RRFM IGORR 2007

LYON, France 11.3. - 15.3. 2007

Transactions

11th International Topical Meeting

Research Reactor Fuel Management (RRFM)
and
Meeting of the International Group on Reactor Research (IGORR)

Centre de Congrès, Lyon, France
11– 15 March 2007

Organised by the
European Nuclear Society (ENS)

and

IGORR: International Group on Research Reactors

in co-operation with the
International Atomic Energy Agency (IAEA)

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These transactions contain all contributions submitted by 9 March 2007.

Session VI

Safety, Operation and Research Reactor Conversion

Session VI - Safety, Operation and Research Reactor Conversion

CONVERSION OF RESEARCH AND TEST REACTORS: STATUS AND CURRENT PLANS

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ABSTRACT

The Office of Global Threat Reduction's (GTRI) Conversion Program develops 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 27 years of existence, 46 research reactors have been converted from HEU to LEU fuels and processes have been developed for producing the medical isotope Mo-99 with LEU targets.

Under GTRI the Conversion Program has accelerated the schedules and plans for conversion of additional research reactors operating with HEU. Also the Program emphasizes the development of advanced high-density LEU fuels to enable further conversions. The Conversion program coordinates with the other program functions of GTRI, most notably the Removal function, which removes fresh and spent HEU fuel from countries around the world. 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 worldwide have been in operation for over 60 years, supporting nuclear science and technology development, as well as providing an important role as a research tool in scientific fields including medicine, agriculture, industry, and basic research. Over 270 research reactors are currently operating in more than 50 countries. Starting in 1954, many research reactors outside the United States were provided under the *Atoms for Peace* initiative. Initial research reactors were fueled with low-enriched uranium (LEU) with a content of U235 of less than 20%. More advanced research reactors desired higher specific power and neutron flux and, to avoid costs associated with the development of higher density LEU fuels, those reactors used high-enriched uranium (HEU) material, with an enrichment of 20% or higher, and typically over 90%, with the existing fuel designs. As HEU fuel became readily available, it turned into the usual fuel for research and test reactors, even for some that had initially operated with LEU fuel.

As worries increased over the potential use of HEU in the manufacturing of nuclear weapons, concern grew about the possibility of HEU-fueled research reactors becoming a source of the material. In response, the U.S Department of Energy 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.

A significant use of research reactors is in the production of medical isotopes, Molybdenum-99 (Mo-99) in particular. Although Mo-99 can be produced by neutron activation, it is more widely produced by fission of U^{235}, through the irradiation of HEU targets. A significant fraction of the HEU exported by the U.S. is for the fabrication of targets for the production of Mo-99. Therefore, in the mid-1980s the Conversion Program was extended to include, in addition to the conversion of research reactors, the development of technology for Mo-99 production with LEU material.

The Conversion Program was initially focused on U.S.-supplied reactors, but in the early 1990s it expanded and began to collaborate with Russian institutes with the objective of converting Russian-supplied reactors 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 is not only that of converting the HEU-based reactors and Mo-99 production processes to use LEU, but also to remove the HEU material from the facilities and provide a secure disposition. The Conversion Program has therefore been coordinating its activities with programs for the secure disposition of HEU material. These programs include GTRI's Removal program function, which coordinates the repatriation of U.S.-origin and Russian-origin fresh and spent research reactor fuel.

CONVERSION STATUS UNDER GTRI

The Conversion Program identifies 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 removal of the 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 collaborate with the United States on reactor conversion programs, and (3) requirements for very specialized LEU fuel which would be too costly and time consuming to develop.

Since the inception of the Conversion Program, 48 of the 129 reactors have been converted to LEU fuel or have shutdown prior to conversion. The list of converted or shutdown reactors is shown in Table 1.

Table 1, List of reactors converted or with conversion initiated

	Country	City	Reactor		Country	City	Reactor
1	Argentina	Ezeiza	RA-3	25	Netherlands	Delft	HOR
2	Australia	Lucas Heights	HIFAR	26	Netherlands	Petten	HFR
3	Austria	Vienna	TRIGA II	27	Pakistan	Rawalpindi	PARR-1
4	Austria	Niederosterreich	ASTRA	28	Philippines	Quezon City	PRR-1
5	Brazil	Sao Paulo	IEA-R1	29	Romania	Pitesti	TRIGA II

	Country	City	Reactor		Country	City	Reactor
6	Canada	Chalk River	NRU	30	Slovenia	Ljubljana	TRIGA MARK II
7	Canada	Montreal	SLOWPOKE Montreal	31	Sweden	Studsvik	R2-0
8	Canada	Hamilton	MNR McMaster	32	Sweden	Studsvik	R2
9	Chile	Santiago	La Reina	33	Switzerland	Wuerenlingen	SAPHIR
10	Colombia	Bogota	IAN-R1	34	Taiwan	Hsinchu	THOR
11	Czech Republic	Prague	Sparrow	35	Turkey	Istanbul	TR-2
12	Denmark	Roskilde	DR-3	36	USA	Ames IA	UTR-10
13	France	Sacley	OSIRIS	37	USA	Atlanta GA	GTRR
14	Germany	Berlin	BER- I I	38	USA	Charlottesville VA	UVAR
15	Germany	Geesthacht	FRG-1	39	USA	East Lansing MI	Ford
16	Germany	Juelich	FRJ-2	40	USA	Manhattan NY	MCZPR
17	Germany	Zittau	ZLFR	41	USA	College Station TX	NSCR
18	Greece	Athens	GRR-1	42	USA	Columbus OH	OSURR
19	Iran	Tehran	TRR	43	USA	Gainesville FL	UFTR
20	Japan	Ibaraki-Ken	JRR-4	44	USA	Lowell MA	UMLRR
21	Japan	Ibaraki-Ken	JMTR	45	USA	Narragansett RI	RINSC
22	Libya	Tajoura	Critical Facility	46	USA	Rolla MO	UMRR
23	Libya	Tajoura	IRT-1	47	USA	Schenectady NY	RPI
24	Mexico	Ocoyoacac	TRIGA Mark III	48	USA	Worcester MA	WPI

Under GTRI, DOE has established targets for the conversion of 129 HEU-fueled research reactors. The current goal is to convert the remaining 81 reactors in the list of candidates by the year 2018. Of the 81 remaining research reactors within the scope of the Conversion Program, 53 can be converted with existing LEU fuels, while the remaining 28 require the development of advanced high density fuels to allow their conversion. In this vein, a new very 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 on the development of advanced high density fuels, particularly U-Mo fuels, which will make feasible the conversion of these remaining 28 research reactors. The goal is the qualification of the advanced fuels by 2010.

It must also be noted that, with one exception, all new research reactors over 1 MW designed by Western countries since the inception of the Conversion Program have been fueled with LEU.

Increased security concerns in recent years have led to the establishment of the Global Threat Reduction Initiative (GTRI) by the U.S. Department of Energy's National Nuclear Security Administration. Secretary Abraham announced this initiative in May 2004. A follow up conference for the International GTRI

partnership at the IAEA in Vienna in September 2004 established the framework for international collaborations in meeting the goals of the program. The overall GTRI objectives include securing radiological material in addition to fissile material. The Conversion Program is an integral part of GTRI. ANL provides technical coordination for the entire program and Idaho National Laboratory provides the technical lead for fuel development.

The Conversion Program has also been coordinating 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 RERTR program from very early in the 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 legislation 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 burnup 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.

CONVERSION CURRENT ACTIVITIES

This year, the Conversion Program has undertaken activities towards the conversion analysis and fuel procurement of several facilities simultaneously. These include domestic as well as international activities and covers reactors of US and Russian origin.

The Program is also developing within the U.S. and under international collaborations with a number of partners high density LEU fuel and aiding in the development and demonstration of the basic technology for the production of Mo-99 with LEU targets. In the Mo-99 production with LEU, the Conversion Program is optimizing the effort to reduce the volumes in the target dissolution process and the minimization of waste streams.

CONCLUSION AND FUTURE DIRECTIONS

The overall objective of the Conversion Program is the reduction and eventual elimination of the use of HEU in civil applications. The Program develops the technical means (conversion analysis and high density LEU fuels) to enable the conversion of research and test reactors that use HEU fuel, and LEU target and process technology to make possible the efficient production of Mo-99 without the use of HEU.

In order to accomplish these goals interaction is occurring with multiple facilities and analysis is being initiated for the conversion of multiple reactors. Progress is being made in the collaboration with Russia in the fuel development as well as in the collaboration for the conversion of Russian-supplied reactors.

The Conversion program has the ability to establish incentives for accelerating the conversion of research reactors. Domestically, NNSA can purchase the LEU fuel for the

university reactors thus facilitating the scheduling of the conversion. Internationally, it is possible to provide an incentive in the form of LEU fuel supply with an equivalent lifetime to that remaining in the HEU fuel it replaces. Coordination with the Remove function of GTRI allows the repatriation programs the establishment of incentives in the form of return of spent fuel or supply of LEU fuel in exchange for return of fresh HEU fuel.

In the next few years the Conversion Program is expected to accelerate further, as many reactor conversions need to occur annually to meet the GTRI schedules. The technical efforts to establish agreements with the reactor operators, and the development and procurement of fuel will increase rapidly to meet the challenges. This will require policy efforts to approach facilities that have not joined the conversion effort as well as technical efforts to develop a conversion approach for reactors that are technically more challenging.

SAFARI-1: ADJUSTING PRIORITIES DURING THE LEU CONVERSION PROGRAM

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ABSTRACT

In July 2005, the South African Department of Minerals and Energy authorised the conversion to Low Enriched Uranium (LEU) of the South African Research Reactor (SAFARI-1) and the associated fuel manufacturing at Pelindaba. At that stage the proposed scheduling allowed approximately three years for the full conversion of the reactor, anticipating simultaneous manufacturing ability from the fuel production plant.

Initial priorities and regulatory agreements were allocated with the intention to manufacture and produce two Lead Test Assemblies (LTAs) from the Pelindaba plant (Phase I) and use these as qualification of manufacturer as well as initiation of the SAFARI-1 conversion (Phase II). Delays in the demonstration of sufficient confidence in the manufacturing ability to enable local fuel licensing and qualification have resulted in minor readjustments of these Phases.

Delays in the initial schedule that allowed for the insertion of the two South African LTAs during the 1st quarter of 2006 were pre-empted by the acquisition of 2 LEU silicide elements of SA design manufactured by AREVA-CERCA. These two LTAs are currently undergoing testing in SAFARI-1 and have to-date completed up to 8 cycles of irradiation.

As a further precaution to the potential delays in the fuel-manufacturing Phase, a reload (760 plates) of LEU silicide element fuel plates were purchased and will be assembled locally to enable the SAFARI-1 conversion program to continue according to schedule.

This paper will trace the developments of the above in order to reflect the current status and the planned correlation of the Phase I and Phase II programs according to latest expectations.

1. Introduction

The development of the South African nuclear industry has been comprehensively reported at recent RERTR and other international conferences. In particular, the availability of large resources of natural uranium, the formation of the Atomic Energy Board (AEB) of South Africa, the establishment of the 1st South African Fundamental Atomic Research Installation (SAFARI-1) at the South African Nuclear Energy Corporation (Necsa) site at Pelindaba and the reactor's subsequent first criticality on 18 March 1965, have been well documented [1,2,3,4].

SAFARI-1, initially a 6.67 MW tank-in-pool type light water reactor, based on the Oak Ridge Reactor (ORR), was purchased from the USA but was soon modified to enable operation at 20 MW. The reactor is currently capable of functioning at 30 MW [6] but operational levels are maintained at a maximum of 20 MW, pending regulatory authorisation. The reactor was initially fuelled with Highly Enriched Uranium (HEU) sourced from the USA and elements manufactured either in the USA or the UK. In later years (post 1981), the reactor has been fuelled solely with HEU allocated from the South African HEU inventory (45 and/or 93%). At the same time, target plates required for a now well-established ^{99}Mo production programme at Necsa are also manufactured from this original SA HEU inventory (45%).

2. SAFARI-1: The Role in Necsa's Isotope Production and R&D Programmes

SAFARI-1, which is owned and operated by Necsa on behalf of the Department of Minerals and Energy (DME) is currently utilised mainly as a client service to perform irradiations for NTP Radioisotopes (Pty) Ltd (NTP) for the production of radioisotopes for medical application (national and export) as well as for the production of Neutron Transmutation Doped (NTD) silicon.

There are also pneumatic and fast pneumatic systems utilised for Neutron Activation Analysis (NAA). Utilisation of beam-ports for institutional (academic) purposes is encouraged and Neutron Diffraction and Neutron Radiography facilities are well utilised, whilst a Small Angle Neutron Scattering (SANS) facility, subsidised by the IAEA, is under development.

In support of safe operation and the commercial needs of NTP, SAFARI-1 applies an integrated management system, incorporating Quality, Health, Safety and Environment (QHSE). The reactor's Quality Management System (QMS) is fully certified according to ISO 9001 (2000) and it implements an incorporated Environmental Management System (EMS), fully certified according to ISO 14001 (2004) [5]. The current licence of SAFARI-1, as authorised by the National Nuclear Regulator (NNR) endorses operation of the reactor to 2020, but requires assurance that the proposed operational plan is justified, not only by the current safe operation of a well utilised RR but also by establishment of a longer term sustainability plan.

3. Conversion Strategy: HEU to LEU in View of Operation and Commercialisation

As reported earlier, the DME (July 2005) authorised the conversion to Low Enriched Uranium (LEU) of SAFARI-1 and the associated fuel manufacturing at Pelindaba over a period of approximately 3-4 years [3].

The original strategy to align SAFARI-1 as a multipurpose semi-commercial facility today provides the backbone of a strong medical isotope supply facility for SAFARI-1's major customer – NTP Radioisotopes (Pty) Ltd. This Necsa subsidiary is subsequently responsible for providing the major source of the reactors operational income by processing and distributing, amongst others, fission product isotopes (e.g. ^{99}Mo, ^{131}I) for medical applications. This successful marketing achievement currently positions SAFARI-1 as one of the top 5 reactors supplying services to producers of ^{99}Mo internationally. SAFARI-1 has a subsequent responsibility to ensure continuity of good quality supply of irradiated products and services.

As a result of commercial requirements, the reactor currently operates on a cyclic programme of ~5 weeks full power operation at 20 MW and shuts down for essential maintenance and fuel reload/reallocation over a period of 3-4 days. The resultant ~312-317 FPD operation implies a demanding average availability in excess of 84%.

These high commercial expectations in terms of product supply and operational efficiency require a reliable continuity of provision of quality isotopes to the medical industry, both for the well-being of fellow humans as well as for the financial sustainability of the reactor.

4 Postulated Conversion of SAFARI-1

4.1 The Impact of LEU Conversion – Operational and Commercial

Earlier theoretical postulations modelled on the current HEU utilisation, indicated operational efficiency losses of ~8%, with slightly smaller penalties in the fast-to-thermal flux ratios for the LEU conversion. The latter could impact on the levels of utilisation for the irradiation services such as fission isotope production and NTD of silicon. In view of the DME's authorisation to progress with the conversion of SAFARI-1, together with provision of the necessary funding to ensure satisfactory conversion of both the reactor and the manufacturing process over a period of ~3-4 years, the conversion project was initiated during 2005.

The project was split into two major phases for regulatory purposes:

Phase I Establishment of a qualified local fuel (LEU) manufacturing ability; and
Phase II: Transition of SAFARI-1 core from HEU to LEU Fuel.

4.2 Phase I: Manufacturing Ability

As indicated above, all fuel supplies for the operation of SAFARI-1 after the mid 80's were sourced using local HEU and from assemblies (fuel elements and control rods) manufactured at Pelindaba.

The technology applied had been developed and established locally but was based on the ORR fuel design criteria, using initially 45% and then later 90% ^{235}UAl alloy. The first assemblies (19 flat-plate) had HEU loadings maintained at 200g - ^{235}U but were later modified to 300g - ^{235}U per assembly. In terms of this, the equivalent loading of 340 g ^{235}U per LEU assembly - corresponding to a uranium density of 4.8 gm/cm^3, maintaining the same geometric profile - was confirmed as feasible.

Typical challenges experienced during the local manufacturing development program resulting in minor delays in this phase of the conversion programme, are elaborated on at this and previous conferences [7,8].

The manufacturing test and qualification programs are currently scheduled for completion and supply of the 1st local Lead Test Assemblies (2 LTAs) during 2007.

4.3 Phase II: Preparation for SAFARI-1 Conversion

Due to the delays experienced in the manufacturing conversion program and in view of the understood commitment to ensure that the conversion of the reactor takes place as postulated, it was agreed for regulatory purposes that two approaches would be used:

- Demonstration of the ability of the core management processes to predict the impact of LEU addition to the core in terms of both operational and commercial efficiency by gradual transition, i.e. selectively starting with one and gradually adding more LEU assemblies. This is a combination of benchmarking the existing core management software (SAFI-2000 and OSCAR-3) against experimental measurement (flux wires at each interim fuel cycle):
    - Two LTAs were imported from AREVA-CERCA and were installed, under conditional regulatory requirements from the NNR, into the SAFARI-1 core during January 2006.
    - The two LTAs have to-date completed 8 cycles of irradiation and have achieved ~60% burnup on a predicted End-of Life (EOL) of ~70%.

- The LTAs are being visually examined between cycles (during the shutdown period) and individually validated in terms of integrity of gap measurements for 12 of 18 channels.
- Predetermined acceptance criteria (deviation of less than 0.1 mm from manufactured specifications) for unrestricted reinsertion of the LTAs into the next cycle have been established with the NNR - the required authorisation is provided by the SAFARI-1 Reactor Safety Committee. To-date all results have been in full compliance with the specifications.

- In view of the delay in the manufacturing program qualification, a core of LEU silicide plates (760) has been acquired from AREVA-CERCA. Assembly qualification, using these plates and locally manufactured components, is proceeding. It is anticipated that the first fuel assemblies will be ready for insertion into SAFARI-1 early in 2007.

- The final approach remains unchanged to that reported earlier, viz. the demonstration, using the imported fuels as benchmark, of the suitability of the locally manufactured fuel. For this purpose, as mentioned, the first of the SA LTAs is expected to be loaded during 2007, followed by successive local LTAs according to regulatory authorisation and as available.

In both cases, international and locally manufactured LTAs, the benchmarking will consist mainly of inter-cycle monitoring of the fuel condition, i.e. visual and gap-measurement verification of the cooling channels as set out above.

4.4 Regulatory Expectations Regarding SAFARI-1 Conversion

It is not expected that there will be any major operational deviations during the reactor conversion process – this is supported by communications with management of the sister reactor HFR at Petten and their experience as reported elsewhere [9]. As previously discussed, however, the conversion must be done systematically in a controlled manner that ensures optimum utilisation of the South African HEU inventory and at the same time guarantees the continuity of quality service to clients, particularly in the field of isotope supply. This requires good coordination of the systematic conversion of the reactor together with an acceptable licensing approach.

The following regulatory authorisations have been (or are being) negotiated:
- Initial irradiation of the two CERCA LTAs to demonstrate compatibility of the LEU with the current HEU core during conversion;
- Irradiation of successive additional LTAs either of South African origin after qualification of the local manufacturing process and/or of South African assemblies using CERCA manufactured plates;
- Systematic conversion of SAFARI-1 to LEU Fuel assemblies over a period of the next 3 years – this will require a significant revision of the current Safety Analysis Report [6] to incorporate thorough reapplication of the relevant risk analyses and transient and accident conditions analyses using e.g. the thermal-hydraulic code RELAP.

In general, the continuity of operation of the reactor should not be unnecessarily challenged, either by quality or financial efficiency. This implies that any inability to continuously supply the reactor with good quality locally or internationally manufactured fuel, due to possible fuel failure, must be matched according to schedule and finances of alternative supplies.

Furthermore, deviations from an operational schedule should not have significant negative impacts on supply of service to stakeholders.

Secondly, the fuel inventory should be utilised to optimise efficiency, i.e. ensure fuel discharge burn-up is in line with current HEU levels of utilisation (~60%). This requires selective matching of current HEU fuel inventories, local fuel manufacturing schedules for supply and the selective backup of international suppliers, which in this case, due to the efficiency of the current UAl manufacturing plant (Pelindaba) could impose a significant financial penalty.

5. Conclusion

The conversion of the South African research reactor SAFARI-1 and the related local fuel manufacture to LEU utilisation was authorised by the Department of Minerals and Energy (DME) in July 2005. The conversion has proceeded to the stage where the manufacturing qualification of the local facilities, although delayed somewhat due to technical complications, is imminent (during 2007). At the same time, in order to ensure that systematic conversion of the reactor is feasible, the purchase and irradiation of 2 Lead Test Assemblies from AREVA-CERCA proceeded during 2006. Currently these LTAs have successfully completed 8 irradiation cycles and ~60% burnup. Further backup inventory has been acquired by the purchase of a reload of LEU plates for local assembly and utilisation in SAFARI-1 as may be required should further local manufacturing qualification delays be experienced. Regular utilisation of LEU in SAFARI-1 is expected over a transition period of ~3 years – under regulatory authorisation, which will require revision of the current Safety Analysis Report and review of the applicable risk assessment and transient applications.

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RESULTS OF 14MW RESEARCH REACTOR CORE CONVERSION MEASURED AT LOW POWER

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ABSTRACT

The full conversion of the 14MW TRIGA Research Reactor was completed in May 2006, thanks to international cooperation and commitment of Romanian scientists. The conversion was achieved gradually, starting in February 1992 and going on step by step in August 1996, March 1998, October 2000, March 2004 and completed in May 2006. Each step of the conversion was achieved by removal of HEU fuel, replaced by LEU fuel, accompanied by a large set of theoretical evaluation and physical measurements intended to confirm the performances of gradual conversion. After the core full conversion, a program of measurements and comparisons with previous results of core physics and measurements is underway, allowing data acquisition for normal operation, demonstration of safety and economics of the converted core. Low-enriched uranium TRIGA fuel elements behavior in continuous utilization is of special interest due to material differences between the constituents of high uranium content in the alloy and technological variables – as compared to the initial design of this fuel.

1. INTRODUCTION

Characteristic for the last decades, it is obvious the tendency of decreasing the research infrastructure for nuclear power development due to the lack of new human resources and to the depreciation of a number of operating research reactors.

The 14MW TRIGA research reactor, operated by the Institute for Nuclear Research in Pitesti, Romania, is a relatively new reactor, commissioned 26 years ago. It is expected to operate for another 15-20 years, sustaining new fuel and testing of materials for future generations of power reactors, supporting radioisotopes production through the development of more efficient new technologies, sustaining research or enhanced safety, extended burnup and verification of new developments concerning nuclear power plants life extension, to sustain neutron application in physics research, thus becoming a center for instruction and training in the near future.

The pillars for the future utilization of the TRIGA Research Reactors and of the Post-Irradiation Examination Laboratory in Pitesti, Romania, are the following:

I – Safety, Reliability and Availability

-- a proved safety of the 14MW TRIGA-MTR;

-- high flexibility of experimental and testing programs application correlated with post-irradiation laboratory;

-- project of power increase to 28-30 MW in order to achieve a flux of 3-3.5x10^{14}n/cm^2s.

-- full core conversion (done in 2006);

-- existence of an European reliable fuel manufacturer;

-- complementary utilization of Annular Core Pulse Reactor ((TRIGA-ACPR) for special safety experiments;

-- no major issues concerning the spent HEU fuel return in the country of origin and solutions until the horizon of 2019 for LEU fuel;

-- a large refurbishing and modernization program undertaken to cope with ageing and obsolesce of equipment and to satisfy the actual requirements in terms of safety and reliability which will be accomplished during the next year.

II – International cooperation and utilization for research on the development of new materials for power reactors

III – Increasing energy demand

The full conversion of the core was a necessary step to ensure the continuous operation of the reactor. The core conversion took place gradually, using fuel manufactured in different batches by two qualified suppliers based on the same well qualified technology for TRIGA fuel, including some variability which might lead to a peculiar behavior under specific conditions of reactor utilization.

In order to survey and prove the performance of TRIGA 14MW LEU fuel, a program of measurements and comparisons, using previous results of HEU and Mixed Core, is performed over the entire converted core.

2. Technical Objectives of Conversion

The objectives of reactor core management in the process of conversion were to ensure safe, reliable and best use of existing HEU in core until its complete removal from the core and, at the same time, to apply identical general requirements to LEU fuel, perform the operational tests on the whole converted core.

The testing program for a complete demonstration of parameters of the first LEU core (now under development) provides a series of tests and measurements at progressive levels of power: 2MW, 5MW, 10MW and at full power, i.e., 14MW.

The full converted core configuration [Fig.1] takes into consideration the previous experience in the utilization of mixed core and the safety limits requirements, previously approved. The requirements are presented below:

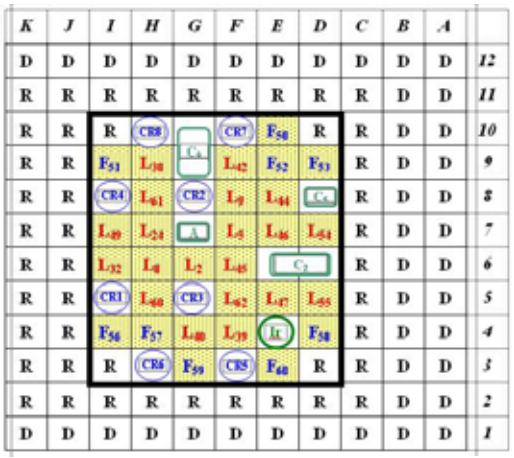

Fig 1. The full converted core configuration

-- Number of LEU fuel assemblies in core: 29;
-- Fulfill the initial concept of core design to have one-side-drive, containing control rods and another side experimental vertical irradiation channels;
-- Reactivity of bank control rods roughly equal to the double of core reactivity, to ensure flat power distribution;
-- To satisfy the criteria of safe shutdown with most effective control rod blocked out of core;
-- Maximum temperature in fuel rods (central temperature) should remain within the limits of the previously approved value at full power (14 MW);
-- To ensure a sufficient number of in-core vertical irradiation places in order to increase availability and utilization;
-- To demonstrate practical results of the entire program related to TRIGA-14MW core conversion, starting with initial design, analysis and first fabrication of LEU-TRIGA fuel until full conversion of the core, which will result in fitness for fuel service in agreement with the utilization program maximum burnup / less spent fuel and “O” cladding defects;
-- To provide input for the qualification of fuel for storage.

3. Results of TRIGA-LEU core analysis and neutron flux determination at low power (2 MW)

The previous analysis of core physics, considering the increase of U-235 and less Erbium, provides a design of 29 fuel assemblies with a higher reactivity, using only fresh LEU fuel assemblies.

Due to the gradual conversion of the core, started in 1992, some 60% of fuel assemblies record an average burnup of 35%. Distribution of fuel assemblies in the core configuration, considering used fuel and fresh fuel, was subject to an initial design and analysis, in order to comply with the above criteria.

The computer codes and library used for these analyses were developed and verified in the past, during gradual conversion. For the LEU fuel supplied by CERCA – France for full conversion, the isotopic composition is given at the level of each fuel rod pellet, which allows a more detailed analysis as compared to the previous data, where the isotopic composition was known as a mean per fuel rod/assembly. Considering the above data, 11 fresh fuel assemblies were selected for the full conversion of the core. The fresh fuel was located at the periphery of core fuel assemblies, referred as F_n in Fig. 1. The prompt temperature coefficient of reactivity computed for the fully converted core is 10% higher than for HEU core at the beginning of the fresh core life. This coefficient will decrease with burnup and with Xe-135 poisoning, but remains higher than the HEU core coefficient, able to control safety in case of fast reactivity transients. The delayed neutron fractions β in the instances presented above are practically identical differences: below 1%. Following the analyses results, another parameter displays a significant modification: the lifetime of prompt neutrons depends on core burnup and configuration of experimental channel.

Power peaking factors are identified in each fuel pin by using tri-dimensional DFA computer code and POW for proper selection of instrumented fuel pins location in core the maximum power peaking factor value is 2.127.

Using the power peaking factors the maximum clad temperature and fuel centerline temperature are determined by using PARET computer code; the maximum temperature in fuel center is 617ºC and clad 114ºC at 14MW power. Lifetime of the designed core configuration will be 2 to 2.5 years for a utilization of 5500 hours/year. The forecasted refueling may occur at the beginning of 2009 with 2 or 3 fuel assemblies removing similar amount spent fuel assemblies from core center (Fig. 2).

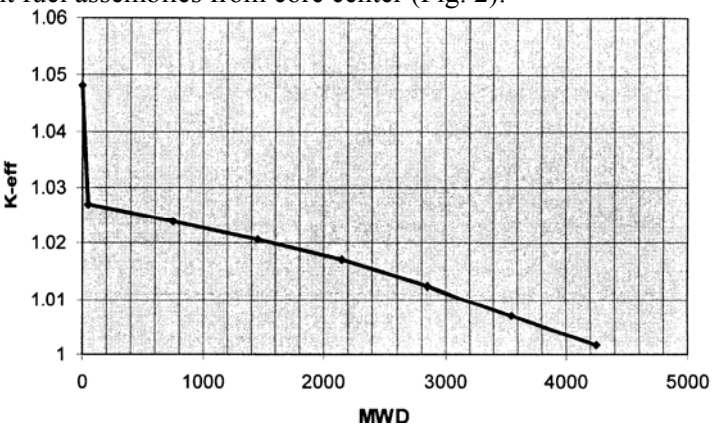

Fig 2. First LEU core lifetime

4. Thermohydraulic Analysis

The configuration of the converted core with LEU fuel is similar to the configuration of the initial designed HEU core, in terms of number of fuel assemblies, number of control rods, geometrical dimension of fuel assemblies and fuel pins, pitch, designed flow, coolant channels and water temperature. Some of the features are still different between these types of fuel concerning fuel "density" (specific weight), thermal capacity and thermal conductivity of material. The above considerations lead to a similar behavior of the initial core within the normal range of operation: the maximum temperatures of the clad and fuel are similar and DNB 2.7 – 2.8.

The RELAP5 computer code was extensively used for transient analysis concerning Loss of Flow Accident (LOFA) and Reactivity Insertion Accident (RIA).

LOFA results, considering main pumps and emergency pumps shutdown, do not necessarily present differences between HEU and LEU and the natural convection mechanism of residual heat removal is safe to prevent any fuel and clad temperature increase. RIA results, considering typical accident of TRIGA-14MW Safety Analysis Report initial power 1W, reactivity insertion 1% in 0.3 seconds for fresh cold core. The behavior of LEU reactor core presents a peak power of 230 MW and a fuel temperature of 286ºC in comparison to HEU fuel, where power peak is 550 MW at a fuel temperature of 810ºC. The difference is determined by prompt temperature coefficient of reactivity, higher for LEU fuel and with better performances in this case of LEU.

5. Neutron Flux Determination at Low Power

The most controversial penalty of research reactor core conversion concerns the neutron flux in some reference in-core locations, where analytical determinations and previous measurements in HEU core or mix core were performed and considered as reference.

Measurement of neutron flux axial distribution: the 14MW TRIGA core is a relatively short core (57.5 cm) and it is well reflected by a complete lateral beryllium reflector, but in vertical direction it is not reflected. The axial distribution is similar to the cosine shape of many other reactors, with some non-uniformities and an asymmetrical maximum and a peculiar ratio between the maximum and the mean value of thermal neutrons and a peculiar value of ratio thermal over fast neutrons.

The relatively large number of efficient control rods operating in core, partially inserted at each level of power, may disturb the axial distribution and the value of the flux in the entire core.

For the scope of axial profile determination in all accessible locations, mostly in irradiation channels, in the core center and in other places – as beryllium channels or inside the fuel assemblies – an experimental device equipped with a mobile self-powered neutron detector is currently used for such determinations or for local monitoring of some non-instrumented in-core experiments.

The local flux perturbation of measuring device is not significant. The diameter of stainless steel cladding is 1.4 mm, the common shape of axial

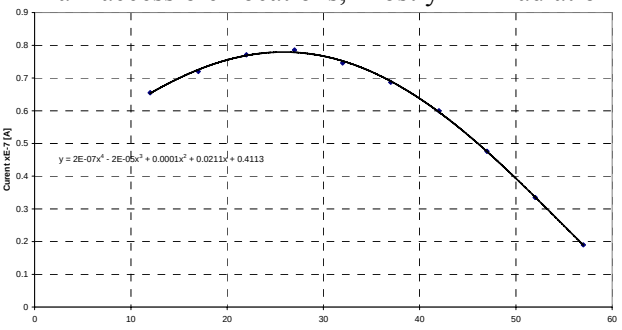

Fig 3 Axial thermal neutron flux distribution, P=1MW

thermal neutron flux distribution at P=1MW is shown in Fig. 3

Flux spectrum determination in central irradiation channel is based on techniques of selected neutron multi foils activation. Irradiation data were processed by using dedicated instruments. During activation of foils the reactor power remained at the pre-selected level, the control rods in a defined configuration while the water and fuel temperature remained stationary.

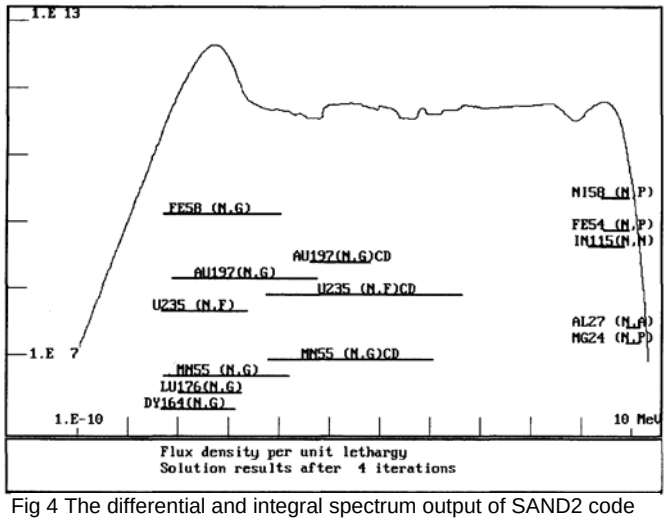

Fig 4 The differential and integral spectrum output of SAND2 code

The entire process of flux spectrum determination is subject to a measuring procedure where all parameters are kept under control: irradiation time, structure of isotopic composition of sets of detectors, decay time, gamma spectrometry settings, calibration data acquisition, input data for unfolding, etc. The differential and integral spectrum output of SAND2 code on 621 energy groups for central irradiation channel are shown in Fig. 4

Due to the special core configuration after full conversion, containing previously used fuel in the central area of the core, some flux flattening occurs. A flux decrease in the center of the core with regard to measurements performed with new fresh HEU core in 1980 and measurements performed in October 2006, can be discussed as 8% but, at the same time, now the thermal flux is increasing in other irradiation channels as 3-4%. The controversy on real values of flux spectrum cannot be settled now and, in fact, few percents regarding many other advantages have only theoretical value.

The core burnup after the full conversion is highly heterogeneous, for four fuel assemblies L38, L39, L40, L42 are in core since February 1992 and the burnup is higher that HEU fuel assemblies by 3-5%. The refueling of the core with LEU fuel assemblies continued in 1996: 3 F.A., 1998: 4 F.A., 2004: 3 F.A., and 2006: 11 F.A. thus completing the conversion. The comparison of LEU fuel with HEU fuel rods with regard to dimensional stability at high burnup is useful in the evaluation of overall results of conversion, including economics.

6. Continuous Evaluation of TRIGA-LEU Fuel Behavior during Irradiation

The behavior of TRIGA-LEU fuel during expected long duty cycle, more than 15 years in-core, accumulating a high burnup, over 43% of the initial U235 with a large number of power cycles (350-500), is difficult to be assessed by analysis and simulation.

The continuous survey of the irradiation conditions, operation history duty factors and extensive inspection and post-irradiation examination allow the consolidation of an entire set of data that sustain the safety of utilization of this fuel and prevent fuel failure.

This becomes also important because now, in the fully converted core, three batches of LEU fuel are utilized, 2 batches manufactured by General Atomics and one (fresh fuel) manufactured by CERCA – France. The references for LEU fuel behavior are the result of previous HEU utilization. The extensive in-service examination of the fuel is rendered feasible by the conditions within the reactor and Hot Cells Facility. The 14MW TRIGA reactor core is located in a large pool connected to the post-irradiation examination hot cells by an underwater transfer channel. The design allows easy and safe handling of fuel and installation of the additional examination equipment. In the reactor pool the clean primary water, free of contaminant fission products, the on-line and off-line monitoring system allows instant identification of minor clad defects – if any. The permanently installed pool-side devices allow a rough control of all fuel rods elongation, bending as well as visual inspection. The permanently installed under-water neutron radiography facility allows radiographic inspection of some selected fuel rods. The under-water gamma scanning (temporarily installed) allows burnup determination of each fuel rod in inspection campaigns, to confirm the in-core analytical power distribution, to produce a large set of information for fuel management, with results in fuel economy and safety, and also allows the spent fuel qualification before packing and shipment to the country of origin.

The post-irradiation laboratory allows a highly accurate examination of LEU fuel by non-destructive and destructive examination, as follows:

-- direct visual inspection, through magnifying periscope, with digital photography;

-- profilometry of fuel elements, diametral increase, ovality, local asymmetry, swelling, bending, axial relative data distribution;

-- gamma scanning and tomography: burnup axial distribution, peaking factors;

-- destructive examination, plenum pressure on composition, metallography of fuel cladding, mechanical properties of cladding.

The entire set of data processing from pool side examination and in hot cells results are used for the optimization of fuel utilization and determination of local conditions in the irradiation channel for in-core experiments.

From the first batch of LEU fuel 18 fuel elements have been selected for periodic examination, following the methods listed above.

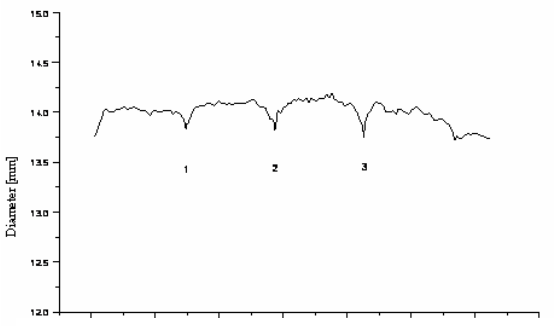

Fig 5 Distribution of values of the mean diameter along the fuel rod - LEU

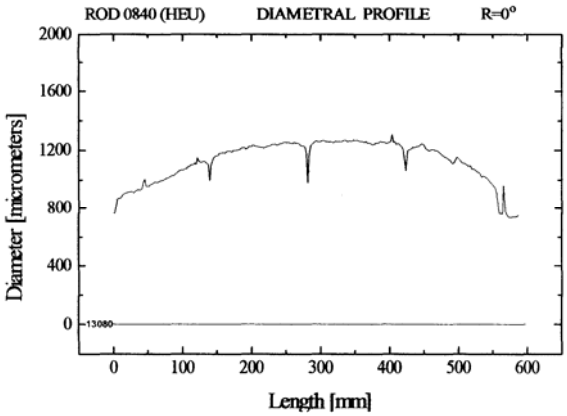

Fig 6 Distribution of values of the mean diameter along the fuel rod - HEU

The periodic examination was performed in 1993, 1995, 1996, 1997, 2000, 2001 and 2006. At the beginning of the examination some local swelling (protuberances) occurred on the clad surface. Further examination showed that this was only at the beginning number of protuberances remaining constant. It was found that this will not be harmful for fuel behavior. The maximum diametral increase of fuel rods with high burnup was 2.4%. Fig. 5 shows the distribution of values of the mean diameter along the fuel rod. The smaller diameters are associated to fuel-pellets interfaces. For comparison, the diametral profile of one HEU fuel elements presented in Fig. 6. Fuel rod image with pellet interface is presented in Fig. 7.

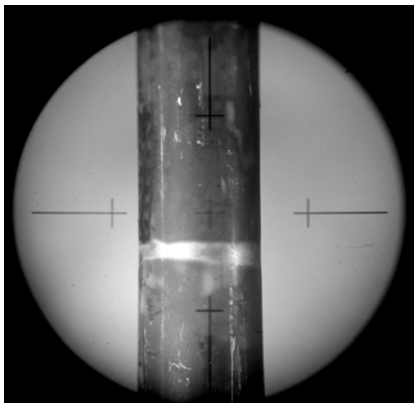

Fig 7 Fuel rod image with pellet interface

Burnup determination was performed for similar fuel rods selected for dimensional control. The burnup distribution, power profile and peaking factors are similar foe LEU and HEU fuel elements. For long term measurement and comparison the fission product Cs-137 was selected to avoid the incertitude produced by accumulation and decay of other fission products. Figs. 8 and 9 show similarities and differences produced by manufacturing technology.

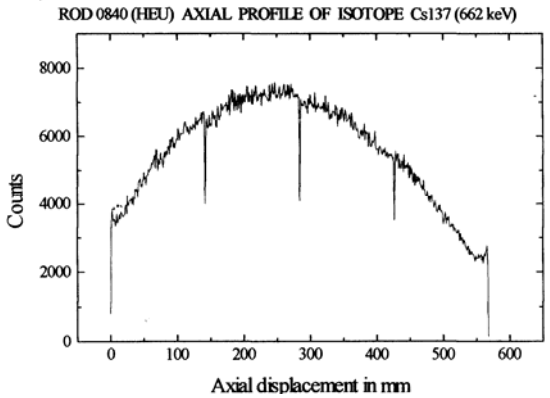

Fig8. Axial profile of gamma activity for Cs137

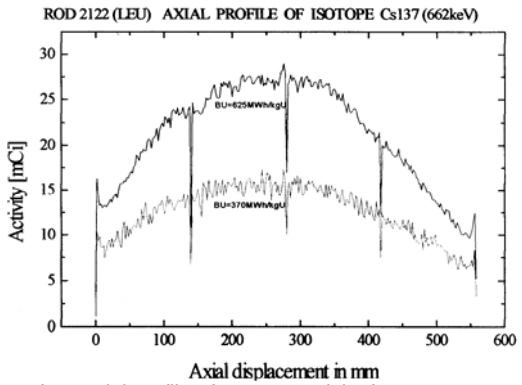

Fig9. Axial profile of gamma activity for Cs137

The TRIGA nuclear fuel is a hydrided Uranium, Zirconium and Erbium alloy. The process produces a ZrH1.65 matrix with a dendritic structure alpha - phase of uranium being located on the limits of zirconium hydride grains. The difference is that uranium thickness in HEU fuel is 1μm and 5 μm in

LEU fuel, producing a quasi-continuous uranium matrix. Above 330ºC this matrix allows the transport of hydrogen from the high temperature area inside the fuel rod to the lower temperature. The increased hydrogen concentration is accompanied, at lower temperature, by phase transitions produced by H/Zr ratio from initial delta to alpha + delta or epsilon - phase, which may result in an increase of the alloy volume. Some modifications of the fuel element profilometry as compared to HEU profile can be associated to these phenomena.

Destructive examination of LEU fuel

The outer diameter of the fuel cladding was 13.7 – 13.8 mm, clad thickness 0.40 mm without signs of internal or external corrosion. Fuel swelling consumed completely the radial gap, producing a mechanical bonding. Due to the high ductility of incoloy 800 the clad will reproduce all internal modifications of fuel swelling, allowing the measurement of diameter variation. The circumferential cracks located at clad proximity can be associated with specific volume transformation due to phase modification or a high stress area produced by temperature gradient (Fig. 10).

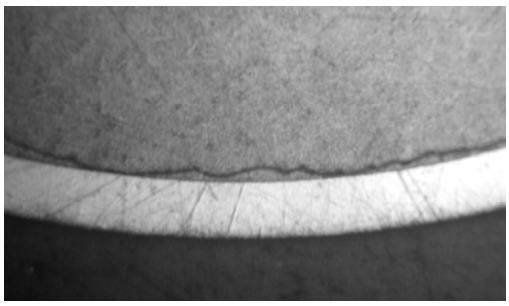

Fig10. Fuel circumferential cracks located at clad

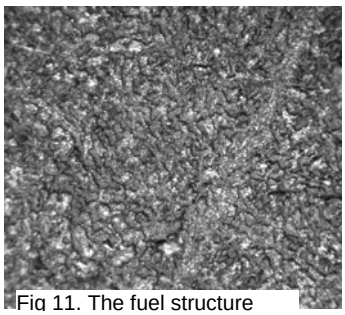

Fig 11. The fuel structure

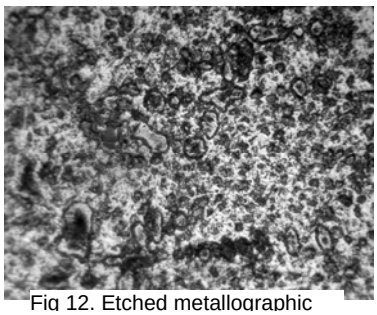

Fig 12. Etched metallographic

On the other hand, the microstructure of a central area is representative for metallic alloy operating at high burnup and high temperature, the micro-pores accommodating the fission products. The fuel structure is not affected by normal porosity (Fig. 11). An etched metallographic sample (Fig. 12) shows the internal micro-structure of delta-phase of zirconium hydride and a fine alpha structure of uranium dispersion.

Some non-homogeneities are recognized at pellets edge where, due to fast cooling during pellets manufacturing, some of zirconium is not completely melted or is segregated (Fig. 13). These aspects need further effort for analysis and correlation.

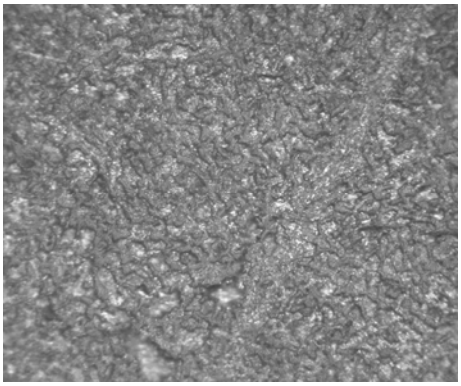

Fig 13. Fuel structure (X100)

7. Conclusions

The conversion of the 14MW TRIGA reactor core was successfully accomplished throughout a relatively long period of time. The gradual conversion allows the accumulation of a large amount of experimental data which, to some extent, prove and confirm the results of previous analyses which funded the conversion. A careful approach of full converted core, based on continuous evaluation of analysis and experimental data, will allow the progressive increase of power and reactor operation, with demonstrated safety margins in terms of operation and fuel behavior.

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Some Requirements For The Conversion of the Syrian MNSR Core

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To be presented at the RRFM-2007
Lyon (France), 11-15 March -2007

Abstract:

The conversion of the core of the Syrian MNSR from the use of HEU to the use of LEU would require some changes in the values of both the flux and the total thermal power of the reactor. Some dispersion and ceramic fuel types are considered.
In this paper a comparison between the neutronic parameters of the core for the reference case and the other cases, in which the LEU fuel is used, is done and differences are emphasized.

KEYWORDS
Reactor, MNSR, Comparison, LEU, HEU, Fuel, Core.

1. Introduction

Some studies have been performed on the conversion of the core of the Syrian MNSR[1-4]. These studies considered some fuel types like the dispersion ones in general (U-Al_x-Al) [2-4], besides to UO_2 fuels [1]. The general conclusion for the dispersion fuel types was that these fuels have low densities so that special configurations of the core should be used (the reflector characteristics are essential to the adjustment of the initial excess reactivity). An other configuration has been considered [4] in which a mixed fuel (some rods contain HEU fuel, and others containing LEU fuel) was employed. In the case of the UO_2 fuel, different results were obtained [1,5]. In a previous work of some colleges [1] a UO_2 fuel with 5.45 g content of ^{235}U/fuel element was used. The paper indicated a configuration in which only 199 fuel elements were necessary to have about 4.579 mk for the initial excess reactivity. In other works [5] two types of UO_2 were considered : the UO_2 as a dispersed fuel, and a ceramic pellet fuel fabricated by Zircatec (CANADA). In the following a comparison between the different solutions is made.

2. Methodology

Since the approach to the calculations of a new fuel would require principally the quantity of uranium and the total number of fuel elements it would be convenient to adopt a model of the reactor(and in ;particular of the core) that considers the conservation of matter in the copre rather than a very detailed model in which the single pins are to be described.
A very simple of the reactor has been constructed (see Fig. 1). The reactor is formed of the core (central gray zone), the annulus reflector (side purple zone), the

bottom reflector(bottom purple zone), the tank wall(external blue zone), the internal irradiation sites (two parts: the yellow zones inside the annulus reflector and the underlying dark purple zones), the upper grid (brown zone at the upper end level of the annulus reflector), the shim tray base (dark purple zone lying above the upper grid with a thin layer of water in between, and the control rod (the black zone in Fig.1).
The external irradiation sites, the upper, bottom, and lateral frames were eliminated. The basement was eliminated as well. Other less important components were eliminated too.
The Codes WIMSD4 [6] and CITATION [7] are both used here as a cell and core-calculation codes, respectively. The same number of neutron groups and the same limits of these groups, which pertain to the previous models [9], are adopted here.
This model run faster than the model described in [9] for the diminished no. of components forming the reactor. The saving in the running time would be about 30% (this saving refers to the total saving which comprises the time of running WIMSD4 for the cell calculations, plus the time of running CITATION for the core calculations).

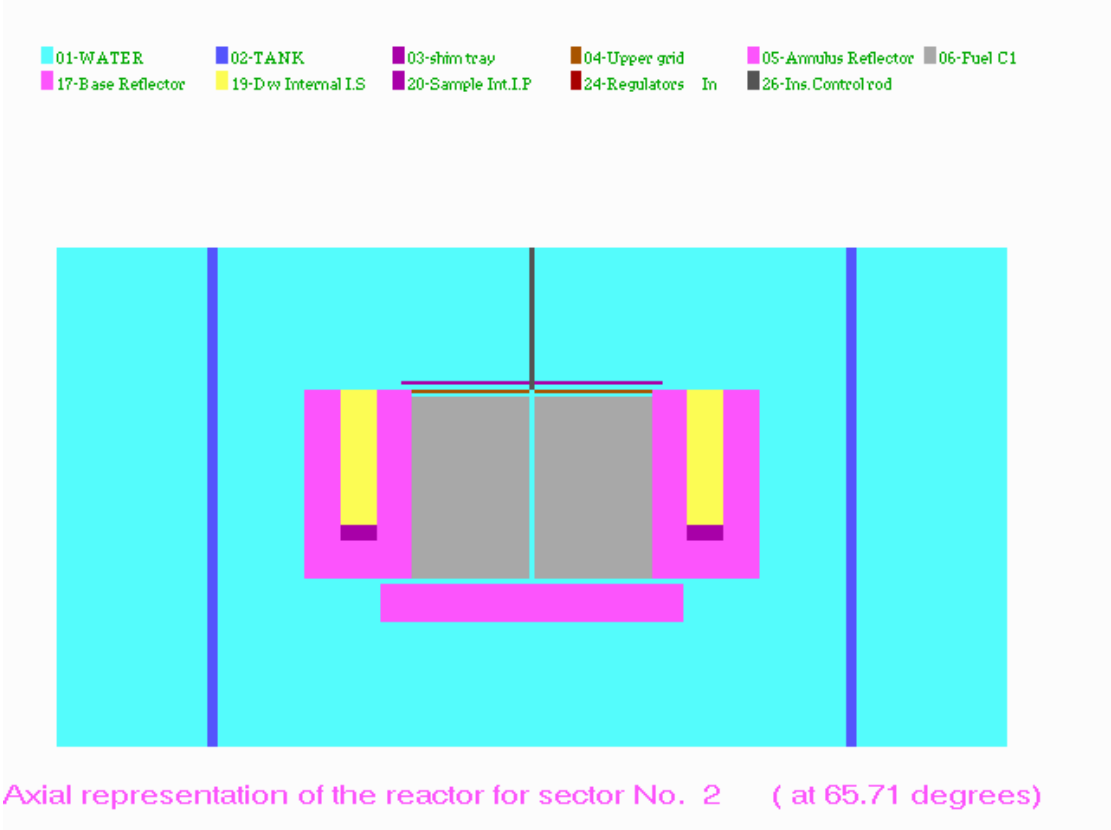

Fig. 1 . The new model of the Syrian MNSR used for the core calculations.

3. Results and Discussion

Using the above described model for the Syrian MNSR the following parameters would be found (see Tab. 1) for the actual reactor using HEU fuel.
There are 3 dummy elements made of aluminum having the same external diameter of the fuel rods plus other 4 tie rods connecting the upper and lower grids. They are assumed to be made of aluminum too.

Table 1. The actual reactor characteristics resulting from the new reactor model.

Flux in the Internal Irradiation Sites ($*10^{12}$)				Initial Excess reactivity (mk)	Fuel Type	No. of Fuel rods	No. of Dummy elements	No. of Tie Rods
Group 1	Group 2	Group 3	Group 4	3.9385	$U\text{-}Al_4\text{-}Al$	347	3	4
.19064	.36495	.51986	.99017					

This model produces data which have a fairly good agreement with the experimental ones [8] (the thermal flux in the internal irradiation site is ~ 1. 10^{12} n/cm^2.s, and the initial excess reactivity of the reactor is ~3.94 mk). The type of fuel is obviously a dispersion one.
Moving to other fuels like the UO_2-Al dispersion fuel we would find for the Syrian MNSR the results of Tab. 2 when the above described model is used:

Table 2. The reactor characteristics resulting from the use of UO_2-Al dispersion fuel.

Flux in the Internal Irradiation Sites ($*10^{12}$)				**Initial Excess reactivity (mk)**	**Fuel Type**	**No. of Fuel rods**	**No. of Dummy elements**	**No. of Tie Rods**
Group 1	Group 2	Group 3	Group 4	3.8323	UO_2-Al	223	0	4
.18028	.34399	.48581	.94584					

As results from Tab. 2 only 223 fuel rods are required. The thermal flux in the internal irradiation sited would decrease of about 5% only. In this model the thickness of the clad is maintained equal to .6 mm, while the old fuel is replaced by the new one only. The other dimensions of the fuel rods are maintained constants as well. In this case the Syrian MNSR would require to increase the power of about 5-6% to recover the decrease in the flux value in the internal irradiation sites, since these sites are very important in the reactor utilization for the Neutron Activation Analysis.

If a ceramic UO2 fuel pellets (with Zr cladding) were used instead the results contained in Tab. 3 would be found.

Table 3. The reactor characteristics resulting from the use of UO_2 pellets cladded with Zirconium.

Flux in the Internal Irradiation Sites ($*10^{12}$)				Initial Excess reactivity (mk)	Fuel Type	No. of Fuel rods	No. of Dummy elements	No. of Tie Rods
Group 1	Group 2	Group 3	Group 4	3.8442	UO_2 ceramic	203	0	4
.178531	.34015	.48067	.95104					

It appears that the fluxes are similar to that of the case of UO_2-Al dispersion fuel, but both cases (of LEU fuel) are different from the HEU actual case of about 5% in terms of flux in the inner irradiation sites. This would imply that the reactor power be raised by the same percent at least.

Acknowledgment

The author thanks Professor I. Othman, Director General of the Atomic Energy Commission of Syria for his encouragement and continued support.

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FUEL MANUFACTURE IN SOUTH AFRICA: THE ROAD TO CONVERSION A PARTNERSHIP NECSA-AREVA CERCA

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ABSTRACT

The South African Research Reactor Fuel Manufacturing Facility (MTR Fuel) was established in the 1970's to supply SAFARI-1 with Fuel and Control Rods. Local capability was developed in parallel with the SA uranium enrichment program to meet the varying needs of the Reactor. In July 2005 the South African Department of Mineral and Energy authorised the transition of SAFARI-1 from HEU Fuel to LEU Silicide Fuel, which includes the conversion of the MTR Fuel Facility.

Past experiences and the status of the MTR Fuel Facility are discussed. Future plans for the local manufacture of LEU fuel and on-going co-operation with AREVA CERCA, the French manufacturer of Research Reactor fuel elements, is explained and elaborated on.

1 Introduction

The birth of the South African Nuclear industry and subsequent programs has been extensively documented in a wide variety of books, publications and journals. Extensive reporting of the commissioning and operation of SAFARI-1 Research Reactor as well as the various conversion, enrichment, PWR fuel fabrication and other strategic projects have dominated discussions for many years [1] [2].

SAFARI-1 (1st South African Fundamental Atomic Research Installation) a tank-in-pool type light water reactor based on the Oak Ridge Reactor was constructed in the early 1960's to meet the needs of resident research and development scientists and selective isotope production needs. The 6.67MW reactor went critical on 18 March 1965 and was subsequently modified to enable operation at 20MW. The reactor was fuelled with Highly Enriched Uranium (HEU) manufactured either in the USA or UK.

SAFARI-1 was regularly operated at 20MW until 1977, at which stage international restrictions on the supply of fuel elements were enforced by political boycott actions. The reactor schedule was adjusted to an operational level of 5MW to support intermittent R&D programs. These developments spurred on the acceleration towards the establishment of a Fuel Fabrication Facility for the local supply of Fuel and Control Rods.

The chronological graph (Figure 1) explains the utilisation of SAFARI-1 since 1965. The left-hand Y-axis indicates the actual MWh operation of for each 3-month period and the right-hand Y-axis the cumulative MWh of operation since start-up.

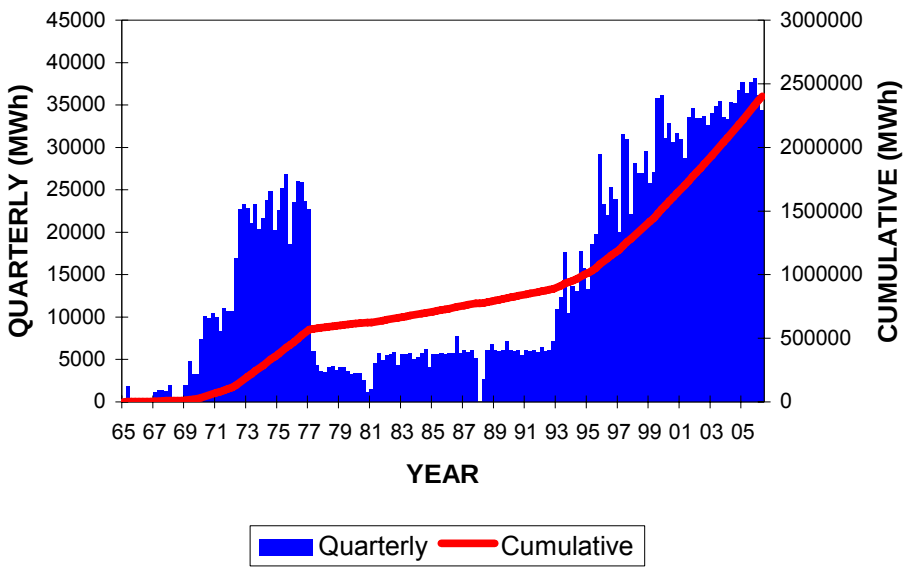

Figure 1: SAFARI-1 Power History (MWh/quarter and Cumulative)

Table1 indicates the HEU Fuel Source and the transition from international to local fuel.

Date	Description of Activity	Utilisation Level (%)	HEU /Fuel Source
1965 (March 18)	First criticality	-	USA
1965 - 1969	Conversion from 6.67 MW to 20 MW and establishment of experimental facilities	< 5%	USA
1970 – 1977	Utilisation of experimental facilities for R&D	~60%	USA/UK
1977 – 1981	Political boycott – fuel sparingly used for R&D	~10%	USA/UK (Inventory)
1981 - 1987	Local fuel manufactured with ~45% enrichment	~15%	**SA**
1988	Shutdown - plant maintenance and refurbishment	-	-
1989 – 1993	Moderate operation - optimised supplies and R&D	~20%	**SA**
1993 →→	Democratic Government - lifting of political boycott HEU (45&90%) allocated to target & fuel program Initiation of semi-commercial program	>80%	**SA**
2005 →→	Continuation of commercial and R&D allocations Initiation of LEU conversion program (~4 years)	>80%	**SA**

Table 1: SAFARI-1 Chronological Utilisation

2 HISTORICAL DEVELOPMENT – SOUTH AFRICAN FUEL MANUFACTURE

It is clear from Figure 1 that the political climate, the priority of the South African strategic programs and the impact of boycotts related to the supply of nuclear fuel to South Africa significantly affected SAFARI-1 utilisation for the period 1977 to 1993. It was in this period however (late 1970's) that the decision was made to establish local fuel manufacturing capabilities at Pelindaba. A Fuel Element Production Facility (Elprod) was constructed and commissioned in order to manufacture UAlx HEU fuel elements and control rods. The technology applied had been developed and established locally, based on ORR fuel design criteria using initially 45% enriched material and later 90% ^{235}U. The first assemblies (19 flat plates) had a loading of 200g ^{235}U, which was subsequently upgraded to 300g ^{235}U. The first local fuel (45%, 200g) was loaded into SAFARI-1 in 1980/1981 and the first local 90% (200g) HEU elements in 1994. 300g ^{235}U elements were loaded for the first time in the first quarter of 1999.

Of particular significance was the successful development and manufacture of fuel elements of 45% enrichment. As is well known the UAlx system at higher concentrations of U presents particular fabrication difficulties (casting, rolling etc), which were successfully overcome. These developments have made an ongoing contribution to the successful ^{99}Mo production program – target plates to this day are manufactured from 45% enriched material using technology developed in the 1980's!

Figure 2: SAFARI-1 FUEL ELEMENT (19 Fuel Plates)

3 FUEL AND TARGET PLATE MANUFACTURE AT Necsa – CURRENT STATUS

The status of fuel manufacture at Necsa for the past 10–12 years has been consistent and stable in terms of manufacturing quantities with routine manufacture of HEU fuel elements and control rods taking place to meet SAFARI-1 requirements. The MTR Fuel Group consists of a Uranium Chemistry Section (Uchem) primarily tasked with the recovery of HEU and melting and casting UAlx ingots and a Fuel Fabrication Section (Elprod) responsible for fuel plate, target plate, component manufacture and assembly of fuel elements.

The HEU recovery and conversion facility (Uchem), which was incorporated into the MTR Fuel Fabrication Group in the early 1990's has focussed on recovery of HEU from a variety of uraniferous materials remaining after the closure of the strategic programs. Uranium metal is currently recovered from all forms of "scrap material" and processes typically include:

1) dissolution of solid forms and subsequent manufacture of uranyl nitrate

2) liquid-liquid extraction of impurities (TBP process)
3) conversion of uranyl nitrate to ADU and UF_4
4) calciothermic reduction to U metal
5) melting and casting of UAlx for fuel plate meat

This facility and processes have proven to be an invaluable source for the recovery of enriched uranium for fuel and target plate manufacture.

The Elprod Facility continues to manufacture fuel, control rods and target plates as well as all components required in the manufacture of fuel elements. Fuel Element End Adapter castings with a complex inner profile, which have proved problematic for many years due to casting defects are now manufactured by CNC machining from extruded aluminium alloy. Although fabrication costs are somewhat higher, excess machining capacity and the significant improvement in quality (particularly welding) have made the alternative fabrication method a preferred option.

The MTR Fuel Group has manufactured 722 Fuel Elements and 131 Control Rods to date (about 15800 fuel plates) with no fuel failures directly attributed to fuel quality in SAFARI-1.

Currently about 40 Fuel Elements per annum and 8 –9 Control Rods are manufactured. Target plates are manufactured in accordance with NTP Radioisotopes (Pty) Ltd requirements for the supply of ^{99}Mo.

4 THE WINDS OF CHANGE – HEU to LEU

In July 2005 the South African Department of Mineral and Energy affairs authorised SAFARI-1 and the MTR Fuel Plant to convert from HEU to LEU fuel over a period of 3-4 years (i.e. a phased conversion of the core).

Experimental U_3Si_2 (4.8g/cc) work has been underway at MTR Fuel since ~2002 albeit on a development scale and with assistance from Argonne National Laboratories and the French research reactor fuel manufacturer AREVA CERCA.

Typical challenges confronted the development team, which included resolving technical problems associated with arc-melting, communiting and selection of particle size, die design, selection of cladding, rolling schedules, homogeneity, stray particle presence and typical dog-bone formation - a host of technical issues experienced at some stage or other by most manufacturers of silicide fuels of higher density.

Challenges facing the MTR Fuel group in this conversion phase include:

1) Uninterrupted production (and subsequent phasing out) of HEU fuel elements and control rods in accordance with the SAFARI-1 conversion requirements
2) Ongoing production of target plates for ^{99}Mo irradiation.
3) Development (including resolving of technical problems) and manufacturing qualification of Silicide Fuel (4.8g/cc).
4) Manufacture of 2 LEU Silicide LTA's (Lead Test Assemblies) for irradiation in SAFARI-1 and subsequent limited LTA manufacture utilising the development facility.
5) Specification of a Silicide production facility for powder and core manufacture (processes and equipment).
6) Design, licensing, construction, cold and hot commissioning of Silicide production facilities (2-3 year project).

It soon became apparent that Necsa and the MTR Fuel group did not have the resources (manpower) to address the production, technical development and production facility design and other requirements in parallel, and that technical co-operation with an experienced manufacturer would have significant benefits.

The MTR Fuel Group and AREVA CERCA, a long-standing business associate of Necsa, held preliminary discussions in the middle of 2005 where the concept of potential technical co-operation was discussed and finally agreed upon.

5. CERCA EXPERIENCE IN SILICIDE FUEL MANUFACTURING AND MTR CONVERSION

CERCA, a subsidiary of AREVA, has been in charge of manufacturing and supplying research and material test reactor fuel assemblies for more than forty years and is the world leader in its field. CERCA supply covers a large range of products, in terms of geometries (flat or rolled plates, tubular or ring-shaped elements) as well as enrichments (HEU, MEU, LEU), and fully satisfies the technical and scientific needs of customers demanding quality and safety.

Since 1960, CERCA has manufactured over 300 000 fuel plates, about 20 000 fuel elements of 70 designs, delivered to 40 research reactors in 20 countries.

Thanks to this broad supply, CERCA has gained a high quality experience feedback. This is employed to the benefit of reactor operators by providing them with high performance fuels.
One of the ways of development CERCA has been working on for more than 20 years concerns improvement of low-enrichment fuel performance levels by increasing ^{235}U load per plate without changes in external geometry. This has been achieved with silicide U_3Si_2 fuel (standard and control fuel assemblies) that CERCA manufactures on an industrial scale and routinely delivers worldwide.

By mid-2006, CERCA has manufactured 2700 U_3Si_2 fuel assemblies (about 55 000 plates), for customers distributed in Australia, France, Germany, Greece, Netherlands, Japan, Canada, South Africa, Sweden, Switzerland, Taiwan and Turkey.
CERCA manufacturing plant, located in Romans in the south of France, has a global manufacturing capacity of 20 000 plates. The production is around 11 000 plates per year, 50% of which are U_3Si_2 fuel plates.

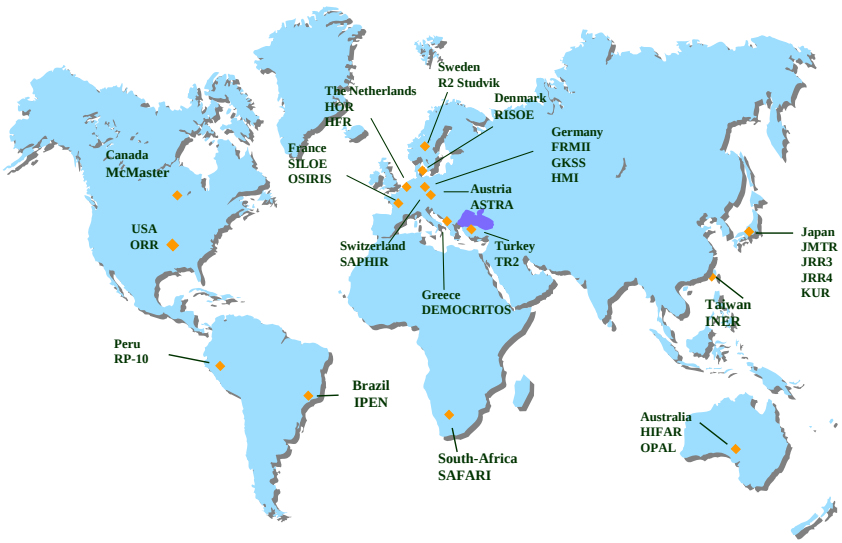

Figure 3: CERCA U_3Si_2 customers

Since the first fuel test assemblies, delivered to the Oak Ridge Research Reactor in 1983, considerable progress was accomplished in the production standards. From that time new production equipment and processes have been implemented to fulfill the specific silicide fuel requirements.

For example:

- A new arc melting furnace has been supplied
- A dedicated powder manufacturing line has been installed
- New core pressing tools have been designed
- New rolling sequences have been developed
- Numerical X-ray machine has been developed to assist operators for fuel core length adjustment

CERCA has also implemented a quality system for inspection of the fuel plates at each step of the manufacturing:

- UT inspection: special UT inspection equipment enables detection of delamination within the meat that cannot be detected with a blister test
- RT inspection and film examination: controlling of the stray particles by microscope on X-rays films and also accurate dimensional inspection of the cores
- Homogeneity inspection: numerical X ray machine and software systems for obtaining a map of uranium density in the plates and knowing the exact density value at each square centimeter
- Dimension check and surface defect check on finished plates
- Final inspection including contamination check

In addition to the manufacturing experience, there have been many in-reactor tests and examinations, from 1982 to 1998, involving U_3Si, with a Uranium density between 2 and 6 g/cm^3. This extensive program of test and development covered various design parameters, such as meat chemical composition and densities. Several production parameters were also varied, keeping the industrial conditions. Industrial conditions, as opposed to laboratory conditions, refer to manufacturing in the same workshop, with the same people, using the same equipment, work instructions, Quality system etc as for the standard production.

At each step of the development, CERCA has been driven by the reliability of the solution proposed and has endeavoured to carry out all developments with standard manufacturing tools, on full size plates and with a significant number of elements. This has allowed CERCA to overcome the all too well-known gap between R&D with few test elements and standard production in large series. As a result, the prototype fuel assembly loaded in HFR-PETTEN in 2003 has exceeded 75% of burn up and is exhibiting an excellent behavior.

With continuous contacts with the research reactor community around the world and thanks to its large manufacturing experience, CERCA knows very well what is important from a safety standpoint, as far as fuel manufacturing is concerned, and can address individual customer needs and any special quality requirements in consistency with safety authorities' demands.

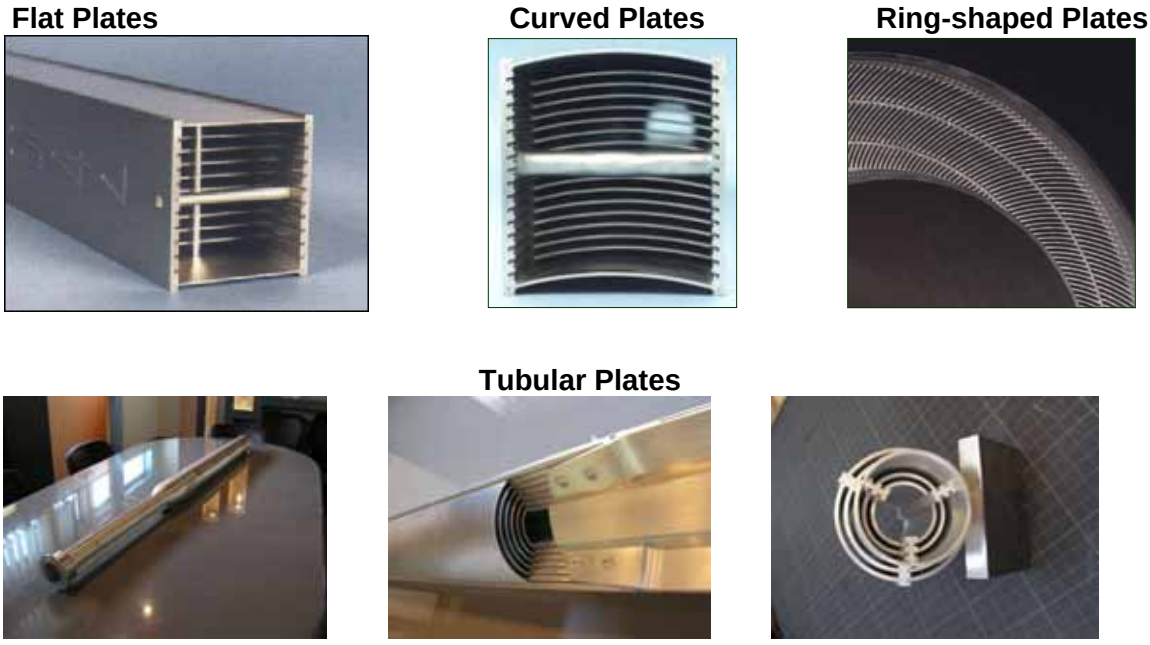

Figure 4: main fuel designs manufactured by CERCA

CERCA Experience in Research Reactor Conversion

CERCA has been involved for more than fifteen years in international cooperation for U_3Si_2 fuel assembly supply and reactor conversion. It has collaborated several times for reactor conversion process, for MTR and TRIGA reactors, in countries such as Japan, Germany, the Netherlands, the USA and France. Presently, CERCA is collaborating with the USA regarding the conversion of the Washington State University Triga reactor and RPI MTR reactor in Portugal.

6. CO-OPERATION BETWEEN NECSA AND CERCA: A LONG TERM RELATIONSHIP

Necsa and CERCA commenced a formal partnership from early 2004 by means of a Memorandum of Understanding and have been working together in different fields such as the supply of two LTA's for SAFARI-1, fuel plate supply, uranium delivery, fuel plate evaluation and preliminary evaluation of a concept silicide fuel plant design (powder and core manufacture).

To support the MTR manufacturing conversion process, the objective is to provide Necsa with a) technical assistance and validation of manufacturing qualification on a limited scale utilising a laboratory facility for powder and core manufacture and b) technical cooperation and expertise for establishing a qualified production line for routine LEU fuel manufacture and mastering the new manufacturing processes.

Various proposals were tabled and a point has been reached where two phases of technical assistance and co-operation are under way..

6. 1 On-going technical assistance initiative:

The first phase of this collaboration is focusing on the manufacturing of powder and cores performed on a laboratory scale. CERCA is evaluating the current MTR Fuel processes, proposing improvements and validating the LTAs manufactured by Necsa through verification of compliance to the Necsa technical specification. This assistance has been in progress from February 2007, at the Pelindaba manufacturing facility.

6.2 Longer term cooperation agreement:

The second step will consist in assistance for the manufacturing on a production scale. CERCA will perform:

- Evaluation of Necsa existing production processes and equipment
- Training of Necsa staff on the different steps of manufacturing of silicide fuel and inspection processes. This training will be performed at CERCA manufacturing plant in Romans-France.
- Technical assistance during starting of the new equipment at the MTR Fuel plant at Pelindaba
- Validation of the first production run manufactured by Necsa

This assistance is scheduled to take place during 2007 and 2008.

7. CONCLUSION

South Africa has committed itself to the conversion of the SAFARI-1 research reactor and the associated fuel and control rod manufacturing from HEU to LEU utilisation. Significant in-house progress has been made regarding development of the applicable manufacturing techniques, together with selective assistance from both the Argonne National Laboratories and the research reactor fuel manufacturer AREVA CERCA. Final stages of development to achieve qualified licensed fuel initially on an experimental scale (Lead Test Assemblies) and later on a full production scale are being addressed in conjunction with CERCA.

This cooperation is strengthening the long-lasting relationship between Necsa and CERCA and developing mutual benefit for the two companies.
Thanks to this assistance and experience provision, Necsa will be able to manufacture the LEU fuel elements for the SAFARI-1 reactor and gain additional skills and competences in the manufacturing field. On its side, CERCA will gain a new experience in research reactor conversion.
Moreover, this operation will contribute to meeting the international requirement on global reduction initiative.

8. REFERENCES

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[2] CSB Piani, South Africa and the SAFARI–1 Scenario: On the Road to Conversion – From HEU to LEU, RERTR 2005

NEUTRONIC CALCULATIONS FOR CONVERSION OF ONE-ELEMENT CORES FROM HEU TO LEU USING MONOLITHIC UMO FUEL

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ABSTRACT

The use of monolithic UMo fuel of highest density might be a viable option to convert high-flux research reactors from HEU to LEU fuel especially in case of one-element-cores. As a challenging example we used the M^3O code to investigate the potential of monolithic UMo for the conversion of German FRM-II. To meet the conversion requirements of a maximized flux, a minimized enrichment and an acceptable cycle length requires a global optimization routine. Neutronic calculations of variations of fuel element parameters (meat, cladding and cooling channel thickness, height, density transition radius) show the potential of monolithic LEU fuel. Results of a parameter space study indicate the chance to find LEU options while keeping important operational constraints (like power peaking, heat flux etc.). Additionally, we report on progress in developing a search routine for the global optimization problem, which should be generally usable for other cases as well.

1. Introduction

In a rapidly changing world with nations seeking to acquire nuclear weapons the policy of non-proliferation and programs to reduce risks associated with nuclear weapon usable materials become more and more important striving for a more proliferation-resistant technology use. The global threat reduction initiative (GTRI) e.g. has the ambitious goal to convert a targeted number of about 100 remaining HEU reactors worldwide to LEU fuel by the year 2014.
Fortunately the development of UMo-dispersion-fuels made significant progress by addition of Si to suppress excessive pillowing due to porosity effects under high fission rates, high fission densities and temperature conditions [1]. Thus conversion of many reactors comes within reach using dispersion fuels. Although monolithic UMo-fuel, due to its high density, still is the best candidate for a conversion to LEU in case of high-flux single-element reactors. In particular, only with monolithic fuel a conversion to an enrichment level of 20% (19.75%), which is the international accepted limit for LEU[1], could be achieved in principle for the challenging case of the FRM-II. Therefore monolithic UMo fuel might be the viable option to foster the international efforts to phase out the usage of civilian HEU for research reactors and thus reducing the risks of proliferation and nuclear terrorism.
After the first confirmation of good irradiation behaviour of monolithic miniplates in the RERTR-4 experiments [4,5], several further irradiation experiments have been performed [6,7]. Additionally, a variety of fabrication techniques are going to be developed [8,9]. A minor drawback in the development is the discovery of small porosities at the meat-cladding interface, due to the same mechanism as in dispersion fuel [6], but this is not so troublesome since the interface area is much smaller in monolithic than in dispersion fuel.

2. Need for Optimization and Adequate Tools

In the case of the FRM-II first results using the M^3O-Code, developed in our group [10,11], showed that the most straightforward way for a conversion to LEU - the simple replacement of the uranium-silicide-HEU fuel by UMo-monolithic-LEU fuel - yields a reactivity loss at BOL and therefore an

[1] Assessing the proliferation potential of research reactor fuel shows that an enrichment of 20% is the best compromise between U235 content in the fuel and plutonium production [2]. Calculating the critical masses shows that 50% enriched uranium only triples the critical mass [3].

unacceptable fuel element lifetime of just a few days. Therefore the usage of monolithic LEU in the current FRM-II fuel element HEU-design is not possible due to these $k(eff)_{ini}$ losses in the core [12].
The two principle strategies to increase the initial reactivity to reach an adequate cycle length are the usage of enrichment levels above the LEU limit or geometrical changes to the core and/or fuel plate geometry. Of course, enriching beyond LEU contradicts the international goal for the conversion of research reactors and therefore has to be considered problematic with regard to non-proliferation objectives.
The more interesting strategy to increase $k(eff)_{ini}$ is the modification of geometrical design parameters of the fuel element. Such modifications are very limited as long as the most relevant dimensions (inner and outer radius) of the fuel element geometry are considered to be unchangeable. Calculations considering first tentative modifications of particular variables like the height of the fuel element, cladding, meat and cooling channel thickness have shown the potential of these modifications to increase $k(eff)_{ini}$ [13].
However, the full optimization problem does not only consist of an increased initial reactivity to achieve an adequate cycle length. It includes an optimized overall performance in parallel to minimizing the enrichment requirements for a given fuel element geometry down to LEU. Furthermore the overall performance of a neutron source cannot be adequately assessed by the maximum flux in the moderator tank only. The performance of a neutron source is a function of availability per year (cycle length, downtime), the neutron flux at different positions in the moderator tank, the number of beam tubes and the efficiency of the neutron guides, instruments and experiments.[2]
The full optimization problem is non-linear as both objective functions $f_j(x_i)$ and constraints $C_k(x_i)$ are complex functions of the reactor design variables x_i (cf. [17]). A routine to solve this problem would either use global optimization algorithms or genetic algorithms. Both approaches need widely automated tools for input deck generation and execution and processing of computer-jobs to handle the large number of files. Considerable progress on these tools was made although the full optimization routine is not automated completely yet and some additional modules have to be implemented.[3]

3. Summary of First Results

With these improved tools we already carried out a parameter study [13,17,18] varying several fuel plate dimensions x_i (meat, cladding and cooling channel thickness and the length of the fuel plate). The first goal was the investigation of the functional dependence of $k(eff)_{ini}(x_i)$ on these parameters. For this parameter-space-study the enrichment was kept constant at 19.75% (LEU) and the initial reactivity served as new objective function ($f(x_i)=keff_{ini}(x_i)$).
To no big surprise the results indicated that cladding thickness is very important to gain more initial reactivity. The thinner the cladding the more initial reactivity is gained as the volume gained by reducing the cladding can be filled with fuel if the number of fuel plates is kept constant. The Effect of cladding thicknesses as low as 200 µm were evaluated. The meat thickness to get an optimum for the initial reactivity lies between 0.7 and 0.9 mm (cf. fig. 1) except in the case of very thin cladding and cooling channel, for which a trend to thinner meat thicknesses can be observed. Regarding the cooling channel thickness it can be observed that variations down to smaller values are not attractive. The optimum can be obtained between 2.6 and 3.4 mm within the studied range. Wider cooling channels in combination with less plates in the fuel element are increasing the moderator to fuel ratio, thermalize the spectrum and therefore can increase the reactivity. If the cooling channel thickness is increased, the more sensitive is $k(eff)_{ini}$ on increasing meat thickness.[4] Increasing the active height of the fuel element, directly raises $keff_{ini}$. In conclusion, the study showed that thicker meat, thinner cladding and wider cooling channels as compared to the current HEU geometry with additional enlargement of the height of the fuel plate can increase $k(eff)_{ini}$ significantly [17,18].[5]

[2] See [11, 14] for more details on this discussion.
[3] As a faster alternative to the MCNP part of M^3O we consider the future use of TART [15] coupled to MCMATH [16] or ORIGEN for burnup calculations.
[4] See [18] for more details.
[5] These changes to the fuel plate geometry reduce the number of plates in the fuel element.

Two promising combinations of parameters were chosen from the parameter study (LEU1 and LEU2) and were investigated in more detail (cf. tab. 1). The maximum flux in the moderator tank is suppressed by 15.4% and 16.2% for LEU1 and LEU2, at the position of the cold source by 13.7% and 14.7% respectively. Using a marginal 10% increase in reactor power to 22 MW the flux losses decrease to 7.1% and 5% for LEU1 and LEU2 respectively at the maximum flux position and 7.9% and 6.2% at the cold source. Although using 19.75% enriched fuel the LEU1 modification achieved a cycle length of $T_{cycLEU1} \approx 50$ days for 20 MW, which effectively reduces the overall performance per year by additional 1%.[6] In case of a marginal increase in power of 22 MW the higher burnup will lead to a decreased cycle length of 45 days which corresponds to additional 4% performance loss.

	HEU	LEU 1		LEU 2	
Thermal Power	20 MW	20 MW	22 MW	20 MW	22 MW
Active Height	70 cm	80 cm		84 cm	
Meat thickness	0.60 mm	0.80 mm		0.80 mm	
Cladding thickness	0.36 mm	0.20 mm		0.25 mm	
Cooling Channel thickness	2.2 mm	3.0 mm		3.0 mm	
Enrichment	93%	20%		20%	

Table 1 Chosen parameter specifications for LEU1 and LEU2 in comparison with the current HEU design of FRM-II.

4. Steps to Further Optimization: Power Peaking, Flux Performance, Zr-Cladding

With the parameter space study it could be shown that there is a potential for increasing $k(eff)_{ini}$ [17,18] without changes to the outer fuel element geometry. However, studying the parameter space only to increase $k(eff)_{ini}$ neglects the need to optimize the flux itself and to keep operational constraints like power peaking and heat flux. A first approach to prevent power peaking in the fuel plate is to adjust the density transition radius in the fuel plate and the transition ratio. To minimize the power peaking in the fuel plate our calculations propose an interval in the transition radius 10.3-10.8 mm (HEU design: 10.59 mm) keeping power peaking below 2.0. (At 10.74 mm power peaking is reduced from 1.93 to 1.88, $k(eff)_{ini}$ increases from 1.169 to 1.172, and flux loss increases by additional 0.5%.)

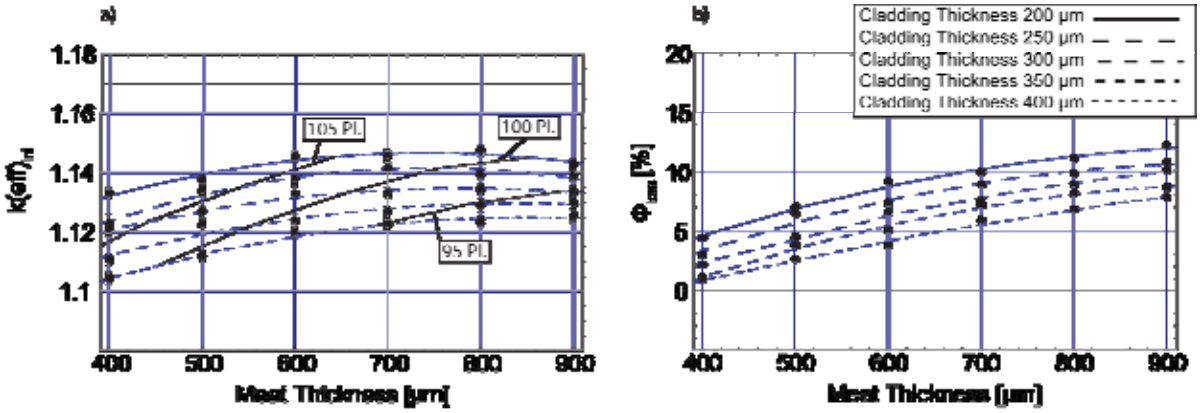

Fig. 1 Calculation results for different meat and cladding thicknesses (cooling channel thickness 3.0 mm, active height 70 cm, LEU). Each point represents a M^3O calculation with a particular fuel element geometry of FRM-II. The plate number differs from core to core. a) Initial reactivity for different fuel element dimensions. Solid black lines connect points with constant number of fuel plates. b) Flux losses in reference to the flux in the current HEU geometry.

Broadening the view again to the full optimization problem the effect on the flux performance is evaluated. Fig. 1 shows an example from the studied range of 125 different fuel element geometries, each geometry with a different number of fuel plates in the core. The desired increase of $k(eff)_{ini}$ can be achieved by thickening the meat (Fig. 1a), since increasing the meat thickness and keeping cooling

[6] Availability of the source per year taking the current downtimes into account. (The nominal cycle length of the current HEU design is 52 days.)

channel and cladding thickness constant effectively increases the amount of fuel in the core and thereby increases $k(eff)_{ini}$. However, as the content of U238 is increased, self-shielding reduces the flux as well (Fig. 1b). Similar opposed trends can be observed by reducing the cladding thickness as again the amount of U238 is increased in the core.
These results show again the need for a routine to find the optimum between the complementary needs to increase $k(eff)_{ini}$ (thus increasing the cycle length) and to minimize flux losses.
In case of widening the cooling channel thickness the trends are not opposing with respect to $k(eff)_{ini}$ and flux: for wider cooling channels $k(eff)_{ini}$ increases and flux losses decrease.
We have also investigated these effects at the position of the cold neutron source (CNS): the influence of increased meat thickness on flux losses is less important; flux losses at the CNS and at the position of maximum flux are similar regarding the cladding thickness; widening the cooling channel thickness has nearly no influence on the flux at the CNS.[7]
Therefore, widening of the cooling channel will raise the performance, while meat and cladding thickness have to be further optimized beyond the first LEU guesses, probably going to thinner claddings.
First irradiation tests of Zr-cladding fuels fabricated by the Argentine fuel development program and irradiated in RERTR-7a showed good preliminary results [19]. This indicates the possibility to fabricate fuel with very thin cladding as low as 150 µm. To assess the potential of Zr-cladding (Zr-4). for monolithic fuel, we calculated LEU-options with different meat and cooling channel thicknesses and the original active height of 70 cm. Fig. 2 shows that in case of very thin claddings an effective choice of meat thickness can be much lower than expected first. Furthermore the flux can be optimized by meat thickness changes without compromising the initial reactivity, while taking wider cooling channel into account.

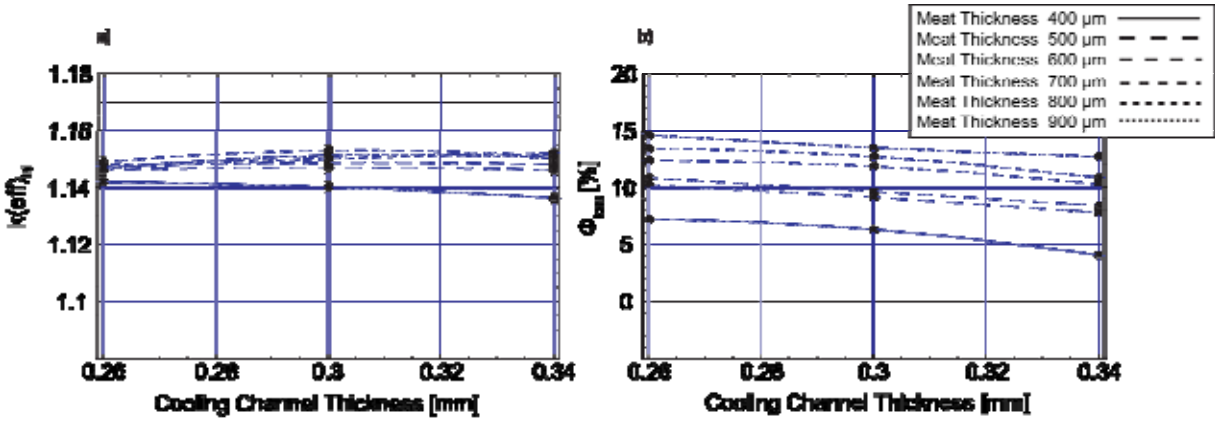

Fig. 2 Results for different meat and cooling channel thicknesses (active height 70 cm, 150 µm Zr-cladding, FRM-II, LEU). Each point represents a M^3O calculation with a particular fuel element geometry. The plate number differs from core to core. a) Initial reactivity for the different core dimensions. b) Flux losses in reference to the flux in the current HEU geometry.

5. Conclusion and Outlook

The presented results show, that a global optimization strategy is needed to scrutinize the effects of changes to fuel element geometry in depth. The results also indicate that it might be possible to reach the ultimate goal of optimizing enrichment, reactor performance, and safety margins. Cycle length and flux at instruments can be optimized by appropriate choice of design parameters for the fuel element (in particular wider cooling channels and suitable meat and cladding thickness). Calculations for very thin Zr-cladding show that there is an additional potential to increase $k(eff)_{ini}$ and thus raise the cycle length.
The current work focuses on the full implementation of a search algorithm (genetic algorithm) for all variables, especially the implementation of a radially (or even axially) shaped meat thickness to

[7] Please note that the flux losses in fig. 1 only give the general trend. It has to be kept in mind that the considered fuel element variations will have a shorter cycle length as in the HEU design since $k(eff)_{ini}$ is lower than 1.17., which – at least as a rule of thumb –determines the value of the cycle length.

optimize flux and power distribution. We will improve our reactor model to assess the axial flux in greater detail. This implies mainly the integration of the control rod movement into the code. Additionally, we will concentrate on the operational constraints by integrating models for heat transport in the plates and the core.

Financial support by Berghof Foundation for Conflict Research, Berlin is gratefully acknowledged

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ANALYSIS OF AN LEU FUEL WITH SPATIALLY-DEPENDENT THICKNESS IN TWO DIMENSIONS

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ABSTRACT

Prior studies have shown that the neutron flux at the cold source location of the High Flux Isotope Reactor is diminished at end-of-cycle by 15% when the reactor is fuelled with low enriched uranium in place of the current, high enriched uranium. An increase in operating power could mitigate this penalty. The operating power is limited by the predicted onset of incipient boiling calculated under a set of assumptions related to measurement uncertainties, analysis techniques, and defined safety margins. Reducing the spatially-dependent local power densities at the exit of the coolant flow from the core would allow for increased operating power with no change to the current assumptions. By manufacturing fuel in which the thickness of the fuel is varied (graded) in the axial direction as well as continuing the current practice of grading in the radial direction, operating power can be increased.

1. Introduction

In August of 2005, staff at the Oak Ridge National Laboratory (ORNL) were requested by the National Nuclear Security Administration (NNSA) of the United States Department of Energy (DOE) to begin studies of the conversion of the fuel for the High Flux Isotope Reactor (HFIR) from high enriched uranium (HEU) to low enriched uranium (LEU). The criteria established by the NNSA were:

1) ensure that the ability of the reactor to perform its scientific mission is not significantly diminished,
2) work to ensure that an LEU fuel alternative is provided that maintains a similar service lifetime for the fuel assembly,
3) ensure that conversion to a suitable LEU fuel can be achieved without requiring major changes in reactor structure or equipment,
4) determine, to the extent possible, that the overall costs associated with conversion to LEU fuel does not increase the annual operating expenditure for the owner/operator, and
5) demonstrate that the conversion and subsequent operation can be accomplished safety and that LEU fuel can meet safety requirements.

A previous study (Ref. 1) had shown that it was not possible to meet these criteria with uranium silicide fuels. The potential qualification of uranium molybdenum fuel (UMo) which has a significantly higher density than uranium silicide led to the request that reactor performance with UMo be examined.

1.1 Description of HFIR

The HFIR (Fig.1) is a pressurized, light-water-cooled and -moderated, flux-trap-type reactor that currently uses U_3O_8-dispersed-in-aluminum, HEU fuel and operates at 85 MWth. The reactor core (Fig. 2) consists of two annular fuel elements, each approximately 61-cm high (fueled height is 51 cm). At the center of the core is a 12.70-cm-diameter, cylindrical hole, referred to as the “flux trap target” region, which contains 37 vertical, experimental, target rod sites.

The HFIR fuel elements, surrounding the flux trap, contain fuel plates having an involute-shape in the radial direction. The fuel elements are separated by a narrow water gap. The inner element contains 171 involute-shape fuel plates, and the outer element contains 369 involute-shape fuel plates, as detailed in Fig. 3. The fuel plates are a sandwich-type construction with a fuel-bearing cermet and aluminum filler bonded to a cladding of type-6061 aluminum. The fuel plate and adjacent water gap thickness are each 1.27 mm. The HEU oxide is distributed (graded) along the arc of the involute aluminum plate, as seen schematically in Fig. 3. At the start of each fuel cycle, fresh inner and outer elements are loaded so there are no fuel management issues as in other reactors.

Figure 1. Configuration of HFIR core and reflector.

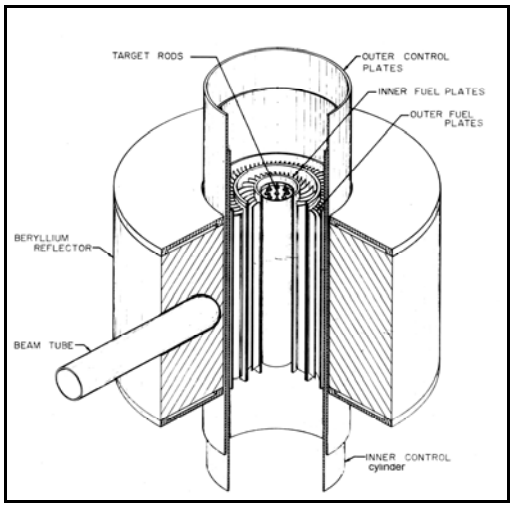
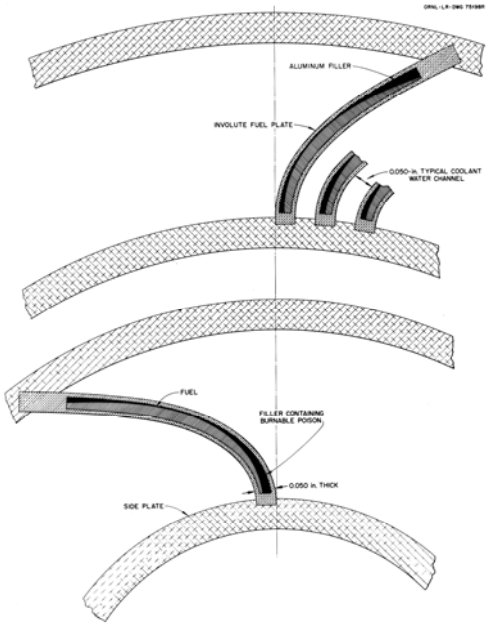


Figure 2. The core of HFIR, showing the inner (IFE) and outer (OFE) fuel elements.

Figure 3. Schematic of radial fuel grading (fuel meat) in the current HEU fuel plates in the HFIR fuel elements.



Control plates, in the form of two thin (approximately 1.3 cm), europium/tantalum-bearing concentric cylinders, are located in an annular region between the outer fuel element and the beryllium reflector (see Fig. 1). These plates are driven in opposite directions and significantly impact the power density at the outside edge of the outer element.

The control plates and fuel elements are surrounded by a concentric ring of beryllium (Be) that serves

as a reflector and is approximately 30-cm thick. This Be reflector is subdivided into three regions: the inner removable reflector, the middle semi-permanent reflector, and then the outer permanent reflector. The Be reflector is surrounded by a light water and a steel pressure vessel. In the axial direction, the reactor is reflected by light water.

Nominally, there are eight fuel cycles in a calendar year (8 reloads, 26 day cycle length). Maintaining fresh fuel inventory requires the manufacturing of over 4000 fuel plates per year. The lifetime of the reactor is limited by fluence to the pressure vessel but with the current vessel and at the current power level and availability factor, the reactor is expected to operate for another 30 years, potentially requiring the manufacture of 130,000 UMo fuel plates.

1.2 Purpose of study

Implementation of the objectives mentioned previously was achieved through the creation of an assumptions and criteria document (Ref. 2). All studies conducted to date have been limited to changes in the region-between-the-clad (fuel meat region) of the HFIR plates, shown in Fig. 3. No changes have been assumed for any of the reactor operating conditions (inlet, outlet temperatures, system pressure, etc) or for the geometry of the reactor core (diameter, materials, fuel plate dimensions, etc.). Various physics-related parameters related to safety (reactivity coefficients), performance (flux levels in irradiation and beam tube locations), and safeguards (dose levels and actinide content) were studied and were documented in Ref. 3.

A conclusion of these studies was that the current level of reactor performance as defined by parameters listed above and in Ref. 2 could not be maintained with LEU fuel with the reactor operating at 85 MW. From those studies, the hypothesis was developed that if the reactor power could be increased to the original design level of 100 MW – the HFIR was operated at 100 MW for more than 20 years – the performance parameters could be maintained at their current levels. One method of obtaining this higher power level consistent with the original assumption of only considering changes to the fuel meat region was to smooth the power distribution by grading (or tapering) the fuel thickness in both the axial and radial directions (currently the fuel is graded only in the radial direction). Preliminary results of these studies are presented in this paper.

1.3 Fuels under consideration

As noted in Ref. 2, the reference fuel form was uranium-molybdenum alloy with 10 wt% molybdenum (U-10Mo). Consideration was given to a monolithic fuel form – a rolled metal foil and also to a dispersion of U-10Mo spheres in Al powder. Upon review of the results in Ref. 3 (the U-10Mo dispersion fuel could not reach lifetime requirements), materials specialists suggested consideration of a denser, 7 wt.% uranium-molybdenum alloy (U-7Mo) at a slightly higher packing fraction than had been assumed for the U-10Mo studies. The characteristics of the U-7Mo fuel were:

1) 55 volume percent for U-7Mo,
2) 45 volume percent for Al,
3) uncoated U-7Mo spheres with silicon in matrix (Si not in neutronics calculaton) yielding,
4) a uranium density of 8.7 g/cm^3 (maximum in current HEU fuel is 3.2 g/cm^3)

As will be presented in the next section, this dispersion fuel was found to give equivalent neutronic performance to the U-10Mo monolith. Consequently, both monolithic and dispersion fuels fabricated from uranium-molybdenum remain candidates for a HFIR LEU fuel.

2. Results of 1-D grading

Fuel profiles for the inner and outer elements – grading profiles as shown in Fig. 3 but for LEU fuels – for monolithic (U-10Mo) and dispersion (U-7Mo) fuels are shown in Fig. 4 (inner element) and Fig. 5 (outer element).

Fig. 4. Inner element LEU fuel profiles

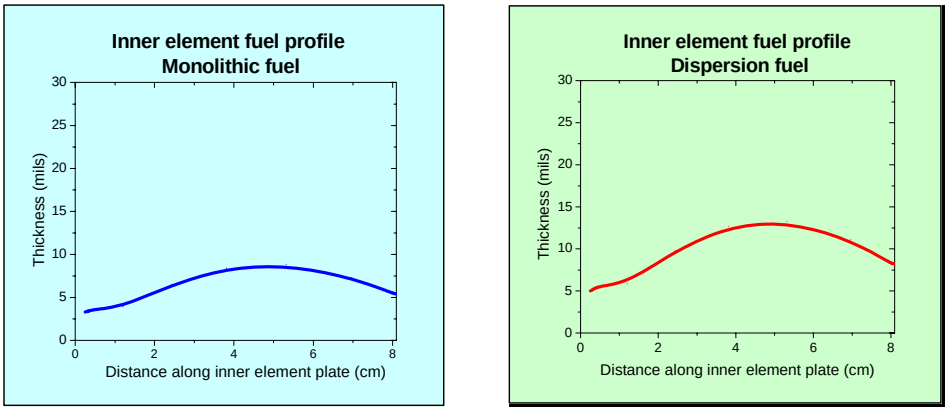
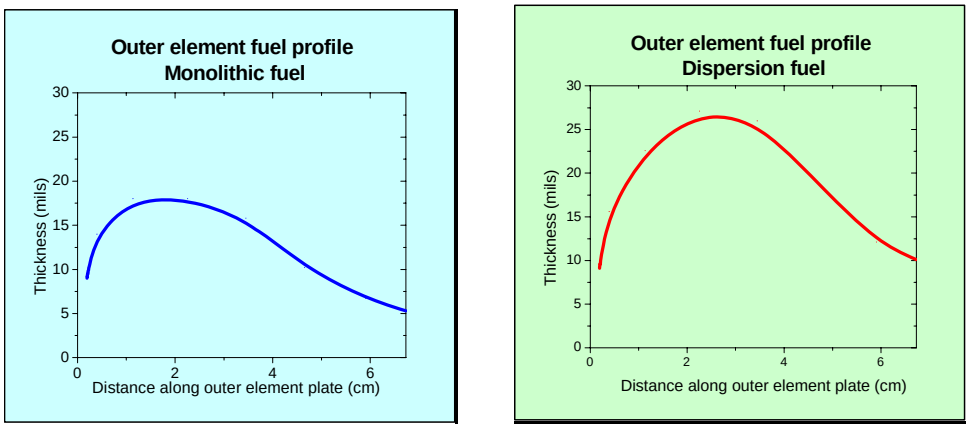


Fig. 5. Outer element LEU fuel profiles



The performance for selected parameters for the two fuels is provided in Table 1. Monolithic fuels could be constrained by minimum foil thickness that can be economically fabricated. Dispersion fuel is constrained by the available meat thickness in the current HFIR fuel plate.

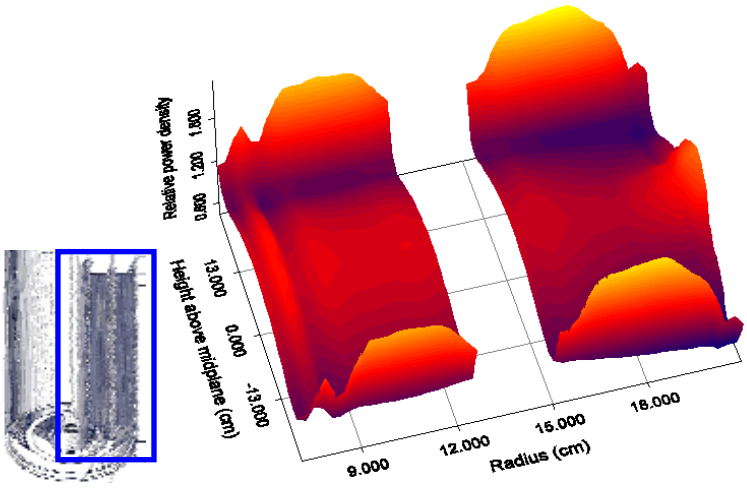
Table 1. Selected LEU performance parameters

Parameter		HEU	LEU monolith	LEU dispersion
Peak thermal flux in reflector (n/cm^{2}s)	Beginning of life	1.1(10^{15})	1.1(10^{15})	1.1(10^{15})
	End of life	1.7(10^{15})	1.5(10^{15})	1.5(10^{15})
Peak thermal flux in central target (n/cm^{2}s)	Beginning of life	2.6(10^{15})	2.5(10^{15})	2.6(10^{15})
	End of life	2.7(10^{15})	2.5(10^{15})	2.5(10^{15})

3. Performance from two-dimensional fuel grading

Fig. 6 shows the power profile in the HFIR for the "only-radially-graded" monolithic fuel design. The coolant in HFIR flows downward through the core and the lower axial edge peak is the limiting location for avoiding incipient boiling. Fuel densities at the upper and lower edges of the fuel elements were reduced and the operating power resulting from the thermal hydraulic margins identified in Ref. 2 was recalculated for a new power distribution based on Monte-Carlo-normalized diffusion theory. The height of the reduced-density axial zones has not been optimized but reducing the fuel density by 50% over the top and bottom 2.5 centimeters of the fuelled region results in an estimated beginning-of-life operating power of 102 MW and an estimated end-of-life operating power of 97 MW. EOL reflector peak thermal flux equals the current HEU value.

Fig. 6. Beginning-of-life power distribution for "only radial graded" fuel
(diffusion theory; profile for blue highlighted region of core)



4. Changing the assumptions – alternative methods for achieving 100 MW

If the requirement that "only changes to the interior of the fuel plate (the meat) can be considered" is removed, then at least two additional methods might allow increasing the power level of the HFIR and should be analyzed. However both methods would result in additional tests to extend the safety basis for the reactor. Tests that would, perhaps, not be needed if the requirement were retained.

4.1 Poisoned end plates

During recent discussions with fuel fabricators, the request was made to investigate the use of poisoned endplates – neutron poison placed in the unfuelled, axial portion of the fuel plates (5 cm at each end of the HFIR fuel plate do not contain fuel). The fabricator expressed the opinion that production of plate cladding that is "poisoned in some locations but not others" might be a simpler, cheaper, and more reliable process than a two dimensional fuel grading process. Exposure performance data for poisoned clad joined to un-poisoned clad would be needed.

4.2 Redefining input to safety analyses

Ref. 2 contains a list of assumptions related to fuel fabrication tolerances that are input to the HFIR safety basis. New fuels, especially monolithic fuels, could be fabricated to tighter tolerances than existing specifications therefore providing additional margin that could result in increased operating power. Data to support new tolerances would be needed. Improvements in thermal hydraulic methods to account for turbulent mixing within the HFIR core could also provide margin for increase in operating power. Data to validate computer program modifications would be needed.

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NEUTRONIC ANALYSIS FOR CONVERSION OF THE GHANA RESEARCH REACTOR-1 FACILITY USING MONTE CARLO METHODS AND UO_2 LEU FUEL

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Abstract

Monte Carlo particle transport methods and software (MCNP) have been applied in the modelling, simulation and neutronic analysis for the conversion of the HEU-fuelled core of the Ghana Research Reactor-1 (GHARR-1) facility. The results show that the MCNP model of the GHARR-1 facility, which is a commercial version of the Miniature Neutron Source Reactor (MNSR) is good as the simulated neutronic and other reactor physics parameters agree with very well with experimental and zero power results. Three UO2 LEU fuels with different enrichments, core configurations, core loadings were utilized in the conversion studies. The nuclear criticality and kinetic parameters obtained from the Monte Carlo simulation and neutronic analysis using three UO_2 LEU fuels are in close agreement with results obtained for the reference 90.2% U-Al HEU core. The neutron flux variation in the core, fission chamber and irradiation channels for the LEU UO_2 fuels show the same trend as the HEU core as presented in the paper. The Monte Carlo model confirms a reduction (~ 8% max) in the peak neutron fluxes simulated in the irradiation channels which are utilized for experimental and commercial activities. However, the reductions or "losses" in the flux levels neither affects the criticality safety, reactor operations and safety nor utilization of the reactor. Employing careful core loading optimization techniques and fuel loadings and enrichment, it is possible to eliminate the apparent reductions or "losses" in the neutron fluxes as suggested in this paper. Details of the Monte Carlo simulated criticality and kinetic parameters, power as well as the neutron flux distributions are presented in the paper.

1.0.Introduction

The Ghana Research Reactor-1 (GHARR-1) facility has been operating since its commissioning in March 1995 [1]. GHARR-1 is a commercial version of the Miniature

Neutron Source Reactor (MNSR) owned by the Ghana Atomic Energy Commission and operated by the National Nuclear Research Institute (operating organization) of the Commission. The facility was acquired with the assistance of the International Atomic Energy Agency (IAEA) under its Project and Supply Agreement (PSA) framework

As noted for all MNSR facilities operating presently, the GHARR-1 core employs a highly enriched uranium (HEU) fuel assembly. The 90.2% enriched core has a fuel burnup of 1% [2-4] and designed as a lifetime core with of 10 years before next cycle operations. Future cycle operations will require a replacement of the spent or depleted core with either HEU or low enriched uranium (LEU) cores.

Several external factors influence considerations or decisions to use LEU fuels as likely candidate replacement core for the GHARR-1 facility. It is indeed most likely that the front-end of the GHARR-1 nuclear fuel cycle for the next cycle operation will consists of an LEU core of different fuel type. For this and other reasons, 3-D Monte Carlo model has been developed for the simulation of nuclear criticality safety and neutronic analysis for the GHARR-1 facility. The neutronic analysis has been performed on a number of probably fuel types to establish a neutronic design basis for the GHARR-1 HEU-LEU conversion programme. In this paper, some results of the neutronic analysis using different enrichments of UO_2 (LEU) are presented.

In this paper, the brief description of the GHARR-1 Monte Carlo model and the results of the simulation and neutronic transport analysis for the HEU-LEU conversion studies using the MCNP transport codes are presented.

2.0.The GHARR-1 Facility

Technically, the GHARR-1 is a commercial version of the Miniature Neutron Source Reactor (MNSR) with thermal power rated at 30kW. The initial core configuration is a compact core consisting of a single fuel assembly of 344 U-Al (admixed in aluminum matrix) pins enriched to 90.2% [2-6]. Cold clean excess reactivity for fresh core is limited to about 4mk (1/2 β_{eff}) and the integral reactivity worth of the control rod is about -7mk, providing a core shutdown reactivity margin of -3mk. The fuel elements, dummy and tie rods are arranged in ten concentric zones or rings of structural lattices distributed about a central control rod guide tube. The core is under-moderated with an H/U atom ratio of 197. Under-moderation of the compact core contributes to a high negative temperature coefficient (-0.1mk/^{o}C) to boost its inherent safety properties [3]. Additionally, the excess reactivity of the reactor is limited to $\rho_{ex} \leq$ ½ β_{eff} [7]. This ensures that prompt criticality is not possible. Because of the safety provided by the combination of the reactor's limited excess reactivity (4mk under normal conditions) and its self-limiting power excursion response (due to its negative temperature coefficient), it is inconceivable that any situation could arise for reasons of safety that the reactor be quickly shutdown.

Core cooling is achieved by natural convection using light water. MNSR reactors have small and compact cores which facilitate neutron escape in both axial and radial directions. These reactors therefore employ heavy reflection on the side and underneath

the fuel cage with thick annulus and slab of beryllium alloy material respectively, to minimize neutron loses and conserve neutron economy as shown in the cross-sectional view (Fig.1).

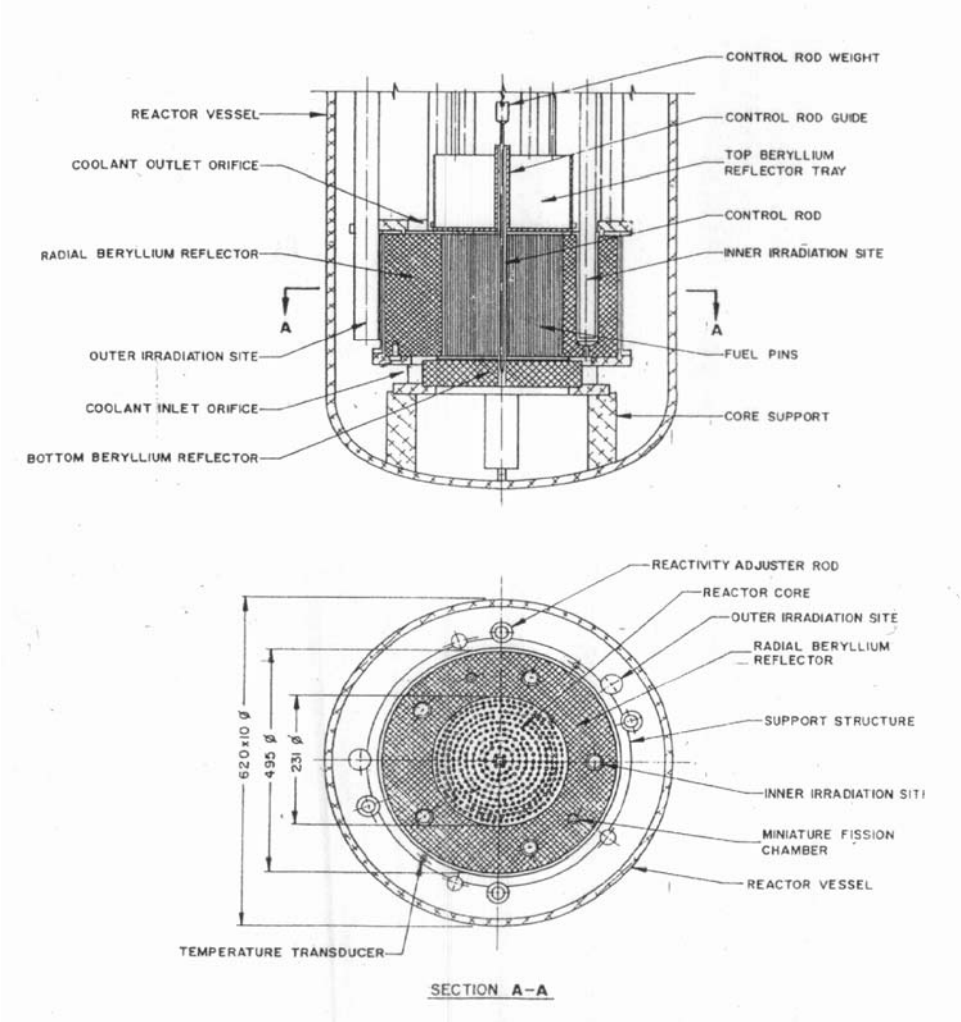


Fig.1: Cross section through the GHARR-1 reactor

3.0. Neutronic Analysis: 3-D Model and Method of Analysis

The versatile and multipurpose Monte Carlo particle transport code MCNP has been used in the development of a Monte Carlo model and subsequent simulation of neutronic behavior and reactor physics design characteristics of the GHARR-1 facility using both HEU and LEU cores. A detailed of the physical description of the GHARR-1 Monte Carlo model has been reported [1,8]. In particular, the model can be easily adapted with little modifications to other operating MNSR reactors with different core configurations.

Neutronically, the GHARR-1 Monte Carlo model provides for the simulation of reactor physics parameters such as nuclear criticality and core reactivities, neutron flux distribution in some selected locations of the reactor. In particular, neutron transport

simulations were done for clean fresh cores (zero burnup). Nuclear criticality calculations were performed to determine k_{eff} and corresponding core excess reactivities. For better statistics in this present analysis, the criticality specifications in the MCNP model provided for 400 kcode cycles and 500,000 source particles giving a total of 200 million neutron particle source histories. A 3-group neutron energy structure condensed from an initial 7-subgroup structure of the Hansen-Roach continuous neutron energy group was used in the neutronic analysis. The special $S(\alpha\beta)$ scattering feature of the MCNP code was applied in the nuclear model to treat thermal scattering in beryllium and hydrogen in light water for the reflector material and water regions respectively of the Monte Carlo model [1,6,8].

The MCNP plots of the GHARR-1 Monte Carlo model showing the core configuration (344 fuel elements), and vertical cross sectional views of the reactor in operational (control rod withdrawn) mode is presented in Figs. 2-3 respectively.

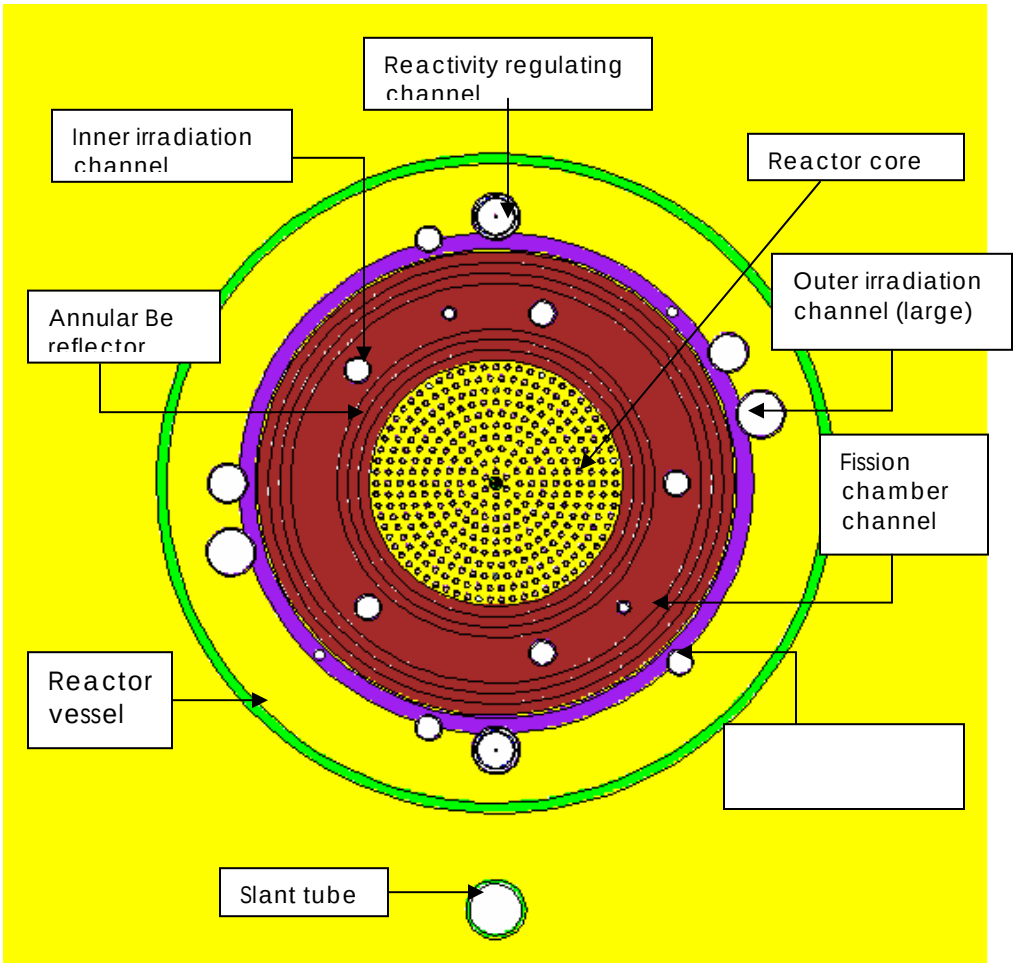


Fig. 2: MCNP5 plot of GHARR-1 core configuration showing fuel region (reactor core), channels for irradiation, fission chamber, regulating, slant and annular beryllium reflector

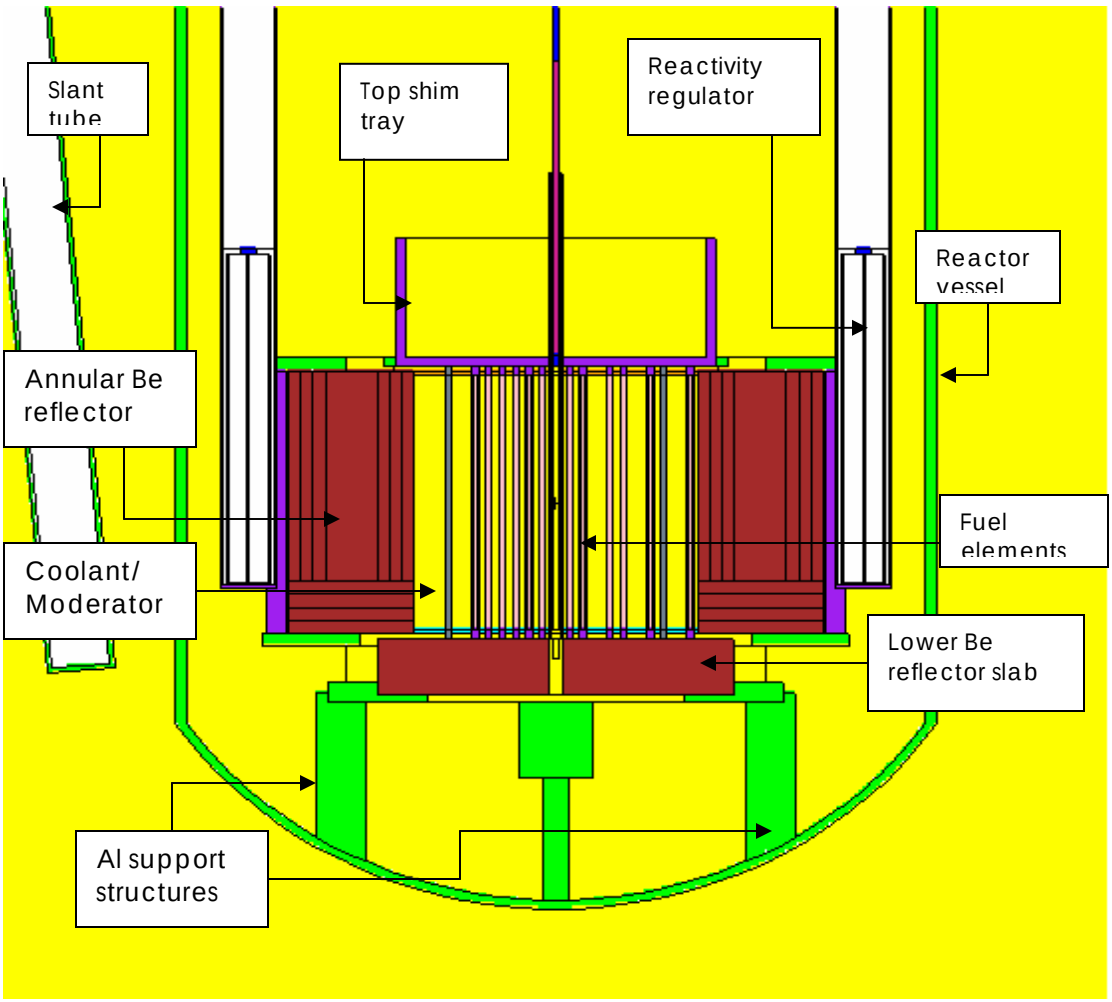


Fig.3: MCNP plot of vertical cross section of GHARR-1 reactor (control rod in full withdrawn position) showing structural supports, reactor vessel, etc.

In this work, the reference HEU (90.2% U-Al) and three other UO_2 LEU "candidate" cores with different lattice configurations and enrichments were considered in the Monte Carlo neutronic analysis for the GHARR-1 HEU-LEU conversion feasibility studies using the MCNP4c and MCNP5 [6] transport codes. The three UO_2 LEU cores consisted of 344 fuel elements (12.6% enriched, 6 Al dummies), 201 fuel elements (19.75% enriched, 149 water dummies), and 238 fuel elements (19.75% enriched, 112 Al dummies) were modelled.

4.0. Results and Discussions

The GHARR-1 Monte Carlo model developed using the versatile MCNP particle transport code has proved to be very effective for global neutronics analysis and simulation of reactor physics design parameters of MNSR reactors, including HEU-LEU core conversions studies. For purposes of brevity, only a few results are presented in this paper. A comparison of the Monte Carlo nuclear criticality (k_{eff}) calculations obtained for reference HEU and LEU candidate cores are indicated in Table 1.

Table 1: Monte Carlo (MCNP4c) simulated criticality parameters for GHARR-1 Facility: HEU vrs. LEU Fuel

Fuel type	Criticality parameter & Reactor status				
	Control rod withdrawn		Control rod inserted (shutdown)		
	k_{eff}	Excess reactivity (mk)	k_{eff}	Rod worth (mk)	Shutdown margin (mk)
HEU: U-Al (90.2%) 344 FE, 6 Al dummies	1.00454 ± 0.00007	4.5195	0.99701	-7.5526	-3.0331
LEU: UO_2 (12.6%) 344 FE, 6 Al dummies	1.00454 ± 0.00007	4.5195	0.99797	-6.5834	-2.0624
LEU: UO_2 (19.75%), 238FE, 112 Al dummies	1.00481 ± 0.00006	4.7870	0.99862	-6.1985	-1.4115
LEU: UO_2 (19.75%) 201FE, 149 H_2O dummies	1.00434 ± 0.00006	4.3213	0.99785	-6.5040	-2.1827

From this table, it is observed that the Monte Carlo calculated criticality (k_{eff}) values obtained for the HEU and zirc-4 clad UO_2 LEU cores are in close agreement. In particular, the 12.6% enriched UO_2 core using the same core configuration as the reference 90.2% HEU, yielded the same k_{eff} result of 1.00454. The simulated control rod worths for the reference HEU and various LEU cores also agree well with experimental results (6.8mk-7.0mk) reported for the centrally located control rod for the Ghana MNSR facility [2,7]. The lowest rod worth value of -6.2mk corresponding to a maximum deviation of 9% -11% was calculated for the UO_2 core (19.75%, 238 fuel elements with 112 Al dummies). However, in accordance with requirements for reactivity control and limiting conditions (OLC) for safe operation of the Ghana MNSR facility, the minimum and maximum reactivity worths of the cadmium control rod clad with stainless steel shall be 5.5mk and 7mk respectively [9]. Thus, for a core with excess reactivity of 4mk, the corresponding minimum and maximum shutdown margins shall therefore be -1.5mk and -3mk respectively. From the results of the Monte Carlo criticality safety simulations, all three UO_2 LEU cores satisfy these conditions and thus qualify on this basis as suitable candidate LEU fuels for conversion of the HEU-fuelled GHARR-1 facility and hence any other MNSR cores [6,8].

A simulation of the fission energy deposition on each fuel element and lattice zone provided for the establishment of the reactor power distribution across the GHARR-1 core for both HEU and LEU fuel assemblies. In particular, the fission power and fluxes peak at the centre of the fuel channels which was selected as the geometrical centre of the core in this model. The same trend was observed for the UO_2 cores with different configurations and enrichment considered in this work. In general, for a given axial location, the fission power and fluxes were observed to be higher as the number of fuel

elements in the fuel lattice zones increased. For brevity, graphical representation of the results of the Monte Carlo simulations of the fission energy deposited in for all the 10 fuel lattice zones in the GHARR-1 HEU reference and UO_2 cores (19.75% enriched, 238 FE + Al dummies) are illustrated in Fig. 4 and 5 respectively [6,8].

Fig. 4: Monte Carlo Simulation of Axial Variation of Fission Power Deposited in GHARR-1 Core (HEU Fuel)

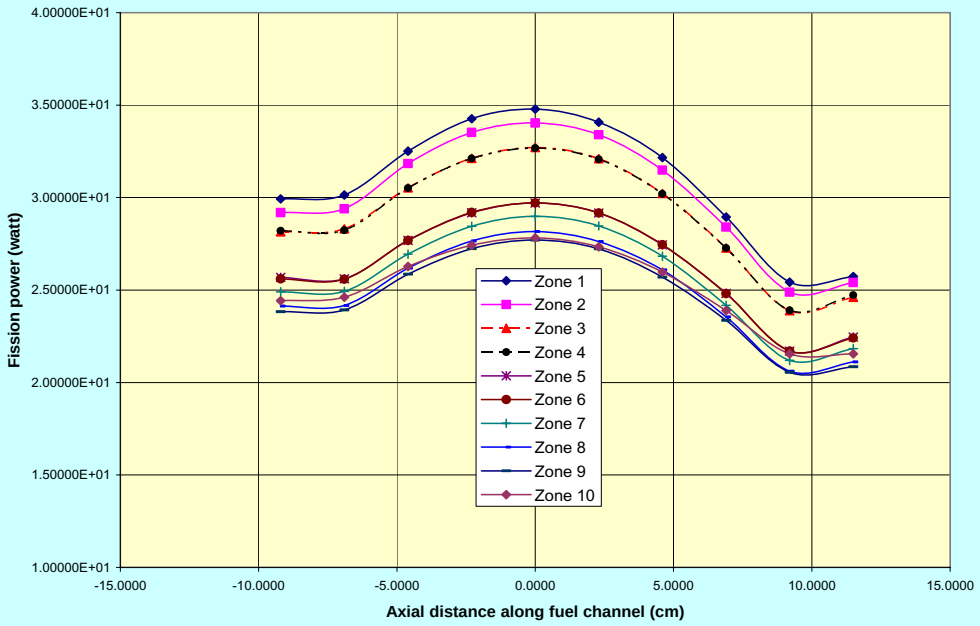
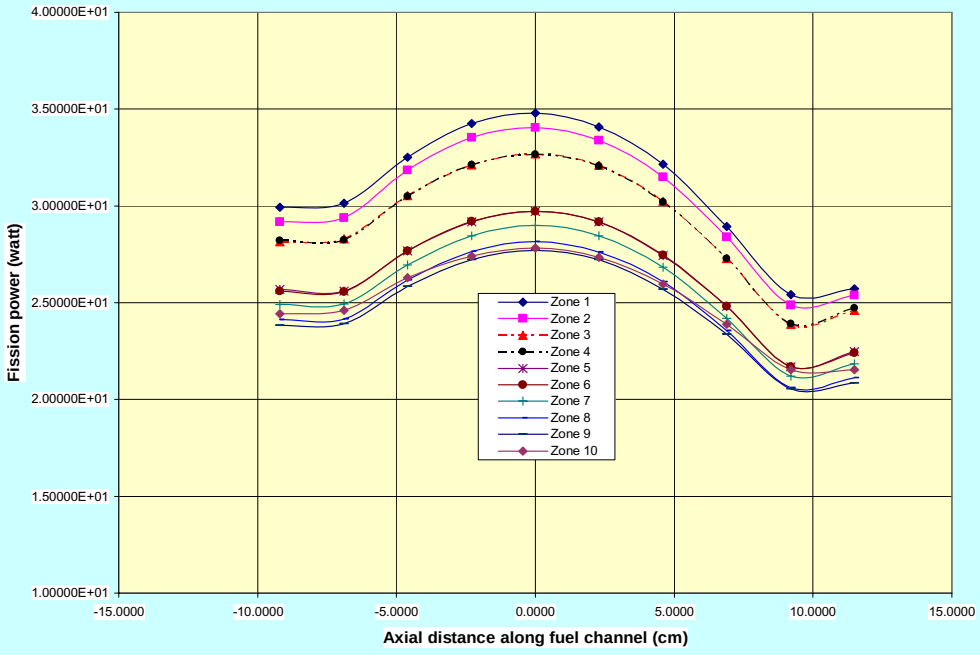
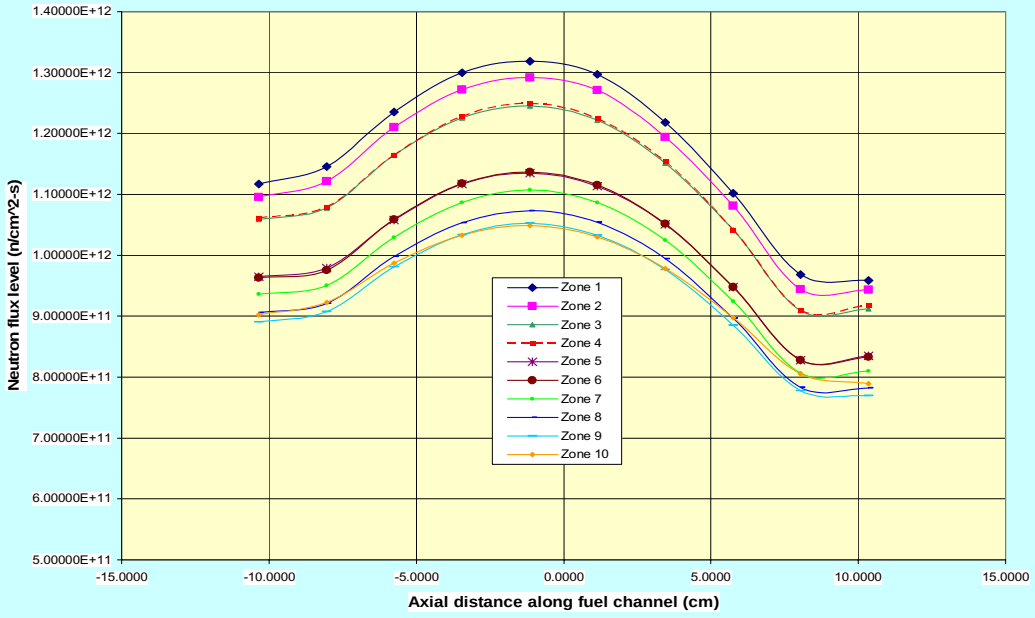


Fig. 5: Monte Carlo Simulation of Axial Variation of Fission Power Deposited in GHARR-1 Core (UO2 LEU fuel: 19.75%, + Al dummies)

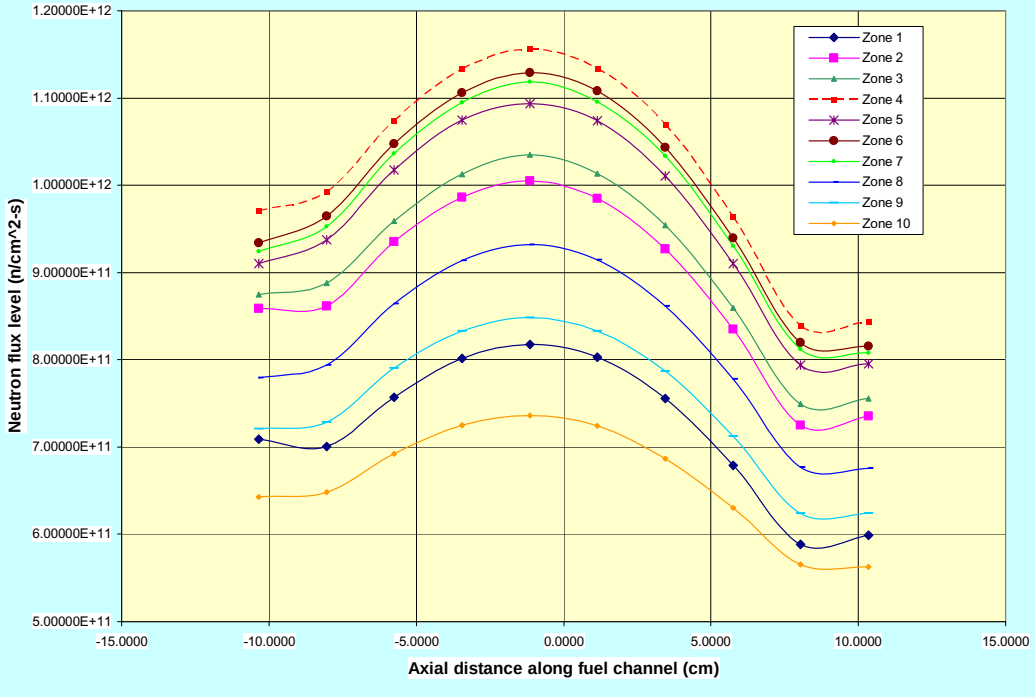


The Monte Carlo neutronic simulation of the axial neutron flux (thermal) distributions along the fuel rod channels in the HEU and LEU core regions are shown in Fig. 6 and 7 respectively. The corresponding experimental plot for the axial neutron flux intensity measured at *three* locations *between fuel lattices* is depicted in Fig.8.

Fi.g 6: Monte Carlo Simulation of Axial Neutron (Thermal) Flux Distibution in GHARR-1 Core (HEU)



Fi.g 7: Monte Carlo Simulation of Axial Neutron (Thermal) Flux Distribution LEU Core: UO2-19.75% (Al dummies)



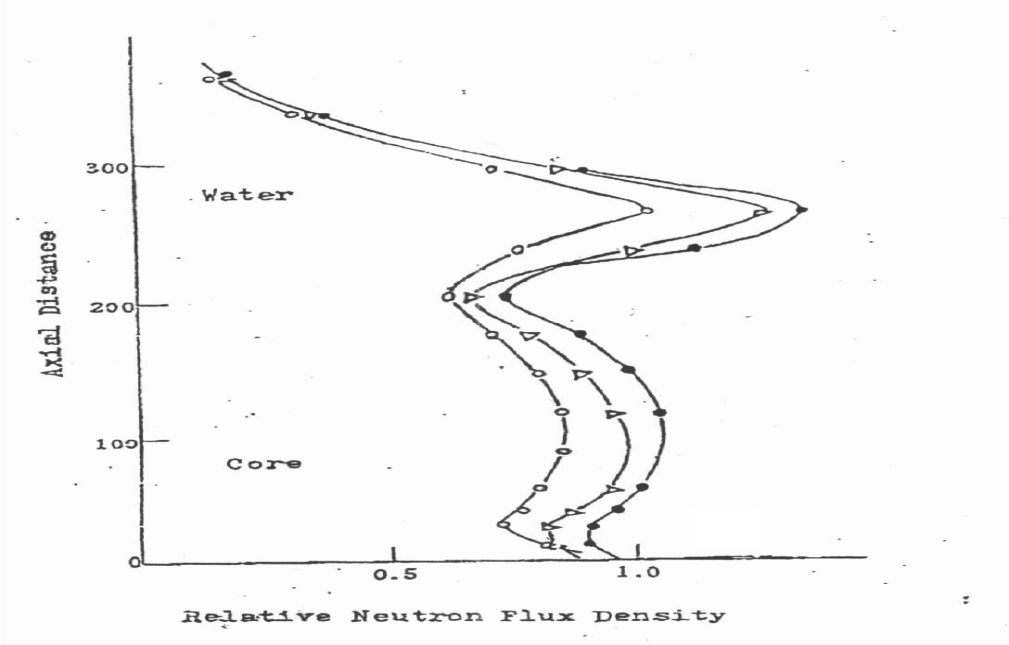


Fig.8: Relative axial flux density (experimental)

As observed from these plots, the experimental and Monte Carlo simulated results both show the same trend in the parametric (power and flux) distributions in the core. The observation shows that the GHARR-1 MCNP model is very good and thus establishes the fact the Monte Carlo model can be accurately and successfully utilized as an excellent tool in performing neutronic analysis of the MNSR reactor.

A comparison of the neutron flux levels simulated in the inner irradiation channels for HEU and LEU fuels are described in Fig.9. The neutron flux "trade-offs" or losses reported for three of the LEUs considered in this study are within 6%-10% with respect to the reference HEU core. In general, the GHARR-1 facility is operated at half full power (15kW) corresponding to a thermal neutron flux of 5.0E+11 n/cm^2.s. These "losses" will however, not affect utilization of the Ghana MNSR facility since the simulated results show that all fuels are capable of producing thermal fluxes within 10% of the rated maximum of 1.0E+12 n/cm^2.s recorded experimentally for the reference HEU core. Thus, neutronically, it can be concluded that all the three UO_2 LEU fuels qualify as candidate LEU options for core conversion of the GHARR-1 facility.

Simulation of the kinetic parameters of the Ghana MNSR using the Monte Carlo approach was also performed in the neutronic analysis for the GHARR-1 HEU-LEU conversion studies. Table 2 shows the Monte Carlo (MCNP5) simulated results for the delayed neutron fraction for the reference HEU and candidate LEU cores. The results reported in the GHARR-1 SAR [9] were computed using the diffusion code EXTERMINATOR-2. From Table 2, the results of both the Monte Carlo (MCNP5) and the diffusion (EXTERMINATOR -2) calculations are agreeable.

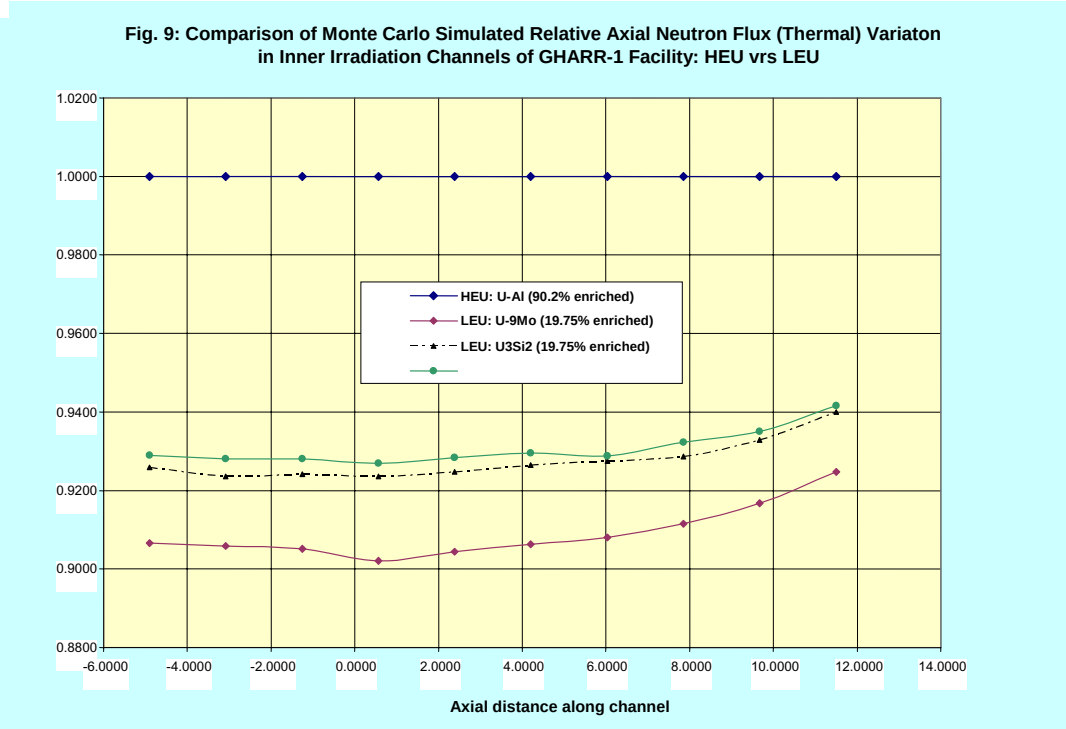


Table 2: Kinetic parameters of GHARR-1

	Criticality		Effective delayed neutron	
Fuel	k_{eff} (Total)	k_{eff} (delayed)	β_{eff} MCNP5	β_{eff} (EXT-2)
HEU fuel: U-Al alloy (90.2%)	1.00454	0.99618	8.3541E-03	
LEU fuel: UO_2, 12.6% 344 FE, 6 Al dummies	1.00454	0.99621	8.3240E-03	8.0800E-03
LEU fuel: UO_2, 19.75% (238FE + 112 Al dummies)	1.00481	0.99636	8.4402E-03	
LEU fuel: UO_2, 19.75% (201FE + 153 H_2O dummies)	1.00434	0.99609	8.2543E-03	

5. Conclusion

Neutronic analysis based on the Monte Carlo transport approach has been performed for the 30kW (th) Ghana MNSR (GHARR-1) facility. In particular, the neutronics analysis has been extended to the GHARR-1 HEU-LEU core conversion studies. The Monte Carlo model correctly simulates the neutronics and other design parameters of the MNSR reactor, as seen from the demonstrated results. Thus, the GHARR-1 Monte Carlo model developed using the versatile MCNP particle transport code has proved to be very effective for global neutronics analysis and simulation of reactor physics design

parameters of MNSR reactors, including HEU-LEU core conversions studies, as evident in its application to other MNSRs.

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JRR-3 MAINTENANCE PROGRAM UTILIZING ACCUMURATED OPERATION DATA

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ABSTRACT

JRR-3(Japan Research Reactor No.3) has been operated for more than 15 years after the modification, without significant troubles by carrying out maintenance such as the preventive maintenance (mainly time based maintenance) for the safety-grade equipment and the breakdown maintenance for the non-safety-grade equipment. Unscheduled shutdowns caused by aged non-safety-grade equipment have been increasing, and the resources have been decreasing year by year.

In this situation, JRR-3 maintenance program is reviewed about safety, reliability and economic efficiency.

This report offers the policy of the maintenance review and the future direction of maintenance programs.

1. Introduction

JRR-3(Japan Research Reactor No.3) achieved the first criticality in 1962, and shut down in 1983. In 1985, the modification started in order to upgrade the utilization performance. The modified JRR-3 was 20MW thermal power and light water cooled and reflected pool type reactor, and achieved the criticality in 1990. All the facilities except the reactor building were renewed in the modification. JRR-3 has been operated safely for more than 15 years without significant troubles.

JRR-3 had some reactor shutdown troubles caused by aged non-safety-grade equipment. The resources of human-power and costs tend to decrease year by year. In such situation, the maintenance program at JRR-3 is reviewed in order to search an efficient maintenance.

2. System of Maintenance Program

Maintenance is divided into the preventive maintenance (P.M) and the break down maintenance (B.M). (see Fig.1).

P.M is the maintenance to keep a function of equipment, which includes the time based preventive maintenance (T.B.M) and the condition based preventive maintenance (C.B.M).

T.BM is the one to carry out replacement and inspection on the planned time. T.B.M is subdivided into the periodic maintenance (PE.M) and the age based maintenance (A.M). PE.M. is the maintenance at regular intervals. A.M. is the one on planned operation time.

C.B.M consists of two steps. At the first step, the condition of equipment is observed continuously or intermittently. At the second step, the repair or replacement work is planned based on the observation results.

B.M is carried out when abnormality or malfunction of equipment occur.

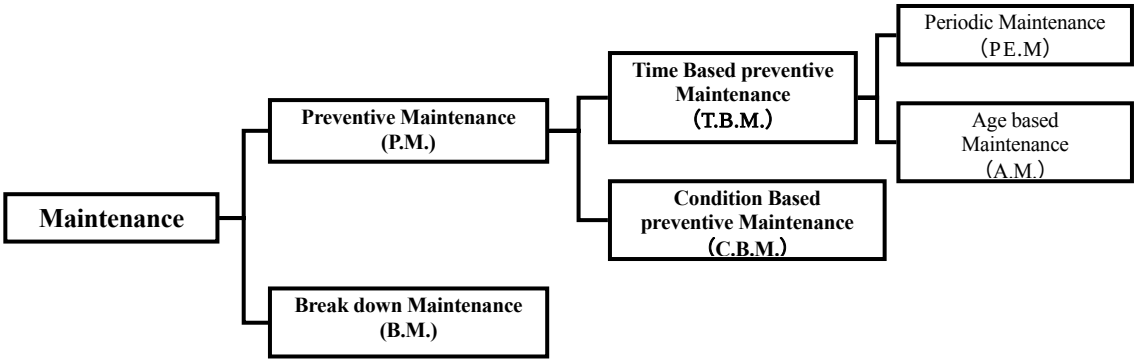

Fig.1 System of maintenance

T.B.M is effective to the parts of equipment that consumption and deterioration would be highly related with operation time. T.B.M is, however, not effective in costs if the parts have long remaining lifetime.

C.B.M has a possibility to cut down maintenance cost when both parts and equipment have been used to the maximum lifetime.

3. Present Maintenance Program

JRR-3 has two types of maintenance. One is for the safety-grade equipment including the reactor system, primary cooling system, siphon-break valve, secondary cooling pump, etc. These correspond with PS and MS-1, 2 and a part of class 3 on the "Guide for Safety Design of Water-cooled Research and Test Reactor Facilities in Japan".

The other one is for non-safety-grade equipment such as the secondary cooling fan, exhaust system and air supply system that are not directly related with safety of reactor.

The safety-grade equipment has been kept in good condition by carrying out overhaul and replacement of parts on T.B.M. The frequency of T.B.M. was decided based on the proposal by the equipment maker. The non-safety-grade equipment has been mainly done B.M.

Number of troubles at JRR-3 has increased because of non-safety-grade equipment aging from 2005. From these troubles, it was found that the accumulated data had not been used effectively. For example, the data was just only used to compare with the safety judgment criteria. It was necessary to utilize the accumulated data for searching symptom of abnormality.

4. Review of Maintenance Program

4.1. Review Policy of Safety-grade Equipment

Even for the same equipment, the equipment condition would change according to the atmosphere and operating time. In such a situation, the maintenance program of safety-grade-equipment is reviewed whether the equipment has received excessive maintenance or not. The review policies are outlined below.

(1) Keep established safety and reliability
(2) Refer to the accumulated maintenance data
(3) Optimize T.B.M.(by extending or shortening maintenance intervals) if the optimum maintenance interval can be judged by the accumulated maintenance data
(4) Consider to shift the maintenance program from T.B.M to C.B.M
(5) Continue the present T.B.M if it is not applicable to (3) or (4)

4.2. Review Policy of Non-Safety-grade Equipment

Maintenance program of non-safety-grade equipment is reviewed from the viewpoint of adoption of C.B.M. The review policies are outlined below.

(1) Keep established safety and reliability
(2) Refer to the accumulated maintenance data
(3) Set up performance indicator(P.I) by utilizing the accumulated data

Here, P.I. means the criteria in order to keep the performance of equipment.

5. Evaluation of Accumulated Operational Data

5.1. Review of Maintenance for Safety-grade Equipment

As an example of maintenance for the safety grade equipment, a flush of heat-exchanger is evaluated.

The primary cooling water flows through the body of the heat exchanger, and the secondary cooling water flows through the tube. Scale and corrosion product build up in the tube.

Condition of the heat exchanger has been maintained by the overhaul and the flushing. The overhaul has been carried out only once in the past. The flushing to keep the over-all heat transfer coefficient constant has been carried out 3 times per year (1 time per 2 or 3 cycles; here, 1 cycle is 4 weeks operation) as T.B.M. Figure 2 shows the relation between the number of flushing maintenance and the coefficient. Recovery of the coefficient was not observed from 2004 to 2005 regardless of the periodic flushing maintenance. This is because the flushing maintenance in 2004 to 2005 was carried out for the heat exchanger in good condition.

This result indicates that the present flushing maintenance has a possibility of excessive maintenance, and the flushing should be shifted to C.B.M. In other words, the timing of flushing maintenance should be planned by observing the coefficient at regular time intervals. The P.I was, then, set to about 1900W/ m^2 • K.

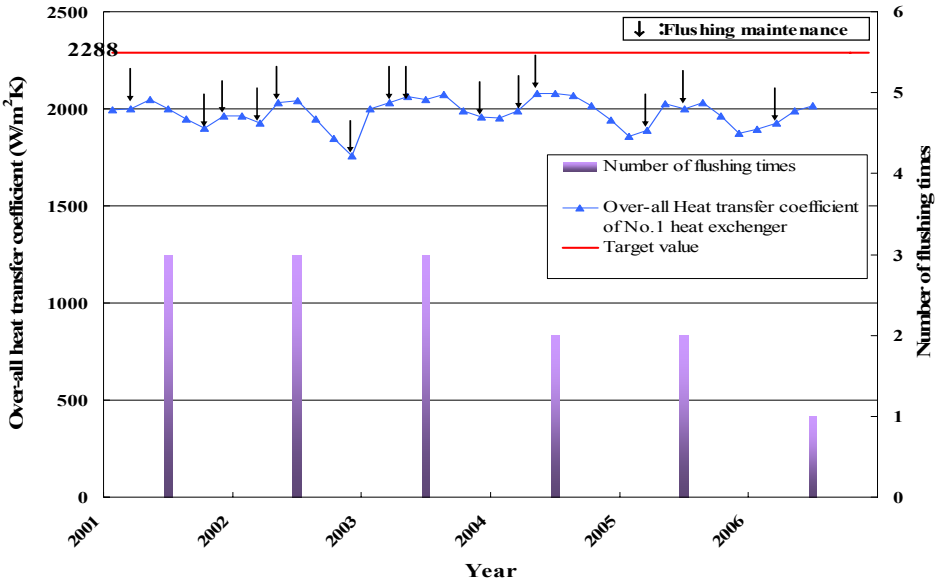

Fig.2 Relation between flushing maintenance and Over-all Heat Transfer Coefficient

5.2. Review of Maintenance for Non-Safety-grade Equipment

An example is the air blower for ventilation of the reactor room. The air blower consists of motor, shaft and blade. This blower had a trouble in the shaft in 2006. The blower has neither the periodic maintenance nor the overhaul in the past. But, the vibration data of bearing were accumulated every

year. Fig.3 shows the vibration data.

The vibration had been stable till 2003. But it increased after 2003, and then the bearing was replaced in 2005. Nevertheless, it did not recover to the stability (about 50μm) before 2003. If the bearing vibration data were observed long-term, it would be possible to find out other causes like the shaft wastage aside from the bearing wastage. It was necessary to measure the size of the shaft at the bearing replacement on 2005.

From this review, it was decided for the blower that the vibration measurement and the sound detection inspection should be done as P.I. once per half a year. And the management-criterion of P.I. is 60μm at the bearing. If the vibration approaches to the criteria, the overhaul which includes both the bearing replacement and the shaft size-measurement would be planned.

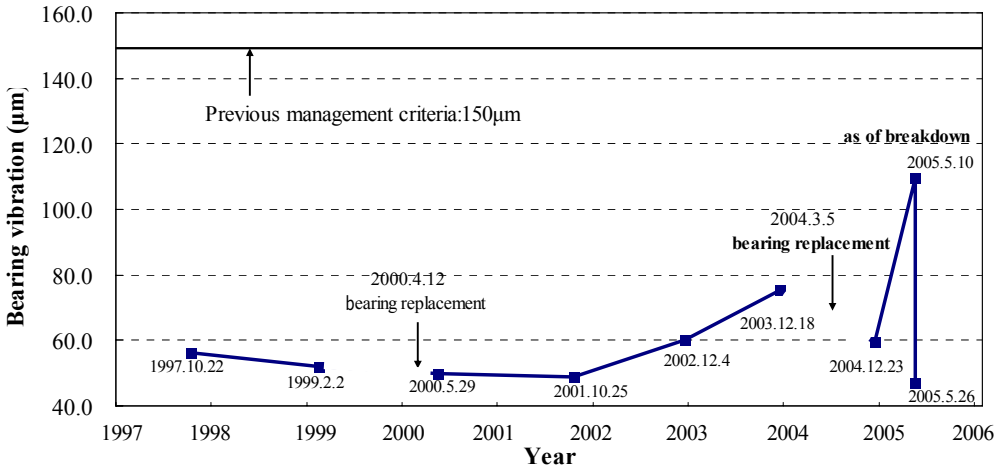

Fig.3 Vibration data of air blower

6. JRR-3 Maintenance Program in Future

6.1. Maintenance Program of Safety-grade Equipment

JRR-3 continues to optimize T.B.M because the maintenance of some equipment may have excessive maintenance. In addition, JRR-3 continues to shift the maintenance from T.B.M to C.B.M. and takes more data such as size measurement or vibration at before-and-after of overhaul. The aging speed of parts and equipment may become clearer by the data, and optimizing the interval of T.B.M would be possible. Therefore, it is expected that safety and reliability of equipment would go better.

6.2. Maintenance Program of Non-Safety-grade Equipment

JRR-3 continues to shift the maintenance of non-safety-grade equipment to C.B.M. and searches P.I by utilizing accumulated data. In adoption of C.B.M, it needs to search an effective maintenance without losing the safety and reliability.

7. Conclusions

Effective use of the maintenance resource can be expected by establishing the effective maintenance program using the accumulated data.

In JRR-3, by optimizing T.B.M and adopting C.B.M, reliability and condition of equipment would be better. Furthermore, future data accumulated by this maintenance program would be useful to JRR-3 safety operation.

RADIATION PROTECTION DESIGN OF THE FRM II

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ABSTRACT

The major radiation protection design features in the design of the FRM II – already in operation since 2004 – are presented like zoning, building structure, and separation of systems, shielding concept and access concept. All these planning topics resulted in the radiation survey program for the commissioning of this research reactor.

1. Introduction

The FRM II is a research reactor with 20 MW thermal reactor power. It is moderated with D_2O and cooled with H_2O. The core consists of a single fuel element and one central control rod. The D_2O filled moderator tank containing the fuel element is situated in a pool with H_2O. The general contractor was Siemens KWU; the work was performed by AREVA NP GmbH on behalf of Siemens. The operator now is the Technical University of Munich (TUM). The concept design was developed during the early 90s, the construction took place from 1996 till 2001. After a long, additional phase of intensive review by authorities until 2003, the hot commissioning started during 2004. The reactor reached its full power for the first time in August 2004. It is to be emphasised that the same rules and criteria for radiation protection were applied as for commercial nuclear power plants.

2. Zoning

2.1 Type of Zones

Following the German Radiation Protection Ordinance the following zone types must be distinguished:

- Public Area – In case of the FRM II this zone begins at the fence of the facility. Assuming the continuous presence of an individual there the annual dose must not exceed 1 mSv.
- Monitored Area – This zone comprises all within the fence except the reactor building. Considering the real presence in a certain location the annual dose of an individual there must not exceed 6 mSv.
- Restricted Area – This zone comprises the reactor building, an attached cellar area, the activity monitoring room of the vent stack and a so-called neutron guide tunnel in the neutron guide hall (Fig. 5). The annual dose of an individual must not exceed 20 mSv.
- Exclusion Areas: – These zones are special, limited areas – normally rooms where the dose rate exceeds 1 mSv/h. Examples are the chamber for the primary pumps and coolers, the filters for the primary and D_2O coolant, the He-gas system, the neutron guide tunnels in the reactor building (Fig. 1) and in the neutron guide building (Fig. 5) and the buffer stores for radwaste (Fig. 3).

2.2 Room Classes

The rooms in the restricted area were classified as follows:

- Class 0 – maximum dose rate ≤ 5 µSv/h, no contamination therefore no request for specific protective clothes, unlimited presence; example: experimentation hall (Fig. 1) in the reactor building.
- Class 1 – maximum dose rate ≤ 5 µSv/h (locally ≤ 10 µSv/h), potential low contamination, therefore request for protective clothes against contamination but unlimited presence; example: maintenance floor below experimentation hall (Fig. 3).
- Class 2 – maximum dose rate ≤ 1 mSv/h, potential contamination, limited presence without special radiation protection permission; example: pump rooms of purification systems (Fig. 4).
- Class 3 – dose rate can exceed 1 mSv/h, potential contamination, normally locked, access only with special radiation protection permission; example: chamber for pumps and coolers of the primary cooling system and purification of D_2O system.

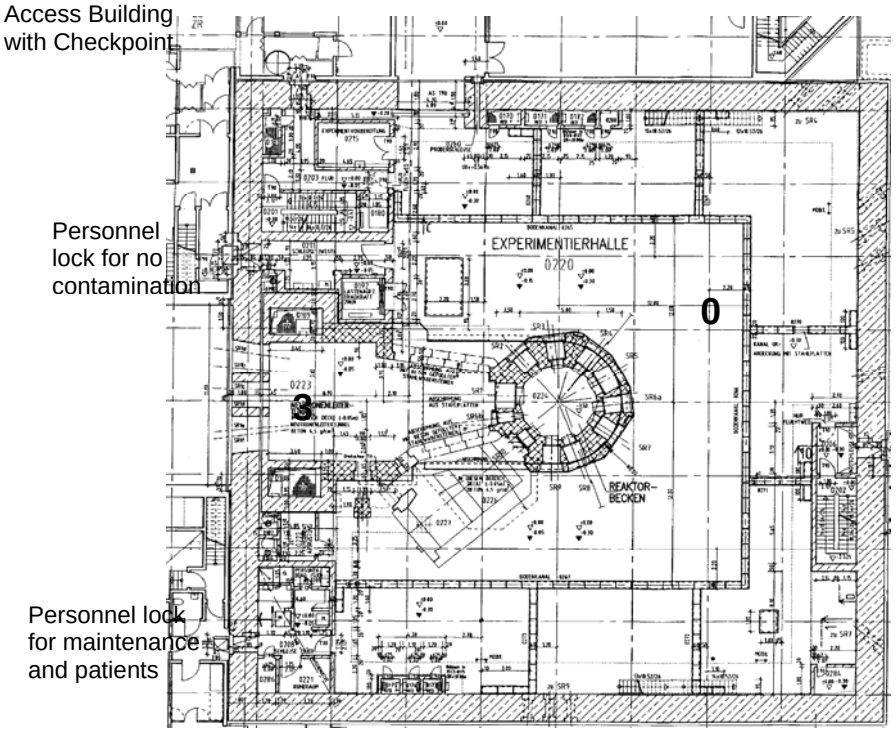

Fig. 1 Experimentation Hall

3. Building Structure and Separation of Systems

The building structure reflects very clearly the zoning and the principle of separation of systems especially with primary coolant (Fig. 2+4) separated from the moderator system (Fig. 3) and both separated from systems which do not contain radioactive media. The ventilation air flow is directed from rooms with low contamination potential to such with potential higher contamination potential.

4. Shielding

4.1 Shielding Targets

Targets for the shielding design were keeping the dose limits based on the zoning concept (see 2.1) and keeping the dose rate limits due to the room classes (see 2.2). Furthermore in case of areas or locations with a significant occupancy time during the year the annual dose to an individual executing his regular work there shall not be higher than 10 mSv considering the anticipated annual presence (examples: reactor hall in Fig. 2, experiment hall in Fig. 1).

4.2 Shielding Materials

The following materials were used for shielding:

- Normal (ordinary) concrete, $\rho \geq 2.1$ g/cm³ - in general

- Heavy (hematite) concrete, $\rho \geq 3.6$ g/cm³; examples: the upper wall sections of the reactor pool, the lower wall sections of the fuel pool, and the ceiling of the chamber for the primary cooling system
- Heavy (hematite) concrete with steel shroud, $\rho \geq 4.5$ g/cm³; examples: the lower sections of the reactor pool in core height, the operator's side of the hot cell, and the lateral walls of the neutron guide tunnel in the reactor building
- Steel and lead plates, examples: 2 floor sections of the fuel pool, the operator's side of the hot cell, the fuel transfer duct, active zones of the D_2O filters, and partially the heavy concrete walls of the neutron guide tunnel
- Water – in case of the combined reactor and fuel pools.

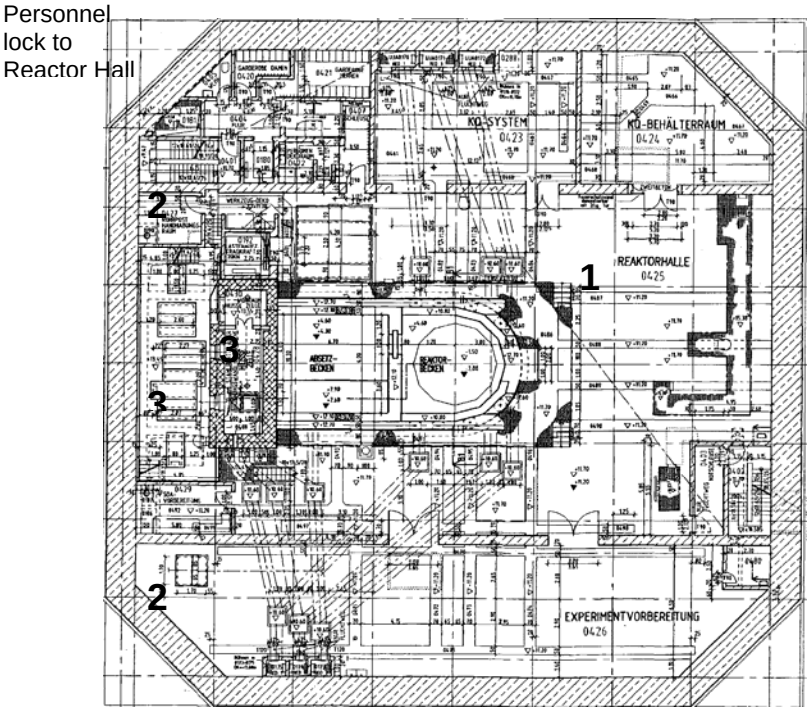

Fig. 2 Reactor Hall

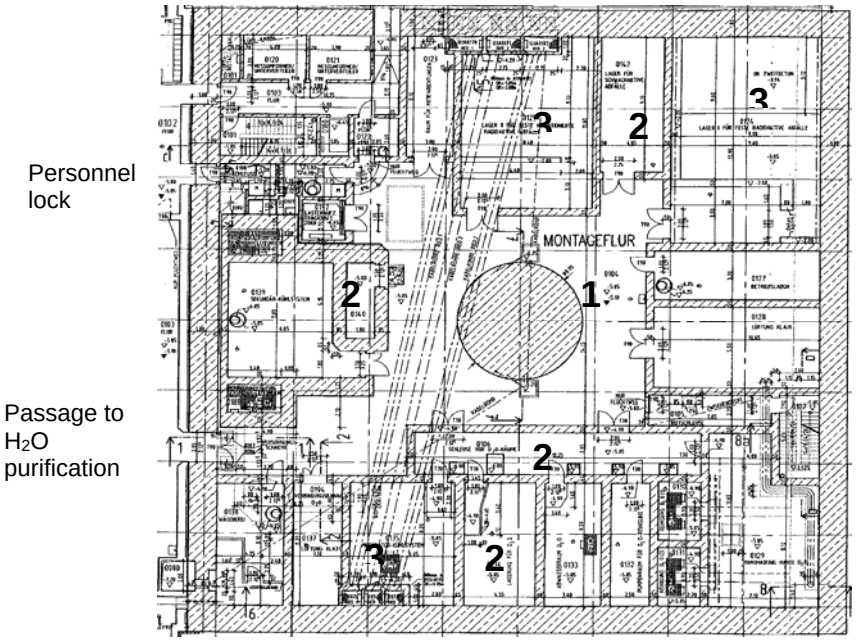

Fig. 3 Cellar of Reactor Building

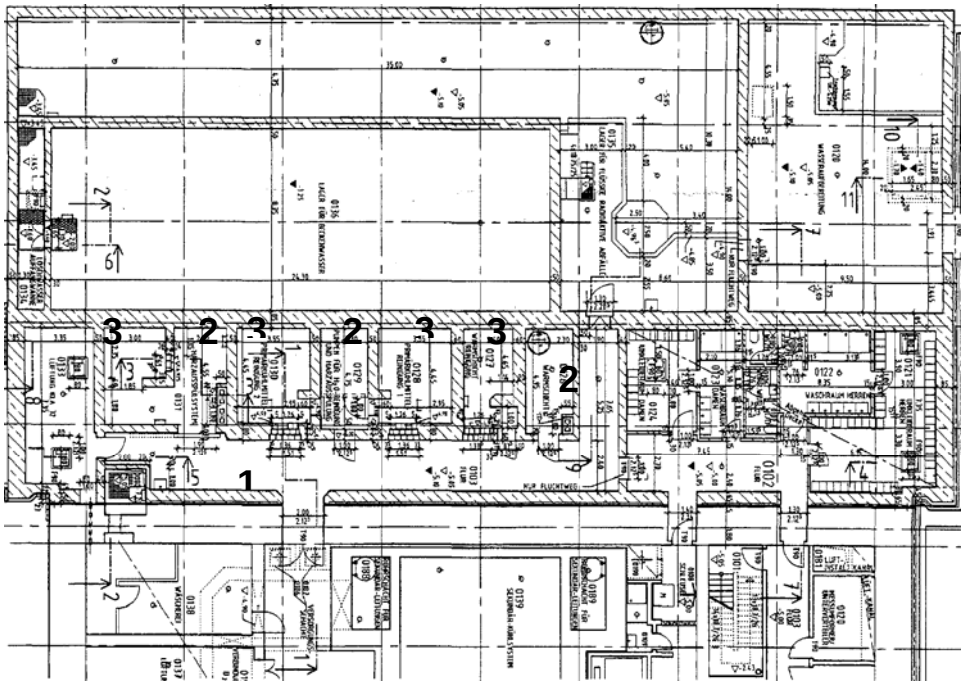

Fig. 4 Cellar for H_2O Purification and Liquid Waste

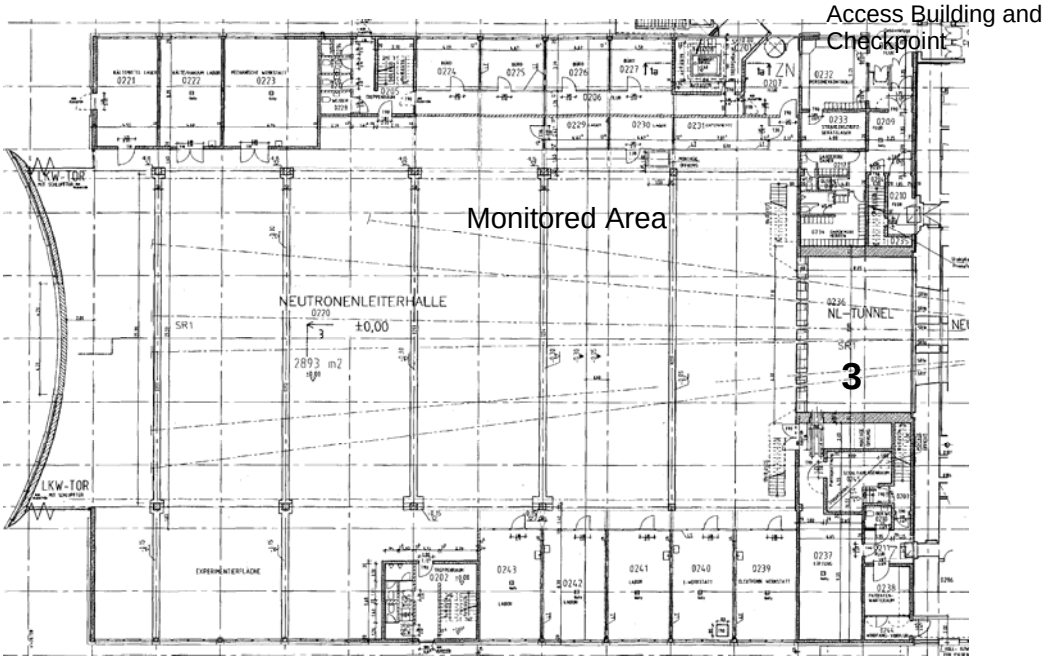

Fig. 5 Neutron Guide Hall

5. Access Concept

All access to the controlled area as well as to the experimentation areas outside the restricted area is via a general security checkpoint in the access building in front of the reactor building (Fig. 1, top).
From there the entrance into the reactor hall is via the personnel lock in the North West corner on the same building level.
In case of the cellar the corresponding lock is in the North West corner on the same level (Fig. 3).
The experimentation hall can be accessed via two different routes in a case by case dependence:

- During experiment operation without contamination potential the personnel lock in the North West corner is used with protective clothes; that lock is equipped with a hand-feet contamination monitor only (Fig. 1).

- During states with contamination potential the personnel lock in the South West corner is used which in addition serves as access for patients receiving cancer irradiation (Fig. 1); protective clothes must be worn.

Even the neutron guide hall (Fig. 5) does not belong to the restricted area, persons leaving it are required to use a hand-feet contamination monitor at its two exits. The entrance is separated from that to the restricted area, but the access route is via the access building, too.

6. Commissioning

6.1 Radiation Survey Program

The radiation survey program was conducted during the 7 phases of the hot commissioning at 0.2, 2, 6, 10, 14, 18 and 20 MW reactor power. It comprised 5 sub-programs:

- A dose rate survey program for gamma and neutron dose and dose rate,
- Regular control and evaluation of readings of dose rate and activity monitors, comparison of raised indications with design values and reference thresholds,
- Measuring of the activity concentration in systems by sample collection, evaluation of these samples in laboratories with high resolution gamma-spectrometry and comparison of the results with design values,
- Measuring of the ^{41}Ar activity concentration in various ventilation sections by sample collection with compressors, evaluation in a laboratory and comparison of the results with design values,
- Measuring of the ^{3}H activity in D_2O and D_2 systems by sample collection, evaluation in a laboratory and comparison of the results with design values.

During the execution of all 5 sub-programs the found raised indications were correlated with operation conditions. The results were regularly discussed with the authorised experts from the surveillance authority.

6.2 Results

In general the admissible levels for dose rate and activity release via the stack were not exceeded. Exceptions were caused – and this could be logically proved – from operation conditions, unforeseen events a. s. o. In a few cases local modification and adaptation of systems were performed to eliminate unforeseen increased dose rate for the operator (examples see Fig. 6 and 7 below).

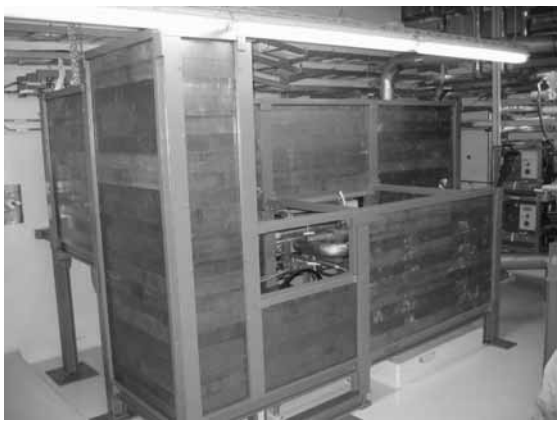

Fig. 6 Lead shielding around the D_2O cooler and the associated pump and valves

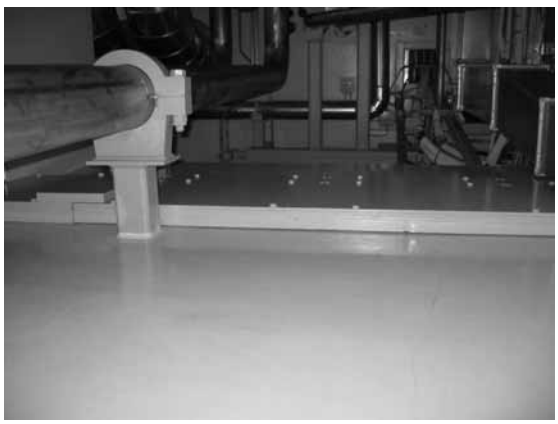

Fig. 7 Additional steel protection of a concrete gap in the floor of the tunnel for D_2O piping

7. Summary, Conclusion

The radiation protection design fulfils the standard regulation for nuclear plants without exemptions for research purposes. The design allows further modification and adaptation of the facility. After now approx. 3 years of operation no single case of exceeding legal limits for radiation exposure of personnel, scientists and the public has occurred.

IDENTIFICATION OF A LEAKING TRIGA FUEL ELEMENT AT THE NUCLEAR REACTOR FACILITY OF THE UNIVERSITY OF PAVIA.

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During a periodical activity of characterization of the ionic-exchange resins of the demineralizer of the primary cooling circuit of the TRIGA Mark II reactor of the University of Pavia a small but detectable amount of ^{137}Cs contamination was measured. Since the reactor has been running for several hundreds of hours at full power without showing any anomaly in the radiometric and thermo-hydraulic parameters, the reactor was brought at the nominal power of 250 kW for one hour and a sample of water was collected from the reactor tank and analysed in a low-background gamma-ray detector.
As a result a small amount of fission products were detected in the reactor pool water (few Bq/g) suggesting the existence of a possible clad defect in one ore more fuel elements. Since no halogens such as iodine and bromine were detected in the sampled water, the more probable hypothesis, also supported by literature, seemed to be a micro-fissure in the neck of an instrumented fuel element. A dedicated apparatus for reactor pool water sampling and on-line spectroscopy measurements was realized. The leaking fuel element was identified and removed from its position and the reactor was back in regular operation after 2 months from leakage detection.

1. Introduction

During a periodical activity of characterization of the ionic-exchange resins of the demineralizer of the primary cooling circuit of the TRIGA Mark II reactor of the University of Pavia a small but detectable amount of ^{137}Cs contamination was measured. As a consequence of the unusual fission products activity detected a campaign of gamma-ray spectrometry was implemented in order to evaluate the origin and the importance of the release.

2. Measurements campaign

Using a HGe (1.72 keV FWHM – 31.3% efficiency – 58.5 Photo Peak/Compton) many analysis were performed and the most significant ones are presented below.
Notice that all the results of the measurements presented have a relative error less than 10% and that the MDA was evaluated according to the "*RISO*" methodology.

2.1 *Measurement of a sample of the ionic-exchange resins of the demineralizer of the primary cooling circuit* (Activity reported to last day of operation of the reactor at nominal power):

Radioisotope	**$T_{1/2}$**	**Bq/g**
Co-60	5.26 y	129,8
Zn-65	244 d	17,4
Mn-54	312 d	123,8
Co-58	70.8 d	86,4
Cs-134	2 y	2,86
Cs-137	30.2 y	245
Cr-51	27,7 d	265

Tab.1 – Activity of the ionic-exchange resins of reactor primary cooling circuit

2.2 Smear-Test of the external surface of a fresh fuel element.

The measure was performed in order to exclude the hypothesis that the clad of the new SST fuel elements were contaminated by a small amount of fission products. The measure didn't show any presence of such contamination (MDA = 2,75 10^{-4} Bq/cm^2 at 661 keV).

2.3 *Measurement of a sample of the water filter of the primary cooling circuit* (Activity reported to last day of operation of the reactor at nominal power):

Radio-isotope	**$T_{1/2}$**	**Bq/g**
Eu-152	13,6 y	10,5
Co-60	5,26 y	36,6
Fe-59	44 d	11,6
Eu-154	8,8 y	1,5
Zn-65	244 d	4,9
Sc-46	83,8 d	3,5
Mn-54	312 d	2,5
Co-58	70,8 d	1,04
Cs-134	2,0 y	0,38
Cs-137	30,2 y	1,19
Cr-51	27,7 d	204

Tab.2 – Activity of a sample of the water filter of the primary cooling circuit

2.4 *Measurement of a sample of water collected from the reactor pool before to operate the reactor test at nominal power* (Activity reported to last day of operation of the reactor at nominal power):

Radio-isotope	**$T_{1/2}$**	**Bq/g**
Co-60	5,26 y	5,60 10^{-04}
Zn-65	244 d	4,10 10^{-04}
Mn-54	312 d	2,80 10^{-04}
Co-58	70,8 d	3,50 10^{-04}
Cs-134	2,0 y	< 9,33 10^{-05}
Cs-137	30,2 y	1,50 10^{-03}

Tab.3 – Activity of a sample of reactor pool water before reactor operation test

2.5 Measurement of a water sample collected from the reactor pool after one hour of operation at 250 kW nominal power.

The sample was collected 30 cm below the pool water surface with the primary cooling system of the reactor switched off. The analysis was performed after 90 min from the collection of the sample giving the following results:

Radio-isotope	**$T_{1/2}$**	**Bq/g**
Cs-138	32,2 m	7,08
Xe-138	14,13 m	5,51
Ar-41	109 m	14,82
Na-24	15 h	3,58
Mn-56	2,58 h	0,53
Kr-88	2,84 h	0,70
Rb-88	17,8 m	21,67
Kr-85m	4,48 h	0,14
Kr-87	76,3 m	1,04
Cl-38	37,21 m	0,57
Xe-135	9,11 h	0,04

Tab.4 – Activity of a reactor pool water sample after 1 hour of operation at nominal power (short half-life)

The same sample was analysed after 96 hours and 41 min in order to look for long life radioisotopes. The results of the measurement were reported, for comparison, to the same time of the data displayed in Tab.4.

Radio-isotope	$T_{1/2}$	Bq/g
La-140	40,22 h	$4{,}46\ 10^{-03}$
Na-24	15 h	3,68
Co-60	5,26 y	$5{,}63\ 10^{-04}$
Mn-54	312 d	$6{,}66\ 10^{-04}$
Co-58	70,58 d	$7{,}69\ 10^{-04}$
Cs-134	2,0 y	$< 1{,}74\ 10^{-04}$
Cs-137	30,2 y	$1{,}84\ 10^{-03}$
Cr-51	27,7 d	$2{,}13\ 10^{-03}$

Tab.5 – Activity of a reactor pool water sample after 1 hour of operation at nominal power (medium and long half-life)

2.6 Measurement of aerosol sample collected above the pool water surface.
The measurement was performed sampling 3 m^3 of air on a active-carbon and absolute filter (porosity 4,5 μm). The spectrometry of the filter was performed 60 min after the sampling and besides natural radioisotopes only 22 Bq/m^3 of ^{138}Cs was detected.

2.7 Measurement of aerosol sample collected in the off-gas channel of the reactor.
The measurement was performed sampling 50 m^3 of air on a active-carbon and absolute filter (porosity 4,5 μm) and only natural radioisotopes were detected (MDA for $^{138}Cs = 7{,}5\ 10^{-2}$ Bq/m^3; MDA for $^{137}Cs = 5{,}80\ 10^{-4}$ Bq/m^3).

3. Preliminary evaluation of the release

In order to evaluate the typology of the release of the fission products the relative abundance of the noble gas was calculated and compared with the specific activities measured in the sampled water. The results of the calculation as a function of the Fission Reaction Rate (*R*) are reported below:

^{138}Xe $[\lambda_{Xe}N_{Xe}(1h) = 5.97\ 10^{-2}\ R]$
- direct production (Y= 4.81 %) $\lambda_{Xe}N_{Xe}(1h) = 4.56\ 10^{-2}\ R$
- production from ^{138}I (Y = 1.49 %) $\lambda_{Xe}N_{Xe}(1h) = 1.41\ 10^{-2}\ R$

^{135}Xe $[\lambda_{Xe}N_{Xe}(1h) = 3.79\ 10^{-4}\ R]$
- direct production (Y= 0.0785 %) $\lambda_{Xe}N_{Xe}(1h) = 5.70\ 10^{-5}\ R$
- production from ^{135}I (Y = 6.28 %) $\lambda_{Xe}N_{Xe}(1h) = 2.36\ 10^{-4}\ R$
- production from ^{135m}Xe (Y = 0.178 %) $\lambda_{Xe}N_{Xe}(1h) = 8.57\ 10^{-5}\ R$

^{88}Kr $[\lambda_{Kr}N_{Kr}(1h) = 7.69\ 10^{-3}\ R]$
- direct production (Y= 3.55 %)

^{87}Kr $[\lambda_{Kr}N_{Kr}(1h) = 1.075\ 10^{-2}\ R]$
- direct production (Y= 2.56 %)

^{85m}Kr $[\lambda_{Kr}N_{Kr}(1h) = 1.85\ 10^{-3}\ R]$
- direct production (Y= 1.29 %)

The ratios between the calculated abundances and the measured specific activities of noble gas in the reactor pool water sample (see Tab.4) are displayed in Tab.6:

Radio-isotopes	Calculated Ratio	Measured Ratio
$^{138}Xe/^{135}Xe$	157	138
$^{87}Kr/^{88}Kr$	1.40	1.48
$^{138}Xe/^{87}Kr$	5.55	5.30

Tab.6 – Ratio between calculated abundances and measured specific activities of noble gas detected in the reactor pool water sample

From this evaluation it seemed clear that the noble gas were release promptly in coincidence with the reactor operation at the power of 250 kW.
For what concerns the specific activity in air of ^{138}Cs (22 Bq/m^3) measured above the pool water surface, it was consistent with the specific concentration in water of the radio nuclide, presuming an evaporation coefficient of $\sim 10^{-3}$, realistic for the reactor pool water temperature of 40 °C.
On the contrary, the increase of the specific activity in water of ^{137}Cs after the operation of the reactor for 1 hour at the power of 250 kW (comparison between Tab. 3 and Tab.4), it was not consistent with the hypothesis of a prompt release unless about 30% of the fuel elements of the core were fissured. This hypothesis seems to be not realistic because, since no halogens such as iodine and bromine were detected in the sampled water, about 24 fuel elements should present a micro-fissure. A possible explanation of this anomalous increase was that ^{137}Cs could be dissolved into the moisture that could be accumulated inside the damage fuel element when the reactor was not in operation and which could be released all at once when the fuel element was heated up.

4. The sampling and detection apparatus

Following the experience of University of Utah Nuclear Engineering Laboratory, a sampling and measurement apparatus was realised with the following components (see Fig.1):

- an aluminium anticorodal tube (length 6 m ∅ 25,4 mm) with a funnel terminal (∅ 55 mm)
- a hydraulic pomp OMAN mod. ALM25 in SST (max flow 43 lt/min – Prevalence 12 m)
- a rubber tube for water circulation (∅ 25.4 mm)
- a HPGe Ortec GMX (n-type, Coaxial Detector, Be Window) - FWHM 2.57 keV at 1.33 MeV ^{60}Co – Efficiency 30% - Photopeak/Compton = 46/1

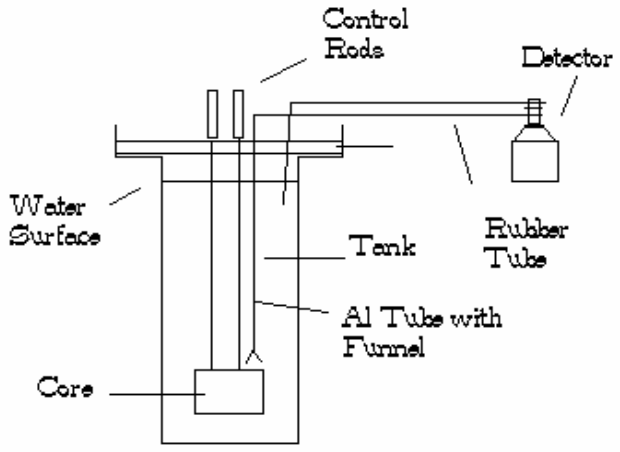

Fig. 1 - Sampling and detection apparatus lay-out

Water was collected from the superior grid of the reactor core by means of the aluminium tube in different sectors and in different fuel element positions and was counted on-line using the HPGe detector positioned in the Radiochemistry Laboratory (about 20 m distance from the reactor top).
In order to allow the decay of short half-life radioisotopes such as ^{19}O, ^{16}N, the pump suction flow was reduced to 5 lt/min, that means that the sampled water took at list 2 minutes to reach the detector.
The radioisotope considered for the measurements was ^{138}Xe that presents three well defined and clean gamma-peak at 258 keV, 434 keV, 1768 keV.

5. Identification of the leaking Fuel Element.

The reactor core was virtually divided into 4 sector: N°1 (Sud-Est), N°2 (Sud-West), N°3 (Nord-West), N°4 (Nord-Est).
After testing the sampling apparatus before to start the reactor, the reactor was operated at the power of 1 kW and the water was sampled in all four sector at a distance of about 15 cm from the superior grid of the reactor. In this condition no fission products were detected in the water.
Thus the reactor power was raised up to 50 kW (i.e. a fuel temperature about 45 °C) and the same investigation in all four sectors was repeated, but still no fission products were detected.

The reactor power was then raised up to 100 kW (i.e. a fuel temperature about 90°C) and finally fission product were detected starting from sector N°3.
Unfortunately the specific activity measured in each sector ended up to be of the same magnitude suggesting two possible explanations: there were more than one fuel element fissured positioned in different sectors of the reactor core or the water of the pool mixed up very fast in proximity of the superior grid preventing the possibility of identifying the sector of origin of the release.
Anyway it was clear that a more systematic analysis, fuel element per fuel element, should have been performed.
Thus the aluminium tube was lowered down towards the grid in such a way that the funnel covered just one fuel element position.
The water sampling was repeated until when two SST clad fuel element, close one to the other in sector N°3 (position C5 and C6), seemed to be the possible origin of the release. These two elements, though, were close to three instrumented fuel elements and, knowing from literature that these kind of elements are more likely to undergo fissure, the hypothesis of their involvement in the release was still the more probable.
In order to verify this last hypothesis, the two SST clad fuel elements were moved to another position of the core where they were measured again. As expected no fission product were detected. On the contrary, fission products were detected again in position C5 and C6 where two different fuel elements, previously verified for the absence of leakage, were inserted.
At that point it was clear that the release was caused not by the SST fuel elements in position C5 and C6 but by one or more fuel elements positioned nearby. Since three instrumented fuel elements were positioned close to positions C5 and C6 (in position D7, D8 and B3), the oldest of the three was removed from the core.
The measurements in position C5 and C6 were repeated and no fission products were detected.
The reactor power was raised up to 250 kW and the measurement of the water sampled in all four sector was repeated showing no presence of fission products.
A sample of water was collected from the reactor pool after one hour of operation of the reactor at the power of 250 kW with the primary cooling system switched off and it was measured in a low-background gamma detector. No fission products were revealed in the water.

6. Conclusion

As expected the fission products leakage was due to a micro-fissure of a fuel element that released noble gas only when the fuel element was heated up to a temperature around 90 °C, i.e. at the reactor power of about 100 kW. The fuel element identified as the origin of the release was the oldest SST clad instrumented fuel element present in the core.
The fuel element was removed from its position and stored in a rack of the reactor pool under 4 m of water shield. In this condition the element will not release any fission product any more but it will be necessary to condition it in a proper way after at least a couple of year of cooling down.
The reactor was back in regular operation within two months from the detection of the fission products in the ionic-exchange resins of the primary cooling circuit and no other anomalies were reported. As a routine operation, the reactor pool water is now sampled and measured with a low-background gamma-ray detector every month before the reactor start-up and after half-hour of operation of the reactor at full nominal power.

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THE DALAT NUCLEAR RESEARCH REACTOR OPERATION AND CONVERSION STATUS

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ABSTRACT

This paper presents operation and conversion status of the Dalat Nuclear Research Reactor (DNRR). The DNRR is a pool type research reactor which was reconstructed from the 250 kW TRIGA-MARK II reactor. The core is loaded with WWR-M2 fuel assemblies with 36% enrichment. The reconstructed reactor reached its initial criticality in November 1983 and attained its nominal power of 500 kW in February 1984. The DNRR is operated mainly in continuous runs of 100 hrs, once every 4 weeks, for radioisotope production, neutron activation analyses, training and research purposes. The remaining time between two continuous runs is devoted to maintenance activities and to short runs. Until now four fuel reloading were executed. The reactor control and instrumentation system was upgraded in 1994. And now the reactor control system is being replaced by new one. The study on fuel conversion has been done. Now we are working for realizing fuel conversion of the DNRR.

1. Introduction

The Dalat Nuclear Research Reactor (DNRR) is a pool type research reactor which was reconstructed from the 250 kW TRIGA-MARK II reactor. During reconstruction, some structures of the former reactor such as the reactor aluminium tank, the graphite reflector, the thermal column, the horizontal beam tubes and the radiation concrete shielding were retained [1]. The reactor core, the control and instrumentation system, the primary and secondary cooling systems as well as other associated systems were newly designed and installed. The core is loaded with WWR-M2 fuel assemblies (FA) with 36% enrichment. The reconstructed reactor reached its initial criticality in November 1983 and attained its nominal power of 500 kW in February 1984. The DNRR is operated mainly in continuous runs of 100 hrs, once every 4 weeks, for radioisotope production, neutron activation analyses, training and research purposes. The remaining time between two continuous runs is devoted to maintenance activities and to short runs. The reactor control and instrumentation system was upgraded in 1994. And now the reactor control system is being replaced by new one.

In April 1994, after more than 10 years of operation with 89 fuel assemblies, the first fuel reloading was executed. The 11 new fuel assemblies were added in the core periphery, at previous beryllium element locations. After reloading the working configuration of reactor core consisted of 100 fuel assemblies. The second fuel reloading was executed in March 2002. The 4 new fuel assemblies were also added in the core periphery, at previous beryllium element locations. After reloading the working configuration of reactor core consisted of 104 fuel assemblies [2]. Figure 1 shows core configuration with the 104 fuel assemblies. The third fuel reloading by shuffling of fuel assemblies was executed in June 2004. The shuffling of 16 fuel assemblies with highest burn up in the centre of the core and 16 fuel assemblies with low burn up in the core periphery was done. The working configuration of reactor core kept unchanged of 104 fuel assemblies. The fourth fuel reloading was executed in November 2006. The 2 new fuel assemblies were loaded in the core periphery, at previous locations of wet irradiation channel and dry irradiation channel. After reloading the working configuration of reactor core consisted of 106 fuel assemblies.

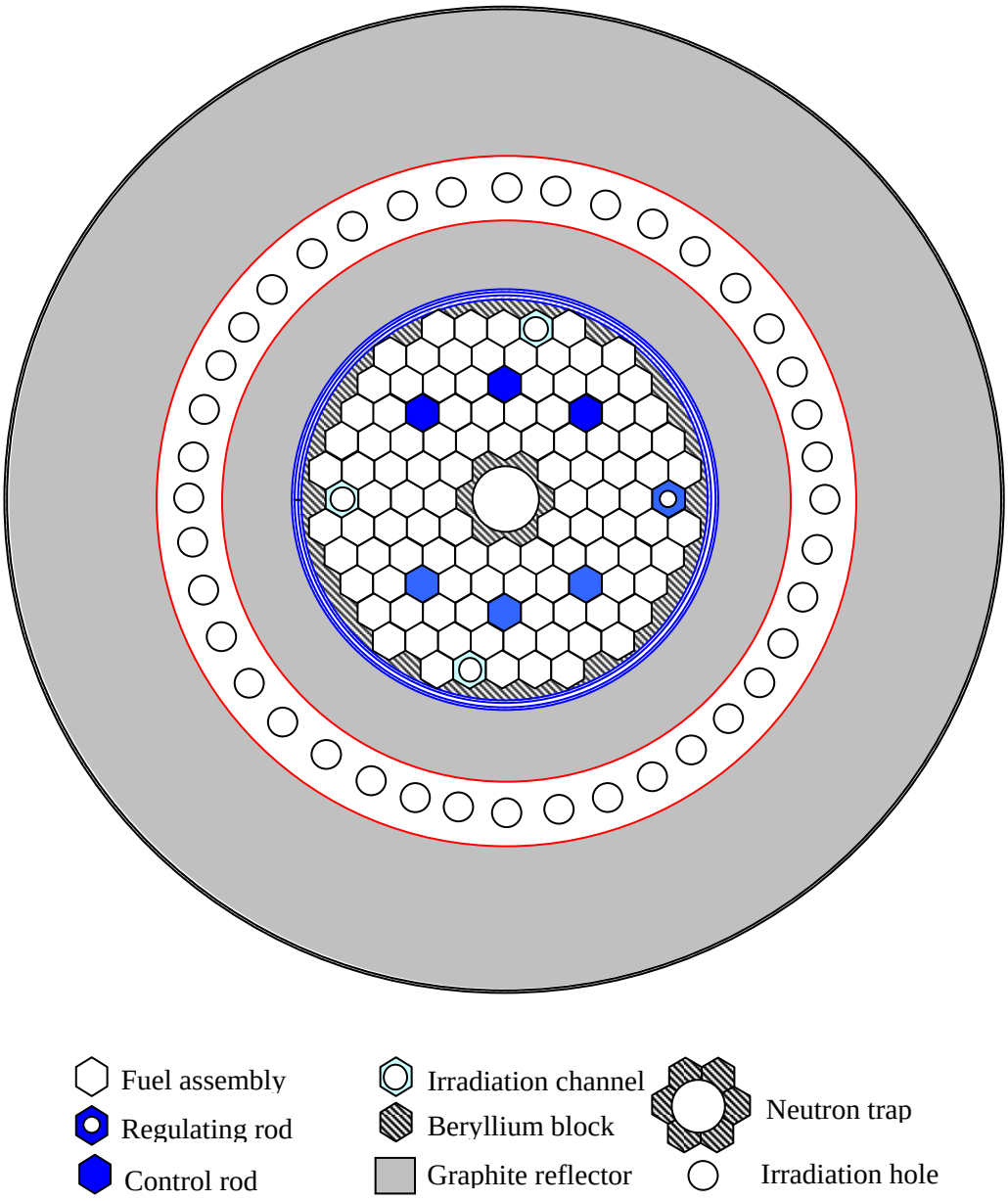

Figure 1. Core configuration with the 104 fuel assemblies

The study on safety analyses for inserting either 36 fresh HEU WWR-M2 fuel assemblies stored at the DNRR or 36 LEU WWR-M2 fuel assemblies with 19.75% enrichment to be procured before additional reactivity is required for continued operation around April 2006 has been done. The results of study showed that operation time of mixed core by inserting 36 LEU FA last much longer than 36 HEU FA. Neutron flux performances at irradiation positions are not significantly changed. The insertion of fresh LEU WWR-M2 fuel assemblies instead of fresh HEU WWR-M2 fuel assemblies will keep the reactor operating as safe as current core. Now we are working for realizing fuel conversion of the DNRR.

2. Reactor operation status

Total operation time at nominal power of the DNRR from March 1984 to December 2006 is 29790 h. The total energy released was 595 MWd. Figure 2 shows record of operation time of the DNRR. Figure 3 shows status of radioactive production at the Dalat Nuclear Research Institute. The number of unexpected scrams in the last 22 years is 256, mainly due to unstable working of the local electrical supply network (70%), due to equipment failures (20%) and human errors (10%).

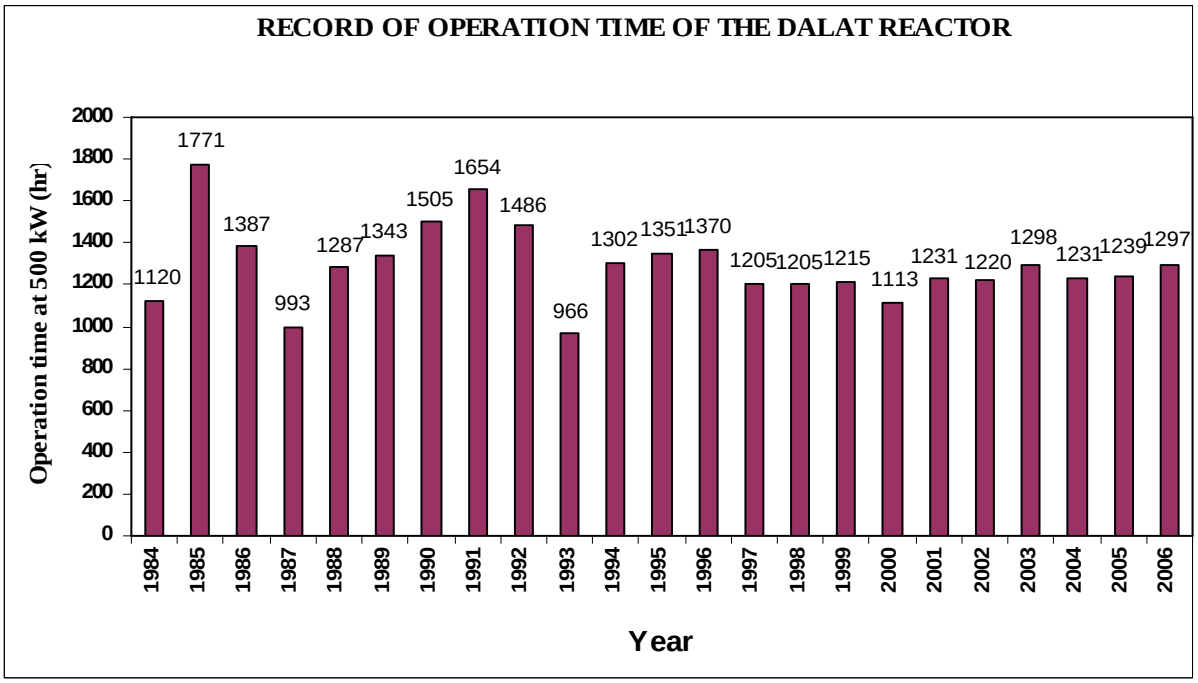

Figure 2. Record of operation time of the Dalat reactor

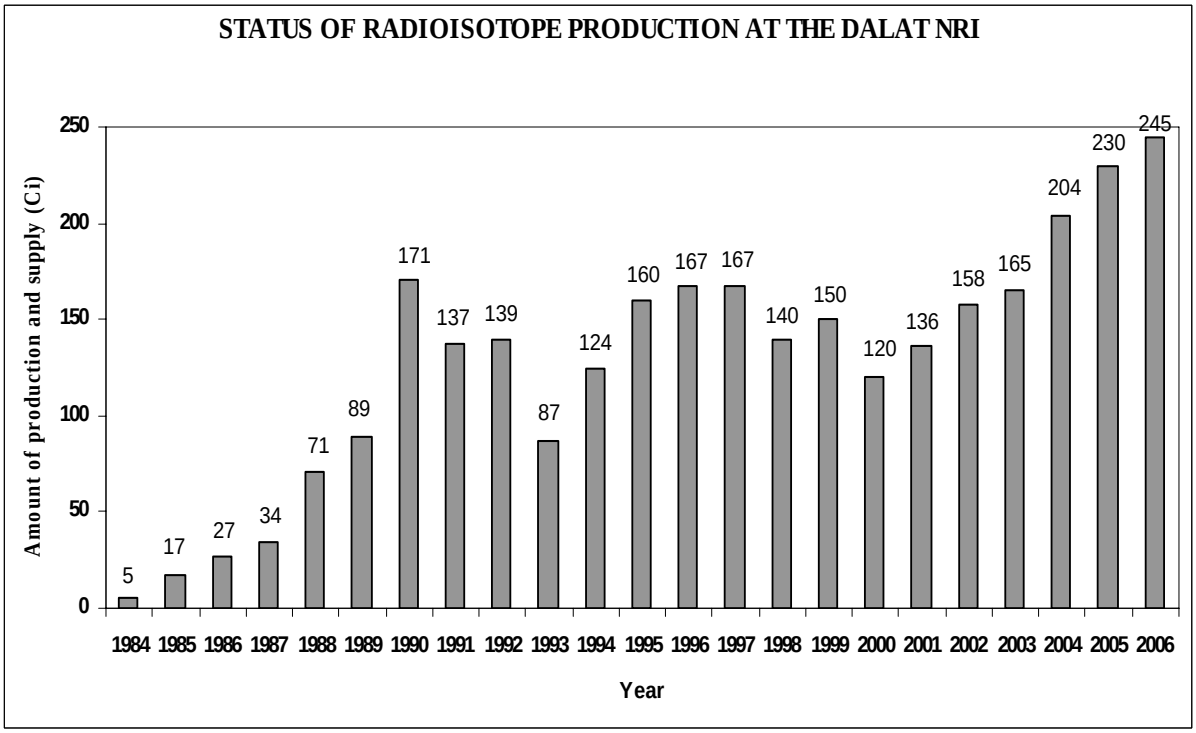

Figure 3. Status of radioactive production at the Dalat Nuclear Research Institute

Reactor control and protection are affected by six control rods composed of boron carbide and an automatic regulating rod composed of stainless steel. The reactor control and instrumentation system was upgraded in 1994. Now we are carrying out project to replace the reactor control system by new one except control rods. This project is supported by International Atomic Energy Agency and Vietnam Government. Equipments are supplied by company SNIIP-SYSTEMATOM, Russia. New control system ensures the safety, control, check and monitoring of the reactor facility by means of the following channels and equipment: channels for monitoring of reactor power and period by thermal neutrons flux density (NFME channels); channel for monitoring of process parameters; channels for

logical processing of signals from NFME channels, from technological and supporting systems and for generation of control signals for protection safety system and for normal operation system; channel for automatic power regulation; channels for reactivity monitoring; channel for monitoring of control rods position; information channels for displaying operative information at control panel; buttons and keys of control panel and equipment for archiving, diagnostic and recording. We started replacement work on 9 December 2006. The work will be fulfilled in March 2007. And then the DNRR will be operated with new control system.

3. Reactor core conversion status

The core of the DNRR utilizes fuel of aluminium-uranium alloy of Soviet-designed standard type WWR-M2, enriched to 36%, clad in aluminium alloy. Each fuel assembly contains about 40.2 g of U-235 distributed on three coaxial fuel tubes (fuel elements), of which the outermost one is hexagonal shaped and the two inner ones are circular. The fuel layer of U-Al alloy with a thickness of 0.7 mm is wrapped between two aluminium alloy cladding layers of 0.9 mm thickness. The study on reactor core conversion has been done in the framework of joint study between RERTR program at Argonne National Laboratory (ANL) and Vietnam Atomic Energy Commission (VAEC) [3] and Ministerial research theme on DNRR core conversion [4]. Each LEU (enrichment of 19.75%) fuel assembly contains an average of 49.7 g 235U with UO2-Al dispersion fuel meat. The fuel layer with a thickness of 0.94 mm is wrapped between two aluminium alloy cladding layers of 0.78 mm thickness. The neutronic calculations indicated that around April 2006 if 36 fresh HEU WWR-M2 fuel assemblies or 36 fresh LEU WWR-M2 fuel assemblies are inserted without fuel shuffling over the next four operating cycles, the core could operate for an additional 10.5 years or 14.1 years, respectively. A comparison of the operating times for each of the four reload cycles is presented in Table 1.

Cycle	HEU or LEU FA Inserted per Cycle	Total HEU or LEU FA Inserted	FPD with HEU Fuel	Cum. Years Oper. Using HEU	FPD with LEU Fuel	Cum. Years Oper. With LEU
1	8	8	143	2.8	183	3.5
2	8	16	265	5.1	344	6.6
3	10	26	413	7.9	546	10.5
4	10	36	547	10.5	733	14.1

Table 1: Summary of operating times for Incremental Insertion of 36 fresh HEU or 36 fresh LEU fuel assemblies beginning in April 2006

Shutdown margins for four reloaded cores range from -4.1% to -4.41% which are much greater than the required value of -1%. At the Neutron Trap, the fast and thermal flux of mixed fuel core (using HEU and LEU fuel) decrease only several thousandths compared to those of HEU core. In all other comparison locations, the fast flux is essentially the same. In these same locations the thermal flux has been reduced 1 to 3.4% as more LEU fuel is reloaded. The neutron flux performance of fast and thermal fluxes in all irradiation positions is compared in Table 2. The total power peaking factors for the mixed fuel cores are slightly smaller (~1%) than those in the corresponding HEU cores. The calculated temperature feedback coefficients and kinetics parameters are not so different between the current core and mixed fuel core. The requirement of thermal-hydraulic safety margin for the mixed fuel core in normal operational condition is satisfied. The Accident analyses for reactivity insertion, maximum positive reactivity insertion and fuel cladding failure were done. The results of analysis for reactor power, fuel cladding temperature and coolant temperature in the investigated cases of positive reactivity insertion have no much the differences between the two cores. The results of analysis for the fuel cladding failure shown that the changing from HEU core to mixed fuel core will not affect significantly on the MHA consequences. Thus, the insertion of fresh LEU WWR-M2 fuel assemblies instead of fresh HEU WWR-M2 fuel assemblies will keep the reactor operating as safe as current core.

	Cycle 1		Cycle 2		Cycle 3		Cycle 4	
	Thermal	**Fast**	**Thermal**	**Fast**	**Thermal**	**Fast**	**Thermal**	**Fast**
Dry Irradiation Channels								
Cell 13-2	0.999	0.995	0.992	0.995	0.974	0.998	0.972	1.002
Cell 7-1	0.994	0.993	0.997	0.995	0.994	0.994	0.966	0.989
Wet Irradiation Channel								
Cell 1-4	1.000	0.994	0.996	0.997	0.983	1.005	0.967	1.006
Neutron Trap	0.997	1.001	0.997	1.000	0.997	0.994	0.995	0.992

Table 2: Neutron flux performance comparisons for four reload cycles: LEU/HEU ratio

Now we are working for contracts between Russia, Vietnam, USA and the International Atomic Energy Agency for Nuclear fuel manufacture and supply for DNRR and Return of Russian-origin non-irradiated highly enriched uranium fuel to the Russian Federation. According to the plan we will received 36 new LEU fuel assemblies in the second half of 2007. Then we will execute fuel reloading by using LEU fuel and create mixed core with 104 fuel assemblies as shown in Figure 1.

4. Conclusions

The DNRR is operated mainly in continuous runs of 100 hrs, once every 4 weeks, for radioisotope production, neutron activation analyses, training and research purposes. Total operation time at nominal power of the DNRR from March 1984 to December 2006 is 29790 h. The total energy released was 595 MWd. Now we are carrying out project to replace the reactor control system by new one except control rods. The replacement will be fulfilled in March 2007. And then the DNRR will be operated with new control system.

The study on reactor core conversion has been done. The results of study showed that operation time of mixed core by inserting 36 LEU FA last much longer than 36 HEU FA. Neutron flux performances at irradiation positions are not significantly changed. The insertion of fresh LEU WWR-M2 fuel assemblies instead of fresh HEU WWR-M2 fuel assemblies will keep the reactor operating as safe as current core. Now we are realizing conversion project for DNRR.

5. References

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RRFM IGORR 2007

LYON, France 11.3. - 15.3. 2007

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