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Session III

Reactor operation, fuel safety and core conversion

THE CONVERSION PROGRAM

Authorities, Activities and Plans for the Minimization of High Enriched Uranium Through the Global Threat Reduction Initiative

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ABSTRACT

The Office of Global Threat Reduction's (GTRI) Conversion Program develops and implements the technology necessary to enable the conversion of civilian facilities using high enriched uranium (HEU) to low enriched uranium (LEU) fuels and targets. The Conversion program mission supports the minimization and, to the extent possible, elimination of the use of HEU in civil nuclear applications by working to convert research reactors and radioisotope production processes to the use of LEU fuel and targets throughout the world. During the Program's 30 years of existence, 55 research reactors have been converted from HEU to LEU fuels, and processes have been developed for producing the medical isotope Molybdenum-99 with LEU targets. Under GTRI, the Conversion Program has accelerated the schedules and plans for the conversion of additional research reactors operating with HEU. This paper summarizes the current status and plans for conversion of research reactors, in the U.S. and abroad, the supporting fuel development activities, and the development of processes for medical isotope production with LEU targets.

INTRODUCTION

Nuclear research and test reactors have been in operation for over 60 years and have served a variety of uses from pure nuclear science, to nuclear technology development, to roles as research tools in non-nuclear scientific fields including medicine, agriculture, and industry. To date, there are over 270 research reactors currently operating in more than 50 countries worldwide. The expanded use of research reactors began in 1954 under The Atoms for Peace initiative. Initially, the majority of these research reactors were fueled with low-enriched uranium (LEU), however as technology developed reactors began requiring higher specific power and neutron flux, and to avoid costs associated with the development of higher density LEU fuels, these reactors began using high-enriched uranium (HEU) material. This change allowed existing fuel designs to be used.

As worries increased over the potential use of HEU in the manufacture of nuclear weapons, concern grew about the potential of HEU-fueled research reactors becoming a source of the material. In response, the U.S Department of Energy (DOE) initiated a conversion program in 1978 to develop the technology necessary to reduce the use of HEU fuel in research reactors by converting them to LEU fuel. Argonne National

Laboratory (ANL) and Idaho National Laboratory (INL) are the technical lead laboratories for the program.

Beyond the research activities for research reactors described above, a significant purpose of research reactors is the production of medical isotopes, Molybdenum-99 (^{99}Mo) in particular. Although ^{99}Mo can be produced by neutron activation, it is more widely produced by fission of ^{235}U, through the irradiation of HEU targets. In fact, a significant fraction of the HEU that the U.S. exports every year is for the fabrication of targets for the production of ^{99}Mo. In the mid-1980s the Conversion Program was expanded to include, in addition to the conversion of research and test reactors, the development of technology for the production of ^{99}Mo with LEU material.

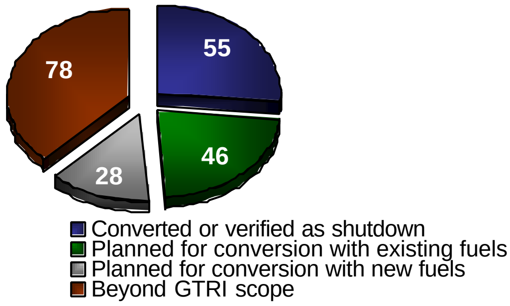
Another expansion of the Conversion Program occurred in the early 1990s, when the Program, which initially focused on reactors supplied with U.S.-origin HEU, began to collaborate with Russian institutes with the objective of converting reactors supplied with Soviet- or Russian-origin HEU to the use of LEU fuel. Since 1995, a fuel development program specifically intended to support the conversion of Russian-supplied reactors, including irradiation and qualification of fuels in Russian test reactors, has been underway.

The ultimate objective of the Office of Global Threat Reduction (GTRI) is not only the conversion of HEU-based reactors and ^{99}Mo production processes to use LEU, but to remove the HEU material from the facilities and provide for its secure disposition. The Conversion Program therefore coordinates its activities with programs which focus on the secure disposition of HEU material, programs like GTRI's Removal program, which coordinates the repatriation of U.S.-origin and Russian-origin fresh and spent research reactor fuel.

CONVERSION STATUS UNDER GTRI

The Conversion Program has identified 207 research and test reactors worldwide that are or were fueled with HEU fuel. The program has compiled a list of 129 of these research reactors with the objective of converting them to LEU fuel. The current list contains U.S.-supplied, Russian-supplied, and Chinese-supplied facilities. The selection of facilities for inclusion in the list is based on the potential for converting the reactor to LEU fuel (availability of LEU fuel, either already qualified or under development) and the existence of a secure disposition path for the removed HEU fuel. The remaining 78 HEU-fueled reactors have been excluded from the Conversion Program scope for a variety of reasons, including (1) classification as defense related facilities, (2) location in countries that currently do not fully collaborate with the United States on reactor conversion programs, or (3) requirements for very specialized LEU fuel which would be too costly and time consuming to develop.

Since the inception of the Conversion Program, 55 of the 129 reactors have been converted to LEU fuel or have shutdown prior to conversion. Under GTRI, DOE has established targets for the conversion of 129 HEU-fueled research reactors. The current goal is to convert the remaining 74 reactors in the list of candidates by the year 2018. Of the 74 remaining research reactors within the scope of the Conversion Program, 46 can be converted with existing LEU fuels, while the remaining 28 require the development of advanced high density fuels to allow their conversion. A new high-density UMo fuel is under development that will allow the conversion of 19 reactors, the remaining 9 reactors may be able to use the UMo fuel as well, but further analysis is needed. The program is focusing much effort on the development of these advanced high-density fuels, particularly UMo fuels, with the goal of qualifying these advanced fuels by 2010.



The Conversion Program also coordinates with other agencies, including the State Department, the Nuclear Regulatory Commission (NRC), and the International Atomic Energy Agency (IAEA). The IAEA has supported the objectives of the Conversion Program through departments concerned with nuclear security and technical cooperation. The role of the NRC is important, as regulator for U.S. university reactors and as the agency that approves the export of HEU material.

Current U.S. law authorizes HEU exports for reactors that have agreed to convert to LEU fuel once a suitable fuel is qualified for their facility. This policy has been instrumental in encouraging the conversion of research reactors with high utilization that require significant annual amounts of fresh HEU fuel. Many reactors, however, have a very slow rate of burn-up and require no new fuel in the immediate future. To encourage the conversion of these reactors, the Conversion program has developed an incentive program that allows the procurement of LEU fuel that would provide a service life equivalent to that of the HEU fuel in the reactor. The number of conversions per year has accelerated significantly since GTRI took over management of the Conversion program. Since the announcement of GTRI the Program accelerated the conversion rate, with a total of sixteen in the last three years.

AUTHORITIES FOR IMPLEMENTATION

From its beginning in 1978, the Reduced Enrichment for Research and Test Reactors program, now the GTRI Conversion Program, has expanded its scope and strengthened its mandate. Today the Program enjoys various levels of support from within the Department of Energy up to the President, including several international agreements. In 1986, the Nuclear Regulatory Commission (NRC) issued a rule on "Limiting the Use of Highly Enriched Uranium in Domestically Licensed Research and Test Reactors. This set the mandate that research reactors must convert to use LEU if it is available and qualified for use in the reactor. It also states that U.S. Government funds would be used to implement the conversion. In 2004, Secretary Abraham committed the U.S. to converting its domestic research reactors to use LEU in a speech to the IAEA, and created the Office of Global Threat Reduction within the NNSA. RERTR became the Reactor Conversion program and a pillar of this office. In 2007, in the third meeting of the Global Initiative to Combat Nuclear Terrorism, the U.S. issues a joint statement with Russia. The Statement calls for, among other things, "minimizing the use of highly enriched uranium…in civilian facilities and activities". Along with these political authorizations, the United States Congress continually authorizes the expansion and increased funding of the Reactor Conversions Program, which now includes 129 domestic and international reactors.

CONCLUSION AND FUTURE DIRECTIONS

In the next few years the Conversion Program is expected to accelerate further, as many reactor conversions will continue to occur. The technical efforts to establish agreements with the reactor operators, and the development and procurement of LEU fuel will increase rapidly to meet the challenges. Meeting this goal will also require increased policy efforts to engage the governments and facilities that have not yet joined the conversion effort as well as technical efforts to develop a conversion approach for reactors that are technically more challenging.

COMMISSIONING OF THE NEW LEU CORE OF THE PORTUGUESE RESEARCH REACTOR

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ABSTRACT

The 1 MW Portuguese Research Reactor (RPI) switched from high-enriched uranium (HEU) to low-enriched uranium (LEU) in September 2007. The core conversion was done under IAEA's Technical Cooperation project POR4016, with financial support from the US and Portugal. The safety analyses for the core conversion were made with the assistance of the RERTR program. This paper presents the measurements done during the start-up program and compares them with an as-built MCNP model. The performance of the new LEU core is compared to that of previous HEU cores.

1. Introduction

The Portuguese Research Reactor (RPI) is a 1 MW, pool-type reactor, built by AMF Atomics and commissioned in 1961. The activities currently underway in the RPI cover a broad range from irradiation of electronic circuits to calibration of detectors for dark matter search, as well as by more classical subjects such as neutron activation analysis. Most of these activities use in-pool irradiations.

The RPI was commissioned in 1961 with LEU fuel. However, it was later converted to HEU fuel for economic reasons. In 1999 Portugal declared its interest to participate in the Foreign Research Reactor Spent Nuclear Fuel Acceptance Program (FRRSNF). A commitment was made to stop using HEU after May 12, 2006 and return all HEU fuel until May 12, 2009. The core conversion to LEU was done within IAEA's Technical Cooperation project POR4016 with financial support of the US and Portuguese governments. An extension on the use of HEU until May 31, 2007 was granted by the Department of Energy, in order to minimize the downtime of the reactor. The actual conversion was done in September 2007. Table 1 summarizes the main milestones of the project.

A feasibility study was performed during 2005 with the assistance of the RERTR program at Argonne National Laboratory. Uranium silicide (U_3Si_2-Al) dispersion fuel with a density of 4.8 g/cm^3 was selected because of its widespread use in research reactors and for the relatively large number of manufacturers. The feasibility study also had the goal of minimizing the number of assemblies required for operation during the current FRRSNF acceptance window. The new LEU standard assembly has ^{235}U loading of 376 g vs. 265 g for an HEU standard

* Work supported by the U.S. Department of Energy, National Nuclear Security Administration, under Contract No. DE-AC02-06CH11357.

assembly. With this design the core size remained unchanged, at 12 assemblies, and only 14 assemblies are required for operation until May 2016 [1]. The number of plates (18 for standard and 10 for control assemblies) was kept the same as for the HEU fuel.

Milestone	Planned	Effective
Commitments for funding	Mid 2005	As planned
Feasibility study	End of 2005	As planned
Safety studies	Mid 2006	End of 2006
Project and Supply Agreement	Mid 2006	Early 2007
Fuel manufactured	End of 2006	As planned
Regulatory Approval	End of 2006	August 2007
Conversion	Early 2007	September 2007

Tab. 1: Milestones for the conversion project

The results of neutronic studies, steady-state thermal-hydraulic analyses and accident analyses demonstrated that the RPI could be operated safely with the new LEU fuel [2]. The submission of the safety documentation for approval suffered a 6 month delay from planned. The IAEA initiated the review of the documents shortly after their reception. Revised documents were submitted in June 2007 addressing the issues raised during review. The IAEA provided a letter of support for the conversion in late June and the licensing body of the RPI approved the conversion in August 2007.

The most challenging aspect of this project was the conclusion of the required tripartite agreement between the IAEA and the US and Portuguese Governments, which involved several interactions with the two governments, the IAEA and the European Commission.

2. Conversion

Fig. 1 shows the initial LEU core configuration. LS1 through LS7 are standard assemblies and LC1 through LC5 are control assemblies, NS is a Sb-Be neutron source, FC a fission chamber and the DA are hollow dummy assemblies. The hollow dummy assemblies were introduced in the LEU core in order to improve the thermal hydraulic safety margins [2].

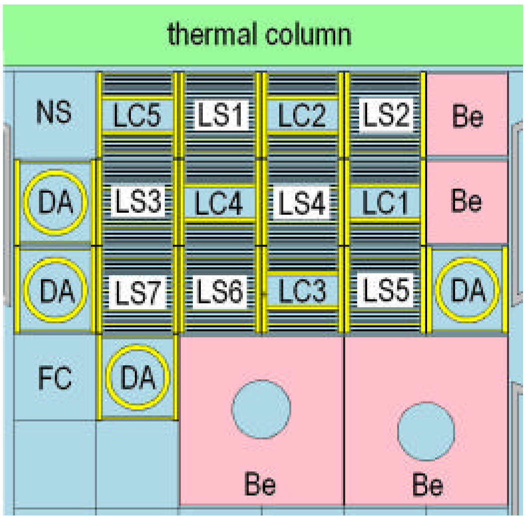

Fig. 1. Initial LEU core configuration, adapted from MCNP model of core.

The shim-safety rods B1 to B4 are mounted in assemblies LC1 to LC4; the regulating rod, BR, in LC5. The regulating rod was calibrated using the positive period method. The shim-safety rods were calibrated in pairs B1/B2 and B3/B4 by comparison with a known displacement of the regulating rod. At the end of these calibrations, the safety parameters of Table 2 were determined, where B1 through B4 represent the shim-safety rod worth. The quoted uncertainties of 3% derive directly from the uncertainty in the calibration of the regulating rod and its propagation to the other parameters through the calibration process.

Parameter (%?k/k)		Description	Required in OLC	Measured
1	Core Excess Reactivity	E	< 4.80	4.11 ± 0.12
2	Total Shutdown Subcriticality	E – (B1+B2+B3+B4+BR)	< -3.00	-9.09 ± 0.27
3	Min. Shutdown Subcriticality	E – (B1+B2+B3)	< -1.00	-4.73 ± 0.14
4	Regulating Rod Worth	BR	< 0.60	0.33 ± 0.01

Tab. 2: Compliance with Safety Parameters

All safety parameters obtained from the rod calibrations satisfy the requirements of the OLC.

3. Neutron fluxes

Thermal, epithermal and fast neutron fluxes were measured in 13 grid positions, including the 4 hollow dummy assemblies in positions 62, 63, 13 and 54, as shown in Fig. 2.

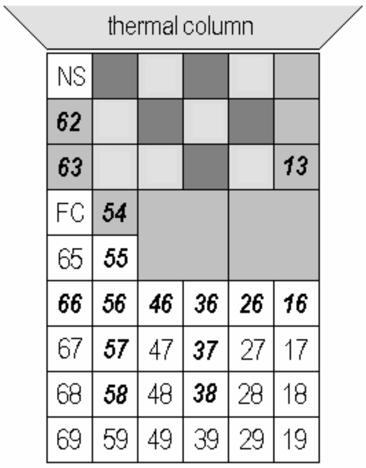

Fig. 2. Plot of core grid showing highlighted in bold and italic the positions where neutron fluxes were measured.

The RPI does not have a regular fuel cycle, with a standard core configuration. Configurations with up to 15 HEU assemblies were previously used; configurations up to 13 LEU assemblies are now foreseen. For the purposes of flux comparisons, the best match with the current LEU core is the first HEU core [3], implemented in February 1990; it is not a perfect match, since the HEU core had one Be reflector in position 13 and the fission chamber in position 54.

Table 3 compares the measured thermal fluxes at core mid-height. Measurements were done at 1 MW and 100 kW. The average ratio between the thermal fluxes measured in the HEU and LEU cores is 0.9 $\pm$ 0.3, covering two orders of magnitude of the values. We are conservatively assuming an uncertainty of 10% and 20% for the measured LEU and HEU flux values, respectively. From the available data there is no clear loss or gain of thermal neutron flux with the conversion to LEU. Furthermore, the LEU core has 2 additional irradiation positions, inside the hollow dummy assemblies in positions 13 and 54, which have thermal neutron fluxes of 1.9×10^{13} and 1.8×10^{13} n/cm^2/s, respectively.

Grid position	**LEU thermal flux (n/cm^2/s) ± 10%**	**HEU thermal flux (n/cm^2/s) ± 20%**	**Ratio HEU/LEU (± 22%)**
55	7.7E12	5.4E12	0.7
56	1.7E12	1.2E12	0.7
46	2.8E12	2.6E12	0.9
36	3.9E12	3.2E12	0.8
26	2.8E12	3.0E12	1.1
57	2.8E11	2.4E11	0.9
37	5.0E11	4.5E11	0.9
38	5.0E10	5.6E10	1.1

Tab. 3: Comparison between thermal neutron fluxes for HEU and LEU comparable cores.

Gamma dose rates were also measured in all free grid positions, at mid-height of the core, using a Radiotechnique Compelec CRGA11 ionization chamber. The measurements were done at a power of 100 kW and extrapolated to 1 MW using the ^{16}N linear channel. The ratio of HEU to LEU values is 1.1 ± 0.2 covering one order of magnitude of the values.

4. Updated MCNP model

The MCNP core model used in the feasibility and safety studies [1,2] was updated using the extensive data provided by the fuel manufacturer CERCA. Measured values for the uranium isotopes, impurities in fuel meat and cladding were introduced, as well as measured values for the plate and clad thickness.

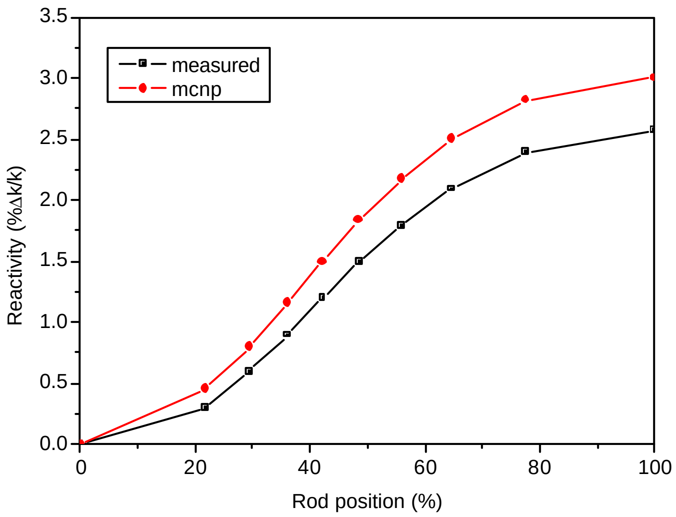

Fig. 3. Integral rod worth curve of shim-safety rod 1: measured vs. MCNP calculated values. The lines were drawn to guide the eye.

Since there is considerable shadowing between the shim-safety rods in this compact core, the integral worth of the rods was calculated by simulating the actual rod positions that were used in the measurement. The same procedure was applied before for the HEU cores with excellent results [1]. Only preliminary results are shown here. A comparison of calculated and measured values in determining the worth of shim-safety rod B1 is plotted in Fig. 3. The integral worth was measured to be 2.6 ± 0.1 %?k/k and calculated to be 3.0% ?k/k.

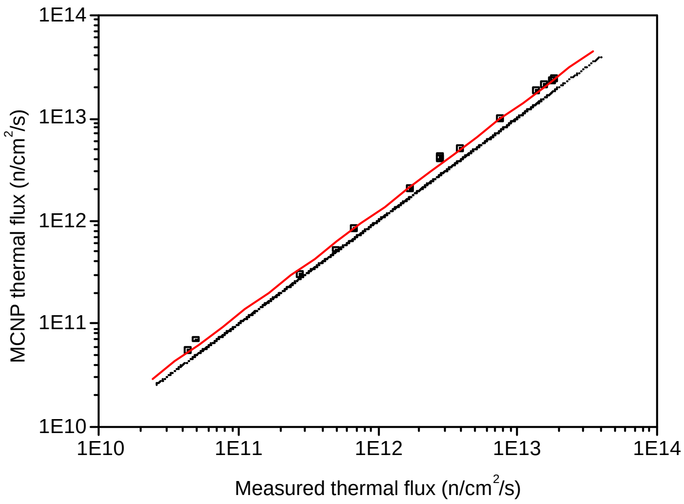

Fig. 4. Thermal neutron fluxes: measured vs. MCNP values. The top line is a least-squares linear fit; the bottom line shows a 1:1 ratio.

Figure 4 shows preliminary results of the calculated thermal neutron fluxes vs. measured values. Calculated values are along a straight line with a small offset to the 1:1 relationship over nearly 3 orders of magnitude.

Conclusions

The RPI switched from HEU to LEU in September 2007 within IAEA project POR4016, with financial support from the US and Portugal. For in-pool irradiations, the new LEU core has the same performance as a comparable HEU core. The core change also allowed the introduction of two high-flux positions which did not exist before, increasing the pool irradiation capabilities. Work in progress includes the measurement of neutron fluxes and gamma dose rates in the beam tubes and improvements in the as-built MCNP model of the core.

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INL/EXT-07-12604

University Reactor Conversion Lessons Learned Workshop for Texas A&M University Nuclear Science Center

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ABSTRACT

The Department of Energy's Idaho National Laboratory, under its programmatic responsibility for managing the University Research Reactor Conversions, has completed the conversion of the reactor at the Texas A&M University Nuclear Science Center Reactor. With this work completed and in anticipation of other impending conversion projects, INL convened and engaged the project participants in a structured discussion to capture the lessons learned. This lessons learned process has allowed us to capture gaps, opportunities, and good practices, drawing from the project team's experiences. These lessons will be used to raise the standard of excellence, effectiveness, and efficiency in all future conversion projects.

CONTENTS

ACRONYMS

ANL	Argonne National Laboratory
DOE	U.S. Department of Energy
GA	General Atomics
HEU	highly enriched uranium
INL	Idaho National Laboratory
LEU	low-enriched uranium
NNSA	National Nuclear Security Administration
NRC	Nuclear Regulatory Commission
NSC	Nuclear Science Center
SNF	spent nuclear fuel
TAMU	Texas A&M University

University Reactor Conversion Lessons Learned Workshop for Texas A&M Nuclear Science Center

1. INTRODUCTION

The Department of Energy's (DOE) Idaho National Laboratory (INL), under its programmatic responsibility for managing the University Research Reactor Conversions, has completed the conversion of the reactor at the Texas A&M University Nuclear Science Center (TAMU NSC). This project was successfully completed through an integrated and collaborative effort involving INL, Argonne National Laboratory (ANL), DOE (headquarters and the field office), the Nuclear Regulatory Commission (NRC), the universities, and the contractors involved in analyses, fuel design and fabrication, and spent nuclear fuel (SNF) shipping and disposition. With this work completed and in anticipation of other impending conversion projects, INL convened and engaged the project participants in a structured discussion to capture the lessons learned. The objectives of this meeting were to capture the observations, insights, issues, concerns, and ideas of those involved in the reactor conversions so that future efforts can be conducted with greater effectiveness, efficiency, and with fewer challenges.

2. BACKGROUND

As part of the Bush administration's effort to reduce the amount of weapons-grade nuclear material worldwide, the National Nuclear Security Administration (NNSA) has established a program to convert research reactors from using highly enriched uranium (HEU) to low-enriched uranium (LEU) fuel.

The research reactor conversion effort is a critical step under the Global Threat Reduction Initiative's Reduced Enrichment for Research and Test Reactors program. As part of this program, NNSA is minimizing the use of HEU in civilian nuclear programs by converting research reactors and radioisotope production processes to the use of LEU fuel and targets. The HEU is weapons-grade nuclear material that can be used to make a nuclear weapon or dirty bomb. The research reactors are secure and are used for peaceful purposes; however, by converting these reactors to use LEU, a significant step is made toward ensuring that weapons-usable nuclear material is secure and safeguarded.

Among the list of research reactors targeted for conversion in 2006 were the University of Florida and Texas A&M University.

Reactor conversions include analyses, LEU fuel fabrication, reactor defuel and refuel activities, HEU packaging and transportation, and reactor startup.

3. LESSONS LEARNED PROCESS

The process for capturing the lessons learned from this project involved taking the schedule of the project activities and focusing feedback and discussion on each respective activity. The feedback and lessons learned discussions were held in an open discussion workshop, including all participating team members and their representatives. To promote a more expedient discussion at the workshops and to help the project team focus on the higher priority areas, a survey was developed and sent to project participants before the workshops. The survey invited those involved in the project to score and offer comments with regard to the projects activities in which they were involved. The survey was formatted with a 5-point Likert scale, where 1 was low or "extremely challenging," and 5 was high or "exceptional." The surveys were collected and scores were entered and averaged for each activity. The average score for each activity is identified in Section 6 of this document.

Based on survey scores and comments, the workshop agenda was established and timeframes were estimated. Consistent with expectations based on the survey results, the workshop discussions were brief for the unremarkable areas and more extended and detailed in those areas of greatest significance. The detailed lessons learned were captured and the themes and general conclusions were then drawn. The general conclusions and themes tend to apply to all activities (almost as operating principles) and will benefit future project teams and project managers. The more detailed lessons learned align to given activities and apply to the project manager and those involved in the given activity, as that activity is undertaken.

4. LESSONS LEARNED

4.1 General Conclusions

This project was clearly a success. Nonetheless, there were many detailed lessons learned regarding both technical and project management aspects. The specifics are provided in the following sections; however, some general elements are key to the success of future conversion and spent fuel shipping projects. Future projects will be conducted most effectively, efficiently, and with a minimum of risks, interference, and interruptions if the following are an integral part of the project:

- **Project team composition**, which includes a project team composed of individuals who are critical thinkers, flexible, and committed to the project results (the following was extracted from the comments submitted: "Having the right people who were willing to buy into the common vision and mission was critical. Everyone had a great personal work ethic. Having a single person who is solely dedicated to the project [allowing that person to stay in contact with all parties involved and to identify and track issues] was instrumental in the success of the project.").

- **Communication**, including inclusive communications and exchange that provides for effective sharing of needs, expectations, roles, responsibilities, data, assumptions, schedules, and facility and equipment constraints.

- **Use of expertise**, including confidence in and effective utilization of the varied expertise and experience of the team members.

- **Proactivity** and individual levels of initiative.

- **Early initiation** includes the earliest possible initiation of planning and activities at every step in the project process, thereby minimizing the likelihood of time-critical situations.

- **Verification and re-verification** of data, analyses, specs, assumptions, performance expectations, and equipment fit and function throughout the project.

- **Clear and common understanding**, including clear expectations of roles, responsibilities, technical variables, and technical results.

- **Knowledgeable and informed stakeholders** who can advocate for the project, remove barriers, and support decisions and adjustments needed to ensure project success (e.g., public, political, and administrative).

- **Compile reactor data** includes assembly or compilation of the historical documents that reveal what is known and unknown about the reactor.

- **Value-added government oversight**, in which the public interests are served, objectivity is retained, but NRC's experience and expertise is available to the project.

The above list comprised the general themes of the lessons learned meeting. The detailed lessons learned were discussed in the order of project activities, from initiation to closeout, and are provided in the following sections.

4.2 Lessons Learned Meeting Summary

The Lessons Learned Workshop for the Texas A&M University Nuclear Science Center convened on February 21, 2007, at the General Atomics (GA) facilities in San Diego, California. The following were attendees at the workshop:

Dana Meyer, INL	John Bolin, GA
Eric Woolstenhulme, INL	Jason Yi, GA
Doug Morrell, INL	Ken Mushinski, GA
Dale Luke, INL	Pierre Colomb, CERCA
Jim Wade, DOE-ID	Helios Nadal, CERCA
Parrish Staples, DOE-NNSA	Jim Matos, ANL
Scott Declue, DOE-SRS	Jim Remlinger, TAMU
Alexander Adams, NRC	W Dan Reece, TAMU
Bill Schuser, NRC	Jamie Adam, NAC
Anthony Veca, GA	

The following was the agenda for the workshop:

8:00 Welcome and introductory remarks, establish ground rules, and review agenda

8:30 Presentations

- TAMU NSC TRIGA Reactor Performance Analysis—TAMU NSC
- TRIGA Fabrication Process—TRIGA International

9:00 Discuss and collect lessons learned by each major activity area

- Initiating Conversion Project
- Conversion Proposal Process

10:15 Break

10:30 Discuss and collect lessons learned by each major activity area (continued)

- Fuel and Hardware Development and Procurement

12:00 Lunch

1:00 Discuss and collect lessons learned by each major activity area (continued)

- Core Conversion
- SNF Shipment

2:20 Break

2:35 Discuss and collect lessons learned by each major activity area (continued)

- Other areas needing to be addressed

3:35 Next steps and assignments

4:10 Closing remarks

4:30 Adjourn

5. PRESENTATIONS

5.1 Texas A&M University Nuclear Science Center TRIGA Reactor Performance Analysis

Dr. Dan Reece summarized the TAMU NSC reactor conversion in his presentation. Dr. Reece concluded that many things went very well, but there were a few problems. Dr. Reece also gave his perspective on the lessons to be learned from the conversion work. Highlights from Dr. Reece's presentation include the following:

- The difference between calculated values for fuel element temperatures and the actual measured values of the new core
- The apparent conflict between calculated values for neutron fluxes and the fluxes derived from foil experiments in the new core
- The importance of interactions and relationships with the various regulators and conversion team members

- The importance of planning and coordination for the project
- The difficulty of locating specific details about the old core.

5.2 TRIGA Fabrication Process

This joint presentation covered the ongoing research concerning the difference between the calculated values for fuel element temperatures and the actual measured values of the new NSC core. Additionally, it was shown that the NSC fuel elements fabricated by CERCA were produced in compliance with GA technical specifications and CERCA's quality assurance requirements. The fuel elements were delivered on time and in accordance with the initial manufacturing schedule.

The process for assembling TRIGA elements was discussed. The point was made that inserting the meats into the cladding is a difficult process because of tight cladding tolerances. About 60% of the fuel elements must have the fuel meats pressed into the cladding. Only meats and cladding with a large gap actually just slide in.

For the instrumented fuel elements, the meat diameters were within tolerance, but at the small end of the ID tolerance. The cladding ID was larger than is allowed per the drawings, but it was determined that it was within the safety analysis report specifications and was cleared for use. This configuration translated to a larger than nominal gap between the meat and the cladding. This gap reduces heat transfer from the meat to the cladding and causes the fuel temperature to be higher than optimal. As the meat swells from operating the reactor, the gap will decrease and the temperature will be lower.

The ostensible decrease in neutron flux was also discussed. The matter needs further investigation and foil testing and the results will be documented in a report by GA.

6. LESSONS LEARNED

The detailed lessons learned were discussed in order of project activities, from initiation to closeout, and are provided in the following sections.

6.1 Initiating Conversion Project

6.1.1 Initiation

The average survey score was 3.88.

Issues	Recommendations
Some reactor specifications were difficult to ascertain and came late in the project. Some of this was because the contract with GA was finalized later than optimum.	Early involvement of GA is imperative to better understand the core and project implications (e.g., fuel and hardware). Also, GA should be invited to the reactor early in the process, with procurement and analysis aspects being a key focus.

Issues	Recommendations
The initial license amendment followed an old example rather than following the NRC guidance document, NUREG-1537. This resulted in some unnecessary rewriting.	Follow NUREG-1537 rather than relying on previous amendments. Reviewing past requests for additional information from NRC may also be of benefit.

6.2 Conversion Proposal Process

6.2.1 Proposal Preparation

The average survey score was 2.83.

Issues	Recommendations
An interactive request for additional information resolution meeting with all parties involved was a key activity. This was much more effective than trading phone calls and emails. The face-to-face and open, direct communication was key. This reduced the required time to complete the process by a factor of 10.	Teamwork is critical to success and efficiency of the proposal process.

6.2.2 Contract Negotiation

The average survey score was 3.0.

Issues	Recommendations
The procurement process on both sides (i.e., government and university) is problematic. Lack of a mutual understanding in the procurement process lends to bogging down the process.	Promote communications and negotiations between the principle project parties before going to the procurement agents. Once the terms are understood, then the procurement people can be brought in to complete the process. Involve both procurement agents early on to ensure that time is not lost negotiating differences between processes and waiting for additional information later. Early initiation involvement and coordination of contracts/procurement staff are crucial.

6.3 Fuel and Hardware Development and Procurement

6.3.1 Fuel Specifications and Drawings

The average survey score was 2.20.

Issues	Recommendations
Specifics about the fuel and hardware procurement were confusing because of the varied opinions and individual spreadsheets.	It would be helpful to get everyone together at the onset and create a format for presenting the fuel and hardware information that everyone agrees to and understands. Drawings and other historical documents could be presented at the initial meeting. The various parties could discuss the data to ensure mutual agreement on what needs to be ordered. One person could be charged with keeping the fuel and hardware spreadsheet updated and issued to the interested parties.
Specifics about the fuel and hardware procurement were confusing because no cluster assembly information was provided to the university.	See above recommendation. Also, GA could provide information about which upper and lower adapters (and other hardware) are required for the various cluster types.
The gram loading for the fuel elements was on the low end of the required range.	The project should advise TRIGA International to load the elements on the heavy side to maximize the amount of fuel in the core. This maximizes the per element value when considering the dollars spent on fabrication, shipping, usage, and disposal of a fuel rod.
Having the fabrication data for the new fuel earlier in the process would be helpful.	This effort must be worked with the university to ensure that all needed information is provided in the data packages. As a minimum, the data packages should be included with the fuel shipment. Caution must be taken to properly handle proprietary information.

6.3.2 Fuel Inspection

The average survey score was 4.00.

Issues	Recommendations
The fuel receipt inspection worked well at the reactor and at CERCA.	The right people were involved in the inspection (i.e., vendor, quality assurance personnel, and receivers). A coordination meeting was held before the inspection so that everyone involved was well advised and clearly understood their rolls. A source inspection was conducted at the manufacturer site in France before shipment so that the receipt inspection at the university was less complex and time intensive.
After inspection, it was unclear who took ownership of the fuel.	There needs to be a clear transfer of responsibility so that it is understood who owns the fuel at any given time. A signature process could be devised that formally documents and completes the ownership transfer.

6.3.3 Preparation of Facility for Fuel Receipt

The average survey score was 3.60.

Issues	Recommendations
The truck/trailers arrived at NSC with the containers positioned toward the front of the trailers and with some of the containers turned sideways; this precluded access with a pallet jack or forklift.	Information about the shipping trucks and loading configuration is important to expedite the receipt of the fuel at the reactor. Ii would be best if the trailers had a side-loading capability to make it easier to unload the shipments with a forklift. The INL should facilitate communications between the shipper and reactor. The INL should consider writing truck specifications into the contract with the shipping company.

6.3.4 Reassembly

The average survey score was 3.33.

Issues	Recommendations
It may take specific training to open and reassemble the shipping containers for return shipment.	Dave Capp at the INL was this person for the TAMU NSC project. He did a great job. The INL needs to secure a similar individual on all future projects.

6.4 Core Conversion

6.4.1 Fuel Removal

The average survey score was 3.33.

Issues	Recommendations
Fuel removal went well at NSC.	Video taping of the processes will serve as a great resource for those who must perform the tasks later. It may be beneficial to have the core parameters measured and documented before the reactor is shutdown for refueling (i.e., fuel temperatures, neutron flux, and control rod positions). The measurements may be useful in analysis following restart.

6.4.2 Refueling

The average survey score was 3.50.

Issues	Recommendations
Personnel turnover at the universities can sometimes cause a loss of drawings, specifications, and other documents. This can make converting the reactor and SNF shipments a significant challenge.	Early notification of the documentation needs by the INL should be made to the university. This will allow more time for locating the information.

Issues	Recommendations
Hardware for NSC had to be re-machined because of lack of information. GA was quick to respond to all issues identified; therefore, the issues were resolved quickly.	An early start can also allow time for reactor personnel to physically verify reactor components before procurement of the parts. Because of this issue, we must pay greater attention to the details of the reactors.
The instrumented fuel elements read higher than expected from the earlier analysis.	Instrumented fuel elements cladding and fuel meat gaps must be tighter to ensure that the actual readings are more representative of the core analysis.
Thermocouple leads on the instrumented fuel elements were too long for the NSC configuration. The NSC cut the leads, but then required a half day to re-work the lead wires.	The correct length should be identified before fabrication at CERCA. Cutting the thermocouple leads is standard practice, but had it been considered ahead of time, the materials and capabilities could have been in place onsite to significantly reduce the time and effort required.

6.5 Spent Nuclear Fuel Shipment

6.5.1 Cask Determination

The average survey score was 3.67.

Issues	Recommendations
The SNF shipment activities are very difficult for universities that do not normally ship SNF.	Updated guidance from NRC regarding SNF shipping would be helpful. The INL should consider contracting with other companies or experienced shippers to help the licensees. The DOE could consider taking ownership of the shipping rather than NRC. It is important to field-verify all procedures, plans, and such before shipping.
Not everyone with a need to know had copies of the SNF shipping orders, specifically, some information needed to be included in shipping documents prepared by others. This was caused, in part, by a Safeguards Information “blackout” for information from NRC.	Safeguarded Information issues have been resolved at NRC. This situation should not occur in the future.

Issues	Recommendations
The cask was identified much later than appropriate by INL. The tardiness of the contract with the cask vendor caused delays in the facility preparations. This caused unnecessary stress and work for NSC.	The INL needs to make cask arrangements as soon as possible. The cask vendors need to make detailed site assessments early in the project. Drawings and procedures need to be supplied to the reactor as soon as possible. The project should make early visits to the university and discuss the tasks associated with SNF shipping.

6.5.2 Transportation Plan/Security Plan

The average survey score was 3.0.

Issues	Recommendations
Transport and security plans can be time-consuming and labor intensive.	The project should get the most effective and reliable sources to carry out the functions of developing the plans.
Guidance form NRC regarding HEU shipments was not as clear or up-to-date as it could have been.	The current guidance should be updated. The NRC suggests we work with one of the current licensees to get better understanding of the current regulations.

6.5.3 Route Assessment

The average survey score was 3.2.

Issues	Recommendations
Communication about the route assessment documents was sometimes inefficient.	It was suggested to involve other subject matter experts during the route assessment. Communication lines between all parties (i.e., shipper, INL, cask vendor, and other facilitating companies) need to be open.

6.5.4 Certification of University Quality Assurance Programs

The average survey score was 3.0.

Issues	Recommendations
Certifying as an SNF shipper can be extensive.	Begin activities early and the program should provide assistance to the facility, as needed.

6.5.5 Facility Preparations for Spent Nuclear Fuel Activities

The average survey score was 3.60.

Issues	Recommendations
The SNF shipping preparations are wide-ranging and often difficult.	Need to ensure early, comprehensive planning with attention to detail. Start the process to procure support equipment (e.g., cranes) early. This worked well for us.

6.5.6 Required Shipping Data Preparation

The average survey score was 2.5.

Issues	Recommendations
Required shipping data preparations can be laborious and resource intensive.	Use of the parametric study on TRIGA fuel burnups for completing the required shipping data radioisotope and decay heat tables would be very effective. The university may need to check and validate the applicability of the standard decay heat data.

6.5.7 Shipping Documentation

The average survey score was 3.0.

Issues	Recommendations
Shipping documentation, such as SNF Transportation Plans and the Bill of Lading, were very involved for an unfamiliar shipper.	The INL's help was invaluable. The university always felt that they had an ally and knowledgeable resource to facilitate the process. The project university also had confidence in the experts and could trust their advice and experience during document development.

6.5.8 Cask Loading

The average survey score was 3.67.

Issues	Recommendations
The SNF roles and responsibilities were well defined going into the SNF shipping activities.	The NSC had been informed early in the project that they were in charge and responsible for the activities. All other entities also understood this at the outset of the project. This hierarchy resulted in effective working relationships between the project entities. We need to maintain this level of rigor and discipline for future conversion projects.

Issues	Recommendations
The cask sat loaded at NSC over the weekend. This was an unfavorable situation for the shipper.	Many notifications and logistics have to be worked out for the moment the shipment leaves the facility. Changes to planned shipping dates are difficult if not impossible to effect. The SNF loading was to begin on Monday. It was estimated that loading would take about 5 days to complete, thereby finishing on Friday. Weekends are not the preferred times to start shipments; therefore, the INL shipping coordinator felt that it was best to leave the weekend for schedule contingency in the case loading took longer than expected. The project needs to fully communicate this thinking and the firm shipping dates for the university. In future shipments, the project needs to consider the trade-off between shipping on a weekend or leaving the loaded cask at the facility for the weekend.

6.5.9 Receipt Facility Preparation

The average survey score was 3.33.

Issues	Recommendations
There was some confusion on who was making arrangements for the return shipments of the Nuclear Assurance Corporation equipment. Just days before the shipment, it was found that the arrangement for a truck had not been made.	It needs to be clearly established, well in advance of the cask loading dates, who is responsible for planning and executing the tasks for all legs of the shipments. This includes equipment shipment to and from the various facilities.

6.6 Other issues

6.6.1 Safeguards Information

The average survey score was 3.0.

Issues	Recommendations
There was a bit of confusion regarding what constitutes safeguards information and who can have access to it.	The various entities involved with the project need to clearly understand their responsibilities and limitation under this order. The project should consider holding an onsite meeting to clarify the policies with the project team.

7. ROUND ROBIN

In concluding the discussion of the lessons learned, all participants were invited to reiterate, summarize, or offer any other lessons learned. The following list provides their final thoughts:

- Well defined goals and responsibilities are essential to success. All team members must understand their responsibilities. Because of division of responsibilities at INL, it was confusing to NSC who at INL was in charge of some tasks.
- It is important for the project team to understand that if a task can be done early then it should be. Performing tasks just-in-time would have caused the NSC conversion to fail because of unexpected, last-minute tasks and issues. In other words, completing tasks early will allow the project to be flexible enough to address the last minute challenges.
- The NSC project went well in spite of the minor setbacks and challenges. The project will be held to a higher standard of performance next time.
- There will be some weeks/months after the project where parties will need to work together to get some things accomplished and review present issues of conversion.
- The next lessons learned analysis needs to include a specific "what went well" column so that we can capture the things that worked.

CONCLUSION

This lessons learned process has allowed us to capture gaps, opportunities, and good practices, drawing from the project team's experiences. The process is inclusive and offers an opportunity for every entity that "touched" the project to share from its experience. These lessons will be used to raise the standard of excellence, effectiveness, and efficiency in all future conversion projects. Despite making improvements to successive projects by addressing the lessons we have learned on this project, conducting a lessons learned activity will be vital to each conversion project as technologies, regulations, and other aspects of the environment change and influence success. It is recognized we cannot become complacent, nor adopt a mindset that the process has been "perfected."

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University Reactor Conversion Lessons Learned Workshop for the University of Florida

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ABSTRACT

The Department of Energy's Idaho National Laboratory, under its programmatic responsibility for managing the University Research Reactor Conversions, has completed the conversion of the reactor at the University of Florida. With this work completed and in anticipation of other impending conversion projects, INL convened and engaged the project participants in a structured discussion to capture the lessons learned. This lessons learned process has allowed us to capture gaps, opportunities, and good practices, drawing from the project team's experiences. These lessons will be used to raise the standard of excellence, effectiveness, and efficiency in all future conversion projects.

CONTENTS

ACRONYMS

ANL	Argonne National Laboratory
DOE	U.S. Department of Energy
GA	General Atomics
HEU	highly enriched uranium
INL	Idaho National Laboratory
LEU	low-enriched uranium
NNSA	National Nuclear Security Administration
NRC	Nuclear Regulatory Commission
SNF	spent nuclear fuel

University Reactor Conversion Lessons Learned Workshop for the University of Florida

1. INTRODUCTION

The Department of Energy's (DOE) Idaho National Laboratory (INL), under its programmatic responsibility for managing the University Research Reactor Conversions, has completed the conversion of the reactor at the University of Florida. This project was successfully completed through an integrated and collaborative effort involving INL, Argonne National Laboratory (ANL), DOE (headquarters and the field office), the Nuclear Regulatory Commission (NRC), the universities, and the contractors involved in analyses, fuel design and fabrication, and spent nuclear fuel (SNF) shipping and disposition. With this work completed and in anticipation of other impending conversion projects, INL convened and engaged the project participants in a structured discussion to capture the lessons learned. The objectives of this meeting were to capture the observations, insights, issues, concerns, and ideas of those involved in the reactor conversions so that future efforts can be conducted with greater effectiveness, efficiency, and with fewer challenges.

2. BACKGROUND

As part of the Bush administration's effort to reduce the amount of weapons-grade nuclear material worldwide, the National Nuclear Security Administration (NNSA) has established a program to convert research reactors from using highly enriched uranium (HEU) to low-enriched uranium (LEU) fuel.

The research reactor conversion effort is a critical step under the Global Threat Reduction Initiative's Reduced Enrichment for Research and Test Reactors program. As part of this program, NNSA is minimizing the use of HEU in civilian nuclear programs by converting research reactors and radioisotope production processes to the use of LEU fuel and targets. The HEU is weapons-grade nuclear material that can be used to make a nuclear weapon or dirty bomb. The research reactors are secure and are used for peaceful purposes; however, by converting these reactors to use LEU, a significant step is made toward ensuring that weapons-usable nuclear material is secure and safeguarded.

Among the list of research reactors targeted for conversion in 2006 were the University of Florida and Texas A&M University.

Reactor conversions include analyses, LEU fuel fabrication, reactor defuel and refuel activities, HEU packaging and transportation, and reactor startup.

3. LESSONS LEARNED PROCESS

The process for capturing the lessons learned from this project involved taking the schedule of the project activities and focusing feedback and discussion on each respective activity. The feedback and lessons learned discussions were held in an open discussion workshop, including all participating team members and their representatives. To promote a more expedient discussion at the workshops and to help the project team focus on the higher priority areas, a survey was developed and sent to project participants before the workshops. The survey invited those involved in the project to score and offer comments with regard to the projects activities in which they were involved. The survey was formatted with a 5-point Likert scale, where 1 was low or "extremely challenging," and 5 was high or "exceptional." The surveys were collected and scores were entered and averaged for each activity. The average score for each activity is identified in Section 5 of this document.

Based on survey scores and comments, the workshop agenda was established and timeframes were estimated. Consistent with expectations based on the survey results, the workshop discussions were brief for the unremarkable areas and more extended and detailed in those areas of greatest significance. The detailed lessons learned were captured and the themes and general conclusions were then drawn. The general conclusions and themes tend to apply to all activities (almost as operating principles) and will benefit future project teams and project managers. The more detailed lessons learned align to given activities and apply to the project manager and those involved in the given activity, as that activity is undertaken.

4. LESSONS LEARNED

4.1 General Conclusions

This project was clearly a success. Nonetheless, there were many detailed lessons learned regarding both technical and project management aspects. The specifics are provided in the following sections; however, some general elements are key to the success of future conversion and spent fuel shipping projects. Future projects will be conducted most effectively, efficiently, and with a minimum of risks, interference, and interruptions if the following are an integral part of the project:

- **Project team composition**, which includes a project team composed of individuals who are critical thinkers, flexible, and committed to the project results (the following was extracted from the comments submitted: "Having the right people who were willing to buy into the common vision and mission was critical. Everyone had a great personal work ethic. Having a single person who is solely dedicated to the project [allowing that person to stay in contact with all parties involved and to identify and track issues] was instrumental in the success of the project.").

- **Communication**, including inclusive communications and exchange that provides for effective sharing of needs, expectations, roles, responsibilities, data, assumptions, schedules, and facility and equipment constraints.

- **Use of expertise**, including confidence in and effective utilization of the varied expertise and experience of the team members.

- **Proactivity** and individual levels of initiative.

- **Early initiation** includes the earliest possible initiation of planning and activities at every step in the project process, thereby minimizing the likelihood of time-critical situations.

- **Verification and re-verification** of data, analyses, specs, assumptions, performance expectations, and equipment fit and function throughout the project.

- **Clear and common understanding**, including clear expectations of roles, responsibilities, technical variables, and technical results.

- **Knowledgeable and informed stakeholders** who can advocate for the project, remove barriers, and support decisions and adjustments needed to ensure project success (e.g., public, political, and administrative).

- **Compile reactor data** includes assembly or compilation of the historical documents that reveal what is known and unknown about the reactor.

- **Value-added government oversight**, in which the public interests are served, objectivity is retained, but NRC's experience and expertise is available to the project.

The above list comprised the general themes of the lessons learned meeting. The detailed lessons learned were discussed in the order of project activities, from initiation to closeout, and are provided in the following sections.

4.2 Lessons Learned Meeting Summary

The Lessons Learned Workshop for the University of Florida convened on February 22, 2007, at the General Atomics (GA) facilities in San Diego, California. The following were attendees at the workshop:

Dana Meyer, INL	Anthony Veca, GA
Eric Woolstenhulme, INL	Jason Yi, GA
Doug Morrell, INL	Ken Mushinski, GA
Dale Luke, INL	Jim Matos, ANL
Jim Wade, DOE-ID	Ali Haghighat, UF
Parrish Staples, DOE-NNSA	Benoit Dionne, UF
Scott Declue, DOE-SRS	Roy Boyd, STS
Alexander Adams, NRC	Chip Shaffer, BWXT
Bill Schuser, NRC	

The following was the agenda for the workshop:

8:00 Welcome and introductory remarks

- Establish ground rules and review agenda

8:30 Discuss and collect lessons learned by each major activity area

- Initiating Conversion Project
- Conversion Proposal Process

10:15 Break

10:30 Discuss and collect lessons learned by each major activity area (continued)

– Fuel and Hardware Development and Procurement

12:00 Lunch

1:00 Discuss and collect lessons learned by each major activity area (continued)

– Core Conversion

– SNF Shipment

2:20 Break

2:35 Discuss and collect lessons learned by each major activity area (continued)

– Other areas needing to be addressed

3:35 Next steps and assignments

4:10 Closing remarks

4:30 Adjourn

5. LESSONS LEARNED BY PROJECT ACTIVITY

The detailed lessons learned were discussed in order of project activities, from initiation to closeout, and are provided in the following sections.

5.1 Initiating Conversion Project

5.1.1 Initiation

The average survey score was 3.88.

Issues	Recommendations
Open communication between the university and the program went a long way in resolving a question of roles and responsibilities. In this case, the program analysts wanted to conduct the analyses, while the university believed they should perform them. The university saw it as an opportunity to thoroughly understand their reactor. A meeting was held to discuss the university's desires, rationale, and subsequently their capabilities and scope of analyses, and it was agreed to allow the university to do the analyses, with the program analysts providing guidance and expertise, as needed.	A valuable lesson learned in this regard was for the program to understand and respect the university's objectives, and the related programmatic benefits, and assist them as needed to accomplish their goals. With regard to the question of who would do the analyses, we needed confidence in each others' respective capabilities, clarity, and agreement of roles based on those capabilities, and subsequent demonstration of those capabilities in the undertaking of the project.

Issues	Recommendations
The university team was segregated a bit and it was not clear if all the necessary information was being shared appropriately.	A kick-off meeting with the university, designer, fabricator, analyst, shipping support, and shipper should take place as soon as possible to facilitate formal and systematic documentation of ALL technical and functional requirements for the entire project in a technical and functional requirements document. This would clarify roles, expectations, and requirements, and especially ensure that each piece of the design/specification could be verified against those requirements. Technical and functional requirements documents would be signed and become the "binding" document that everyone must abide by. Doing this will help eliminate many of the design problems that were experienced on this project. It would be a living document that gets revisited at each review.
Insufficient coordination of reviews caused delays and confusion.	Explicitly discuss "who else" needs to be "on board" to determine the support needed and establish essential contacts for review and information. Direct the university to provide, at the preliminary meetings, a list of those individuals that they want to review drawings, specs, and such.

5.2 Conversion Proposal Process

5.2.1 Contract Negotiation

The average survey score was 3.0.

Issues	Recommendations
Delays were experienced in the contracting process due, in large part, to lack of understanding of the work and time constraints by the contracts representatives.	Involve contracts/procurement people early in the process to promote an understanding of the work that mitigates nonessential delays.
Procurement and contracts personnel play a pivotal role in managing risks and clarifying obligations through the contracting process. However, their effectiveness can be suboptimized if they are ill-informed and are not involved early.	Start negotiations early to ensure the procurement process is less troublesome. Involve procurement personnel from both parties early, so that all parties are informed and working together.

5.2.2 Proposal Preparation

The average survey score was 2.83.

Issues	Recommendations
The age and history of any given reactor potentially allows for the likelihood that changes have occurred in designs, equipment, functionality, and such. These changes impact the design, analysis, and any number of activities on these projects.	Advise university early (at the start of the process or at the initial phase of the analysis) to recover and provide any historical documents, geometries, specifications, and such that are available. They also need to identify what information is missing so they can conduct whatever activities are necessary to fill those data gaps.
Lots of time was spent up front trying to determine format, content, and such. A clearer guideline of what the format (and some boilerplate) would be extremely helpful in preparing the proposal.	Now that it has been published, we need to use the NRC guide/template when preparing the proposal.
Although proposals are not due until a specific date, involvement of NRC to conduct upfront negotiations and clarify expectations and contractual obligations DURING proposal development would greatly improve the process.	Involve NRC in the proposal process as soon as reasonable regarding those areas where NRC involvement is stipulated (i.e., before the postal worker drops it off).
Proposal preparation went well. Lots of interaction back and forth with a clear, comprehensive plan and identification of who was responsible for what.	Embrace a collaborative and interactive operating philosophy, yielding constructive and clear communication and exchange.
The NRC oversight was value-added yet remained objective. Several aspects of the proposal can only be decided by NRC; therefore, early, open involvement is crucial. Use NRC as a technical resource/sanity check, and not just for answering administrative-type questions (e.g., changes to technical specifications), puts NRC in a position to “advocate” the conversion proposal on behalf of the university. Anytime the proposal preparer questions how NRC might react to a point, he/she needs to call and ask.	Use NRC as a technical resource/sanity check and not just for answering administrative-type questions. Anytime the proposal preparer questions how NRC might react to a point, he/she needs to call NRC and ask.

Issues	Recommendations
There is a risk in preparing the conversion proposal while developing the fuel, because gaps, tolerances, and such must be known, documented, and understood.	Complete the design before preparing the conversion proposal. This will ensure the correct design specs are included. The proposal can then move forward with significantly minimized risk. Transmit final drawings for fuel design to NRC to support their review of the analyses.
Picking overly restrictive tolerances causes safety limits to come down. Any future changes in design means analyses have to be revisited and sometimes revised. Over conservatism in tolerances may make fabrication nearly impossible. For example, the University of Florida proposal asked for a ±1 mil tolerance across a 26-in. element. This was rigorously discussed internally at the University of Florida and ANL (who conducted the analysis), but was not discussed with the designers at INL who would have resisted such a limited tolerance.	Be less restrictive during the analysis so that we are not so limited/restricted in the design. The fabricator and the designer MUST collaborate very closely at every phase of the process, almost as if they were the same entity, so that nothing is lost or overlooked. Better lines of communication between those conducting the analysis and those who are designing/fabricating the fuel are essential. This will go a long way to resolving the impacts of gap tolerances, design changes, and such. Involve ALL parties (e.g., analysis, design, fabrication, and university) in ALL conversations that will impact them directly or indirectly. Err on the side of inclusion and let people opt out.

5.2.3 Submittal of Proposal

The average survey score was 3.20.

Issue	Recommendations
Some confusion existed on whether the submittal should be paper copy or electronic and how many copies were needed.	Call NRC when ready to submit the proposal and ask the question.

5.2.4 Requests for Additional Information

The average survey score was 4.50.

Issues	Recommendations
After issuing the request for additional information, NRC visited the university to discuss their resolutions/dispositions to the questions. This was extremely effective and worked to expedite the question resolution process.	Continue this practice.
Before collaborative dialogue with NRC, the university and ANL prepared a draft response to the request for additional information so that discussions during the visits/proposal review were focused on the content of the response rather than on understanding and clarifying the request for additional information. This significantly accelerated the process.	Continue this practice.

5.2.5 Final Review and Comment on Proposal

The average survey score was 4.50.

Issues	Recommendations
This worked really well. Daily telecons to discuss and resolve issues and the willingness of participants to give and take to make it work was invaluable. Great interaction, initiative, listening, flexibility, and such.	Continue these practices.
The common vision and mission were critical.	Communicate these at the start of the project to all concerned, and continue to refer to them throughout the project.
Everyone had a great personal work ethic.	As much as practicable, select team members with established track records of success and excellence.
We had a single person (Dana for INL and Benoit for the University of Florida) that was solely dedicated to the project (allowing that person to stay in contact with all parties involved and to identify and track issues). This was instrumental to the success of the project.	Identify a key point of contact for the program and for the university to act in these roles.

5.2.6 Conversion Order

The average survey score was 3.50.

Issue	Recommendation
The NRC conversion order process went very smoothly. NRC provided great support and quick response to the proposal. This was highly appreciated.	Keep NRC informed; respect their role while leveraging their experience and expertise.

NOTE: Many of these issues are discussed with regard to collaboration and clarification between designers and fabricators. Communication and misunderstandings appear to be the biggest issue. Designers and fabricators (and analysts) need to talk openly and often. Inclusive (i.e., all parties) communications is critical.

5.3 Fuel and Hardware Development and Procurement

5.3.1 Fuel Specifications and Drawings

The average survey score was 2.20.

Issues	Recommendations
Design decisions did not include all essential members of the University of Florida team.	Advise the university about how critical it is to communicate and disseminate information among its own team.
Many players do not have experience reading drawings.	Assistance from other departments or organizations should be enlisted to assist the university in areas where it is needed.
The INL prepared mockups of components, and then when the University of Florida changed the specifications based on an analysis, INL would have to redo the mockup. This is expected; however, open and frequent communication can significantly minimize the impacts of those occurrences and the rework involved.	Anticipate an iterative process and advise those involved that the process will be that way. The design and specifications will change. We need to be ready for it and not resist when such changes come.
Absence of spacing and tolerance specifications created confusion.	Spacing requirements and tolerances need to be clearly documented on the drawing.

Issues	Recommendations
Assumptions with regard to design, fit, and function proved invalid, requiring correction.	Identify and document requirements such as spacing, tolerances, fit-up, and such in a technical and functional requirements document. Test all assumptions and VERIFY. Check the details early on, perhaps as early as the initial kickoff meeting. Perform mockups of designs to verify the designs work. Include mockups as part of the critical path so they are not forgotten. Verifying assumptions, specs, designs, and such is especially critical when continuity has been interrupted or extended in the process.

5.3.2 Fuel Fabrication Statement of Work and Procurement Documents

The average survey score was 2.78.

Issues	Recommendations
Issues regarding fuel fabrication quickly arose nearing the end of the process (e.g., questions on fabrication process, quality assurance programs, and channel spacing.)	Advise the university to become familiar with the fuel fabrication company's quality assurance documents and process. Involve the university in review and verification of the fabricator's quality assurance program. Ensure the preliminary meeting between all parties (e.g., university, analysts, designers, and fabricators) occurs to discuss what each party will get at each phase of the process. These same parties should be included in status and issues conversations throughout the process. Communicate all requirements for analyses and fabricability with all affected organizations.
The INL/DOE relied on the licensee to maintain the relationship with NRC and generally did not get involved with that relationship. When changes had to be made due to fabrication and analyses issues, NRC was not informed in a timely manner.	Advise and encourage the licensee to communicate openly with NRC regarding changes to fuel design and such. Need to ensure design is COMPLETE before submission of the proposal.

Issues	Recommendations
The magnitude of support needed to accommodate the changes in design and analyses was overwhelming at times due to constraints in time.	Planning and funding needs to anticipate making resources available to handle the simultaneous work.

5.3.3 Fuel Inspection

The average survey score was 4.00.

Issues	Recommendations
The blue books did not come with the fuel (i.e., several weeks delayed). The inspections were accomplished using advanced email or faxed copies rather than the final books. Could not verify individual plates because the serial numbers are too small to read and the plates were fastened into the elements. Having the blue books would have helped alleviate this problem because the books would have documented the inspectors' conclusions that the plates were as indicated on the drawings.	Ensure the quality assurance documents are provided up front. ACTION: BWXT will check to see why the blue books were not sent with the fuel.
Markings, labeling, and data were incomplete or scattered.	Pull together all markings, labeling, and data before inspections. Conduct both source inspections and receipt inspections. Advise the university to go to the fabricator and inspect the fuel before shipping.

5.3.4 Preparation of Facility for Fuel Receipt

The average survey score was 3.60.

Issues	Recommendations
The University of Florida was very restricted in their receipt area. Knowing what size of trucks could be accommodated was very helpful in coordinating the receipt of fuel. Communication of logistics between the university and the shipper was critical to successful receipt of the fuels.	Ensure the university and shipper communicate with regard to logistics, restrictions, and such.

Issues	Recommendations
Several different types of 6M drums were used at the University of Florida. The hardware needed for these drums was not communicated to the university.	Have shipper advice the university about the type of 6M containers (e.g., drawings and opening mechanisms) that will be arriving, so that the right tools are onsite at the receipt location.

5.3.5 Reassembly

The average survey score was 3.33.

Issue	Recommendation
Shipping assistance had to be provided to the university to return the empty canisters because the University of Florida was not familiar with the process (e.g., paperwork).	Make time early in the process to inform the university about the requirements for return shipment.

5.4 Core Conversion

5.4.1 Fuel Removal

The average survey score was 3.33.

Issues	Recommendations
A 90-day shutdown period is required before shipping the SNF. This timeframe needs to be closely coordinated with the university to ensure reactor needs are met and all implications of the shutdown are considered.	Make the university aware of the 90-day requirement and advise them to consider the implications of the schedule on reactor operations and research.
Contractors assisting the university with activities had unescorted access at the facility. Having Secure Transportation Services qualified as secondary operators at the reactor facility was instrumental during operational activities. This enabled them to move around and get things done without having to be constantly escorted.	Have contractors qualified as secondary operators at the reactor facility, and provide them with unescorted access.

5.4.2 Refueling

The average survey score was 3.50.

Issues	Recommendations
Several activities (e.g., maintenance, measurements, and disassembly) were required that could have been carried out earlier. This created a backlog as those activities became critical path and created additional schedule impacts.	Consider these activities early on, identify those that can be done earlier in support of conversion and schedule them. Add additional maintenance-type activities explicitly to the schedule so that they can be considered in the timing of the project. Activities that can be performed before receiving new fuel and reactor startup should be done as soon as possible, so as to not interfere with critical activities.
There was some unfamiliarity with the tools/equipment that needed to be resolved real-time during refueling activities. During loading, reactivity measurements were not reconciling with the University of Florida's calculations, causing uncertainty, questions, and undue stress on the operation. Reactivity at intermediate loading had not been calculated.	Require the university to have a comprehensive plan for refueling so they have a basis to reconcile differences between the analysis and the core measurements. This will be a formal commissioning/startup plan that compares calculated reactivity to measured values at intermediate loading during the refueling process. If possible, provide for onsite expertise to resolve startup issues during refueling. In the absence of onsite expertise, have a detailed plan and procedures with lots of hold points. Clarify explicit roles and responsibilities (e.g., what-ifs and ways to respond).
The university encountered unanticipated situations with regard to support equipment operability or function. Numerous questions arose as to how to respond to the arising issues.	Check all needed equipment (maintained and verified as operable) BEFORE you get to the critical point where it is needed. Conduct routine maintenance and pre-activity walk downs/ inspection of all needed equipment.

5.5 Spent Nuclear Fuel Shipment

5.5.1 Cask Determination

The average survey score was 3.67.

Issues	Recommendations
The university found the process for shipping SNF/cores offsite overwhelming due to the volume of orders and the regulations that applied. Even though lots of guides and documents are available, the pure volume of details and the uniqueness of what needs to be done takes time and coordination.	Anticipate the likelihood of trepidation and the sense of being overwhelmed. Be prepared to provide encouragement, support, and guidance. Develop a generic guide and a workshop to discuss shipping issues and put those who will be responsible for shipping in contact with those who have already done it. ACTION: Scott Declue will schedule a workshop to discuss the related issues and draft a guide in support of SNF shipping.
Lots of information was gained during walk downs. This was especially valuable when done in the preplanning stages. It opened the door for lots of questions to be addressed early on.	Continue to conduct these walk downs as a matter of practice.

5.5.2 Transportation Plan/Security Plan

The average survey score was 3.0.

Issue	Recommendation
Transportation and security plans are usually developed in tandem so the appropriate information can be conveyed, where allowed, with the parties. On this University of Florida effort, we were under a security information lockdown due to regulatory changes regarding safeguarded information, and were not able to share everything we needed to share.	The lockdown is over now, so this should not be a problem in the future. Need to begin the fingerprinting process early, and make it appropriately and effectively inclusive (include shippers).

5.5.3 Route Assessment

The average survey score was 3.2.

Issue	Recommendation
The route assessment was performed late in the process.	Conduct the route assessment as early as possible. Anything being shipped from a new location needs to have the route assessed as early as possible.

5.5.4 Certification of University Quality Assurance Programs

The average survey score was 3.0.

Issues	Recommendations
Universities are, in large part, unfamiliar with establishing a quality assurance program and writing a quality assurance plan.	Refer to other experienced universities, such as MURR (Missouri), for guidance to the NRC guidance.

5.5.5 Facility Preparations for Spent Nuclear Fuel Activities

The average survey score was 3.60.

Issue	Recommendation
Proactive, early involvement in preparing facilities for SNF activities is critical to success	Encourage and facilitate the inclusion of those involved in SNF activities in early discussion and preparations.

5.5.6 Support Equipment/Tools for Spent Nuclear Fuel Activities

The average survey score was 3.60.

Issues	Recommendations
A lid was built in accordance with the drawing; however, no one realized that the drawing was looking up at the lid. Subsequently, the lid was inverted. The error was caught during an unplanned dry run that was conducted during a project delay; therefore, no time was lost. Had there not been a delay, the project would have been hard pressed to correct the error.	(1) Pay closer attention to detail, and (2) conduct dry runs of newly designed equipment.
Each facility has its own equipment needs.	Identify specific equipment needs as early as possible.

5.5.7 Appendix A Preparation

The average survey score was 2.5.

Issues	Recommendations
Identification numbers on the fuel did not match the identification numbers listed in Appendix A.	Convey the importance of fuel element identification numbers to the shipper. If a discrepancy is found in the numbers, it should be documented and faxed to the field office immediately for response and resolution.
The university was not experienced nor prepared for the requirements of Appendix A submission. The preparation can be cumbersome, complex, and confusing.	Advise licensees of the requirements of the Appendix A submittal. Prepare a simplified guidance document (similar to a 1040A tax form) to show licensees how to prepare Appendix A. ACTION: Scott Declue will schedule a workshop to review Appendix A requirements and come up with a plan for providing the needed guidance.

5.5.8 Shipping Documentation

The average survey score was 3.0.

Issues	Recommendations
Required labels on the cask were torn off during transport due to harsh weather conditions.	Harsh weather conditions need to be considered when affixing labels.
Photos were taken of the BMI cask before shipping, showing the labels were in place before leaving the university.	Continue this practice of taking photos. They can be essential in providing information as a verification mechanism to regulating entities, especially when things change during transit.

5.5.9 Cask Loading

The average survey score was 3.67.

Issue	Recommendations
The lid for cask loading required rework.	Performing a "dry-run" of loading activities is essential to identifying problems with procedures, equipment, and such.

5.5.10 Receipt Facility Preparation

The average survey score was 3.33.

Issue	Recommendations
The University of Florida needed to have SNF shipped offsite in an extremely compressed schedule due to many factors (e.g., availability and scheduling of BMI casks and security issues of holding HEU in storage at the university). Additionally, hurricane force rain and winds impacted transport.	Advise university of the need for comprehensive planning, attention to detail, and anticipation of all relevant factors in preparing, scheduling, and shipping SNF. Have flexibility to relax the schedule if safety issues are a concern.

5.6 Other Issues

5.6.1 Safeguarded Information

The average survey score was 3.0.

Issues	Recommendations
The safeguards information issues have been resolved at NRC.	Submit fingerprints and other such information as soon as possible.

6. ROUND ROBIN

In concluding the discussion of the lessons learned, all participants were invited to reiterate, summarize, or offer any other lessons learned. The following list provides their final thoughts:

- There were lots of challenges on this project, but the team pulled together to meet those challenges and complete the project on schedule. Well done.

- The key to success was that everyone had the same goal and worked together to accomplish it.

- Next time we decide to use cones on the fuel plates, we need to taper them and not use hard edges. They do not go into the box very easily when they have hard edges.

- If we decide to have a workshop (e.g., initial orientation to the work and expectations), let us consider a single, comprehensive document and guidance that will address all of these issues with appropriate templates. It would be ineffective to pull all these people together in separate meetings to discuss each issue separately. A single guidance document and workshop would be the most efficient way to address it.

- Everyone in the project was working at or near capacity; therefore, the stress level was very high. It is great to work with people who can perform under such circumstances and know their limits so the work is (was) appropriately managed.

- It takes some time after refueling for the university to get the reactor up and running and to get operations back to normal. During this time, new operating procedures have to be written and operators have to be trained to the new procedures. The message here is that you will not start conversion on Monday and be back to full operation the next Monday. The transition and startup time after refueling needs to be planned for and coordinated. Additionally, the universities must prepare and have knowledge of reactor physics with appropriate onsite expertise. Certain parameters are needed to run tests in the reactor, and many of the operators do not have the reactor physics knowledge to do it. Depth of knowledge is the issue.

- Need to add operator training to the commissioning/startup plan that is discussed above. This is where the analysis information is conveyed to the operator. New operating procedures also need to be written, trained to, and implemented.

- Steps to success—communicate, plan, verify, and communicate. We need to involve future licensees in the next lessons learned meeting so they can have the information up front.

- Do not submit the conversion proposal and application until the information is full and complete.

- How issues are handled when they arise is a good indicator of the strength of the team. This was a great team.

7. ACTIONS

Scott will take the lead to establish a workshop to address activities needed for SNF shipping.

BWXT will check to see why the blue books were not sent with the fuel.

8. CONCLUSION

This lessons learned process has allowed us to capture gaps, opportunities, and good practices, drawing from the project team's experiences. The process is inclusive and offers an opportunity for every entity that "touched" the project to share from its experience. These lessons will be used to raise the standard of excellence, effectiveness, and efficiency in all future conversion projects. Despite making improvements to successive projects by addressing the lessons we have learned on this project, conducting a lessons learned activity will be vital to each conversion project as technologies, regulations, and other aspects of the environment change and influence success. It is recognized we cannot become complacent, nor adopt a mindset that the process has been "perfected."

Safety Aspects of Research Reactor Core Fuel Conversion from Highly Enriched Uranium to Low Enriched Uranium

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ABSTRACT

A number of safety issues are associated with research reactor core fuel conversion from highly enriched uranium (HEU) to low enriched uranium (LEU). These issues include aspects of LEU fuel qualification, reactor core neutronic and thermal-hydraulic analysis for steady state and transient conditions, and safety analysis. This paper discusses these issues on the basis of the IAEA safety requirements for research reactors. Other issues such as the commissioning programme for the converted core and preparation or update of licensing documents, in particular the Safety Analysis Report (SAR) and the Operational Limits and Conditions (OLCs), are also presented and discussed.

1. Introduction

In addressing increasing international concerns on security and non-proliferation, many of research reactors, through the “Reduced Enrichment for Research and Test Reactor (RERTR) Programme”, have been converted from HEU to LEU fuel. Other research reactors are currently implementing (or considering) core fuel conversion projects. Several safety aspects need to be considered in the implementation of such projects. These aspects include issues related to LEU fuel qualification programme, neutronic and thermal-hydraulic design, safety analysis, and licensing process.

In the frame of its programmes, the IAEA as an international coordinator continues to play an important role in supporting projects on research reactors core fuel conversion from HEU to LEU. These programmes include assistance activities to research reactors to ensure safe implementation of core conversion projects and associated reactor modifications [1].

The following sections present the safety aspects of the core fuel conversion projects and discuss, on the basis of the IAEA Safety Standards [2, 3, 4, 5, 6], the relevant issues and topics that need attention and careful considerations when implementing such projects.

2. Qualification of the New Fuel

Core fuel conversion projects involve use of new (or modified) fuel design and, in some cases, refurbishment of or modification to the reactor systems. Prior to its use, a newly designed (or modified) fuel has to be qualified by irradiation tests simulating the intended operating conditions. These qualification tests are aimed at demonstrating that the fuel design limits and safety criteria are not exceeded during steady state and transients conditions.

A qualification programme using lead test assemblies is usually performed for the verification of the mechanical, thermal-hydraulic, and neutronics performance of the new fuel. This programme includes non-destructive and destructive post irradiation tests. These tests allow, *inter alia*, for assessment of microstructure evolution during irradiation, measurement of the coolant channels gap profile, measurement of fuel volume change and cladding thickness and estimation of swelling, determination of relative longitudinal and transverse burnup, and detection of blisters or corrosion indication on the cladding. The qualification programme includes also power ramp tests, reactivity initiated accident tests, and loss of coolant accident tests [3].The analysis of the results of the qualification tests allows the determination of the fuel utilization limits.

It should be mentioned that there are several safety issues that are related to the installation of experimental devices for fuel qualification tests. Insertion of fuel assemblies to be qualified will change the value of the reactor reactivity shutdown margin, and may have influence on the decay heat of the core and its radioactive inventory. The accident analysis of the reactor may be also influenced. Detailed safety analysis demonstrating the safe implementation of such experiments should be performed. This analysis should include assessment of the interaction between the reactor core, irradiation device, and fuel under qualification as well as studies on potential radiological risks associated with the use of the irradiation device. The installation and performance of this type of experiments requires authorization from the regulatory body [2, 5].

3. Neutronic Analysis

The primary objective of the neutronic analysis associated with core fuel conversion projects is to ensure safe and optimum use of the fuel in the reactor, while remaining within the limits imposed by the design of the fuel assembly and those related to the operation of the reactor, which are based on the Operational Limits and Conditions (OLCs) derived from the safety analysis. A secondary objective is to meet the requirements of the reactor utilization programme (e.g. neutron flux to experiments).

The neutronic parameters important to safety are affected by core fuel conversion. Evaluation of these parameters should be covered by the neutronic analysis. This analysis should include determination of the core excess reactivity, shutdown margins, and reactivity feedback coefficients. These parameters are subject to limitations specified in the OLCs. The reactivity feed back coefficients (in addition to the reactor kinetic parameters which need to be also determined by the analysis) are used in the safety analysis.

Some research reactors undergoing fuel conversion have a second shutdown system (e.g. drainage of the moderator or injection of a neutron absorber). In these cases, the neutronic calculations should include analysis of the shutdown capability of that system and any change of its reactivity worth.

The reactivity worth of experiments could be affected by changing the fuel in the core from HEU to LEU. The neutronics analysis should demonstrate that the values of the reactivity worth of the experiments are kept within the reactivity limits specified in the OLCs.

The neutronic analysis for the core fuel conversion should also cover determination of the detailed power distribution across the reactor core and verification that the nuclear power peak factor remains below the value specified in the OLCs. These parameters are used as an input to the steady state thermal-hydraulic calculations. The nuclear power peak factor is the most important neutronic parameter for the evaluation of the thermal-hydraulic safety margins.

4. Thermal-hydraulic Analysis

The objective of the thermal-hydraulic analysis is to ensure that the heat generated in the core can be adequately removed, so that the fuel and clad temperatures are kept within acceptable values in all operational states and design basis accidents. This analysis should demonstrate that the reactor can be operated with adequate safety margins against the thermal-hydraulic critical phenomena and should cover both forced and natural circulation cooling conditions.

The critical phenomena that are of concern in the thermal-hydraulic design are departure from nucleate boiling and flow redistribution. Although it is not by itself a critical phenomenon, the onset of nucleate boiling is also considered in the analysis as a measurement of the approach to a heat transfer crisis. In addition, the analysis should also demonstrate that the coolant velocity through the reactor core is adequately below the value of the critical velocity.

The analysis should take into account the uncertainties due to fuel fabrication tolerances, deviations in the construction process, simplifications made in the thermal-hydraulic models and possible deviations in the operational conditions. These uncertainties may have significant influence on the results obtained and should be treated using a conservative approach.

The calculations associated with the thermal-hydraulic analysis should be performed for the reactor core hot channel and cover all planned core configurations, including the configuration with the minimum core size. The computer codes used for these calculations should be qualified for their validity to use in research reactor's analysis. Where practical, use of an instrumented fuel element in the converted core will allow for an experimental validation for the thermal-hydraulic calculations.

The results of this analysis form the basis for defining the safety limits, applicable to the forced and natural circulation modes of reactor operation, and safety system settings for the relevant parameters such as reactor power, coolant flow rate, pressure difference across the core, and coolant temperature at the core inlet (or outlet).

In some cases within core fuel conversion projects, there may be a need to operate a research reactor with HEU and LEU fuel loaded into the core (i.e. mixed core). In these cases, it should be ensured that the values of the nuclear power peak factor are kept within the acceptable limits.

5. Safety Analysis

The results of the analysis of some of the Postulated Initiating Events (PIEs), originally performed for the HEU fuel, could be affected by changing fuel assemblies to LEU. Therefore, the reactor safety analysis should be revised in the framework of the core fuel conversion projects.

The objective of the revised safety analysis for the core fuel conversion is to demonstrate that the reactor can be kept within the safety conditions established in the design. It should also demonstrate that the radiological consequences of the design basis accident do not modify the conclusions of the analysis presented in the Safety Analysis Report (SAR). The scope of the safety analysis revision should cover the event sequences, the evaluation of the consequences of the PIEs, and a comparison of the results of the analysis with the radiological acceptance criteria and design limits.

For core fuel conversion and in performing the revision of the safety analysis, the PIEs that were originally considered in the case of HEU fuel should be compared to the list of PIEs recommended by the IAEA [4], and completed as necessary. Operating experience from the reactor under consideration, or from similar reactors (including examination of event reports and the database of the IAEA Incident Reporting System for Research Reactors), can be also used to supplement the list of the selected PIEs.

Particular emphasis should be put on review of the design basis accident and on the PIEs that involve criticality and positive reactivity insertion. These PIEs includes criticality and erroneous handling of the fuel (e.g. error in fuel insertion, fuel storage criticality, dropping of transfer flask on the fuel, etc.), start-up accident, inadvertent ejection of control rods, unbalanced control rod positions, and insufficient shutdown reactivity. These PIEs also include erroneous handling of experiments or experimental devices, and maintenance errors with reactivity devices.

The results of the revised safety analysis should be reflected in an updated version of the SAR and OLCs. These results should also be used, as appropriate, in the revision of the operating procedures, periodic testing and inspection programmes, and emergency planning.

6. Commissioning Programme

Core fuel conversion from HEU to LEU is a project with major safety significance. The IAEA Safety Standards require implementation of a formal commissioning programme for this category of research reactor modification [5, 6]. This programme should be aimed at demonstrating not only the functionality of the modification but also its safety. This programme should also demonstrate that all safety requirements and intent of the design stated in the SAR are met for the converted reactor core.

The commissioning programme for a core fuel conversion project should cover the following:

- Description of the organizational set-up for the project as well as the roles and responsibilities of the individuals involved;
- Stages of the commissioning process (pre-operational tests, initial criticality and low power tests, and power rise tests), including the planned tests, and the associated pre-requisites and schedule;
- Commissioning test procedures including administrative procedures;
- Management system/Quality assurance programme for commissioning that includes verification, review, audits, and treatment of non-conformances.

The commissioning tests for core fuel conversion projects should include, in particular:

- Approach to criticality;
- Measurements of the shutdown margin, reactivity worth of the control rods, and core excess reactivity;
- Flux measurements and estimation of power peak factors;
- Measurements of reactivity feedback coefficients;
- Measurements of reactivity worth of the in-core and reflector experimental devices such as irradiation loops, rigs, and capsules;

- Measurements of reactivity worth of the second shutdown system, as applicable;
- Calibration of neutronics instrumentation and adjustment of the safety system settings, on basis of thermal balance measurements;
- Verification of tools and equipment for handling the new LEU fuel.

In addition, the commissioning tests should cover all the administrative procedures associated with the core fuel conversion projects.

7. Licensing Process

The IAEA Safety Standards require that projects with major safety significance such as the core fuel conversion be subject to authorization from the regulatory body prior to its implementation [2, 5].

Before loading the new LEU fuel into a research reactor core, the operating organization should submit to the regulatory body for review and assessment the fuel design and its qualification results (verification of mechanical, neutronic, and thermal-hydraulic limits), input data for prediction and monitoring of the reactor LEU core behaviour, revised safety analysis, and the corresponding commissioning programme. In its application for the LEU operating license, the operating organization should submit to the regulatory body an updated SAR with the results of the commissioning programme, and updated OLCs and emergency plan.

8. Conclusion

Research reactor core fuel conversion from HEU to LEU is a project with major safety significance, which requires authorization from the regulatory body prior to its implementation. Detailed safety analysis for the converted core should be performed to demonstrate that the reactor can be kept within the safety conditions established in the design. A formal commissioning programme should be established to verify that the reactor can be operated according to the design intent and in compliance with the OLCs. Updated safety documents (SAR, OLCs, and emergency plan) form the basis for the licensing process of the converted core.

The IAEA will continue to provide assistance to Member States to ensure safe implementation of core fuel conversion projects.

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IMPACT ON FUEL CYCLE COSTS OF CONVERSION TO LOW ENRICHED URANIUM FUELS

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ABSTRACT

Strenuous efforts are being made worldwide to convert research reactors to use LEU fuels instead of HEU fuels, in particular by USDOE's GTRI program. Considerable progress has been made with more than fifty reactors converted as of the date of this paper, and many more expected to do so in the next several years. The number of reactors converted is now sufficient for meaningful statistics relating to the conversion process to be compiled and analyzed.

One of the key issues for the operators of research reactors planning or considering conversion of their reactors is the impact on operating costs, and in particular on fuel cycle costs.

This paper examines the experience to date with reactor conversion and compares fuel cycle costs before and after conversion. Key items affecting fuel cycle cost are analyzed to provide an informative empirical guide that will be useful to guide decision making during the conversion process.

1. Introduction

Strenuous efforts are being made worldwide to convert research reactors to use low-enriched uranium (LEU) fuels instead of highly-enriched uranium (HEU) fuels, in particular by the USDOE's GTRI program. Considerable progress has been made with more than fifty reactors converted as of the date of this paper, and many more expected to do so in the next several years.

A critical issue for reactors that undergo LEU conversion is to maintain the performance of the reactor, by changing to new fuel types with higher uranium densities, by reconfiguring the core and so on. The discussion, as evidenced by the papers presented at the RRFM and RERTR Annual Meetings, etc., focuses on the impact of conversion on the technical and safety parameters of the reactor, with particular attention paid to maintaining neutron flux density. As an example, the recent paper by Glaser [1] took as its reference the INFCE criterion that "any loss in the overall reactor performance" such as flux per unit power "should not be more than marginal".

Little attention is paid in the published papers, however, to the impact of conversion on the economics of reactor operation, even though this is clearly important to reactor sustainability after conversion. The decisions made during the conversion process will potentially shape the economics of reactor operation through the impact they have on both the cost and revenue aspects. Issues such as the cost of fuel acquisition and spent fuel management, and operating parameters such as the cycle length, impact the operating costs, while changes in neutron flux densities and irradiation facilities may change the functional capability and potential revenue generating capability of the reactor

Economic sustainability is a major challenge for many research reactors worldwide. In a funding climate where direct governmental support for many facilities is diminishing, all changes and new activities have to be evaluated in terms of their impact on the economic future of the reactor. In that context, a systematic evaluation of the potential impact on the economics of operation would be a logical part of the planning for the conversion of a reactor to use LEU fuels, with the results of that evaluation taken into account in the development of the criteria for fuel design and other decisions. This would help to mitigate any negative economic impact.

The published literature provides little insight into the impact of conversion on the economics of operation, and each reactor operator must construct its own framework for such an assessment. An understanding of whether, and how, reactor conversion has affected operating costs of those reactors that have already converted, or are at advanced stages in the planning for conversion, will help inform those reactor managers that are currently considering conversion. The purpose of this study, therefore, was to provide an initial examination of the impact on the economics of operation of converting a research reactor to use LEU fuels, and from that examination to determine whether any general patterns are evident in the conversion projects to date.

2. Framework for the Economic Analysis

The current study project began with anecdotal evidence that, at least in some cases, the special fuel designs required for reactor conversion were more expensive to fabricate than the HEU fuels used prior to reactor conversion, and that the fuel fabrication costs had not been formally included in the conversion decision making.

In order to provide a more comprehensive view of the economic impact, a questionnaire was developed and combined with a telephone survey to elicit information about the impact of conversion on reactor operations and the costs of operation. This was supported by a literature search, primarily from papers to RERTR and RRFM Annual Meetings.

Obtaining statistically comparable data presented some difficulty. For example, in the case of fuel acquisition, direct comparison of pre-conversion and post conversion costs was often not feasible because of the rapid changes in the international market prices for enriched uranium, the irregular, infrequent purchases of HEU fuel prior to conversion and, in certain cases, the commercial confidentiality of information. In addition, it is not universally safe to assume that HEU fuel would have continued to be routinely available under the preexisting terms and conditions had the reactor not been converted.

Information that was easier to collect included issues such as changes to the operating cycle length, re-configuration of the core, changes in reactor power level and changes in spent fuel discharges.

The questionnaire and the telephone survey were designed to cover all the main issues noted above, as well as to solicit other items of importance to the respondent reactor managers. The data was tabulated to allow comparisons and conclusions to be drawn, without compromising any of the proprietary information in the individual responses.

3. Results of the Analysis

1. Most reactor operators surveyed did not use explicit economic optimization criteria during planning for conversion, see Figure 1. Those that did were primarily those with a strong commercial focus, for example [2].

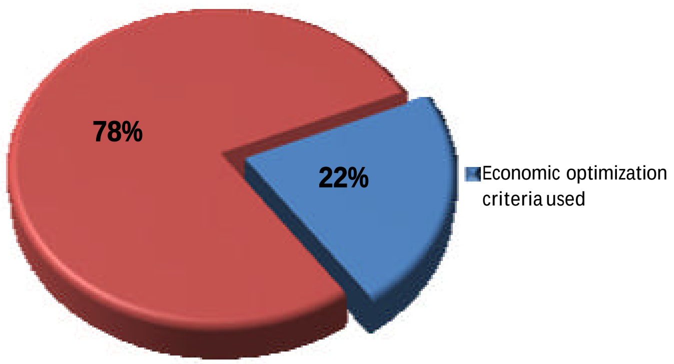

Figure 1: Use of Economic Optimization Criteria during Conversion

2. Increased operational costs were reported only by those reactors that refuel during normal operations. Smaller reactors (< 0.5MW) reported little or no requirement to purchase or dispose of fuel post conversion, and therefore no quantifiable impact on operating costs following conversion. In all cases surveyed, these smaller reactors were able to maintain the neutron flux densities, shut down margins, and other technical parameters needed to fulfill their mission, and so are not vulnerable to changes in income as a consequence of LEU conversion.

3. Similarly, those reactors that are primarily involved in activities that are not sensitive to the exact flux provided, for example, teaching or geochronology, were not sensitive to the potential reduction in flux and this was, thus, not a "flux penalty".

4. For the larger reactors, there were no trends or rules that applied to all. Differences between facilities dominated the analysis, even for reactors that use similar fuel types.

5. Although an unavoidable reduction in flux density is often discussed as a likely consequence of conversion to LEU fuels, this was not supported by the data. As many reactors maintained their neutron flux density as suffered reduced neutron flux densities, see **Figure 2**.

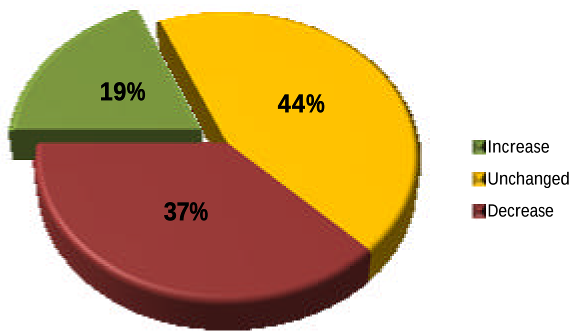

Figure 2: Change in Neutron Flux Density Following Conversion

6. In certain cases a reduction in neutron flux density as a consequence of conversion was dismissed as insignificant by the reactor manager. In others, an increase in reactor power has been used or proposed to maintain the neutron flux densities

required for the reactor's planned activities. The planners for the SAFARI reactor, for example, have restating the criteria for conversion in terms that include "limited loss (preferably none) in the maximum production capacity", and "no increase in the fuel cost per production unit" [2]. These observations suggest that preserving neutron flux density per unit power is less relevant in today's context than preserving the capability to execute a strategic plan or reactor mission.

7. Some reactor operators reported increases in the fuel acquisition costs, and in the costs of fuel fabrication. As noted above, however, it is difficult to isolate the reasons for the cost increases. Infrequent purchases of fuel prior to the date of conversion mean that changes in the market cost of enriched uranium cannot be separated from the costs of fabrication at this stage of the analysis.

8. Several reactors reported an increase in fuel acquisition costs because of increased consumption of fuel assemblies, due either to a lesser duty cycle for the fuel, or to the need to increase reactor power to offset a reduction in neutron flux density. An increase in the number of fuel assemblies consumed not only increases the new fuel procurement costs, but also potentially increases the spent fuel management costs.

9. Spent fuel management costs were recognized as a very significant issue, but for several of the reactors surveyed, the true cost of spent fuel management was masked by the Foreign Research Reactor Spent Nuclear Fuel (FRRSNF) Acceptance program. For the period the FRRSNF program is in operation, many reactor operators are not exposed to increased costs of spent fuel management, either on the basis of number of fuel assemblies discharged, or because of the choice of fuel meat. However, at the end of the program, a step change in spent fuel costs is expected, with the potential to threaten reactor sustainability. One of the respondents noted that that this could become a disincentive for some reactors to convert. This observation suggests that an analysis of the spent fuel management costs would be of particular significance in planning conversion, and it also creates a strong linkage between the FRRSNF acceptance and GTRI conversion programs.

10. There was no clear picture on the impact of security costs. Although in principle, removal of all HEU from a reactor site might be expected to reduce the costs of security, this was not supported by the limited data available. In most cases, changes and upgrades in nuclear facility security standards over the past several years have masked the savings that may have resulted from conversion to LEU fuel.

11. Conversion can affect regulatory costs, with two of the respondent reactors noting that proposed increases in reactor power to offset a reduction in neutron flux density would result in additional regulatory work and potential costs.

4. Conclusions and Recommendations

Conversion to LEU fuels can significantly impact the economics of reactor operation in terms of both revenues and costs. Therefore, strategic planning and economic analysis should be integral to the planning for the conversion of a research reactor. Such an analysis would highlight issues that are potentially relevant, and provide information on how they might be optimized to reduce, or avoid, a negative economic impact. For example, at what point does a change in the flux density become economically or functionally relevant? A cost benefit analysis based on the reactor's strategic plan would show at what point a reduction in neutron flux density becomes a flux penalty, and whether it is relevant to the future sustainability of the reactor. The analysis would also quantify the level of increased cost and effort that can be tolerated when correcting the problem.

In this regard, economic *sustainability* of reactor operations according to the strategic and business plans for the reactor would be a more valuable means of determining conversion priorities than isolated technical criteria such as neutron flux density per unit power.

The current analysis should be developed further, both to improve the statistical analysis of the economic impact of LEU conversion, and to develop a mechanism and guidelines for assessing the economic impact. This would assist research reactor operators who are planning to convert their reactors to use LEU fuels to fully understand the potential impact of conversion on their operating costs and to optimize their plans to ensure sustainability.

5. References

[1] Alexander Glaser, *Performance Gain with Low-Enriched Fuel and Optimized Use of Neutrons*, The 29th International Meeting on Reduced Enrichment for Research and Test Reactors (RERTR) September 23-27, 2007, Prague, Czech Republic

[2] WE Stumpf, AP Vermaak, and G Ball *Key Considerations in the Conversion to LEU of A Mo-99 commercially producing reactor: SAFARI-1 of South Africa* 2000 International Meeting on Reduced Enrichment for Research and Test Reactors October 1 - 6, 2000 Las Vegas, Nevada, USA

OPTIMIZATION STUDIES FOR CONVERSION OF THE MIT REACTOR TO LEU FUEL

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ABSTRACT

Studies are underway as to the optimization of fuel plate thickness and number of plates per fuel element of a monolithic U-10Mo LEU fuel and configuration design for the MIT Reactor. MCNP and REBUS-PC neutronic models are used for determination of radial and axial power peaking, which are used in the thermal-hydraulics codes MULCH-II to determine the limiting safety system setting (LSSS) for maximum power level, based on onset of nucleate boiling (ONB) in the hottest channel. The pressure drop across the core and corresponding pressure on the reactor core tank is calculated to assure that pressure limits are not exceeded.

These studies have shown that core configurations exist for which a reactor power level greater than 6 MW is achievable with LEU fuel. In addition, burnup modelling has shown a significantly longer operating cycle using LEU fuel than with HEU fuel.

1. Introduction

The MIT Reactor (MITR-II), contains a hexagonal core that contains twenty-seven fuel positions in three radial rings (A, B, and C), as shown in Figure 1. The reactor is currently licensed to operate at 5 MW, with an upgrade to 6 MW expected soon. Typically at least three of these positions are filled with either an in-core experimental facility or a solid aluminium dummy element to reduce power peaking. The remaining positions are filled with standard MITR-II fuel elements. Each rhomboid-shaped fuel element contains fifteen aluminium-clad fuel plates using HEU (93% enriched) in an aluminide cermet matrix with a fuel thickness of 0.76 mm (0.030 in.) and a length of 61 cm (24 inches). The cladding of each fuel plate has 0.25 mm longitudinal fins to increase heat transfer to the coolant. The fuel has an overall density of 3.7 g/cm^3, with a total loading of 506 g ^{235}U in each element (445 g ^{235}U prior to 1980).

The core is light water moderated and cooled and is surrounded by a D_2O reflector. Boron impregnated stainless steel control blades are present at the periphery of the core at each of the sides of the hexagon. In addition, fixed absorbers of boron-stainless steel can be installed in the upper twelve inches of the core in a hexagonal configuration between the inner and second fuel rings as well as in three radial arms extending to the edge of the core.

Several reentrant thimbles are installed inside the D_2O reflector, delivering greater neutron flux to the beam ports outside the core region. Beyond the D_2O reflector, a secondary reflector of graphite exists in which several horizontal and vertical thermal neutron irradiation facilities are present. In addition, the MITR Fission Converter Facility is installed outside the D_2O reflector. This facility contains eleven partially spent MITR fuel elements for a delivery of a beam of primarily epithermal neutrons to the medical facility for use in Boron Neutron Capture Therapy (BNCT).

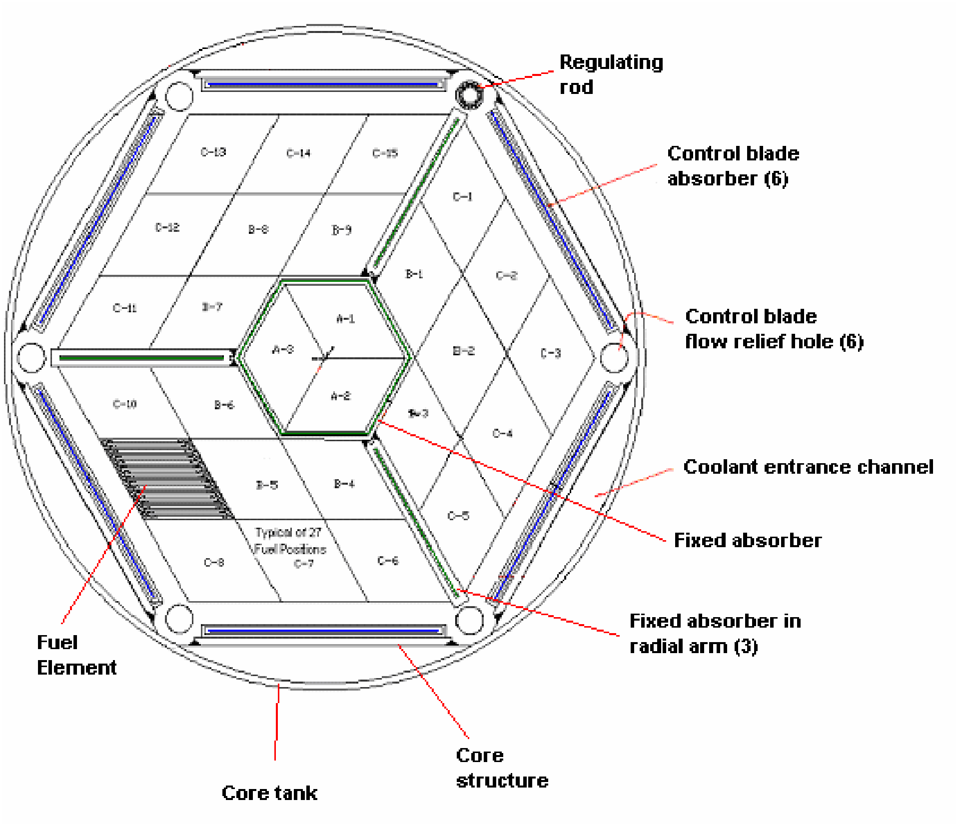

Fig. 1. The MIT Reactor core

2. Modelling

The MITR core has been modelled using the Monte-Carlo transport code MCNP for the current HEU configuration as well as for studies of conversion of the MIT reactor to LEU fuel. In addition, the WIMS-ANL 1-D transport code has been used for generation of neutron multigroup cross-section libraries, along with the REBUS-PC code for fuel cycle analysis. [1]. The REBUS-PC model uses a triangular-Z matrix, necessary because of the rhomboid shape of the MITR fuel elements. [2]. These models have been fairly well validated using operational data from HEU core #2, which consisted of twenty-two new (445 g ^{235}U) fuel elements and five aluminium dummies in-core with no fixed absorbers.

For thermal-hydraulic modelling, the multichannnel analysis code MULCH [3] is used to determine the limiting condition of onset of nucleate boiling (ONB). ONB in the hottest channel is the basis of the reactor limiting safety system settings (LSSS).

For conversion studies, monolithic U-Mo fuel with 10 Mo (U-10Mo) has been chosen as the target fuel. This fuel has a uranium density of 15.3 gU/cm^3. For the LEU fuel, a nominal cladding thickness of 0.25 mm was chosen with 0.25 mm fins added. Dispersion fuels of lower densities were not considered because of the dificulty of achieving criticality under similar core configurations.

3. Fuel Design Optimization

MCNP studies have shown that direct replacement of HEU aluminide fuel with U-10Mo monolithic fuel will result in a flux loss of at least 10% to all experimental facilities if the reactor remains at the same power level. Thus, it was necessary to optimize the fuel design for delivery of the maximum appropriate energy neutron flux to experimental positions. This can be done by core design and by maximizing the reactor power level. The design was optimized by varying the number and thickness of fuel plates. A number of designs were included, but since power generation per plate becomes larger with a lower number of plates, only fuel elements with 15 plates per element and above were considered.

Radial power peaking (hot channel factor – the ratio of the heat generated by the hottest plate to that of the average plate) was determined for each proposed design by calculation using MCNP. These values were generated using fresh fuel for the entire core, thought to be the most conservative case. This peaking was then input into MULCH for determination of the LSSS for the design. An additional constraint is that of the pressure on the reactor core tank. A nominal pressure limit of 172 kPa (25 psi) is given for the design, but the origin of this value is uncertain. For conservatism, the current operating pressure of 103 kPa (15 psi) was used as an operating limit for fuel design.

4. Results

MCNP results comparing HEU core #2 (15 plates per element with 0.762 mm fuel thickness and 0.381 mm cladding), equivalent dimensions with a U-10Mo LEU core, and LEU with 0.508 mm and 0.381 mm thick fuel are shown in Table 1. The flux values shown are at a reactor power level of 5 MW. The 12" beam port flux is seen to be representative of all ex-core experimental facilities.

This table shows a significant softening of the in-core neutron spectrum with reducing plate thickness, as would be expected, since more water is present. This also results in a higher K_{eff} for the thinner plate cases. It may be possible to take advantage of this higher excess reactivity to further reduce peaking and increase neutron fluxes. Plate power peaking is lower for thinner plates, with a hot channel factor of 1.65 for the 0.508 mm case, as compared to 1.73 for the 0.762 mm case.

Given the significant reduction in fluxes with LEU, it is apparent that at least a 10% increase in reactor power will be necessary to maintain the HEU equivalent fluxes.

	HEU core #2	**LEU core #2 (0.762 mm)**	**LEU 18 plate 0.508 mm**	**LEU 18 plate 0.381 mm**
K_{eff}	1.00275	1.0050	0.9966	1.0006
In core experimental fast flux (> 1MeV)	6.57E13	6.44E13	6.36E13	6.27E13
In core experimental thermal flux	2.76E13	6.88E12	1.75E13	2.10E13
12" beam port flux	7.36E12	6.79E12	6.63E12	6.51E12
Hot channel factor	1.43	1.73	1.65	1.61

Table 1. MCNP results of selected cases

The core tank pressure loading calculations are shown in Figure 3. All points below the HEU pressure drop curve meet the pressure limit criterion. Thus, there are several options available for the 0.508 and 0.381 mm fuel cases, up to 18 and 19 plates per element, respectively.

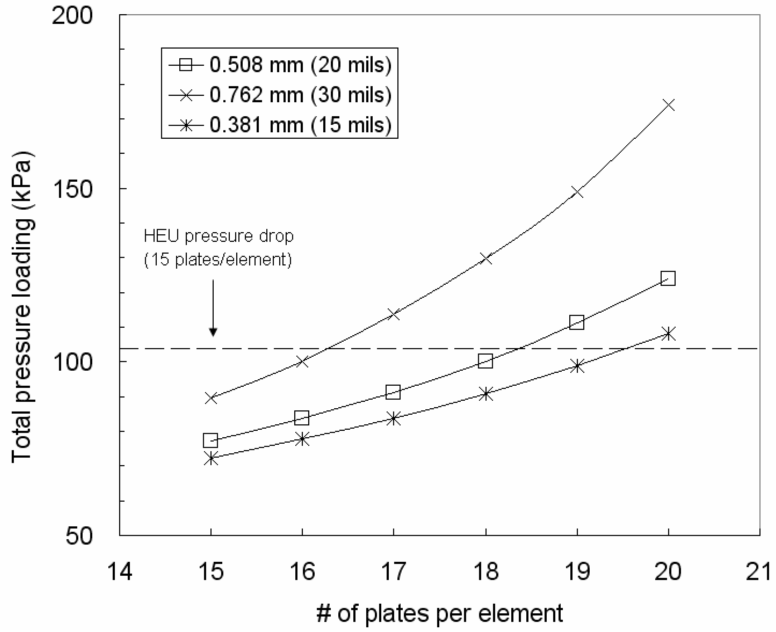

Figure 3. Core tank pressure loading with varying plate dimensions

The LSSS power as calculated by MULCH is shown in Figure 4. Since there remains some uncertainty of the hot channel factor (HCF), three different HCFs were assumed, including a highly conservative 2.0. Because the goal is to maintain neutron fluxes at or above those of the HEU core, only cases with an LSSS power level above the HEU level of 7.4 MW would meet this criterion. Given the pressure limitations above, the most promising case is that of 18 plates per fuel element. Provided that the HCF can be reduced to 1.6 with fuel management, the LSSS power with 18 plates per element is 9.6 MW, which would allow reactor operation at 8 MW without modifications to the core tank structure.

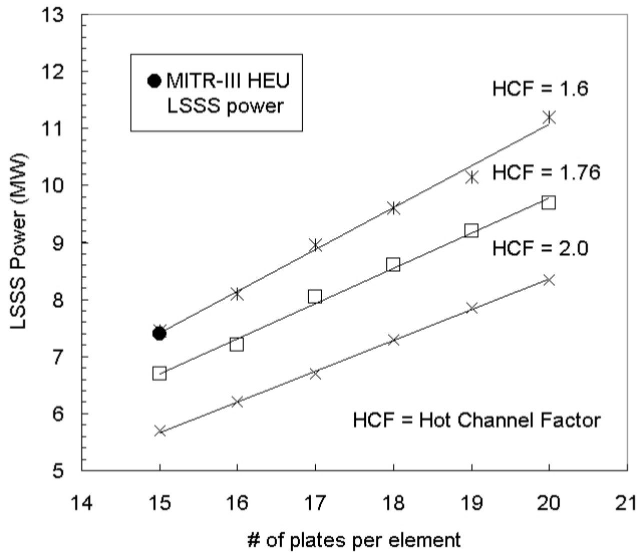

Figure 4. LSSS power for 0.508 mm thick fuel.

4.1 Burnup results

The REBUS results of burnup at 6 MW with the 0.508 mm thick LEU fuel case as compared with the current HEU case is shown in Figure 5. This clearly shows significantly lower reactivity loss with LEU fuel, indicating that an LEU core may be able to run twice as long prior to refuelling.

The REBUS model will be used to determine a fuel management strategy designed to reduce power peaking.

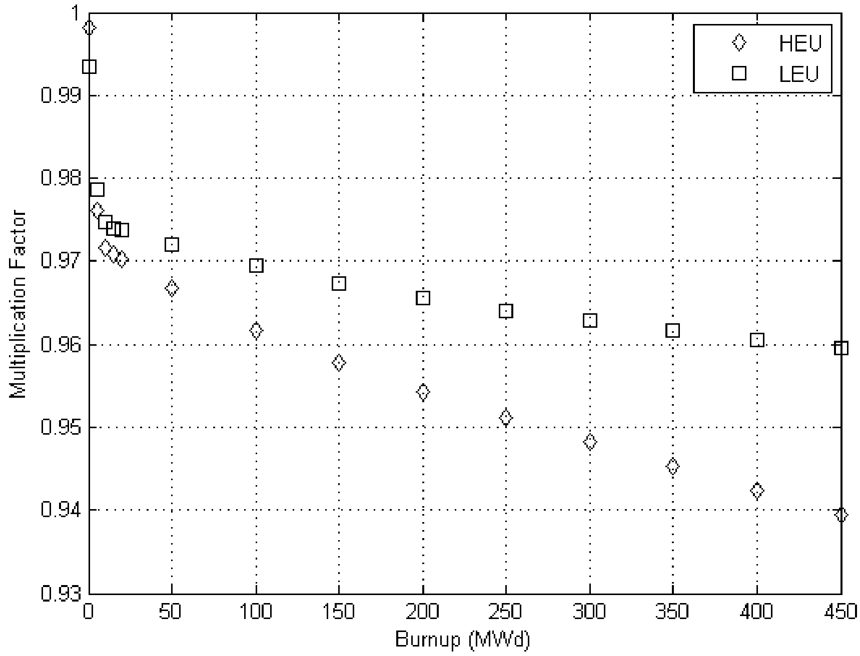

Figure 5. Burnup comparison between 0.508 mm LEU fuel and 0.762 mm HEU fuel

With the given constraints listed above, the calculational methods employed here show that an LEU fuelled MIT Reactor operating at power levels above 6 MW is achievable.

5. References

1. T. Newton, J. Stillman, and A Olson, "Reactor Core Design and Modelling of the MIT Research Reactor for Conversion to LEU," Proceedings of the 2007 International Meeting of the Reduced Enrichment for Research and Test Reactors, Prague, Czech Republic, September, 2007.
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OPTIMIZED CONTROL ROD DESIGN OF THE REACTOR BR2

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ABSTRACT

At the present time the BR2 reactor uses control rods with cadmium as neutron absorbing part. Due to the burn up of ^{113}Cd in the hot spot during reactor operation, the presently used rods for reactivity control of the BR2 reactor have to be replaced by new ones. Considered are various types control rods with full active part of cadmium, hafnium, europium oxide and gadolinium oxide. Options to decrease the burn up of the control rod material at the hot spot, such as use of stainless steel in the lower active rod part are discussed. The changing of the control rod characteristics and the perturbation effects on the reactor neutronics are investigated during 1000 EFPD of reactor operation. The calculations are performed for the full scale 3-D heterogeneous geometry model of BR2 using MCNP&ORIGEN-S combined method, MCNPX 26E. An optimal design is given and a new control rod type is chosen for the reactivity control of the reactor BR2.

1 INTRODUCTION

Historically, the earliest reactivity control of the BR2 reactor core has been maintained by control rods with full length made of cadmium as absorbing material. The experience has shown that the lower edge of the control rod, which is exposed to the highest thermal neutron flux, is burning out under irradiation mainly due to depletion of the dominant cadmium isotope ^{113}Cd. The objective is optimization of the control rod design with focus on the choice of main absorbing material in the active rod part. Combinations of black absorber with grey one, (e.g. stainless steel) both to diminish the burn up of the black absorber and to flatten the neutron flux and power distributions in the core have to be considered. The old control rod design has to be optimized or proposed new one if this is demanded by the choice of the new absorbing material. The existing procedures for experimental determination the control rod worth have to be revisited to satisfy the new irradiation conditions. The chosen new control rods must provide the necessary negative reactivity for adjustment of the power level during ~ 30 effective full power days and they should not disturb too strongly the neutron flux and power distributions in the BR2 core. The chosen main control rod absorbing material should have both high thermal and high epithermal absorption cross sections, but should not burn too fast in order to be used at least during 25 BR2 operating cycles. A detailed comparative analysis for the control rod life of various absorbing materials during long time of irradiation (~ 1000 EFPD) is available in the Master Thesis of X.Liu [1].

2 CALCULATION METHODOLOGIES

The methods of the reactor kinetics lay in the basis of the experimental techniques for determination of differential control rod worth at the BR2 reactor. A simple type of kinetic measurement can be performed if we make a small perturbation (insertion of a positive reactivity) in the critical reactor and then to measure the asymptotic period of the resultant core transient. One can derive the reactivity worth of the perturbation from a measurement of the asymptotic positive period using the in-hour equation:

$$r = \frac{\Lambda}{Tk_{eff}} + \sum_{i=1}^{7} \frac{b_i}{1 + l_i T} \qquad (1)$$

The perturbation theory [2-3] is applied for estimation of reactivity values of the partially inserted control rod and used in the experimental technique for determination of total control rods worth at the BR2 reactor. A hybrid Monte Carlo & perturbation method devoted to improve the experimental procedure for estimation of the control rod worth is proposed [4]. MCNPX [5] is used for steady-state flux and spectra calculations and calculations of control rod worth for the full-scale 3-D realistic heterogeneous geometry model of BR2. Detailed 3-D isotopic depletion calculations of the absorbing rod material are performed using automatic Monte Carlo burn code MCNPX 2.6.E [6], which has been validated on the fuel depletion calculations of the BR2 reactor [7]. Detailed 3-D space dependent distribution of the isotopic fuel depletion in the whole reactor core is performed by the coupled MCNP&ORIGEN-S method [8-9] using about 4000 to 6000 cells with varied burn up.

3 CR MODIFICATIONS AND CHOICE OF OPTIMAL PROJECTS

3.1 Reference Control Rod

The BR2 reactor uses shim control – safety rods, which provide both the coarse normal operational control and the safety control. Each mobile rod consists of two sections. The lower section is a beryllium assembly cooled by water. The upper section is a round cadmium tube clad with aluminum on both sides. The cadmium section is completely inserted in the active core when the rods rest on their shock absorbers. A capsule containing approximately 190g of cobalt particles is inserted between the lower end of the cadmium section and the upper end of the beryllium assembly.

3.2 Control rod candidate materials

The material selected for control rods should have a good absorption cross section for neutrons and have a long lifetime as an absorber. Materials with very high absorption cross section may not be preferred because they disturb strongly the neutron flux in the vicinity of the rod and generate big reactivity perturbations in the core. They can burn out rapidly unless transmuting into another isotopes having also high absorption cross section. Materials that resonantly absorb neutrons are often preferred to those that merely have high thermal neutron absorption cross sections. The path length traveled by the epithermal neutrons in a reactor is greater than the path length traveled by thermal neutrons. Therefore, a resonance absorber absorbs neutrons that have their last collision farther from the control rod than a thermal absorber. This has the effect of making the area of influence around a resonance absorber larger and it is useful in maintaining a flatter flux profile. The most commonly used elements for reactivity control in research reactors are presented by rods or plates of strong neutron absorbers (such as boron, cadmium, hafnium, gadolinium, europium or combination of these materials with grey absorbers), which can be inserted into or withdrawn from the core.

3.3 Impact of various factors on the CR parameters in the reactor BR2

According to the operation experiences of BR2 confirmed by MCNPX calculations, many factors affect the control rod total worth [1]. The burn up of cadmium reduces the total control rod worth by ~ 0.5 $ (360 pcm) and strongly affects the shapes of total and differential CR worth. The poisoning of the beryllium represented by both helium – 3 and Li – 6 absorption reduces the total control rod worth up to ~ 1.7 $ (1224 pcm) compared to fresh beryllium matrix. The presence of strong absorbers in the core (experimental samples) and burnable poisons in the fuel (B_4C and Sm_2O_3) increases the total control rod worth up to ~ 0.8 ÷ 1.0 $ (576 ÷720 pcm). The accumulation of fission products in the depleted fuel increases the total control rod worth up to about ~ 1.0 $ (720 pcm). The maximum worth has control rod with fresh beryllium follower; minimum worth has control rod with light water or Al follower and medium control rod worth is for poisoned beryllium follower. Accounting for photo neutrons reduces the control rod worth by about 10%. Increasing the aluminum cladding around the absorbing control rod material reduces the total control rod worth up to ~ 1.2 $ (864 pcm). The location of CR close to the core centre increases the total rods worth up to about ~1.8 $ (1296 pcm).

3.4 Criteria for control rod life

A criterion for the changing of the absorption properties of the CR is the behavior of macroscopic absorption cross sections of the CR material during long term of irradiation. This criterion gives an idea for the depletion of the rod material and changing of the local rod absorption properties, but it doesn't draw the actual behavior of the CR in the union of the whole reactor core. The fraction of the CR absorption at typical critical position (Sh~500 mm) is about 6-8% from the total absorption in the reactor (this value depends on the reactor load and on the critical height at BOC). Therefore additional criterion related to the behavior of the $k_{eff} = f(A/F)$, i.e. the change of the macroscopic absorption (A) and fission (F) processes in the whole reactor core due to insertion of a given type CR has been determined [1].

3.5 Control rod modifications

Considered have been several type CR with geometry and design as for the Reference CR: ***Cd+Co*** – Reference CR with cadmium and cobalt in the lower active absorbing rod part with Al cladding; ***Cd+Cd*** – CR with full length of cadmium with Al cladding; ***Cd+AISI304*** – CR with cadmium and AISI304 in the lower active part; ***Hf+Hf*** – CR with full length of hafnium with and without Al cladding; ***Hf+AISI304*** – CR with hafnium and AISI304 in the lower active part, without cladding; CR with full length of ***stellite***; $\boldsymbol{Eu_2O_3}$ and $\boldsymbol{Gd_2O_3}$ – CR with full length of europium or gadolinium oxide without cladding. The thickness of the absorbing material for all types is 5 mm; the outer diameter is 61 mm and the inner diameter - 51 mm. The full length of the absorbing part, including the use of grey material in the lower part is 895 mm. The length of the section with grey material in the lower part is 140 mm and the thickness is equal to the thickness of the absorbing material. 6 identical CRs are located in the same channels, occupied by the Reference CR.

4 ANALYSIS OF THE CALCULATION RESULTS

4.1 Calculation methodology for the burn up of the CR material

The burn up of the CR absorbing material is evaluated during ~ 33 consecutive operating cycles, each about 30 days long, which is equivalent to ~ 1000 EFPD of reactor operation. A typical BR2 reactor core load, which remains the same in each cycle, is used in the calculations. The following calculation methodology is applied: 6 CR with fresh absorbing material are loaded into the BR2 core at beginning of the 1^{st} operating cycle, BOC1. The densities of the CR material at EOC1 are used as initial densities at the BOC2, etc. up to the 33^{rd} cycle.

4.2 Comparison of macroscopic absorption cross sections during 1000 EFPD

The comparison of the macroscopic absorption cross sections for different CR types is given in Fig. 1. The total macroscopic cross sections remain almost constant for all considered rods during sufficiently long time of irradiation T ~ 650 EFPD. After T ~ 650 EFPD the macroscopic cross sections for cadmium rods drastically decrease due to the rapid burn up of the dominant isotope ^{113}Cd. The macroscopic cross sections for hafnium and europium rods and combination of these absorbers with stainless steel in the lower rod part, remain almost constant up to ~ 1000 EFPD. The absolute values of the macroscopic cross sections for fresh and burnt CR material are summarized in Table I.

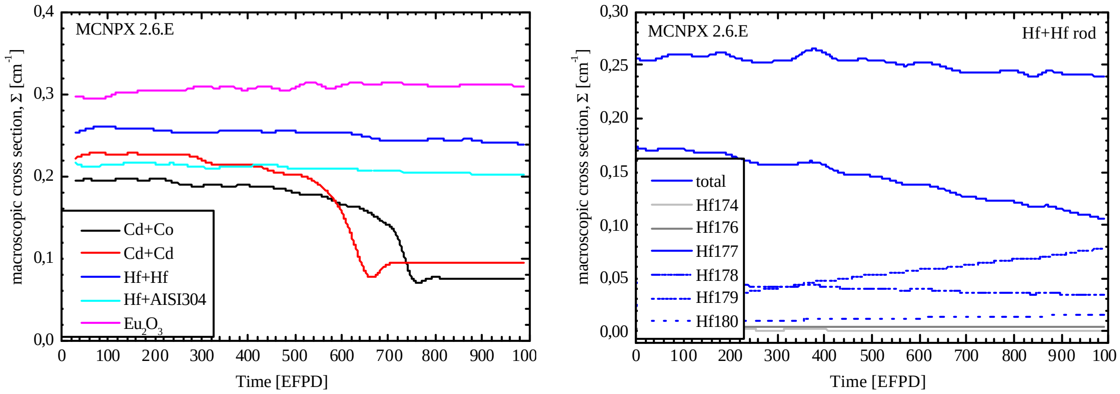

Figure 1. Comparison of effective macroscopic absorption cross sections for various CR types.

Table I. Calculated by MCNPX 2.6.E macroscopic absorption cross sections Σ [cm^{-1}] in fresh and burnt CR material.

	Cd+Co	Cd+Cd	Hf+AISI	Hf	Eu_2O_3	Gd_2O_3
T=0	0.20	0.23	0.22	0.27	0.30	0.28
T=600 EFPD	0.16	0.08	0.21	0.25	0.31	0.26
T=1000 EFPD	0.08	0.10	0.20	0.24	0.31	–

4.3 Comparison of microscopic absorption cross sections during 1000 EFPD

The macroscopic cross is defined as:

$$\Sigma = N < s >_{eff}, \left[cm^{-1} \right] \tag{2}$$

Where, Σ is function of two variables: the atomic density N and the effective microscopic cross section $< s >_{eff}$, which is defined as:

$$< s >_{eff} = \frac{\int s(E)\Phi(E)dE}{\int \Phi(E)dE} \tag{3}$$

The evolution of the atomic densities of the dominant nuclides and the microscopic effective cross sections, defined with Eq. (3) are given at Fig. 2. The effective microscopic cross section of ^{177}Hf remains almost constant during the irradiation (the decrease of the density is compensated by production of the other Hf – isotopes), while for cadmium the microscopic cross section increases during the first ~ 10 to 20 cycles. However, the product of the atomic density and the effective microscopic cross section, i.e. the macroscopic effective cross section Σ remains constant during long irradiation time ~ 650 EFPD for the both Cd and Hf rods. Detailed explanations of these effects can be found in [1].

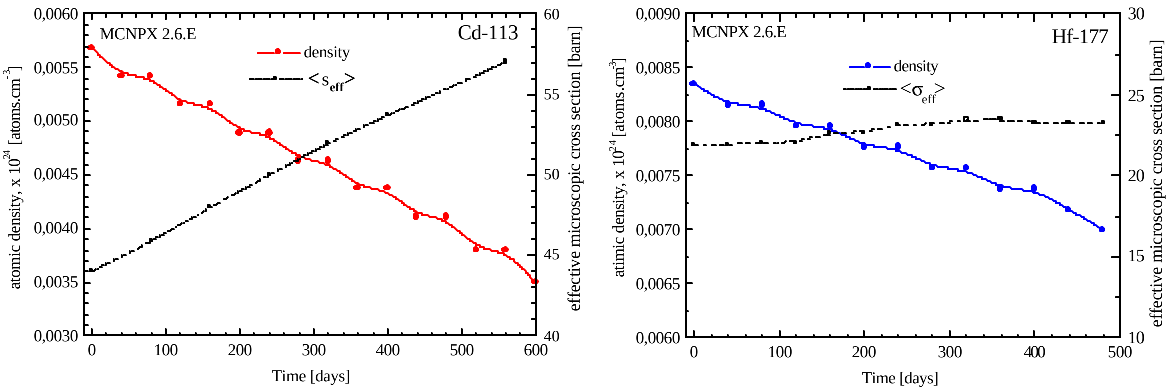

Figure 2. Evolutions of microscopic effective absorption cross section and atomic density of dominant isotopes in cadmium and in hafnium.

4.4 Comparison of neutron spectra in the lower part of CR during 1000 EFPD

The changing of the spectrum in the lower part of the depleted rod has been investigated during long irradiation period for T ~ 1000 EFPD. The comparison of the neutron spectra for different types CR at T=0 and T=1000 EFPD is given in Fig. 3. It is seen that for cadmium rods the thermal fluxes increase drastically after ~ 600 EFPD which is related with the complete depletion of ^{113}Cd, having high thermal absorption cross section.

4.5 Comparison of activity and nuclear heating in CR

Detailed calculations of the activity in the different CR types have been performed using MCNPX 2.6.E [1]. For all CR types the major contribution into the activity comes from the lower part of the rod, being exposed to the maximum thermal neutron flux. For the Reference Cd+Co rod the dominant nuclide is ^{60}Co. The dominant nuclides in the activity of Eu-rods are ^{152}Eu and ^{153}Sm. For Hf+Hf rod the major contributor to the activity is ^{181}Hf. The nuclear heating in the lower part of cadmium CR types was calculated using MCNPX [1]. The main contributions to the total heating in cadmium give neutrons (more than 90%). The contribution from prompt and captured γ-rays Q^{pr+cap}_{γ} into the total heating is about 12% to 14% from the heating Q_n, caused by neutrons. Additional contributions into the total heating come from the delayed γ-rays Q^{del}_{γ}.

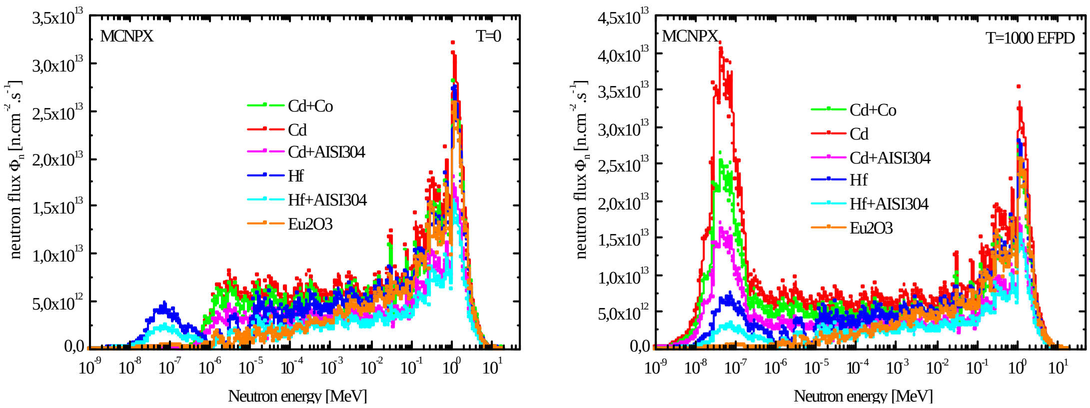

Figure 3. Neutron spectra in the lower active part of the CR.

4.6 Comparison of control rod worth

Detailed calculations of the rod worth for various CR types have been performed by MCNPX for the beginning of the control rod life and after long time of irradiation, taking into account the detailed axial burn up of the CR absorbing material. The total worth for the different types CR is compared at BOC and EOC of the 1st and the 30th operation cycle and presented in Table II.

Table II. Comparison of total worth (in units of $ and pcm in the brackets) for different control rod types accounting for the axial burn up of the absorbing material during irradiation.

CR type	BOC		EOC	
	T=0 (1st cycle)	T=1000 EFPD (~ 30th cycle)	T=30 EFPD (1st cycle)	T=1030 EFPD (~ 30th cycle)
Cd+Co	13.4 $ (9648 pcm)	13.0 $ (9360 pcm)	14.5 $	14.0 $
Cd+Cd	13.6 $ (9792 pcm)	13.3 $ (9576 pcm)	14.7 $	14.3 $
Cd+AISI304	13.2 $ (9504 pcm)	12.9 $ (9288 pcm)	14.3 $	13.9 $
Hf+Hf	15.8 $ (11376 pcm)	15.7 $ (11304 pcm)	17.1 $	17.0 $
Hf+AISI304	15.6 $ (11232 pcm)	15.5 $ (11160 pcm)	16.7 $	16.7 $
Eu_2O_3	17.5 $ (12600 pcm)	17.5 $ (12600 pcm)	19.0 $	18.8 $

The curves of the total control rods worth for fresh CR at T=0 and for burnt CR at T~1000 EFPD are given in Fig. 4 The curve of the total worth of control rods with full cadmium length decreases significantly after ~ 650 EFPD of irradiation. The curves of total worth for Cd+AISI304 rod and for Reference Cd+Co rod also decrease during irradiation, but less. For all other CR types Hf+AISI304, Hf+Hf, Eu_2O_3 the changing of the curves of total CR worth during T ~1000 EFPD is negligible.

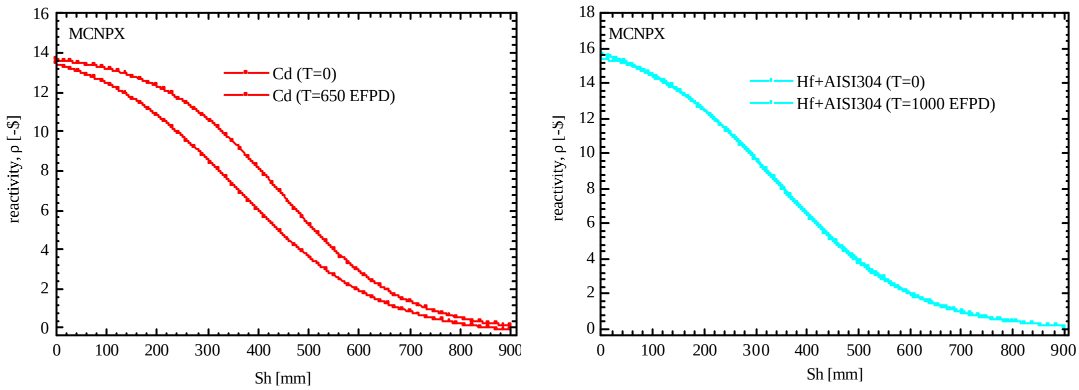

Figure 4. Comparison of total CR worth $R0 = r(0) - r(900mm)$ for fresh (T=0) and burnt (T ~ 1000 EFPD) absorbing material in cadmium rod and in hafnium rod.

The comparison of the differential CR worth for fresh and burnt CR material in different CR types is given at Fig. 5. The burn up of the CR material affects strongly the curve of differential worth for cadmium rod, which maximum is reduced and shifted to the lower rod positions Sh. The differential worth for hafnium and europium rods almost does not change with burn up of the CR material. Prolonging the black absorber with a grey material shifts the maximum of the differential worth curve to the lower positions Sh of the CR motion. A compromised decision can be found reducing the length of the grey material which will be used when choosing the optimal rod design.

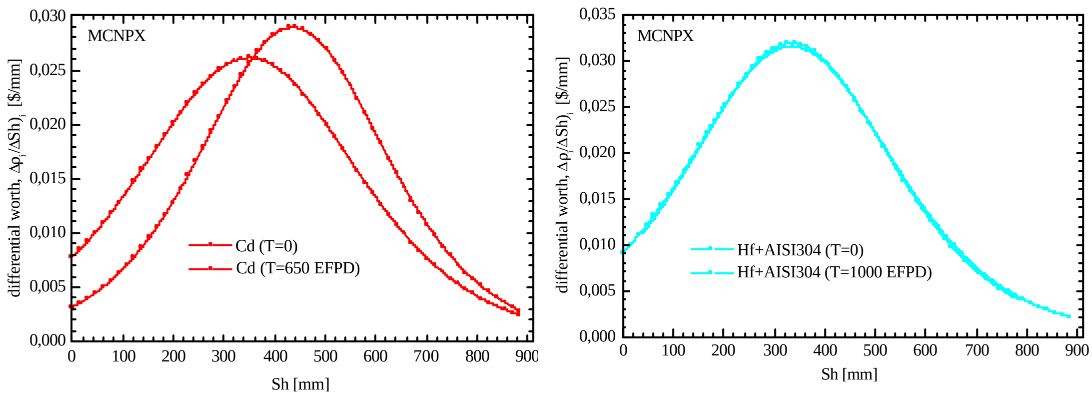

Figure 5 Comparison of differential worth $\Delta r_i / \Delta Sh_i$ in fresh (T=0) and burnt (T~1000 EFPD) absorbing material in cadmium and in hafnium rod.

4.7 Comparison of reactor neutronics characteristics for different CR types during BR2 fuel cycle

Accurate criticality calculations have been performed by MCNPX for loaded different CR type in the reactor core, using same fuel load (see Table III). The positions of the CR with burnt cadmium will be lower than the position with fresh cadmium due to the burn up of the lower active rod edge and reduction of the cadmium length. The maximum decrease of the position Sh at BOC after long term of irradiation has the cadmium rod (about 70 mm), the decrease of Sh for Cd+Co rod is less (~ 40 mm). For all other rods (hafnium, europium, gadolinium) the position Sh at BOC remains practically constant during many operating cycles. The criticality variations for few CR types during long term of irradiation are shown in Fig 6. After the 15th cycle the tendencies of the k_{eff} for cadmium rods are quite different from those for the other CR types. Close to the 25th operating cycle, the values of the k_{eff} for cadmium rods increase very rapidly, that is related to the depletion of ^{113}Cd. The tendencies of k_{eff} for the rest CR types remain almost same up to the 25th cycle [1].

Table III. Comparison of the positions Sh [mm] at criticality for different CR types at T=0 (loaded fresh CR absorbing material) and for burnt CR absorbing material after ~ 25 operation cycles equivalent to ~750 EFPD. Calculations by MCNPX.

Rod type	T=0	T=750 EFPD
Cd+Co	440 mm	404 mm
Cd+Cd	500 mm	429 mm
Hf+Hf	535 mm	522 mm
Hf+AISI304	450 mm	450 mm
Eu2O3	555 mm	554 mm

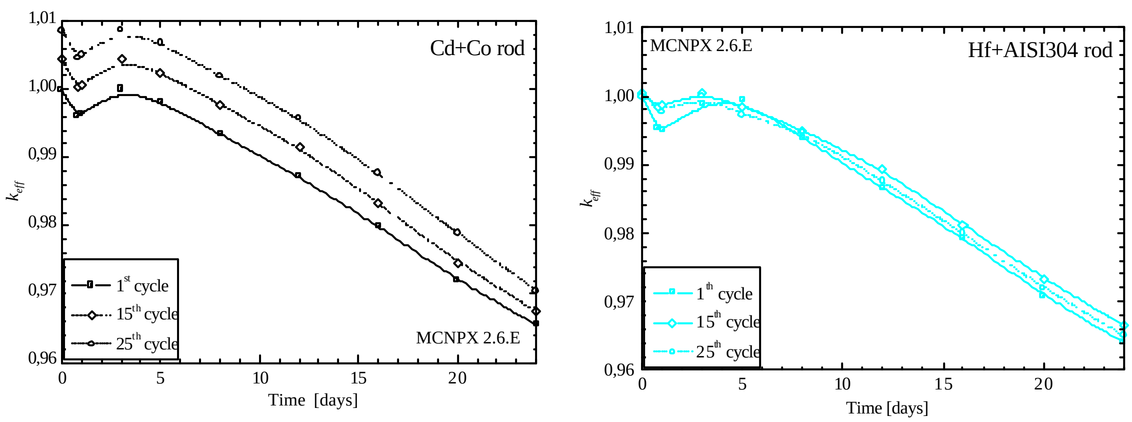

Figure 6. Evolutions of k_{eff} for Reference CR and for Hf+AISI304 rod during long irradiation time T ~ 750 EFPD (the position Sh for a given CR type is kept constant during the whole irradiation period).

4.8 Control rod effects on neutron flux distributions

The strongly absorbing nature of the control elements causes major perturbations of the neutron flux in the vicinity of the control rod and also affects the overall flux and power distribution of the reactor core. The detailed axial distributions of thermal, epi-thermal and fast neutron fluxes in the axis of typical fuel and reflector channels have been calculated in function of CR position Sh and successively increased produced energy during a typical operation cycle [1]. The strongest perturbations of the axial neutron flux distributions are observed in reflector and fuel channels, located near to the channels of CR location and also in the axis of fuel channels in the central crown. The perturbation of the flux distributions in the axis of the FE in the central channel H1/C is lower. Maximum perturbation effects on axial flux distributions have rods with full absorbing length from europium and cadmium. The axial distributions of neutron fluxes for CR rods, composed as a combination of grey material and black absorber are similar to those for the Reference CR. Maximum perturbation effect on the axial neutron flux distributions is observed at lower positions of the CR (Sh=400 to 500 mm). For high positions Sh, the differences in the axial neutron flux distributions for the various CR types practically disappear at Sh > 700 mm.

5 SUMMARY

Optimization studies for choice of new CR type of the BR2 reactor are presented. The changing of the rod absorption properties is evaluated during 33 consecutive operating BR2 cycles, which is equivalent to ~1000 EFPD. The calculations of the fuel depletion and the depletion of the CR absorbing material are performed using MCNP&ORIGENS combined method and MCNPX 2.6.E. The maximum worth has the Eu - rod (17.5 $ or 12600 pcm) and minimum worth has Cd - rod (~ 13.5 $ or 9720 pcm). The Total CR worth increases at EOC for all CR types by about 8% due to the depletion of ^{235}U and accumulation of fission products. The burn-up of the absorbing material affects the total CR worth of cadmium rods by about 3% - 4% and less than 1% – the worth of hafnium rod. The burn-up of the absorbing material affects strongly the shape of the curves of total and differential CR worth for cadmium rods. For hafnium and europium rods the total and differential worth curves do not change during irradiation. The maximum of the differential curves of CR made from a combination of black and grey absorber is shifted to the lower positions of the CR due to the reduced length of the black material. The macroscopic absorption cross section Σ_a [cm^{-1}] for cadmium rod remains almost constant during ~ 650 EFPD. For T ~ 650 - 1000 EFPD Σ_a ~ 20% from the initial value (^{113}Cd is totally burnt, residual Σ_a is due to other Cd – isotopes). Σ_a remains constant till ~ 1000 EFPD for all other rods – Hf, Hf+AISI304, Eu_2O_3. The maximum value of the activity is for Cd+Co rod and minimum for Hf+AISI304 rod. CR types with full length of Eu_2O_3 depress more strongly the axial distributions of neutron fluxes: this effect is sensitive at Sh ~ 400 to 500 mm. For Sh > 650 mm the axial distributions of neutron fluxes for all CR types are the same. The thermal fluxes in cadmium rods increase strongly after T ~ 650 EFPD (due to burn up of absorbing material). For hafnium and europium rods the change of the neutron spectra in the whole energy region is not significant during long irradiation time T ~1000 EFPD.

6 PROPOSED NEW CR TYPE: *HF+AISI304*

Analyzing the behaviour of the considered absorbing materials during long irradiation time, an optimal design is given for the ***Hf+AISI304*** rod as a new CR type. The hafnium rod almost does not burn during long time of irradiation (the main isotope Hf-177 is depleted, but this is compensated by production of other Hf-isotopes). The insertion of rod with full length of hafnium increases the total control rod worth and improves the differential worth in comparison with the

Reference Rod. The Hf+AISI304 rod has total worth equal to 15.6$ (or 11232 pcm) , which is smaller than for the Eu - rod (17.5 $ or 12600 pcm), but bigger than for Cd - rods (~ 13.5 $ or 9720 pcm). Although europium isotopes have the best absorption properties and also europium rod will not burn during very long time of irradiation, the europium rod has not been chosen as candidate for the new BR2 rod, because of the high fabrication costs. The hafnium rod is easier to be fabricated than cadmium one, because Al cladding is not needed. The hafnium rod will be heavier than the cadmium rod due to the higher atomic mass. However the preliminary simulation tests performed at the BR2 reactor have shown that the time for scram will be equivalent for the both rods. Several optimization modifications have been made for the hafnium rod, which are summarized in Table IV. The primary neutronics evaluations were made with geometry model and dimensions as for the Reference CR (N°1 in Table IV) with full length of hafnium (N°2) and for hafnium rod prolonged with stainless steel AISI304 in the lower active part (N°3). Detailed burn up calculations up to ~ 1000 EFPD have been performed by MCNPX 2.6.E for the lower rod part and it was obtained that the stainless steel is not burning. The axial distributions of the neutron fluxes in typical fuel and reflector channels are depressed for the Hf+Hf rod by ~ 10% compared to the Reference CR (see Fig. 7). This can be improved prolonging the lower part of the hafnium rod by stainless steel with length L=140mm. However, the application of stainless steel in the lower rod part shifts the maximum of the differential worth curve toward lower positions Sh. The reduction of the AISI304 length from L=140mm to L=70mm significantly improves the curve of the differential CR worth which can be seen from Fig. 8 and slightly worsens the axial distributions of neutron fluxes. Increasing the thickness of AISI from δ(AISI)=5mm to δ(AISI)=10mm improves the differential CR worth almost for all positions of Sh (see Fig. 8) and does not change significantly the axial distributions of the neutron fluxes (thermal and fast). The final optimized dimensions for the new CR type are highlighted in red color in Table IV.

Table IV. Optimization dimensions of the new CR type for the BR2 reactor.

N°	Rod type	Black absorber			Grey absorber		
		D_{out}	D_{in}	L	D_{out}	D_{in}	L
1	Cd+Co	61	51	755	61	51	140
2	Hf+Hf	61	51	895	0	0	0
3	Hf+AISI304	61	51	755	61	51	140
4	Hf+AISI304	63	53	755	63	53	140
5	Hf+AISI304	64	54	755	64	54	140
6	Hf+AISI304	64	52	755	64	52	140
7	Hf+AISI304	64	54	825	64	54	70
8	Hf+AISI304	64	54	825	64	44	70
9	Hf+AISI304	64	54	825	64	50	70

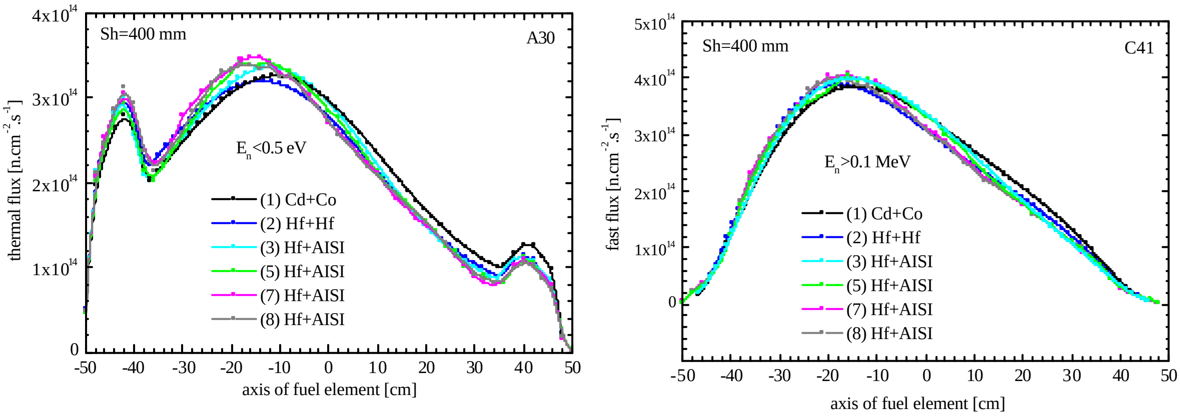

Figure 7. Comparison of axial distributions of thermal and fast fluxes in typical fuel channels for different optimization dimensions of the Hf+AISI304 rod, given in Table IV.

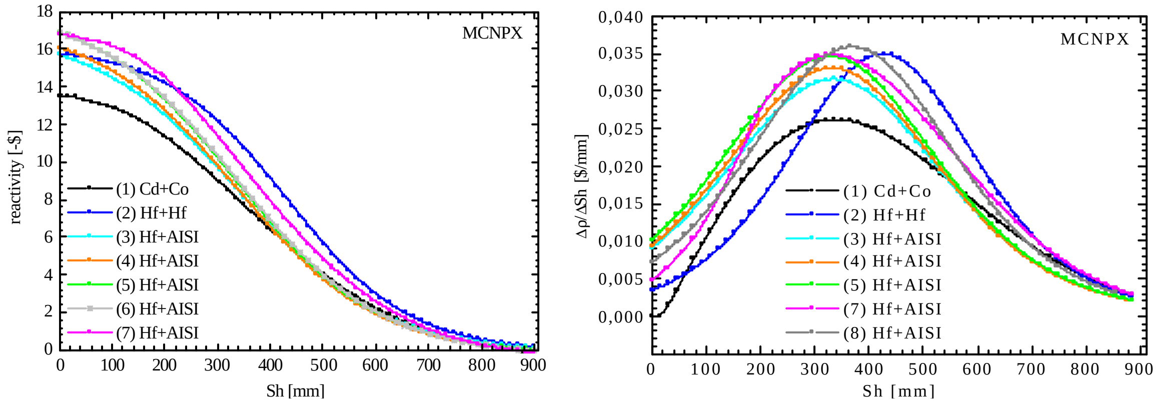

Figure 8. Comparison of total and differential worth for different dimensions of Hf+AISI304 rod, given in Table IV.

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STEP REACTIVITY TRANSIENT THERMAL HYDRAULIC AND SAFETY ANALYSES OF A PROPOSED HEU & LEU CORE FOR SAFARI-1 RESEARCH REACTOR

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ABSTRACT

The SAFARI-1 research reactor is a tank-in-pool type reactor operated at a nominal core power of 20MW and single phase liquid water with nominal temperature not exceeding 100°C.
The application of the Best-Estimate methods constitutes a real necessity in order to get a more realistic vision of the system behaviour. The aim of the current work is an in-depth study and safety analysis of the conversion process of the SAFARI-1 research reactor from HEU to LEU using the System thermal-hydraulic RELAP/SCDAP Mod 3.2 code. A set of transients was analysed during the process of the conversion. In this paper the fast transient related to a reactivity induced transient was analysed and the Step Reactivity Insertion was chosen as example. The Onset of Nucleate Boiling Temperature was predicted and results from different core configuration were compared (0% LEU, 54% LEU and 100% LEU). Differences between different core configuration results are then emphasized and discussed.

1. Introduction

The conversion of the SAFARI-1 research reactor from HEU to LEU requires an in-depth safety analysis to see the behaviour of the reactor during the conversion process. The safety analyses presented in this work is part of the global deterministic study done to determine the behaviour of the reactor during a set of transients. Such transients can happen during the operating of the reactor in different modes (low power, medium power, normal power). In this study we considered different configurations of the core - 24, 26, 28 and 32 fuel assemblies - to have a global analysis of present operation of the reactor and possible future geometries. The best estimate code RELAP/SCDAPMod3.2 [1] was used for comparison with a previous study to validate our RELAP5 model against some measured parameters in the reactor. The modelling of the main features of the thermal-hydraulic design of the reactor and the information regarding the integrity of the fuel cladding, and hence of the fuel as a whole, is maintained for all operational and anticipated abnormal conditions. The requirements of IAEA

Safety Series NS-R-4 [2], SS-35-G1 [3], and SS-35-G2 [4] form the basis of the safety evaluation of SAFARI-1 in this paper.
The objective of this analysis is to demonstrate that the proposed design meets safety and licensing requirements and the safety design criteria. The quantitative analyses have been performed with computer code RELAP/SCDAP and all the assumptions made are conservative. The numerical calculations show that the reactor goes through a series of safe states following the occurrence of Design Basis Initiating Events. The description and analysis of each Design Basis Initiating Event and event sequence is presented.

2. Reactor and modelling characteristics

The aim of the safety analyses covered in this work is to deterministically verify the operational safety of the reactor for a wide range of operating condtions and to provide input to the Probability Risk Assessment. The operating conditions include a range of core loadings and operating powers. Several operating modes, defining the maximum core thermal power for the number of operating shut down or primary coolant pumps, are identified. In addition, a range of core loadings from 24 to 32 standard SAFARI-1 fuel elements (SSFEs), identified as a realistic range to meet current and future needs, is included in the analyses of all initiating events. This is accomplished by modelling the core at four points in this range (namely 24, 26, 28 and 32 SSFEs), and confirming that the behaviour of intermediate core loadings can be deduced by interpolation to an acceptable degree of accuracy. Inclusion of the 26 and 28 SSFE cores, which are historically the core loadings for which the most information is currently available, support such an approach. All core loadings are modelled with six control rods.

2.1 Core parameters

The nominal operational characteristics, which remain valid for all equilibrium core loadings within the above range, are given in Table 1. Some of the parameters listed in the table are supplied as input to RELAP/SCDAP.

Table 1: Nominal Operational Characteristics

Parameter	1-Pump Mode	2-Pump Mode
Core power (MW)	0 – 10	0 – 20
Core inlet temperature (°C)	40.0	40.0
Core ?T at maximum power (°C)	4.30	5.50
Peak fuel clad temp.[1] (°C) 24 SSFE	94.3	115.25
26 SSFE	84.2	103.48
28 SSFE	82.7	99.75
32 SSFE	88.0	107.35
Coolant inlet pressure in core (kPa_{abs})	188-196	218-240
Clad temp. for ONB, T_{ONB} (°C)	128±1	131±1
Core flow rate 24 SSFE	1905; 3.9	3006; 6.1
and velocity (m^3/h; m/s) 26 SSFE	1911; 3.5	3033; 5.9
28 SSFE	1943; 3.4	3062; 5.6
32 SSFE	2002; 3.2	3104; 5.1

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Power distr. factors - axial; radial; total	1.64; 2.13; 3.50	1.64; 2.13; 3.50
Engineering hot spot factor Fully Applicable to RELAP5 model	1.34 1.19	1.34 1.19
Control rod bank worth ($) all cores (Represents minimum allowable)	20.0	20.0
Control rod insertion delay[2] (ms)	600	600
Control rod insertion time[3] (ms)	500	500

2.2 Reactor core neutronics model

Reactivity coefficients as function of moderator temperature, moderator density and fuel temperature have been calculated at BOC for the various different fuel types and given in Figures 1 and 2. Three-dimensional core calculations were performed for this purpose using the in-house OSCAR3 code System [5]. The calculation of the effective delayed neutron fraction and prompt neutron generation time at BOC was performed using MCNP [6], making use of the OSMINT [7] (Oscar-3 MCNP INTerface) code to transfer isotopic inventory data. The prompt approximation method was used, but the results show some inconsistent behaviour in prompt neutron generation time; therefore we used the results from the previous calculations reference [8] to have in-depth analysis of the MCNP result. The results of the calculation are shown in Table 2.

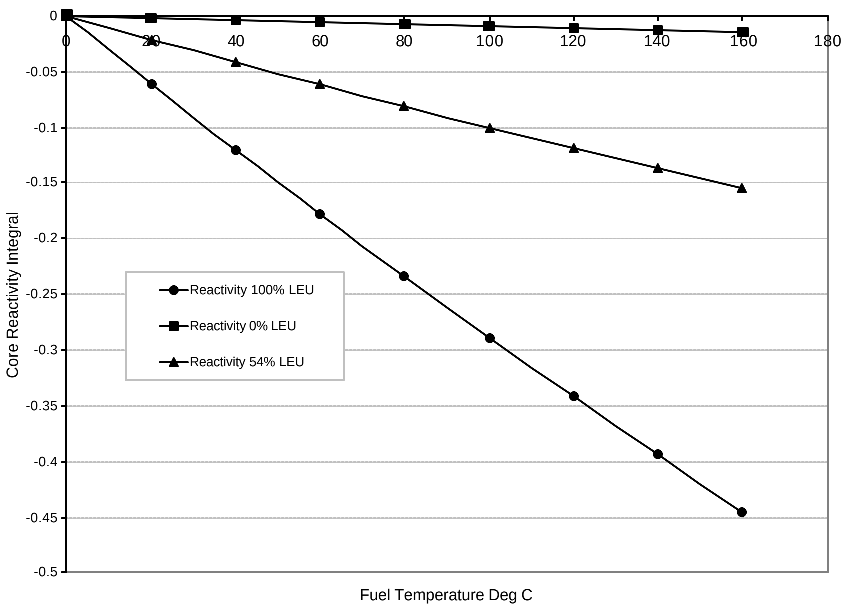

Fig 1 Doppler feedback

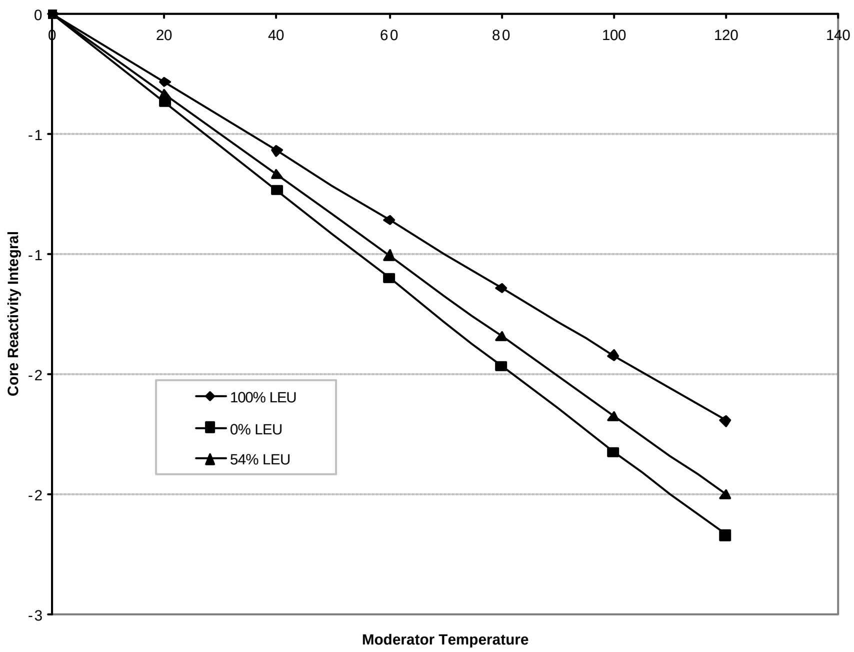

Fig 2 Moderator Temperature ^{o}C

Table 2: Delayed Neutron Fraction and Prompt Neutron Generation Time

Fuel Description	Delay Neutron Fraction	Prompt Neutron generation time (x 10^{-6} s)
HEU 300 g 19 plates	0.00750	57.9
MEU 225 g 19 plates	0.00742	60.5
LEU 340 g 19 plates	0.00717	48.7

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3. Transient analysis

A sudden or stepped or uncontrolled reactivity insertion resulting in adverse transient conditions can arise from one of the following events:

- Insertion or removal of an experiment while the reactor is at power (e.g. by the hydraulic rabbits or thimble facilities).
- Displacement of voids in the core by water. Since the only way of introducing or ejecting voids into/out of the core is by means of an experiment, this is the same as the above
- Displacement of an improperly secured experiment
- Movement of a fuel assembly, due to it having been improperly seated during a reload.
- Sudden movement of a control rod, e.g. sticky or jerky movement during withdrawal.

The generally accepted practice in experimental use of research reactors is to limit the reactor period to a minimum of 10s, since this corresponds with the response time of a typical automatic power control system such as that of SAFARI-1. By the well known In-hour formula this resolves into a reactivity worth of $0.21 for the SAFARI-1 core, which is independent of core loading.

A reactivity worth of $0.21 therefore forms an upper operational bound on sudden changes to be expected in the core and is used as the basis for the analyses in this section. However, it is pointed out that, although $0.21 will not lead to a safety action on reactor period, the power increase is generally rapid enough, especially if not limited by the auto controller, to cause a safety action on over power.

Unless otherwise stated, the reactivity insertions are considered to take place instantaneously, or rather, within the transient advancement time step used by RELAP/SCDAP, which is consistently set at 0.05 s. It is pointed out that a truly instantaneous insertion is impossible. All real uncontrolled reactivity insertions take place over some small but finite time, which can be calculated.

3.1 Results and discussion

For this paper we compared the worst case which is the maximum reactivity insertion that will not lead to SCRAM in each case (LEU, MEU and HEU). In all cases, the 24 SSFE core at a power of 20 MW produces the highest fuel clad temperatures, this is clearly because it has the highest power loading per fuel element. In this transient it was assumed an over power trip at 120% power. A series of three plots is presented in Figures 3 to 5 which trace the reactivity, total power and the peak clad temperature.

The increase in reactor power causes a Reverse (Driven SCRAM) on overpower in each case. The highest peak fuel clad temperature occurs in the 2-pump (20 MW) case, at 127°C, below T_{ONB} by 5.04°C. The minimum DNBR in this case is 4.00, which is greater than the minimum specified in design basis which is 2. These results clearly show that no conditions develop that may lead to fuel damage.

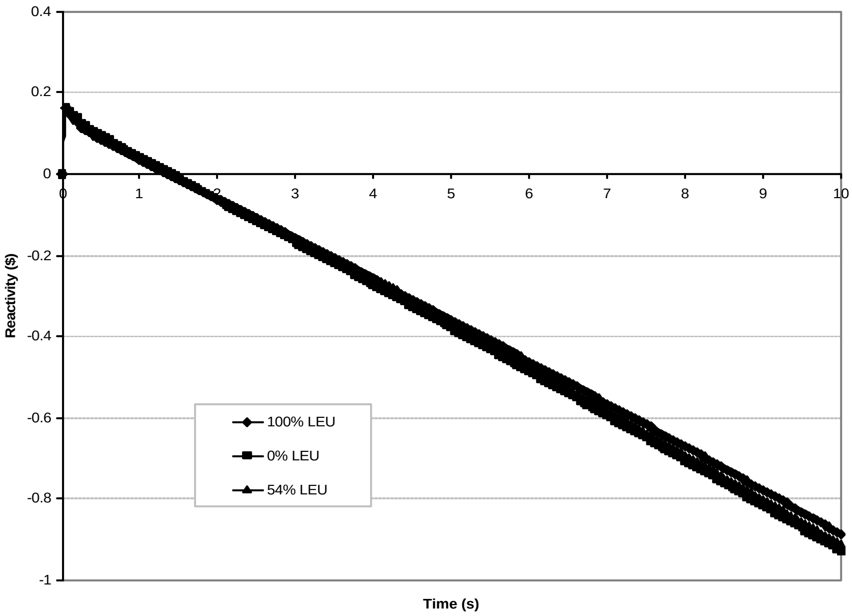

Fig 3 Reactivity ($)

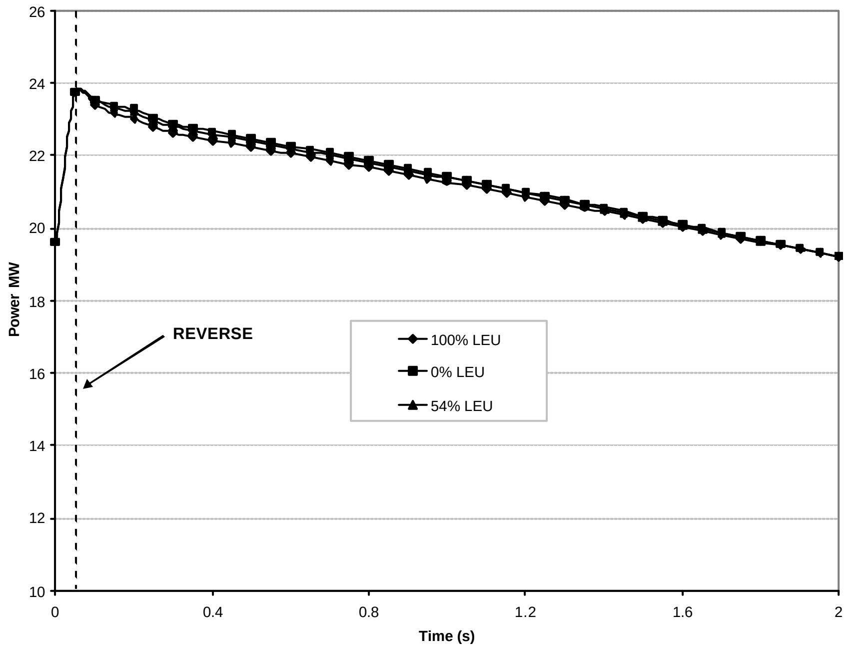

Fig 4 Total power (MW)

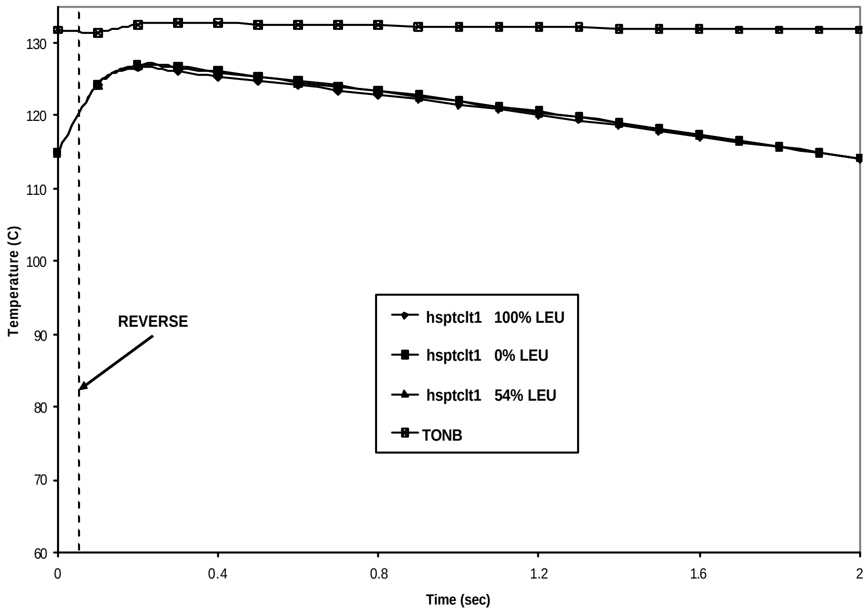

Fig 5 Hot spot temperature (℃)

4. Conclusion

The SAFARI-1 reactor was safely shut down without coolant boiling and with hot spot clad temperature below TONB by a margin of 5 ^{0}C according to design basis, after step reactivity insertion transient modelled with RELAP5/SCDAP using the HEU, LEU or MEU core. The power histories and hot spot clad temperature of the three cores were very close during the transient. Therefore the use of the LEU, HEU or MEU core has no impact upon the results of the transient modelled in this paper

The total isothermal reactivity coefficient becomes more negative as the fuel enrichment is reduced for a fresh core. As each core depletes, the reactivity coefficients become more negative than in the fresh core.

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THE FIRST EUROPEAN FOCUSSING COLD NEUTRON SOURCE
- OPERATIONAL EXPERIENCE AND NEUTRONICS RESULTS -

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ABSTRACT

A cold neutron source is one of the most important components of a research reactor. For this reason GKSS installed a cold neutron source (CNS) at the FRG-1 in 1988. Around 60% of all neutron scattering instrumentations are using cold neutrons. Principal component of this CNS is the moderator chamber shaped like a discus. The moderator is supercritical gaseous hydrogen. In order to increase the yield of cold neutrons, a study was made for a new layout of the moderator chamber in 2003. The new fundamental design of the moderator chamber is based on a hemispherical shape, thereby increasing the cold neutron flux by approx. 60% with the use of focusing effects. The study of all relevant parameters was done by AREVA NP early 2006. The licensing procedure, the fabrication, exchange of the moderator chamber, installation and successful set in operation program took from May 2006 to June 2007 including the participants of the independent experts.

1. Introduction

Long wavelength (cold) neutrons with high intensity are indispensable probe for the study of the microstructure and dynamics of condensed matter. These are necessary for its macroscopic characterization in applied as well in basic for example in material-, biological- and polymer research. With the existent CNS the number of long wavelength neutrons with wavelength > 0.4 nm were increased by a factor of more than 20 compare to the thermal flux. For a further increase of the important cold neutron flux, which feed more than half of the neutron scattering instrumentations, the moderator chamber of an existing spare unit should be replaced by a new one. Model of the new layout were the focusing moderator chambers of the American research reactors MURR and ORNL. These new moderator chambers resulted in gain factors between 50 to 150%.
The following conditions formed the basis for the design and licensing procedure of the GKSS moderator chamber:

- ➔ Simple design (hemispherical shape) and fabrication
- ➔ The same material specification for the new moderator chamber as for the existing one
- ➔ The same technical inspection as for the existing one
- ➔ The same incident conditions (pressure, melting etc.) as for the existing one
- ➔ Comparable nuclear heating for the new and exiting chamber

The consideration of all of these conditions led to a brisk licensing procedure of only 4 month.

2. Optimisation studies of the new moderator chamber

The Research Reactor FRG-1 is operated with a reactor core of 12 fuel elements in a 3x4 matrix arrangement. At three sides this core is surrounded by Beryllium reflector elements, the fourth side faces a block reflector of Beryllium with several holes containing the tops of the azimuthally arranged beam tubes SR6 to SR9.
The cold neutron source (CNS) is installed inside beam tube SR8 just a few millimetres outside the core outer boundary. Main parts of the CNS are a cylindrical vacuum chamber (AlMg3) arranged inside the beam tube SR8 filled with helium and a moderator chamber inside the vacuum chamber with the shape of a discus (Fig. 1).

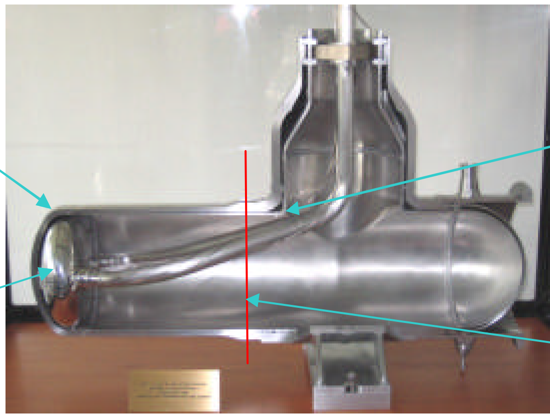

Fig. 1: Inpile section of cold neutron source FRG-1, longitudinal cut through a prototype

The moderator chamber is part of a cold neutron source system operated with supercritical hydrogen at about 25 K and a pressure of 15 bar. The hydrogen serves as moderator for thermal neutrons and as coolant for the heat transport to the cryogenic helium refrigerator outside the reactor pool. The surrounding vacuum chamber provides a good thermal insulation to the beam tube and the reactor pool. The advantage of this medium at these operating conditions is to be always gaseous but with a density of about 90% of that of liquid hydrogen.
In the course of the FRG-1 core compaction in 1999 the complex geometry of core, Beryllium reflector, tangential beam tubes and cold neutron source was modelled with the Monte Carlo computer code MCNP [1]. This included the detailed consideration of each single fuel plate, all structure materials, coolant, Beryllium reflector around the core and all beam tubes. An example is shown in Fig. 2.

An evaluation of existing literature about focusing cold neutron sources [e.g. ref 2] together with the requirement for a simple geometry which had to fit into an existing spare part of the CNS lead to a basic geometry for the new moderator chamber consisting of two hemispherical shells with a cylindrical elongation at its core distant end. An important advantage of this geometry is the mechanical stability of sphere and cylinder with respect to the need of small wall thicknesses to reduce the heat generation in the structure material. The implementation of the cold neutron source into the MCNP model is shown in Fig. 3.

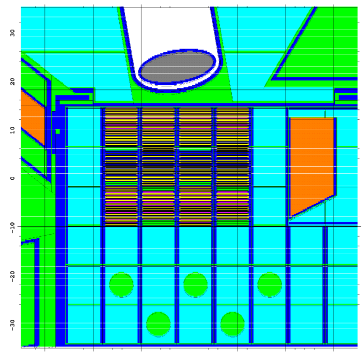

Fig. 2: Cross section through core, reflector and beam tubes

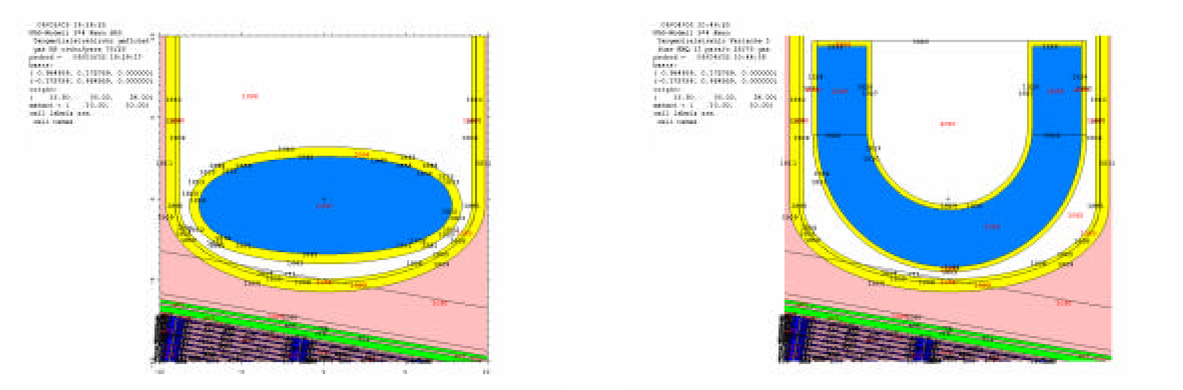

Fig. 3: MCNP model of moderator chamber for reference design (left figure) and optimised design

For optimisation of the geometry of the moderator chamber a sequence of calculations was performed with MCNP for one reference burn up configuration by variation of the moderator thickness and the length of the cylindrical part. The assessment of the results and the selection of an appropriate geometry of the moderator chamber was made considering only those neutrons which had a chance to pass the neutron guide and to reach the experimental set up outside the reactor pool. As a characteristic result Fig. 4 presents the calculated mean gain factors for all neutrons in the range of interest comparing both types of moderator chambers.

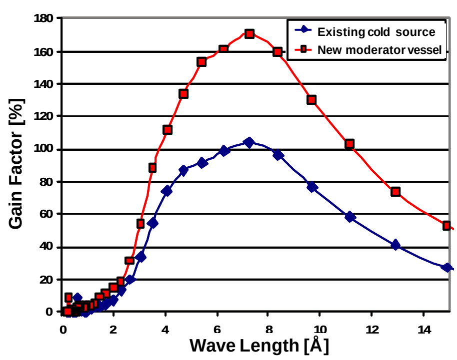

Fig. 4: Calculated gain factor

Fig. 5: Installation of the new chamber

3. Fabrication and installation

The course for the exchange of the old moderator chamber against the new one in the AREVA NP workshops in Erlangen was as follows (main steps):

- cutting off the top of beam tube SR8, vacuum chamber and hydrogen pipes at position indicated by red line in Fig. 2 (the radial gap between beam tube and vacuum chamber was only 0.15 mm),
- fabrication and installation of new moderator chamber (in progress at end of January 2007),
- re-installation of tops of vacuum chamber and beam tube SR8 respecting the original dimensional requirements (Fig 5),
- X-raying of welds and pressure tests (end of April 2007).

After transport of the new in-pile part to the FRG-1 the installation of the in-pile part was done by the operation team of FRG-1 at the end of May and the beginning of June 2007 (Fig. 6). An existing work instruction which was examined during the first installation 1988 has been applied for the installation.

4. Commissioning and validation

A part of the licensing procedure was the installation of a set in operation program. This program contains all steps from the inspection of the spare unit before beginning of the work up to the CNS operation during full reactor power. After the installation of the in-pile part in the reactor pool warm/cold leak tests are accomplished, before hydrogen is filled into the plant. The operating parameters (cooling power) of the CNS with the new focusing moderator chamber are then determined by means of a heater in the helium refrigerator. The most important proof of the CNS is the determination of the operating parameters during reactor operation. For this test the reactor was operated in different power ranges. These tests were accomplished successfully for the two operating conditions (standby operation T = -35°C; normal operation T = 25 K). The final point of the commissioning program was the release of the CNS for normal operation. A first measurement of the cold neutron gain factors at the experiment Nero yields appr. 40% more cold neutrons, which is in a good evidence to the MCNP calculations (Fig. 7).

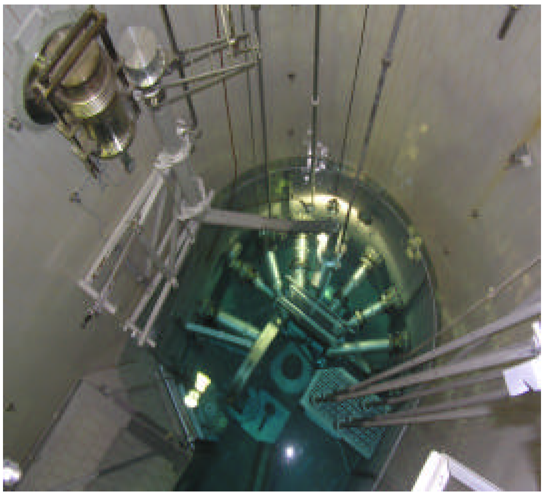

Fig.6: Installation of the CNS Inpile liner

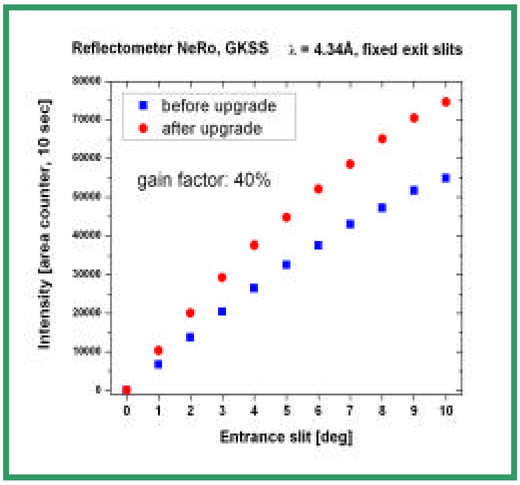

Fig. 7: Gain measurement at Nero

5. Summary

GKSS has already realized a continuous increase of the neutron flux by 2 core compactions and by the installation of the first elliptical CNS. The installation of the focussing moderator chamber is a new step for a further increase of the important cold neutron flux. With the additional gain of cold neutrons by approx. 60%, the FRG-1 results in an interesting middle flux neutron source available to the national and international user community.

6. References

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THE ENERGY RELEASE AND FUEL BURN-UP DETERMINATION METHODS IN THE MIR REACTOR

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ABSTRACT

To determine heat release and burnup of fuel in the driver and experimental channels of the MIR reactor, calculation and experimental techniques have been elaborated on the basis of the thermal balance method. These techniques account mutual exchange of photon and neutron emission between fuel channels as well as heat losses in the reactor cooling pool. Computer codes have been developed for the on-line determination and representation of heat power and burnup of the driver and experimental fuel assemblies (FAs) in the reactor data-measurement system. The distribution of fission rate and burnup over the FAs is calculated using a Monte-Carlo-based 3D code. The experimental determination of fission rate and burnup distribution of FAs as well as the burnup value of fuel elements is determined by the gamma-spectrometry method in the reactor hot cell. The paper presents the description and some peculiarities of methods used to determine heat power and burnup of driver and experimental fuel in the MIR reactor.

1. Introduction

The MIR reactor is a thermal heterogeneous reactor with a metal beryllium reflector and moderator. The core is arranged of hexagonal beryllium blocks 148.5mm in flat size located in a triangle grid with a 1.5mm gap between them. Channel bodies are installed in the central axial holes of the blocks to locate 49 working FAs and 11 experimental facilities. [1]. Each experimental channel is surrounded with six channels with driver FAs and 3÷5 control rods. By varying burnup of the working FAs during reloading and location of control rods around the experimental cells, it is possible to maintain simultaneously the testing conditions practically in all experimental channels. The number of experimental channels can vary from one reactor cycle to another. The experimental channels can be replaced with the driver ones. Several experimental FAs of various design, with different fissile material content, power, form and coolant can be tested in the different experimental channels of the reactor at the same time. The above peculiarities show that the core neutron-physical characteristics change in the wide range depending on the core arrangement and reactor operation modes. It should be mentioned that in different areas of reactor there could be FAs of different power that can vary in 10 times and more during some reactor cycles.
Another important feature of the MIR reactor is that the driver and experimental FAs are located in separate channels and the coolant is supplied separately in each channel. There are gages for pressure, temperature and flow rate control both on the supply and outlet tubes of the driver and experimental channels. So, heat power of the driver and experimental channels is determined by the thermal balance method (TBM) that became the basis of the reactor design. The method is based on the measurement of coolant thermo-dynamic parameters at the inlet and outlet channel pipelines.
To use TBM correctly in the MIR reactor, the mutual exchange of energy between the photon and neutron emission channels should be accounted as well as convection heat losses to the reactor pool. Up-to-date measurement and calculation system allow for on-line determination and registration of heat rate and burnup over the core section in the working and experimental channels.

2. Thermal balance method

A thermal balance equation can be presented as follows:

$dU = dQ + dA - dL$, where:

dU – change of the coolant of internal energy (enthalpy variation), determined by the measurement of coolant thermo-dynamic parameters at the channel inlet and outlet;

$dQ = dQ_{FA} + dQ_{struc}$ – heat amount released at the channel active part level due to fuel fission in the FA as well as neutron and photon reactor emission;

dA – heat released when the forced-circulation coolant loses its energy to overcome local resistance and friction forces in the channel;

dL – amount of heat removed to the reactor pool.

The amount of heat released in the fuel rods can be presented as follows:

$dQ_{FA} = dU + dL - dA - dQ_{struc}$

Having differentiated this equation with respect to time, we can have heat rate in the fuel rods:

$N_{FA} = \frac{dU}{dt} + N_{loss} - \frac{dA}{dt} - N_{struc}$, where:

N_{FA} – heat power of FA; $\frac{dU}{dt} = \Delta I \cdot G$ – change of internal energy per a unit of time;

g – variation of coolant enthalpy in the channel;

$\Delta I = f(\Delta T, P)$ – specific enthalpy variation; G – coolant flow rate through the channel;

$N_{loss} = k_1(T - T_p)^2 + k_2(T - T_p)$ – heat loss rate to the reactor pool in the area between temperature gages at the loop channel inlet and outlet;

T – average coolant temperature in the channel;

T_p – reactor pool water temperature;

k_1, k_2 – constants determined by processing the experimental results of heat losses versus coolant temperature gradient in the channel and reactor pool;

$\frac{dA}{dt} = \frac{\Delta P}{r} \cdot G$ – work done by coolant per a unit of time to overcome local resistance and friction forces;

ΔP – pressure change in the channel to overcome local resistance and friction forces;

r – coolant density.

$N_{struc} = N_{Rg,n} + N_{FAg,n}$ – heat rate in the channel structural materials and coolant due to photon and neutron (g- and n-) emissions of the reactor and FA itself.

$N_{Rg,n} = k_3 \overline{N}_{RFA}$ – radiation heat rate in the channel structural materials and coolant due to g- and n- emission of the reactor surrounding FA;

$\overline{N}_{RFA}$ – average heat power of six FAs surrounding the channel;

k_3 – constant determined either by a special reactor experiment and/or by calculation using the MCU-RR code.

$N_{FAg,n} = k_4 N_{FA}$ – radiation heat rate in the channel structural materials and coolant conditioned by FA g- and n- emission, where:

k_4 – constant determined by calculation using the MCU-RR code, N_{FA} – heat power of FA.

The total energy (fission power) rate of the FA fuel rods W due to fuel fission is determined by the following relation:

$W_{FA} = k_5 \cdot N_{FA} - k_6 \overline{N}_{RFA}$, where:

k_5 – calculated constant that accounts the energy carry-over of photon and neutron emission out of the channel that is determined by the calculation code,

k_6 – calculated constant that accounts heat rate in the FA due to photon and neutron (*g*- and *n*-) emissions of the reactor.

To evaluate the contribution of photon and neutron emission to heat rate in the MIR core components, a model was developed using the MCU-RR code. It implements an algorithm of a 3D co-modeling of photon and neutron lines by the Monte-Carlo method accounting an energy dependence of photon and neutron cross-sections interaction with a substance [2, 3]. The radiation heat rate was calculated as a sum of two main components conditioned by the interaction of photons and neutrons with materials.

Dg – specific heat rate (W/cm^3) of photons that is determined as:

$$D_g = g \cdot \int_{E_{min}}^{E_{max}} j\left(E\right) \frac{m_a}{r}\left(E\right) E dE,$$

where:

g – material density, g·cm^{-3};

$j\left(E\right)$ – photon flux density, MeV^{-1}·s^{-1}·cm^{-2};

? – photon energy, MeV; E_{min} = 0.01 MeV; E_{max} = 9 MeV;

$\frac{m_a}{r}$ – mass factor of energy absorption, cm^2·g^{-1}.

Dn – specific heat rate (W·cm^{-3}) due to elastic scattering of fast neutrons on substance atoms that is generally determined as:

$$D_n = g \cdot \frac{2N}{(A+1)^2} \int_{E_{min}}^{E_{max}} j\left(E\right) s_S \left(1-\overline{\mu_n}\right) E dE,$$

where:

Y- material density, g·cm^{-3};

$N = 6.02 \cdot 10^{23}$ mol^{-1} – Avogadro constant;

? – relative atomic mass of an element;

s_S – microscopic cross-section of elastic neutron scattering, cm^2;

$\overline{\mu_n}$ – average cos of neutron scattering angle;

? – impacting neutron energy, MeV;

$?_{min}$ = 0.1 MeV; E_{max} = 10.5 MeV.

In group notion (NG – number of groups)

$$D_n = g \cdot \frac{2N}{(A+1)^2} \sum_{i=1}^{NG} \overline{j_i} \cdot \overline{s_{Si}} \cdot \overline{\left(1-\overline{\mu_n}\right)} E\rho E,$$

all the values under the summation sign are averaged within an i-group.

The contribution of other reactions (n,2n), (n,α), (n, n'+γ) to the heat rate in beryllium (D_{other}) makes up 5 % from the sum:

$$D_\Sigma = D_a + D_n + D_{other}.$$

3. Calculation of heat rate and burnup

The distribution of fuel fission and burnup rate in FA is calculated by the MCU code that allows a neutron transfer equation to be solved by the Monte-Carlo method on the basis of the estimated nuclear data for 3D geometry systems. In all calculations, neutron interaction cross-sections are used in the energy range 1eV – 10.5 MeV in a 26-group format of the BNAB constant system. Resonance characteristics of cross-sections are used in the form of sub-group parameters. In the thermal neutron area range 0 – 1 eV, cross-sections are presented in a 40-group decomposition with an equal rate pitch. Differential spread cross-sections are calculated accounting chemical relations, crystalline structure and material temperature.

To calculate the change of the isotopic composition of the reactor materials during the reactor cycle, the BURNUP code is used [4]. Numerical models of the reactor core are made accounting the dimensions, forms and materials of fuel rods, FA structural components, control rods and components having significant effect on the physical properties. In the MIR reactor the total energy release per fission is 200.1 MeV including 192.7 MeV of fission products, prompt and delayed gamma, beta, neutron and 7.4 MeV due to neutron capture (n, γ) reactions.

The account of the average energy Wn, released in the reactor due to neutron capture reactions per a fission, calculated as [5]:

$$W_n = \left(\bar{n}_f / k_{ef} - 1\right)\overline{W},$$

where:

$\bar{n}_f$ = 2.43 - average neutron number per a fission;

k_{ef} – effective neutron breeding factor in the conditionally critical task;

$\overline{W}$ – weighted average energy released in neutron-nuclide reactions, MeV;

$$\overline{W} = \sum_r d_r W_r \,/\, \sum_r (1 - n_r) d_r \;;$$

d_r – portion of neutrons participating in an r-reaction, except for the fission reaction;

W_r – energy released in an r-reaction and as a result of decay of its products (negative for an endothermic reaction);

n_r – number of secondary neutrons resulted from an r-reaction.

The specific decrement of heavy nuclides - ^{235}U and nuclides fissioned by thermal neutrons - is considered as a characteristic of the change of the fuel nuclide composition. The specific decrement of heavy nuclides is characterized by the loss and accumulation of ^{235}U, ^{238}U, ^{239}Pu and ^{241}Pu nuclei FP. Thus, the decrement of fissile nuclides is equal to the difference of fissile materials mass sums at its starting and final points. Fissile materials decrease due to the fission reactions and radiation capture and replenish due to the conversion of ^{234}U into ^{235}U and accumulation of ^{239}Pu and ^{241}Pu from ^{238}U. The calculation data are approximated by the least-squares method by the formulas:

$$\frac{M_{U5}(0) - M_{U5}(Q)}{Q} = A_{U5} - B_{U5}Q;$$

$$\frac{M_F(0) - M_F(Q)}{Q} = A_F - B_F Q,$$

where:

Q – FA total energy release;

$M_{U5}(Q)$, $M_F(Q)$ – mass of ^{235}U and fissile nuclides in FA;

A_{U5}, B_{U5}, A_F, B_F – constants characterizing absorption and fission of nuclides.

The calculation models are verified using the experimental results of the heat rate measurements obtained by the thermal balance method as well as using the in-pile measurements of neutron flux by direct charge gages and activation indicators. To verify the distribution over FA and burnup, gamma-scanning results are used.

4. Method for determination of FP content and burnup in fuel rods by gamma-spectrometry.

To determine the FP linear density in fuel are used experimental data obtained when measuring the gamma-quanta counting rate of ^{137}Cs fission product. The constants of ^{137}Cs are studied well enough (half-life – 30.1 years; gamma-quanta energy – 661.6 keV; gamma-quanta per decay – 84.6%, radionuclide burnup is insignificant). To use ^{137}Cs as a reference nuclide to determine the distribution and absolute burnup, data on the operating conditions

and maximal fuel temperature are to be available. In UO_2 pellet fuel, ^{137}Cs migration from uranium dioxide is not observed up to ~1200º? [6].
The method for determination of FP linear density is as follows: gamma-spectrometry facility is used to measure the intensity of gamma-quanta with energy of 661.6 keV emitted in radioactive decay of ^{137m}?? daughter isotope in the ^{137}Cs decay chain both in the fuel standard sample (FSS) with the known ^{137}Cs content (I^{fss}) and in the fuel rod under examination – (I^{fr}).
The total ^{137}Cs content in the fuel rod that should accumulate during heavy atoms fission, if there is no any spontaneous decay, is calculated by the following formula:

$$b_{Cs}^{fr} = \frac{I^{fr}}{I^{fss}} \cdot m_{Cs}^{fss} \cdot k1 \cdot k2 \cdot \frac{k_c^{fss}}{k_c^{fr}},$$

where:

m_{Cs}^{fss} – content of ^{137}Cs on a unit length of the FSS active part at the moment of its certification, g/cm;

$k1 = e^{-l \cdot (t_{mes} - t_{fss})}$ – correction factor for ^{137}Cs decay in FSS from the moment of its certification (t_{fss}) to the measurement (t_{mes});

($t_{mes} - t_{fss}$) – time period from the attestation to the measurement, day;

λ=6.29027 10^{-5}, day $^{-1}$ – ^{137}Cs decay constant accompanied with a 661.6keV gamma-quanta escape;

$$k2 = \frac{\grave{O}_N \times l}{\sum_{j=1}^{\hat{e}} n_j \times (e^{-l \times (t_{mes} - t_j)} - e^{-l \times (t_{mes} - t_{j-1})})}$$

where:

$?_N$ – total time of measured fuel rod operation at power, day;

$$\grave{O}_N = \sum_{j=1}^{\hat{E}} (t_j - t_{j-1}) \times s ,$$

where: σ=1 at $N_j > 0$; σ=0 at $N_j = 0$;

k – number of piecewise constant areas on the fuel rod thermal power change diagram;

$j = (1 \ldots k)$ – index corresponding to the test stage;

t_j – time corresponding to the completion of a j-stage and beginning of a j+1 stage of testing;

$n_j = \frac{N_j}{\overline{N}}$ – relative fuel rod power on the measured area at the j-stage of testing;

N_j – linear fuel rod power on the measured area at the j-stage of testing;

k_S^{fss}, (k_S^{fr}) – factors of 661.6 keV gamma-quanta self-absorption in FSS and fuel rod.

The FP content in the examined fuel rod (b^{fr}) is determined as a ratio of the ^{137}Cs content (b_{Cs}^{fr}) to the effective ^{137}Cs release during heavy atoms fission (ε_{eff}):

$$b^{fr} = b_{Cs}^{fr} / e_{eff},$$

where:

$e_{eff} = S_i(e_{iT}\, g_{iT} + e_{i?}\, g_{i?}) \frac{137}{\grave{A}i}$ – ^{137}Cs effective mass release per one fission of heavy atoms;

e_{iT}, $e_{i?}$ – ^{137}Cs atoms cumulative release per one fission of an i- heavy atom (^{235}U, ^{238}U, ^{239}Pu, ^{241}Pu), rel. unit [7];

$g_{i?}$, $g_{i?}$ – fraction of an i- heavy atom fission with respect to either thermal (th) or fast (f) neutrons is calculated;

$?_i$ – mass number of a fissioned heavy atom.

5. Conclusion

To determine heat rate of the MIR experimental and driver FAs, an improved thermal balance method is used that accounts heat rate due to neutron and photon emission in the structural materials of the experimental facilities and channels. The estimated error of the determined heat power and average burnup of experimental FAs makes up ~4.2% at =500kW of heat power and ~2.0% at =500 kW of heat power. The estimated error of the determined heat rate and average burnup in the driver FAs makes up ~4.7%.

To perform PIE in the reactor hot cell, a technique is used for determination of fission fragments accumulation by gamma-spectrometry using the standard samples. The different release of fragments during heavy nuclei fission is accounted as well as photon emission self-absorption in the fuel rods. Rod-type standard samples have been fabricated and the technique has been certified. The confidence error limits of the determined linear fission fragments accumulation make up 7%. The basic error of the burnup determination is conditioned by an uncertainty of FP release constants of different isotopes under different neutron spectra.

6. References

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REACTOR UPGRADE OF AGN-201 IN KHU, KOREA

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ABSTRACT

AGN-201 has been used for an educational purpose in Kyung Hee University since December 1982. Because of aging of I&C system and low neutron flux level at core (about $3.0X10^6$ #/cm^2-sec at rated thermal power, 0.1 watt), a refurbishment project was done with government support in order to increase availability of reactor for general students and researchers in Korea. The main goals of refurbishing of AGN-201 are summarized as three. The first is to up-rate flux level by 100 times for higher flux and to build an additional shielding structure around the reactor. The second is to have an additional digital operating console parallel to renovated original analogue console. The third is to set up various experimental coursesand legal operational systems to open reactor facility to general users. Project period was extended from planned period of 2.5 years to 4.4 years because of tough regulatory review procedures before issuing a construction permit Upgraded reactor, AGN-201K is now in operation since October 2007.

1. Introduction

AGN-201 is a zero power reactor installed at more than 20 universities around the world since 1960s'. One of them in Colorado State University was decommissioned and moved to Seoul campus of Kyung Hee University (KHU) in 1976. Based on simple regulation at that time, reactor was reinstalled at Suwon campus after two years construction period and started operation in December 1982. Since then this reactor was used only for students in Kyung Hee University, mainly for reactor experimental courses of senior grade. Up to the mid 90's, a TRIGA reactor in Korea Atomic Energy Research Institute (KAERI) at Seoul has been opened for all student in Korea. After the decommissioning of 2 TRIGA reactors, a new 30 MWt research reactor, HANARO in KAERI at Daejeon is the only reactor opened to university students. Because of busy schedule of operation for research and its own characteristic, student experiment has not been organized for reactor experiments since then. Now there is a strong need of utilization of AGN-201 in KHU for students. With AGN-201 a few research activities was done at late 80's for reactivity measurements by Rossi Alpha technique. However, there has not been active utilization because of a limitation of low thermal flux and facility obsolescence.
Since 2003, a refurbishing project of AGN-201 has been funded by government because that is the only one education reactor available to Korean students. About 150 students should take a reactor experiment course for a requisite credit national wide every year. Four major goals of refurbishing of AGN-201 are summarized as the followings. The first is to up-rate flux level by 100 times in order to enhance the reactor utilization opportunity for researches and education. An additional shielding structure and temperature control system should be built for a year-round utilization with enhanced biological shielding. The second is to have an additional digital operating console parallel to renovated original analogue console. A new console is not a replacement of obsolescent analogue console but provide a backup monitoring & control panel. A digital console was placed in parallel and interlocked to analogue console. Digital console provides user-friendly interface for students. The third is to set up various experimental courses

and facility with proper operation and maintenance procedures complying with current regulatory rules.
This paper explains the features of reactor and technical issues to be solved for licensing during refurbishment project period of 4 years and 4 months.

2. Features of Previous AGN-201

AGN-201 is installed in an isolated reactor building in the engineering school complex. The size of reactor is 2.4 m height, 2 m diameter and outer shape is shown in Fig.1. A cylindrical shape homogeneous core is a stack of disks which consist of polyethylene and 19.5 w/o UO_2 powder. The size of core is 23.75cm height and 25.8cm diameter. There is no active cooling system for a tightly-sealed core canister because rated thermal power (0.1 watt) is very low enough to be cooled with air. There are multiple biological shields around the core; 20cm thickness graphite, 10cm thickness lead, steel reactor tank and 50cm thickness water tank. Material compositions of 4 control rods are the same with fuel disks. Excess reactivity predicted for a fully inserted condition is about 0.18%?k/k and is controlled by a fine control rod.

Fig. 1 Reactor Hall of AGN-201, KHU

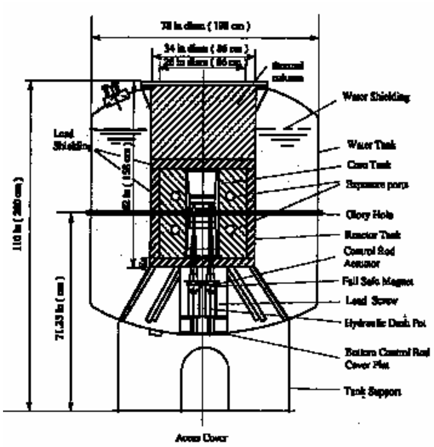

Fig. 2 Cross-sectional View of AGN-201

Operating console is too obsolescent to accommodate new demands for experiments as shown in Fig.3. There are three single wired neutron instrument channel connected to console from two BF3 ion chamber and one proportional counter. There are three shutdown signals from these chambers and additional interlock shutdown signals; shielding water low-temperature signal activating below 16? , shielding water low-level signal, earthquake vibration-signal. An additional device for reactor safety against abnormal power excursion was a thermal fuse at the central part of core. This part is designed to be melt at the 120? in advance of fuel melting at 200? in case of power excursion and make bottom-half core drop down resulting in sub-criticality due to the separation of core.

Fig. 3 Reactor Console of AGN-201, KHU

The same kinds of reactors are now in operation in 3 universities in the USA (University of New Mexico, Texas A&M University and Idaho State University) and one in Italy (University of Palermo). They were all up-rated to 5 watts more than 15 years ago and are ready to be licensed or have been licensed as AGN-201M.

3. Technical Issues in Refurbishment Project

3.1 Shielding Design for Reactor Power Up-rate

In order to increase flux level upto $3.0x10^8$ #/cm^2-sec, rated power will be increased by 100 times to 10 watts from 0.1 watts. There is no impact on cooling except biological shielding. Expected temperature increase in fuel is about 24? in maximum at adiabatic condition.

Required thickness of concrete wall around the reactor was calculated by MCNP-4/C code with r-z geometry model. As shown in Fig.4, shielding structure should have two moving doors at both front and top sides. Concrete wall thickness should be 60 cm as shown in Fig. 5 in order to meet 2.5 mrem/hr dose-limit at the outer surface of wall.

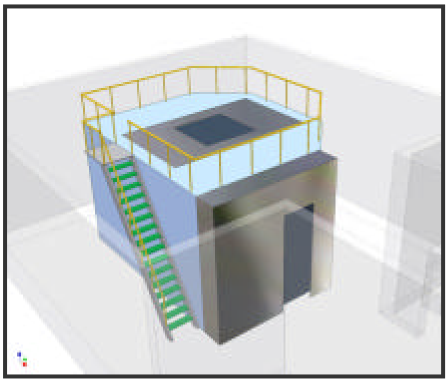

Fig. 4 Shielding Block Design

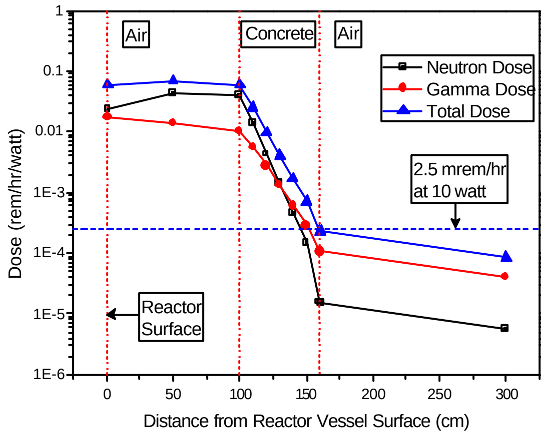

Fig. 5. Shielding Calculation Results of MCNP-4

Because of weight of door, thickness of steel is designed to be 25 cm for front door and 15 cm for the top doors. It was calculated that 15 cm be thick enough for gammas but not for neutrons. Therefore 25 cm thick paraffin block was installed inside of shielding structure.

Because of high reactivity feedback coefficient of -0.0275%/? , AGN-201 is very safe against over-power transients. However, reactor cannot be reached to critical at hot summer days because of high feedback and very low excess reactivity of 0.18%. Therefore, reactor cavity inside of shielding structure was equipped with air-conditioning systems as a non-safety feature. Installation of shielding structure with moving doors needs a change of reactor protection system. Additional interlock shutdown signals were installed for safety. Manual shutdown signal can be generated inside of shielding structure. Reactor console cannot operate when one of the doors are open. Gamma radiation dose rate and air temperature at both inside and outside of shielding block is now monitored and recorded all the time.

3.2 Operating Console Upgrade

A new digital operating console was installed parallel to the current analogue console. In order to avoid licensing procedures for console addition, digital console is designated only for monitoring and fine-power control but not for shutdown. All safety shutdown function is kept by analogue console with original safety logics. In order to enhance information display for operators and students, man-machine interface design was applied to digital console with reactivity meters and database management system. Fig. 6 and 7 are examples of human machine interface design in a digital console which is activated by touch screen mode.
Because of up-to-dated regulation rules of Korea Institute of Nuclear Safety (KINS), AVR and UPS should be installed at both consoles. Cable ducts for power and signal should be separated for independency. Most of cables and connectors in old analogue console were replaced with new parts and cables.
One of the technical issues on the addition of digital console was the assurance of physical separation of two consoles. Separator was owned by operator at the analogue console and operator can allow the function of fine control rod movementto student at the digital console. He can switch back to analogue console at any time by separator. Q-class parts were used for the safety circuits and tested by KINS inspectors.

Fig. 6. Layout of Digital Console

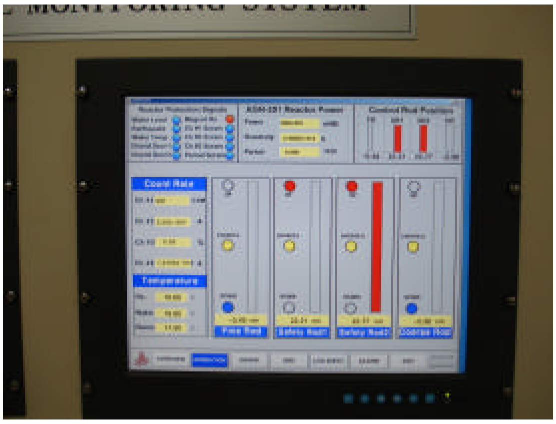

Fig. 7. Example of Human-Machine Interface – Operation Mode Screen

3.3 Safety Evaluation on Hypothetical Radiation Exposure Accidents

Safety evaluation was done in three aspects before license permit for reactor power up-rating. Inherent safe features against reactivity accidents were not changed much from those of a reactor before up-rating. Secondly, increased radiation dose hazard due to power increase can be protected by additional shielding. Integrity of shielding structure against earthquake should be measured in detail because of heavy weight of concrete walls of 60 cm thickness. This third item among many licensing issues made project delayed.

Reactor building for AGN-201 was built in 1980 when there were no seismic design criteria. Current rules in Korea want to evaluate a safety-grade seismic analysis for all structures within a reactor building without exceptions for the research reactor category. Instead of following current rules focused on commercial nuclear power plants, hypothetical radiation hazard was evaluated based on initiating event of earthquake, but complying with non-safety grade seismic analysis for reactor building. Many conservative assumptions were applied for radiation release analysis.

1) It was assumed that reactor itself be broken into pieces because of earthquake of level-7.

2) Accumulated amounts of radio-isotopes were calculated by ORIGEN-2 with assumption that reactor be operated continuously at full power for 6,400 hours concerning 40 yr lifetime.

3) Radiation release was calculated for inert gas of Kr and Xe as well as Iodine based on fission gas release diffusion model with diffusion coefficients which were 10 times higher than those predicted at operational temperature.

4) Exposure rate was evaluated by ‘Handbook of Dose Coefficient v.2.5.3’ developed by KAERI. Internal dose was evaluated for element type iodine based on 24 hours breath assumption with respiration rate of $1.2m^3/hr$.
Calculated values are shown in Tab.1 and those are very low compared with limit values from NUREG-1537, ANSI/ANS 15.7 and IAEA-TECDOC -403.

Table 1. Internal and external exposure rate from hypothetical radiation accident

	Internal Dose by Iodine (mSv)	External Dose by Kr, Xe and I (mSv)	Sum of exposure dose (mSv)	Limit of ANSI/ANS15.7 (mSv)	Ratio to Limit
Thyroid Dose	0.448	0.00382	0.45182	15	3.01 %
Effective Dose	0.0238	0.00362	0.02742	5	0.55 %

4. Future Plans

Project period was extended from planned period of 2.5 years to 4.4 years because of tough regulatory review procedures before issuing a construction permit. After concluding a license review procedure, construction was done during 4 months. Start-up test and pre-inspection was done by KINS during a month of Oct. 2007. Upgraded reactor, AGN-201K is now in operation since October 2007.
Even though reactor power is increased by 100 times, rated power level is too low to accommodate a lot of experiments related to radiation utilization. Neutron radiography and neutron activation analysis for carbon dating are demanding experiment for future generation, but flux level at the beam port is too small. More design work should be done for installing sophisticated and expensive devices at the thermal column area.
An incident of control rod cladding failure was reported from Idaho State University. Because of material degradation in control rod manoeuvring system, the same problem can be happened in AGN-201, KHU. Spring, positioning meters and motors should be replaced at the next stage of project. If the same incident is happen, there is no way to obtain extra control rods which contains the fuel composition the same with core fuel. Extra fuel material should be reserved for replacement in future.
For the opening of AGN-201 for national wide utilization, an extensive effort should be done for designing of one-week experimental course including theory and practice.

Acknowledgements

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ANALYSIS OF THE CALCULATIONAL TECHNIQUES OF HEAT DEPOSITION VIA MCNP IN COMPARISON WITH A CALORIMETER EXPERIMENT AT SAFARI-1 RESEARCH REACTOR

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ABSTRACT

The accuracy of predicting any physical parameter (simulating natural behaviour) via MCNP is not a straightforward problem, where obtaining a value with a low variance is not the only requirement for good Monte Carlo calculations. Factors which affect the accuracy have been discussed in literature. These factors amongst others are; the physical models of particle transport, their production, destruction and physical models approximations. It is discussed in simulating a Calorimeter experiment to predict the heat deposition, where the contribution of the material compositions to the problem accuracy is determined and their source of error are quantified. Other factors have been determined through the course of this work and are considered for future work. The nonanalog techniques applied for more efficient calculations are out of the scope of this paper and are only mentioned briefly. The paper concludes with the preferred methodology to predict heating and areas of future work. The pulse height estimator with neutron analog absorption and full particles transport was the best estimate for the heat deposition; however a large margin of error sources makes it questionable.

1. Introduction

One can perform Monte Carlo calculations with MCNP easily, getting answers with very low variance possibility, where the variance is the only goal to be minimized, without being able to judge if the problem solved has simulated the natural physical behaviour.
In literature it has been discussed explicitly how to do an efficient Monte Carlo calculation in terms of computing time and problem statistics, e.g. in deep penetration problems when the analog simulation fails to give answers.
Factors affecting the problem accuracy that need to be minimized are now discussed.
The problem-modelling factors are constituted of the geometrical configuration and material isotopic which are addressed to be future work factors to be minimized. The geometrical configuration of the SAFARI-1 core is being built with the OSCAR-3 MCNP Interface system (OSMINT) which includes an engineering data file containing the detailed geometry, and where the engineering tolerances are to be optimized for the whole core. OSMINT detects the whole core configuration, isotopic and bank withdrawal positions from OSCAR-3 at a cycle snapshot. The isotopic from OSCAR-3 plays a major role to define the local radiations environment spectra (Calorimeter core), where the material composition is the main severe source of error. To predict the margin of error in comparison with the Calorimeter experiment heat deposition, a pre-comparison with flux profiles has been conducted to correlate for this error, using a Cu wire activation measurement to compare with MCNP reaction rate results.
Abovementioned code factors are discussed, and the cross section representation still needs to be optimized in future work. The continuous energy neutron libraries contain photon production data in expanded format and at maximum neutron cross section energy of 150

MeV when possible are used, in order to determine the actual photon spectrum. Photoatomic and electron libraries of recent recommended evaluations are used and photonuclear data are not used for the entire problem.
This paper discusses the physical measurement by the calorimeter, and the results of using different physics treatments in neutron, photon and electron transport, production and destruction, and cross section libraries JEFF-3.1 and LANL/T16 to determine the heat deposition in the calorimeter core.
Assuming there is no error from the plant data (Thermal power, control bank positions, average core coolant temperature) and the measurement has been conducted in a well-defined environment, and the user error of checking outputs, input errors, nonanalog techniques applied and physical measurements are minimized.

1.1 Experiment

Description of the Calorimeter
A French patent calorimeter was adopted for this experiment. The instrument constitutes a core (molybdenum) 1.143 cm in diameter with a height of 4 cm embedded in an atmosphere of air (which is mainly nitrogen), which is then encased in a leak tight stainless steel 316 vessel (so called "jacket") of inner and outer diameter of 1.2 cm and 1.3 cm respectively and a corresponding height of 6 cm. Embedded in the jacket and the core are thermocouples which detects temperature variations by sending a signal to the monitors or recording devices. The essence of this calorimeter is to maintain the temperature of the jacket to the temperature of the water in the reactor pool during the course of the experiment.

Description of the Experiment
Gamma and neutron radiation inside the reactor core deposits energy in metals, therefore causing their temperature to increase. In-core calorimetry is a method used in the SAFARI-1 reactor to determine heat deposition in materials since the rise or change in temperature is related to heat deposition in a medium. A calorimetric experiment was conducted in the SAFARI-1 core to measure heat deposition in molybdenum and stainless steel (304) core calorimeters respectively. This experiment was done at beginning of cycle 0704-1 (April 2007) at a thermal power of 5 MW. The objective of this experiment is to characterize heat deposition in different materials for the purpose of future fuel irradiations, isotope production, rig design and other material irradiations.
In physics, the heat is considered as the amount of energy transferred due to the difference in temperatures between two systems. And in the calorimeter experiment, once radiation particles undergo nuclear reactions inside the calorimeter material core, the core temperature will rise to a stable temperature, and the measuring device will determine the amount of specific heat (W/gm) deposited by radiation in the calorimeter core.
The heat produced, due to mainly photon radiation, deposit all or part of their energy in the calorimeter core, and the neutron photon and secondary photon spectra in the core will define the amount of heat deposited.

2. Material Composition

▪ The core simulator in OSCAR-4 uses the Analytic Nodal Method (ANM) to solve the neutron diffusion equation. Due to the fact that core geometry is modelled by assembly sized homogeneous nodes (with a single set of broad group cross sections per node obtained using an assembly transport solver) and that nodal diffusion methods in general are inaccurate close to strong absorbers, the isotopic of the neighbouring fissionable assemblies will be a source of error. When the reaction rates from MCNP and the Cu activation measurements are compared for the axial layers corresponding to the position of the calorimeter it can be seen that this error is approximately 4% (Figure 1).

3. Physical Models and Cross Section Representations

▪ The MCNP results are normalized per source weight, and depending on the source units used (particles) the result units are determined. The average energy released per fission calculated for core 0704-1 is 1.83851E+02 ±3.67703E-02 MeV/fission, where per source values are normalized to the reactor power by the factor 4.14220E+17 ±8.28273E+13 per second. The k-eff is not used where MCNP inherently normalizes the fission source inside the reactor core at each criticality calculation cycle by k-eff to keep the fission source constant which could be identified from the estimated fission source entropy stability during the active cycles.

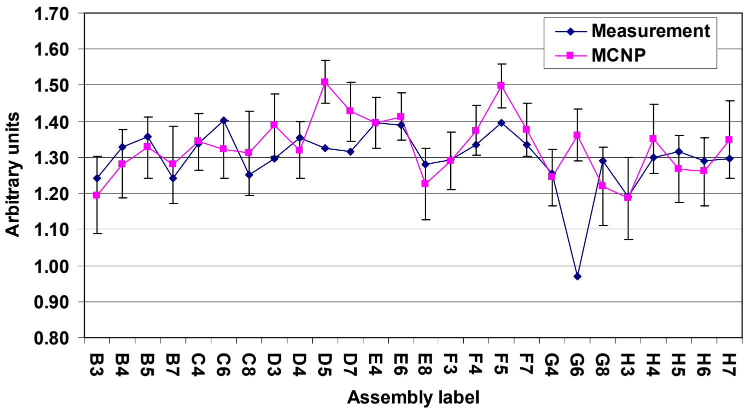

Figure 1. Cu measurement values compared with the MCNP reaction rates.

▪ The phase-space (spatial) neutron, photon and electron importance functions have been generated for optimal tally specific neutron/photon/electron geometrical weight window bounds (using a superimposed mesh), with an iterative process used to asses the conversion of the importance functions and clear improvement in the computational efficiency (computing time and statistics), where using the optimal weight window bounds minimized the time required to reach better statistics showing a clear improvement of the FOM (Figure Of Merit) by two orders of magnitude on average.
▪ The neutron implicit capture and the analog capture in the photon detailed physics treatment which are the default have been used. The neutron-induced photon production are not biased, where MCNP by default uses the minimum weight low bounds of -1 which produce photons depending on the source and current cell importance and source neutron weight and when applying the weight window, the photon production low weight bounds specified to be equal to weight window bound.

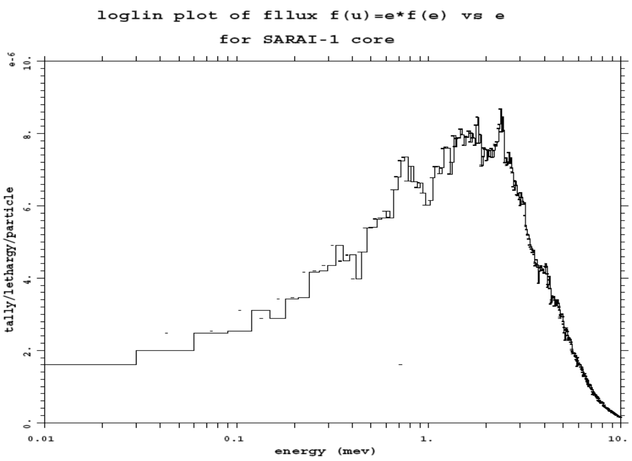

Figure 2. A loglin plot of the lethargy-normed neutron flux vs e for SAFARI-1 core.

▪ Figure 2 is the lethargy-normed neutron flux inside the SAFARI-1 core, where the neutron spectrum won't be altered by the different photon and elector physics treatments in the reactor core and the calorimeter core, unless the photonuclear data were used and cross section representation changed as shown in Figure 3. The effect of applying the photonuclear data on these spectra will be discussed in future work.

▪ Figure 3 is a linlog plot of energy-normed photon flux for the SAFARI-1 core. Results from JEFF-3.1, which lacks the photon production data for certain heavy isotopics, and results from calculations performed with LANL/T16 that are identified by a".69c" all other isotopes in the system used ENDF ".24c & .42c" cross sections with ".60t" S(α,β) data for light water and Beryllium. The plots show the difference in photon spectrum due to the presence of more accurate photon production data for certain heavy isotopes and due to the change in the whole core neutron spectrum. The calculated k-eff for both was respectively 1.01728 and 0.9923 with an estimated standard deviation of 0.0005 for both.

More investigation to minimize the error of cross section representation will be discussed in future work.

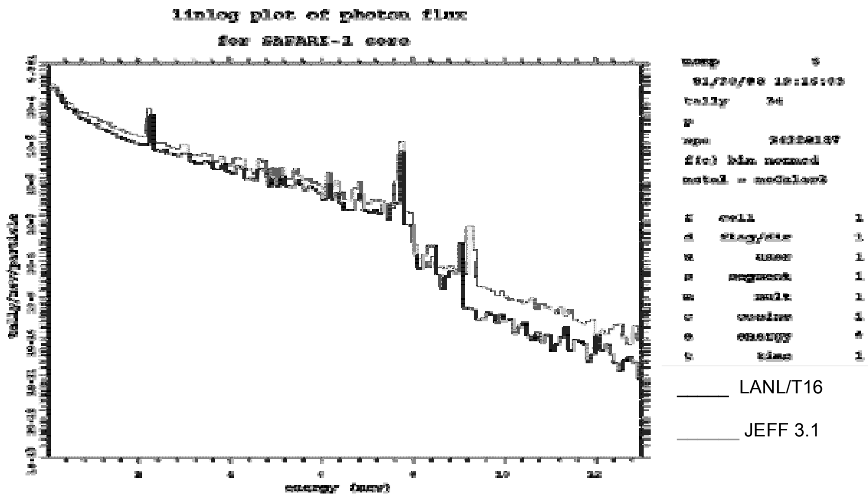

Figure 3. A linlog plot of photon energy-normed flux in the SAFARI-1 core. Using LANL/T16 vs JEFF-3.1 nuclear data libraries.

▪ The rest of the analysis was performed using the JEFF-3.1 nuclear data library. The contribution of the error from cross section representation will be obtained by comparing calculated and measured heat depositions after correcting the error coming from material composition.

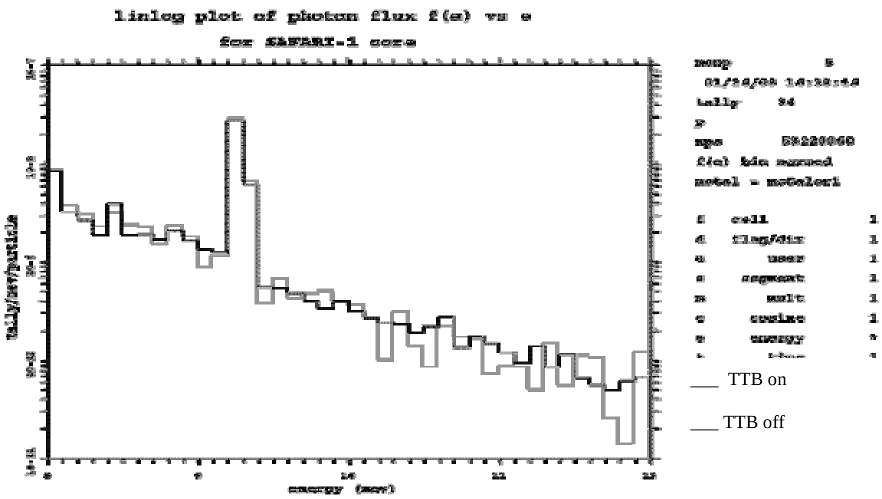

Figure 4. A linlog plot of photon energy-normed flux for the SAFARI-1 core, with the thick target Bremsstralung on and off respectively, in full electron transport problems.

▪ Figure 4 is the photon energy-normed flux in the SAFARI-1 core from 8 MeV to 12 MeV, the increase in photon total flux is due to the Bremsstralung photons which were accounted for and biased to higher energy. Below 8 MeV the difference in photon spectra were of insignificant importance.

▪ Figures 5 is the photon spectrum inside the calorimeter core from .01 MeV to 10 MeV, the right-side plot is the loglin lethargy-normed flux, and the visual area under the curve accurately represents the contribution to the total flux, which shows an increase in the total area under the curve at energies from 2 to 10 MeV when the secondary photon transport and creation were accounted for.

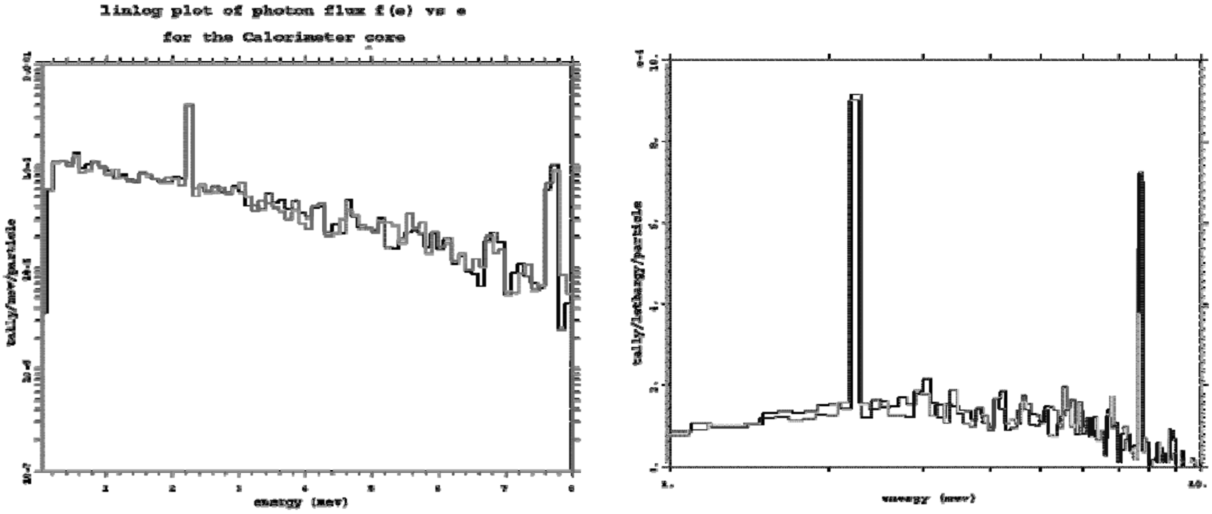

Figure 5. Left is the linlog plot of photon energy-normed flux in the Calorimeter core. Right the loglin VAA plot of photon lethargy-normed flux in the Calorimeter core.

▪ Figure 6 is a loglin lethargy-normed heat deposition, the visual area under the curve accurately represents the contribution to the total photon heat deposition. Where the solid line represents results from the model where photons produce electrons which create secondary photons. The gray line represents results from the model where the secondary photons are not created (it was accounted for in the first photon's deposition), and the electrons deposit all their energy locally, which is the interpretation of the heating number used as a flux energy-dependent multiplier to estimate the heat deposition.

In order to obtain the heat deposition correctly, while the heating number is used, the entire problem should be run without tracking electrons and creating secondary Bremsstralung photons, because the heating number assumes instantaneous local deposition of all secondary electron energy. Therefore, the deposition site of electrons and secondary photons will not be accurate. The heat deposition was over predicted by 3% when the photon and electron transport is applied due to the conflict in using the heating number as an energy-dependent multiplier in the photon/electron transport problem.

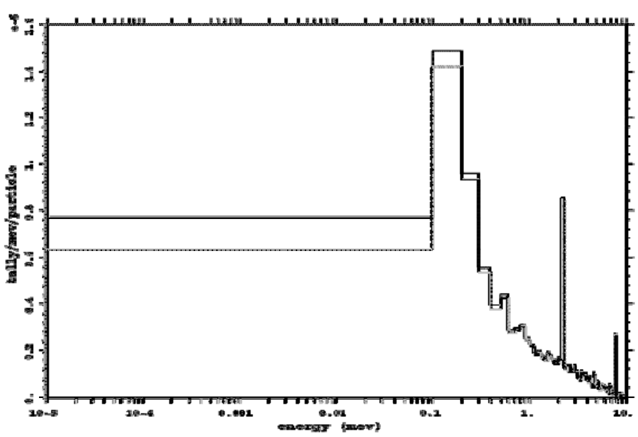

Figure 6. A loglin plot of photon lethargy-normed heat deposition in the Calorimeter core. solid line: electron transport and thick target Bremsstralung TTB. Gray line: TTB turned off.

▪ In order to avoid the approximation in the heating number throughout the course of neutron/photon/electron tracking, and after it was shown that the contribution of neutron heating is insignificant, the pulse height tally is used to estimate the photon/electron heat deposition. It is compared with the track length flux estimator, where nonanalog neutron capture is applied. Figure 7 is a loglin plot of lethargy-normed heat depositions.

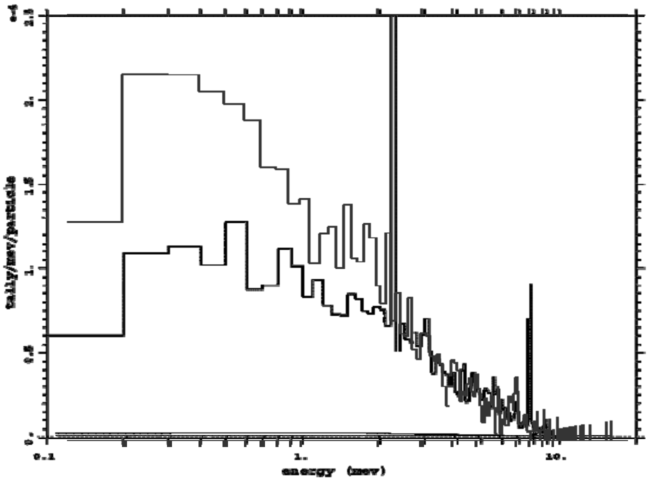

Figure 7. A loglin plot of photon lethargy-normed heat deposition in the Calorimeter core, solid line is the track length estimator and gray is the pulse height estimator, and the small gray line attaching the x-axis is the electron contribution.

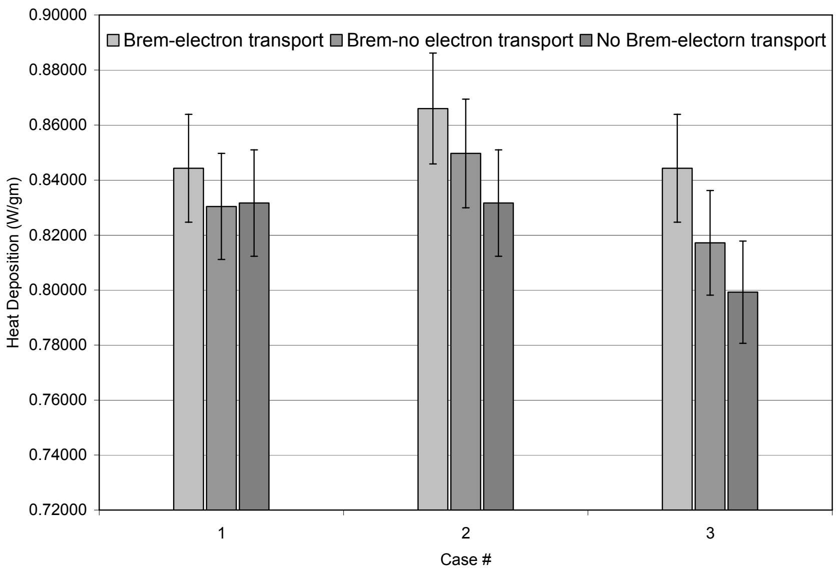

Figure 8. The results of calculating the heat deposition in the Calorimeter core. Each case has: Estimator 1 is the track length estimator with the heating numbers and cross sections as energy-dependent multipliers, with the electron transport and secondary photon production. Estimator 2 is over predicted where the secondary photons were accounted for in terms of the heating number and on the secondary photon/electron transport. Estimator 3 is the pulse height estimator with a full photon/electron transport and analog neutron capture.

In Table 1 the second column shows the main results of full neutron/photon/electron transport heat depositions in the Calorimeter core. The third column shows the results after correction with the estimated source of inaccuracy from material compositions provided by OSCAR-4, the fourth column shows the error from other sources of inaccuracy which have been earmarked for future work to add all the 3%.

Table 1. Results compared with measurement

Estimator	Brem-electron transport	After Correction	C/E-1
1	0.84426	0.86388	-3.26038%
2	0.88111	0.90160	0.96303%
3	0.84496	0.86460	-3.18004%
Measured	0.893	0.893	---

4. Conclusions

The aim of this work was to explicitly address the sources of error in modelling the SAFARI-1 reactor core, by examining all the factors that affect the accuracy and may lead to a systematic error. The present sources of uncertainty to calculate the heat deposition accurately in comparison with the Calorimeter experiment is also quantified.
Factors that must be addressed in future work are; the cross section representation which approximately contributed 3% to the error, and isotopic transferred from OSCAR-4 which was found to contribute 4% in the region close to the Calorimeter core, and 14% for the over all reactor core on average, and the effect of using the photonuclear data which has not yet been applied.
The comparison between physical models for neutron/photon/electron transport concludes that the pulse height estimator with neutron analog absorption and full particles transport was the best estimate for the heat deposition. And showed a 3.3% increase of the heat deposition when a full electron transport was applied to the problem with the pulse height estimator.

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SAFETY ANALYSES OF REACTIVITY INITIATED ACCIDENTS (RIA) AND ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS) OF THE BUDAPEST RESEARCH REACTOR

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ABSTRACT

The foreseen fuel enrichment reduction in the Budapest Research Reactor (BRR) and the Hungarian Atomic Energy Agency's decision made the revision of the Final Safety Analysis Report and repeating some limiting analyses - both for the present and for the new fuel - necessary. According to the authority decision, modernized tools of the analyses were applied. A special version of the KIKO3D 3D neutronics-thermohydraulic dynamic code, created originally for the VVER-440 power plant core, was developed for the Budapest Research Reactor. The program is coupled to the ATHLET system thermal hydraulic code. After reviewing the tools used for the RIA analyses, their special validation for the BRR analyses, the preparatory and analysis activities, the applications of the codes for RIA and ATWS are presented. The analyses proved the reactor safety, nevertheless for some extremely low frequency RIA ATWS events it was found that the reactivity released due to the decreasing xenon poisoning, and the resulting positive feedback is not negligible. Because the maximum reactivity insertion due to the control rod movement is limited by a built in automatic mechanism, a suitable time interval for the operator action – obtained from the analyses - is available.

1. Introduction

A special version of the KIKO3D 3D neutronics-thermohydraulic dynamic code, developed originally for the VVER-440 power plant core, was elaborated and validated for the Budapest Research Reactor, which is a 10 MW pool type reactor with 36 % enriched WWR-M2 fuel. The code is coupled to the ATHLET system thermal hydraulic program, which has also been validated by using the energetic and physical startup measurements. New analyses in the FSAR became necessary on one hand due to the authority requirements and on the other hand because of the Low Enriched Uranium (LEU, 19 %) fuel introduction.

After the preparatory work (handbooks and validation of the codes, calculations of the initial core states and the neutronic frame parameters; reactivity coefficients, etc.), the analyses were carried out by using the ATHLET, KIKO3D, coupled ATHLET-KIKO3D and the LOCASYM codes. First, the different groups of the initiating events were reviewed and the limiting cases were selected. The different groups, the limiting cases and the codes used are presented in Table 1.

A common set of frame the parameters were set up, covering all cycles including the transient ones, where the high and low enrichment fuel elements are mixed. An advantage of this treatment was that only AHLET-KIKO3D calculations had to be carried out for both fuels. The subcriticality of the fresh fuel storage and the "inner storage" of the burnt fuel was also analyzed. The safety analysis acceptance criteria are fulfilled in all cases.

Group of initiating events	Calculated limiting case	Code
Loss of Flow	Covered by the full blackout	ATHLET
Decrease of secondary cooling	Covered by the full blackout	ATHLET
Reactivity Initiated Events (RIA)	All cases calculated	KIKO3D, ATHLET-KIKO3D
Loss of Coolant	Most cases calculated	LOCASYM

Tab 1: Initiating event groups, limiting analyses and the applied codes

The following Reactivity Initiated Events were analyzed with and without scram (the latter ones belonging also to the Design Basis).

1. Uncontrolled reactivity insertion after the startup at strongly xenon poisoned state
2. Cold water ingress
3. Sudden unintentional control rod withdrawal
4. Uncontrolled reactivity insertion during reloading
5. Uncontrolled irradiation device withdrawal
6. Uncontrolled control rod withdrawal (different scenarios)
7. Unintentional control rod misalignment

2. Reactor Physics Tools

2.1 Short description of the KIKO3D code

KIKO3D is a three-dimensional nuclear reactor dynamics program developed originally for coupled neutron kinetics and thermohydraulics calculation of VVER type pressurised water reactor cores [1,2], and modified later for other LWRs, like the research reactor or High Performance Light Water Reactor. The code has been developed in the KFKI Atomic Energy Research Institute. Main applications of KIKO3D are the calculation of asymmetric accidents in the core, e.g. start-up of inoperable loop, inadvertent control rod withdrawal. The above -so called - middle-fast transients play an important role in safety analyses. The modelling of the faster transients characterised by the pressure waves spreading at sound velocity is out of the scope of this program. The research reactor version is able to calculate also the burnup and xenon processes and the reshuffling of the reactor. In this way, the initial sates of the transients can be derived, moreover xenon processes during the transients can also be taken into account. KIKO3D is a nodal code, where the nodes are the hexagonal or rectangular fuel assemblies subdivided by the axial layers. The symmetries of full and 1/2 core can be used in the calculations. The neutron kinetics model of KIKO3D solves the two-group diffusion equations in homogenised fuel assembly geometry by a sophisticated nodal method. Special, generalised response matrices of the time dependent problem are introduced. The unknowns are the scalar flux integrals on the node boundaries. The time dependent nodal equations are solved by using the IQS (Improved Quasi Static) factorisation method. The parameterised response matrices are calculated by using the KARATE code system (see 2.2).

2.2 Short description of the KARATE code system

KARATE is a code system to perform core design calculations [5-7]. Originally, it was developed for VVER reactors, but several modules, namely the spectral ones, are also applicable for other reactor types, like VVRSZM, HPLWR.

KARATE has been elaborated to calculate rector cores by a coupled neutron physical-thermal hydraulics model. The main goal of the calculation is core reload design, however, certain problems of the safety analyses amenable to a static code can be analysed. Accordingly, stationary neutron physics and thermal hydraulics models have been implemented. These models are capable of following burnup and Xenon processes but do not allow for calculating faster transients demanded in a safety analysis. The program serves economic core reload design so that the limitations demanded by the safety analysis should be observed. The reload limitations ("frame parameters") demanded by the safety analysis are also available from the calculations.

KARATE involves all the libraries and computer programs which are needed to perform fuel cycle calculations and fuel cycle design. The libraries need refreshment if a new fuel type is being used or if the parameter range of an existing fuel is being extended. The calculation is grouped into 3 levels. A level is connected to the higher one through parameterized data libraries, these libraries provide a part of the input data for the higher level. A level is connected to the lower one also, usually boundary condition is provided for a "Lupe"-like calculation. Input to KARATE are the ENDF/B-VI nuclear data library, engineering data (geometry, core composition etc.). A typical output comprises the power distribution in the core, and reaction rate distributions in selected assemblies.

2.3 Validation of the reactor physics tools, calculations of the "frame parameters"

Before the analyses, the following Budapest Research Reactor validation calculations have been performed [3,4].

- Validation by measurements during the physical startup:
  - Isothermal temperature reactivity coefficient (from which a temperature dependent reactivity curve is derived), see Fig. 1.
  - Control rod worth measurement
- Validation by the measurement performed at nominal power state:
  - Control rod worth measurement
  - Space dependent distributions of different activity foils
- Comparison of power distributions against Monte Carlo, DIF3D and REBUS-3 ANL results [10].

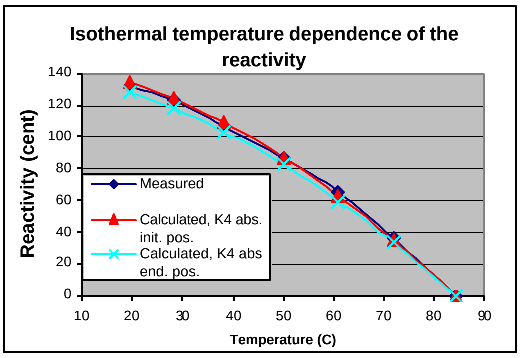


Fig. 1 Measured and calculated reactivity change as a function of the isothermal temperature

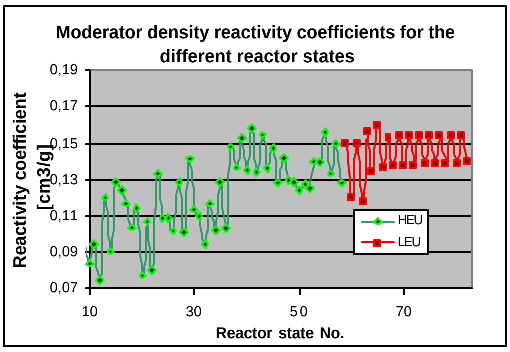


Fig. 2 Moderator density reactivity coefficients for the different reactor states, BOC and EOC of HEU, trans., LEU cycles

Before the analyses, the so called "frame parameters" were determined for all the HEU, transition and LEU cores. The frame parameters are the enveloping values of the most relevant reactor physics characteristics influencing the transient results: reactivity coefficients, control rod worth, power peaking factors, etc. In case of the ATWS, the most important one is the moderator density reactivity coefficient. According to Fig 2, this parameter is changing in a wide range from cycle to cycle. The main reason is the great sensitivity of this parameter on the control rod positions.

2.4 Short description and research reactor specific validation of the ATHLET code

The ATHLET system thermal hydraulic code [8] was developed in GRS (Gesellschaft für Anlagen- und Reaktorsicherheit mbH) for the investigation of the phenomena at normal and abnormal transients, small and large loss of coolant accidents in light water reactors. Besides the point kinetics not used here, the code consists of the following main modules:

- Thermo-fluiddynamics - TFD
- Heat transfer module – HECU
- General Control Simulation Module – GCSM

In the research reactor calculations a detailed nodalization of the primary loop and the secondary side of the heat exchanger was applied.

The code is coupled to the KIKO3D neutron kinetics code. Beside the basic verification and validation, the code was validated against the following special Budapest Research Reactor problems.

Measurements at hot standby state:

- Pressure distribution in the primary loop at nominal flow rate
- Flow rate dependent pressure drop in the reactor and in the heat exchanger
- Flow rate dependent coolant level in the reactor pool and in the gravity auxiliary vessel

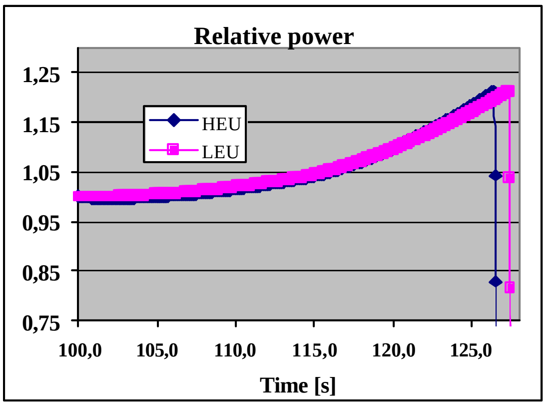
Measurement at nominal power state:

- Blackout (causing pump trip and scram, without single failure)

The nodalization contains a special hot channel for the acceptance criteria fulfillment evaluation. The model was verified by comparison to the ANL PLTEMP/ANL V3.0 channel thermal hydraulic code results [9].

3. Calculation of an unintentional control rod withdrawal with and without scram

In the presented analysis, the „K4" control rod - in an asymmetric radial position - is withdrawn unintentionally. The normal control rod movement velocity of 0,79 cm/s was applied in the analysis. The results obtained in case of scram are shown in Figs. 3,4. The cladding surface temperature does not reach the temperature necessary for subcooled boiling.



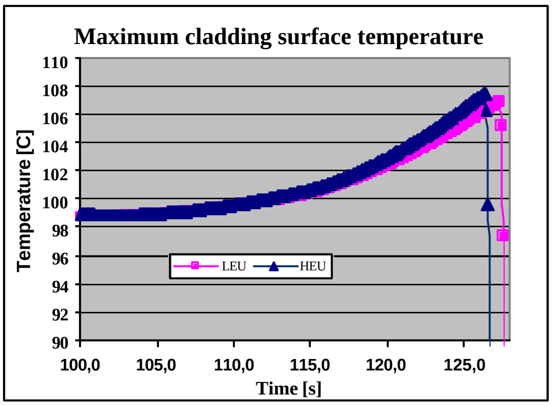


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Fig. 3 Relative reactor power

Fig. 4 Maximum cladding surface temperature

In case of the ATWS, in spite of the extremely low probability, it is postulated that the automatic scram is not actuated. Nevertheless the maximum inserted reactivity is limited according to the built in automatic mechanism, which allows only a limited control rod movement at any event. As a parametric study, a set of the maximum allowed reactivity insertion was investigated, which were 10, 15, 20, 30, 40 cents.

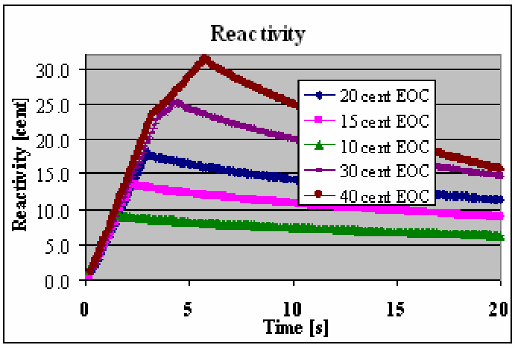

Fig. 5 Reactivity

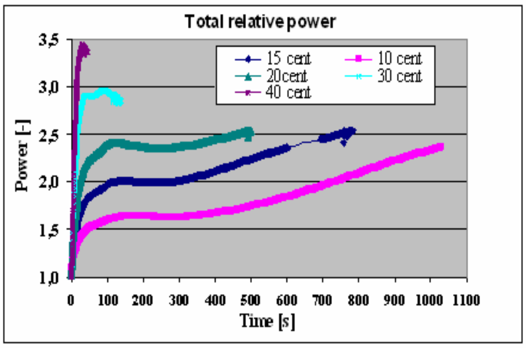

Fig. 6 Total relative power of the reactor

As a consequence of the inserted reactivity (Fig. 5), the reactor power is increasing quickly in the first seconds of the transient (Fig. 6). Following the first period, the power increase starts slowing down (Fig. 6). The reactivity decrease after the first seconds is caused by the fuel and moderator temperature feedback (Figs. 7,8). Later, at about 300 s, a slight reactivity increase (Fig. 9) can be observed again due to the more enhanced xenon depletion at the increasing power. This process - with slow positive feedback between the power and the xenon burnup - leads to a continuous power and temperature increase after 400 s.

The hot channel results are shown in Figs 10-16. The increased hot assembly power (Fig. 10) leads to increasing inlet and outlet temperatures (Figs. 11,12), outlet void content (Fig. 16), maximum cladding temperature (Fig. 13). The enhanced temperature and void content results in decreased hot channel mass flow (Fig 15) because the flow can be directed to other less loaded channels. This positive feed back process finally could lead to a very low mass flow, boiling crisis and temperature runaway. However, a large number of signals, initiated by the increasing temperatures, are warning the operator, who can insert the regulating control rods manually, shutting down the reactor even without the scram rods, and at least –2$ subcriticality can be reached.

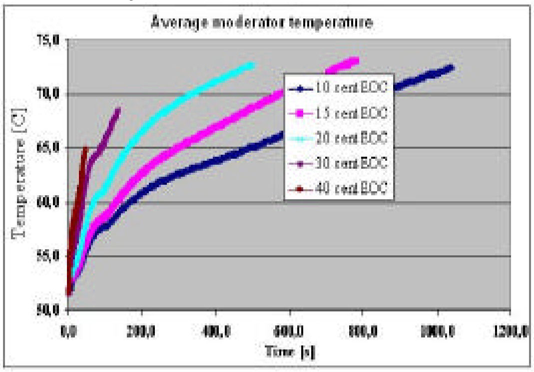

Fig. 7 Average moderator temperature

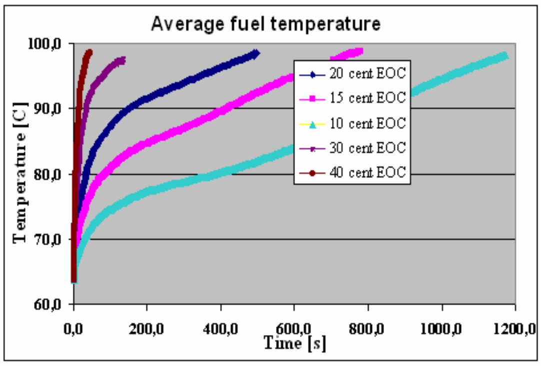

Fig. 8 Average fuel temperature

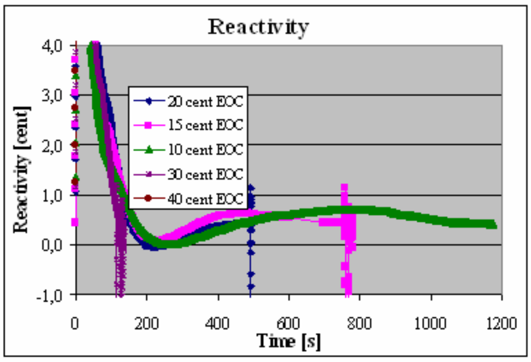

Fig. 9 Reactivity

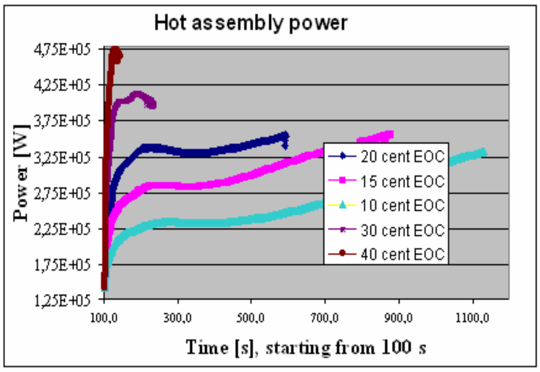

Fig. 10 Hot assembly power

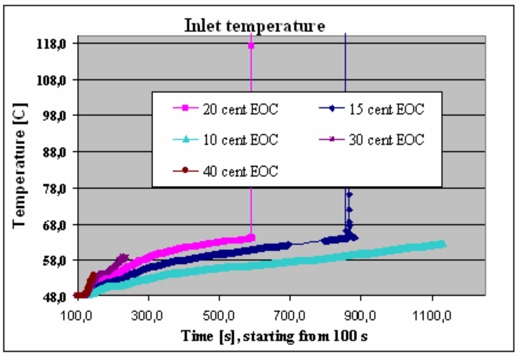

Fig. 11 Inlet temperature

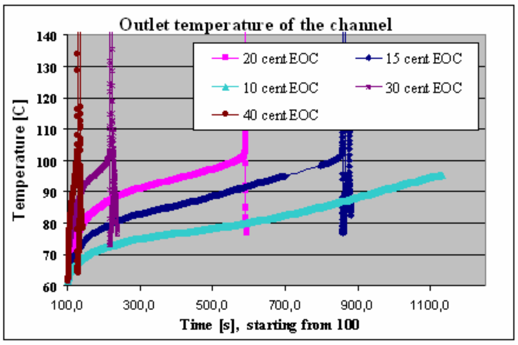

Fig. 12 Subchannel outlet temperature

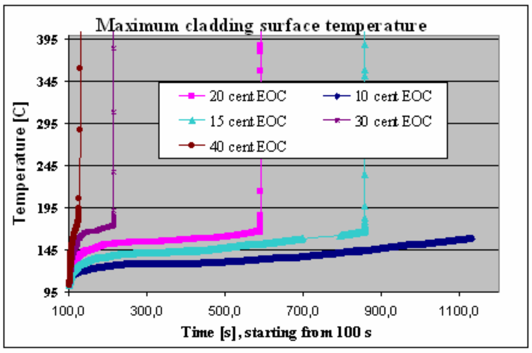

Fig. 13 Maximum cladding temperature

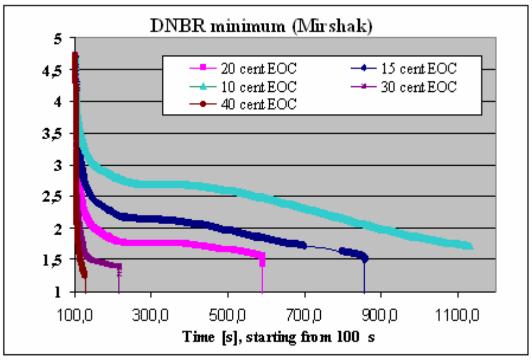

Fig. 14 DNBR

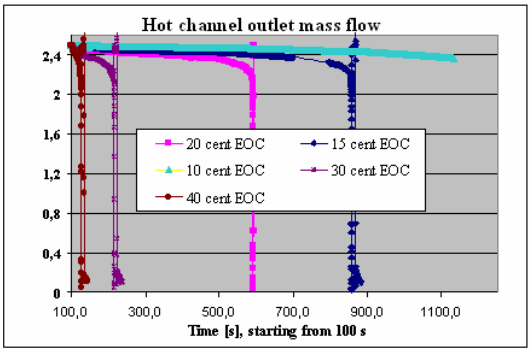

Fig. 15 Hot channel outlet mass flow

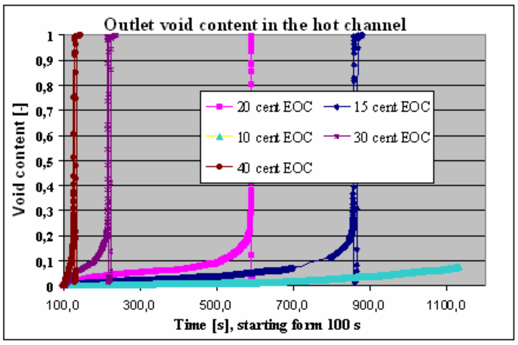

Fig. 16 Outlet void content in the hot channel

It was found, that at a given time, which is depending on the inserted reactivity, flow blockage in the most loaded parallel channels can lead to very high cladding and fuel temperatures. At this phase, DNBR - according to the Mirshak correlation - is also close to the unity. The time interval available for the operator action depends on the inserted reactivity according to Table 2.

Inserted reactivity [cent]	Time for the operator [s]
10	> 1000
15	757
20	491
30	115
40	27

Tab 2: Time interval available for the operator at different values of the reactivity insertion

Summary

Modernized tools, the coupled KIKO3D neutronics and ATHLET thermal hydraulic codes were used for the RIA and ATWS analyses of the Budapest Research Reactor. Both codes were validated against the startup and operational measurements of the research reactor, furthermore against special problems solved by other codes. The acceptance criteria were fulfilled for all the transients. For extremely low frequency RIA ATWS events it was found that the reactivity released due to the decreasing xenon poisoning, and the resulting positive feedback is not negligible. Because the maximum reactivity insertion due to the control rod movement is limited by a built in automatic mechanism, a suitable time interval for the operator action – obtained from the analyses - is available.

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