from the measured

activation values. From the relative power distribution and the actual reactor thermal power the power delivered by an element can be calculated. The U-235 mass reduction is then simply estimated by multiplying the power delivered by the element during the time step by the constant factor. By doing this for all elements, the mass of U-235 depleted in the reactor may be determined.

Although the approach is straightforward and accepted by the regulatory authorities, it has serious shortcomings. These are summarised as follow:
- The amount of U-235 depleted for a given thermal power is not constant. As will be shown later this assumption is not necessarily too bad for a reactor loaded with high enriched uranium (HEU). No such constant value can however be assumed for a low enriched uranium (LEU) element. This is of course well known and is simply due to the increased importance of the other fission isotopes (mainly plutonium) created due to the increased neutron capture in LEU fuel (higher U-238 content).
- The power distribution in the core during a cycle could be quite different from the distribution at low power and at the beginning of the cycle when the measurements are made. The power distribution will change due to xenon build-up, operating temperature distributions, change in burnup distribution, control bank positions, insertion of irradiation samples during operation and many more.

As part of the cycle and reload planning the copper wire activation measurements were used to determine power peaking and hotspot values and formed part of the SAFARI-1 operating licence requirements. A few years ago the operating requirements were amended and the OSCAR-3 calculated powers and hotspot factors were introduced in the place of the copper wire measurements. Naturally the need to perform copper wire measurements every start-up was discarded (only done intermittently) with a significant saving in the downtime between reloads.

Since the copper wire measured values were no longer available the method used to estimate the SAFARI-1 inventory reporting was also adjusted. The calculated thermal flux distribution from OSCAR-3 at the beginning of cycle was selected to simulate the copper wire activation activities and used as input to the U-235 inventory calculations. This was a slight improvement since the true reload configuration could be taken into account and more realistic control bank position and xenon equilibrium conditions could be simulated. The history change during the cycle, although available from OSCAR-3, was never implemented and thus the beginning of cycle power distribution is essentially still assumed for the duration of the cycle.

3. Assembly burnup and U-235 masses

Since the traditional method used to determine inventories (or more specifically the U-235 content) for SAFARI-1 is based on a constant relationship between the amount of U-235 burned and the power delivered, this relationship is studied in more detail. The amount of U-235 burned in an element in order to produce one mega watt day (MWd) was compared for different fuel elements as a function of burnup. A variation in this value can be expected due to variations in U-235 enrichment, the element burnup and neutron spectrum. The most important factor will be the enrichment since it determines the possible formation of other fissionable isotopes (Pu from U-238 and U-236 from U-235) with burnup. Other factors such as material compositions and element design will affect the neutron spectrum.

Three SAFARI-1 assemblies were considered and are called HEU300, HEU200 and LEU340 (envisaged). Table 1 provides a summary of the specifications. Single assembly burnup calculations were performed with the HEADE two-dimensional transport code. The code uses the collision probability method within a cell, while a low-order interface current method [4] is used to connect these cells in the total assembly calculation. Calculations are performed with a 172 fine-group neutron cross-section library [5] based on JEF-2.2. For each case the quantity of U-235 burned in a particular time step and the power produced in the time step (given in MWd) were used to calculate the “mass U-235 burned per one megawatt day”. The results are shown in Fig 1.

Table 1. Assembly specifications.

Properties	HEU300	HEU200	LEU340
Number of plates per fuel element	19	19	19
U-235 loading per fuel element [g]	300	200	340
Uranium content per fuel element [g]	333	222	1721
Enrichment [%]	90%	90%	19.75%
Meat material	UAl_x-Al	UAl_x-Al	U_3Si_2-Al
Weight percentage of U in U-Al alloy	28%	19%	-
Uranium density in meat [g/cm^3]	0.916	0.611	4.73
Meat thickness [cm]	0.0508	0.0508	0.0508
Cladding thickness [cm]	0.0384	0.0384	0.0384
Coolant gap [cm]	0.2932	0.2932	0.2932
Fuel height [cm]	59.37	59.37	59.37
Effective planar dimensions [cm]	7.71 cm x 8.1 cm	7.71 cm x 8.1 cm	7.71 cm x 8.1 cm

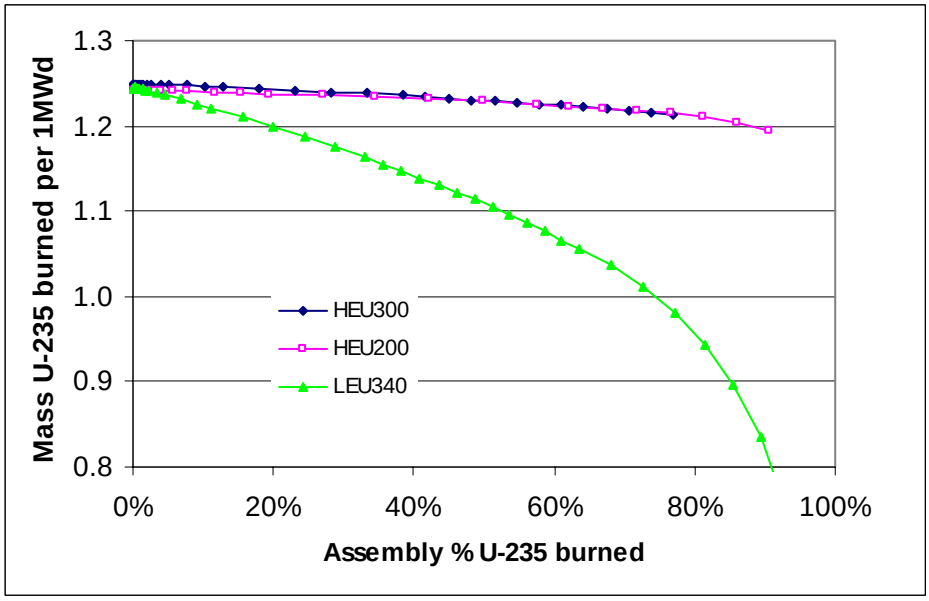

Fig 1. Mass of U-235 burned per 1 MWd as a function of burnup.

It is immediately apparent that the U-235 burned per MWd is not constant as assumed in the historical approach. The value decreases slightly for both the HEU elements while a significant decrease is realized for the LEU assembly. The results at 0% burnup compare favourably since in all three cases U-235 is the major contributor to the fission power for fresh fuel. In case LEU-340 the contribution to fission by the plutonium built-up increases with time. For example, at 50% U-235 burnup, the element contains more than nine grams of Pu-239, making a considerable contribution to the fission power. This results in a sharp decline in the U-235 that is required to deliver 1 MWd of power. For example this value has decreased from 1.24g/MWd to only 1.1g/MWd for the LEU-340 element with 50% U-235 burnup.

The values for HEU-300 and HEU-200 are similar. This is expected since both elements contain 90% enriched material. Although the elements have exactly the same geometry, the 300-gram element will result in a harder neutron spectrum due to its higher uranium fraction (28% versus 19%) in the U-Al alloy. This affects the fission product and actinide isotopic compositions. With burnup the HEU U-235 burned per MWd changes from about 1.25g/MWd for fresh to 1.23g/MWd for 50% U-235 burned. This is of course also due to the contribution of other fissionable isotopes but in this case the effect is much smaller due to the small amount of U-238 present.

4. Examples from the SAFARI-1 reactor

The SAFARI-1 reactor is a 20MW tank-in-pool type material testing reactor owned and operated by NECSA at its Pelindaba site near Pretoria, South Africa. An 8 Δ 9 grid houses 28 fuel elements, 5 control rods, 1 regulating rod, in-core irradiation facilities and reflector elements (see Fig 2). The core is fuelled with MTR type fuel elements with 19 plates. The core currently contains HEU-300 fuel elements, up from a loading of 200 grams U-235 (HEU-200) in the late 1990's, while conversion to LEU is envisaged.

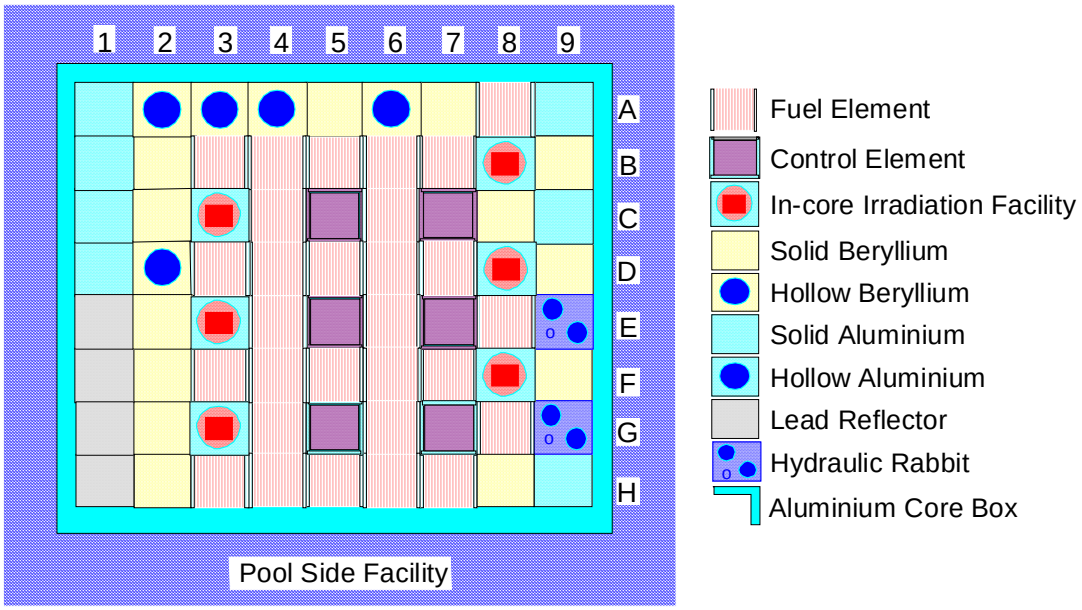

Fig 2. Top view of SAFARI-1 core.

In order to evaluate the importance of burnup history effects and core position on the "mass U-235 burned per MWd" ratio, comparisons were made between HEADE results and SAFARI-1 cycle calculations for HEU-300. This is presented in Fig 3. It shows the ratio as obtained from the HEADE depletion calculation and results from two fuel elements from SAFARI-1 OSCAR-3 analysis. From the OSCAR-3 core follow calculation the U-235 mass remaining per 300g U-235 HEU assembly as well as the burnup of the assembly at the end of each cycle was obtained. These parameters were used to determine the mass of U-235 required to deliver a unit amount of energy, and the result plotted as a function of the percentage of U-235 mass depleted in the assembly. The first lead test assembly (LTA) during the conversion from HEU-200 to HEU-300 was selected (called "HEU-300 LTA"). A second fuel element was taken after the conversion has been completed, called "Production". The burnup of the two elements, over several cycles, was traced.

Whereas the 2-D HEADE assembly calculation assumes an infinite array of HEU-300 elements, the two HEU-300 elements ("HEU-300 LTA" and "Production") followed the operational history and were typically surrounded by elements and materials quite foreign to the environment assumed in the assembly calculation. Typically a fresh element is introduced on the periphery of the core (row H) neighbouring the pool-side facility after which it will spiral inwards to the centre of the core over its lifetime. This is the history followed by the "Production" element.

The "HEU-300 LTA" element, being the first "heavy element" introduced into the core loaded with HEU-200, was placed in position A8 for the first ten cycles. This was required to avoid power peaking due to the increased U-235 loading. Only after this it was moved to row H where the fresh elements are normally introduced. It was then also spiralled inwards into a core filled mostly by HEU-200 elements. The effects of these different loading positions and surrounding element types on the mass U-235 burned per MWd can easily be seen in Fig 3. As the element spirals inwards with depletion and is surrounded more by its own kind, a value similar to the HEADE assembly calculation can be seen.

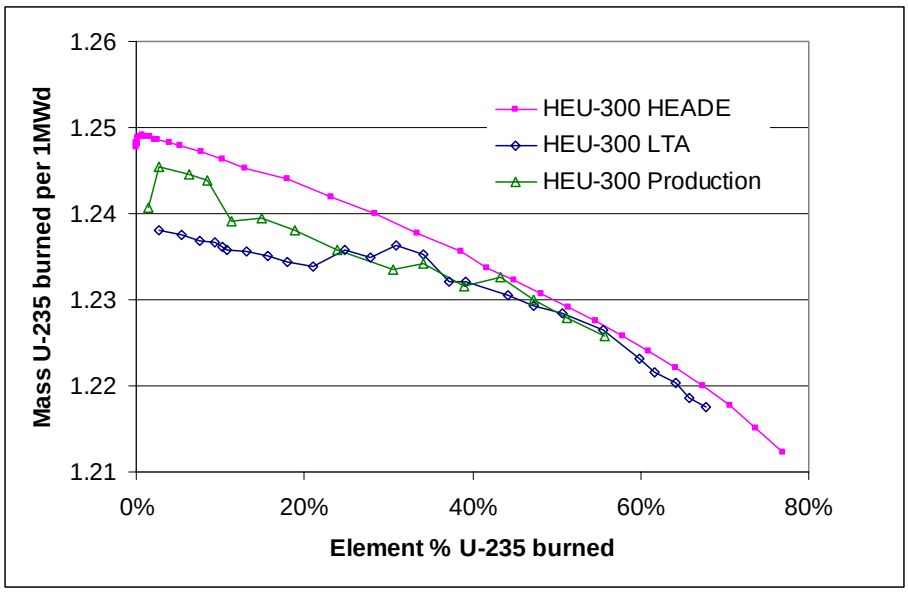

Fig 3. Mass-burnup relation for HEU-300 element histories compared with assembly calculation.

The importance of the fuel element environment on its burnup behaviour and specifically the mass U-235 burned was clearly illustrated for HEU-300. This applies even more during conversion when the element is not placed in the environment as assumed in the HEADE assembly calculation or as represented by a fixed relation (constant or formula) between the mass U-235 burned and power delivered.

In order to quantify the effect the use of OSCAR-3 will have on the SAFARI-1 inventory reports, comparisons were made between the U-235 mass inventory as predicted by OSCAR-3 and the values obtained using the simulated copper wire activation procedure, described above. Three fuel elements were traced in OSCAR-3 and the mass U-235 compared with the inventory reports. The differences in U-235 masses are presented in Fig 4 where the cycle numbers refers to the 1st, 2nd and succeeding cycles where the fuel element was loaded. The legend shows the element serial numbers. In the three examples significant differences of up to 9 grams were found between the two methods. Note that the U-235 mass differences in the total facility could be relatively smaller with assemblies having more or less U-235.

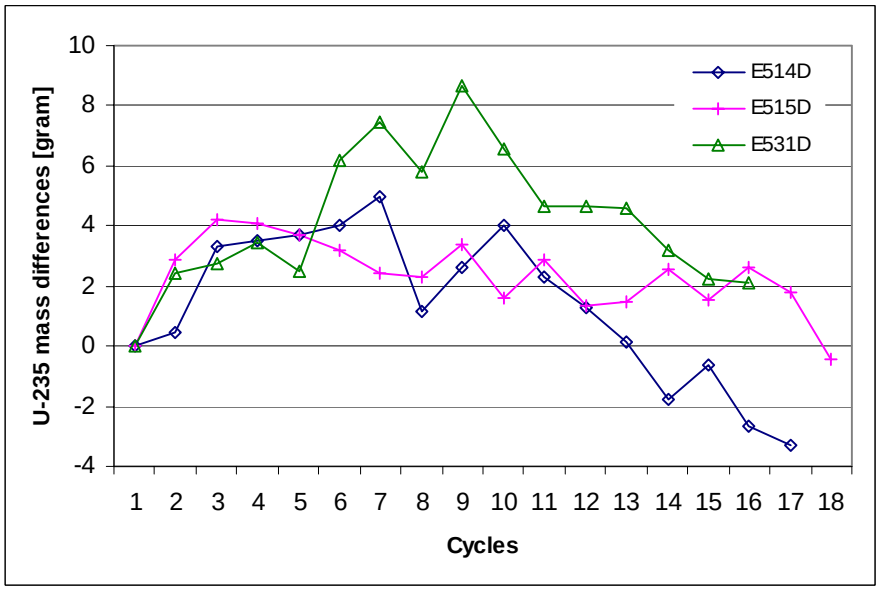

Fig 4. U-235 mass differences [grams] between approximate method and OSCAR-3 calculations.

5. Implementation in OSCAR-3 and MAESTRO

In OSCAR-3 a list of isotopes, to be treated microscopically, can be defined. In practice it means that the depletion calculation will be solved separately for each of these isotopes so that the homogenised atom-densities and thus element masses are known at any time during burnup. By defining all the uranium and plutonium isotopes to be treated microscopically, these masses are available for inventory reporting.

An ASCII file is currently used to store the latest fission material inventory of all elements. The serial number of an element is used as identification and the position, date loaded and masses are stored. The concept of different facilities such as the core, fresh fuel vault or spent fuel pool is already used in OSCAR-3. This facilitates the reporting of the inventory based on the location of the fuel elements. Provision was also made to store comments on each assembly such that it has been cropped or earmarked for storage at the dry pipe store facility. As part of future development a secure format where the integrity of the data can be assured will replace the ASCII file. The use of a general database format is investigated as a possibility. This database can then be used by the reporting features to be included in the OSCAR-3 graphical user interface called MAESTRO or imported into any commercial database for presentation purposes.

6. Conclusions and future work

It was demonstrated that the operating history of a fuel element has a large effect on the fission-isotope content of the element. The amount of U-235 burned per MWd depends on the uranium loading, location in the reactor, enrichment and burnup. These variations cannot be captured by approximate burnup models based on constant or generic relations between U-235 content and the power delivered.

The use of measured or calculated power profiles only at the beginning of a cycle, can introduce significant errors since the power distribution could change significantly due to xenon feedback effects and control bank positions. The combined effect of the approximate methods was shown to be larger than 8 grams in U-235 for one of the fuel elements compared.

In the case of an LEU core the contribution of plutonium becomes important and should be included in inventory reports. The extension of the SAFARI-1 inventory reports to include plutonium is foreseen as soon as the envisaged conversion to LEU commences. The availability of all the data in OSCAR-3 makes it an attractive option to use the code system as a tool to perform not only reload planning and core follow analyses, but also a tool to manage and produce fuel element inventory material reports.

The implementation of an automated inventory reporting system in MAESTRO is being planned, making use of a secured data format. All the uranium and plutonium isotopic masses would be available, with a selection to be reported in an inventory report. Export functions to customised reports are also being planned.

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PERFORMANCE OF THE LEU FUEL AND USE OF HIGHER ^{235}U LOADING IN THE PAKISTAN RESEARCH REACTOR-1 (PARR-1) FUEL

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ABSTRACT

The Pakistan Research Reactor (PARR-1) is a 10 MW_{th} swimming pool research reactor using MTR type fuel. It achieved first criticality on December 21, 1965. After operation of about 25 years it was converted to LEU (20%^{235}U) fuel with power upgraded to 10 MW_{th}. During studies for new fuel design it was found that 290 grams of ^{235}U per standard fuel element will be sufficient to attain about 35% burnup and the fuel was fabricated with this loading. The converted and upgraded reactor attained first criticality in October 1991 and full power in May 1992. Since then the reactor has been in regular operation with satisfactory fuel performance. In 1998 the target burnup was enhanced to 40% and several fuel elements have been discharged after achieving this burnup.
In the next fuel design it is being considered, to increase the ^{235}U loading by about 10%. Preliminary analyses show that burnup can be enhanced to about 50% in this way. Detailed analyses of the mixed cores (containing 290 and 320 gram fuel elements) and then high loading cores are being planned.
At the time of conversion to LEU fuel, several of the HEU fuel elements had very low burnup. It is also being studied to re-use these elements with the present LEU fuel.

1. Introduction

The Pakistan Research Reactor (PARR-1) is a 10 MW_{th} swimming pool research reactor using MTR type fuel. It achieved first criticality on December 21, 1965 and attained full power of 5 MW in June 1966. The original PARR-1 design utilised highly enriched uranium, HEU, fuel (93% ^{235}U). The reactor had been extensively used for training of manpower, production of isotopes and nuclear research. In the late 70's the availability of HEU fuel was discontinued. Therefore, in the 80's planning for conversion to low enriched (LEU: 20% ^{235}U) fuel was initiated in line with the international RERTR programme. There had been pressing demands to provide higher flux for experiments and isotope production. In order to meet these requirements, it was also planned to raise the reactor power from 5 to 10 MW_{th}. The actual conversion project was started in 1988. Detailed studies were carried out about core physics, thermal hydraulics and safety. The reactor and its associated systems were inspected for their condition and usefulness for the extended life.

After operation of about 25 years, the reactor decommissioning activities were started in November 1990. The conversion and upgradation project involved a large number of changes and improvements in the reactor [1]. During this phase, a separate wet storage bay was constructed for fuel and other radioactive components such as the graphite reflector elements and control rod blades. A transfer cask was designed and fabricated. In order to remove the chronic seepage problem the reactor pool was lined with stainless steel. Apart from reactor fuel, major modifications and additions were made in the heat removal system to take-up heat load due to power upgradation from 5 to 10 MW. Both primary and secondary heat removal systems were completely redesigned. New primary pumps, heat exchangers and cooling towers were installed. To ensure safety in case of a loss of coolant accident an emergency core cooling system was installed. Several other renovations were made to replace most of the old systems with new and better equipment particularly in the HVAC, compressed air system and normal as well as emergency power supply systems. Both interior and exterior of the containment

building were thoroughly renovated to remove leakages and the weather effects. Thus, during the conversion and upgradation project not only the fuel was converted to LEU and power was doubled but whole reactor and its associated systems were renovated. In this way life of the reactor was expected to increase by another 15 to 20 years.

2. Design of the previous LEU fuel

The HEU core of PARR-1 utilized 93% enriched uranium fuel in the form of UAl_x-Al alloy. The standard fuel element had 196 g ^{235}U contained in 16 fuel bearing plates with two aluminium dummy plates. Each control fuel element had 98 g ^{235}U contained in 8 fuel bearing plates. Fuel density was 0.7 g/cm^3. LEU fuel for PARR-1 was chosen to be 19.99% enriched and detailed calculations about neutronics, thermal hydraulics and accident analysis were carried out for various fuel types such as U_3O_8-Al_3, U_3Si_2 and different fuel loadings [2]. A comparative study of several fuel element designs having different meat material, uranium loading and number of fuel plates, was carried out to optimize the neutron flux at irradiation sites, fuel burnup and reactor power without compromising on the reactor performance and safety. Based on the results of these studies and keeping in view the specific PARR-1 requirements, standard fuel element having 23 fuel plates with meat density of 3.3 g/cm^3 and loading of 290 g of ^{235}U in the form of U_3Si_2-Al, was selected. The control fuel element was designed with 2 aluminium dummy plates as guides for the control rods and 13 fuel plates having a loading of 164 g of ^{235}U.

During these design studies it was found that uranium loading of 290 grams of ^{235}U per standard element will be sufficient to attain a burnup of about 35% [3]. Since fabrication constitutes a significant part of the total fuel element cost, a higher ^{235}U loading would have been desirable for economic reasons. However, at that time no experience was available to us about behaviour of the new fuel. Moreover, for the first time some very significant changes were being suggested in the fuel design (Table 1). Therefore, a moderate target burnup of 35% was taken to be reasonably good and correspondingly the LEU standard and control fuel elements were designed.

Table 1: Comparison of some parameters of the HEU and LEU fuel elements

Parameter	HEU	LEU
^{235}U Enrichment (% by wt)	93	19.99
^{235}U Loading (g) in Standard FE Control FE	196 98	290 164
No. of Fuel Plates in Standard FE Control FE	 16 8	 23 13
No. of Dummy Plates in Standard FE Control FE	 2 1	 Nil 2
Shape of Fuel Plates	Curved	Straight
Density of Uranium in Fuel Meat (g/cm^3)	0.7	3.3
Water Channel Thickness (mm)	3.12	2.10
Coolant Velocity (m/s)	1.74	3.35
Heat Transfer area (cm^2)	11712	17319
Flow Area (cm^2)	35	31
Average Heat Flux (W/cm^2)	20.3	23.6

3. Experience with the LEU fuel

The converted and upgraded reactor attained first criticality in October 1991. Experimental measurements of various core configurations were carried out in the open and stall pool ends. The measured results verified the design parameters. Full power was achieved in May 1992. Initially a few continuous reactor operations were arranged for test and calibration of the instruments and to study the behaviour of radiation and xenon poisoning. Regular operation was started soon after that. Since then the reactor has been in regular operation and performance of the fuel has been satisfactory [4]. The fuel elements, close to the design burnup of 35%, were inspected regularly and no visible abnormality was observed. Thus in 1998 the core configuration was slightly modified to enhance the target burnup upto 40%. [5, 6]. The reactor has been operated without any untoward incident and at present several fuel elements have been discharged after achieving burnups of 40% or more. Their inspections have revealed no abnormality. Thus, the performance of the LEU fuel has been according to the expectations, even for the burnups higher than the design value.

4. Design of the new LEU fuel

With satisfactory performance of the 290 g fuel elements, it is now being considered to increase the ^{235}U loading in the fuel. Preliminary studies have shown that about 10% enhancement in ^{235}U loading (i.e. from 290 to 320 g per standard fuel element) may be adequate and with that burnup close to 50% may be achieved from the future fuel elements. In this regard, a programme for analyses of the mixed cores (containing 290 and 320 gram fuel elements) and then high-loading cores is being prepared. Results of these analyses will be used to design the future fuel elements for PARR-1. Use of higher ^{235}U loading will reduce the reactor operation cost significantly, as fabrication constitutes a major portion of the total fuel cost.

Initially the 320 g ^{235}U fuel elements will have to be utilised with the existing 290 g fuel elements in the mixed cores. The higher ^{235}U loading fuel can not be randomly used with the lower ^{235}U loading fuel because of the safety reasons. For this purpose, appropriate loading procedures will be established after detailed analyses. At first, the 320 g fuel elements will be loaded at the core sides where neutron flux, and hence the power production, is rather low. After attaining initial burnup they will be moved successively towards core centre. In this way, the reactor core will shift from the 290 g fuel towards the 320 g fuel.

5. Use of the Low Burnup HEU Fuel Elements

At the time of decommissioning for conversion to LEU fuel in November 1990, several of the HEU fuel elements had very low burnup. The average burnup of the previously discharged HEU fuel elements was about 35% whereas some of the HEU fuel elements discharged at decommissioning had burnup varying from 3 to 20%. Therefore, it is also being studied to re-use these elements with the present LEU fuel. Use of the mixed HEU-LEU fuel elements in PARR-1 core needs detailed analysis and approval of the concerned licensing authority. A preliminary analysis of a mixed core containing 28 LEU fuel elements (24 standard and 4 control elements) and 6 HEU elements (5 standard and 1 control element) was carried out. The results have shown [7] that the mixed HEU-LEU core has acceptable characteristics. Flux at the irradiation facilities is comparable to that with the LEU-only core so that reactor utilisation is not affected. The power peaking factors, control rod worth and shutdown margins and other safety parameters are similar to the LEU-only core thereby maintaining same safety levels. Therefore, the leftover HEU fuel can be utilised safely with the LEU fuel. The only restriction will be that the HEU elements will have to be loaded on the thermal column side. This is because the old HEU fuel elements have curved fuel plates and the convex side of the plates is towards the thermal column side. The new LEU fuel elements have straight plates. Same is the case of the graphite reflector elements and the irradiation water boxes.

5. Conclusions

After conversion and upgradation the PARR-1 has been operating satisfactorily for the past 10 years. The LEU fuel has shown excellent performance even for the burnup higher than the design value. Keeping in view good experience of the 290 g fuel, the new fuel may be designed with 320 g of ^{235}U per standard element. This will reduce the over-all fuel (and hence operation) cost. It is also found that some of the partially depleted HEU elements may also be utilised safely along with the existing LEU fuel.

6. Acknowledgments

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Session 4

Spent fuel management, back-end options, transportation

DRAFT EDLOW GROUP PROPOSAL FOR EXTENSION OF DOE FRR SNF ACCEPTANCE PROGRAM

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ABSTRACT

We urge the US Department of Energy (DOE) to extend the Foreign Research Reactors Spent Nuclear Fuel (FRR SNF) Acceptance Program for five years. We request a built-in commitment for an additional extension of five years if the availability of the molybdenum fuels is further delayed and to allow for the return to the US of the last of the remaining silicide spent fuel.

1. Recommendations

We urge DOE to take steps now to extend the FRR SNF Acceptance Program for five years, (through June 2011, with the additional three years needed for the last of the fuel used in 2011 to cool down and be prepared for shipment). In addition, we are requesting a built-in commitment to consider toward the end of the five years an additional extension if more time is needed before the molybdenum fuels can be licensed for use, and for the completion of the return to the United States of the last of the remaining silicide spent fuel.

2. Background

The Edlow Group of eight non-U.S. research reactor operators is most appreciative of the strong commitment of the Department of Energy (DOE) and the Department of State (DOS) to the establishment and implementation of DOE's Foreign Research Reactors Spent Nuclear Fuels (FRR SNF) Acceptance Program, which was launched in 1996. The Edlow Group, and the Governments of the countries in which the research reactors are located, supports the nuclear nonproliferation policy, which provides the foundation for the Acceptance Program, i.e., to minimize the civil use of, and world trade flows in highly enriched uranium. We urge the U.S. Government to stay the course and continue the Acceptance Program so that the goals of the program can be fully realized.

Recent unsettling events on the world scene highlight the importance of stabilizing, securing, and minimizing trade flows of those fissile materials, which can be used directly to make weapons, or in connection with terrorist attacks or threats. Highly enriched uranium is one of those materials. Low enriched uranium cannot be used directly to make weapons and therefore is widely regarded as a more benign material for civil nuclear commerce and use.

3. RERTR Program

The Department of Energy's Reduced Enrichment for Research and Test Reactors (RERTR) program, which has successfully developed low enriched uranium silicide fuels and is continuing to develop alternative new low enriched uranium fuels, is vital to the foreign research reactors participating in the DOE Acceptance Program. The RERTR program has provided the research reactor community with viable alternatives to the highly enriched uranium fuels which most of us were using before the RERTR program was started in 1978. The RERTR program thus makes an invaluable contribution to the longstanding U.S. policy, which, starting with the "Atoms for Peace" program launched in the

1950's, has been to support the peaceful uses of nuclear energy around the world. Our research reactors provide isotopes for medical, industrial, and agricultural applications around the world and to service various power plant operations. They are also used in scientific research and for training the next generation of nuclear scientists and engineers.

The successful development of the new low enriched molybdenum uranium fuels under the RERTR program is essential to a number of research reactors, which plan to continue operating after 2006, when the DOE Acceptance program is now scheduled to end. These fuels, when they become qualified and are licensed in the various user countries, can be reprocessed in France. The shipping of spent fuel to France provides a much-needed means of disposition of the spent fuel. The low enriched silicide fuels, which many research reactors are now using, cannot be reprocessed in France for technical reasons. Once DOE's Acceptance Program ends, we would be left without a "back-end solution" for our spent fuel and most of us would be forced by our governments to shut down our reactors if the molybdenum-uranium fuel is not available in time.

4. **Time Scale of Molybdenum – Uranium Fuel**

There have been major unavoidable and unanticipated delays in the development of the new low enriched molybdenum-uranium fuel. Argonne National Laboratory, which is developing this promising new fuel in cooperation with various vendors, is pressing ahead as fast as they can. However, we understand that the earliest date when the new fuel could be fully qualified is 2006. The original target had been 2003. Even if there are no further slippages beyond 2006 and the fuel is qualified by then, 2006 is much too late for all of the reactors in our group. We cannot make the necessary investments to build the reactor core for a fuel that has not been produced, tested, or qualified. The appropriate nuclear regulatory bodies in each of our countries must license the fuel.

In addition, each of us must satisfy our licensing authorities that we have a specific, viable plan for spent fuel disposition. Finally, after our reactors are operating on molybdenum-uranium fuel, we need three years to complete the remaining shipments to the United States of the last of our silicide-based fuel. (DOE's Acceptance Program as it is presently configured allows three years beyond 2006 for the return of the remaining silicide spent fuel. This "grace period" allows time for fuel that was being used in the reactor in 2006 to be removed from the core, cool down, and then prepared for shipment.)

DRY STORAGE OF THE BR3 SPENT FUEL IN THE CASTOR BR3® CASK

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ABSTRACT

The BR3 reactor was the first PWR plant installed in Europe. Started in 1962, BR3 was definitely shut down on June 30th, 1987. Used at the beginning of its life as a training device for commercial plant operators, it was also used during its whole life as test-reactor for new fuel types and assemblies.
Most of the spent fuel was stored in the deactivation pool of the plant for more than 15 years. The reactor being now in decommissioning, it was decided to remove the spent fuel from the plant. After comparison of different solutions, the long term storage in dual purpose storage casks was selected in 1997.
The selected CASTOR-BR3® cask is designed as a transport and storage cask for accommodating 30 spent fuel assemblies. As a type B(U) cask fitted with shock absorbers, it meets the transport requirements according to the IAEA guidelines and fulfils also the conditions for cask storage.

1. Introduction

The BR3 reactor was the first PWR plant installed in Europe. It is a low rated plant with a net electrical power output of 10.5 MWe. The BR3 was definitely shut down on June 30th, 1987.
Used at the beginning of its life as a training device for commercial plant operators, it was later also used as test-reactor for new fuel types and assemblies. So Mixed Oxide (MOX) fuel, Poison containing fuel pins (Gd_2O_3) and other types of fuels were tested and present in the storage pond.
Some fuel rods, having undergone a destructive analysis, were stored in different laboratories at the SCK•CEN. In total, the BR3 spent fuel comprises the equivalent of almost 200 fuel assemblies corresponding to some 5000 fuel rods.
Beside the spent BR3 fuel, a limited number of spent fuel rods, with equivalent characteristics as the BR3 fuel but irradiated in research reactors outside Belgium and stored in other buildings at the SCK•CEN nuclear site, were added to the inventory of spent fuel to be evacuated.

Various options such as reprocessing and intermediate storage awaiting final disposal were evaluated against criteria as available techniques, safety, waste production and overall costs. Finally the option of an AFR (away-from-reactor) intermediate dry storage of the BR3 and other spent fuel in seven CASTOR BR3® casks was adopted.
Loading of the seven CASTOR® BR3 containers at the SCK•CEN followed by transport to Belgoprocess and storage has taken place at the end of 2002.

2. Inventory and preparation of the spent fuel

During its operating period, the BR3 reactor has had eleven successive core loadings. Each core loading consisting out of 73 fuel elements except the first and the second ones (32 assemblies), most of the fuel elements were irradiated during several cycles. When the BR3 reactor was shut down in 1987, most of the core loadings, except the first two ones (which were reprocessed earlier at Eurochemic) were still onsite. In total, the BR3 spent fuel comprises the equivalent of almost 200 fuel assemblies corresponding to some 5000 fuel rods. As mentioned before, the BR3 power plant was used as a test bench for prototype fuels during his lifetime; this led to the presence of a large variety of spent fuel, which were stored in the spent fuel pond. The assemblies used in the 9 latest core loadings have an hexagonal cross-section. Table 1 gives an overview of the different dimensions and characteristics of the fuel.

Table 1: Overview of the different types of fuel

Type	Rod-∅ (mm)	Cladding material	Overalll engthh (mm)	Isotopic composition (enrichment)				
				U-rods ^{235}U (%)	Mox-rods		Gd_2O_3 (%)	Ratio-rods U / Mox
					^{235}U (%)	Pu_{fiss} (%)		
Z	8.7	Zircaloy4	1235	7.1	3	3.1 - 5.0	-	36/0 Č 10/26
G	10.7	Zircaloy4	1136	5.1 - 8.6	3	3.7 - 7.0	1.35	20/0 Č 0/20
go	9.5	Zircaloy4	1136	5.0 - 8.3	0.3 - 0.7	3.7 - 10.3	3.0 – 7.0	28/0 Č 0/28

During the consecutive core loadings, a large number of rods were withdrawn from their original assembly position to perform post-irradiation examinations or another irradiation in a different assembly position. This led to a situation that 25% of the assemblies were not complete and about 380 rods were stored in temporary storage cans.
Therefore, it was necessary to organise large consolidation campaigns in order to complete these fuel assemblies. After this, each assembly was submitted to a leak tightness test using the "Wet sipping" method. The "Wet sipping" is used in several nuclear power plants and is described in an IAEA guidebook on non-destructive examination of water reactor fuel. An external independant safety inspectorate verifies the results of the leak tightness tests.

Figure 1: Canister for non-intact and individual fuel rods

For the non-intact and cut fuel rods a canister was developed to have an airtight confinement of the fuel. The canister consists essentially of a cylindrical metallic bottle, sealed by welding to guarantee an airtight confinement similar to the intact fuel pins tightness, and is provided with an inner tubing basket. The tube arrangement serves to lock the physical location of the non-intact fuel rods, preventing the creation of a critical mass by accumulation. Each canister can contain 15 equivalent fuel pins. The loading, welding and testing of the canisters were performed in a hot cell.

3. CASTOR BR3®

The CASTOR BR3® is derived from the CASTOR THTR and is designed as a transport and storage cask for accommodating each 30 BR3 spent fuel assemblies. As a type B(U) cask fitted with shock absorbers, it meets the transport requirements according to the IAEA regulations and fulfils the conditions for cask storage in Germany. For the transport, the cask needs a Type B(U) licence from the Belgian competent authority.

The necessary modifications to the original THTR design refer to an additional neutron shielding in the cask wall, lid and bottom as well as to an adapted design of the basket. The basket is designed with closed loading tubes and can be handled separately by a crane, even in loaded condition. This allows unloading a cask if some leak on the primary lid is detected.

The main cask components are:

- ü The cask body constructed out of ductile cast iron foreseen of an internal cavity, which is coated to avoid corrosion. For neutron shielding, 30 axial boreholes are machined in the cask wall, and are filled with polyethylene rods.
- ü The primary lid, equipped with an inner metallic and outer elastomer seal, closes the cavity and acts as first independent safety barrier both for the leaktight confinement of the fuel and for the radiological shielding.
- ü The secondary lid, also equipped with an inner metallic and outer elastomer seal, serves as an additional shielding as well as a second independent tightness barrier.
  A pressure sensor system is mounted in the secondary lid to monitor the pressure in the inter-lid space hence providing a continuous check of the leak-tightness of the CASTOR cask during the long-term storage period.
- ü The inner basket consists of 30 hexagon tubes, which are mounted on a bottom plate. The tubes are made of borated stainless steel and the bottom plate of stainless steel and the latter is provided with some small holes (diameter approx. 20 mm) for dewatering purposes. Apart from these small holes in the bottom plate, the basket structure is closed and designed for being handled in loaded condition.

Table 2: Dimensions and weights of the CASTOR BR3 parts.

Part	Height (mm)	∅ (mm)	Wall thickness (mm)	Weight (tonne)
Cask body	2493	1428	370	20
Cask cavity	1652	690	-	-
Primary lid	270	911	-	1.3
Secondary lid	70	1122	-	0.6
Basket	1638	685	-	1
Protection plate	66	1370	-	0.8
Inventory	150	(Hex) 85	-	0.8

A continuous monitoring of the pressure in the interlid space is carried out, to check for potential primary or secondary lid sealing leakage (the interlid space being slightly pressurized to a pressure of 6 bar). Finally an independant seal has been placed on each cask closure by the EURATOM/IAEA for safeguard purposes.

Due to the diversity of the spent fuel and in order to simplify the calculations, 4 fuel cases were defined which covered the most important and worst case of loading. The main characteristics of these fuel cases (residual heat production, neutron and gamma radiation) were used to perform the safety calculations of the CASTOR BR3. These four fuel cases are covering each assembly and canister, which will be evacuated in a CASTOR BR3®.

To obtain the necessary transport licence, a complete safety report has been issued by GNB (the manufacturer of the CASTOR BR3®). The following main items have been checked according to the IAEA regulations for the safe transport of radioactive material (1996 Edition No ST-1):

- ü Study of the sealing system and activity retention of the Castor BR3®.
- ü Criticality calculations
- ü Shielding calculations
- ü Mechanical analyses under accident conditions
- ü Thermal analyses

For the storage licence, additional safety calculations had to be performed like:

- ü Cask behaviour in case of an earthquake
- ü Mechanical interaction of casks in the storage facility
- ü Airplane crash onto the lid system
- ü Drop of the cask in the storage facility
- ü Thermal analyses of the cask in the storage building
- ü Shielding calculations for the cask storage building
- ü Long-term behaviour of the cask components

Figure 2: Casting of the first CASTOR BR3®.

The fabrication of the Castor BR3® started in March 2000 and was finished in December 2001. The quality assurance on the design and the fabrication is performed according to the GNS/GNB quality management system based on the international standard DIN EN ISO9001. In addition to this QA follow up, a spot check follow up was performed by an independent organism (AIB) on behalf of the SCK•CEN.

4. Time schedule of the project

The major steps of the project can be summarised as follows:

- mid 1995-1997: evaluation of different options for a dry storage of the spent fuel BR3 resulting in a decision to store the spent fuel in CASTOR BR3® casks at the Belgoprocess nuclear site (central Belgian interim radioactive waste store)
- 1998-mid 2000: preparation of licensing documents for transport and storage
- mid 2000-end 2001: fabrication of 8 CASTOR BR3® casks
- 1999 – beginning 2001: preparation for evacuation of fuel assemblies
- mid 1999 - June 2002: evaluation of licensing documents by Belgian safety authorities
- mid 2001 – end of 2001: construction of storage building
- May 2002: cold test for loading and storage
- from September 2002: transport and storage of 7 CASTOR BR3® casks.

5. Experience gained during the execution of the loading and storage operation

A lot of improvements could be implemented after the evaluation of a cold test, which allowed us to verify the procedures and the different equipments to be used and also to optimise the radiation protection. As a result, all seven loading, transport and storage operations were performed without any problems and on schedule.

Figure 3: Loading under water and storage of the CASTOR BR3® cask (one can notice at the top of the casks, the connections for the pressure sensor)

As mentioned before, a complete safety report for transporting and storing the CASTOR BR3® casks had to be prepared. When we compare the measured values with the ones given in this safety files we can state that conservatism is well incorporated in these evaluations.

Table 3: Comparison between calculated and measured characteristics

	Calculated	Measured	Ratio
Contact dose rate (ɔ + neutron)			
Cask side wall	468 µSv/h	80 µSv/h	÷ 6
Cask lid side	53 µSv/h	5 µSv/h	÷ 10
Cask bottom	670 µSv/h	52 µSv/h	÷ 13
Temperature rise	20 ° C	5 ° C	÷ 4
Dose rate storage building.	2.0 µSv/h	0,2 µSv/h	÷ 10

6. Conclusions

Studies were launched to evaluate all possible solutions for the BR3 experimental and "exotic" spent fuel, i.e. reprocessing, dry storage in containers and dry storage in canisters. These studies were intended to compare the different options against criteria like available techniques, safety, waste production and overall costs. For the BR3 spent fuel the dry storage in CASTOR BR3® containers was chosen.
The spent fuel has been evacuated as radioactive waste and the casks are stored in a dedicated building at the Belgoprocess nuclear site.

The back-end management of research reactor fuel is very specific and special attention should be paid on record keeping of historical information, in order to have a clear view on the fuel inventory characteristics.

This project allowed the SCK•CEN to build up an important know-how in the field of spent fuel management. Especially the management of research reactor fuel, while this fuel is very specific and not directly comparable with spent fuel of commercial nuclear power plants.

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U.S. SPENT FUEL TRANSPORTATION SECURITY IN THE POST 9/11 WORLD

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ABSTRACT

Following September 11, 2001, many new questions have been raised in relation to the security measures for nuclear activity. The level of security increased to another dimension because of the threat of terrorism. Also, competent authorities have realized that transportation of radioactive and nuclear material is very sensitive because the material was transported on public roads, rail, sea and air during its transit from the shipper to the receiver facility. In the United States, the formation of the Homeland Security advisory system and Interim Compensatory Measures from the NRC, along with other security measures, have affected the way we transport spent nuclear fuel and fresh uranium.
This paper describes the challenging and demanding ways that security is planned, provided and maintained in support of spent fuel shipments in the United States. It also presents a status of the impact of the new security rules on the Foreign Research Reactor Return program to DOE-Savannah River Site.

Introduction

The regulatory framework for transport of spent nuclear fuel in the United States is contained in the US Code of Federal Regulations, Title 10 for the Nuclear Regulatory Commission (NRC) and Title 49 for the Department of Transportation (DOT). While historic shipments (pre-1990) were made in strict compliance with the regulations, the Urgent Relief and Foreign Research Reactor program shipments were subject to a number of "extra-regulatory" requirements intended to assure the safety and security of the materials. As time passed, a process of selection took place where some of the requirements were retained and extended to other shipments, while others were abandoned.
Around the year 2000, spent fuel transport once again was receiving elevated attention due to impending Government action on approval of Yucca Mountain as the spent fuel repository for the United States. As other grounds for appeal were lost, opponents of Yucca Mountain turn to the public risks associated with transportation as the basis of their opposition. This was the environment on September 11th when multiple coordinated acts of terrorism redefined the threat scenarios for those assigned to protect public safety. Investigations since September 11 have evaluated the likelihood of various threats, potential safety impacts of the threats on varying designs of nuclear packages, and the need to reassure the public as to the safety of commercial transport of hazardous materials in the United States. The effects of these assessments on requirements for transport are still emerging. Due to the length of time required for rule making (making formal changes to the Code of Federal Regulations), changes made by the NRC to date have been in the form of Regulatory Advisories (shortly following 9/11), and more recently in the form of Interim Compensatory Measures. This process is resulting in near term uncertainty as to the specific requirements to be met for a given shipment but will no doubt have lasting effects on our future way of doing business.

2. Safeguards & Security Recommendations and Regulations

The regulations for protection of Nuclear Material in transit in the United States are contained in 10CFR 71, providing the package design requirements, 10CFR73 providing for security of materials in transit, and 49CFR171 – 180 providing transport requirements for hazardous materials in transit. In

addition, DOE and international shipments are also subject to DOE Orders and Directives and the IAEA standards respectively.
More specifically, security requirements during the transport of spent nuclear fuel in the United States are defined in 10 CFR 73.20 thru 73.37, and 73.72. These include:

- Protection of Safeguards Information. This includes the composite transportation physical security plan, schedules and itineraries for specific shipments, details of vehicle immobilization features, alarms & communication systems, capabilities of Local Law Enforcement Agency (LLEA) response, locations of safe havens, limitations of communication systems, and procedures for response to safeguards emergencies.
- Records
- Planning and Scheduling
- NRC Route Approval. This includes identification of the shipper, consignee, carrier, LLEA, fuel & packaging information, primary & alternate routes, planned rest & vehicle fueling stops, safe havens, and escort instructions.
- Communications Center
- Notifications to US Nuclear Regulatory Commission (NRC) and states along the route.

Commercial shipments of spent nuclear fuel must be made in compliance with the regulations above. From a practical standpoint, pressure from specific States or contractual requirements imposed by DOE or other shippers has often dictated that transporters comply with additional requirements. Those in most common use include satellite tracking during a shipment, and imposition of the Commercial Vehicle Safety Alliance (CVSA) Level VI inspection, immediately prior to movement of spent nuclear fuel. This inspection is often referred to as the "enhanced" CVSA inspection. It is different from other motor vehicle inspections, in that it allows "zero" tolerance on all inspection criteria prior to initiation of transport.

3. New Requirements Since September 11, 2001

Since the terrorist attacks of September 11, 2001, the focus on homeland security, and the international environment associated with the military actions currently underway in Afghanistan, an in-depth review of nuclear material shipments, both international and domestic, was initiated by several agencies within the US Government. This review continues today and ultimately can be expected to culminate in rule-making action to amend Titles 10 and 49 to the Code of Federal Regulations.
However, near term actions in response to the attacks and resultant intelligence information include the NRC issuance of several Safeguards and Threat Advisories in order to strengthen responses to a potential attack on a nuclear or regulated activity. Nuclear transportation advisories, outlining additional security-related steps to be taken during the transport of spent nuclear fuel, have also been issued. The NRC has also communicated with other federal, State and local government agencies and industry representatives to discuss and evaluate the current threat environment, in the process of conducting a review of its safeguards and security programs and requirements. Pending rule making, the NRC has determined that certain compensatory measures are required to strengthen the protection given to nuclear material shipments against the threat of radiological sabotage. On October 10, 2002, the NRC published in the Federal Register, an order for implementation of these advisories, which have been combined and refined into a final document, titled "Interim Compensatory Measures (ICM) for the Transportation of Spent Nuclear Fuel". In addition, on December 6, 2002, Implementation Guidance for Interim Compensatory Measures for Shipments of Spent Fuel Greater than 100 Grams was issued to the licensees and their transport providers. These documents deal with the current environment and dictates requirements for safeguards in the elevated threat situation presently experienced. However, the content is designated Safeguards Information and details are released on a "need to know" basis. As a result, we are precluded from addressing any of the detail in the heightened security environment other than to acknowledge that it exists. These measures have been determined by the NRC to be *"prudent, interim measures to address the current threat environment in a consistent manner."* The requirements are to be met by November 2, 2002 or upon the next shipment, whichever is later.
Sufficient to say, today's expanded security measures, including satellite systems, enhanced and early preplanning and communications to appropriate responses forces prior shipment, vehicle safety

inspections, LLEA involvement and federal oversight, ensure the safe transport of spent nuclear fuel in the United States. More consideration is also required for contingencies, training, personnel screening, etc.
These requirements will remain in effect pending notification from the NRC that a significant change in the threat environment has occurred, or the NRC determines other changes are needed. Permanent measures will be issued after completion of the full assessment of the threat and the potential risk from sabotage of a spent fuel shipment.

4. Impact of the New Requirements on the FRR Program

The spent fuel shipments performed under the Foreign Research Reactor return's program are more complex than those above since all of these shipments are international. As a result, they must comply with the US, DOE and international regulations discussed previously. In addition, many different organizations are involved in an FRR shipment:

- DOE Headquarters
- DOE Operations Offices at Savannah River and Idaho
- US Department of State
- Other US Federal Agencies (Federal Railroad Administration, US Coast Guard, Department of Transportation - Research and Special Programs Administration, Nuclear Regulatory Commission, Federal Bureau of Investigation, Environmental Protection Administration, Federal Emergency Management Agency, etc.)
- Reactor Operators
- Transportation Contractors
- Foreign Governments

Because of the complexity of the interfaces between the multiple organizations and the need to clarify their roles and responsibilities, DOE dictated that special planning be applied. For the DOE's Transportation Services contract as a whole and for each shipment, a transportation plan and a security plan were required. Master Transportation and Security Plans were the first tasks required of the Transportation Services contractors. The shipment specific security plan, based on a Master Security plan, is issued because the threat to DOE's assets will vary depending on the location, timing and route selected. This plan, the details of which are Safeguards Information, addresses the following issues:

- Roles and Responsibilities
- Shipment Schedule
- Shipment Tracking
- Communications
- Threat Assessment
- Physical Protection Requirements
- Emergency Preparedness

Shipments of spent nuclear fuel under the Foreign Research Reactor program have always been subject to elevated safety and security requirements. Before the restart of the return program in 1994, DOE performed an extensive safety, security, and environmental assessment in order to ensure that the shipments would meet public and governmental expectation for protection of the population. This evaluation is documented in the "Final Environment Impact Statement (EIS) on a Proposed Nuclear Weapons Nonproliferation Policy concerning Foreign Research Reactor Spent Nuclear Fuel" EIS DE/EIS-218F, February 1996 (Reference 1). As part of this process, a lengthy and extensive set of public hearings and meetings with States' representatives transpired. The outcome of this process was a set of negotiated "extra-regulatory" protections considerably exceeding the requirements of the regulations in effect at the time. Early shipments under the Urgent Relief Program were distinguished by having airborne surveillance, a security inspection (lead locomotive) of the entire rail route from receipt port to DOE receiving site, local law enforcement escort for the locomotive, on-board escort, satellite tracking, additional notifications beyond those in the regulations, and enhanced inspections of all elements of the transport path, cranes, locomotives, rail cars, and tractor-trailers. While many of these requirements have been deleted in recent shipments, many still remain. As a result, the level of security as defined by DOE for the Return's Program shipment has routinely exceeded the regulatory requirements. Consequently, the ability of the program to adapt to interim compensatory measures has

been quite straightforward and not unlike the program response to other extra-regulatory transportation protections imposed on the FRR shipments.
As a result, FRR shipments have continued to be performed, in compliance with regulatory requirements and those imposed by NRC Order, and accommodated without significant cost or schedule impact to the program. Interestingly, a ship was on its way to Charleston on 9/11. The casks were unloaded at Charleston and safely transported by train to Savannah River Site using the security protections enforced by the shipment security plan. Three other shipments from Europe have been performed successfully in December 2001, June 2002 and September 2002. More shipments are expected this year
The 9/11 attacks have not impacted significantly the process used to perform the shipments. That process dictates preparation of a security plan based on the threat assessment applicable to the shipment. While in the past, this has primarily been focused on international threats; the homeland security threat level is now also a factor. No *"Other Than High Income Country"* shipments have been performed since 9/11, due in part to the security threat in candidate shipping countries.

5. NAC's Response to New Requirements

Safeguards and security expertise has always been a core capability in the inventory of disciplines that NAC offers its customers in meeting their transportation needs. NAC has a history of performing very high-level security activities for the Government. This has included NAC's current work in North Korea, our prior work in Iraq and the Republic of Georgia, as well as during FRR shipments from Colombia, Indonesia, Thailand, Philippines, and other areas of the world subject to in-country or transportation security threats. Consequently, NAC has developed a comprehensive security posture that includes DOE security clearance for NAC personnel, classified information storage capability, secure telephone and fax capability, and the full spectrum of all other transportation security provisions. NAC's response to NRC's new requirements has been planned and implemented in a manner similar to that with which the "extra-regulatory" FRR requirements were addressed. We see nothing in the Interim Compensatory Measures that NAC would not be able to meet consistent with customer shipment schedules. We also understand that we need to be responsive and anticipate any additional changes for strengthening security levels during transportation that might be prompted by further escalation in the threat level. We understand that reactive Communication, real time tracking, contingency procedures and training are key aspects for improvement of the protection of nuclear shipments.
Consequently, NAC has decided to operate its own Communication Center in our Headquarters in Norcross, GA. This Communication Center has been operated for the first time in January 2003 for a shipment of spent MTR fuel elements from Brookhaven National Laboratory, NY to Savannah River Site, SC. Tracking, and communication equipments have been implemented. Also, follow-up and contingency procedures have been developed with personnel training. This shipment have been continuously tracked and appropriate notifications performed. NAC Communication Center will be operated for all the spent fuel shipments performed in future. NAC has been able to avoid any significant cost impact in implementing the measures in force to date, due to the inventory of capabilities available in-house. However, at some level, impact on shipment cost is inevitable.

6. Summary

The planning process used for the FRR program and the extra-regulatory security provisions used on FRR and other sensitive Government programs have prepared NAC well in responding to the NRC directives and to customer requirements. Because of NAC's comprehensive security capabilities, changes to date have been absorbed with little impact. However, the process of intelligence collection and threat assessment continues. NAC and other transportation service providers will continue to be tested by changing security requirements as the process of rulemaking and interim compensatory measures matures.

A LOOK AHEAD

AN EVALUATION OF THE ADEQUACY OF THE TRANSPORTATION INFRASTRUCTURE TO HANDLE POTENTIAL RETURNS FOR THE BALANCE OF THE FRR PROGRAM

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ABSTRACT

Original DOE projections for Foreign Research Reactor spent fuel to be returned to the United States under the FRR program identified approximately 23 metric tons of eligible material to be shipped to DOE's Savannah River Site and Idaho National Engineering and Environmental Laboratory. Since the initiation of receipts under the "urgent relief" program, only about 4 metric tons of the 23 identified have been shipped. The paper will evaluate the potential range of candidate material likely to be shipped to the US and the adequacy of the transportation infrastructure to support the range of material shipments. Availability of certified casks and vessels, ability of receipt facilities to accept shipment rates, potential budgetary constraints and other considerations will be evaluated. Decisions by reactor operators that might affect the availability of transportation and receipt capability, such as delaying shipment until late in the program term, will be highlighted.

1. Introduction

The United States Department of Energy (DOE) adopted the Nuclear Weapons Nonproliferation policy concerning Foreign Research Reactor Spent Fuel in May 1996. The Final Environmental Impact Statement (EIS) issued in February 1996 estimated about 20 metrics tons of heavy metal representing approximately 22,700 elements to be eligible for return to the US between 1996 and 2006. Eligible fuel is required to be discharged from the reactor by 2006 and the transportation window is open until May 2009 in order to allow cooling time prior shipment. 104 Research and Test Reactors located in 41 countries are eligible for inclusion in the EIS.

In December 2002, DOE selected NAC International (NAC) to be its sole transportation services contractor for the balance of the program. This has placed a unique responsibility on NAC to understand the range of participation likely, the timing for potential shipments, and the capability of the transportation infrastructure to fully support the program.

2. Current Status

In the 7 years of implementation, the Foreign Research Reactor (FRR) Spent Nuclear Fuel (SNF) Acceptance Program has completed 25 shipments safely and successfully. 27 countries have participated so far, returning a total of 5,537 irradiated fuel elements to the United States for management at the DOE sites in South Carolina and Idaho. 23 of the 27 shipments were delivered to Savannah River Site (SRS) in South Carolina. The other 4, containing TRIGA® spent nuclear fuel elements, were transported to Idaho National Engineering and Environmental Laboratory (INEEL). A total of 143 casks (131 to SRS, 12 to INEEL) have been utilized to perform these shipments. Based on the initial expectations of the EIS, 32 percent of the MTR fuel element shipments and 20 percent of the TRIGA® fuel element shipments have been completed after seven years of program implementation.

3. Evaluation thru 2009

Unless the FRR Fuel Return Program is extended beyond 2009 by the United States Government, only three years remain for candidate fuel irradiation and six years for transportation. The participation of the international community in the fuel return program is presently expected to be considerably lower than that estimated in the EIS. We have evaluated an updated potential range of candidate material to be shipped to the United States.

- **MTR Fuel Elements**

A bounding number of 17,800 MTR fuel elements were originally estimated to be shipped to SRS. As of the end of 2002, 4,576 MTR fuel elements have been transported to SRS, with a bounding number of 13,000 being eligible for shipment during the 6 remaining years. Some countries, however, have not yet confirmed their total or even partial participation in the program. This is the case for Belgium, France, Israel, Pakistan, UK, Netherlands, South Africa, Iran, Brazil, and Canada. Of these countries, Belgium* (representing 1,766 elements), Canada* (representing 2,831 elements) and France* (representing 1,982 elements) have large amounts of uncommitted fuel that could significantly impact transportation estimates. It should be noted that some of the above countries have indicated they will (or may) participate; however contracts with the United States Government have not yet been executed. Furthermore, uncertainties exist for some other countries like South Korea that now have the status of High Income Country and about 500 fuel elements remaining to be shipped. With these considerations, the range in possible shipments is very large but the expected amount of fuel to be transported by May 2009 is only around 4,200 to 4,500 fuel elements.
*Based on EIS estimates.

- **TRIGA® Fuel Elements**

A bounding number of 4,900 TRIGA® fuel elements was estimated to be shipped to INEEL. 961 TRIGA® fuel elements have been transported to INEEL with nearly 4,000 still eligible for the remaining years of the program. Participation of some countries such as Mexico, Malaysia, Brazil, and Bangladesh is not currently expected. Our current evaluation is that approximately 1,500 to 2,000 remaining TRIGA® fuel elements should be transported to INEEL by 2009.

4. Cask Usage

Based on the experience of the program to date and reasonable expectations for the near term, we can consider an average of 35 MTR or 80 TRIGA® fuel elements per package to be representative of typical cask capacities for the program. Consequently, we can estimate that the number of cask shipments to be transported between 2003 and May 2009, based on the expected fuel return estimates, should roughly be:

- 120 casks for MTR
- 25 casks for TRIGA®

This means that an average of 24 cask loads of research reactor fuel should be shipped to the United States each year.

5. Transportation Infrastructure Adequacy to Meet the Potential Shipment Schedule

- **Cask Availability**

The type B fissile casks currently licensed and used under the FRR program are the TN-7/2, GNS-16, GNS-11, GE 2000, TN-MTR and the NAC-LWT casks. The LHRL120 casks and JAERI casks are licensed, but are only used for specific needs of their owners (ANSTO and JAERI, Kyoto University, etc.).

The table below identifies main information on the available casks.

Spent Fuel Cask ID	Number of Casks	Type of Fuel Elements	Comments
TN-7/2	2	MTR, DIDO	
GNS-16/GNS-11	4	MTR, TRIGA®	
TN-MTR	1 or 2	MTR, DIDO	
GE-2000	2	MTR, TRIGA®	Limited foreign validation
NAC-LWT	8	MTR, DIDO, TRIGA®	
TOTAL	**17 or 18**		

Considering that 8 additional JAERI casks are available for an annual shipment, the average number of 24 casks/year should be considered as 16 casks/year. Each cask should be able to perform a minimum of two shipments a year which means that the cask availability appears consistent with the expected program need and current DOE unloading rates. However, participation by any of the large uncommitted fuel sources will strain cask availability.

- **Cask Licensing Issues**

Licensing issues can and do frequently have significant impact on cask availability and utilization. All the casks mentioned above are at least licensed to the requirements put forth in the 1985 IAEA Safety Series. The TSR-1 regulations adopted under the IAEA in 1996 will be applicable in Europe in 2004 with a possibility to continue using 1985 designs with a Multilateral Approval. The U.S. Nuclear Regulatory Commission (NRC) is currently reviewing the TSR-1 regulations for adaptation to the U.S. Code of Federal Regulations. The final rules should be issued and adopted by the NRC in 2003 for implementation no later than 2006. No major immediate impact is expected on the casks mentioned above. Since most spent fuel casks are licensed for only a three to five year period, most of casks will need to be re-certified and consequently will need new foreign validations in the countries in which they are used.

Due to the variations in the characteristics of research reactor fuel elements, additional licensing and possibly equipment manufacturing will be required to transport fuel elements not covered under the current approvals. Twelve months is typically required for licensing efforts, including license application preparation, issuance of the certificate of compliance in the country of origin and then validation in all countries where the cask is used or is in transit. This means that advanced planning remains a key consideration for on-time shipment performance. This will be even more important will as the 2009 shipment window is tightening. For example, none of the casks mentioned above can accommodate Canadian NRU type or KMRR South Korean fuel elements should decisions be made to include them in the program. Another aspect to consider is cooling time authorized for each fuel form and each cask. Research reactor operators will need to plan their shipments to account for the minimum cooling time allowed for their selected casks and for the expected shipment volume since the last two or three years of the program are expected to see extremely high cask utilization rates.

Cask licensing should not be an issue to meet the program needs but it remains a sensitive aspect and only good anticipation will preclude issues with new or “not yet shipped” fuel types. To prevent any difficulties, it is strongly recommended that the research reactor ensure the compatibility of their fuel elements with the casks selected to transport them a minimum of two years in advance of their intended shipment.

- **Transportation Availability and Capacity**

Approximately 5 or 6 INF-2 and INF-3 ships are expected to be available for the coming shipments. It is unlikely that any additional ships will become available. In 2000, DOE was successful at increasing

the number of casks allowed in single shipment from 8 to 16. As a result, it is not expected that vessel capacity or quantity will be a limiting factor in the transportation model. As in the past, surface transportation (including rail and truck) is not expected to be a limiting factor in the fuel return program as well.

- **Transportation Security Issues**

Our paper entitled "Security post 9/11" describes the impact of the new security measures in the US since 9/11. Historically, the level of security applied by DOE for the Return's Program shipments has exceeded the regulatory requirements. Consequently, the ability of the program to adapt to the increased requirements reflected in interim compensatory measures has been quite straightforward and has not impacted significantly the process used to perform the shipments. However, we also understand that we need to be responsive and anticipate any additional changes for strengthening security levels during transportation that might be prompted by further escalation in the threat level.

6. Environment - Threat Risk – Safety

The new homeland security threat level implemented after 9/11 is now also a factor. No *"Other Than High Income Country"* shipments have been performed since 9/11, due in part to the security threat in candidate shipping countries. In addition, the RERTR meeting scheduled for 2001 in Bali was cancelled due to concerns relative to terrorism. It is difficult to predict how it is going to evolve in future. This security threat might impact the program. However, the FRR fuel return program substantially contributes to limiting the proliferation threat of nuclear materials. This certainly adds U.S. and international political support to the program.

7. Summary

The existing transportation infrastructure should allow research reactor operators to successfully transport their fuel elements to the United States. However, this conclusion is based on shipments being performed on a regular basis and not waiting for the two last years of the program. This is also based on current estimates of participation, omitting the high quantity fuel producers mentioned earlier. Comprehensive licensing and shipment schedule planning is also a key factor for completion of the shipment during the remaining window. Shippers need to confirm compatibility of their fuel elements with the available casks at least two years before their intended return date. The process of intelligence collection and threat assessment continues. Security threats in candidate shipping countries and throughout the international community are difficult to predict. Recent schedule impacts from the aftermath of 9/11 are likely to continue and could even grow more significant should additional threats or events become a reality. Reactor operators need to carefully consider shipping as much fuel as possible as early as possible. It is very likely that some eligible fuel will not be able to be included in the program due to "as late as possible" shipment strategies being considered by some reactor operators.

Our challenge is to meet all of your shipment needs for the remaining years of the program. Anticipation, communication, planning and teamwork between DOE, NAC and the international research reactor community are the key factors to our success.

RETURN TO THE RUSSIAN FEDERATION OF IRRADIATED FUEL ASSEMBLIES FROM THE INSTITUTE OF NUCLEAR PHYSICS, REPUBLIC OF UZBEKISTAN

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ABSTRACT

The Russian Federation is jointly implementing with the USA and the IAEA the program providing return to the Russian Federation of irradiated fuel assemblies from foreign research reactors constructed with the assistance of the former Soviet Union. The main goal of the program is to diminish the stockpile of highly enriched uranium in the form of fuel assemblies at research reactor sites, it also attempts at convincing the countries with operational research reactors to switch over to a lower enriched fuel.

The research reactor at the Institute of Nuclear Physics, Uzbek Academy of Sciences was elected to be the first candidate for a "pilot" transfer to the Russian Federation of irradiated research fuel assemblies. The reason for this selection is the institute's maximum preparedness in the institutional and technological aspects.

The irradiated fuel assemblies will be returned in full compliance with the provisions of the recent Russian legislation on management of foreign irradiated fuel assemblies. The legislation endorses a mandatory state ecological expertise of the uniform project for the return of irradiated fuel assemblies and implementing specialized programs for environmental rehabilitation of radiation contaminated sites.

1. Foreign spent nuclear fuel from research reactors, built under technical assistance of the former Soviet Union Introduction

In the 60-ies and 70-ies of the past century the former Soviet Union carried out an intensive construction program of research reactors abroad. In contrast to commercial reactors decisions on the feasibility of research reactors construction were mainly based on political grounds. As a result a large number of research reactors was built in developing countries with unstable political and economical systems, which did not contribute to the non-proliferation goals.

As a rule, the reactors were supplied with fresh fuel assemblies, containing HEU (36-90 % enrichment for U-235). At the beginning of the 70-ies the Soviet Government took a decision to start the process of converting foreign research reactors to lower enrichment levels, however with the disintegration of the USSR these works found no continuation.
Normally, spent fuel from foreign research reactors was not returned to the country-manufacturer with the exception for the SNF from the former Soviet republics and the return operation of the spent fuel assemblies from the Iraq Nuclear Research Center in Baghdad, that was accomplished in 1993 by air in Accordance with the UN Security Council Resolution. Accordingly, to the current moment large quantities of highly enriched SNF assemblies have been accumulated at the research reactors' on-site SNF storages.

In May 1997 the USA announced the start of a large-scale SNF return program from foreign research reactors, built to US-designs. The program covered research SNF from 41 states to be finally disposed of on the US territory.

The major program goal is reduction of HEU quantities, contained in SNF assemblies at the research reactor sites and convincing the states, operating the given reactors to convert them to low-enriched fuel.

However, from the US perspective the issue of nuclear proliferation risk reduction in the area of research reactor fuel can not be resolved completely without the involvement of the Russian Federation, the second largest world supplier of corresponding facilities, technologies and fuel.

In view of this, starting from 1999 “Techsnabexport” (the company, authorized by the Government of the Russian Federation to conclude foreign deals related to the management of the spent nuclear fuel from foreign nuclear reactors) has carried out several rounds of negotiations with the operators of foreign research reactors. Preliminary consultations have shown a keen interest of a number of foreign states to perform regular shipments of their spent research nuclear fuel to the Russian Federation. However, taking into the account limited budgets of the research reactor operators to finance operations on the spent fuel management and in some cases their complete absence, large-scale fuel shipments seem to be hardly possible without the assistance of various sponsors (IAEA, USA, EUROATOM, private funds, etc.).

The United States have expressed their readiness to provide financial assistance to the customer-states in returning their fuel to Russia on condition of a gradual transition of the research reactors operators to fuel with enrichment of less then 20 % for U-235).

At the same time trilateral meetings in the frame IAEA-Russia-USA started to discuss possibilities for the return of the spent nuclear research fuel to the Russian Federation.

As the result of technical missions of the IAEA-Russia-USA experts to the research reactor sites in Uzbekistan, Ukraine and Yugoslavia, performed in 2001, the Republic of Uzbekistan was selected to be the first state, from where the return of the spent fuel assemblies could be executed. The complex political situation in Middle Asia and the maximum technical and organizational readiness for the take-back operation of the Republic of Uzbekistan were taken into consideration.

2. Spent nuclear fuel of the research reactor of the Republic of Uzbekistan Academy of Sciences Institute of Nuclear Physics

The Republic of Uzbekistan Academy of Sciences Institute of Nuclear Physics was founded in 1956. For the present moment it is one of the largest scientific institutes in Asia. Currently the institute carries out fundamental and applied research in nuclear and elementary particle physics, solid body physics, radiochemistry, biology, element analysis and many other areas.

Initially the power of the VVR-SM reactor, which was put into operation in September 1959 was 2 Mwt. In the soviet times the reactor was used both for military and civil purposes. In 1971-1979 the reactor was upgraded and its power was increased to 10 Mwt.

From 1959 to 1971 the reactor was loaded with fuel assemblies with the enrichment up to 10 % for U-235. After modernization in the period from 1979 to August 1998 the reactor was powered by fuel assemblies with the enrichment up to 90 % for U-235. In September 1998 the specialists from the “Kurtchatov Institute” converted the reactor operation from fuel with 90 % enrichment to fuel with 36 % enrichment.

Starting from October 2000 works are being carried out on further enrichment reduction down to 19,7 % for U-235. Apparently, future activities on conversion to low-enriched fuel will depend on terms and volumes of financial assistance, allocated by foreign sponsors. Return of spent fuel from Uzbekistan was organized in 1974-1991. In total 449 fuel assemblies were transported to the Soviet Union during that period.

The Government of the Republic of Uzbekistan is consecutively pursuing the policy of turning Central Asia into a nuclear weapons-free zone and is naturally perplexed with the presence of sizeable quantities of highly enriched nuclear material on its territory. The particular danger is the location of the research reactor close to the combat zone in Afghanistan (the reactor is located only 137 km away from the Uzbek-Afghan border).

In accordance with the agreement an IAEA mission with US and Russian experts participation was organized to the research reactor site in July, 2001. In the course of the mission the complete check of the transport-technological scheme was performed, including spent fuel assemblies loading into the transportation casks TUK-19, transportation of TUK to the railway station by motor vehicle (distance – 7 km) and the loading of TUK-19 into special carriages. The transport-technological scheme has proven itself viable. There is still personnel available at the reactor, who participated in past time return operations, as well as the equipment used during those transportations.

The overall quantity of spent nuclear fuel assemblies (mainly with the high level of enrichment) stored currently at the on-site storage facility is 259 pieces. 256 assemblies are planned to be transported to the Production Complex “Mayak” of the Russian Federation (the reprocessing plant for the research and commercial spent nuclear fuel) in 4 passages. There exist no difficulties of technical character for reprocessing the given types of FA at the “Mayak” plant

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3. Legal aspects for the return of the spent nuclear fuel assemblies from the research reactor of the Republic of Uzbekistan Academy of Sciences Institute of Nuclear Physics

As it has been mentioned before, return of spent fuel from foreign research reactors to the Russian Federation was not possible in the 90-ies not only due to financial difficulties of the reactors operators, but also for reasons of legal character.

The Order for the import into the Russian Federation of foreign reactors spent nuclear fuel, enacted by the Government of the Russian Federation in 1995, is not valid for research reactor fuel.

Only in the framework of the new Russian federal legislation in the field of foreign reactors spent nuclear fuel management, adopted in 2001, an opportunity has emerged for the implementation of the research reactor spent fuel tack-back program.

Currently work is underway to elaborate and harmonize legislative acts of the Russian Government that would in detail regulate the order of import into the Russian Federation of the foreign reactors spent nuclear fuel, that would be closely connected with various special environmental programs for the clean-up of radioactively contaminated areas. To this day only two out of five Government acts have been approved off.

Taking into the account the request of the Republic of Uzbekistan Government to provide assistance for the take-back of the spent nuclear fuel assemblies from the Institute of Nuclear Physics in 2003 and in view of the absence for the present moment of the full package of legislative acts the Government of the Russian Federation has issued a special Direction to study the possibility of the tack-back of the

spent nuclear research reactor fuel from the Institute of Nuclear Physics of the Academy of Sciences of the Republic of Uzbekistan on condition of a positive finding of the State Ecological Expertise on a set of documents, substantiating general reduction of radiation impact and upgrading the environmental safety level as the result of the tack-back into the Russian Federation of the given SFA and execution of special environmental arrangements at the Production Complex "Mayak", that would be financed from the means, received from the contract on the SFA import.

4. Preparatory works related to the return into the Russian Federation of the SNFA from the research reactor of the Republic of Uzbekistan Academy of Sciences Institute of Nuclear Physics

Adoption by the Government of the Russian Federation of a special Direction allowed starting preparatory arrangements for the organization of the return operation from the Institute of Nuclear Physics.

A contract has been signed, which is currently being implemented, between "Techsnabexport" and the Institute of Nuclear Physics on the preparation of a set of documents, substantiating the possibility for the return of the research nuclear spent fuel, that would be subject for the evaluation and approval of the State Ecological Expertise. The major scope of works in the framework of the given contract is being financed by the US party.

The works are being performed in 3 stages:

Stage 1 – Preparation of the set of documents.

Stage 2 – Obtaining of approvals on the levels of Ministries and Offices of the Russian Federation.

Stage 3 – Evaluation and approval of the State Ecological Expertise at the Ministry of Natural Resources of the Russian Federation.

In case of a positive conclusion of the State Ecological Expertise the contract will be signed on the return of 256 spent nuclear fuel assemblies from the Institute of Nuclear Physics to the Russian Federation.

Practical implementation of the given project will facilitate:

- Strict adherence of the Russian Federation, in its nuclear weapons state status, to the safeguards mechanism for non-proliferation of nuclear weapons.
- Support of stability in the region of Middle Asia.
- Implementation of ecological clean-up measures for rehabilitation of regions, radiologically contaminated as the result of the nuclear arms race in the 50-ies and 60-ies of the past century.

THE TEST RIG FOR INSPECTION OF SPENT FUEL ASSEMBLIES OF RESEARCH REACTORS OF THE STATE SCIENTIFIC CENTER OF THE RUSSIAN FEDERATION RESEARCH INSTITUTE OF ATOMIC REACTORS

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ABSTRACT

The report describes the technical characteristics and possibilities of the multifunctional test rig for inspection of spent fuel assemblies (SFA) of the research reactors (RR) stored in the water pools at the central storage facility (CSF) of spent nuclear fuel (SNF). The test rig (TR) will allow us to check the presence of the fuel assembly in the storage, to assess its self-protection, to identify the fuel assembly and the fuel type, to detect failures of the structural integrity during long-term storage and to check the fuel rod cladding leak-tightness. It is supposed that the TR constructed under the program of upgrading and automatization of the material control and accountability (MC&A) system at the RIAR will substantially reduce the possibility of unauthorized handling of spent nuclear fuel.

1. Introduction

The systematic control and accounting of nuclear materials becomes an urgent task in different countries and in the industrial enterprises. Control over handling of fissile materials and SFAs of the RRs includes the following operations:
-Arrangement of the automated circulation of documents, accounting of the certificate data by the irradiation conditions, calculation of the amount of the fissile materials and the decay heat during discharge from the reactor;
-Control over storage and movements of the SFAs;
-Random inspection of the SFAs at the CSF and at the intermediate storage of the RR using special portable equipment;
-Shipment of SFAs for long-term storage (outside the RR) with the attached certificate containing all the attendant data, calculation values of the fission products (FP) activity and the decay heat at the moment of shipping;
-Cladding integrity control (CIC) of fuel rods in the SFA, visual inspection, confirmatory measurements, weighing when received at the CSF;
-Check of the nuclear material (NM) quantity in the inventory items and the authenticity of data declared in the accounting documents during the long-term storage of SFAs.

2. Purpose and Application of the Test Rig

The complication of the task of the systematic control and accounting of nuclear materials in the RR's is conditioned by the variety of the fuel types and irradiation conditions. Most of the RIAR research reactors have been in operation for more than 30 years. Within this period the reactor fuel has undergone several upgrades. After 3-5 years of storage in the intermediate pools of the reactor facilities, the SFAs are sent to the CSF for "wet" long-term storage. The CSF consists of three pools, transport entry and the service rooms. Pools No.1 and No.2 are used for storage and pool No.3 for

installation of the containers and loading/unloading of SFAs. The pools are filled with distilled water. The maximum depth of the reloading pool is 10.5 m. The crane trolley rated at 100 ton and the bridge crane rated at 15 ton are used for handling the SFAs at the CSF. All the transport and process operations are carried out under the water layer. The storage facility has certain locations for storing the SFAs from each research reactor. Pool No.3 consists of the receiving cell that is used for accommodation of the transport containers and the refueling cell intended for refueling of the SFAs. More than 4000 SFAs are currently stored at the CSF. Table 1 provides some characteristics of the RR SFA and the type of their fuel composition. Most of the SFAs contain the highly enriched fuel.

Table 1. Parameters of RR SFAs stored at CSF.

RR	**Fuel**	**Enrichment, % U^{235}**	**Number of FR in FA**	**Cladding material**	**Average storage time, years.**	**Average burnup, U^{235} at. %**
SM	UO_2[1]	90	188	Stainless steel	19.6	37.9
MIR	UO_2	90	4	Al alloy	16.1	37.5
	UAl alloy		4		23.4	40.6
			6		28.3	38.4
VK-50	UO_2	2-3		Zr alloy	21.6	44.2
ARBUS	UO_2	90	3-5	Al alloy	16.2	4.4
	UO_2, Al	90			19	2.1
	UAl_3	36			33.6	5.9
	UAl_3UAl_4	90			22.6	3.9
BOR-60	[2]	90	37	Stainless steel	21.2	11.5
	[3]	56	37		16.9	10.8
	[4]	75	37		10.4	14.2
	[5]	90	37		21.2	7.5
	[5]	75	37		6.2	12.5
	[6]		7		11.1	

Note:
1. Fuel dispersed composition of the cermet type: UO_2 in copper matrix with the beryllium bronze additive (sintered mixture of copper and beryllium powders).
2. Fuel of sintered UO_2 bushings.
3. Mechanical powder mixture (80%UO_2 + 20%PuO_2).
4. Homogenized fuel (80%UO_2 + 20%PuO_2) produced by melting method.
5. Mechanical mixture of UO_2.
6. Blanket assemblies containing sintered UO_2 pellets (depleted or natural U).

During the long-term storage the fuel assemblies can lose the inherent self-protection and be of interest for the terrorist organizations thus becoming an object of theft. This mostly determines the requirements imposed to the material control and accounting system at the RIAR CSF. The main data on the NM quantity used for accounting records and drawing up the material balance are the total NM mass of each type and mass of uranium and plutonium isotopes. The measurement system must provide check/confirmation and in some cases determination of these values in order to identify them with the accounting data. The measurement of the attribute characteristics of SFAs is very important.
It is supposed that the above listed tasks will be solved using the TR developed at the RIAR. The TR is designed to check the condition of the SFAs from the VK-50, BOR-60, ARBUS, MIR and SM research reactors stored at the CSF as well as those received for storage and sent for reprocessing or some other kind of storage.

The purpose of inspecting the SFA is to check its actual availability, identification by the type and kind of fuel, detection of the structural failures, cladding integrity control of fuel rods. To realize this purpose it's necessary to perform the preliminary preparation of the SFA. The fuel assemblies received at the TR are subject to visual examination, weighing and gamma-scanning. The identification number area of the SFA is subject to the preliminary mechanical cleaning. The visual inspection of the SFA outer surface or its wrapper using the video system enables to identify its number or some other identifier, to control the structural integrity and change of the fuel assembly geometry. Then the SFA is subject to the CIC followed by washing and drying. The CIC of the fuel rods allows us to make a decision on the further storage conditions of the fuel assembly. Washing and drying of the SFA are the necessary stages of the work involving the inspection of fuel assemblies and transfer of SFA for "dry storage" or shipment for reprocessing. The cladding integrity control can be provided by registration of the long-lived radioactive fission products with the greatest fission yield from the failed fuel rods. The most suitable for this purpose is ^{85}Kr the half-life period of which is 10.72 years.

3. The TR Design and Layout of the Equipment

The TR consists of the four main components, each of which is placed in an individual case. The case protects the unit from mechanical damage and serves as a biological shielding too.

1. The measuring unit (Fig 1)

It is a low protective metal cylinder, that accommodates gamma-detector with collimator, radiation dose meter and four video cameras. The video system of the process monitoring of the SFA provides visual inspection of the fuel assembly surface in order to perform its identification by the outer appearance and number. It is also intended to detect the defects, to determine the geometrical dimensions, to control keeping of the SFA in the grip and to document the inspection process. The video system_provides an all-round vision of the SFA and consists of five black-and-white TV cameras with optics and electronic units for processing and storage of the video images and mechanical components (channels and periscopes). Gamma-scanning system_provides gamma scanning of the SFA along the length. The measuring unit is placed on the refueling cell of pool No.3

2. The mechanical cleaning unit (to clean the SFA identification number region) with the grips keeping the SFA during cleaning (Fig 1)

The mechanical cleaning unit is made in the form of the mechanical brush placed on the shaft that is set in motion through the bevel gear from the alternating current motor. The brush can move to the assembly axis depending on the assembly diameter.

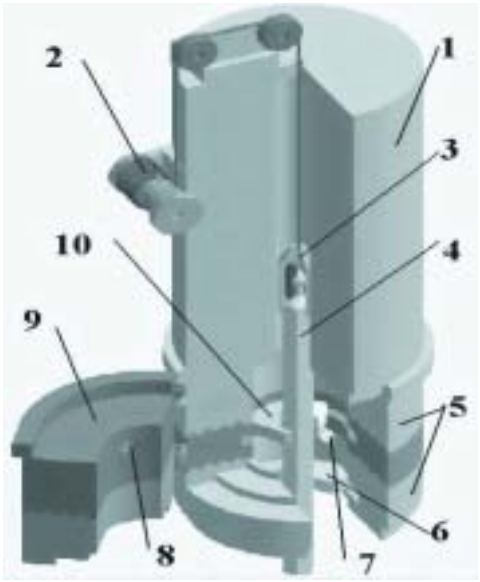

Fig 1. Measuring unit, refueling container and mechanical cleaning unit

1-Refueling container;
2-Container winch with weighing system;
3-SFA lifting Grip; 4-SFA; 5-Measuring and mechanical cleaning units;
6-Mechanical brush; 7-TV camera hole;
8-CdZnTe detector with collimator;
9-Removable shielding;
10-Grips for keeping SFA during cleaning.

3. The CIC system, washing and drying unit (Fig 2)

The CIC, washing and drying unit is placed in the immediate vicinity of the refueling cell. It is a casing that accommodates all the auxiliary equipment. The casing provides radiation protection from the fuel assembly under inspection. At the top of the casing there is a leak-tight lid, set in motion by the drive placed on the outer casing surface. The heater is placed in the middle of the casing. The purpose of the CIC system is to detect the failure of the fuel rod in the fuel assembly by measuring the activity of the long-lived gaseous FP. Inside the vessel there is a tank to accommodate the SFA. The tank ensures the reliable isolation of the fuel assembly from the environment. The tank has a mechanism keeping the SFA in the upright position. Due to its design the tank can be fast removed together with the SFA in case of its failure.

Electric heating and automatic temperature control system is intended for stimulation of the gaseous FP release from fuel assemblies. The maximum temperature on the SFA surface can achieve 230^0C. Washing and drying system is intended for the technological preparation of the SFA for the inspection. The washing system is used to prepare the washing solutions, to supply them into the tank and to remove into the special sewage system with further washing of the tank by the distilled water. Drying provides for the possibility of vacuuming the tank containing the fuel assembly and drop of the pumped air into a special ventilation. The SFA are washed by the solutions running through the tank. The surplus moisture can be removed from the SFA by heating or by vacuuming the working cavity.

4. The refueling container with weighing system (Fig 1)

Weighing system provides weighing of the SFA in the range from 5 to 255 kg with an accuracy of 0.1 kg.

The TR is placed in pool No.3 nearby the standard refueling cell. The flowing detector of the CIC system and all the secondary equipment, the server of the measuring system and the control panel of the video system are installed in the non-contaminated room. The refueling container of the lower loading and the special grips are used for transportation of SFA from the refueling cell to the TR. Spent fuel assemblies 68-200 mm in diameter and 910-2550 mm in length can be examined in the TR.

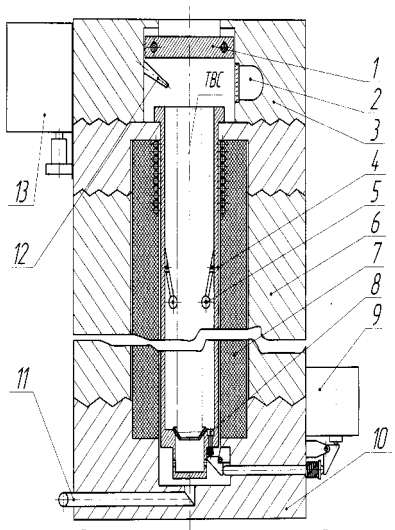

Fig. 2 The CIC system, washing and drying unit.

1-Upper leak-tight lid; 2-Video camera for SFA surveillance; 3-Biological shielding; 4-Removable tank of CIC system; 5-Fixtures holding SFA in vertical position; 6-Casing; 7-Heater; 8-Drainage valve; 9-Drainage valve drive gear; 10-Lower part of the casing; 11-Outlet collector; 12-Inlet collector (supply of water, washing solution, air); 13-Lid drive gear

4. Description of the TR Operation

According to the standard procedure, the SFA is placed into the refueling cell, which is kept free before inspection. The refueling container is installed on the measuring unit. The grip sinking from the refueling container is linked with the fuel assembly and raised to the identification number region where it is stopped to perform its mechanical cleaning.

The position for the proper fixing of the fuel assembly is determined visually using the video system. The weight of the fuel assembly is fixed by the weighing system placed on the refueling container. Upon completion of the mechanical cleaning the video system automatically identifies the SFA number and records its image and information about the SFA on the computer disc of the video system.

Then the SFA is raised with a certain step into the refueling container. The raising step is determined by the opening angle of the video camera periscopes and the detector collimator, which must be adjusted for each group of SFA in dependence on the required sensitivity parameters of the measuring system.

The portable unit equipped with the CdTe gamma-detector for inspection of SFAs in the intermediate storage was tested in the BOR-60 reactor [1,2]. At present the detectors of this type are rather widely used for safeguard measurements [3]. The investigations suggest that it is possible to measure the isotopic composition by the relation of the intensities of the characteristic gamma-lines from the main plutonium and uranium isotopes in the SFA radiation spectrum. Based on measurement of the SFA activity and dose rate in the fixed geometry it will be possible to assess the degree of the SFA self-protection, to make a conclusion on the type of fuel and in some cases to assess its burnup, to distinguish one type of the SFA from another or a mockup from the irradiated fuel assembly. Moreover, the SFA gamma-spectrum received during measurement performed with the inspection purpose can be an additional identifier of the SFA and can be attached to its certificate as a characteristic or tracer inherent to this or a group of similar SFAs.

5. Summary

With an advent of the SFA TR the SSC RF RIAR will improve and automate the MC&A system of spent nuclear fuel of the research reactors and will close the chain of the MC&A systems starting from the receiving of “fresh” fuel and finishing by shipping the spent nuclear fuel for reprocessing. A set of measures described above and the use of the test rig will reduce the possibility of an unauthorized handling of nuclear materials and their theft at the stage of the long-term storage.

6. References

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A NEW GENERATION IN THE FAMILY OF PACKAGES FOR TRANSPORTATION

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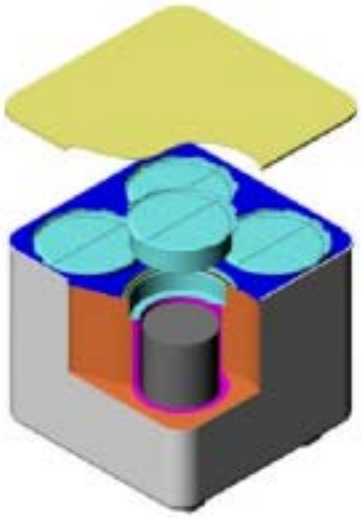
ABSTRACT

A new generation of packages has been developed by COGEMA LOGISTICS to meet the various needs and requirements of the Laboratories and Research Reactors all over the world, following the 1996 Regulations. A large quantity of packages existing till then had been used successfully for many years but could not comply to the changing regulatory context. These new packages are adapted to various options, and are able to propose worldwide and long-term solutions for international transports of all kinds and sizes of materials from and to any site

1. TN-UO2

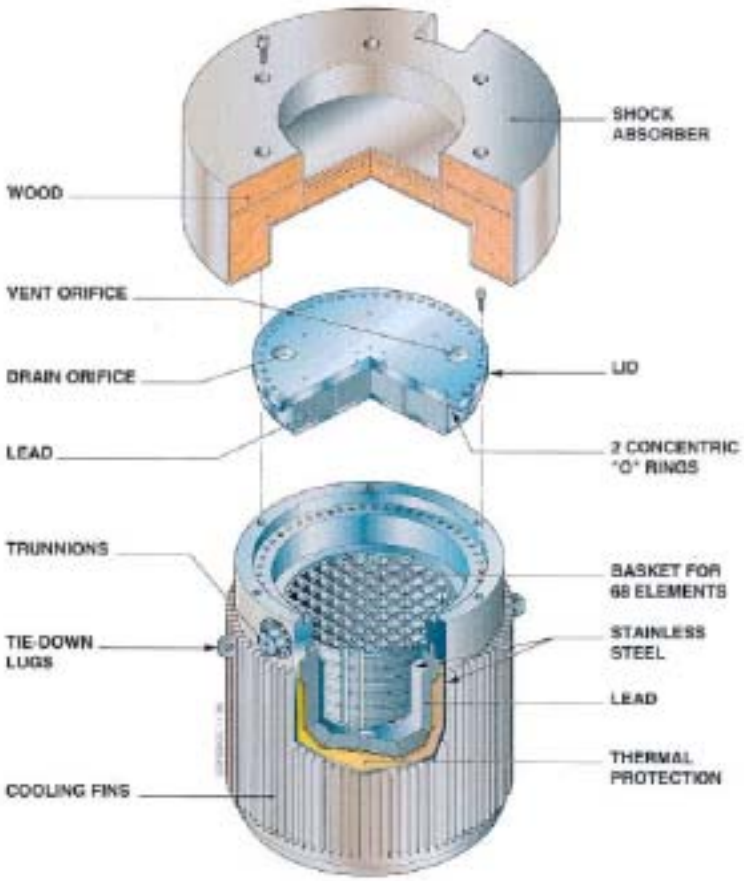
The TN-UO2 is a stainless package meeting IAEA 1996 ST-1 requirements, type A fissile, which can be used to ship several kinds of non irradiated materials, including reprocessed uranium. Its multipurpose design and internal equipment enables to accommodate with many different contents, such as UO_2 powders, UO_2 pellets, as well as wastes and metallic uranium up to 10 % enrichment. The TN-UO_2 is rather a small package, user friendly, and easy to handle. Its outer dimensions are: diameter 400 mm, and height 805 mm. The cavity inner content dimensions are: diameter 260 mm, and height 580 mm. The inner capacity is 23.7 liters. Its maximum gross weight is 95 kg, and the maximum payload is 32.5 to 38 kg, depending on the enrichment. The TN-UO2 is used today on a daily basis in Europe, with more than 600 packages owned by French Companies: FBFC, COGEMA LOGISTICS. The package has also received in September 2002 an agreement by the French Authorities for air shipments, which has been validated in Australia. Due to increasing needs of transport of higher enrichments, an extension of the existing agreement has been applied for solid uranium enriched up to 20 %, including air shipments. As there are also further needs for high-enriched uranium, another extension is expected for 2003 for metallic uranium with enrichments up to 95 %, including air shipments. The TN-UO2 is presently licensed in France, Sweden, and Australia. Validations are in progress in Belgium, the Netherlands, Germany, the United Kingdom, the USA, and Canada.

2. TNF-XI



The concept and the technology of the TN-UO2 can be versatile, and can be adapted to fit some specific needs or requests: a derivative version of the package, called TNF-XI, has been designed and manufactured to meet Japanese market specific requirements. There are presently 800 of these TNF-XI packages to be used in Japan. The TNF-XI package is already licensed in France and Japan for contents up to 10 % and in Great Britain for contents up to 5 %. Its agreement or validation is in progress in the USA, Canada, Sweden, Belgium and Great Britain for contents up to 10 % enrichment.

3. TN-MTR



The TN-MTR is a B(U)F package which has been developed for the shipments of spent fuel of Research Laboratories. The TN-MTR presents a cylindrical cavity, in which today 7 different kinds of

baskets are available to comply with the characteristics of the spent fuel to be shipped. The body is covered with cooling fins, and a shock absorber is fitted on the top of the package, in transport configuration. Up to 76 assemblies can be shipped in one package. The outer dimensions of the package are: 1,610 mm height, and 1,600 mm diameter without the shock absorber. With the shock absorber, the height is 2,080 mm and the diameter 2,008 mm. The inner cavity has a diameter of 960 mm and a height of 1,080 mm.

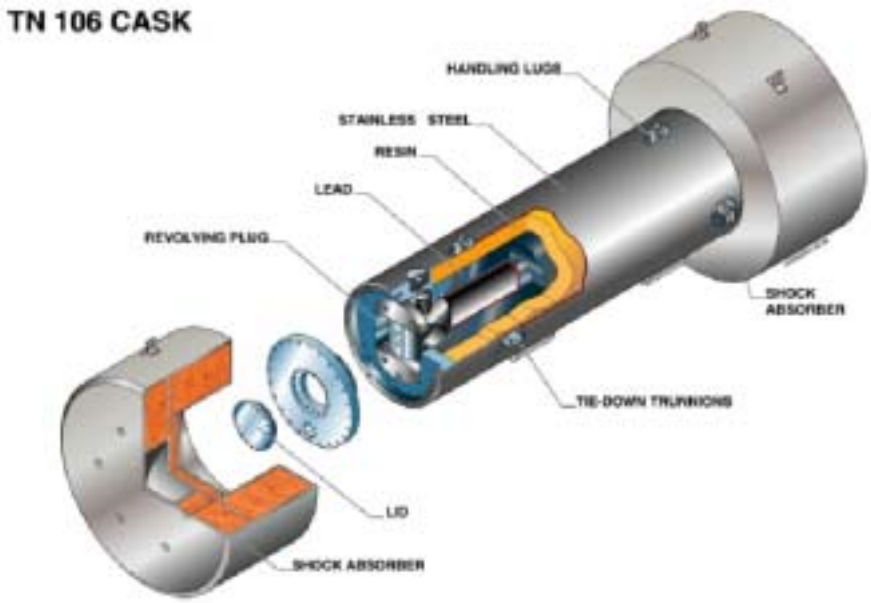
The 7 different baskets which can presently be used are:

- RHF: with 3 cylindrical holes for RHF fuel,
- MTR-4: with 4 cylindrical holes for inner containers of diameter 336 mm maximum,
- MTR-44: with 44 square holes for elements,
- MTR-52: with 52 square holes for elements,
- MTR-52-S: with 52 sleeved square holes for elements, especially designed for the USA,
- MTR-61: with 61 square holes for elements,
- MTR-68: with 68 square holes for elements + 8 for smaller elements.

Other baskets can be created to suit new needs. 4 packages belong to French owners (COGEMA LOGISTICS, CEA) and are daily used, particularly in France and Belgium. Several spot shipments are made in other parts of the world: European countries, the USA, Australia.

The TN-MTR concept enables either dry or easy wet loading and unloading on the sites. A transfer system can be used in order to facilitate the operations. A recent example of international operations took place in 2002, with a shipment of spent fuel from Denmark to the US. The wet loading of the fuel took only one hour at Risö. The delivery at Savannah River was completed to the great satisfaction of the US DOE. A renewed agreement has been obtained in France and Belgium, and validations are in progress in Germany, Denmark, Australia and the USA.

4. TN-106



The TN-106 is a brand new B (U) F package, meeting IAEA TS R1 regulations, and used for transport of irradiated fuel rods and pins in replacement of the old TN-6 family:

- irradiated fuel rods and pins with uranium oxide enriched at 10 % maximum,
- irradiated fuel rods and pins with plutonium oxide or MOX with 12.5 % Pu maximum,
- fuel rods and pins from Fast Breeder Reactors with 45 % Pu maximum,
- non fissile solid radioactive materials,
- fuel elements with solid metallic uranium mixed to other metals (MTR, Triga, UNGG).

It enables both dry or wet loading and unloading, either in vertical or horizontal position, in order to meet the needs or requirements of Research Reactors or Laboratories, worldwide. There is something exclusive, which makes the TN-106 a very original package: its design is based on a modular concept, allowing manufacturing of a series of packages with various useful cavity length from 1,000 to

3,200 mm, in accordance with the Research Reactor or Laboratory interface, needs or requirements. A variety of internal arrangements may be designed for the TN-106 packaging, such as baskets, racks, capsules, etc. Specific internal arrangements can permit an increase of allowable fissile mass.
The TN-106 has received in May 2002 a 5 years agreement from French Authorities for shipments by road, rail and sea. It is the first time that a license authorizes variable length of cavity. This agreement has been validated in Switzerland, and other validations are presently under way in the US, Denmark, Sweden, the Netherlands and Germany.
For transport purpose, the TN-106 is tied-down by its 4 trunnions on a transport chassis, which can be loaded into an ISO container. It could also be covered by a tarpaulin. The associated equipments consist of the transport chassis, the handling lifting beam, an IP2 box containing some tools, and the transport ISO container. 2 Handling lugs fixed onto the package enable also the handling of the package on its chassis. 2 shock absorbers are fixed on the package in transport configuration.
The outer dimensions of the package are: Overall diameter: 1,458 mm with the shock absorbers, 958 mm including the trunnions, without the shock absorber, 820 mm excluding the trunnions, without the shock absorbers. The overall length is 1,778 to 3,978 mm without the shock absorbers, and 2,424 to 4,614 mm with the shock absorbers. As for the internal characteristics, the internal diameter is 203 mm, and the useful cavity length 1,000 to 3,200 mm. The empty package weights from 5.8 to 12.4 tons without the shock absorbers. The 2 shock absorbers weigh each 0.6 ton, and the overall loaded package with the 2 shock absorbers weigh from 7.2 up to 14.4 tons, depending on the length.
COGEMA LOGISTICS owns presently one TN-106, whose length is 2,200 mm. Another one is likely to be built in the near future, with a length of 2,400 mm. The first test shipments have occurred in October 2002 in France, at great satisfaction of the CEA units of Saclay, Grenoble and Cadarache, and now the operational shipments are starting.

5. RD-26

The RD-26 is a stainless steel type B(U) package, which can be used for multipurpose shipments of various materials, such as:

- alpha contaminated technological wastes put into 118 liters drums (maximum 100 kg),
- UO_2 powder, pellets, or part of fuel rods put into specific drums (maximum 70 kg), 3
- liquid wastes, organic effluents and aqueous solutions in plastic bottles (maximum 150 kg).

The package concept presents one single containment barrier on the lid and orifice plug. A venting system can be proposed as an option.

The outer dimensions are: 1,145 mm height, and diameter 860 mm.

The inner dimensions of the cavity are: 780 mm height, and diameter 513 mm.

The mass of the empty package is 460 kg.

The filled package can weigh up to 530 to 610 kg maximum, depending on the contents.

The RD-26 can be handled on any site with classical means, such as forklifts, cranes and slings. Up to 12 packages can be loaded into a 20’ ISO container using a specific rack. Today, 72 packages are in operation, and more than 30 shipments are performed every year.

6. TN-CIEL

The TN-CIEL is an innovative concept of mobile tank vehicle type IP2, intended for road transportation of waste radioactive and potentially corrosive liquids: liquid effluents, concentrates of boric acid (H_3BO_3) with soda (NaOH) and phosphate, Co 60 nuclides. The tank is licensed LSA-II with respect to the IAEA recommendations, and, in Europe, with respect to the ADR road transport regulations for class 7 (radioactive materials), but also for the class 8 (corrosive materials). The particularity of the liquid wastes is that it crystallizes at ambient temperatures. Therefore it must be transported at a minimum of 60° C. The TN-CIEL tank makes it possible by heating the cavity of the tank at a temperature regulated between 60 and 70° C by electrical resistance. Moreover the TN-CIEL tank is equipped with an agitator in order to preclude precipitation in some colder zones. can be connected to one of the three manholes allows loading and unloading of the radioactive liquids in the tank from and to any container. The maximum allowable tank volume is 5 m^3 of low specific activity liquids. The TN-CIEL mobile tank is now used on a daily basis for transports operations between the 18 EDF NPP to the French incineration facility CENTRACO Marcoule. The total mass of the vehicle, under 40 tons, enables transport on any type of European road.

7. Conclusions

There are also several other projects adapted to the specific needs of several international customers. All this new range of packages meets the latest requirements of the regulations or IAEA recommendations. Far more than the former generation, this new family enables multipurpose shipments, of a multiplicity of materials, adapted to a maximum of international sites, avoiding many problems of agreements and validations of specific contents.

TRANSPORT AND STORAGE CASKS FOR IRRADIATED FUEL ASSEMBLIES FROM RESEARCH REACTORS

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ABSTRACT

For irradiated fuel assemblies from various research reactors, the Consortium NCS/GNB has designed transport casks as well as combined transport and storage casks. The transport casks of the type GNS 16 and TN 7/2 and the combined transport and storage cask of the type CASTOR® MTR 2 and CASTOR® BR3 are described in the following sections. There is a large amount of experience with both transport casks GNS 16 and TN 7/2 in the research reactors such as those in Jülich, Munich, Berlin, Geesthacht, Hannover, Heidelberg, all in Germany, Petten (NL), Risö (DK), and Sao Paulo (Brazil). Starting in the year 1998, 18 casks of the type CASTOR® MTR2 were loaded in the German reactor Rossendorf with VVER-M and VVER-M2-fuel assemblies of Soviet type as well as EK-10-fuel assemblies of Soviet type. The identical cask type has been loaded four times at the Joint Research Center in Petten, Netherlands with HFR fuel. In 2002, seven casks of the type CASTOR® BR3 were loaded for interim storage at SCK-CEN in Mol, Belgium with research PWR fuel.

1. GNS 16 Transport Cask

The GNS 16 transport cask mainly consists of a cask body made of a sandwich construction (stainless steel/lead/stainless steel) and a massive lid made of stainless steel.

The cask body is formed by the side-wall (40 mm stainless steel), the inner liner (30 mm stainless steel), the bottom plates belonging to the side-wall and the inner liner as well as the head ring. The side-wall and the inner liner are welded to the head ring. The hollow space formed by the side-wall and the inner liner is completely cast with lead. The lead thickness at the side-wall is 153.5 mm and at the bottom 135 mm.

The 325 mm thick lid closes the cask cavity with its sealing system. Together with the head ring, the inner liner and the inner liner bottom, it forms the leak-tight containment of the cask. The lid is attached by means of 16 cylindrical bolts (M30). Handling connections for draining and drying the cask as well as for the test of its leak-tightness are located in the lid. During transport, these handling connections are tightly closed by means of a closure lid and/or closure bolts.

A protection plate is installed above the lid, protecting the lid system from dust, moisture and mechanical influences during handling and transport of the cask.

Different fuel baskets can be inserted in the cavity of the cask for accommodating the following contents:

- 33 square-shaped MTR-fuel assemblies of enrichment levels HEU (max. initial enrichment: 95.1 M% U-235), MEU (max. initial enrichment: 45.7 M% U-235) and LEU (max. initial enrichment 20.3 M% U-235) with a max. decay heat of 40W/fuel assembly.

- 28 tubular MTR-fuel assemblies of enrichment levels HEU, MEU and LEU with a maximum decay heat of 26 W/fuel assembly.
- 90 TRIGA-fuel assemblies of enrichment level LEU with a maximum decay heat of 1W/fuel assembly.

In order to handle the cask, two trunnions are arranged opposite each other on the head ring. The transport of the GNS 16 cask is performed in a special 20 ft container. For this purpose, shock absorbers are installed at the top and bottom ends.

The overall dimensions (without shock absorbers) of the GNS 16 are as follows:

- Overall height: 1535 mm
- Outer diameter: 1180 mm
- Cavity height: 944 mm
- Cavity diameter: 723 mm

The mass of the loaded cask without shock absorbers is approx. 13 100 kg. The total mass of both shock absorbers is approx. 2020 kg.

The GNS 16 complies with the international regulations of the IAEA (International Atomic Energy Agency) for package designs of the type B(U) and the nuclear safety class I.

2. TN 7/2 Transport cask

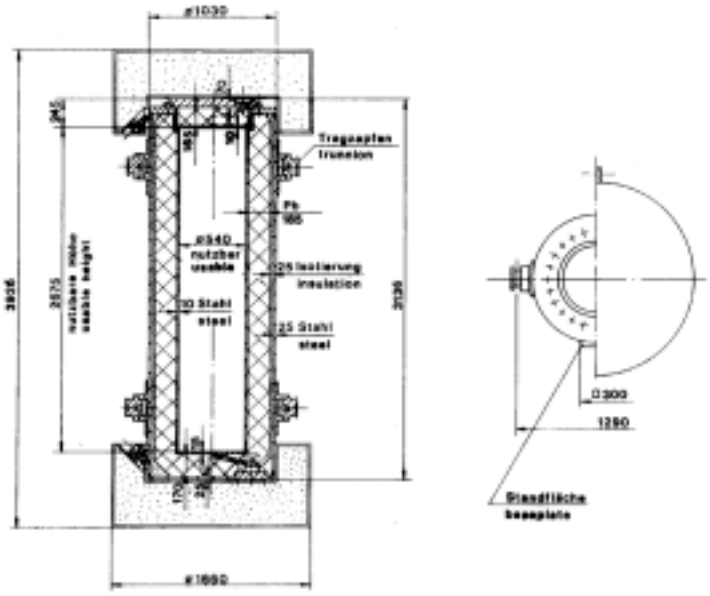

Fig 1. TN 7/2 cask.

Weights :

- Total (without transport frame)	20 720 kg
- Shock absorber (top)	780 kg
- Shock absorber (bottom)	780 kg
- Lid	760 kg
- Transport frame	1 200 kg

This cask has a maximum transport capacity of either 64 cut square section MTR fuel assemblies or 60 cut round section MTR fuel assemblies, the weight of the cask being 21 t. The U-235 enrichment varies from 20 to 93 % according to the type of fuel. The fuel assemblies are loaded under water into the transport baskets. These are then placed one above the other in the cask. Due to the weight of the cask, the latter can only be used in ponds which have sufficiently

powerful cranes. A 20' Container is used to transport the TN 7/2 so that transfer from road to rail or ship causes no problems.

3. CASTOR® MTR 2 Transport and Storage Cask

The CASTOR® MTR 2 serves for the transport as well as the dry interim storage of irradiated fuel assemblies from various research reactors. The cask mainly consists of a thick-walled cylindrical cask and a testable and monitorable double-lid system.

The cask body is made of ductile cast iron and is cast as a hollow cylinder closed on one side with a smooth surface and subsequently machined. The wall thickness of the cask body is 354.5 mm at the side-wall and 360 mm at the bottom.

The 280 mm thick primary lid is made of steel with high ductility at low temperatures. With its sealing system, it closes the cask cavity and forms, together with the cask body, the leak-tight containment under normal conditions. The primary lid is attached to the cask body by means of 28 cylindrical bolts (M30). The sealing system of the primary lid consists of a metal and an elastomer seal.

Above the primary lid, there is a second, 60 mm thick lid, the so-called secondary lid. The secondary lid forms, with its sealing system, a second, independent sealing barrier for the cask. Together with the cask body, it can form an alternative leak-tight containment. The secondary lid is also made of steel with high ductility at low temperatures. Its sealing system is similar to the one of the primary lid and it is attached by means of 28 cylindrical bolts (M30). A connection for filling the space between the primary and secondary lids with helium is located in the secondary lid as well as a pressure switch. For the storage of the cask, the inter-lid space is filled with an overpressure compared with the cavity and the outer ambience. This overpressure and thus the leak-tightness of the cask are continuously monitored by means of the pressure switch during the whole storage period.

Different fuel baskets can be inserted in the cavity of the cask for accommodating the following contents:

- 33 square-shaped MTR-fuel assemblies of enrichment levels HEU (max. initial enrichment: 93.2 M% U-235), MEU (max. initial enrichment: 45.5 M% U-235) and LEU (max. initial enrichment 20.0 M% U-235) with a max. decay heat of 25W/fuel assembly.
- 28 tubular MTR-fuel assemblies of enrichment levels HEU, MEU and LEU with a maximum decay heat of 25 W/fuel assembly.
- 90 TRIGA-fuel assemblies of enrichment level LEU with a maximum decay heat of 1W/fuel assembly.
- 147 VVER-M or VVER-M2-fuel assemblies of Soviet type with a maximum initial enrichment of 36.6 M% U-235 and a maximum decay heat of 1W/fuel assembly.
- 28 EK-10-fuel assemblies of Soviet type with a maximum initial enrichment of 10.0M% U-235 and a maximum decay heat of 1W/fuel assembly.
- 5 compact fuel assemblies with a maximum initial enrichment of 93 M% U-235 and a maximum decay heat of 165W/fuel assembly.

For handling of the cask, two trunnions are bolted to the cask body. The transport of the CASTOR® MTR 2 cask is performed in a special 20 ft container. For this purpose, shock absorbers are installed at the top and bottom ends of the cask.

For storage of the cask, a lid protection plate is installed above the secondary lid and a bottom protection plate below the cask bottom.

The dimensions (without shock absorbers and protection plate) of the CASTOR® MTR 2 are as follows:

- Overall height: 1631 mm
- Outer diameter: 1430 mm
- Cavity height: 920 mm
- Cavity diameter: 721 mm

The mass of the loaded cask without shock absorbers and protection plate is approx. 15800 kg. The total mass of both shock absorbers is approx. 2130 kg. A schematic presentation of the combined transport and storage cask CASTOR® MTR 2 is shown in Figure 2.

The CASTOR® MTR 2 complies with the international regulations of the IAEA for package designs of the type B(U) and the nuclear safety class I. Dry interim storage of irradiated fuel assemblies from research reactors in the CASTOR® MTR 2 has been approved by the German licensing authorities.

4. CASTOR BR3 Transport and Storage Cask

The CASTOR® BR3 is designed for the transport and interim storage of up to 30 spent fuel assemblies and/or spent fuel canisters from the Belgian pressurized water reactor BR3 at Mol. The cask was loaded in the wet spent fuel storage pool at the reactor site and transported by road to a nearby interim storage facility. For the transport the casks received a type B(U)F license in Belgium.

The cask consists of a thick-walled cylindrical cask body made of ductile cast iron, for which the interior cavity is closed by a double lid system consisting of a primary and secondary lid. These lids are each bolted to the cask body and sealed with metal seals.

For the proper localization of the fuel assemblies and/or fuel canisters, the cask cavity contains a basket with 30 hexagonal fuel assembly positions. For the handling of the cask during loading, unloading and transferring, there are two bolted trunnions each at the top and bottom side of the cask. During transport over public roads, the cask is provided with impact limiters at the bottom and lid side of the cask in order to reduce the shock loads during an hypothetical assumed transport accident.

In order to minimize the risk of contamination during the wet loading in the reactor storage pool, the cask and its components have been designed to ensure a proper decontamination by using smooth surfaces, filling gaps and holes with silicone rubber to avoid water access and providing a painting which is easily cleanable.

The main dimensions and handling masses are given in Table 1. Fig. 2 shows a sketch of the cask in transport configuration with both impact limiters. Fig. 3 shows a sketch of the top view of the cask without lids.

Main dimensions of the CASTOR® BR3

Overall diameter	1428	mm
Overall length	2493	mm
Thickness of the cask side wall	368.5	mm
Diameter of the cavity	691	mm
Length of the cavity	1652	mm
Diameter of the impact limiter	2136	mm
Cask length with impact limiters	3063	mm

Masses of the CASTOR® BR3

Handling configuration (Transport)	27600	kg
Handling configuration (Storage)	24400	kg

Table 1 Main dimensions and handling masses

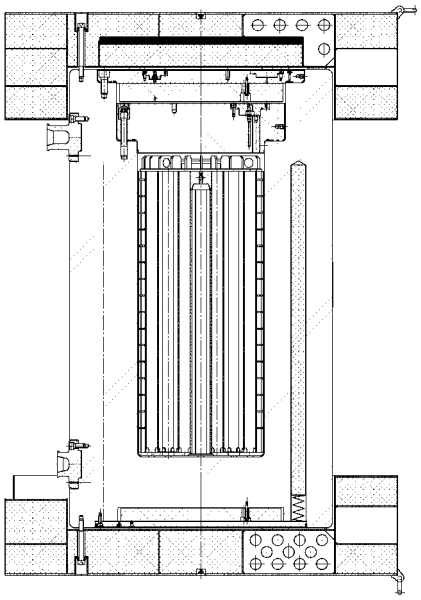

Fig. 2 CASTOR® BR3 - Transport configuration configuration, Cross section

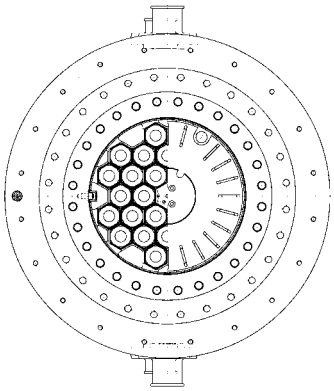

Fig. 3 CASTOR® BR3 - Transport Top view without lids

The cask lid system consists mainly of a primary lid with devices necessary for dispatching the cask after loading under water and a secondary lid with devices necessary for monitoring the leak tightness during long term interim storage. Both lids create the typical double lid system of most of the CASTOR® type transport and storage casks.

The so-called primary lid closes directly the cask cavity. It serves as the first barrier against activity release from the cask cavity and as a highly effective gamma shield. The lid is fixed by stud bolts and cap nuts. In the sealing area of the lid supporting surface there are two grooves as a seat for the lid gaskets. The inner groove is designed for the assembly of a metal gasket and the outer groove contains an elastomer gasket. With these gaskets, a check volume for the leak test of the inner metal gasket is defined. The principal sealing function for the cask cavity is performed by the metal gasket only. The primary lid contains an opening for drainage and drying the cask cavity which is closed by a lid equipped with a metal gasket for leak-tightness purposes

The secondary lid normally is put onto the cask before transport. It serves as an additional shielding as well as a second independent and long term stable leak-tight barrier. The lid is sealed analogue to the primary lid. The secondary lid contains two openings into the volume between primary and secondary lid. One opening is closed by a quick connection valve and a closure plate which is equipped with a metal gasket for leak--tightness purposes. The quick connection valve is used to adjust the specified pressure in the interlid space during the storage period. The other opening is used for mounting a pressure sensor which is used for monitoring the pressure in the interlid space in the storage facility to check the leak-tightness of the double barrier system during cask storage.

For the handling and lifting of the cask, two trunnions each are bolted to the cask wall in the lid and the bottom area of the cask. The bottom trunnions are used for securing the cask onto a transport frame for road transportation.

THE LEAKING RTR ALUMINIDE SPENT FUEL MANAGEMENT LA HAGUE REPROCESSING PLANT FLEXIBILITY

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ABSTRACT

Besides the known advantages of waste volume reduction and long term radiotoxicity minimisation, the RTR spent fuel management by COGEMA, through the proven back-end solution "treatmentconditioning" by reprocessing, offers a possible final solution for these RTR aluminide spent fuels declared unsound, "defective" or "questionable" by the reactor operator considering the concern that RTR aluminide spent fuels stored under water or in a dry storage can undergo corrosion when storage time becomes decades.
The normal acceptance criteria by the La Hague reprocessing plant require that the aluminide fuels be declared "sound" by the research reactor operator. Spent fuels which do not satisfy this condition may be submitted to additional measures or restrictions for their acceptance by the La Hague reprocessing plant.
The final conditions for acceptance of unsound RTR spent fuels by the La Hague reprocessing plant were defined by the French Safety Authority (DGSRN) after a thorough safety assessment by the IRSN.
This paper presents an overview of the fruitful cooperation between SCK•CEN and COGEMA and of their efforts to reach an approval by the French Safety Authorities.

1. Introduction

For six years now, SCK• CEN and COGEMA work close together for the spent fuel element management of the BR2 research reactor. The reprocessing of RTR spent fuel as a sustainable back-end management allows to answer to the respective national safety authorities and radwaste management entities, taken into account the general European policy of a sustainable development.

Today, 850 spent fuel elements from the BR2 research reactor were transported by road and delivered to the COGEMA La Hague reprocessing plant.

With this mutual intensive experience, SCK•CEN and COGEMA developed and proposed respectively to the Belgian and French safety authorities, a procedure leading to manage correctly and safely the specific case of the leaking fuel elements.

2. Status of the leaking fuel elements

During the operating of the BR2 research reactor, some abnormally high concentrations of fission products were occasionally observed in the primary circuit.

The procedure now foresees that at shutdown all fuel elements undergo a thorough sipping test. Some of them are then declared leaking –"PF"- on basis of acceptance criteria developed by experience.

Leaking spent fuel elements are stored for years in an interim storage pool at the reactor site.
The systems and the rigorous operating instructions in use at the BR2 site (use of demineraized water, continuous demineralization, monitoring of the water parameters such as pH, conductivity and radioactivity) guarantee optimal conditions for an interim storage.

However, experience learns that the behaviour of the RTR aluminide spent fuel elements during a long-term dry or wet interim storage is not always adequate. There is a significant potential for corrosion despite the implementation of various precautionary measures in the research reactor pools or storage wells.

It is the responsibility of the research reactor operator to define a sustainable and stable conditioning to guarantee a non dissemination of radioactive materials for the future decades.

3. Acceptance criteria

COGEMA can accept the spent fuel elements which are declared sound by the research reactor operator after a cooling time of at least one year.

The authorization licenses for reception, interim storage and processing delivered by the French safety authorities (DGSNR) require the respect of the following basic conditions:

- Before the departure of the transport cask, a radioactive control of the cask atmosphere has to be carried out to verify the spent fuel element integrity. The transport is authorized only if the measured activity of the cask atmosphere is less than 1.85 MBq/m^3.

- During the unloading operation at the reprocessing COGEMA La Hague plant, an activity measurement of the rinsing water used through the cask has to be carried out. The measured activity has to be less than 3.7 GBq/m^3. If it is not the case, all the RTR spent fuel elements have to be put in leaktight bottles.

COGEMA can, however, manage nearly all fuel declared unsound by research reactor operators, provided a careful and specific additional examination is conducted case by case.

4. Acceptance procedure for BR2 leaking spent fuels

In the framework of this specific acceptance procedure, some radioactive isotopes were selected analysis in the sipping test water.

The main radioactive isotopes analyzed were:

- Beta and gamma spectrometry
  ^{106}Ru, ^{134}Cs, ^{137}Cs, ^{144}Ce: representative long-lived isotopes still released after years of cooling;
  ^{131}I, ^{133}Xe: most representative short-lived isotopes for detection of leaking fuel elements directly after shutdown of the reactor;
  ^{85}Kr: representative long-lived gaseous isotope still released after years of cooling.
- Alpha spectrometry
  ^{234}U, ^{235}U, ^{238}U, 238Pu, ^{239}Pu, ^{241}Am: representative isotopes

A standard duration of 100 hours was defined for sipping test from various observations and measurements carried out by the BR2 reactor operators.

It was also observed that:

- the released beta and gamma activity during the sipping test is increasing linearly with time
- the released rate of ^{137}Cs is exponentially decreasing with the cooling time. with the spent fuel cooling is exponential and decreasing

Considering this experience and the results, it was agreed to adopt the following sipping test conditions:

Beta and gamma activity release rate measured during a sipping test of minimum 100 hours should be under 10 kBq/h.

The following three additional acceptance criteria were issued by the COGEMA La Hague reprocessing plant for the leaking fuel elements:

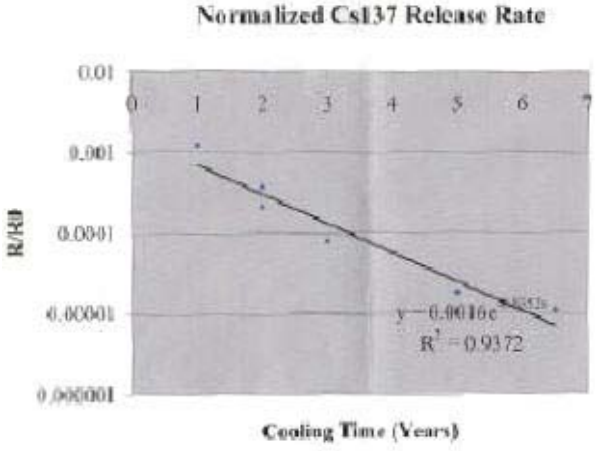
- mechanical integrity of the spent fuels;
- no dissemination of fissile material:
- beta and gamma activity release rate, measured by a sipping test of minimum 100 hours should be lower than 10 kBq/h.

As most of the observed leaking fuel elements were sip tested a first time when unloaded of the reactor, the following methodology was defined by SCK•CEN and COGEMA :

- the fuel element (B201) with the largest recorded release rate (after unloading from the reactor) is identified;
- the relative Cs137 release rate in function of the cooling time (between 1 and 6.5 years) is measured by sipping tests for the fuel element B201 and others

  (R/Ro) = 0,0016 Exp (-0,8352 x ÄT)

  where R: activity release rate, Ro: activity release rate at time 0



- the allowable Cs 137 activity release rates:
- it is established that the ^{137}Cs release is representing minimum 50% of the total released activity
- for a cooling time of 5.5 years (unlimited number of leaking fuel elements in the transport cask): (R/Ro)= 1.62 10-5

It means that to satisfy the 10 kBq/h limit at time of acceptance by COGEMA, any leaking fuel element should satisfy the following relation when sipped at time 0 (unloading from the reactor:

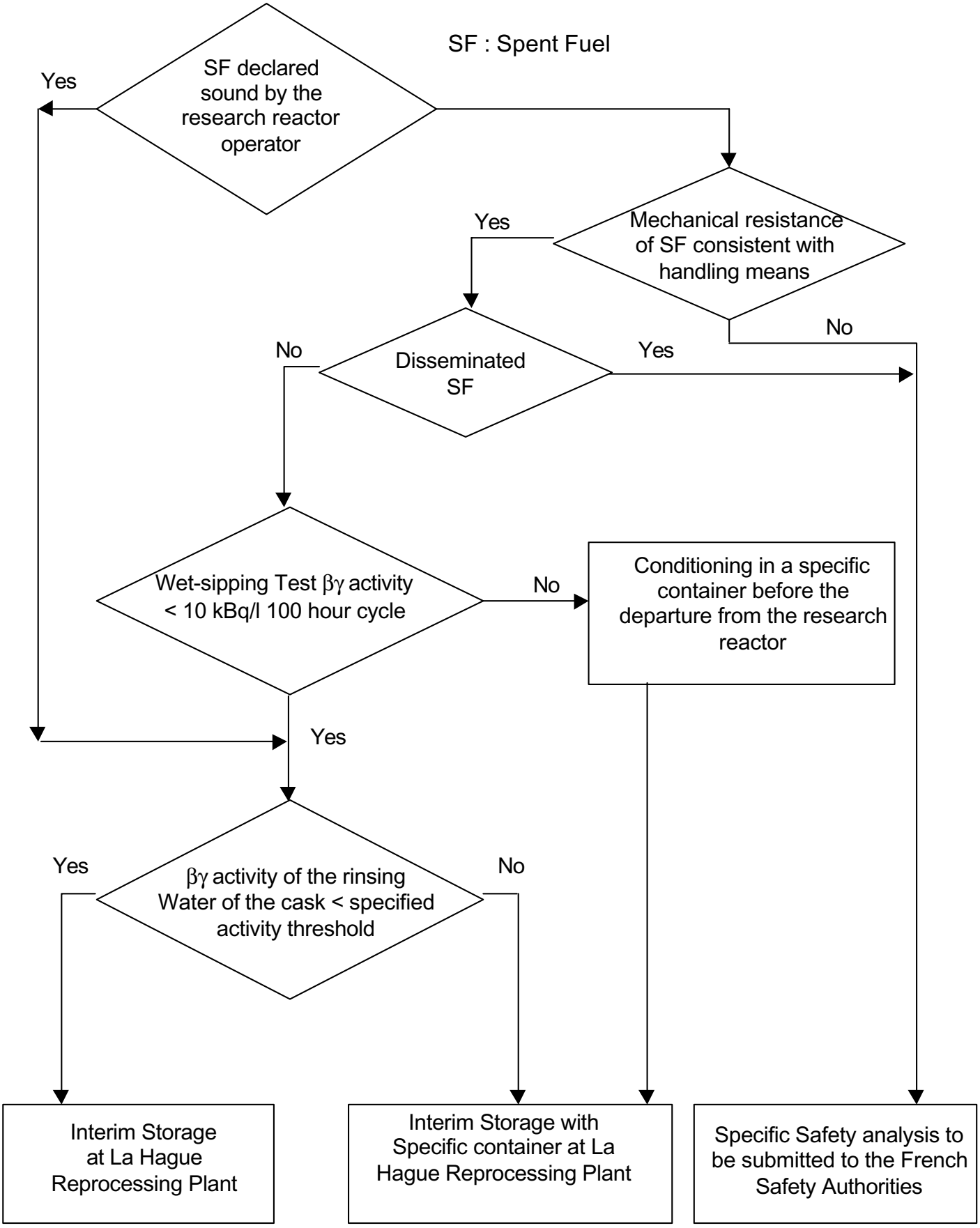
A (Bq/l) < 1.48 109 (Bq/l) 137Cs

where A is the measured ^{137}Cs volumic activity.

5. Conclusions

Additional restrictive criteria and specific analyses undertaken by COGEMA and a research reactor operator allow the acceptance of defective spent fuels by the COGEMA La Hague reprocessing plant. The experience gained by SCK•CEN and COGEMA through a specific acceptance procedure of some spent fuels declared leaking –"PF" is a concrete example. Thanks to a fruitful collaboration, leaking spent fuels could be safely transported and received by the COGEMA reprocessing La Hague plant during the year 2002, respecting strictly the French safety authorities requirements.

Annex: Diagram on the leaking spent fuel acceptance



TRANSPORTATION FOR REPROCESSING OF THE SPENT NUCLEAR FUEL (SNF) OF TVR ITEP RESEARCH REACTOR AND PROPOSALS FOR SNF MANAGEMENT PLANS FOR THE RA REACTOR

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ABSTRACT

The TVR heavy water research reactor was deployed at Moscow Institute of Theoretical and Experimental Physics. In 1990, the final batch of the spent nuclear fuel from this reactor was shipped to Production Association (PA) "Mayak" for reprocessing. The SNF removal was a stage of the reactor decommissioning activities.
The designs of the TVR reactor and its fuel elements are similar to the RA reactor designs. Two ways of the RA reactor SNF transportation to PA "Mayak" have been considered: in aluminum barrels and in additional canisters using respectively TUK-32 and TUK-19 shipping casks. The practical experience and the equipment used to prepare for the TVR reactor SNF removal can be helpful to the RA reactor personnel in finding the best way to perform these engineering operations.

1. Introduction

The 2.5 MW TVR heavy water research reactor (Institute of Theoretical and Experimental Physics, Moscow) was commissioned in 1949 and was successfully operated up to 1986. Then, after its examination, decommissioning activities were started.
During the reactor operation period, spent fuel (fuel elements (FEs) along with channels enclosing them in the reactor core) were transferred using a standard technology from the core to the spent fuel pool filled with water, whence, after being cooled, it was shipped to PA "Mayak" for reprocessing. In 1987, the final SNF load was transferred from the core to the spent fuel pool.
In 1990, after three years of the last SNF batch cooling in the spent fuel pool, the fuel was removed from the spent fuel pool, transferred to shipping casks and shipped to PA "Mayak" based on the TVR reactor decommissioning feasibility study and process.
The RA heavy water research reactor (Vinca Institute, Belgrade, Yugoslavia) is an analog of the TVR reactor. The reactor was shut down in 1984. No spent fuel has ever been removed from the RA reactor site. Spent fuel is contained in about 8100 FEs kept in:

- two hundred stainless steel channels (about 2600 FEs);
- thirty aluminum barrels (about 5000 FEs);
- dry channels in the reactor core (about 500 FEs).

The stainless steel channels and aluminum barrels are kept in four connected pools filled with water. Since water is not purified in the spent fuel pools, the current quality of the water in them differs much from the normal quality [1].
In 1998-1999, the aluminum barrels were vented and gas bubbles discharged from the barrels showed that the barrels were leaky. The results of the water sampling in individual barrels with spent FEs suggested that the water activity in terms of ^{137}Cs was up to $1.3 \cdot 10^6$ Bq/ml [2]. Further SNF storage under the existing conditions (in water pools) can soon cause large-scale radioactive contamination of the spent fuel pool water. The solution of this problem requires consideration of the following SNF handling options:

- SNF transfer to "dry" storage;
- SNF removal for reprocessing.

Owing to the similarity of the TVR and RA reactors, the TVR decommissioning experience can be used to solve the problems concerning the handling of the RA reactor SNF.

2. Equipment and technology for shipping the TVR reactor SNF for reprocessing

In 1990, the final SNF batch from the TVR reactor was removed to PA "Mayak" for reprocessing. A dedicated TK-5M shipping cask was used for the SNF shipment. To meet the applicable safety requirements, some of the shipping cask units were modified, namely: there was developed and manufactured a transport canister with a unique lid enabling remote SNF canister locking and providing for the canister serviceability under normal conditions and in emergencies. The shipping cask weight was not more than 5500 kg. Each shipping cask contained 132 fuel elements.
Technologically, the SNF handling process consisted of the following main stages:

- transportation of fuel channels with spent fuel elements from the spent fuel pool to the TVR reactor's dry storage facility;
- remote cutting of the fuel channels;
- spent FEs withdrawal from the fuel channels, FEs loading into canisters within cans and remote locking of the canisters;
- transportation of the cases with the SNF canisters from the dry storage facility to a dedicated temporary storage facility in the reactor's upper water shield;
- transfer of the cases from the temporary storage facility to the shielding shipping casks;
- loading of the containers on a motor vehicle and their shipment to a transshipment station for being loaded into a container car.

The complex for the spent FEs transfer from the fuel channels to the canisters within cases was deployed in the vault of the dry storage facility in a concrete biological shield on the reactor upper platform.
To reduce the time for the train loading with shipping casks, the SNF was loaded into the shipping cases in advance. The filled up cans were accumulated and temporarily stored in a dedicated temporary storage facility in the reactor top's iron-water shield.
The transfer complex consisted of the following major components:

- a machine for cutting off the tails of the fuel channels with the spent fuel elements;
- a device for item-by-item transfer of the spent FEs to canisters;
- a device for remote installation and locking of the canister lids;
- the upper biological shield with a transport hatch and a peephole;
- a TV camera and electric lamps.

The temporary intermediate storage facility was equipped with three pulse instrumentation channels for monitoring the SNF case lattice multiplication properties.
After their manufacture and assembly, the equipment components and rigs for the SNF handling operations were tested under normal and abnormal conditions of operations on a step-by-step operation basis and as a complex with the FEs mock-ups.
The main stages of SNF reloading from fuel channels to transport cases and schema of the TVR reactor with the temporary storage of loaded cases are shown in Figs. 1. and 2.

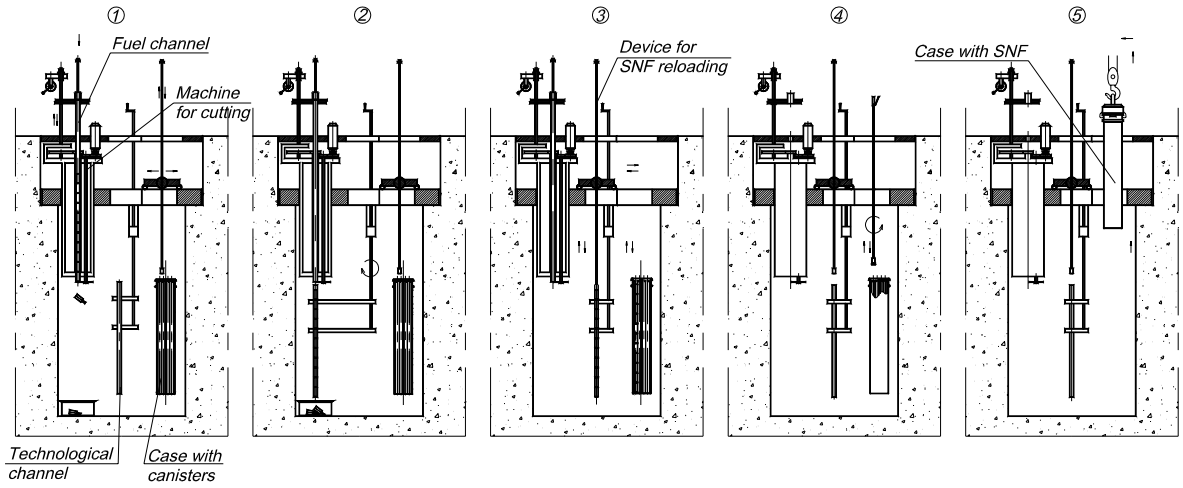

Fig.1. The main stages of SNF reloading from fuel channels to transport cases

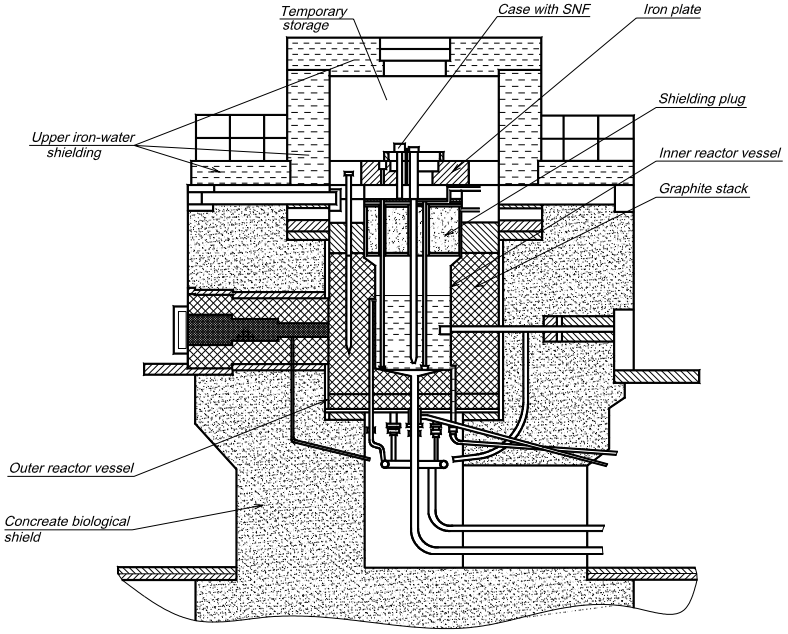

Fig.2. Schema of the TVR reactor with the temporary storage of loaded cases

3. Proposals for the handling of the RA reactor spent nuclear fuel

Currently, it has been resolved to decommission the RA reactor. The purposes of and the procedure for undertaking these activities should be determined by the Decommissioning Program under development. Similarly to decommissioning of other reactors, the major tasks to be attained in the RA decommissioning will be:

- SNF handling;
- equipment and structures dismantling;
- solid and liquid radioactive wastes handling.

Due to deteriorated states of the structural and fuel materials and abnormal storage conditions, the reactor SNF handling is becoming an independent problem requiring urgent solution.

The most cardinal way to begin with solving the problem of normalizing the situation at the RA reactor site is to implement the fuel handling option aimed at the fuel removal to PA “Mayak“ for reprocessing. In accordance with Russian regulatory documents, the major requirement to the spent fuel assemblies of research reactors is that there should not be:

- any loss in the structural integrity, major deviations from the shape and dimensions and any other damages making their loading/unloading to cans more difficult;
- damages to the gripper device;
- leaky parts leading to the fuel composition contact with the storage facility water (microdefects as “gas leakage” through the cladding is permitted).

The conditions for shipping defective spent nuclear fuel should be stipulated additionally. As a rule, the major requirement to the shipping of defective SNF is its loading into leak-tight canisters. The peculiarity of storing the spent nuclear fuel from the RA reactor is the FEs storage in batches (from 12 fuel elements kept in stainless steel channels up to 180 kept in Al barrels). Both leaky and leak-tight fuel elements are kept together in separate channels and barrels. The FEs (each FE separately or the batch as a whole) should be checked for leak-tightness during SNF preparation for transportation and defective FEs should be sealed additionally. FEs can be also additionally sealed as part of the entire FEs batch in which defective FEs have been found. The other approach to preparing the SNF for transportation is to regard all FEs as conventionally leaky without preliminary check and, accordingly, seal them additionally.

The SNF from the RA reactor can be reprocessed at PA “Mayak” as the TVR reactor SNF. The reprocessing plant is technologically capable of receiving the RA reactor SNF both in shipping canisters and in Al barrels but it should be mentioned that the reprocessing of SNF from Al barrels is much more difficult then the same process for SNF from shipping canisters.

Currently, TUK-19 and TUK-32 shipping casks owned by PA “Mayak” and RF SRC NIIAR are used in Russia to remove the SNF of research reactors. These casks meet current Russian and international requirements and can be licensed for being used in SNF transportation to PA “Mayak” from foreign countries.

Spent fuel elements can be placed into in the TUK-19 casks after being transferred to special canisters. The TUK-32 have dimensions large enough to be capable of accommodating both special canisters and Al barrels with fuel elements. Special cases should be developed and manufactured for placing canisters and Al barrels in the shipping containers. Depending on the transport container type selected for the SNF removal, the procedure for the SNF preparation for transportation should be determined.

The TUK-19 shipping cask includes a container, a case (slightly structurally different from the case used for the SNF removal from ITEP) and twelve canisters. The container is filled with air or an air mixture with an inert gas. The empty shipping cask weight is not more than 4705 kg. The internal container cavity has the diameter of 220 mm and the height of 1400 mm.

The most suitable vehicle for the shipping cask transportation is a special TK-5 freight car. Motor and water transport may be used if required. The cask is positioned vertically in the car. Each TK-5 car can contain up to eight TUK-19 shipping casks. The TK-5 car can be transported on railways with a gauge line width of 1520 mm and 1435 mm. The axle load does not exceed 18 tons.

Currently, two TK-5 cars equipped with sixteen TUK-19 containers are used.

To be loaded into the TUK-19, the spent FEs of the RA reactor should be transferred to canisters to be placed, in turn, into cases. The spent FEs arrangement in the TUK-19 cask is shown in Fig. 3. The TUK maximum capacity is 132 fuel elements. To transport all SNF from the RA reactor, not less than four special train runs will be required (2112 FEs/run).

The spent FEs can be placed into canisters and shipping cases using the technology, equipment and tools used for the removal of the SNF from the Russian TVR reactor after being modified for the RA reactor conditions. The upgraded shipping complex can be deployed in the existing dry pool or in one of the four pools filled with water (after its sluice gate is opened and it is dried) in the spent fuel pool building.

The TUK-32 shipping cask is intended for the SNF removal from SRC NIIAR research reactors. Altogether, there are three TK-32 containers carried in one TK-VG-18 car. A TK-VG-18 car can be transported on railways with the gauge line width of 1520 mm and 1435 mm. The axle load does not exceed 22 tons. The TK-32 container weight is about 37470 kg. The internal container cavity has the diameter of 775 mm and the height of about 3.5 m and is filled with air during transportation. The TK-32 use enables the SNF removal in the 20 aluminum barrels currently containing about half of all FEs. The said vessels have the height of about 800 mm and the diameter of 300 mm and can be placed in a container in several layers. The FEs Al barrels arrangement in the TUK-32 shipping cask is shown in Fig. 4.

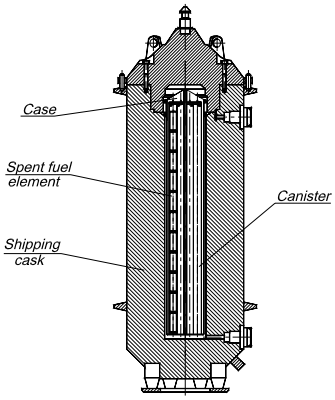

Fig.3. FEs arrangement in the TUK-19 shipping cask

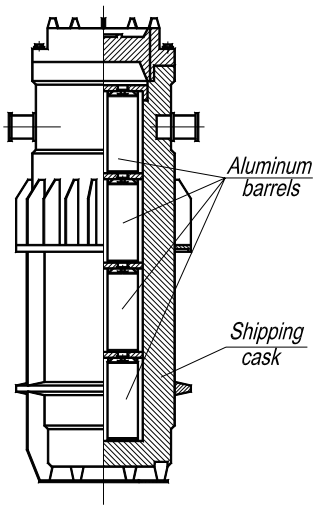

Fig.4. FEs Arrangement in the TUK-32 shipping cask

Nuclear safety assurance of the SNF containers will be given keen attention in the development of a new shipping cask with the TK-32 container. This is likely to require the use of neutron absorbers (e.g. boron steel). Preliminary studies show that three TK-32 containers are capable of containing all RA reactor SNF inventory and its removing in one special train run.

The use of the TK-32 involves major difficulties in its handling at the RA reactor site with the crane equipment's lifting capacity of not more than 5 tons. Besides, the load on the KT-VG-18 car axle exceeds 18 tons, which will require special additional measures to enable the special train run on railways in European countries.

To compare the options of the SNF transportation arrangements and transportation as such and select the best option, the following factors should be taken into account:

- radiation impact on the attending personnel, population and environment;
- costs of transportation and reprocessing;
- costs of acquiring equipment with costs estimates for its subsequent decontamination or disposal;
- time required to prepare the SNF for transportation and remove it from the Vinca Institute territory.

4. Conclusion

In 1990, the last spent fuel batch was removed from the TVR reactor. The developed technology and specialized equipment for the SNF transfer to shipping casks have proved their reliability and safety.

Due to the identity of the TVR and RA reactor designs and their fuel, the experience and equipment used in the FEs preparation for loading into shipping casks in preparing the SNF from the TVR reactor, Moscow, can be used as the basis for the RA reactor SNF removal.

The existing shipping casks can be used for the RA reactor SNF transportation for reprocessing after being modified. The reprocessing plant is technologically capable of receiving the RA reactor SNF both in shipping canisters and in Al barrels but it should be mentioned that the reprocessing of SNF from Al barrels is much more difficult then the same process for SNF from shipping canisters.

The SNF handling options should be compared by way of feasibility studies (with considering the option of the SNF transfer to long-term dry storage) prior to the best option selection.

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RTR SPENT FUEL REPROCESSING: NITRIC ACID DISSOLUTION OF ALUMINIUM ALLOYS

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ABSTRACT

The process proposed by COGEMA for reprocessing spent fuel from research reactors is based on an initial step in which the fuel plates are completely dissolved in nitric acid. The CEA therefore initially examined the dissolution of aluminium, the material used for the fuel cladding. Unirradiated samples of the various aluminium alloys used by CERCA for the fabrication of these assemblies were dissolved experimentally in hot nitric acid. The study focused on AG3NE alloy, for which the dissolution rate was measured at temperatures ranging from 70°C to the boiling point (~110°C) and for solution acidity values from 3 to 9 N. The dissolution kinetics of the other alloys were then measured at the boiling temperature and $[HNO_3]_0$ = 9 N. The alloy composition was found to have a significant effect on the results.

The effects of the reactor residence time on the materials were assessed by measuring the dissolution rates of samples maintained for 200 hours in demineralized water at 90°C. For some alloys this treatment resulted in a significant increase in the initial rate, which then gradually diminished. The rate increase can be attributed to the increased material surface area in contact with the solution due to surface corrosion.

1. Introduction

The currently planned dissolution process calls for complete dissolution of the fuel assemblies after chopping off the most massive structural components. In this case, aluminium would account for the largest fraction of the material to be dissolved. Despite the experience acquired to date by the CEA and COGEMA in reprocessing fuels of this type, before adapting the present processing unit it was advisable to obtain additional data on the nature of the gaseous reaction products as well as on the dissolution kinetics of the aluminium alloys found in past, present or future fuels.

2. Dissolution reactions – Nature of the reaction off-gases

The dissolution of aluminium is described by a set of reactions differing by the nature of the off-gas compounds: nitrogen oxides (NO, NO_2 and N_2O), nitrogen and hydrogen. The predominance of one or more of these compounds appears to depend mainly on the acidity of the medium, although no explicit relation has been proposed. From the process standpoint, it is important to define the nature of the off-gas stream arising from dissolution of this material not only under standard operating conditions but also under conditions of low acidity that can result in greater hydrogen production according to published reports.

In order to identify the reactions involved, dissolutions were carried out on AG3 aluminium alloy plates in the test facility shown in **Figure 1**. The solution was maintained at a temperature of 90°C.

Nitrogen was supplied to the bottom of the reactor to homogenize the solution and to transfer the gaseous products to the analysis system without oxidizing NO to NO_2. The NO, NO_2, N_2O and H_2 concentrations were continuously measured by mass spectrometry after condensing the steam.

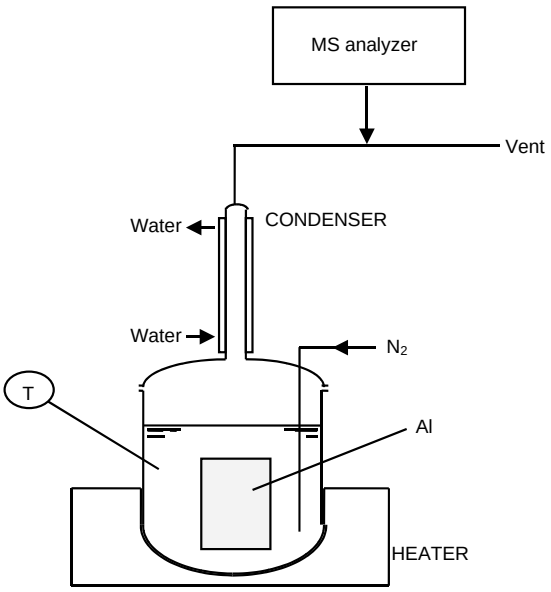


Figure 1. Experimental facility

At the same time the H^+ et Al species were monitored in solution to measure the acid consumption during dissolution.

Table I summarizes the results obtained for four acidities in the dissolution medium. Although the N_2O concentration remained near zero in all the experimental media, the NO_2 concentration was observed to rise with the acidity, at the expense of NO.

Nitric oxide was the most abundant compound in every case, with a molar concentration ranging from 95% to 75% as the acidity increased from 0.5 to 10 N.

Table I. Distribution of nitrogen oxides depending on the acidity of the dissolution medium

Initial acidity ($mol \cdot L^{-1}$)	Final acidity ($mol \cdot L^{-1}$)	X_{NO}*	X_{NO2}*	X_{N2O}*
0.5	–	0.94–0.96	0.03–0.05	< 0.01
3	2.4	0.94–0.95	0.05–0.06	< 0.01
6	5.4	0.82–0.88	0.11–0.14	0.01–0.04
10	9.6	0.76–0.78	0.21–0.24	< 0.01

* Molar fraction of each compound ($NO - NO_2 - N_2O$)

The solution analysis results indicated that 4.2 moles of acid were consumed per mole of dissolved aluminium. A comprehensive balance of the nitrogen compounds (HNO_3, $Al(NO_3)_3$, NO and NO_2) showed that the dissolution is controlled primarily by the following two reactions:

$$Al + 6HNO_3 \rightarrow Al(NO_3)_3 + 3NO_2 + 3H_2O$$

$$Al + 4HNO_3 \rightarrow Al(NO_3)_3 + NO_2 + 2H_2O$$

Hydrogen was monitored during successive dissolutions at decreasing acidity (from 6 N initially to 1 N) showing a slight increase in the quantity hydrogen produced (with respect to the quantity of aluminium dissolved) as the acidity of the medium diminished. The increase was very slight in every case, and the mean measured quantities ranged from 4×10^{-3} to 6×10^{-3} mole of H_2 per mole of dissolved Al.

3. Dissolution kinetics

This study was carried out essentially with samples of AG3NE alloy, the principal material used for the fuel cladding. The dissolution kinetics were also measured for various other alloys under specified temperature and acidity conditions. All dissolutions were carried out on thin (1–1.5 mm) plate sections in solution with an initial acidity of 9 N (final solution: $[H^+] = 3$ N and [Al] = 35 to 40 g/L). The kinetics were determined by monitoring the quantity of aluminium measured in solution (ICP-AES analysis) over time, and expressed in terms of the material/solution contact surface area.

3.1 Dissolution of AG3NE alloy

Figure 2 shows the dissolved aluminium content versus time at three test temperatures. The linear plots indicate constant kinetics both over time and as a function of the acidity (since the acidity simultaneously diminished at a linear rate). Tests performed at different acidities confirmed this result. **Figure 3** demonstrates the significant effect of the temperature on the kinetics.

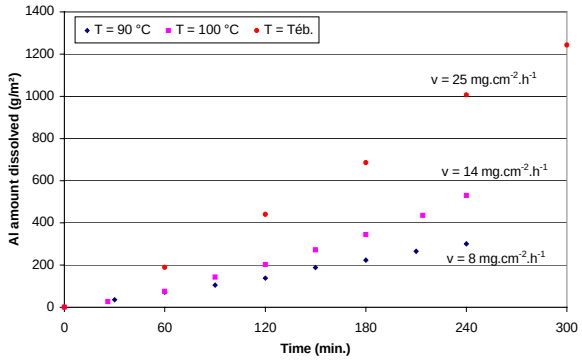


Figure 2. Aluminium dissolution versus time

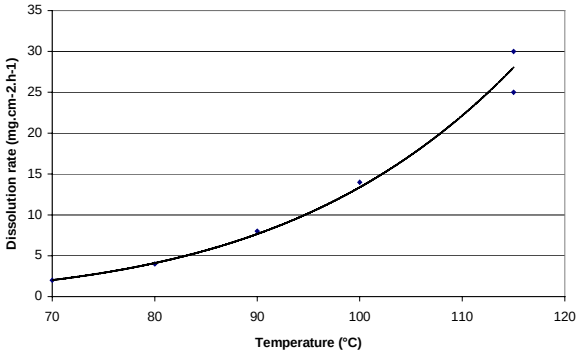


Figure 3. Effect of temperature on the dissolution kinetics

The preceding experiments on unirradiated samples were followed by the dissolution of samples that had been treated to reproduce the effects of residence in a reactor. Samples were immersed in demineralized water at 90°C. Microscopic observation and X-ray diffraction analysis revealed the presence of boehmite AlO(OH) on the surface; this compound is observed on the surface of fuel plates after irradiation. As shown in **Figure 4** the samples exhibited a very different dissolution profile than the untreated samples.

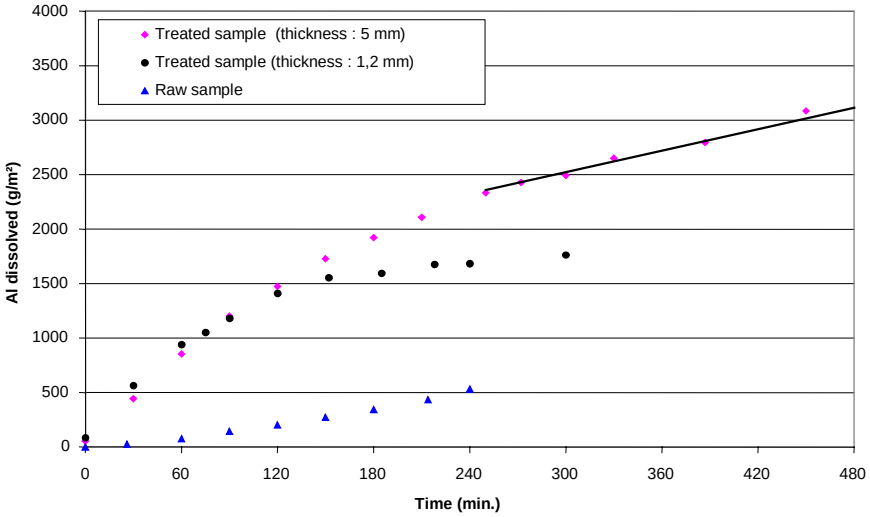
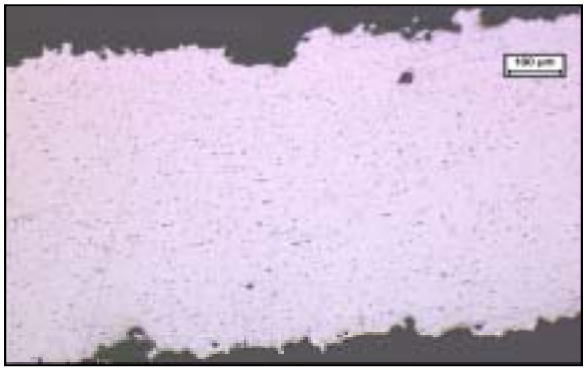


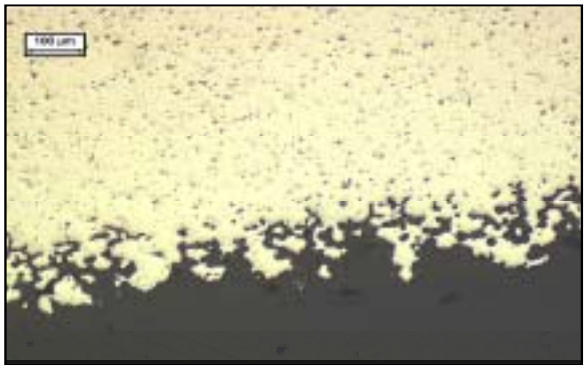
Figure 4. Effect of simulated reactor residence time on the dissolution kinetics at 100°C

The initial rate was much higher (90 ± 10 mg·cm^{-2}·h^{-1}) and diminished over time. With thicker samples the initial kinetics were the same as for the thin samples, then diminished to a constant value of the same order of magnitude as for the untreated samples under identical conditions. This observation suggests that the phenomenon is limited to the sample surface.

The higher apparent dissolution rate appears to be attributable to an increase in the contact surface area between the solid and solution rather than to an increase in the reaction rate. This was confirmed by SEM examinations of polished cross sections from partially dissolved samples.



(a) Untreated plate
after 4 hours of dissolution



(b) Corrosion-treated plate
after 30 minutes of dissolution

Figure 5. SEM observations of polished cross sections from partially dissolved samples: effect of simulated reactor residence time

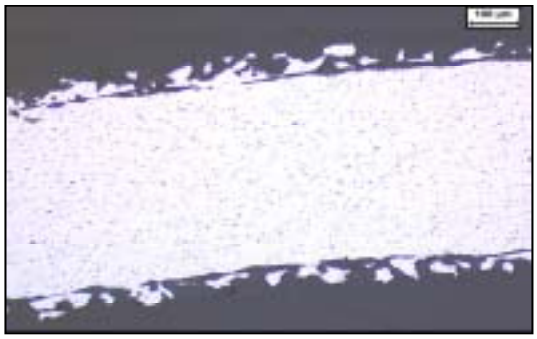


Figure 6. Corrosion-treated plate
after 75 minutes of dissolution

The two SEM images of samples at the same degree of dissolution progress (**Figure 5**) illustrate the difference in the metal surface area accessible to the solution.

Figure 6 shows how the surface area diminishes as the reaction progresses, confirming the drop in the apparent dissolution rate observed in **Figure 4**.

These observations are consistent with aluminium corrosion studies indicating the sensitivity of this metal to pitting corrosion in aqueous media at near-neutral pH values.

3.2 Dissolution kinetics of different aluminium alloys

Table II. Dissolution rates of different Al alloys ($[H^+]_0$ = 9 N; T = bp)

Material	Dissolution rate ($mg \cdot cm^{-2} \cdot h^{-1}$)	Main constituent elements (wt%)				
		Mg	Fe	Ni	Si	Min
AG3NE	25 to 30	3.5 to 4.5	0.5	–	0.3	0.2 to 0.7
AG5NE	165	4.9 to 5.6	0.5	–	0.4	0.2 to 0.7
AlFeNi	20	0.8 to 1.2	0.8 to 1.2	0.8 to 1.2	-	0.2 to 0.6
6061	25 to 30	0.8 to 1.2	0.7	–	0.4 to 0.8	0.15
A5	15	0.05	0.4	–	0.25	0.05

The nature and elemental concentrations in aluminium alloys can affect the dissolution kinetics; measurements were therefore carried out with 4 other commonly used materials. The experimental conditions were the following: $[H^+]_0$ = 9 N, boiling temperature. **Table II** indicates the dissolution kinetics measured on the untreated materials and the weight percentages of the main elements in these alloys. Comparable results were observed for all the materials except AG5NE alloy, which dissolved at a much higher rate.

However, this difference in dissolution behaviour observed for AG5NE alloy is not confirmed by the dissolution plots of these same samples previously immersed for 200 hours in demineralized water at 90°C (**Figure 7**). Although the samples were not analyzed prior to dissolution, it can be assumed that different aluminium alloys exhibit different behaviour in this aqueous environment designed to represent reactor operating conditions. Alloys with significant magnesium content (AG3NE and AG5NE) appear to be subject to the greatest variation in their initial dissolution rates.

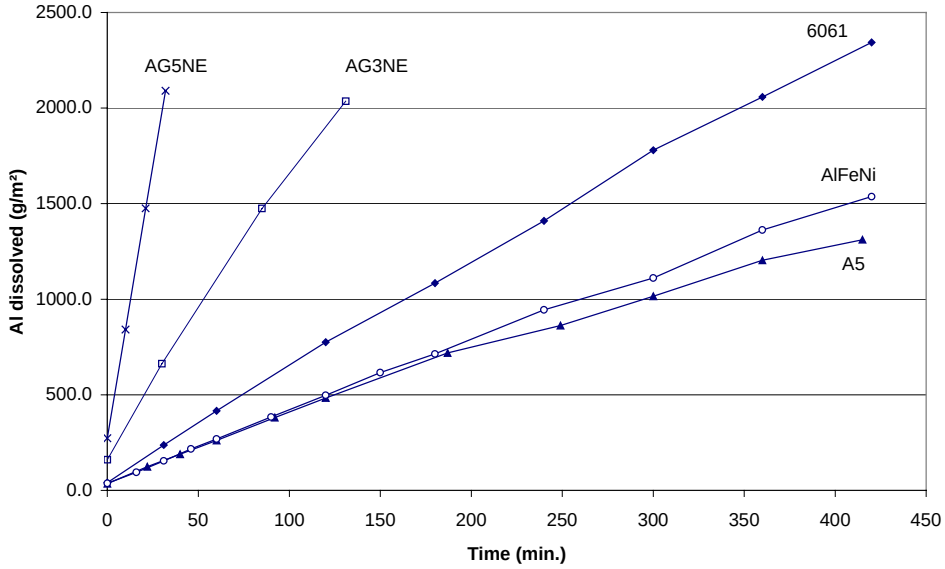


Figure 7. Dissolution of various Al alloys after immersion in demineralized water at 90°C (Dissolution: $[H^+]_0$ = 9 N; T = bp)

4. Conclusion

This study of aluminium dissolution in nitric acid was limited to the conditions planned for reprocessing research reactor fuel : batch dissolution producing a solution with an Al concentration of about 40 g/L and 3 N acidity.

The gaseous dissolution products were initially identified and found to consist of nitrogen monoxide and dioxide compounds in proportions that varied with the acidity of the medium; NO was the majority compound in every case. Hydrogen was also observed, the quantity produced remained very small : 5×10^{-3} mole of H_2 per mole of Al dissolved.

The dissolution kinetics of AG3NE alloy were unaffected by the acidity of the medium but increased significantly with the temperature.

In addition, major differences could be observed depending on the nature alloy dissolved.

Finally, the treatment applied to various samples to reproduce the corrosion sustained by the material in a reactor increased the initial dissolution rate to an extent that depended on the nature of the alloy. Tests with AG3NE alloy showed that the apparently faster kinetics were due in fact to the increased surface area on samples corroded by water.

FiR 1 REACTOR AND THE SPENT FUEL MANAGEMENT

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ABSTRACT

The FiR 1 –reactor, a 250 kW Triga reactor, has been in operation since 1962. The main purpose to run the reactor has been lately the Boron Neutron Capture Therapy (BNCT). According to the current operating license of our reactor we have to achieve a binding agreement between our Research Centre and either the domestic Nuclear Power Companies about the possibility to use the Olkiluoto final disposal facility for our spent fuel or USDOE about the return of our spent fuel back to USA. If we want to continue the reactor operation beyond the year 2006, the domestic final disposal is the only possibility.

At the moment the domestic final disposal is the primary alternative, but it seems still to be reasonable to be prepared to both possibilities: the domestic final disposal and the return to the USA offered by USDOE. Because the cost estimates of the both possibilities are on the same order of magnitude, the future of the reactor itself will decide, which of the spent fuel policies will be obeyed. In less than two years' time it will be reviewed, if the results of the BNC treatments are satisfactory for the continuation of the treatments. If the BNCT and other irradiations develop satisfactorily and seem to have positive future, the reactor can be kept in operation beyond the year 2006 and the domestic final disposal will be implemented. If, however, the development of the BNCT will not develop positively and there will be lack of money, there is no reason to continue the operation of the reactor and the choice of USDOE alternative is natural.

1. Introduction

The FiR 1 reactor, a 250 kW Triga reactor, has been in operation since 1962. The main purpose to run the reactor has been lately the Boron Neutron Capture Therapy (BNCT). The epithermal neutrons (1 eV – 10 keV) needed for the irradiation of brain tumor patients are produced from the fast fission neutrons by a moderator block consisting of $Al+AlF_3$ (FLUENTAL$^{\text{TM}}$) developed and produced by VTT. The material gives excellent beam values both in intensity and quality and enables the use of a small research reactor as a neutron source for BNCT purposes. Twenty-one patients have been treated since May 1999, when the license for patient treatment was granted to the responsible BNCT treatment organization. The treatment organization has a close connection to the Helsinki University Central Hospital. The BNCT work dominates the current utilization of the reactor: three or four days per week for BNCT purposes and the rest for other purposes such as the neutron activation analysis and isotope production. Figure 1 describes the general layout of the BNCT facility at the FiR 1 reactor. The facility gives a high epithermal neutron field, 1.1×10^9 n/cm^{2}s with a very low fast neutron and gamma component.

During the next two or three years the back end solutions of the spent fuel management will have a very important role in our activities and in the possibility to continue the operation of the reactor. According to our current operating license we have now about two years' time to achieve a binding agreement between VTT and the domestic Nuclear Power Plant Companies about the possibility to use the final disposal facility of the Nuclear Power Plants for our spent fuel. In this case we can continue the operation of the reactor as long as there is reasonable work to do with the reactor and the funding is in order. Naturally we can also make an agreement with the USDOE within the well-known time limits.

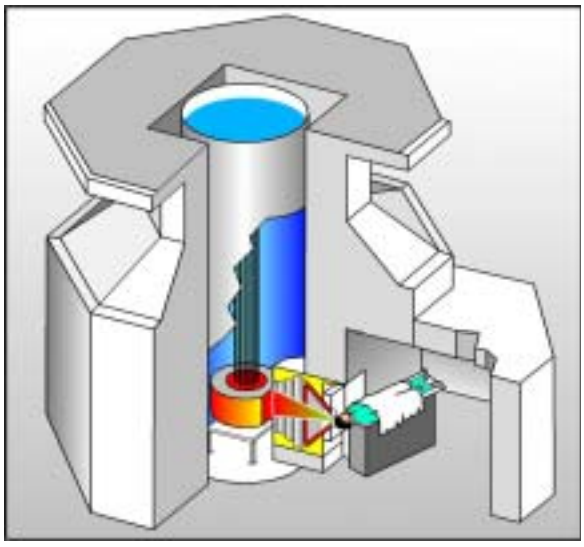

Figure 1. BNCT Facility at the FiR 1 –reactor

2. Final disposal solution in Finland

The Finnish Nuclear Power Companies founded in 1995 a separate company Posiva to develop the technology and carry out safety analysis and site investigations for implementing the spent fuel final disposal. In 1999 Posiva submitted an application for a decision in principle for a final repository to be built at Olkiluoto, on the western coast of Finland. Olkiluoto is also one of the two nuclear power plant sites in Finland. At the end of the year 2000 the Finnish government approved the application and sent it to the parliament for ratification. The ratification took place in May 2001. Separate licenses still will be needed for the construction of the facility, scheduled to start in 2010, and also for the operation, 10 years later. The government alone will grant these licenses.

For the final repository the spent fuel will be encapsulated in airtight copper canisters and situated in the bedrock at a depth of 500 m. The safety of this deep underground repository is based on multiple natural and engineered barriers. Each canister contains 12 normal fuel assemblies from nuclear power plants. The present concept for Triga fuel elements is that the elements will be loaded in containers, which have the same outer dimensions as the nuclear power plant fuel assemblies. This ensures that the Triga fuel will be easily handled in the final disposal facility and also loaded in the heavy copper canisters.

3. Current situation of the FiR 1 reactor

In Finland also the research reactor must have a nuclear waste management plan, which contains among others a part for spent fuel management. The plan describes the methods, the schedule and the cost estimate of the whole spent fuel management procedure starting from the removal of the fuel from the reactor core and ending to the final disposal. The cost estimate of the nuclear waste management plan has to be updated annually and every fifth year the plan will be updated completely. The plan has been based on the assumption that the final disposal site will be somewhere in Finland. Now we know that the final disposal facility for the spent fuel of the nuclear power plants will be situated in Olkiluoto. The final disposal facility is supposed to be in operation in 2020.

We have had already for thirteen years an agreement in principle about the possibility to use the final disposal facility of one of the Finnish Nuclear Power Companies. Later this agreement was transferred to the joint nuclear waste management company Posiva. According to the current operation license of our reactor we have to achieve a binding agreement between our Research Centre and either Posiva or USDOE about the back end solution of the spent fuel. This means that the said agreement in principle is not sufficient any more. The binding agreement with Posiva is the only alternative, if we want to continue the reactor operation beyond the year 2006. Obviously the idea is that the binding agreement has to be established during the time when there are still two possible agreement partners left. Before

we can start the real negotiations about the final disposal of our spent fuel with Posiva, we have to prepare a safety study about the behaviour of the Triga fuel in the final disposal surroundings.

The operation license of our reactor will expire in 2011. It is very probable that there will be certain waiting time from the shut down of the reactor to the opening of the final disposal facility. Therefore there have to be a sufficient interim storage for the spent fuel before the transportation to the final disposal facility. After enlargement work in 1997 we have sufficiently storage capacity for the fuel in the reactor building. So far we have used it as dry storage. In addition to the domestic final disposal solution there is still the USDOE alternative available until 2006.

4. Safety of the Triga fuel in the final disposal repository

In order to start the negotiations with the Nuclear Power Companies or their representative Posiva we have to prepare a safety study about the long-term behaviour of the Triga fuel in the final disposal repository. As was already mentioned the Triga fuel elements will be loaded in containers, which have the same outer dimensions as the nuclear power plant fuel assemblies. We need from 3 to 5 such containers for all the Triga fuel elements. The containers can easily be loaded into the heavy copper canisters, which have 12 positions to be loaded. For the criticality reason the Triga containers will be situated in the outer zone of the canister and the inner zone will be left empty. In practice the empty positions will be loaded with dummy assemblies made of cast iron. It can be shown that the system is critical safe. This is important, because if the criticality safety would demand the fuel to be divided to two or more canisters, the expenses would also be about twice or more compared to the one canister alternative.

5. Decisions to be done

If we want to utilize the USDOE policy and return our fuel back to USA, it means in practice that the whole inventory of the irradiated fuel should be sent to USA at the same time. Thus the return of the fuel back to USA means in other words the shutting down of the reactor permanently. In our case the important question should be: Are we willing to shut down the reactor within the time limits declared by the USDOE?

The BNCT work is today the main purpose to run the reactor. The amount of BNCT irradiations is still rather low: twenty-one irradiations during the first forty months. However, at the end of last year (2002) the funding of the BNCT project was reorganised and the new arrangement ensures sufficient funding both for the reactor operation the BNCT work for the next two or three years. This means that we can now start a very intensive period of BNC treatments. During that time we together with the treatment organization have the opportunity to show that the BNCT irradiations will give good results and will be needed also in the future. If the BNC treatment results are promising and the positive trend seems to continue, the funding of the BNCT project will obviously also continue. In this case there is no reason to use the USDOE alternative. Instead it is reasonable to continue the reactor operation beyond the year 2006, which means inevitable also the choosing of the domestic alternative for the treatment of the spent fuel. If, however, the results of the BNC treatments do not fulfil the expectations, the funding of the reactor will be stopped after the said period. This leads inevitably to the permanent shut down of the reactor. In that case the USDOE alternative seems to be the right one.

6. Conclusions

We have now ahead of us a restricted period of time, during which we have the possibility to show that the BNCT method works successfully. After less than two years, in 2004 we are in the situation, when we have to decide in every case, if the reactor will continue the operation or if it will be shut down. As the criterion there will be the possible success of the BNCT. Ultimately the funding organization of the BNCT will decide. It will simply either continue funding or stop funding. After the decision of the reactor operation the choosing between the USDOE and the domestic back end solution will be rather easy, because the expenses of both of the spent fuel management alternatives seem to be of the same order of magnitude.

EXPERIENCES FROM HEU FRESH FUEL TRANSPORTATION FROM YUGOSLAVIA TO RUSSIA

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ABSTRACT

This paper shows the experiences gained in the recent transport (August 2002) of the fresh high enriched uranium (HEU) fuel elements from Yugoslavia back to Russian Federation for uranium down blending. In this way, Yugoslavia gives its contribution to the Reduced Enrichment for Research and Test Reactors Program and to the world's joint efforts to prevent possible terrorist action against nuclear material potentially usable for production of nuclear weapon.

1. Introduction

During spring and summer of 2002, Yugoslav government officials had made the agreement with officials of US and Russian governments to return of all unused HEU fuel elements back to the country of origin - Russian Federation. Decisions on final shutdown and decommissioning of the RA research reactor and the HEU fresh fuel shipment, were made by the Government of Republic of Serbia and the Government of the Federal Republics of Yugoslavia in July and August 2002. Preparations for the shipment of more than 5 thousands HEU fuel elements were done in the "Vinča" Institute of Nuclear Sciences, Belgrade, Yugoslavia and in the Branch Federal State Institute "Safe Transportation of Nuclear Materials" (STNM Institute), Dimitrovgrad, Russian Federation. Safeguards department of the International Atomic Energy Agency, Vienna (IAEA), Austria and non-proliferation specialists from the Oak Ridge National Laboratory, operated for Department of Energy (DOE), USA, took active monitoring part during the preparation and transport of the HEU fuel elements. The shipment was realised during mid-August 2002 under highest physical protection available in the Vinča Institute and during the transport of the fuel to the Belgrade airport.

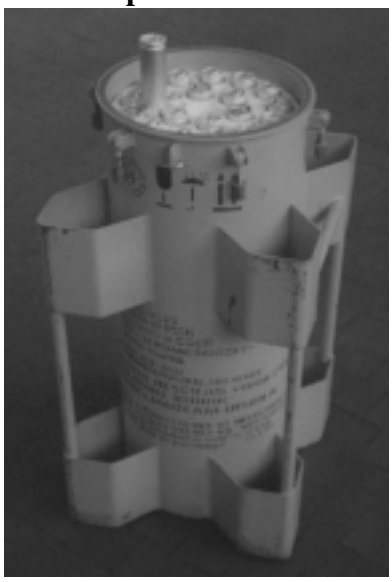
2. Description of TVR-S fuel assemblies

All the fresh HEU fuel assemblies (FA, "slugs") at the Vinča Institute were bought in ex-USSR during 1976-1985. These elements have been used for the operation of 6.5 MW heavy water research reactor RA and for experiments at the RB heavy water critical assembly. The HEU fuel elements, known as the TVR-S type of FA [1], were produced in the Novosibirsk Chemical Concentrates Plant (NCCP). The slug is 11.30 cm long cylinder with 3.72 cm OD that contains 80 % enriched U in form of UO_2 dispersed in Al matrix [2]. The Al matrix has length 100 mm and ID/OD 31/35 mm. Mass of ^{235}U nuclide is (7.7 ± 0.3) g. The fuel layer is covered by 1 mm thick Al cladding. Inner tube, made of Al ("expeller") within the slug serves to adjust the coolant flow rate. Top and bottom of the slug are covered by the 3 mm thick Al "stars" with sprockets. The Al, used in the construction of the TVR-S slugs, is known as the SAV-1 alloy (0.985 weight fraction of Al with very low contents of neutron high-absorbing impurities, e.g., B and Cd). Average mass of the TVR-S slug is 162 g.

About 1400 HEU slugs were spent during the operation of the RA reactor, by the end of 1984. The total amount of the fresh HEU fuel elements at the RB and RA reactors at Vinča Institute in 2002 was 5046. The fresh and spent nuclear fuel elements at the Vinča Institute are under regular safeguards control of the IAEA. The HEU fresh fuel elements were stored in original, NCCP made containers used for transport and storage (Figure 1). Beside police guards engaged for physical protection of the

reactors' building, the storage rooms for the fresh HEU elements at both reactors were also under surveillance by the automatic electronic alarm system applied since 1996.

3. Description of used IP-2 NCCP transport packages



Inspection of the existing old NCCP packages has been carried out in spring 2002 by US experts from the ORNL and LANL and Russian experts from the MINATOM, ATOMSPECTRANS and the STNM Institute. They found out that these containers do not fulfil the modern standards and requirements [3] for the shipment of fission material. So, the STNM Institute has proposed the use of Russian industrial package IP-2 type TK-S15 and TK-S16, produced by the NCCP. These containers are designed and regularly used for transport of fresh FA (but not TVR-S type) that are used for operation of various types of Russian research reactors. Thus, new packaging procedures were proposed by the STNM Institute, according to the available space within TK-S15 and TK-S16 containers, and according to the safety requirements in respect to the Criticality Safety Index (CSI) and Transportation Index (TI).

Fig. 1 Old NCCP containers

These TK-S15 and TK-S16 containers with new packaging procedure for transport of the TVR-S type HEU fuel elements were certified and licensed on July 1, 2002 in Russia, with validity by the end of the year. The criticality calculations, done according requirements given in [3], were carried out independently in the Fizichesko Energheticheskiy Institut – "FEI," Obninsk, Russia and in the Centre for Nuclear Technologies and Research "NTI" of the Vinļa Institute of Nuclear Sciences, Yugoslavia. Results of the calculations [4] showed that unlimited number of the proposed TK-S15 and TK-S16 packages, filled with the anticipated number of 80% HEU fuel slugs of TVR-S type, can be used in the transport, i.e., that CSI = TI = 0, even in case of the worst assumed accident

The TK-S15 packaging consists of a steel case (container), two covers, heat insulation and wrapper. The inner equipment is a welded construction of 7 Al tubes. Capacity of the TK-S15 is 7 FA. Dimensions of the TK-S15 package are: length 1650 mm, width 400 mm, and height 420 mm. Mass of the TK-S15 is 240 kg. The total of 20 TK-S15 packages were delivered to the Vinļa Institute for the shipment (17 were used, while 3 were the spare ones). The TK-S16 packaging consists of the container steel case - a welded construction as a barrel with double walls. The gap between the walls is filled with heat-insulating material. Inner equipment is a construction of 7 Al tubes welded to spacer grids. Capacity of TK-S16 is also 7 FA. Mass of the TK-S16 is 160 kg. Dimensions of the TK-S15 package are: diameter 655 mm, maximum width 740 mm, and height 1200 mm. The total number of TK-S16 packages, delivered to the Vinļa Institute to be used for fresh HEU fuel shipment was 10.

New packaging procedures were as follows. In each tube of TK-S15 packaging, 13 couples of two TVR-S slugs (the total of 26 slugs, tied together by the Scotch tape) can be stored. In such a way, in one TK-S15 package, the total of 7*26 = 182 TVR-S slugs may be stored. The total mass of ^{235}U per TK-S15 package is 1401.4 g. In each tube of the TK-S16 packaging 8 bundles of four TVR-S slugs (the total of 32 slugs, tied together by the Scotch tape), were placed one above another. In such a way, in one TK-S16 package, the 7*32 = 224 TVR-S slugs may be stored. The total mass of ^{235}U per TK-S16 container is 1724.8 g.

4. Preparatory activities

All preparatory activities were carried out in the Vinļa Institute during June/July 2002, in close cooperation with the safeguards inspectors of the IAEA and experts from the STNM Institute. These activities included: (1) Establishing the management structure and the executive Transport Program Team; (2) Elaboration of the repackaging procedures, preparation of the repackaging area, organisation of necessary logistics support and personnel training; (3) Providing increased physical

protection in the Institute and appropriate police escort during the transport of the fresh fuel elements from the Institute to the airport in Belgrade; (4) Providing all documents, required permissions and certificates issued in Yugoslavia; (5) Providing appropriate transport vehicle(s) and close cooperation with the Customs at the airport in Belgrade; (6) Introduction with the task and training of various supporting teams (e.g., health physics department, medical protection department, fire department) for the regular and possible incidental situations; (7) Preparation of necessary equipment for packaging, radiation protection measurements, marking and sealing of the packages. The whole task was kept as a secret in order to reduce the possibility of a terrorist attack or of any conflicts with the green-peace members during the activities and transport.

The repackaging area was prepared within the RA reactor room. Three additional areas ("arrays") were marked for the locations of 51 existing storage containers, and for the 20 TK-S15 and 10 TK-S16 new transport packages. Two working (packaging) lines ("A" and "B") were set, including all necessary tools and supporting material needed. A spot for fuel element sample control measurements by the IAEA safeguards inspectors was arranged, as well as a location for gamma-ray dose rate measurement, sealing and marking of the loaded containers. Records forms for all procedures were prepared. The radiation control included contamination monitoring of the area, used equipment and new transport packages and gamma-ray dose rate measurements. Staff personnel engaged in the work had protective clothes, gloves, overshoes and TLD, although the gamma-ray dose rate from the fresh fuel elements was very low. A special metal detector gate was set at the only allowed exit/entrance of the RA reactor room in order to prevent any deliberate removal of the fuel slugs from the room. Entrance to the reactor room was allowed only to personnel wearing special badges approved and allowed by the managers of the Transport Program Team. Unless few exceptions, mobile phones and cameras were not allowed inside the RA reactor room during repackaging activities.

As the first step, all HEU fuel slugs used at the RB reactor were unloaded from the core and returned to the old storage containers. These containers were verified for the fuel type and number by the RB staff and the IAEA safeguards inspectors and sealed. Simultaneously, 41 old storage container with the fresh HEU fuel elements at the RA reactor storage site, was unsealed and opened. The total of about 4000 fuel slugs were released from their original protection packaging (paper and plastic foils) and returned back in the containers. Each container was verified for contents, closed and sealed by the IAEA safeguards inspectors, again. All existing storage containers loaded with the HEU fuel elements were transferred from their regular storage places to the RA reactor room, one day before the aircraft with new transport packages arrived from STNM Institute to the Belgrade airport.

5. Packaging and Shipment

The aircraft was unloaded at the airport immediately after lending and all new packages were loaded into the transport vehicle in few hours, including radiation and contamination control and customs procedures. The transport vehicle was escorted by police cars to the Vinļa Institute. The TK-S15 and TK-S16 packages were unloaded at the parking place in front of the RA reactor building, one by one, using lifting carriage, and transferred to the transport entrance of the RA reactor room. There, the packages were reloaded to the transport chart and carried near to marked positions. The existing crane in the RA reactor room was used to unload the chart and locate the package at the desired position. This activity took about 3 hours. The next day, all ten TK-S16 packages were fully loaded, closed, sealed, measured for the TI, labelled and moved back to their position in the reactor room. The HEU fuel slugs were prepared according to the proposed procedure – 4 in a bundle, 8 bundles connected in a row, using strong nylon string that was pulled through central axial holes of any 2 slugs in each bundle by the aid of the special needle prepared. The nylon string was used for moving the whole row of 8 bundles from repackaging table to the aluminium tubes in the TK-S16 package.

In the next two days, all 17 TK-S15 packages were fully loaded, closed, sealed, measured for the transport index, labelled and moved back to their position in the reactor room. The HEU fuel slugs were prepared according proposed procedure – 2 in a bundle, and 13 bundles connected in a row, using strong nylon string that was passed through the central axial holes of both slugs of each bundle by using prepared needle. The nylon string and plastic supporter designed as a long semi-tube was used for moving whole row of 13 bundles from repackaging table to the aluminium tubes in the TK-S16 package. Figures 2 and 3 show the top view of TK-S15 and TK-S16 opened package completely filled by TVR-S fuel elements. The IAEA safeguards inspectors monitored repackaging activities in the RA reactor room and carried out measurements of random selected HEU fuel slugs. About 1 % of all HEU fuel slugs were verified without any remark (Figure 4). The well-trained RA and RB staff completed all packaging activities about 50 % faster than expected during planning.

Fig. 2 TK-S15 loading

Fig. 3 TK-S16 loading

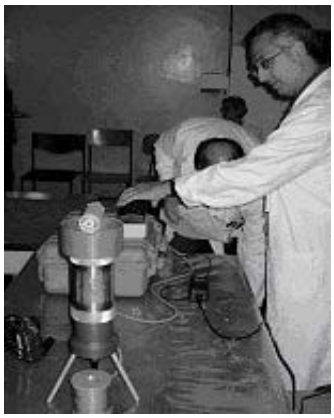


Fig. 4 Safeguard checking

Yugoslav customs officers also put their seals on the containers. The Minister of the Ministry of Science of the Republic Serbia has visited the repackaging activities in the RA reactor room and the departure area at the Belgrade airport during loading the packages with nuclear fuel to the cargo aircraft. Envoy of the Federal Ministry of Economy, responsible for safeguard of nuclear material, accounting and nuclear safety, has monitored activities carried out in the Vinļa Institute, too. The TK-S15 and TK-S16 packages filled with HEU fuel slugs, including the three spare empty ones, were moved from the reactor room to the room transport exit by using the crane and the chart. The lifting carriage accepted the package there and loaded it into the transport truck. This activity took about 2.5 hours, including measuring the TI of the packages loaded on the vehicle.

The Transport Program Team included about 30 persons from operation staff of RA and RB reactors. In the whole activities at the Vinļa site were engaged about 50 persons, including two IAEA safeguards experts, two STNM Institute transportation experts and two ORNL monitoring experts. About 1200 armed policemen, including members of the Special Antiterrorist Unit ("SAJ"), were engaged during loading of the packages on the vehicle at the Vinļa Institute, transport of the fuel elements from the Vinļa Institute to the Belgrade airport and during loading of the aircraft at the airport. The transport itself took place after midnight. Police forces closed for traffic the lines in both directions along the whole route and blocked all intersections on the road. The transport caravan, beside two transport vehicles (one loaded by the nuclear fuel and the second - "dummy one"), included various police escort units and vehicles with experts in transportation, radiation protection, medical protection and fire protection. Transport caravan, escorted also by the police helicopter, moved smoothly along almost the 40 km long route without any incidents in less than 50 minutes. The whole transport, due to closed roads, helicopter escort and heavy police forces engaged, has attracted attention and large curiosity of the Belgrade citizens and public media. The cargo aircraft was loaded in about 3 hours, including fixing packages in their locations in the aircraft, measuring the TI in the aircraft and Customs procedures. The aircraft departed the Belgrade airport the same morning, on August 22 at 08:05, escorted by two Yugoslav military air-fighters to the Yugoslav border. The cargo aircraft landed to the Uljanovsk airport near Dimitrovgrad after about 4 hours of flight time. The successful landing of the aircraft with HEU fuel cargo was immediately reported by Russian experts escorted the shipment. The same day at 12:15 in Belgrade, the Ministry of Science, Technologies and

Development of the Republic of Serbia and the Director General of the Vinļa Institute held the Press Conference about these activities and the success of the whole operation.

6. Conclusion

To carry out the whole fresh HEU fuel shipment operation at the Vinļa Institute, about 50 persons were engaged, including the safeguards inspectors from the IAEA, STNM Institute experts and monitoring experts from ORNL. The executive team consists of 30 experts from the operation staff members of the RA and RB research reactors. Security measures during the whole shipment operation engaged about 1200 policemen, including special task forces, helicopter and two military air-fighters. This operation was the first, planned and realized, shipment of the fresh HEU fuel of the Russian origin from one research reactor, back to the Russia for uranium down blending. This undertaking was very useful experience for various government and non-government institutions and management personnel in Yugoslavia and other two countries being involved. This paper shows the experiences gathered in the organization, planning and carrying such task. For Yugoslav side, the successful conduct of the operation is considered as a trial step for performing the much more complex task in future – the shipment of the spent nuclear fuel from the RA research reactor back to the Russia.

7. Acknowledgement

Authors of the paper are acknowledged to all personnel from Yugoslavia, Russian Federation, USA and IAEA that took part in the operation of shipment of the Vinļa Institute's fresh HEU fuel elements back to Russia.

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DETERMINATION OF RADIOLOGICAL CHARACTERISTICS OF REACTOR RA SPENT FUEL ELEMENTS

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ABSTRACT

This paper gives an overview of the work being performed to investigate the spent fuel characteristics of reactor RA spent fuel storage. An analysis methodology, based on the SAS2H, MCNP-4C/ORIGEN2.1 and KENO-V.a/ORIGEN2.1 utility codes is presented along with information representing the validation of the methods and geometrical models. Example application is given for a radiation characteristics evaluation (radiation sources) of reactor RA spent fuel storage.

1. Introduction

The planning and activities related to the safe transport of spent fuel elements from the reactor RA [1] at Vinca Institute to the reprocessing plant or to the dry storage, performed during last few years, have resulted in development of valid methods for radiological characterisation of spent fuel and shielding analysis of storage containers and transport casks with reactor RA spent fuel elements. The complexity of the reactor RA fuel burnup analysis arises from the end region effect between fuel elements placed inside the fuel channel.

This paper describes three methodologies, being prepared for the analysis of reactor RA spent fuel characteristics (radiation sources, decay heat, and spent fuel isotopics). The first methodology is based on the application of the SAS2H control module [2] from the SCALE-4.4a code system [3] and an approximate geometrical model. The second computational tool employed consists of the MCNP-4C [4] and ORIGEN2.1 [5] codes interfaced by the MOCUP driver [6]. The third methodology is based on the applications of the KENO-V.a [7] and ORIGEN2.1 codes.

Results representing the validation of the methods and geometrical models are also included. Finally, radiation characteristics evaluations (radiation sources) of reactor RA spent fuel storage are presented.

2. Models for reactor RA unit cell

The 6.5/10 MW heavy water moderated and cooled research reactor RA started operation in 1959. It was first operated with 2% enriched uranium metal fuel. The fuel element is aluminum clad hollow cylinder with diameter of 3.7 cm, and height of 11.25 cm, containing metal uranium in tubular form. New fuel with 80% enriched uranium dioxide dispersed in aluminum, having the same geometry shape and amount of ^{235}U per fuel element, was purchased in 1976. From 1976 to 1979, the reactor operated with mixed 2% and 80% enriched fuel core.

The complexity of fuel depletion and decay analysis for the reactor RA arises from the effect of the end region between the fuel tubes of adjacent fuel elements in the reactor RA core. This effect can be accurately account only with detail three-dimensional (3D) model. One 3D model of reactor RA unit cell, used in the MCNP-4C and KENO-V.a Monte Carlo calculations is shown in Figure 1.

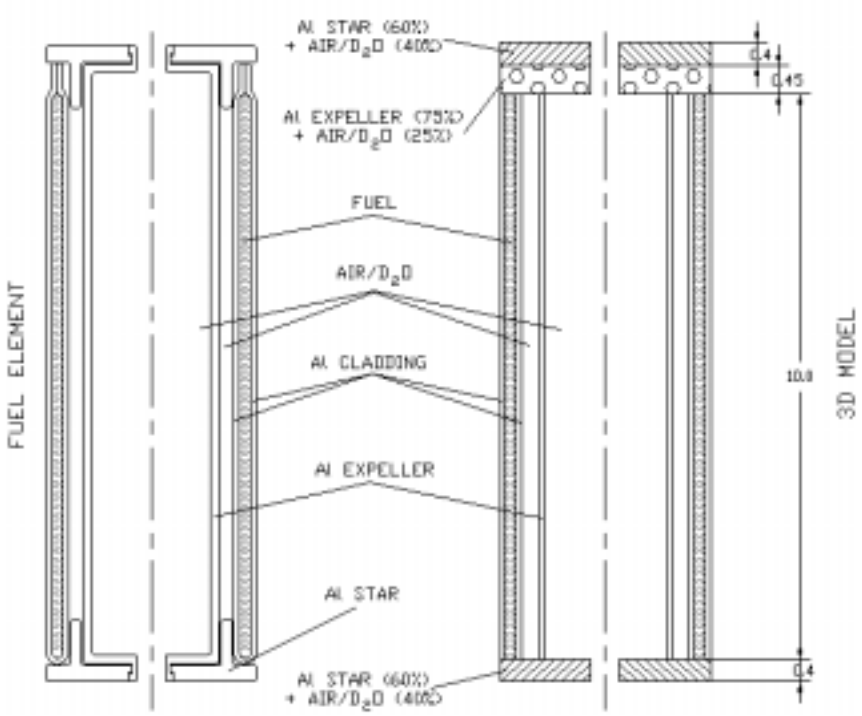

Fig.1. Vertical cross section of the fuel element (slug) and the 3D model

In this model, detailed design specifications for the reactor RA fuel element are modelled as closely as possible. In order to overcome the end region problem, a multiregion cylindrical Wigner-Seitz cell with white boundary conditions at the cell boundary (i.e., with isotropic reflection at the outer cylindrical boundary) is introduced [8]. It corresponds to an infinite array of infinitely long cylindrized cells. In this one-dimensional (1D) model, the aluminum from the end region is added to the inner and outer aluminum cladding and outer cylindrical moderator is enlarged for the volume of heavy water in the end region. The geometry of the proposed unit cell is given in Table I.

Table I. Geometrical Dimensions of the Wigner-Seitz Cell of Reactor RA

Zone	Material	Radius (cm)	
		3D-Model	1D-Model
1	Heavy Water	1.05000	1.00870
2	Clad (Al-Sav)	1.02000	1.20000
3	Heavy Water	1.45000	1.42153
4	Clad (Al-Sav)	1.55000	1.55000
5	Metal U (2% ^{235}U)	1.75000	1.75000
6	Clad (Al-Sav)	1.85000	1.87109
7	Heavy Water	2.05000	2.05000
8	Fuel Channel (Al-Yu)	2.15000	2.16217
9	Heavy Water (Outer Moderator)	Square Cell with Pitch of 13 cm	7.77937

3. Methodology

The analysis of reactor RA spent fuel characteristics (radiation sources, decay heat, and spent fuel isotopics) is based on the application of three procedures. The first, design-oriented procedure is based on the application of the SAS2H control module from the SCALE-4.4a code system and an structured approximate geometrical model, which uses a simplified unit fuel within an infinite lattice for the fuel burnup analysis (*asquarepitch* option for annular cylindrical rods in a square pitch). In this procedure, the five zones *latticecell* 1D model and SAS2H control module are used for calculation of burnup-

dependent fuel composition, and nine zones *multiregion* 1D model (given in Table I) and XSDRNPM code [9] (via CSAS control module [10]) are used for k_∞ evolution.

The second computational tool employed consists of the MCNP-4C and ORIGEN2.1 codes interfaced by the MOCUP driver. In addition to calculating, flux and power distribution, MCNP-4C calculates effective one-group cross-sections for the fuel constituents. These cross sections are used by ORIGEN2.1 code for burnup analysis. The cross-sections used for MCNP calculations are taken from the Vinca library VMCCS [11]; based primarily on ENDF-B/VI data but, if not available, on JENDL3.2 and BROND2 data. For isotopes not included in the MCNP-4C analysis, ORIGEN2.1 uses cross sections from its pre-processed standard libraries. The new ORIGEN2.1 one-group cross sections library (Hwrleu.lib) was prepared by using a detailed (3D) model of reactor RA unit cell and the MCNP-4C Monte Carlo code. This library replacement is very important in regard to the radiological characterisation of reactor RA spent fuel elements.

Recently developed, the third procedure [12] is based on the applications of the KENO-V.a (via CSAS control module) and ORIGEN2.1 utility codes with the ENDF/B-V based 44-group cross section library [13]. In this procedure the standard predictor/corrector steps are included (instead of predictor steps that are used in the MOCUP procedure). The major advantage of using KENO-V.a/ORIGEN2.1 procedure is the decrease in computation time by about a factor of 20 compared with MOCUP. In the both: MCNP-4C/ORIGEN2.1 and KENO-V.a/ORIGEN2.1 reference procedures, the fuel burnup model with 90 main fission products was used.

4. Results

The validity of the Wigner-Seitz cylindrization for the reactor RA unit cell was carefully verified by comparison between the Monte Carlo and transport calculations of the cylindrized cell and the initial square cell in 3D geometry. Results for various burnup conditions obtained as k_∞ evolution without any buckling corrections to the neutron spectrum are shown in Figure 2.

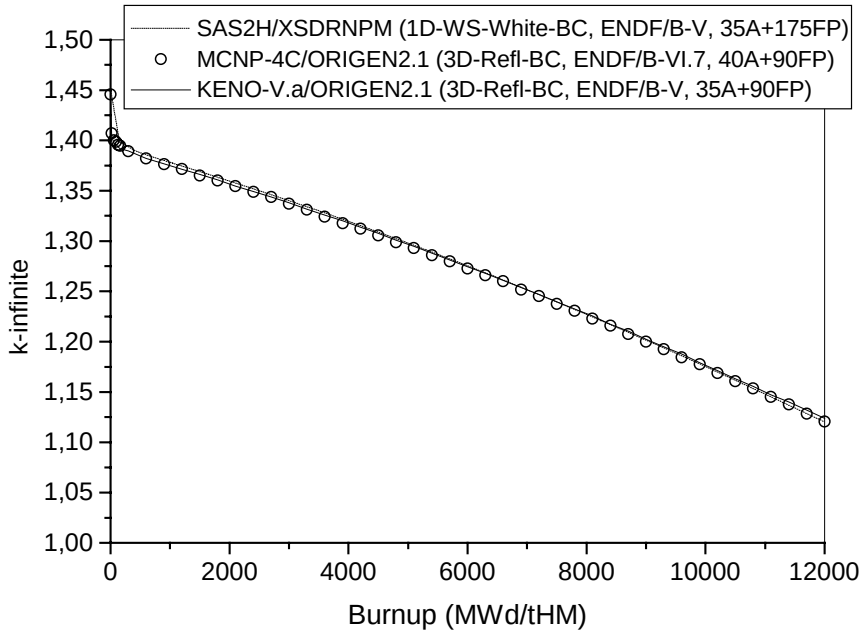

Fig. 2. k_∞ evolution for 1D and 3D models of reactor RA unit cell with metal uranium (2% ^{235}U)

The results obtained for the photon source from one reactor RA fuel element (metal uranium with 2% of ^{235}U) are given in Table II. This Table shows a good agreement between the MCNP-4C/ORIGEN2.1 (MOCUP) and KENO-V.a/ORIGEN2.1 reference procedures. Also, this table

confirms that difference between design-oriented procedure (based on the SAS2H control module with ORIGEN-S [14] code) and reference procedures is less than 10%. The energy spectra of photon source in one fuel element of reactor RA for different fuel burnup and for the cooling time equal 40 years are shown in Figure 3. The neutron and photon source spectra for the reactor RA spent fuel elements, generated by these reference procedures, are subsequently used for the Monte Carlo fixed source analysis.

Table II. Actinides and daughters plus fission products photon source (photons s^{-1}) from one reactor RA fuel element (metal uranium with 2% of ^{235}U)

Burnup [MWd/tHM]	Code system	Cooling time			
		1 day	20 years	40 years	60 years
3000	SAS2H/ORIGEN-S	$3.980 \cdot 10^{14}$	$2.580 \cdot 10^{11}$	$1.590 \cdot 10^{11}$	$0.990 \cdot 10^{11}$
	MCNP-4C/ORIGEN2.1	$3.768 \cdot 10^{14}$	$2.443 \cdot 10^{11}$	$1.594 \cdot 10^{11}$	$1.046 \cdot 10^{11}$
	KENO-V.a/ORIGEN2.1	$3.758 \cdot 10^{14}$	$2.371 \cdot 10^{11}$	$1.547 \cdot 10^{11}$	$1.015 \cdot 10^{11}$
6000	SAS2H/ORIGEN-S	$4.470 \cdot 10^{14}$	$5.050 \cdot 10^{11}$	$3.120 \cdot 10^{11}$	$1.930 \cdot 10^{11}$
	MCNP-4C/ORIGEN2.1	$4.018 \cdot 10^{14}$	$4.530 \cdot 10^{11}$	$2.994 \cdot 10^{11}$	$1.965 \cdot 10^{11}$
	KENO-V.a/ORIGEN2.1	$4.008 \cdot 10^{14}$	$4.518 \cdot 10^{11}$	$2.947 \cdot 10^{11}$	$1.934 \cdot 10^{11}$
9000	SAS2H/ORIGEN-S	$4.590 \cdot 10^{14}$	$7.240 \cdot 10^{11}$	$4.460 \cdot 10^{11}$	$2.760 \cdot 10^{11}$
	MCNP-4C/ORIGEN2.1	$4.233 \cdot 10^{14}$	$6.666 \cdot 10^{11}$	$4.347 \cdot 10^{11}$	$2.856 \cdot 10^{11}$
	KENO-V.a/ORIGEN2.1	$4.226 \cdot 10^{14}$	$6.595 \cdot 10^{11}$	$4.301 \cdot 10^{11}$	$2.826 \cdot 10^{11}$
12000	SAS2H/ORIGEN-S	$4.910 \cdot 10^{14}$	$9.520 \cdot 10^{11}$	$5.860 \cdot 10^{11}$	$3.630 \cdot 10^{11}$
	MCNP-4C/ORIGEN2.1	$4.486 \cdot 10^{14}$	$8.667 \cdot 10^{11}$	$5.651 \cdot 10^{11}$	$3.717 \cdot 10^{11}$
	KENO-V.a/ORIGEN2.1	$4.467 \cdot 10^{14}$	$8.598 \cdot 10^{11}$	$5.605 \cdot 10^{11}$	$3.687 \cdot 10^{11}$

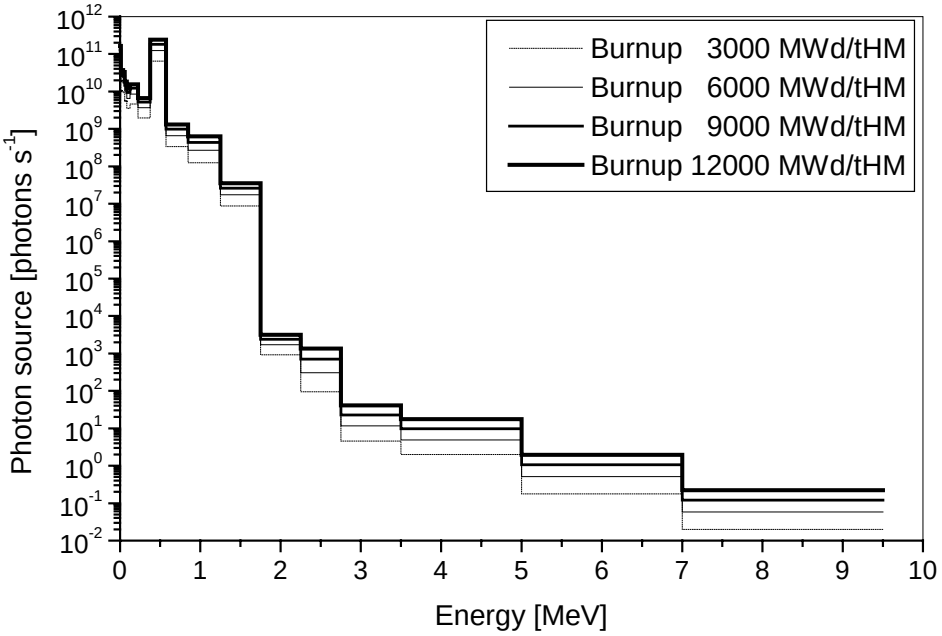

Fig. 3. Energy spectra of photon source in one fuel element of reactor RA with metal uranium (2% ^{235}U)

5. Conclusion

The current work confirms: (1) that the Wigner-Seitz modelling of the very short fuel element in the tubular form, where the realistic cell with this fuel element is considered as an infinitely long cylindrrized cell, is applicable in the neutronics and fuel depletion/decay analyses; and (2) that the procedures MCNP-

4C/ORIGEN2.1 (MOCUP) and KENO-V.a/ORIGEN2.1 are capable to provide the reliable radiological characterisation of reactor RA spent fuel storage.

6. Acknowledgement

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CRITICALITY SAFETY STUDY OF FRESH HEU FUEL ELEMENTS IN TK-S15 AND TK-S16 TRANSPORTATION PACKAGES

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ABSTRACT

This paper presents results of the criticality safety analysis for the fresh HEU nuclear fuel from Vinča RA and RB reactors transported in Russian containers TK-S15 and TK-S16. The goal of this study was to demonstrate the sub-criticality in cases of normal and various hypothetical incidental transport conditions. Analysis was done using MCNP-4B2 Monte Carlo code with VMCCS and TMCCS data libraries. Neutron effective multiplication factors were calculated for several recommended flooding incidental conditions for individual containers and for the infinite square and hexagonal arrays of containers. Results from all analysed cases proved sub-criticality of the system below 0.95, which is enough to allow the transportation without limitations in the packaging geometry and in the number of containers in vehicle. MCNP results are in good agreement with those obtained by KENO V.a code from SCALE 4.4a code system.

1. Introduction

High-enriched uranium fuel slugs of well-known Russian TWR-S fuel assembly type (Figure 1) were used in the RA and RB heavy water research nuclear reactors [1, 2], both operated by the VINČA Institute of Nuclear Sciences, Belgrade. Fuel element ("slug") is a cylindrical layer, 2 mm thick and 100 mm long. It is manufactured as 80%-enriched uranium dioxide dispersed in an aluminum matrix. Both sides of the fuel layer are covered with aluminum cladding, 1 mm thick. Aluminum stars are placed on the top and bottom of the slug, so the total length of the slug is 113 mm. An aluminum expeller (hollow cylinder inside the slug) is designed in order to adjust the coolant flow.

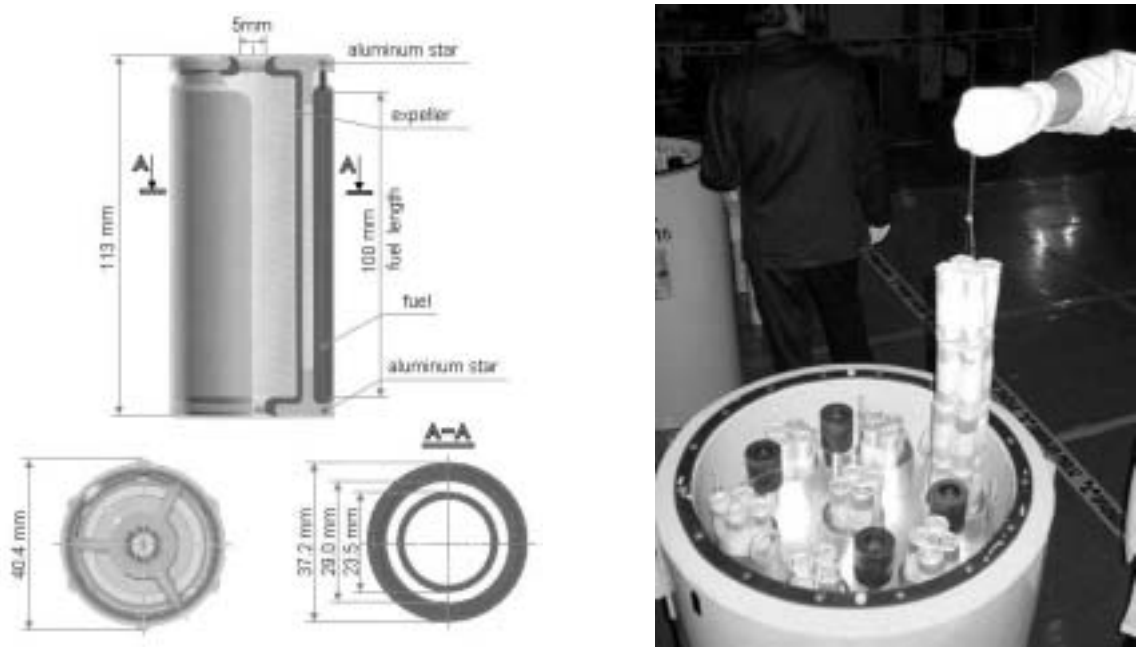

Fig.1 Design of TWR-S fuel assembly and placement of the TWR-S slugs into the TK-S16 containers

Russian TK-S15 and TK-S16 packages (Figure 2 and Figure 3) were originally designed for transportation of IRT-2M, IRT-3M and IVV-10 fresh fuel assembly types. Containers consist of a steel case and a cover. The container case for both types is a welded construction, like a barrel with double walls. The gap between the walls is filled with heat-insulating material. Inner equipment is a welded construction of seven aluminium tubes welded to spacer grids.

This paper is evaluating criticality safety in transportation of fresh fuel TWR-S slugs (with 80%-enrichment in ^{235}U) in TK-S15 and TK-S16 packages. In order to use TK-S15 and TK-S16 containers, which were not originally designed for TWR-S slugs, it was proposed packaging of the slugs as described in Table 1.

Table 1. TK-S15 and TK-S16 packaging

Container Type	Number of aluminium tubes	Number of layers	Number of slugs in each layer	Total number of slugs
TK-S15	7	13	2	182
TK-S16	7	8	4	224

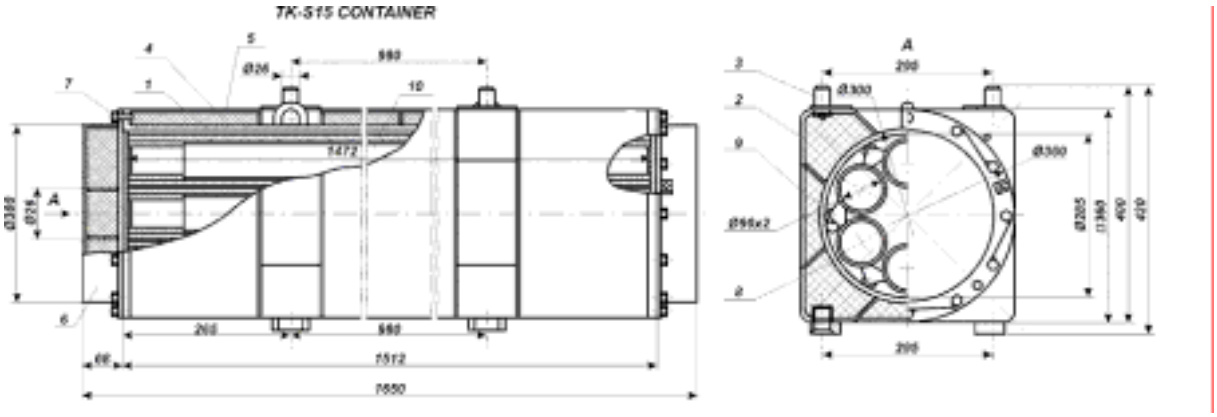

Fig. 2. TK-S15 transport container

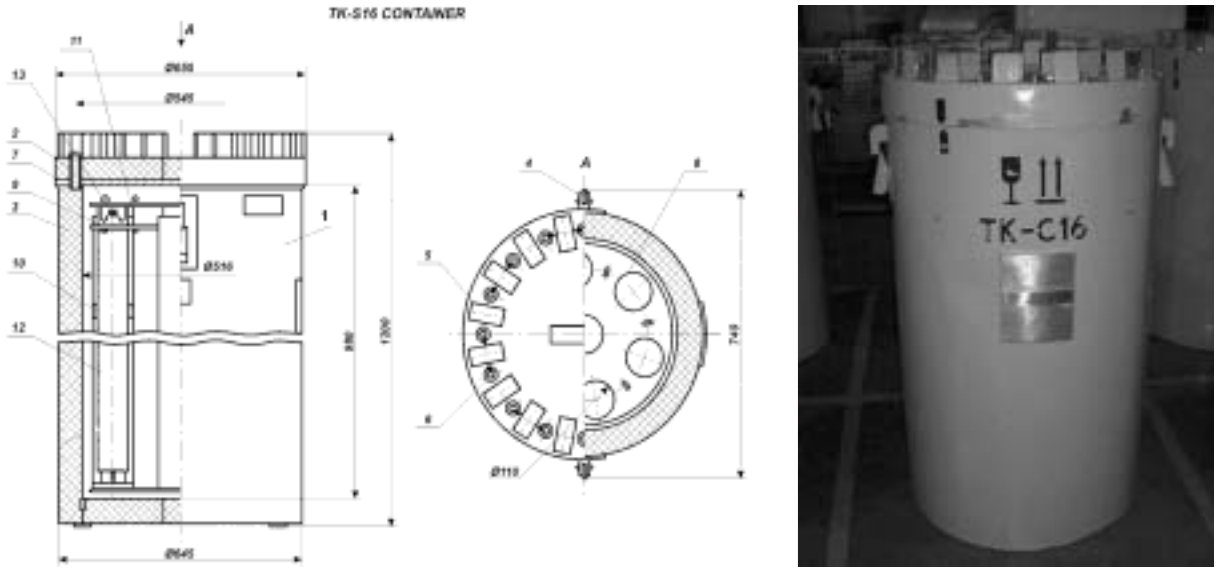

Fig. 3. TK-S16 transport container

2. Fuel element and container modelling

Containers filled with fuel slugs were modelled in 3D geometry using best available material composition data. All calculations were done by using MCNP4B2 computer code [3]. Figures 4 and 5 show both horizontal and vertical cross-sections of the modelled TK-S15 and TK-S16 containers. The horizontal cross-sections present the way seven aluminium tubes were filled with slugs, and vertical cross-sections present that there were thirteen and eight layers along the tubes, respectively.

Analysis was done using Monte Carlo method, incorporated in the MCNP-4B2 computer code, with Vinļa developed VMCCS [4] and originally distributed TMCCS (LANL) nuclear data libraries. According to the detailed material and geometry data available, three-dimensional models of the TWR-S fuel element bundles in transport containers were developed. Neutron effective multiplication factors were calculated for individual containers of each type in several normal and flooding incidental conditions as well as for the infinite square and hexagonal arrays of dry and flooded containers (the worst possible incidental case).

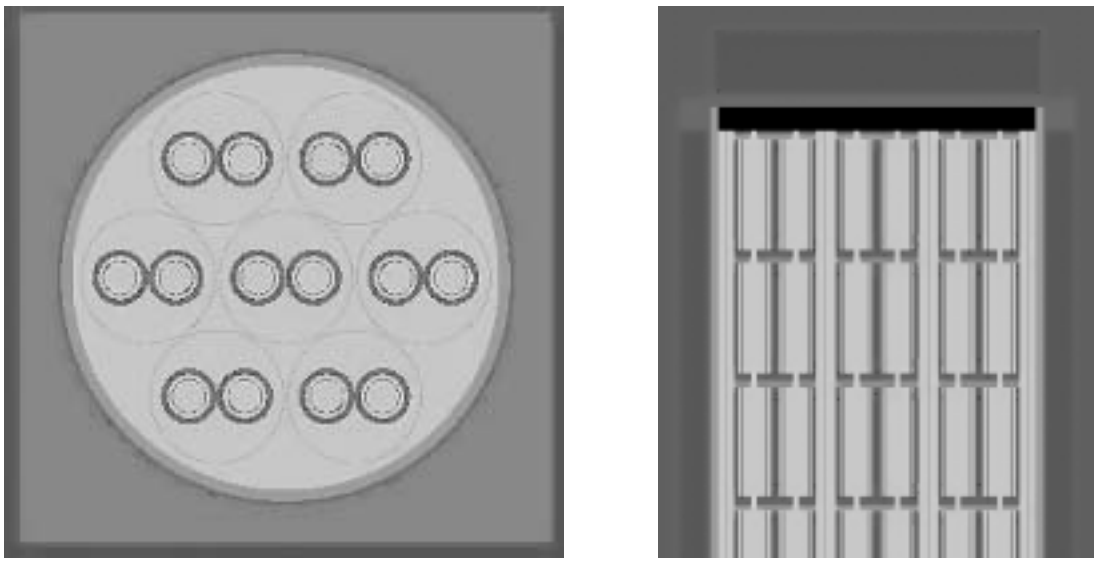

Fig. 4. The model of TK-S15 with TWR-S fuel assemblies

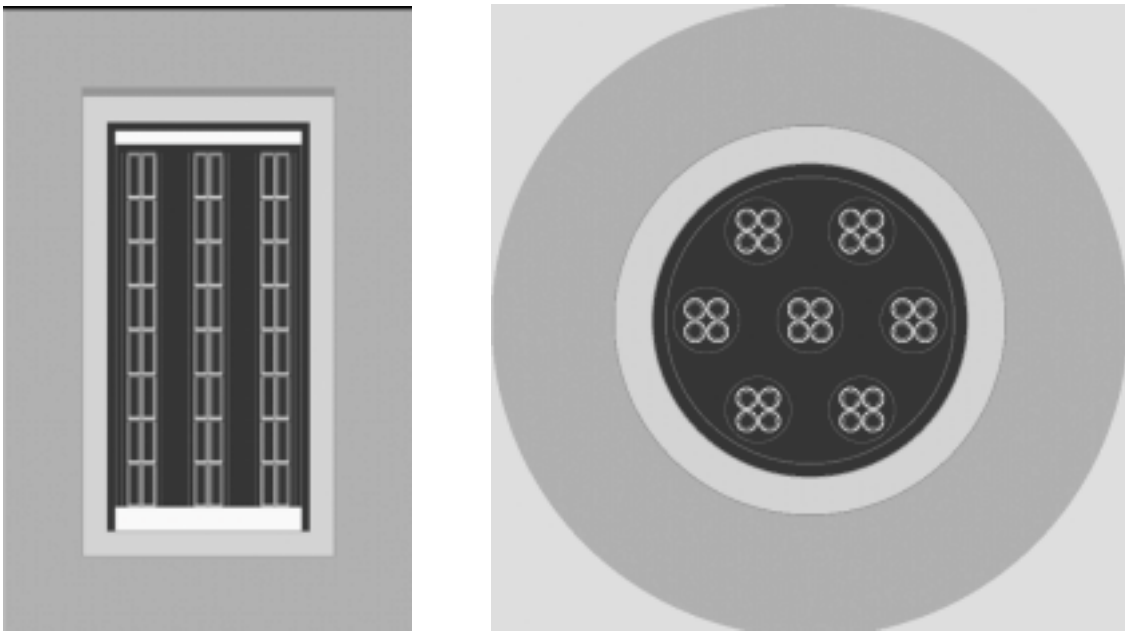

Fig. 5. The model of TK-S16 with TWR-S fuel assemblies

3. Criticality calculations and results

Criticality calculations covered normal conditions of the fresh fuel transportation, and several incidental ones, which theoretically may have occurred or which are recommended for analysis in the IAEA safety recommendation [5] and guides concerning the nuclear fuel transportation. Effective neutron multiplication factors k_{eff} from the MCNP-4B2 calculations are presented in Table 2. The last column shows the results obtained [6] by KENO V.a code from the SCALE 4.4a code system.

Table 2. Results of MCNP-4B2 criticality calculations compared with KENO V.a results

CASE	Container Type	$k_{eff} \pm 1\sigma$	
		MCNP-4B2	KENO V.a [6]
A single container, dry inside, surrounded by 20 cm of air	TK-S15	0.00413 ± 0.00001	-
	TK-S16	0.00443 ± 0.00001	-
A single container, dry inside, surrounded by 20 cm of water	TK-S15	0.15423 ± 0.00039	0.1404 ± 0.0003
	TK-S16	0.13503 ± 0.00025	0.1329 ± 0.0003
A single container, flooded, surrounded by 20 cm of water	TK-S15	0.55285 ± 0.00069	0.5553 ± 0.0006
	TK-S16	0.49735 ± 0.00051	0.4647 ± 0.0006
An infinite square array of containers, dry inside, surrounded by air	TK-S15	0.77544 ± 0.00036	-
	TK-S16	0.01135 ± 0.00001	-
An infinite square array of containers, dry inside, surrounded by water	TK-S16	0.18588 ± 0.00019	-
An infinite square array of containers, flooded, surrounded by water	TK-S15	0.70935 ± 0.00071	0.7262 ± 0.0003
	TK-S16	0.50802 ± 0.00069	0.4751 ± 0.0006
An infinite hexagonal array of containers, dry inside, surrounded by air	TK-S16	0.01320 ± 0.00001	-
An infinite hexagonal array of containers, dry inside, surrounded by water	TK-S16	0.23806 ± 0.00020	-
An infinite hexagonal array of containers, flooded, surrounded by water	TK-S16	0.51561 ± 0.00032	0.4777 ± 0.0006

Results for all analysed cases proved deep sub-criticality of the system, which is enough to allow the transportation of TWR-S fuel slugs (according proposed packaging procedures) without limitations in the packaging geometry and in the number of containers in array. MCNP results are generally in good agreement with those obtained by KENO V.a code from SCALE 4.4a code system. Some disagreements between the results are consequence of slightly different geometry data used in calculations presented in this paper and in ref. [6]. The first analysis of the calculated results shows that they depend very much on the nuclear data for iron. Additional calculations, carried out with iron data from several different nuclear data libraries, were done in order to estimate this sensitivity and significantly different results were obtained. This is another reason for the disagreement between two sets of the results (KENO V.a and MCNP-4B2).

4. Conclusions

According to the IAEA standards [5] transportation of fissile materials is allowed, without any further limitations, if the value of k_{eff} is less than 0.95.

Our evaluation proved deep sub-criticality of a single package and of an array of packages during normal conditions of transport and typical hypothetical incidental conditions.

According to the decision brought by the Yugoslav and Serbian governments in July 2002, related to the RERTR Program [7], all fresh TWR-S high-enriched uranium fuel elements from the RA and RB reactors in Vinļa has been packed in TK-S15 and TK-S16 containers and transported to the country of origin – Russia, in August 2002 [8].

5. Acknowledgements

This work was done as a part of Project "Transport Processes of Particles in Fission and Fusion Systems" supported by the Ministry of Sciences, Technologies and Development of Republic of Serbia under the Contract No. 1958. Authors are very grateful to Mr. MiloševiĆ for useful discussion during the analysis of results.

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RESEARCH REACTOR RECORDS IN THE INIS DATABASE - A BIBLIOMETRIC STUDY -

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ABSTRACT

This report presents a statistical analysis of more than 13,000 records of publications concerned with research and technology in the field of research and/or experimental reactors which are included in the INIS Bibliographic Database for the period from 1970 to 2002. The main objectives of this bibliometric study were: to make an inventory of research reactor related records in the INIS Database; to provide statistics and scientific indicators for the INIS users, namely science managers, researchers, engineers, operators, scientific editors and publishers, decision-makers in the field of research reactors related subjects; to extract other useful information from the INIS Bibliographic Database about articles published in research reactors research and technology. Special attention is devoted to publications related to fuel management and RERTR issues. The quantitative data in this report are obtained for various properties of relevant INIS records such as year of publication, secondary subject categories, countries of publication, language, publication types, literary types, etc.

1. Introduction

Fission reactor research and technology is one of the most important subjects in the scope of the INIS database. Literature in fission reactor categories covers the following topics:

- reactor theory, reactor physics calculation, computations of in-reactor processes;
- design, construction, fabrication and performance of reactor components and accessories;
- design, fabrication and performance of fuel pellets, fuel elements and fuel assemblies, fuel loading procedures;
- fuel fabrication plants, including technical aspects of safety, decommissioning and dismantling;
- systems for control and surveillance of reactors, including safety and computerised control systems, man-machine interaction problems in reactor control;
- technical aspects of safety;
- design, construction, performance, operation, decommissioning and dismantling of specific reactor types and related nuclear power plants as energy sources for electricity generation and other applications including fuel elements, components and accessories, but not control systems;
- real accidents

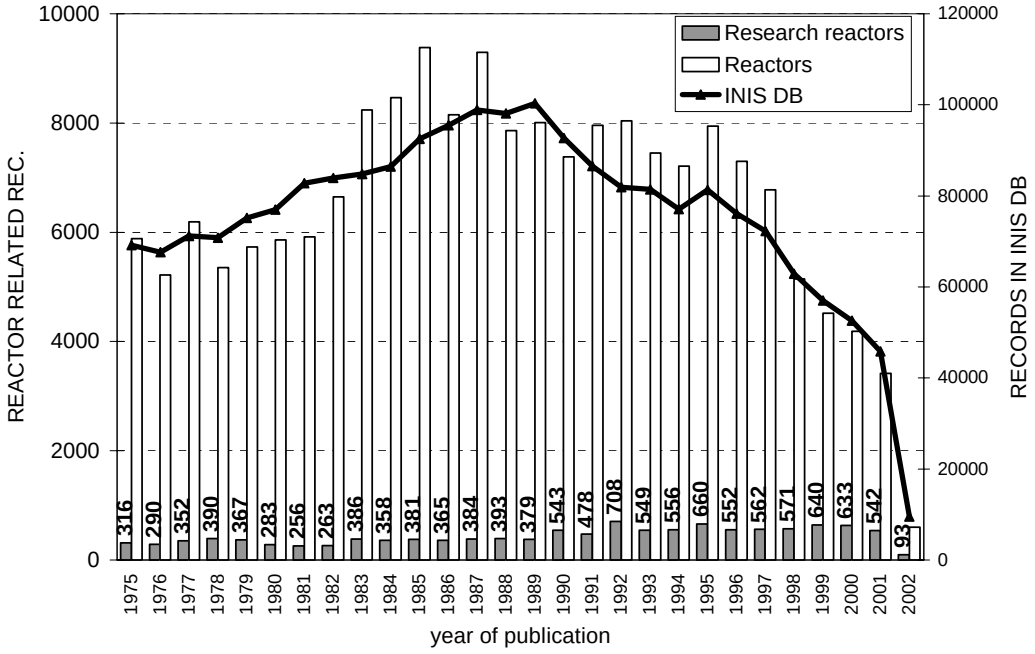
Each record in the INIS Database is assigned a primary subject category. According to the above mentioned fields of interest records related to fission reactors research and technology in INIS Database are assigned a specific subject category.

2. Records related to Research Reactors in the INIS Database

This analysis deals with records in the Research, Test, Training, Production, Irradiation and Materials Testing Reactors, category and covers the period from 1970 until the end of September 2002. The INIS Database is published on CD-ROM by the INIS Secretariat. It comprises seven archival discs covering the period from 1970 through 2000 and one current disc. The last CD taken into account for

the present analysis was labelled “2001-2002/09”. In time dependence analysis, records published before 1975 were not included, since records collected during the start-up period of the INIS Database from 1970-1974, suffer from incomplete literature coverage. The following period, from 1975-2002, was considered as relevant for this analysis. Records input in 2000, 2001 and 2002 must be considered as incomplete, due to the lead-in time for the database.
In September 2002, the total number of records in the INIS Database was 2,339,341 of which 199,010 records fall into fission reactor categories. The total number of records related to research reactors was 13,547. This means that about 8.5% of the records in INIS Database are in fission reactor categories. About 7% of records in fission reactor categories are related to research reactors.
Full text of 562,067 items present in the INIS Database are available from INIS (so called non-conventional literature, NCL) of which 71,405 most recently submitted ones are available from INIS in electronic form. Research reactors related articles available as full text from INIS amount to 6,651, of which 1,547 in electronic form.
The number of records related to research reactors versus year of publication is shown in Fig 1. These data are compared to the number of records in all the categories related to fission reactors and the entire INIS Database. The maximum number of records in research reactor category input in the INIS Database are related to items published in 1992. In the following few years this number was almost constant. This trend is similar to the total number of records in fission reactor categories.

Fig 1. Research reactor records vs. year of publication



2.1. Origin of input

During the considered period, the records in the INIS database were sent by 119 Member countries or organisations, 87 countries/organisations have sent records assigned with primary fission reactor subject categories, and 77 countries/organisations contributed to the total of 13,547 records classified as research reactors. According to RRDB [3], 69 countries have 670 research reactors (operational, shutdown or decommissioned). Share of records submitted by the most important countries is shown in Table 1.

Table 1. Share of country/organization of input

Country/organization of input	Share (%)
USA	38.8
Japan	12.7
IAEA	7.1
Germany	5.8
China	3.0
Brazil	2.8
USSR	2.2
UK	2.2
Canada	1.9
India	2.0

The highest contribution came from countries having the most developed nuclear programs which involve research and development in the field of fission reactor physics and technology. Ten countries/organisations have sent about 80% of all records related to research reactor technology. This indicates the interest in research reactors technology in the mentioned countries. The share of the IAEA (7.1%) is high due publishing of articles, reports, books, in the field of fission reactors resulting from the Agency's task to promote nuclear technology and co-ordinate activities in development of related technology.

2.2. Languages

Table 2. Share of languages

Language of text	Research reactors	INIS DB
English	74%	72%
Japanese	7%	3%
Russian	4%	12%
German	4%	5%
Portuguese	2%	1%
Chinese	2%	1%
French	2%	2%
Other	5%	3%

The majority of records related to research reactors refer to documents written in English (74%) which is somewhat higher than the use of English language in the total INIS Database (72%). One could conclude that the majority of authors from non-English speaking countries are submitting their publications in English, i.e. that the language of research and technology in the field of research reactors is English. The same conclusion is valid for the publications related to all other subjects in the scope of the INIS Database. Share of languages used in publications related to research reactors is shown in Table 2.

2.3. Publication types

Each record in the INIS Database indicates which type of publication it represents. The seven pre-defined types are: book, journal-article, report, miscellaneous (non-conventional literature), patent, computer-medium, translation and audio-visual material. The share of publication types for items related to research reactors is as follows: 46% of records are reports, 29% journal articles, 13% books, 1% computer media, 10% miscellaneous, 1% are patents. This distribution is almost the same as the share of publication types in the INIS Database as a whole.

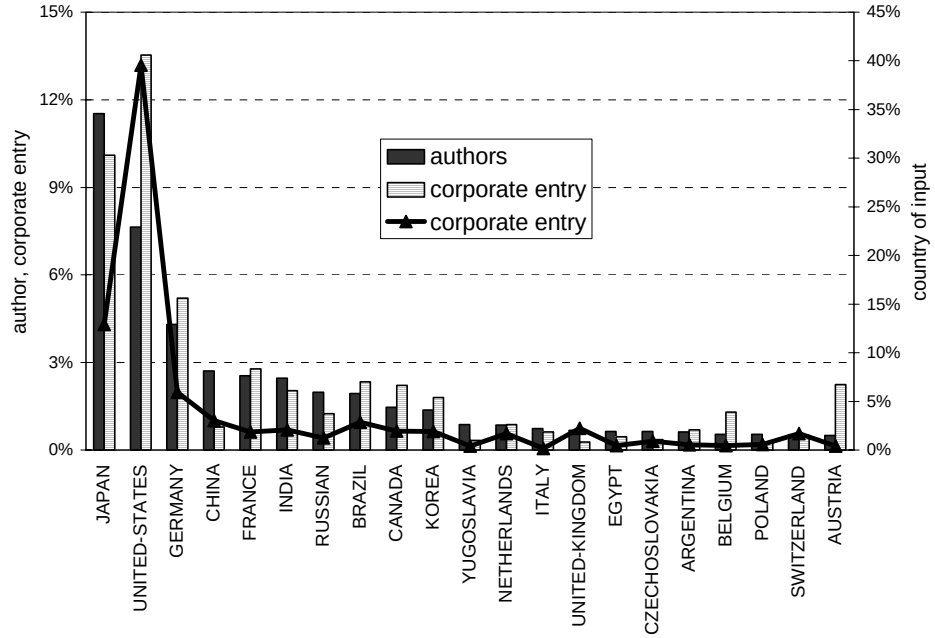
2.4. Literary indicators

Besides record type, each item in the INIS Database is assigned a literary indicator. The following 10 literary types are defined: short communication, conference, dictionary, numerical data, legal material, thesis or dissertation, computer program description, standard or specification, progress report, bibliography. The full text of non-conventional literature is usually available from INIS unless it is marked with the special literary indicator "X" (unavailable from INIS). The majority of records related to research reactors in the INIS Database are articles presented at conferences (65%), which is valid for the entire database as well. The distribution of records by literary indicators is the following: 16% are short communications; 7.8% are numerical data; 1.1% are thesis; 3.3% are progress reports; 0.1% are computer programs. The share of literary indicators for research reactors related items is similar to the share for the database.

2.5. Authors, Institutions and Countries

In the INIS Database, names of authors, institutions and countries of origin. are commonly specified for journal articles and books, but exist in many cases for other record types as well.
Corporate entries include corporate authors i.e. institutions responsible for the scientific content of publications and issuing organizations which are mentioned in the publication as editorially or organisationally responsible for the content of publication. Corporate author entries include academic institutions granting a degree, institutions organizing and/or sponsoring scientific meetings, as well as names of research institutes, laboratories, and names of conferences, symposia, meetings.
Country specific statistics referring to author's countries as well as to corporate entries are presented in Figure 2. These data are independent of the country/organization of input, which are listed as well.

Fig 2. Share of countries in author, corporate entry, and country/organization of input fields



The frequency of appearance of country names in the author field was: Japan contributed 11.4%, USA 7.7%, Germany 4.4%, China 2.5%, France 2.5%, India 2.3%. The comparison of the ranking order of countries according to their appearance in author and corporate entry fields shows a difference. The ranking of country name in corporate entry field is different: USA is leading, contributing 13.8%, followed by Japan (10%), Germany (5.4%), France (2.7%), Canada (2.3%), etc.

3. Research reactor records related to fuel management issues

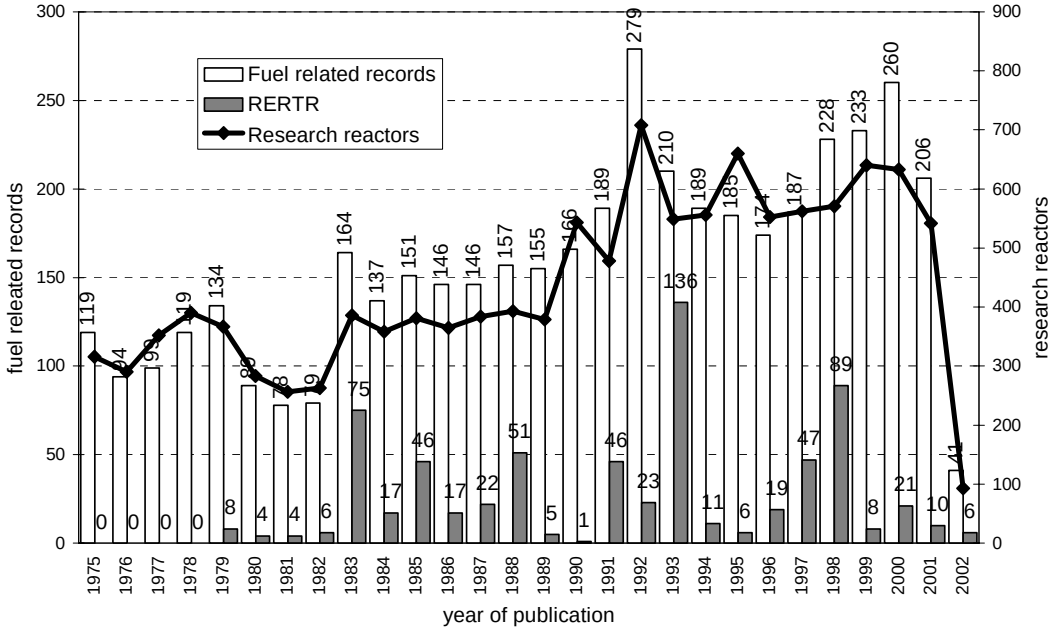
A special analysis was performed to show the significance of fuel management issues in the field of research reactors. Out of 13,547 records related to research reactors 4,881, i.e. 36% were related to fuelling, fuel management, spent fuel problems, new types of fuel elements, of which 678 (14%) mention the RERTR issues explicitly. Time dependence of input of these items into the INIS Database is compared to that of the research reactor records in Figure 3.

4. Concluding remarks

Quantitative data obtained in the present analysis show that 7% of records concerned with fission reactor related subject categories in the INIS Database are related to research reactors. Full text of 562,067 items present in the INIS Database are available from INIS (so called non-conventional literature, NCL) of which 71,405 submitted ones are available from INIS in electronic form. Research reactors related articles available as full text from INIS amount to 6,651, of which 1,547 in electronic

form. Number of countries/organizations that have contributed to the research reactor category was 77 out of 119 INIS Member States. Ten countries/organisations contributed about 80% of records falling into research reactor category. These are the countries with most developed nuclear programs and the highest number of operational and/or shutdown research reactors. This fact indicates the interest in research reactor related research and technology in the mentioned countries. The case of the IAEA shows the interest of the international community in research and technology related to research reactors, since its task is to promote and co-ordinate activities in development of nuclear science and technology.

Fig 3. Research reactor records related to fuel management issues



The majority of records in the research reactor category refer to pieces of literature written in English (74%). One could conclude that the language of research and technology in research reactors is English, which is valid for the publications related to fission reactors and all the subjects in the scope of the INIS Database (72%).

The distribution of the number of research reactor records by record type shows that 46% are reports, almost one third (29%) are journal-articles, and 13% are books. Analysis of the number of records by literary type shows that the majority of records (65%) in research reactor category were published as conference publications.

The appearance of country names in the author field was: Japan 11.4%, USA (7.7%), Germany 4.4%, China 2.5%, France 2.5%, India 2.3%, Russian Federation 1.9%, Brazil 1.9%, Canada 1.9%, etc. The appearance of country name in the corporate entry field is different: USA is leading contributing 13.8%, followed by Japan (10%), Germany (5.4%), France (2.7%), Canada (2.3%), Brazil (2.1%), etc.

Time dependence of number of records related to fuel issues compared to entire number of research reactor records shows that the share was almost constant in time. This is not valid for records dealing with RERTR, which shows two peaks, in 1993 and in 1998.

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STUDY ON INFLUENCE ON CORROSION AT ALUMINIUM-CLAD SPENT FUEL OF RESEARCH REACTOR

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ABSTRACT

Study of influence of corrosion process at the SAV-1 aluminium cladding of the TVR-S type of enriched uranium spent fuel elements of the research reactor RA in the storage water pool is examined within the framework of the IAEA CRP "Corrosion of Research Reactor Aluminium-Clad Spent Fuel in Water". Results of periodic monitoring of chemistry and radioactivity of the water samples, taken from the RA spent fuel storage pool, are given. Examples of corrosion process found at various metal surfaces are given. Initial results of examination of the rack with aluminium and stainless steel coupons, stored earlier in the pool, according to the strategy and the protocol supplied by the IAEA, are described.

1. Introduction

The TVR-S fuel element is 11.25 cm long cylinder with 3.72 cm outer diameter (Figure 1) produced in the Elekhtrostal plant near Moscow (2 % LEU metal fuel) or latter in the Novosibirsk Chemical Concentrates Plant (80% HEU-oxide fuel). These elements are used in 6.5 MW heavy water RA research reactor in Vinļa Institute since 1959 to 1976 (LEU) and from 1976 to 1984 (HEU).

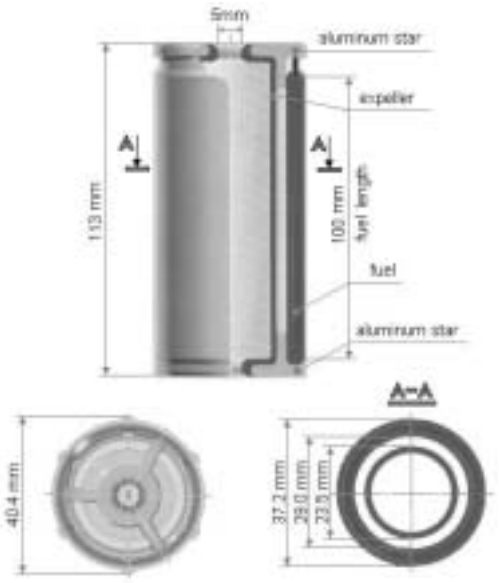


Fig. 1. TVR-S fuel element

The tube-type fuel layer of the TVR-S element has average length 100 mm and ID/OD 31/35 mm. Mass of ^{235}U nuclide in the fuel element is 7.4 g in case of LEU and 7.7 g in case of HEU. It is cladded on inner and outer side by 1 mm thick Al. An inner tube (the "expeller") designed from Al within the fuel element serves to adjust coolant flow rate. Top and bottom of the slug are covered by the Al "stars" (3 mm thick each) connected to the expeller tube. The aluminium, used in the TVR-S fuel elements, is known as the SAV-1 alloy (0.985 weight fraction of Al). The main impurities in the alloy are Mg and Si. Content of impurities with high absorption cross sections for neutrons (B and Cd) are very low. Volume of the TVR-S element is measured at (58 ± 2) cm^3, volume of the SAV-1, excluding fuel layer, is estimated at 40 cm^3. Area of the SAV-1 surfaces in contact with the water is calculated at 420 cm^2.

2. Spent fuel storage

Almost all spent fuel elements of the RA reactor are stored in the temporary spent fuel storage pool Six and a half meter deep pool, within the reactor building, consists of four inter-connected basins and an annex to the fourth basin (Figure 2) [1].

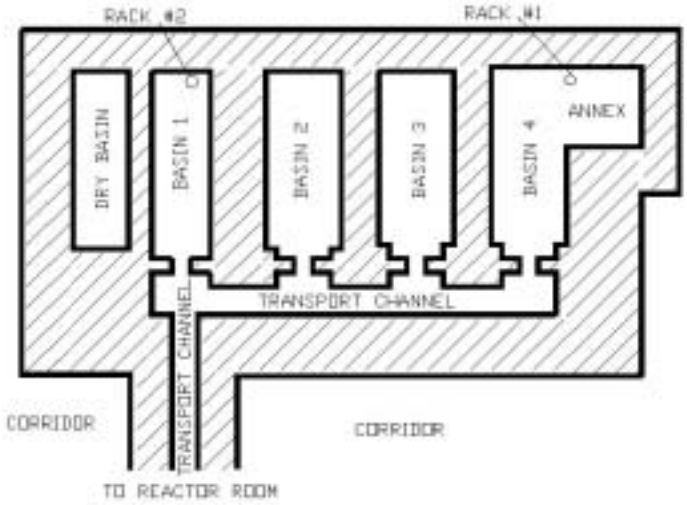


Fig. 2 Cross section of RA reactor spent fuel pool

Each basin has rectangular cross-section with dimension: width 1.60 m (except basin no. 1 that has 1.25 m width) and length 3.80 m. The annex has 1.60 m width and 1.70 m length. Each basin can be isolated (closed, but not hermetically) by a door manufactured from carbon iron. Thin lid plates made of carbon iron cover basins. Thick concrete walls and bottom of the pool are lined by stainless steel plate, 1 cm thick. The pool, filled with about 200 m^3 of stagnant ordinary tap water, is connected by special underground water transfer channel to the reactor body that allows transfer of irradiated fuel elements to the storage area. Due to water vaporisation, tap water is added once per year.

3. Spent fuel inventory

Initially, irradiated LEU fuel elements were stored in 302 original stainless steel channel-type containers (SSC), filled with de-mineralised water, immersed into water of the basins. Each SSC has a carbon iron cylinder at the bottom, acting as the weight against water lift force. It was planned, after 4-5 years of cooling in this temporary storage, to transfer the spent fuel elements back to ex-USSR, but it did not happen. In aim to increase the storage capacity, new storage Al barrels (ALB, known also as the "casks") were designed. Each ALB could be filled with maximum 180 spent fuel elements displaced in 30 Al tubes within barrel body. The ALB is filled with de-mineralised water to cover the fuel elements and sealed. An air gap is left (few centimetres) above water level under the ALB lid. Thin plates of Cd were inserted into water in the ALB to assure sub-criticality. The exact composition of the Al used for ALB construction is not known. It is believed that this Al was produced in ex-Yugoslav factories in sixties. Any difference in Al composition, compared to SAV-1, could increase corrosion rate, because galvanic couples are created. From the beginning of sixties until 1984, about 5000 oldest LEU fuel elements were repackaged from SSC into 30 sealed ALB and stored in two rows (one above the other) in the annex basin to the basin no. 4. About 1600 LEU spent fuel elements and about 900 HEU spent fuel elements remained in SSC. Four hundred eighty HEU spent fuel elements have remained in the drained RA reactor core, since 1984. Design of the TVR-S fuel elements was very reliable. Only one LEU fuel element (from the total of 6656) failed in the core during 18 years of the RA reactor operation. The fuel channel containing damaged fuel element was identified quickly. The channel was replaced in the core by a new one. Failed fuel was stored in a marked SSC in the spent fuel storage pool.

4. Short historical data of pool water

First report about bad quality of water in the spent fuel storage basins, based on visual inspection, was given in the reactor annual report in 1962. Water purification was proposed, but no actions were taken. In 1984, a low activity of ^{137}Cs in water of the storage pool was discovered for the first time and attributed to leaking of the one spent LEU TVR-S fuel element (in 1976) at the bottom of the pool. Periodical monitoring of chemical and radiation parameters of the pool water, water purification system and appropriate regulation rules were proposed. However, none actions were taken until to 1994 when visual inspection of the storage pool discovered thick deposits of sludge at walls and bottom of the pool (about 10 cm), indicating that detail inspection should be made. Experts from the IAEA and abroad were called to help. The first chemical and activity analyses of water samples taken from the basins and SSC and sludge samples collected from the pool bottom were made in 1995. Improper chemical parameters and gamma-ray activity originated from ^{137}Cs and ^{60}Co nuclides were

found. Analyses have shown that the pool water is high corrosive [1] to Al alloys and has activity in range (100 - 130) Bq/mL, attributed to ^{137}Cs nuclide.

Activities of sludge samples were measured in 1996-1997 by Ge gamma-ray spectrometers in the Vinļa Institute and in the IAEA laboratories. Chemical composition of the few sludge samples was determined in radiochemical laboratories of the IAEA [2]. It was shown that main component of the sludge, beside Al oxides, is Fe_2O_3 (about 83% by weight) that gives the dark red – brown colour to the sludge. Visual inspections of TVR-S fuel elements in some SSC and in fuel channels stored in the reactor core have shown that deposits (determined lately to be mainly aluminium-hydroxide) cover Al cladding of the fuel elements. Stains and surface discoloration are present on many of the spent fuel elements that were examined visually during the core unloading and inspections carried out in 1979 – 1984. Creation of these deposits, stains and surface discoloration was attributed to poor chemical parameters and reduced flow rate of heavy water - primary coolant in the RA reactor core.

Some of water samples, taken from several SSC, have shown very high ^{137}Cs activity (2 kBq/mL – 400 kBq/mL) compared to low activity (50 Bq/mL - 400 Bq/mL) measured in samples taken from other SSC. It was an indication that Al cladding of the fuel elements was penetrated as result of corrosion process and that probability of leakage of fission products from particular fuel elements in SSC could be very high. Experts from Russia were engaged to make special equipment for underwater drilling of the ALB in aim to check fuel cladding integrity, activity of water inside the barrels and gas high pressure of corrosion and/or fission products. Sixteen ALB of the total thirty ones were drilled by the end of 2001. Gas high pressure was not detected leading to the conclusion that ALBs were not leak-tight anymore. However, leakage of gas from ALB was detected and recorded by a camera during ALB under water moving. Also, very high total activity of water samples (from 0.5 MBq/mL to 1.5 MBq/mL, attributed mainly to ^{137}Cs), taken from few ALB, was measured. It was concluded that, not only the first fission products barrier (fuel element cladding), but also the second one (ALB walls), are penetrated by corrosion process. It is also believed that Cd strips, placed initially in the ALB to assure sub-criticality, made galvanic couple with Al and increased corrosion processes.

Serious actions for improving conditions and remedy of the water in the pool are under way since 1997. Debris and most of the sludge deposits (about 3 m^3) were removed from the basins of the pool. Many corroded, elements were found and removed, but some large ones remained in the pool. Special equipment was designed for the sludge removing and treatment, using technique developed in the Vinļa Institute for sedimentation and cementing of the sludge-water mixture. Approximately 90 L of immobilised sludge-cement mixture is stored in the 200 L barrels, shielded inside by a 7.5 cm thick wall made of ordinary concrete. About 40 such barrels are stored as LLW in an interim waste storage within Vinļa Institute. It is estimated, according to activity measured in the sludge, that every barrel contains activity of about 110 MBq from ^{137}Cs and 13 MBq from ^{60}Co. Further activities in the spent fuel pool will include: removing the carbon iron structure from basin no. 4, removing rest of the sludge and corrosion deposits using hydro-monitor, pool water purification, and regular monitoring of the water chemistry and ^{137}Cs activity in the pool.

5. Chemistry and radioactivity of pool water

Chemical parameters of the water samples taken from the basins of the storage were done few times per year since 1997 [3], and approximately once per month, since mid-2001. The samples were analysed for the most of the main parameters related to corrosion process. Typical values of the basins water parameters, obtained during 2002, are given in Table I.

Typical average activity of pool water in 2002 is (80 ± 8) Bq/mL from ^{137}Cs nuclide. Activity of ^{60}Co nuclide is under detectable limit (1 mBq/mL). Activity of ^{137}Cs nuclide in the pool water is consequence of cladding failure of few fuel elements due to corrosion process.

Table I Pool water chemical parameters [5]

Water parameter	Mean value	standard deviation of mean
pH factor	8.03	0.10
El. conductivity (μS/cm)	451.3	27.6
Fe ions (mg/L)	0.13	0.01
Cu ions (mg/L)	-	-
Al ions (mg/L)	-	-
Chlorides, Cl (mg/L)	80.0	1.9
Sulfate ions, SO_4 (mg/L)	58.6	2.2
Nitrate ions, NO_3 (mg/L)	0	0
Hardness, dH	6.9	0.1

6. Initial results of the rack#1 examination

To investigate corrosion processes in the pool water, a coupon rack [4] prepared in the IAEA for the CRP, was immersed in water of the annex to the basin no. 4 (Fig. 2, RACK#1) of the RA reactor spent fuel storage pool, near to ALBs, in 1996. It was taken out in summer 2002 (Fig. 3) and will be analysed according to protocol prepared in the IAEA. The new rack (Fig. 2, RACK#2) with Al and stainless steel coupons was immersed into basin no. 1 in March 2002. It is expected that two new racks will be delivered to Vinļa in spring 2003 and immersed in the pool by mid-year. The pH factor of wet surfaces of the rack#1 was about 7, while gamma-ray dose rate was 3.5 μSv/h. The rack#1 was placed in a glass beaker and covered by the pool water from July 2002 to January 2003, when the first examinations were started. We withdrew the rack#1 from the glass beaker and measured the pH of water on external surface of coupons. Two coupons on the top of the rack were dry, so we couldn't measure pH. Values of pH (obtained by using pH paper) on external surface of others coupons were in range from 4.5 to 6.0. The pH factor of the bulk water, i.e., water in the glass beaker, was about 7.0. The rack was disassembled and Al coupons were removed from the rack according to the IAEA Test Protocol. Photographs of the front (Fig. 4, "A") and the back (Fig. 5, "B") of each coupon are presented. Numbers correspond to aluminium alloy types.

Fig. 3. Rack#1

Fig. 4. Front side of coupons

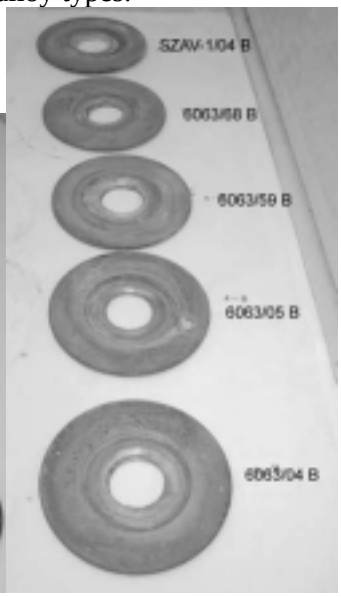
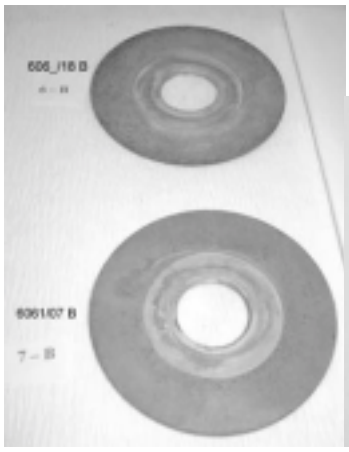
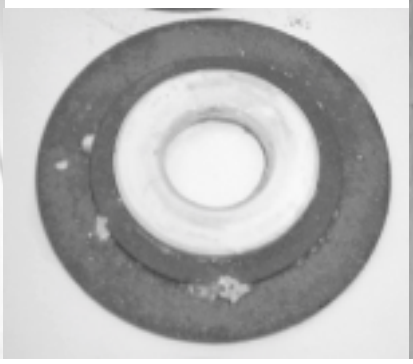


Fig. 5a. Back side of coupons

Front sides of the coupons are covered by dark red sludge (Fig. 4) from the pool water and the corrosion process can not be seen without cleaning surfaces. Also, we can see white deposits (Fig. 6) at front sides of few coupons. Some of these deposits are collected for further analysis. Back side of the coupons (Fig. 5a and 5b) is not covered by sludge and we can see effects of corrosion process (Fig. 6). Pitting is a main localised form of corrosion of aluminium in water basins and could be seen at surface of all coupons. As an example, photograph of back side of coupon marked 606_/18 is given in Fig. 7. Last digit of the number is covered by a black spot. We believe that the black spot is a type of aluminium-oxide. Oxides of different shades of grey colour were observed.





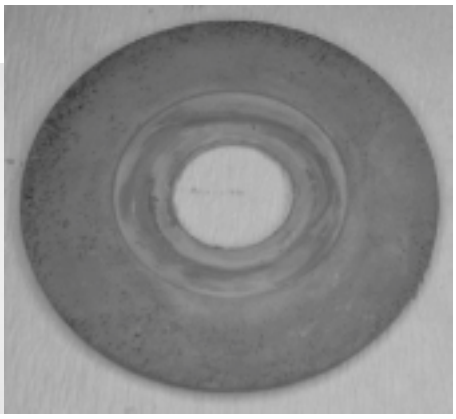


Fig. 5b. Back side of coupons Fig. 6. Deposits Fig. 7. Corrosion process

7. Conclusion

Corrosion of RA research reactor aluminium-clad spent fuel of TVR-S type in water is studied in the Vinļa Institute of Nuclear Sciences, Yugoslavia, as a part of the IAEA CRP. The basic information related to the spent nuclear fuel and the storage pool, including results of monitoring of chemistry and radioactivity the pool water are given. Data on corrosion process of aluminium, obtained from initial analysis of the rack#1, that was immersed in the pool water six years ago, are shown. Corrosion process is confirmed at all aluminium coupons. Numerous white deposits are present, too. Further analyses of the coupons, according to the IAEA Test protocol, will be made in near future.

8. Acknowledgement

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REPROCESSING U-MO SPENT FUELS: RESEARCH PROGRAM

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ABSTRACT

U-Mo fuels, containing up to 10 mass% of molybdenum, are promising low enriched uranium fuels (less than 20 mass% of ^{235}U). Although their reprocessability is ascertained, more precise reprocessing conditions are now under studies. The process would consist at first in a specific dissolution in nitric acid media. The obtained solution could then be diluted into standard UOx type fuels dissolution solution. Uranium and plutonium could be selectively recovered from this feed solution using adapted PUREX process (liquid/liquid extraction using tributyl phosphate). Tests on U-Mo powder are the first step of the planned studies to define dissolution conditions. Dissolution experiments on inactive pieces of U-Mo fuels will then be conducted according to the previously defined conditions. At last, complete reprocessing test of U-Mo spent fuels (irradiated inside a French reactor) - that means dissolution, purification of U and Pu, fission products solution concentration and waste management will be carried out. All this program will be implemented on the French Atomic Energy Commission Center CEA/VALRHO – Marcoule by the end of 2005.

1. Introduction

UMo fuels have been developed since 1978, in the frame of the US program “Reduced Enrichment for Research and Test Reactors”. Although they present relatively low enrichment (less than 20 mass% of ^{235}U) they are very efficient fuels for research reactors, due to their very dense metallic U-Mo alloyed structure.

COGEMA, who is interested in the reprocessing of such fuels in La Hague facility, has initiated reprocessing feasibility studies. The chosen process consists in a specific dissolution of UMo fuels in nitric acid media. The obtained solution could then be diluted into standard UO_x type fuels dissolution solution. Uranium and plutonium could be selectively recovered from this feed solution using adapted PUREX process (liquid/liquid extraction using tributyl phosphate). Optimisation of the operating conditions, concerning mainly the dissolution step and the dilution ratio are under studies.

Presentation of the research program to be conducted on the French Atomic Energy Commission Center CEA/VALRHO – Marcoule is the topic of this paper. Starting with tests on UMo powder, it should go on with experiments on non irradiated fuels and end by experiments on genuine irradiated fuels to test the dissolution step and the complete reprocessing scheme.

2. Description of UMo fuels

UMo fuels are composed of a core made of a uranium molybdenum alloy powder, mixed with aluminium before compacting. This core is then cladded between two layers of aluminium. The molybdenum proportion in the UMo alloy powder is between 5 and 10 mass% but the most common value is 7 mass%. For the whole UMo clad fuel, the proportion of the different element is the following :

- Aluminium: 46 mass%,
- Uranium: 50.2 mass%,
- Molybdenum: 3.8 mass%.

To conduct the studies, different type of material will be available:

- UMo powder used in the fabrication of the fuel,
- Non irradiated (UMo + Al) core or pieces of UMo clad fuel,
- Three UMo clad fuels irradiated inside the French reactor OSIRIS.

3. Description of the planned experiments

Main experiments will concern the dissolution step in nitric acid media. The first point to check will be the solubility of molybdenum (expected to be around 1 g/L) in this media in the presence of uranium and aluminium. These tests will be conducted on UMo powder, at several acidities.

Then, dissolution experiments on non irradiated cores and pieces of UMo clad fuels will be carried out to evaluate the dissolution yields and kinetics in different conditions. For that purpose the evolution of the U, Mo, Al and H^+ concentrations in the dissolution solution will be followed. Reference operating dissolution conditions will be chosen at the end of this parametric study. For safety assessment, a specific test according to the defined reference conditions will be conducted with on line detection of hydrogen and nitrogen oxides (NO, NO_2).

At last, dissolution of pieces of irradiated UMo clad will be carried out to check the chosen reference operating conditions on genuine fuels. Evolution of the ^{85}Kr activity in the vapour phase will be measured to follow the dissolution kinetics. Characterisation of the different flux as solution, gas and solids (quantities, analysis of the radioelements…) will also be conducted.

The UMo fuel dissolution solutions must be diluted in standard UOx type fuel dissolution solutions. The corresponding dilution ratio is still to be determined, considering dilution will decrease the aluminium but increase the uranium amount in the solution. First tests will be carried out using non irradiated UMo dissolution solutions and confirmation will be again checked on genuine UMo Fuels. Stability of the dissolution solutions, before or after dilution will also be evaluated.

Once these first two steps (dissolution and dilution) are validated on pieces of irradiated UMo clad fuel, the complete process, as it is planned to be implemented in La Hague facility, will be tested. That means:

- dissolution of an entire UMo clad fuel,
- clarification of the obtained solution and dilution in a UOx dissolution solution,
- extraction of uranium and plutonium in mixer settlers,
- concentration of the fission products solution.

4. Conclusion

A research program concerning reprocessing of UMo spent fuels has been implemented in the French Atomic Energy Commission Center CEA/VALRHO-Marcoule. Experiments to be conducted on UMo powder, non irradiated and irradiated UMo fuels will provide, by end of 2005, optimised conditions for UMo spent fuels reprocessing in La Hague facility.