



Transactions

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and
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Research (IGORR)**

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Session I - International Topics and Overview on New Projects and Fuel

JHR PROJECT STATUS	9
IAEA'S SUBPROGRAMME ON RESEARCH REACTORS: TECHNOLOGY AND NON-PROLIFERATION	15
REFURBISHMENT AND PERSPECTIVE FOR ILL	24
THE OPAL REACTOR	33
EXPLORING THE CONCEPT OF A NEW GLOBAL INSTITUTION TO PROMOTE BEST PRACTICES FOR NUCLEAR MATERIALS SECURITY	39
THE RENAISSANCE OF FAST SODIUM REACTORS 2007 ASSESSMENT: SITUATION AND CONTRIBUTIONS FROM PHENIX EXPERIMENTAL REACTOR	44
DEVELOPING RESEARCH REACTOR COALITIONS	53
REFURBISHMENT AND ACTIVITIES AT TAJOURA REACTOR	59
IRRADIATION FACILITIES AT THE ADVANCED TEST REACTOR	64
CURRENT AND PROSPECTIVE FUEL TEST PROGRAMMES IN THE MIR REACTOR	69

Session II - New Projects and Upgrades

STATUS OF THE HIGH FLUX ISOTOPE REACTOR AND THE REACTOR SCIENTIFIC UPGRADES PROGRAM	76
NEW MODERATOR CHAMBER OF THE FRG-1 COLD NEUTRON SOURCE FOR THE INCREASE OF COLD NEUTRON FLUX	81
STATUS OF MODERNIZATION AND REFURBISHMENT (M&R) ACTIVITIES OF THE IRT – RESEARCH REACTOR – SOFIA/INSTRUMENTATION AND CONTROL SYSTEM	86
CONCEPTUAL DESIGN OF A PRESSURIZED WATER LOOP FOR THE IRRADIATION OF 6 FUEL RODS IN THE JULES HOROWITZ REACTOR	91
DEVELOPMENT OF HIGH TEMPERATURE CAPSULE FOR RIA-SIMULATING EXPERIMENT WITH HIGH-BURNUP FUEL	97
OPERATIONAL SAFETY; REGULATORY REQUIREMENTS; ADVANCES AND EXPECTED OUTCOMES IN THE GHARR-1 COMPUTERISED CONTROL SYSTEM UPGRADE PROJECT	102



Session III - Fuel Development

CHARACTERISATION AND TESTING OF MONOLITHIC RETR FUEL PLATES	111
2007 REPORT ON DEVELOPMENT PROGRESS ON LEU FUELS AND TARGETS IN ARGENTINA	118
FABRICATION OF MONOLITHIC UAl_2 ROD BY THE POWDER METALLURGY METHOD	123
PHYSICO-CHEMICAL ASPECTS OF MODIFIED UMO/AL INTERACTION	129
THE REACTION ZONE IN THE SYSTEM U-Mo/Al 6061 RELATED WITH THE DECOMPOSITION OF γ U-MO	135
NEUTRON POWDER DIFFRACTION OF U(MO) FUEL IRRADIATED TO 60 PERCENT BURNUP	140
IMPROVED IRRADIATION BEHAVIOUR OF URANIUM-I MOLYBDENUM/ALUMINUM DISPERSION FUEL	145
POST IRRADIATION EXAMINATION OF UMO MONOLITHIC MINIPLATES IRRADIATED IN RERTR-6 & RETR-7	151
COMPREHENSIVE OVERVIEW ON IRIS PROGRAM: IRRADIATION TESTS AND PIE ON HIGH DENSITY UMo/Al DISPERSION	157
NOVEL TRENDS IN FUEL AND MATRIX ALLOYING TO REDUCE INTERACTION	164
POST-IRRADIATION EXAMINATION OF ALFENI CLADDED U_3Si_2 FUEL PLATES IRRADIATED UNDER SEVERE CONDITIONS	170
PROGRAM OF THE TRIAL OF LEAD TEST ASSEMBLIES IN THE WWR-K REACTOR	175

Session IV - Optimisation and Research Reactor Utilisation

VIBRATION MONITORING AND DIAGNOSTIC OF THE IAER1 NUCLEAR RESEARCH REACTOR	181
NEW SILICON IRRADIATION RIG DESIGN FOR OPAL REACTOR	186
MEASUREMENT OF VOID FRACTION IN HYDROGEN MODERATOR USED FOR MODERATOR CELL OF HANARO COLD NEUTRON SOURCE	191
OVERVIEW OF THE IAEA ACTIVITIES ON SAFETY OF RESEARCH REACTORS	
OPTIMIZATION OF THE POOLSIDE FACILITY FOR NEUTRON DOPING OF SILICON IN HIGH FLUX MATERIALS TESTING REACTOR BR2	196



THE MATERIALS SURVEILLANCE PROGRAM FOR THE OPAL RESEARCH REACTOR	201

SESSION V - INNOVATIVE METHODS IN RESEARCH REACTORS PHYSICS

MCNPX 2.6.C VS. MCNPX & ORIGEN-S: STATE OF THE ART FOR REACTOR CORE MANAGEMENT	208
SIMULATION OF IRRADIATION OF A BUNDLE OF MOX FUEL RODS IN THE OMICO EXPERIMENT IN BR2	213
DETERMINING MTR RIA LIMITS USING EXPERIMENTAL DATA	218
SAFETY ANALYSIS OF RESEARCH REACTORS WITH BEST ESTIMATE COMPUTATIONAL TOOLS	225
KINETIC PARAMETERS CALCULATION AND MEASUREMENTS DURING THE OPAL COMMISSIONING	230
DETERMINATION OF SAFARI-1 NEUTRON FLUXES BY MCNPX MODELLING OF FOIL EXPERIMENTS	235
SOPHISTICATED MCNP CALCULATION OF THE FLUX MAP OF FRJ-2 USING A FULLY NODALIZED MODEL	239

Session VI - Safety, Operation and Research Reactor Conversion

CONVERSION OF RESEARCH AND TEST REACTORS: STATUS AND CURRENT PLANS	245
SAFARI-1: ADJUSTING PRIORITIES DURING THE LEU CONVERSION PROGRAM	250
RESULTS OF 14MW RESEARCH REACTOR CORE CONVERSION MEASURED AT LOW POWER	255
SOME REQUIREMENTS FOR THE CONVERSIÓN OF THE SYRIAN MNSR CORE	262
FUEL MANUFACTURE IN SOUTH AFRICA: THE ROAD TO CONVERSION, A PARTNERSHIP NESCA AND AREVA CERCA	266
NEUTRONIC CALCULATIONS FOR CONVERSION OF ONE-ELEMENT CORES FROM HEU TO LEU USING MONOLITHIC UMO FUEL	274
ANALYSIS OF AN LEU FUEL WITH SPATIALLY DEPENDENT THICKNESS IN TWO DIMENSIONS	279
NEUTRONIC ANALYSIS FOR CONVERSION OF THE GHANA RESEARCH REACTOR-1 FACILITY USING MONTE CARLO METHODS AND UO2 LEU FUEL	284



JRR-3 MAINTENANCE PROGRAM UTILIZING ACCUMULATED OPERATIONAL DATA	295
RADIATION PROTECTION DESIGN OF THE FRMII	299
IDENTIFICATION OF A LEAKING TRIGA FUEL ELEMENT AT THE NUCLEAR RESEARCH REACTOR FACILITY OF THE UNIVERSITY OF PAVIA	304
THE DALAT NUCLEAR RESEARCH REACTOR OPERATION AND CONVERSION STUDY STATUS	309

SESSION VII - FUEL BACK-END MANAGEMENT

THE JASON REACTOR: FROM CORE REMOVAL TO FUEL REPROCESSING	315
RESEARCH AND TEST REACTOR FUEL TREATMENT AT THE AREVA NC LA HAGUE SITE	323
REPROCESSING OF RESEARCH REACTOR SPENT NUCLEAR FUEL AT THE PA 'MAYAK'	331
THE U. S. DEPARTMENT OF ENERGY / IDAHO NATIONAL LABORATORY'S RESEARCH REACTOR SPENT NUCLEAR FUEL ACCEPTANCE PROGRAM	336
INVESTIGATIONS TO THE BEHAVIOUR OF RESEARCH REACTOR FUEL ELEMENTS IN REPOSITORY RELEVANT AQUATIC PHASES	345
CORROSION OF SPENT RESEARCH REACTOR FUEL: THE ROLE OF SETTLED SOLIDS	350
UNITED STATES FOREIGN RESEARCH REACTOR (FRR) SPENT NUCLEAR FUEL (SNF) ACCEPTANCE PROGRAMME: 2007 UPDATE	355
PREPARATION AND PERFORMANCE OF THE LARGEST SHIPMENT OF IRRADIATED HEU FUEL ELEMENTS FROM SYDNEY TO THE UNITED STATES	360
THE EXPERIENCE OF SHIPPING SPENT NUCLEAR FUEL FROM UZBEKISTAN TO THE RUSSIAN FEDERATION	366
POSSIBILITY OF A PARTIAL HEU-LEU TRIGA FUEL SHIPMENT	376

Poster Session

THE NEW AREA RADIATION MONITORING SYSTEM OF THE TRIGA NUCLEAR RESEARCH REACTOR FACILITY OF THE UNIVERSITY OF PAVIA	382
ASSESSMENT OF UTILIZATION OF PIN-TYPE FUEL ELEMENTS WITH CERMET FUEL AND LOWER THAN 20% ²³⁵ U ENRICHED URANIUM IN RESEARCH REACTORS OF UGRADED POWER	387
SOURCE OF RADIONUCLIDES IN PRIMARY CIRCUIT WATER OF LVR-15 REACTOR	391



DEVELOPMENT OF INTEGRATED MANAGEMENT SYSTEM FOR THE RESEARCH REACTOR IN SOFIA	395
HOMOGENEOUS SOLUTION REACTOR ANALYSIS FOR ⁹⁹ MO PRODUCTION	399
UPGRADING OF JRR-3/JRR-4 NEUTRON BEAM UTILITIES – FOR COLD NEUTRON BEAM AND BNCT	405
U (AL, SI) ₃ STABILIZATION BY ZR ADDITION	409
A STUDY ON POSSIBILITY OF USE OF LEU MR-6 TYPE FUEL FOR ADS DESIGN	414
THERMAL CONDUCTIVITY OF HEAVY-ION-BOMBARDED U-MO/AL DISPERSION FUEL	419
CHARACTERIZATION OF MONOLITHIC FUEL FOIL PROPERTIES AND BOND STRENGTH	426
SAFETY ANALYSIS OF A 1-MW POOL-TYPE RESEARCH REACTOR	431
OSCAR-3 MCNP INTERFACE (OSMINT5) VERIFICATION AND VALIDATION	445
THE CONVERSION OF TAJOURA CRITICAL ASSEMBLY FROM HEU TO LEU FUEL	451
EFFECTS OF TI IN THE UMO/AL SYSTEM: PRELIMINARY RESULTS	457
FULL CONVERSION OF MATERIALS AND NUCLEAR FUEL - RESEARCH&TEST - TRIGA SSR 14 MW	463
RESULTS OF POST-IRRADIATION EXAMINATION OF THE (U-MO)–ALUMINIUM MATRIX INTERACTION RATE	468
PLACA/DPLACA SIMULATION OF MONOLITHIC/DISPERSE UMO PLATES	473
STRUCTURE STUDIES OF DISPERSED U-MO FUEL AFTER IRRADIATION AND ISOCHRONOUS ANNEALING WITHIN A TEMPERATURE RANGE OF 150-580 C BY THE NEUTRON DIFFRACTION METHOD	478
REMOVAL OF SPENT NUCLEAR FUEL FROM KURCHATOV INSTITUTE RESEARCH REACTORS FOR REPROCESSING: PROBLEMS AND PLANS	483



Session I

**International Topics and Overview on New
Projects and Fuel Developments**

JHR PROJECT STATUS

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ABSTRACT

In Europe, nuclear electricity plays an important role and will stay for the long term a part of the energy mix since it contributes to the energy security of supply and to the reduction of greenhouse gas production.

The Jules Horowitz Reactor (JHR) is a new high performance material testing reactor under construction in Cadarache (France); start of operation is foreseen in 2014.

The JHR is a strategic infrastructure in the European Research Area, open to the international collaboration, to support safety, lifetime management and operation optimisation of current nuclear power plants, development of new types of reactors with improved resources and fuel cycle management, medical applications, material development for fusion reactor...

The design has been completed in 2005 and the preliminary safety report has been issued and is under assessment by the safety body.

A JHR Consortium gathering several industries and research institutes has been settled to finance the construction in order to have a secured and guaranteed access to the JHR experimental capacity.

1. Background

Continuous improvements in the nuclear fission industry require, on the long term, testing the behaviour of the materials and fuels used in nuclear power plants:

1 - Extending the lifespan of Generation II reactors and demonstrating the lifespan of such reactors as EPR (Generation III) is essential for countries having nuclear power plants, whatever their nuclear policy may be in the long-run.

2 - Fuel technology in nuclear power plants is continuously upgraded to achieve better performances and to optimise the fuel cycle, still keeping the best level of safety. As a key part of these performances improvement, it is necessary to experimentally explore the full range of fuel behaviour to determine fuel stability limits and determine safety margins. Fuel is and will stay a strategic topic in the long term (Generations II, III, IV) for nuclear industry.

3 - Besides the industrial goals set for the short and medium term, nuclear energy is also the subject of public policies set for the medium and long term to explore technical solutions towards sustainable development (Generation IV). Innovative materials and fuels are required to resist to high temperatures or fast neutron flux.

Up-to-date and sustainable research infrastructures, such as the Jules Horowitz Reactor (JHR), are necessary to support power reactors developments such as driven in Generation IV forum and to meet the continuous needs from light water reactors that will be in operation all along the XXIst century.

2. JHR technical scope

2.1. For current and coming power reactor technologies

Public and industrial needs for experimental irradiations in support of existing nuclear plants are continuous and well known. Taking benefit of the large available experience from existing MTRs, the JHR has been optimised to perform high quality experiments for existing nuclear reactors.

This encompasses for illustration

- Several capsules and loops, for PWR and BWR fuel or corrosion studies, settled on displacement systems in the reflector. This will provides high quality ramp systems to study transients or power regulated experiments.
- High neutron flux experiments for LWR experiments in the core. As a major stake, dedicated experimental devices are developed to perform low thermal gradients irradiation despite the high gamma heating.
- Improved capacity to perform safety experiments (dedicated alpha cell, displacement systems to manage safe positions after the test).
- On line fission products and helium measurement (at low as well as high level, in gas as well as in water conditions) during the fuel irradiation to optimise fuel microstructures (grain size, homogeneity, additives ...) for the economy (high burn up fuel), the safety (fission products retention for severe accident impact) and the minor actinide management capacity.

2.2. For future reactors

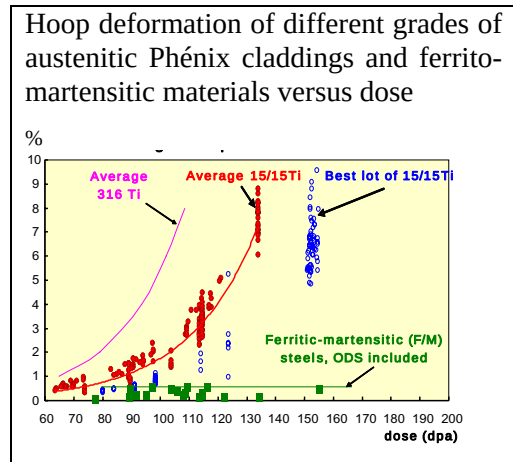
With high flux performances and the flexibility required for in depth experimental investigation, JHR is also optimised to meet future reactors needs.

Innovative materials and fuels which resist to high temperatures and/or fast neutron flux in different environments are necessary: structural materials such as graphite (VHTR and MSR), austenitic and ferritic steels (VHTR, SFR, GFR, LFR), Ni based alloys (SCWR), ceramics (GFR)... Experimental irradiations have to be carried out in order to study microstructural and dimensional evolution, but also the behaviour under stress. New fuels for the different Gen IV systems need also to be qualified in research reactors

The foreseen innovative structural materials are common to fission and fusion application.

For instance, the swelling of austenitic steel claddings limit the burn-up of SFR fuels. More important, implementing low swelling materials may allow reducing coolant channels thickness, which is one of the paths to cope with coolant void reactivity coefficient. Advanced austenitic steels of the type 15Cr/20-25Ni may allow reaching doses of the order of 150 dpa, limited by swelling and the associated loss of mechanical strength. Switching to ferritic/martensitic steels may allow reaching 200dpa. Oxide Dispersion Strengthened (ODS) ferritic steels with ~14% Cr and more could be utilised up to temperature of the order of 900°C, thanks to their improved creep resistance resulting from a dispersion of nanoscale precipitates of Yttrium oxide. Although irradiation data are scarce, the bcc crystalline structure should result in an excellent resistance to swelling.

Going to even higher temperature will require switching from metals to ceramics. Interesting candidate materials for GFR and fusion are Silicon Carbide composites. The SiCf/SiC composites with cubic stoichiometric fibres and matrix are attractive for high temperature application: up to 1000-1200°C in nominal conditions and up to ~1600°C in incidental or accidental conditions. The main issues are (i) the



long term stability of dimension and physical properties, (ii) the irradiation detrimental effect on the inter-phase and its capability of deviating cracks and thus providing reasonable fracture toughness, (iii) the required higher creep strength of the fibre to bear the thermal-mechanical loading in long term service under high temperature and neutron flux, (iv) the type of mechanical damage under irradiation and creep. The behaviour of these materials under coupled irradiation and mechanical stress is a major challenge.

The development and implementation of these advanced metallic alloys and ceramic composites will need breakthroughs in material science, from process development (material fabrication, assembling...) to performance assessment (behaviour under coupled temperature, mechanical stress and irradiation). These challenges require comprehensive tests and in-depth investigations of structural materials and fuel components to be addressed by a high performance experimental irradiation infrastructure such as Jules Horowitz Reactor (JHR) and ultimately in an experimental fast neutron reactors.

3. Situation of Material Test Reactors in Europe

European Material Test Reactors (MTRs) have provided essential support for nuclear power programs over the last 40 years. Associated with hot laboratories for the post irradiation examinations, they are structuring research facilities for the European Research Area in the fission domain.

However, in Europe, MTRs will be more than 50 years old in the next decade and will face increasing probability of shut-down due to their obsolescence. The reactor R2 has been shut down in 2005 and OSIRIS will be shut down at the beginning of the next decade. Renewing the experimental irradiation capability meet not only technical needs but important stakes such as maintaining a high scientific expertise level by training of new generations of searchers, engineers and operators.

Countries	Reactor	Operation	Power (MWth)
Czech Re.	LVR15	1957	10
Norway	Halden	1960	19
Sweden	R2	1960-2005	50
Netherland	HFR	1961	45
Belgium	BR2	1961	100
France	OSIRIS	1966	70

4. The Jules Horowitz material test Reactor

To cope with this context, the Jules Horowitz Reactor Project (JHR) has been launched as a new MTR in Europe to be implemented in Cadarache (south of France); start of operation is foreseen in 2014 [1].

The JHR start of operation is foreseen in 2014. The definition studies are completed (2003-2005). The present development studies (2006-2007) is dedicated to the supply of components, to the qualification of key components, to a major licensing step (the preliminary safety analysis report was submitted to the Safety Body in February 2006), to the preparation of the site in Cadarache. The public consultation and public enquiry were completed respectively in spring 2005 and February 2007 without difficulty.

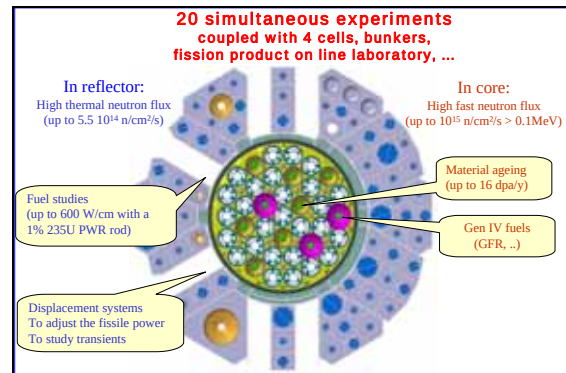
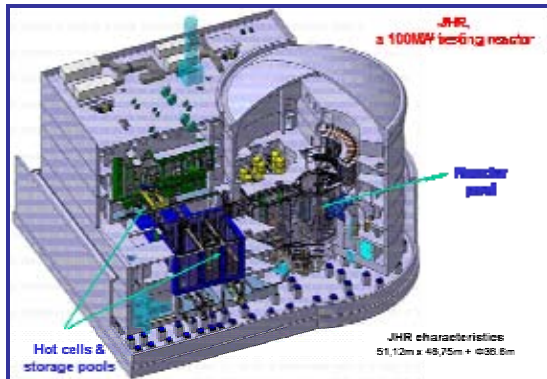
The JHR is a 100MW tank pool reactor.

The core area is inserted in a small pressured tank (section in the order of 740 mm diameter) with forced coolant convection (low pressure primary circuit at 1.5 Mpa, low temperature cooling, core inlet temperature in the order of 25°C). Reactor primary circuit is completely located inside the reactor building.

The reactor building is divided into two zones. The first zone contains the reactor hall and the reactor primary cooling system. The second zone hosts the experimental areas in connection with in pile irradiation (eg., typically 10 loops support systems, gamma scanning, fission product analysis laboratory etc.). The Fission Product Laboratory will be settled in this area to be connected to several fuel loops ether for low activity gas measurements (HTR, ...) or high activity gas measurements (LWR rod

plenum, ...) or water measurements (LWR coolant, ...) with gaseous chromatography and mass spectrometry. Bunkers and laboratories in the experimental area will use 300m² per level on 3 levels.

Hot cells, laboratories and storage pools are located in the nuclear auxiliaries building. The experimental process will make use of two hot cells to manage experimental devices before and after the irradiation. Safety experiments are an important objective for JHR and require an “alpha cell” for an effective management of devices with failed experimental fuel. A fourth hot cell will be dedicated to the transit of radioisotope for medical application and to the dry evacuation of used fuel.



The core (600 mm fuel active height) is cooled and moderated with water. The core area is surrounded by a reflector (water and beryllium elements) which optimizes the core cycle length and provides intense thermal fluxes in this area.

The fuel element is of circular shape. The JHR is designed to be operated with a reprocessable high density low enriched fuel (5U enrichment lower than 20%, density 8 g/cm³). CEA is deeply committed in the development of the UMo fuel within an international collaboration. In case the UMo is not available at the industrial level, the JHR may be started for a limited period with an U3Si2 fuel at typically 27% U5.

Irradiation devices can be placed either in the core area (in a fuel element central hole or in place of a fuel element) or in the reflector area. In core experiment will address typically material experiments with high fast flux capability up to $5 \cdot 10^{14}$ n/cm²/s (resp. 10^{15} n/cm²/s) perturbed fast neutron flux with energy larger than 1MeV (resp. 0,1MeV), that is up to 16 dpa/year with 260 full power operation days per year.

In reflector experiments will address typically fuel experiment with perturbed thermal flux up to $5 \cdot 10^{14}$ n/cm²/s (perturbed thermal neutron flux). Experiments can be implemented in static locations, but also on displacement systems as an effective way to investigate transient regimes occurring in incidental or accidental situations.

5. European and international collaboration

5.1. The JHR Consortium

A JHR Consortium has been set up to finance the JHR construction and to provide to funding Members a secured and guaranteed access rights to the JHR experimental capability.

In early 2007, this Consortium gathers industries and research institutes from several European Member States such as France (CEA, EDF, AREVA), Belgium (SCK), Czech Republic (NRI), Finland (VTT), Spain (CIEMAT as a representative of a pool of industries and public bodies).

JHR is a mature project of European interest and is identified as a research infrastructure of pan-European interest by the European Strategic Forum for Research Infrastructure (ESFRI¹). Following the ESFRI

¹ ESFRI established a European roadmap for the construction of the next generation of large-scale Research Infrastructures in close collaboration with the European Commission and based on an international peer-review.

process and through successive steps in the 7th FP and 8th FP, the European Commission, represented by the Joint Research Centre, will become a full Member of the JHR Consortium in order to have access to JHR experimental capacity for implementing the European Community policy.

Discussions are ongoing with other European and non-European countries to enlarge the JHR Consortium.

The JHR funding process was driven by two principles: i) a balance contribution between private and public funding, a strong commitment from the hosting country with the participation of the European and international fission community. This funding scheme appeared as an effective way to renew research infrastructures managed as user-facilities for the benefit of a broad community.

The JHR Consortium Agreement binds Members contributing to the financing of the JHR construction. CEA is the owner and nuclear operator of the JHR.

Members contributing to the financing of JHR construction will have guaranteed and secured access rights to experimental locations in the reactor in order to perform their Proprietary Experimental Programs. In parallel, a Joint Program will be opened to international collaboration in order to address issues of common interest.

Operation costs are paid only for utilised rights; access rights can be utilised partly or in totality each year. Non utilised access rights can be cumulated from one year to the following.

Non-Member will have access to the JHR facility, under decision of the JHR Consortium Board, and within conditions defined by the strategic and commercial policy of the JHR Consortium.

5.2. *Experimental devices*

The development of JHR experimental devices offers a unique opportunity to develop a new generation of experimental devices meeting up-to-date scientific and technological state of art as well as anticipated users' needs. Development of experimental devices and related programmes requires international collaborations to benefit from the available large experience and to increase the critical mass of cross-disciplinary competences.

Several scientific topics have been assessed in the European 6th framework program JHR coordination action (2004-2005) [2]. In parallel, a new impetus has been put on instrumentation technologies by the creation of a joint lab between CEA and SCK•CEN [3]. Several devices have been investigated within the JHR project to optimise the interfaces with the JHR facility [4, 5].

A new FP6 project (MTR+I3, "MTR plus" integrated infrastructure initiative) has been launch with 18 European partners for the period 2006-2009 with the purpose of building up the European material testing reactor community, of supporting state of the art design of innovative irradiation devices:

Mechanical Testing Device	Mechanical testing devices under mechanical loads
	Corrosion under irradiation
Fuel Testing devices	Neutron screen development for fuel and transmutation studies
	Transmutation
	Power transient systems and neutron screen development for LWRs
	Water Chemistry
	Fission Product measurement
Non LWR Loop design	Gas loop
	Heavy liquid metal loop
	Supercritical water loop
	Miniaturised components
Safety tests instrumentation	

6. Conclusion

The JHR will provide for a large part of the century the experimental irradiation capability in Europe for the benefit of international industries and public stakeholders.

The year 2007 is a major milestone for the JHR project with the launch of the JHR Consortium gathering a first set of funding partners.

The establishment of the JHR Consortium together with the networking of relevant research laboratories is a most important step in the building of the coming generation of R&D competences and infrastructure. This is required to cope with R&D needs to support present and future power reactors.

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IAEA's SUBPROGRAMME ON RESEARCH REACTORS: TECHNOLOGY AND NON-PROLIFERATION

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ABSTRACT

For nuclear research and technology development to continue to advance, research reactors (RRs) must be safely and reliably operated, adequately utilized, refurbished when necessary, provided with adequate proliferation-resistant fuel cycle services and safely decommissioned at the end of life.

The IAEA has established its competence in the area of RRs with a long history of assistance to Member States in improving their utilization, by taking the lead in the development of norms and codes of good practice for all aspects of the nuclear fuel cycle and in the planning and implementation of decommissioning. The IAEA Subprogramme on RRs is formulated to cover a broad range of RR issues and to promote the continued development of scientific research and technological development using RRs. Member States look to the IAEA for coordination of the worldwide effort in this area and for help in solving specific problems.

In this paper a description of the ongoing and planned activities under the IAEA's Subprogramme on RRs for the years 2007-2009 is presented. Special emphasis is put on new international collaborative undertakings, like the new IAEA's Technical Working Group on RRs.

1. Introduction

The IAEA coordinates and implements an array of activities that together provide broad support for RRs. As with other aspects of nuclear technology, RR activities within the IAEA are spread through diverse groups in different Departments. To ensure a common approach a Cross-Cutting Coordination Group on Research Reactors (CCCGRR) has been established, with representatives from all departments actively supporting RR activities.

Utilization and application activities are generally lead from within the Department of Nuclear Applications (NA). With respect to RRs, NA is primarily carrying out IAEA activities to assist and advise Member States in assessing their needs for research and development in the nuclear sciences, as well in supporting their activities in specific fields.

Safety and Security aspects of RRs operation and decommissioning are handled by the Department of Nuclear Safety and Security (NS).

The technological, fuel cycle and operational aspects of RR management are supported by the Department of Nuclear Energy (NE). NE is primarily working to support RR organizations in their pursuit of often diverse strategic objectives within the context of modern RR operational constraints. Today RR operating organizations must overcome challenges such as the ongoing management of

ageing facilities, pressures for increase vigilance with respect to non proliferation, and shrinking resources (financial as well as human) while fulfilling an expanding role in support of nuclear technology development within an evolving “nuclear renaissance”.

In addition, the Department of Nuclear Safeguards is responsible for the control of the fissile material for RR and the Department of Technical Cooperation (TC) supports RR activities for the principal benefit of RRs in developing countries. TC is subsequently supported by NA, NS, and NE who assist in the development and implementation of relevant TC projects within their specific fields of expertise.

The Subprogramme on RRs is under IAEA’s Programme D on Nuclear Science. Implementation of the IAEA Subprogramme on RRs (IAEA code D.2) is shared between NE and NA while separate subprogrammes, managed by NS, deal with RR safety and security. In this paper, only the activities managed by NE and NA under the subprogramme on RRs are presented, including a complete description of the ongoing and planned activities for the years 2007-2009. Special emphasis is put on new international collaborative undertakings, like the IAEA’s Technical Working Group on RRs. The IAEA organization chart is presented in Fig. 1, the Subprogramme on RRs is implemented by the RRs Unit in the Division of Nuclear Fuel Cycle and Waste Technology and the Physics Section in the Division of Physical and Chemical Sciences.

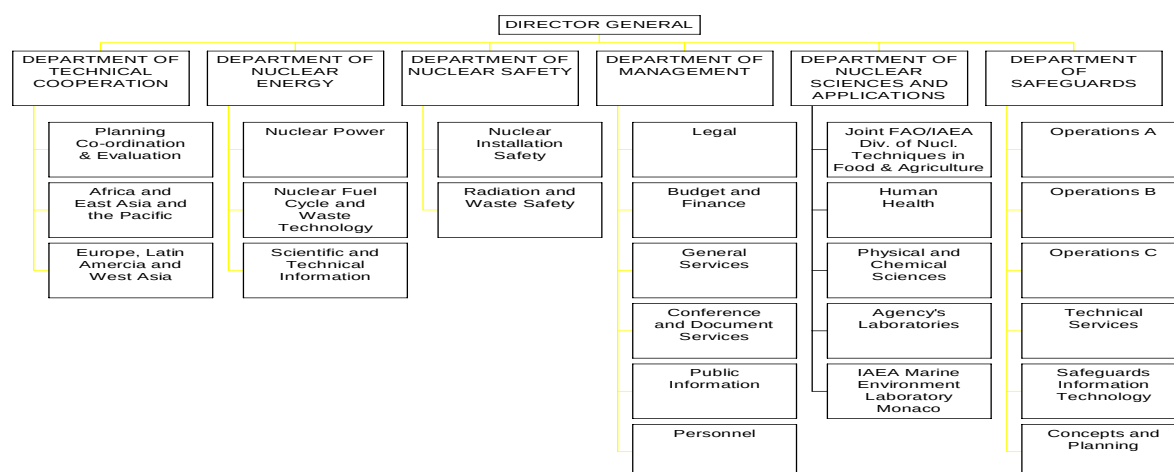


Fig. 1. IAEA organization chart

2. Subprogramme on RRs

For nuclear research and technology development to continue to advance, RRs must be safely and reliably operated, adequately utilized, refurbished when necessary, provided with adequate proliferation resistant fuel cycle services and safely decommissioned at the end of life. Moreover, since about 60% of the operating RRs in the world are over 30 years old, ageing core materials and the technology of ageing management are priority issues in the majority of Member States with aged RRs.

The IAEA has established its competence in the area of RRs with a long history of assistance to Member States in improving their utilization, by taking the lead in the development of norms and codes of good practice for all aspects of the nuclear fuel cycle and in the planning and implementation of decommissioning. This Subprogramme is formulated to cover a broad range of RR issues and to promote the continued development of scientific research and technological development using RRs. Member States look to the IAEA for coordination of the worldwide effort in this area and for help in solving specific problems.

From the traditional support of fundamental research and training, the focus of the Subprogramme has recently moved to helping facilities with strategic planning to increase use in more sustainable areas

such as isotope production and materials modification, in refurbishment and replacement of ageing equipment, in the management of increasing spent fuel inventories and in planning decommissioning. The Subprogramme supports regional and interregional thematic collaborations, networking and centres of excellence for enhanced utilization of RRs.

To contribute to non-proliferation efforts worldwide, support of RERTR and the programmes of returning of RR fuel to the country of origin has been strengthened. To address RR support needed for the evolutionary and innovative nuclear power reactors and fuel cycles, the subprogramme promotes international collaboration to assess projected needs, with a long term time horizon, for RRs on a global and regional basis.

Funding reductions and limited succession planning have strained available resources of a number of RRs, pressurising many facilities to pursue commercial activities to remain in operation. It is in this context that modern RRs are to be used to conduct advanced research in support of innovative nuclear development (in most cases to very aggressive schedules) and training. To support the scientific, educational and commercial demands being placed in present times on RRs, a new project addressing RR Operation, Maintenance, Availability and Reliability has been initiated in 2007.

The main objectives of the RRs Subprogramme are:

- To increase the capabilities of interested Member States to safely and reliably carry out scientific research and technology development at RRs, conduct ageing management, decommissioning, refurbishment and modernization; and
- To enhance the potential of interested Member States to plan new facilities when needed, to cope with RR fuel cycle issues and reduce proliferation risks by conversion from Highly Enriched Uranium (HEU) to Low Enriched Uranium (LEU) of RRs cores and targets used for radioisotope production, and to repatriate fuel to the country of origin.

3. Projects under the Subprogramme on RRs

Organization of the Subprogramme on RRs in projects is shown in Fig. 2.

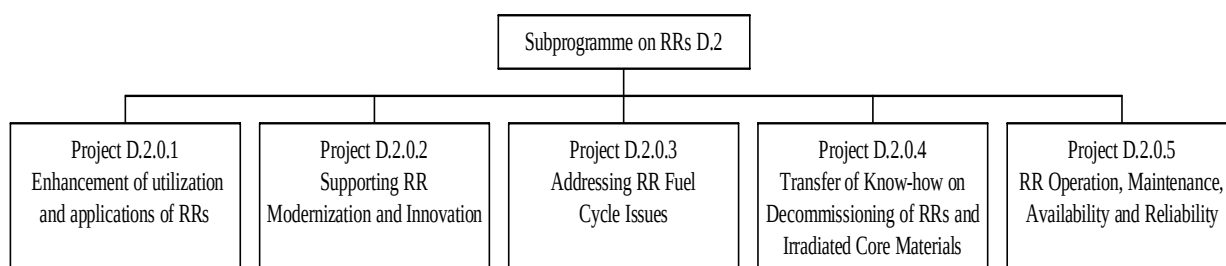


Fig. 2 Projects under the Subprogramme on RRs

A brief description of each one of the projects is given in the following paragraphs.

3.1. Project D.2.0.1: Enhancement of utilization and applications of RRs

RRs have played and continue to play a key role in the development of the peaceful uses of atomic energy. Their contribution to the education and training of scientists and engineers for the whole nuclear community is well documented. In addition they have played an important role in development of science and technology, in the production of isotopes for medicine and industry, in non-destructive testing of materials, in analytical studies, in the modification of materials, in research in various areas of science and in support of nuclear power programmes.

Existing RRs, especially in developing countries, should be supported on an individual level for example in radioisotope production, beam line applications, and analytical services as well as in regional or collaborative efforts in education and training. The sharing of resources will increase the utilization on the one hand and on the other hand pave the way for the decommissioning of under-utilized ageing reactors, without depleting knowledge base and human resources.

We give here an overview of this project formulated to cover the broad range of possible applications and to promote the continued development of scientific research and technological development using RRs.

The main objectives of this project are:

- To enhance RR utilization in Member States for many practical applications, such as isotope production, neutron radiography, neutron beam research and material characterization and testing consistent with RR features; and
- To increase cooperation between different RR centres.

Some of the activities proposed to be carried out under this project are:

- Develop a RR assessment methodology on strategic networking;
- Update RR Database (RRDB). Incorporate user need modification/changes;
- Organize a Technical Meeting (TM) on RR application for materials under high neutron fluence and particle flux in energy sector;
- Coordinate a Coordinated Research Project (CRP) on “Development, characterization and testing of materials using neutrons and complementary techniques” and “Development and application of the techniques of residual stress measurements in materials”;
- Organize a technical meeting on strategic planning and regional networking for sustainability;
- Provide technical support for the IAEA-TC projects involving utilization and applications of RRs;
- Support and participate in meetings pertinent to RRs and neutron based techniques;
- Prepare report on data acquisition and analysis for neutron beam line experiments; and
- Prepare a report on specific application of RRs.

In addition, publication of technical documents based on the output of CRPs and TMs will help in disseminating knowledge and capacity building for RR operators and users.

3.2. Project D.2.0.2: Supporting RR Modernization and Innovation

Member States, especially developing Member States, involved in planning or carrying out refurbishment and modernization of RRs often look to the IAEA for advice and assistance and to exchange information and ideas. Similarly, IAEA assistance is requested when new RRs or major innovative systems, such as in-core loops or cold sources, are being planned or constructed. Regional and interregional thematic collaborations, networking and centres of excellence are being increasingly considered worldwide as an appropriate way to enhance utilization of RRs. This project is designed to fulfil these needs by collecting and sharing relevant information, including best practices and lessons learned.

The main objective of this project is:

- To increase the competence of interested Member States to plan and implement large scale refurbishment and modernization of RRs, and to plan and implement construction of new RRs or major RR systems.

Some of the activities proposed to be carried out under this project are:

- Develop regional RR networks and centres of excellence;

- Provide advice and assistance as requested to RR planning, modernization or refurbishment;
- Hold international workshops on modernization and refurbishment of RRs;
- Coordinate a CRP on innovative methods in RR analysis (2008–2011); and
- Support TC projects on modernization and innovation.

3.3. Project D.2.0.3: Addressing RR Fuel Cycle Issues

The IAEA has been involved for more than twenty years in supporting international nuclear non-proliferation efforts associated with reducing the amount of HEU in international commerce. IAEA projects and activities have directly supported the RERTR programme, as well as directly associated efforts to return RR fuel to the country where it was originally enriched. IAEA efforts have included the development and maintenance of several data bases with information related to RRs and RR spent fuel inventories that have been essential in planning and managing both RERTR and spent fuel return programmes. After the announcement of the Global Threat Reduction Initiative (GTRI) by United States Secretary of Energy Spencer Abraham on May 2004 at the IAEA headquarters in Vienna and following recommendations of the 2004 RERTR meeting, held in Vienna in November 2004, IAEA support of RERTR and the programmes of repatriation of RR fuel to the country of origin have been strengthened and a comprehensive number of new activities have been initiated in 2005 and 2006.

At the back end of the fuel cycle, hundreds of RRs worldwide, both operational and shut down but not yet decommissioned, are storing spent fuel on site. In many cases, this RR spent nuclear fuel (RRSNF) is old (more than 30 years) and physically degraded. Therefore the continued safe, reliable and economic handling, management and storage of RRSNF of all types, standard, failed and experimental, is a serious issue for almost all Member States with RRs. In particular, most RRSNF is aluminium clad which is particularly vulnerable to corrosion. Many Member States, especially those having RRs but no power reactors, are expressing concerns about final disposition of RR spent nuclear fuel. Non-proliferation and environmental concerns associated with RRSNF have become just as important, if not more so, as the above mentioned technical concerns. This project is designed to address these issues.

The main objective of this project is:

- To strengthen the capability of interested Member States having RRs to deal with all fuel cycle issues including fuel development, fabrication and qualification, mitigation of identified health, and environmental vulnerabilities associated with spent fuel management; and to promote conversion from HEU to LEU, repatriation of spent fuel to its country of origin, and regional solutions to the back end of the fuel cycle.

Some of the activities proposed to be carried out under this project are:

- Maintain a database on spent fuel from research and test reactors, publish summary statistics periodically;
- Provide advice and assistance as requested to RRs with corroded or otherwise degraded spent fuel;
- Support spent fuel assessment teams for the preparation for shipment of RR spent fuel;
- Update the RR core conversion guidebook to include conversion to high density U-Mo fuels;
- Prepare a technical document on good practices for the management and storage of RR spent fuel;
- Update the guidelines documents on the technical and administrative procedures required for the shipment of spent fuel;
- Support national projects on RR fuel and fuel cladding;
- Prepare a technical document on the economic aspects of the RR nuclear fuel cycle;
- Support activities related to RR conversion and return of RR spent fuel to the country of origin;

- Prepare a technical document on the use of LEU in accelerator driven subcritical assemblies;
- Coordinate an International Technical Working Group on RRs;
- Coordinate a CRP on small-scale, indigenous production of Mo-99 using LEU or neutron activation;
- Coordinate a CRP on conversion of miniature neutron source RRs (MNSR) to low enriched uranium (LEU);
- Evaluate RR support needed for the innovative nuclear power reactors and fuel cycles; and
- Prepare a technical document on good practices for water quality management at RRs.

3.4. Project D.2.0.4: Facilitating Transfer of Know-How on Decommissioning of Research Reactors and Irradiated Core Materials

A large number of RRs are approaching the end of their useful lifetime and become likely candidates for decommissioning. Within the broader range of nuclear facilities, the decommissioning of RRs presents some unique features including experimental devices, unusual materials, and often proximity to populated areas. A lot of RRs are situated in Member States not having adequate resources for the decommissioning of their reactors. Decommissioning is the inevitable legacy of operation of RRs and needs timely and effective management. This includes management of the materials that result from the decommissioning project. To this end, accurate assessments of the material arising from all sources are required and methods/technologies should be available for Member States to minimize arisings and any environmental impact from the wastes. In many instances the radiation damage mechanisms of core materials, especially after high fluences are poorly understood. With many RRs now beginning to decommission or undergoing extensive refurbishment, it has been pointed out that an opportunity to take samples from the core materials and to study their microstructures is being squandered. Besides providing valuable information for decommissioning waste management, the life extension of RRs and input for improved materials for new reactors, the promotion of information exchange and effective coordinated research effort in this area has the potential to increase the understanding of fundamental ageing mechanisms of reactor structural materials.

The main objectives of this project are:

- To increase the capability in interested Member States with RRs to plan and implement decommissioning; and
- To improve understanding of the ageing of irradiated materials and advanced materials for reactor core applications.

Some of the activities proposed to be carried out under this project are:

- Prepare a technical report on decommissioning of RRs and other small nuclear facilities under constrained resources;
- Prepare a technical document on how to make use of samples from the cores of decommissioning or refurbishing reactors to improve understanding of ageing irradiated core materials;
- Prepare a technical document on cost estimates for decommissioning of RRs;
- Prepare a technical report on pool side inspection of RR fuel; and
- Coordinate a CRP on ageing of irradiated reactor core materials.

3.5. Project D.2.0.5: Research reactor operation, maintenance, availability and reliability

Since the mid 1980's, investment in nuclear RR facilities and infrastructure has decreased significantly compared with earlier decades. Many older facilities have been decommissioned, permanently shutdown, or are faced with probable shutdown in the very near future. Funding reductions and limited succession planning have strained available resources, pressuring many facilities to pursue commercial activities to remain in operation. It is in this context that modern RRs are being tasked to conduct advanced research in support of innovative nuclear development (in most cases to very aggressive schedules) and training. To support the scientific, educational and commercial demands being placed

in present times on RRs, many are looking to optimize operations and maintenance activities to ensure the most cost effective completion of their assigned missions. Many Member States look to the IAEA for advice, ideas and information exchange on these topics. This project aims to fulfil these requests by documenting good practices and lessons learnt as an element for strengthening the operational management.

The main objective of this project is:

- To increase the competence of interested Member States to develop operations and/or maintenance plans and implement these plans to optimize facility availability and reliability.

Some of the activities proposed to be carried out under this project are:

- Prepare a technical document on RR availability and reliability;
- Prepare a technical report on RR quality management system development;
- Coordinate a CRP on on-line monitoring systems for RRs; and
- Support TC projects involving operation, maintenance, availability and reliability improvements.

4. Technical Working Group on Research Reactors (TWGRR)

The TWGRR, a new international collaborative undertaking under IAEA's Subprogramme D.2, will consist in a group of experts to provide advice and support programme implementation, reflecting a global network of excellence and expertise in the area of RRs.

4.1. Scope

The TWGRR will focus its work on activities related to all types of RRs, including critical assemblies, subcritical assemblies and pulsed reactors. Also included in the scope are facilities for: RR fuel fabrication, RR fuel development, RR fuel post irradiation and RR spent fuel storage. All managerial areas involved in the operation of the above listed types of facilities are included in the scope of the TWGRR. The TWGRR will give the necessary attention to all of its relevant aspects, including operation, utilization, nuclear fuel cycle, maintenance, refurbishment, modernization, quality assurance, new designs and decommissioning. The TWGRR will especially address the projected needs for RRs on a global and regional basis with a long-term time horizon. The scope of the TWGRR cuts across all IAEA organizational structures dealing with RRs.

4.2. Functions

The functions of the TWGRR are:

- To provide advice and guidance, and to marshal support in their countries for implementation of the IAEA's programmatic activities in the areas of RR operation, utilization, nuclear fuel cycle, maintenance, refurbishment, modernization, quality assurance, new designs and decommissioning;
- To provide a forum for information and knowledge sharing on national and international programmes development in the area of RR operation, utilization, nuclear fuel cycle, maintenance, refurbishment, modernization, quality assurance, new designs and decommissioning;
- To act as a link between the IAEA's activities in specific area and national scientific communities, delivering information from and to national communities;
- To provide advice on preparatory actions in Member States and the IAEA's activities in planning and implementing coordinated research projects, collaborative assessments and other activities as well as the review of the results on RR activities within their scope;
- To develop and/or review selected documents from the Nuclear Energy Series, assess existing gaps and advise on preparation of new ones, in the scope of their field of activity;

- To identify important topics for discussion at the Standing Advisory Group for Nuclear Energy (SAGNE) and contribute to status reports, technical meetings and topical conferences in the field of RRs;
- To provide guidance to member states in order to improve and optimize the utilization of RRs, in national, regional and extra regional contexts. When considered appropriated, to provide guidance in order to define actions for reactors that have been placed in shutdown condition;
- To identify relevant issues and topics which might increase cooperation among different RR centres, particularly in various regions of the world;
- To encourage and facilitate regional and international collaborative programmes in the construction and utilization of RRs, and to be a forum for discussion of issues related to impediments and challenges that can be faced by the concept of a regional RR park;
- To propose the realization of events that will work as a forum for the exchange of information among the participants in all areas indicated in 4.1. Such events include, technical meetings, workshops, international symposiums and conferences;
- To address the projected needs for RRs on a global and regional basis with a long-term time horizon; and
- To encourage participation of young professionals, as appropriate, in IAEA activities.

4.3. Membership

Members of the TWG on RRs shall be appointed by the Deputy Director General, for Nuclear Energy, following consultation with the respective national authorities or organizations.

Members of the TWG on RRs:

- Shall be recognized experts that worked with RRs having extensive links with national technical communities. There shall be appropriate representation on the Group from RR operators, fuel cycle, materials specialists, designers of RRs, researchers and users of RRs;
- Are to serve for a standard length of four years;
- Shall participate in the Group in their personal capacity and shall provide as appropriate views on national policies and strategies in the technical field; and
- May as appropriate bring experts to provide additional information and share experience in the meetings of the TWG.

The Deputy Director General of the Department of Nuclear Energy may from time to time also co-opt additional members and/or invite observers from other Member States and international or regional organizations on an ad-hoc or continuing basis.

4.4. Methods of Work and Deliverables

The TWG on RRs will determine its own methods of work, including preparation of its Agenda, establishment of special groups, keeping of records and other procedures, and report on its findings to SAGNE. The activity of the TWG on RRs between periodic meetings shall be coordinated by a Scientific Secretary taking into due consideration the relevant recommendations of the TWG and SAGNE. Following each meeting the TWG on RRs shall provide the Deputy Director General with a report on its achievements and recommendations. The report shall be also published on the WEB in a format and content agreeable to all members. The Chairman of the TWG shall communicate to SAGNE recommendations for strategic development or other important topics to be discussed at SAGNE meetings.

4.5. Meetings

The TWG on RRs will meet at regular intervals but no more than once in a year with each meeting lasting up to five working days. Extraordinary meetings may be called when required.

5. Conclusions

The IAEA subprogramme on RRs maintains the focus on the different facets of RRs for their effective utilization and management. In order to address increasingly important non-proliferation concerns, emphasis is put on the support of Member States' work in the framework of the GTRI on RR core conversion from HEU to LEU, conversion from HEU to LEU of targets used for radioisotope production, the repatriation of RR fuels to the country of origin, and the global clean out of RR fissile material, including experimental or exotic fuels and sources. To help achieving an enhanced utilization of RRs, the subprogramme supports the establishment of regional and interregional thematic collaborations, networking and centres of excellence. To address the issue of RR support for evolutionary and innovative nuclear power reactors and fuel cycles, the subprogramme promotes international collaboration to assess projected needs over the long term for RRs on a global and regional basis. To support the scientific, educational and commercial demands being placed at present on RRs, a new project on RR operation, maintenance, availability and reliability has been initiated in 2007.

The new TWGRR will provide a unique forum for information and knowledge sharing on national and international programmes in all technical areas of RR and will provide advice and guidance for implementation of the IAEA's programmatic activities in those areas.

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REFURBISHMENT AND PERSPECTIVE FOR ILL

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ABSTRACT

The ILL has been pursuing three major upgrade operations, involving its reactor, its scientific instruments, and the support facilities offered by the site as a whole. The last ten-year safety reviews of the high-flux reactor were held in 1994 and 2002. The first review followed the replacement of the reactor block. The second focused on the installations' compliance with new Safe Shutdown Earthquake standards (0,6g at 6 Hz), with a major refit programme from 2003 to 2006. During this period reactor operations could nevertheless be maintained for 150 days per year, and the new Key Reactor Components programme was launched. In addition, neutronic studies were performed with a view to reducing the consumption of uranium and being able to explore conversion. In parallel to, and beyond, the refit, the performance of ILL's experimental facilities is being enhanced through the long-term on-going Millennium Programme. Finally, in order to attract more and more scientists to the ILL, ESRF and EMBL, the three institutes are together pushing for a development of their joint site, as well as promoting partnerships for science and technology. ILL is thus ensuring both its users and funders that its reactor remains in state-of-the-art condition; we are now ready for 20 more years of operation, ensuring a reliable flux of quality neutrons for investigative purposes.

Introduction

This presentation develops the following items:

- Previous refurbishment of the reactor (replacement of the Reactor Block ...)
- Refit programme
- Four parallel 10-year investment programmes:
 - The renewal of key reactor components;
 - The provision of new moderators, instruments and techniques;
 - The creation of Partnerships for Science and Technology;
 - The partnership to develop the overall site.

These programmes assume that the lifetime of the Institute will extend at least to 2024 and probably beyond.

The very high flux delivered by the HFR makes possible specific high performance experiments, such as:

- Magnetic structure determination under hydrostatic pressure conditions (above 7 GPa),
- Diffraction studies for minute sample volumes (less than 0,001 mm),
- Physical measurements on atoms which present a large nucleon excess,
- Actinide transmutation in compliance with French regulations on nuclear waste management.

Neutrons provide a powerful tool for investigating nature at all levels, from testing theories about the evolution of the universe to elucidating the complex processes of life. The ILL offers experimental facilities and expertise covering all these areas:

- Chemistry and materials (catalysts, pharmaceuticals, hydrogen-storage materials, environmentally-friendly fuels, earth science ...),
- Engineering (operation of engines and efficient combustion, composite materials, welding and surface treatments ...),

- Magnetism and electronics (exotic magnetic behaviour, molecular magnets, high-temperature superconductors, planetary magnetism ...),
- Liquids and soft matter (plastics, cosmetics, viscosity, multi-component lubricants ...),
- Fundamental physics (basis of quantum mechanics, fundamentals of the gravitational force, refining theories of particles and force, cosmological evolution ...),
- And biology (enzymatic mechanisms, cell membranes, digestive processes, drug delivery and action, gene therapy ...).

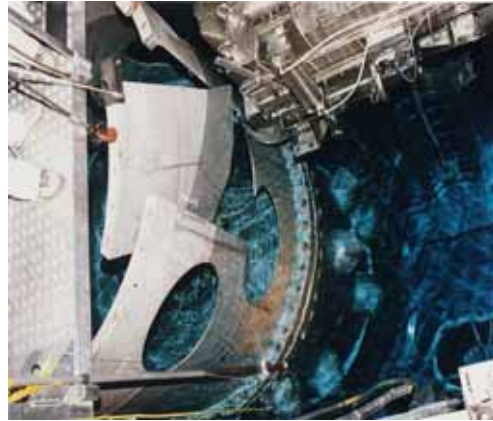
Previous refurbishment of the reactor

The main steps have been:

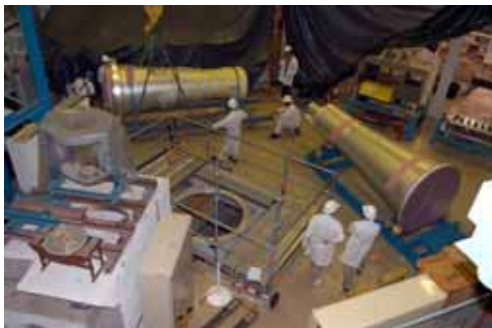
- 1985: A new vertical cold source equipped with a vertical and curved guide tube connected with a turbine. This device feeds ultra-cold neutrons to the experimental instruments.
- 1987: a second (horizontal) cold source. It has been positioned in the front part of a horizontal beam tube. It feeds the second guide hall ILL 22.

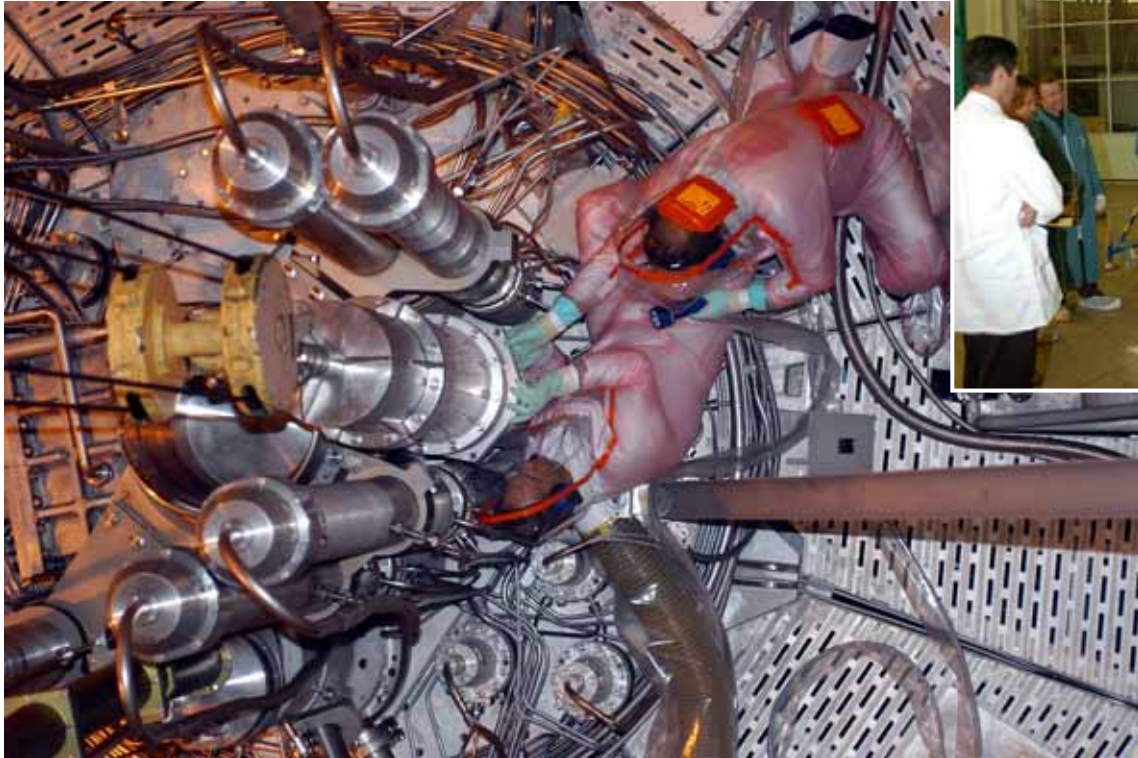


- From 1991 to 1994: replacement of the reactor block consequently the observation of an uncommon trace on the upper antiturbulence grid.

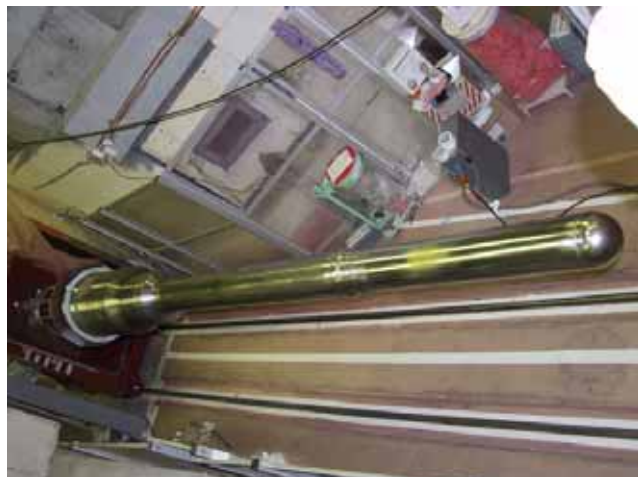


- 2002: replacement of the hot source after 30 years of operation. Some minor modifications were introduced in the light of experience: the thermocouples were secured in position; the central thermocouple now has improved heat resistance; the design has been simplified by eliminating a redundant thermocouple. The source works perfectly at 2000°C, and its three beam lines are highly appreciated by the researchers.





- 2004: replacement of the aluminium beam tube H9 by a zircaloy tube. This has extended its service life, allowing extended reactor operations and reduced radiation exposure for workers.

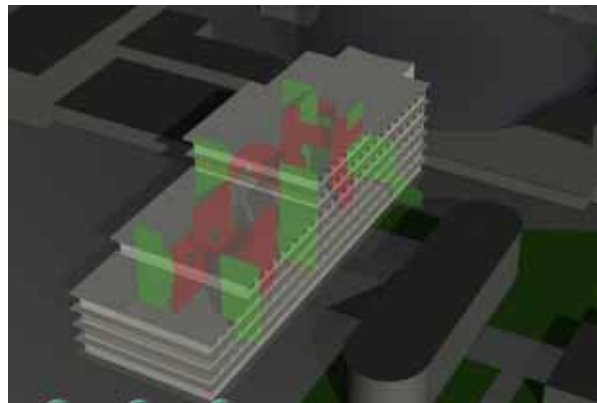


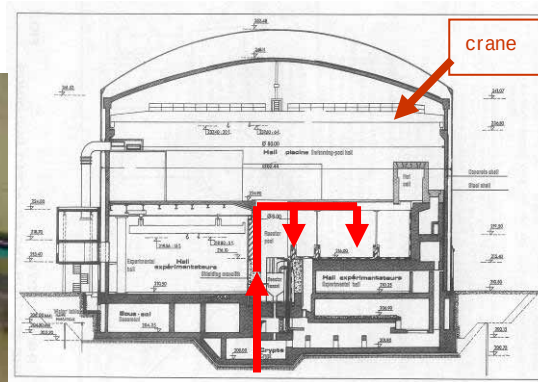
Refit Programme

Following the last safety review in 2002, and in the perspective of 20 more years of operation, a huge amount of work was performed between 2003-2006 to reinforce the reactor and ensure its compliance with Safe Shutdown Earthquake requirements. Throughout the period of this major refit the ILL was able to maintain user service with three 50-day reactor cycles per year. The work included:

- Reinforcement of the transfer canal
- Deconstruction of the concrete structures on the upper floor of the reactor, including the nuclear ventilation system
- 3 new ventilation units to replace the old one
- Comb connexion between the slab (upper floor) and the concrete containment, securing the slab to the containment wall rather than to the wall of the reactor pool, thus reducing stresses
- New seismically qualified circuits: new seismic trip channels, safety valves, leak-tight containment penetrations...
- Modification of the buildings surrounding the reactor: the office building has been reinforced and the front part of the guide halls has been sectioned to avoid contact with the reactor ...
- Doubling of the protection circuits
- Significant reinforcement of security measures (malevolence and theft).

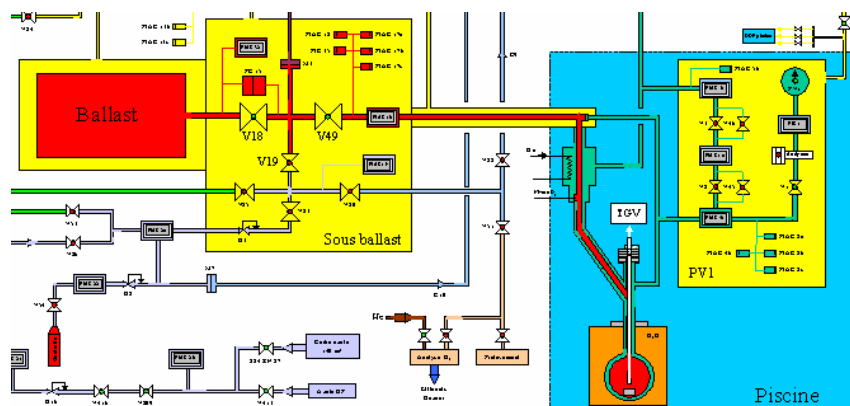
We still need to finalise the EIS-S list ("Elements Important for Safety – Seismic") and reinforce measures against acts of EIS-S-related aggression.





Four parallel 10-year investment programmes

- The aim of the Key Reactor Components programme is to guarantee reliability until 2024. Indeed several important systems have been operating for 35 years. The main focus of this programme is on:
 - Safety rods, 12 new safety rods, project for a new design (**on-going**)
 - **Vertical cold source: renewal of the instrumentation and (digital) control system and of the pressure-resistant housings; addition of a new mimic panel in the control room (accomplished during the Refit Programme, taking advantage of the long 2005-2006 shutdown)**





- **Fuel handling devices: Renewal of the instrumentation and control system with a digital one (done during the Refit Programme, taking advantage of the long 2005-2006 shutdown)**



- Upgrade of the electricity supply (from 15KV to 20 KV) - planned for 2007
- Fuel element: improvements in the use of uranium. This would allow us to take advantage of the package of design tools "Coeur", in collaboration with the CEA. We could then explore and assess the possibilities of using lowly-enriched uranium when it becomes available (on-going)

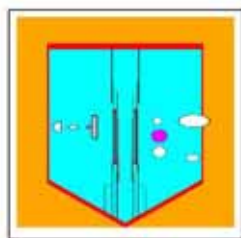
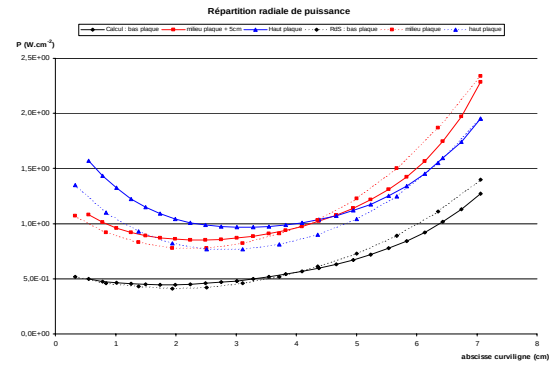
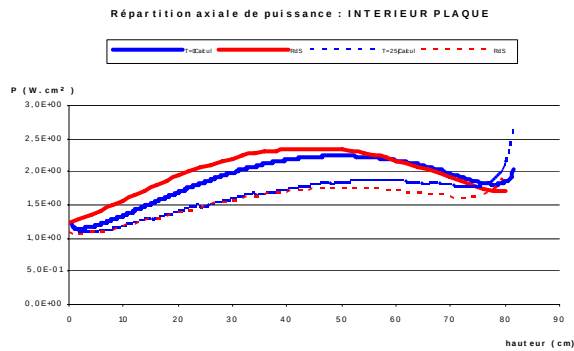


Image 3D du tube à neutrons (base plaque)

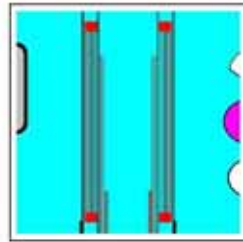


Image 3D du tube à neutrons (base plaque)

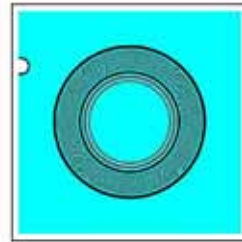


Image 3D du tube à neutrons (base plaque)

- Beam tubes: many will have to be replaced in the near future, and some of them will be manufactured in zircaloy instead of aluminium (on-going)



- A spare vertical cold source cell (in-pile part - the present one is 12 years old)
- Reinforcement of fire, physical, and radiation protection measures, and of the primary circuit components
- Gaseous effluent extraction system: redundancy and seismic qualification (on-going)
- Overhead crane: aseismic bearing pads on the bracket (on-going)
- Nuclear measuring channels
- Renewal of the beam tubes' experimental equipment
- 3rd cold source
- Ultra-cold neutron source

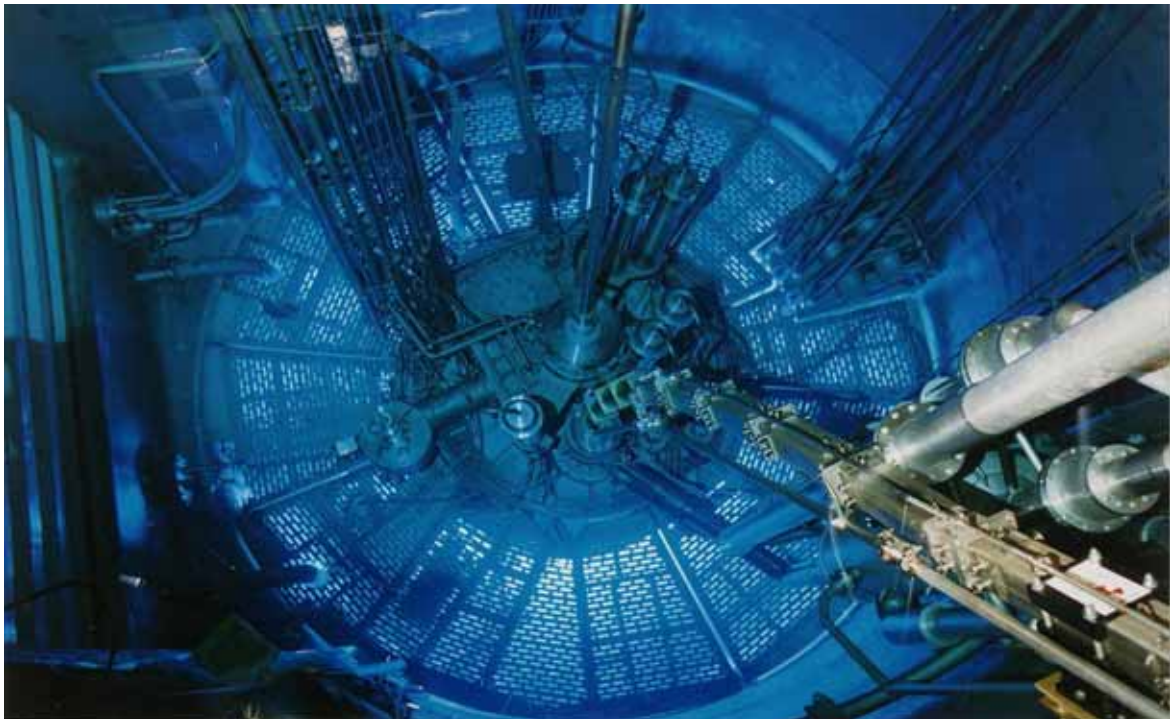
Development of new instruments and techniques:

On the strength of the ten-fold gains in experimental performance achieved by the Millennium Programme's phase M0 (2000–2007), phases M1 and M2 have been launched:

- New high-intensity thermal guides
 - Development of a high-density ultra-cold neutron source
 - Reconfiguration of the instruments in two 5-year phases
 - Increase in public instruments from 25 to 30 by 2011
 - Increase in CRG instruments from 10 to a maximum of 15, according to demand.
- Creation of partnerships for science and technology by capitalising upon the experience of the Partnership for Structural Biology (PSB) laboratory:
 - Partnerships for soft condensed matter, for materials science and engineering
 - An Advanced Neutron Technology Centre surrounded by private engineering companies
 - Partnership for high magnetic fields (ILL and ESRF).
 - Joint development of the site:
 - New entrance (visitor centre, conference and training complex, delivery)
 - Fourth Guest House
 - Common building (library, restaurant)
 - Crèche.

Conclusion

Pushing safety, technological quality and experimental performance, ILL is thus guaranteeing that its reactor, its instruments and its environment remain in state-of-the-art condition; ILL is now ready for 20 more years of operation, ensuring a reliable flux of neutrons for the scientific community.



THE OPAL REACTOR

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ABSTRACT

The OPAL reactor went critical for the first time on 12 August 2006 and achieved full power for the first time on 3 November 2006. This has been a successful project characterised by extensive interaction with the project's stakeholders during project definition and the use of a performance-based turnkey contract which gave the contractor the maximum opportunity to optimise the design to achieve performance and cost effectiveness.

The contractor, INVAP SE, provided significant in-house resources as well as project managing an international team of suppliers and sub-contractor deliver the project's objectives. A key contributor to the project's successful outcomes has been the development and maintenance of an excellent working relationship between the ANSTO and INVAP project teams.

Commissioning was undertaken in accordance with the IAEA recommended stages. The main results of hot commissioning are reviewed and the problems encountered examined. Operational experience since hot commissioning is also reviewed.

1. Introduction

The project to provide a replacement for Australia's HIFAR reactor commenced with Government approval in September 1997 and reached its latest major milestone with the achievement of first full power operation in November 2006. The project has been a successful project for both ANSTO and INVAP. This paper presents, and reflects on with the benefit of hindsight, the approaches used to define the project requirements, choose the supplier and deliver the project, emphasising those good practices that contributed to the project success.

2. Project Planning, Organisation and Implementation

2.1 Assignment of responsibility

Very early in the project the decision was made by ANSTO to procure the facility through the use of a single turnkey contract and to employ a performance-based specification which would necessarily assign to the contractor complete design responsibility. That is, the contractor would not only be responsible for the performance of systems such as process and electrical, but would be responsible for ensuring the delivery of neutrons of the required spectra at the required flux to the required size and number of beam and irradiation facilities.

It took a while for some of ANSTO's stakeholders to embrace this performance-based approach and the general feedback that we had from the then potential contractors was that this was a novel approach. Other than the type of the reactor (pool type), the power (20 MW), the fuel enrichment (maximum 20%)

and the end-users requirements, no other features of the reactor were specified to the Tenderers. The effect was that ANSTO gave significant flexibility to the contractor to produce a cost-effective design for the reactor that could achieve a high performance.

Whilst the contract performance demonstration tests are yet to be completed, the results to date do not suggest that any of performance requirements will not be achieved. If the project were to be restarted now with the benefit of hindsight, neither ANSTO nor INVAP would change this approach

2.2 Stakeholders

In the early stages of the project ANSTO identified a very wide range of organisations and individuals that had an interest in the project. Some of these were obvious, for example government (federal, state and local), users, operators and the local community, but others were less obvious, for example what did ANSTO's public relations group want to be able to achieve? We identified all their expectations, worked towards meeting them, and communicated with them regularly throughout the project. An example of this was that members of the project team met regularly with members of the local community in an open forum which allowed concerns to be addressed.

While we had some early opposition from committed anti-nuclear groups, the local government organisation, and a few individuals of the local community, we have maintained strong broad stakeholder support throughout the project.

2.3 Contractor selection

A two stage process was utilised; a prequalification round followed by the main tender round. In the main round tenderers were required to submit sufficient information by way of conceptual design and calculation to demonstrate that they had a design which was capable of delivering the required performance. As it was recognised that this requirement would cause a significant cost to tenderers, the purpose of the prequalification round was to eliminate all tenderers who failed to convince ANSTO that they had a chance of being successful in the main tender round. While the tenderers were not universally happy with the amount of information ANSTO sought in the main tender round, with the benefit of hindsight, the process served ANSTO very well.

From the tenderers' point of view, this two stage approach was useful in the sense that before committing significant resources to the tender, the tenderers were assured by succeeding in the prequalification process that there was a level playing field, in which all prequalified tenderers had a priori equal chances of being selected.

Tender preparation was one of the most intensive parts of the project for the tenderers, whilst tender evaluation was a very intense part of the project for ANSTO. The tender process, from prequalification to contract signature took almost two years, with the first informative meeting for potential tenderers taking place in September 1998, the prequalification being decided in December 1998, the Request for Tender being issued in August 1999, the tenders being lodged in December 1999, the preferred tenderer being selected in June 2000 and the Contract being signed in July 2000.

Key to the success of the project was that the Tender process assured that the principal's and contractor's goals were aligned: the main goal for the project for both ANSTO and INVAP was for OPAL to be a world class reactor in both radioisotope production and neutron research.

2.4 ANSTO's role post-contract

The contract assigned full design responsibility to INVAP, i.e., INVAP was responsible for the preparation, checking and approval of all design documents. However ANSTO maintained a team of

engineers and scientists who reviewed designs for compliance with the contract requirements and issued acceptances of them prior to manufacture.

During manufacture, installation and pre-commissioning testing INVAP were responsible for the planning and conduct of all inspections and tests. However, the same ANSTO team of engineers and scientists who were responsible for the review and acceptance of designs provided independent witnessing of significant inspections and tests. With components being manufactured in Europe, North and South America, Asia and across a number of Australian states this required a significant commitment of ANSTO's resources, but this high level of independent QA has given significant confidence to ANSTO of the quality of the facility and has been essential in the management of regulatory expectations.

2.5 INVAP's Management Strategy

INVAP decided to carry out all the preparatory work for the project during the tender preparation. All the management plans, the detailed project program, based on a detailed Work Breakdown Structure, the assignment of responsibilities and the organisation chart were defined during the Tender process and submitted to ANSTO with the Tender. This allowed INVAP to very efficiently launch the project once the Contract was signed.

The project management plan imposed several formal and written communication processes between ANSTO and INVAP, whilst several informal communications processes were put in place by ANSTO and INVAP management, including a communication protocol which determined the counterparts in each organisation. Efficient and frequent communications between the two organisations proved to be key to the project success.

INVAP use an integrated team approach for projects: the team responsible for preparing a tender stays with the project for its duration. As ANSTO used a similar approach, most participants in the project have had the opportunity to interact for several years. The building of these long-term partnerships between ANSTO's and INVAP's officers ensured that the goals of both organisations remained aligned.

INVAP used an Earned Value methodology for the project control. The Work Breakdown Structure was used for both project planning and progress control. Formal risk management procedures proved to be valuable.

2.6 Relationship with the regulatory body

In every nuclear project, the management of the interface between the project and the Regulatory Body is fundamental. Key to the success of the project was the cooperation of ANSTO and INVAP in submitting well prepared (mostly by INVAP) and reviewed (by ANSTO) documents to ARPANSA, and frequent and periodic (weekly) meetings between ARPANSA, ANSTO and INVAP helped to obtain the required licences (licence to construct and licence to operate) and authorisations (more than one hundred and thirty authorisations for manufacturing and/or installing safety related components).

3. Hot Commissioning and Operational Experience

The Construction Licence issued by ARPANSA, the Australian nuclear regulator, allowed cold commissioning up to but not including fuel loading. From the issuing of the Licence to Operate, ANSTO took responsibility for operating the facility under INVAP supervision. The commissioning of the reactor was carried out by joint INVAP/ANSTO commissioning teams, while the activities were planned and the daily operations decided by the Commissioning Group, formed by the Reactor Manager, ANSTO's Engineering Manager and INVAP's Design and Commissioning Manager. Key to the success of the

commissioning was the ability of ANSTO, to train, with assistance from INVAP, a full operation crew in time for taking control of the facility.

Stage A cold commissioning tests (74 days), including full system tests with dummy fuel assemblies in the reactor core, were completed in May 2006.

Stage B1 Commissioning

ARPANSA issued the Operating Licence in July 2006 allowing Stage B1 hot commissioning to commence. Three types of fuel assembly were loaded in the first core:

- 212 g U^{235} without burnable poison (BP)
- 383 g U^{235} with BP
- 484 g U^{235} with BP (OPAL standard fuel)

Nine of the full core sixteen fuel assemblies were loaded initially and for each subsequent fuel assembly loaded the control rods were withdrawn and the sub-critical multiplication factor determined. The reactor was taken critical on 12 August 2006 with fourteen fuel assemblies loaded as predicted. The shutdown value of the First Shutdown System with single control rod failure was measured for this first critical core.

The main issue during this testing stage was spurious trips from the nucleonics instrumentation due to electronic noise. This was resolved by close attention to earthing, connections and cable screening.

Stage B1 was completed (5 days), the report issued and ARPANSA approval was received to commence Stage B2.

Stage B2 Commissioning

The full core was loaded and 22 low power tests (up to 400kW) were carried out over 25 days to measure key nuclear and reactivity parameters of the core. The calculated power peaking factor (2.42) was checked by gold wire irradiations and good agreement obtained.

Stage B2 Design verification results

Variable	Value	Design Criteria
Isothermal Feedback Coefficient	-15.74 [pcm/°C]	< 0
Void Feedback Coefficient	-222.89 [pcm/% Void]	< 0
Power Feedback Coefficient	-0.74 [pcm/kW]	< 0
Power Peaking Factor	2.48 [-]	< 3
Shutdown Margin of the First Shutdown System	10067 [pcm]	> 3000
Shutdown Margin (Single Failure) First Shutdown System	6276 [pcm]	> 1000
Shutdown Margin of the Second Shutdown System	10461 [pcm]	> 1000
Safety Factor of Reactivity	2.01 [-]	> 1.5
Shutdown Margin of the First Shutdown System at 0.5 sec	9966 [pcm]	> 2000
Second Shutdown System Reactivity worth in 15 sec	8488 [pcm]	> 3000
Control Rod Plate Reactivity Insertion Rate	19.6 [pcm/sec]	< 20

Issues during Stage B2 commissioning:

- Wide range nucleonics detectors discontinuity as detector changed from pulse to Campbell mode – offsets adjusted and okay.
- Wide range set point for rate enable occurred with detector in pulse mode where the signal is noisy. The rate enable setpoint was raised as this was still within the safety case.
- Failure of a diesel starter motor during a test run. Main cause was identified to be a faulty battery.

Stage C commissioning

Approval was received on 13 October 2006 to commence Stage C commissioning. During this stage the reactor power was increased in steps up to full load (20 MW) which was first achieved on 3 November 2006. Twenty four test procedures were used and more than seventy test records completed.

Issues during Stage C commissioning:

- The CNS turbine was removed so only testing with the CNS in standby (warm) mode was completed.
- The core outlet temperature sensors did not give a true indication of the core outlet temperature. The primary coolant flow path around these detectors was modified and the problem solved.
- Cooling tower performance allowed the operation of the reactor at full power, but extrapolation to the design basis ambient conditions indicated that four of the five fans would not be sufficient for this heat load. The manufacturer has improved the fan performance and further tests are scheduled for March 2007

In addition to the CNS, Stage C testing of some of the irradiation facilities is still outstanding.

Reactor Schedule

The reactor successfully finished its first operating cycle on the 30th of December 2006, after 26 full power days.

Towards the end of the first operating cycle, it was found that the isotopic purity of the heavy water in the reflector vessel is slowly reducing due to a light water leak. The source of the leak has been determined to be a non-structural seal weld associated with the neutron beam tube connection to the vessel. Different repair strategies are being investigated, but the reactor can continue to be operated at full power.

The first reactor refuelling was completed in February. This core is calculated to have the highest PPF and the calculated value (2.49) was confirmed by gold wire measurements (2.48). The reactor is operating at full load 20 MW for testing of neutron beam instruments, commissioning of irradiation facilities and continuing carrying out the contract performance demonstration tests.

CNS commissioning is scheduled to restart mid March.

Commissioning has proceeded to schedule without major problems. ANSTO is now looking forward to completing commissioning and moving into routine operation this year.

4. Conclusion

The OPAL project has been a successful project for both ANSTO and INVAP. Key to the success of the project were:

- Ongoing stakeholder commitment
- A very carefully designed and conducted tendering process
- Effective assignment of responsibilities through the contract
- Goal alignment between the principal and the supplier
- Strong cooperation, enhanced by frequent communications between the parties
- Integrated management teams, both within ANSTO and INVAP.
- Formal management procedures, known, accepted and reviewed by the parties.
- Detailed program, used for both project programming and project control.
- Joint management between ANSTO and INVAP of regulatory issues, including frequent and periodic meetings with the Regulatory Body.
- The timely availability of a complete and fully trained operation crew.

Last, but most important, ANSTO, INVAP and ARPANSA succeeded in assigning excellent people to the project. At the end of the day, any project is as good as the people participating in it.



Reducing the Risk of Nuclear Terrorism through the Creation of a New Forum for Collecting and Sharing Nuclear Material Security Best Practices: The Case for the World Institute for Nuclear Security (WINS)

Corey Hinderstein
March 1, 2007

One of the greatest security challenges of the 21st century is preventing the spread and use of nuclear weapons. The rise of global terrorism has created a new demand for nuclear weapons and a new willingness to use them. There is little doubt that if terrorists acquire nuclear weapons they will use them.

Supplies of highly enriched uranium and plutonium, the necessary materials to make a nuclear weapon, are widely dispersed around the world. Obtaining these essential ingredients is one of the hardest parts of making a nuclear weapon. Since these materials are difficult to make, the most likely way a terrorist organization will get them is through illicit purchase or theft. Terrorists will try to acquire nuclear material from wherever it is easiest to steal or from anyone willing to sell. Terrorists won't necessarily look where there is the most material; they may go to the place where the material is the most vulnerable or accessible.

Vulnerable nuclear material anywhere is a threat to everyone, everywhere. Like most global problems, the defense against nuclear terrorism is dependent upon cooperative and collective global action.

Mission and Need for Nuclear Material Security Best Practices Organization

The world community is aware of the danger of nuclear terrorism. Among other things, the concern of the international community has been translated since 9/11 into several new international instruments to help strengthen our global capacity to keep nuclear materials and weapons out of the hands of terrorists. Key among these initiatives are UN Security Council Resolution 1540; the Amendment to the Convention on the Physical Protection of Nuclear Materials; the International Convention for the Suppression of Acts of Nuclear Terrorism and the creation of the Nuclear Security Fund at the International Atomic Energy Agency (IAEA) with its associated plan for assisting states with implementation of their security obligations, including through the creation of more detailed nuclear security guidelines. While these initiatives form an important legal and institutional architecture, they still fall short.

There are several key reasons that our existing global nuclear security architecture is not yet sufficient. These reasons include inadequate implementing mechanisms for existing

nuclear security initiatives, and the institutional and budgetary constraints of the IAEA, the international organization charged with supporting most of these efforts.

While the various legal and voluntary initiatives described above are important for beginning to create necessary norms and legal frameworks, in very few cases have they been translated into actions. UNSCR 1540 established a Committee to help review the reports required to be submitted by states, but the Committee is not equipped (by either budget or staff) to help states implement the requirements of the resolution. The Convention on the Physical Protection of Nuclear Materials (CPPNM) has no implementing or oversight mechanism, and the Amendment to the Convention has not entered into force (only seven of the 80 states required for its entry into force have ratified it).

The International Atomic Energy Agency does provide assistance to states requesting help with nuclear security issues, but its capacity for action is limited by its budget (with an annual total of about \$15 million per year for all of its nuclear security programs) and its personnel. The IAEA is also limited by its charter to working with states (as opposed to industry for example), and its activities are generally limited to non-weapons materials and facilities.

In addition, despite various bi- and multi-lateral mechanisms for nuclear security cooperation, a comprehensive, global approach to nuclear material security is still missing. Physical protection and MC&A practices vary from country to country and facility to facility, and this is particularly true because establishing the standards for the physical protection of nuclear materials is the sovereign responsibility of the state.

A global best practices organization could be the mechanism for raising the level of global best practices of nuclear materials security in a time urgent way, and serve as a tool for industry and operators who want to stay ahead of the threat. Such an organization could provide a forum for the exchange of experience, lessons learned, and new ideas at the “grass roots” facility-operations level: a forum for practitioners rather than policy makers. In this way, the nuclear materials management community, and all of the partners involved in the organization can reduce the risk of a terrorist event that would threaten the viability of peaceful nuclear activities internationally.

Potential Activities

The primary role of a global best practices organization would be to provide a forum for the exchange of information between operators, industry, governments, and government entities regarding on-the-ground experiences and lessons learned in providing for the security of nuclear materials. Through consultation with the international nuclear materials management community we have concluded that this core mission is not being done by any existing mechanism, and would be valuable to facility operators and managers. We are exploring the creation of such an organization, nominally called the World Institute for Nuclear Security (WINS).

WINS could conduct and facilitate a range of activities, in which entities can choose to participate voluntarily and on a case-by-case basis. For reasons of staffing and financing, it is likely that WINS activities will start from a narrow focus and then broaden with time. It also may be difficult initially to directly involve some military materials or facilities, but nuclear weapons states' and non-NPT states' facility operators and authorities could still participate broadly in the activities of the organization. The process of expanding WINS activities will likely be driven by the confidence of the participants.

An initial activity of WINS should be to collect “best practices” for nuclear material security. WINS will serve as a forum for operators and practitioners to share security strategies that go beyond internationally accepted standards to improve material security. These approaches would contribute to efforts to help facilities implement obligations under UN Security Council resolution 1540 and IAEA INFCIRC/225 Rev. 4.

The IAEA has developed a 2006-2009 Nuclear Security Plan to “achieve improved worldwide security of nuclear and other radiological material.” This is a significant and important achievement. In support of this effort, activities that would complement and supplement the Plan and assist the IAEA in realizing its nuclear security goals should be a major focus of WINS. It will be vital for WINS, in particular in the start-up phase, to work closely with the IAEA to avoid duplication of effort and therefore wasting of resources. There are some areas that the IAEA cannot address and where WINS may be better able to contribute. Some of these areas could include working directly with facilities in the nuclear weapon states, working with non-civilian entities, conducting activities in non-NPT states, and engaging directly with the nuclear industry, including with facilities and operators.

WINS has an important role to play in raising the international awareness of the need for increased attention to nuclear materials security. WINS can also contribute to establishing and building the resource base of experts and services for nuclear material security. These kinds of activities can benefit all groups and individuals working in the field.

We believe WINS could also contemplate the conduct of peer reviews to be carried out on a voluntary basis, and designed to assist facilities in identifying ways that they can improve security implementation. On the one hand, peer review might be one of the most sensitive and difficult activities for a new organization to undertake, and therefore might not be easily incorporated into the initial activities of WINS. However, the experience of the World Association of Nuclear Operators (WANO), which focuses on nuclear safety, demonstrated that the conduct of peer reviews was important in shaping the activities of and support for the organization.

Scope of Materials to be Addressed

Defining the materials to be addressed by WINS activities will impact organizational priorities and shape activities and participation in the organization. The global universe of nuclear materials and facility types is diverse. WINS could address all nuclear and

radioactive materials, be limited to nuclear weapons direct use materials in significant quantities, or cover another subset of materials.

We recommend that the decision on which materials to address should be based on a risk-based assessment that returns to the core rationale for establishing the organization. Under these terms, for example, WINS could define its initial goal as ensuring security of unirradiated direct use materials. This would include highly enriched uranium (HEU), separated plutonium, and fresh MOX. WINS's ability to address the most sensitive materials in the category (e.g., military stockpiles) would depend on the active participation of facility operators and governments responsible for such materials.

Nothing in the definition of this scope should be interpreted as limiting the range of membership and future activities of the organization. Activities geared toward best practices in the management of material as defined above will naturally have potential application for less attractive material. Therefore, participation and information sharing with facilities responsible for other related nuclear and radiological materials should be supported and encouraged.

Potential Participants

Potential participants in the WINS effort could include:

- (1) ***Private Industry***
- (2) ***Government Agencies and Government Entities***
- (3) ***International Organizations***
- (4) ***Non-Government Organizations***
- (5) ***Professional Associations***
- (6) ***Universities***

There are many ways to organize or categorize participation. WANO, for example, is organized primarily based on geographic location of the members. For a diverse participation, such as envisioned for WINS, it may prove valuable to organize participants around technology and facility type. For information sharing purposes, there are likely to be areas that are most valuable for operators of similar facilities, although many security issues can be discussed broadly.

Costs and Financing

Start-up funding could be acquired through voluntary donations from industry, related government entities, NGOs, individuals, associations, professional organizations, and international organizations. WINS will generate sustained funding if it proves to contribute to the interests and values of the nuclear community.

In order for the entity to remain viable over time, we believe it will be important to create a sustainable funding stream. Contributions could be made through in-kind donations, up front commitments, sustaining commitments, and ad-hoc donations.

Challenges

There are many challenges to the creation of an institution to collect and disseminate best practices on nuclear material security. These include, in particular, sensitivities about the sharing of security information between countries and organizations. This should not be an insurmountable barrier as the nature of WINS is not designed to be a public forum, nor is information about specific security measures in place at specific sites necessary.

It is also important to emphasize that participation in WINS will not mean the acceptance of new obligations on the part of facilities or organizations. The goal is information sharing and exchange, not imposing new security obligations.

Finally, some potential participants have noted that there is no generally accepted economic rationale for participation, as there was in the case of safety concerns following the Chernobyl accident for establishment of WANO. To this argument, we respond that the potential global economic costs of a nuclear terrorism event are likely to be substantial and the impact on the nuclear industry may be disproportionate to that experienced by industries in general. The international nuclear community should not wait for a “security Chernobyl” to take steps from preventing a terrorist from accessing nuclear material.

Next Steps

In November 2006, NTI organized and co-sponsored an international “Experts Group” meeting to explore the WINS concept. Twenty-five participants attended from 17 different countries and the International Atomic Energy Agency (IAEA), including government regulators, ministries, and private industry. At the conclusion of the meeting, there was general consensus on the need for WINS and the importance of continuing to advance the concept with support from NTI and other international partners.

As a result of the Experts’ discussion, NTI, in partnership with INMM and the IAEA, is working to carry out three “pilot projects” to demonstrate the value of WINS-type activities to nuclear material managers and facility operators. These activities will be developed through consultation with a number of international partners, including the facility operator communities. We hope to define and carry out a demonstration project in two areas in 2007: plutonium security and highly enriched uranium. Concurrently, NTI is working to cultivate high-level international political support for WINS.

The Renaissance of Fast Sodium Reactors
2007 assessment: situation and contributions from the PHENIX experimental reactor

RRFM/IGORR meeting
12 to 14 march 2007 –Lyon -France.

J Guidez: Director of PHENIX plant.

The first nuclear reactor to produce electrical current was the fast sodium/potassium reactor EBR1, on 20 December 1951 in Idaho (USA) . Following this pioneering experience, France, Germany, Great Britain, USA, Japan, Russia and India launched construction of fast sodium reactors. In the “post –Chernobyl” years, waves of protest against nuclear power grew and swelled, leading to a strong overall slowdown for this reactor type. The SNR300 project in Germany never started up, and was shut down. In Great Britain, PFR was definitely shut down, operation of MONJU in Japan and BN800 project in Russia were frozen, FFTF in the United States shut down, and finally the SPX1 project in France was also stopped. When PHENIX started back up in 2003, there were only three other research reactors operating worldwide: FBTR in India, BOR 60 in Russia and JOYO in Japan, and one power reactor BN600 in Russia.

The Generation IV initiative was the opportunity for global thinking about reactors for the future, referred to as fourth generation reactors. Six reactor designs were selected, including the fast sodium reactor. However, after several years, most of the countries (in or out of GENIV group) have officially announced or confirmed that the fast sodium reactor is their priority reference design. These countries include Japan, China, Korea, Russia (simultaneously with lead reactors), and India. With the GNEP, the United States has announced a project for a fast sodium-cooled, waste-burning reactor. In France, within the scope of the law of 28 June 2006, the country has announced and confirmed the decision to build a prototype scheduled for operation in 2020.

These and other plans are all sustained in a very practical manner by the ongoing production in the field. PHENIX has been operating since 2003, demonstrating the fast reactors’ ability to burn waste. Following the excellent results obtained by the BN600, Russia has re-launched the BN 800 project. China is currently in the process of building a 65-MWT research reactor, scheduled for divergence in 2009. In Japan, work is underway on MONJU for divergence in 2008. In India, a 1200-MWT power reactor is under construction, scheduled for divergence in September 2010, the first of a three-reactor unit.

The stakes behind this renaissance in nuclear power are important indeed. These fast reactors promise to produce world energy for thousands of years through breeding. No production of greenhouse gases. And long-life waste is burned. Moreover, significant progress has been made in terms of safety, reliability, availability and inspectability for this reactor type.

A presentation is made on the experience gained at PHENIX since 1974, and on the industrial validation during his operation, of the points described above.

1. THE PAST UNTIL 2003

The first nuclear reactor to produce electricity was a sodium-cooled fast reactor (NAK), in 1951, the EBR 1 in the United States.

Since that time, 18 fast sodium reactors have operated in many different countries and in 2003, when Phénix started back up, there remained three experimental reactors in the world: BOR60 in Russia, FBTR in India, JOYO in Japan, and one power-producing reactor: BN600 (600 MWe) in Russia.

The enclosed table shows the number of years of operation for these 18 reactors, as of 2007, and shows that accumulated operating experience comes to approximately 379 years. This has led to extremely significant feedback benefiting the fast sodium reactor type.

FAST REACTORS OPERATIONAL DATA 2007				
Reactor (country)	Thermal Power (MW)	First criticality	Final shut-down	Operational period (years)
EBR-I (USA)	1.4	1951	1957	6
BR-5/BR-10 (Russia)	8	1958	2002	44
DFR (UK)	60	1959	1977	18
EBR-II (USA)	62.5	1961	1991	30
EFFBR (USA)	200	1963	1972	9
Rapsodie (France)	40	1967	1983	16
BOR-60 (Russia)	55	1968		39
SEFOR (USA)	20	1969	1972	3
BN-350 (Kazakhstan)	750	1972	1999	27
Phenix (France)	563	1973		34
PFR (UK)	650	1974	1994	20
JOYO (Japan)	50-75/100	1977		30
KNK-II (Germany)	58	1977	1991	14
FFTF (USA)	400	1980	1993	13
BN-600 (Russia)	1470	1980		24
SuperPhenix (France)	3000	1985	1997	12
FBTR (India)	40	1985		22
MONJU (Japan)	714	1994		13
BN-800 (Russia)	2000	Under construction		
CEFR (China)	65	Under construction		
PFBR (India)	1250	Under construction		
Total All Fast Reactors				379

Total per country (years)	
Russia	110
USA	61
France	62
UK	38
Japan	43
Kazakhstan	27
India	22
Germany	14

2. GENERATION IV

The Generation IV initiative has enabled a comprehensive overview for the future, of the possibilities of the reactor types, leading to a list of 6 reactor types, including 3 fast reactors:

- VHTR,
- Gas-cooled fast reactor
- Sodium-cooled fast reactor
- Lead-cooled fast reactor
- Molten salt reactor,
- Supercritical water-cooled reactor

After several years of research and reflection, the actual situation in 2007 with respect to the six possibilities is as follows:

Korea: Korea has announced the choice of the sodium reactor as the GEN IV reactor. However, no actual construction projects are underway (simply the Kalimer research project).

China: China continues its efforts in several fields, particularly in the HTR. However, a 75-MWth sodium-cooled fast reactor is under construction, scheduled for divergence in 2010.



CEFR (China)

This prototype reactor has been described as the start of a series of this type of reactor.

India: India has long seen the fast reactor type as a long-term energy solution for the future.

The FBTR reactor has been operational since 19__ and has applied for a 20-year lifetime extension. This experimental reactor is used to qualify materials and fuel.



FBTR (India)

However, construction of a 1200-MWth reactor was launched in late 2004.



PBFR/India/2007

This reactor should diverge in 2010, and is the first in a series of 3 identical reactors.

Russia: Here too, the choice of sodium reactors has long been made. The recent successful operations of the BN600 has led Russia to request an extension in the life of the reactor.



BN 600 (Russia)

The BN800 reactor was budgeted in 2006. Work has been resumed, and foundations have been poured.

Reactor divergence is scheduled for approximately 2012.

However, Russia continues to work on the lead-cooled fast reactor option, which is the reactor type which equips the nuclear submarines. Russia has strong expertise in this field.

Japan: Within the scope of GenIV, Japan is the leader for the sodium-cooled fast reactors. With Joyo serving as an irradiation reactor, and Monju as a power-producing reactor, Japan has the tools at hand to start such a reactor type up. The Monju reactor had been shut down since the 1992 sodium leak. It was authorized to undertake repairs in 2006 and should gradually start back up, achieving full power between 2008 and 2010.



JOYO (Japan)



MONJU (Japan)

USA: The last American fast reactor (FFTF) has been shut down.

The GNEP project introduced in 2006 calls for an actinide-burning fast reactor. If this reactor is built, it will most likely be a sodium-cooled fast reactor.

France

In France, 2006 was a very active and positive year:

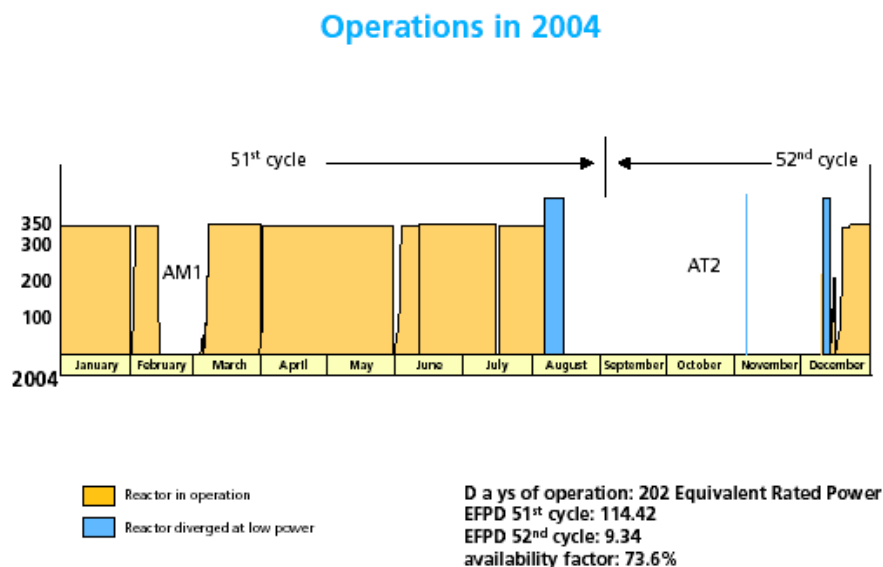
- January 2006: the President of France announced that a GenIV prototype reactor should be operation by 2020.
- June 2006: the law on nuclear waste processing and future was voted. This law also confirms the prototype by 2020.
- December 2006: The Nuclear Energy Council (CEN) confirmed France's nuclear policy for the years to come. The statement is made that the 2002 prototype will be a sodium-cooled fast reactor, and that the gas reactor remains a long-term development option.

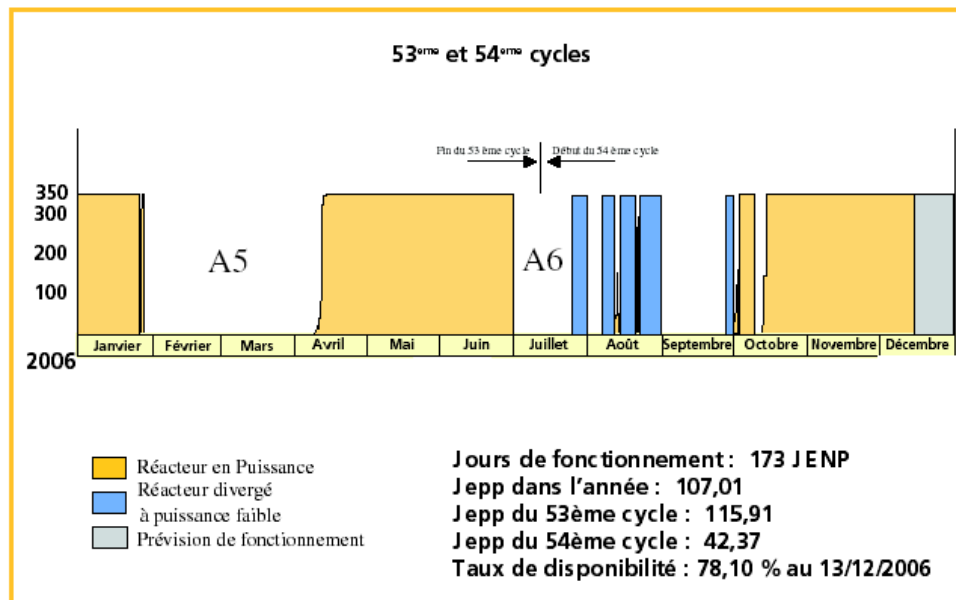
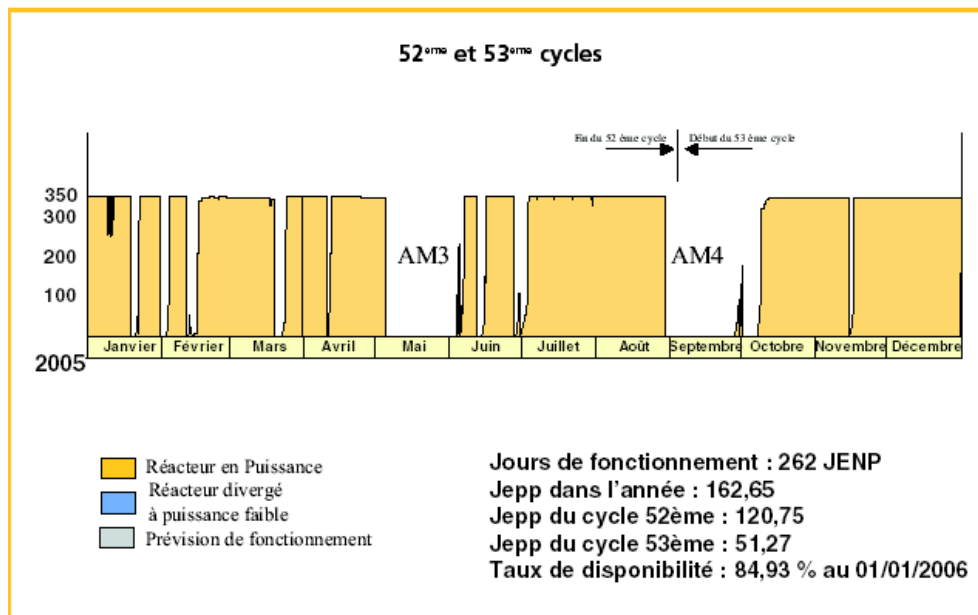
Conclusion: Options remain open on various levels for the potentially promising reactors, in particular the lead-cooled fast reactor (Russia,...), the gas-cooled fast reactor (France), the HTR reactors (China, USA,...).

However, significant feedback on sodium-cooled reactors has emphasized the sodium-cooled solution, and convergence is taking place, at least in the short term, towards the construction of sodium-cooled fast reactors.

3. PHÉNIX OPERATIONS SINCE STARTING BACK UP

During these positive times, the successful operations at the Phénix reactors continues on, since starting back up in 2003.





The 3 diagrams below correspond to the availability at interest rates of 74 %, 85 % and 78 %.

In early 2007, the record for operation without spurious shutdown was beaten on 21 January. The record was established in 1990, with 99 days of operations.

In addition to producing electricity, reactor operations can also include a research and test program, for the purpose of materials development and future fuels, and transmutation experiments.

4. EXPERIMENTS FOCUSED ON DEVELOPMENT OF FUTURE SYSTEMS

An experimental program is being conducted in the PHENIX reactor within the scope of developing future energy systems (FNR-G, FNR-Na, ADS, ITER ...). The purpose is to acquire knowledge on the inert materials under consideration as structural materials for these systems (MATRIX, FUTURIX-MI, and ELIXIR) and on innovative fuel concepts (FUTURIX-Concepts).

The MATRIX program, led jointly with the US-DOE, consists in irradiating, with a target dose of 65 dpa, inert materials such as ceramics (SiC, TiN, ZrN...) and metallic alloys (T91, T92, ODS...) considered as structural materials for the future systems (ADS, FNR-G, ITER). The entire experiment includes over 1000 specimens. Eighteen months after program launch, the experimental rig was placed in the reactor in early 2006.

The FUTURIX-MI experiment, also a collaboration with ITU and USDOE, consists in studying the behavior of refractory materials (Mo- or Nb-based carbide and nitride ceramics) under irradiation and high temperature (approx. 1000°C). These materials are being studied for the FNR-G structures. Work in 2006 focused on producing the internal components (sample holders, DAF ...) for the rig, scheduled to enter the reactor in early 2007.

On the subject of inert materials, the irradiation objectives for the ELIXIR experiment were reached in 2006. The ELIXIR program called for irradiation up to 45 dpa of austenitic steels used for PWR reactor internals, and martensitic steels for fusion reactors such as ITER. The irradiated specimens will be sent to Saclay and Karlsruhe in early 2007.

Research on innovative fuels for future systems (FNR-G) is the objective of the FUTURIX-Concept experiment. This program's researches, produces and irradiates, in PHENIX, special nitride and carbide type fuel concepts: pellets and micro-pellets of fuels inserted in honeycomb-shaped inert structures, beads of fissile and inert particles in TiN or SiC type matrices. The experimental pins were completed in 2006.

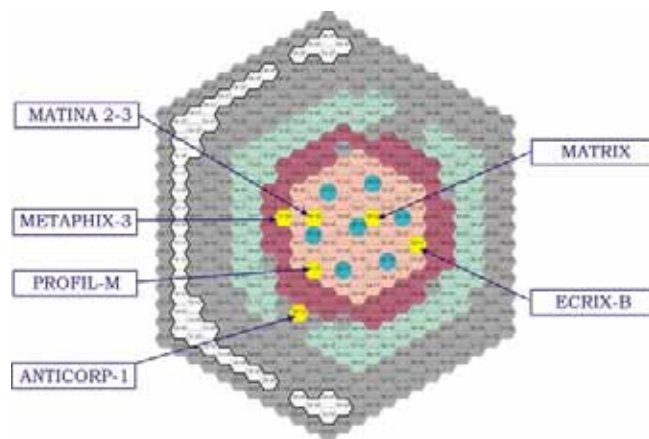
In addition, within the scope of increased plutonium consumption in FNR-Na fuels, the CAPRIX experiment, which contained two UPuO_2 pins with 45 % plutonium content, reached the objective of an average fission rate of 10 at% in 2006.

5. EXPERIMENTS FOCUSED ON TRANSMUTATION RESEARCH

The goal of the experiments conducted at PHENIX is to demonstrate the technical feasibility of transmutation of minor actinides and long-life fission products in a fast neutron reactor. The experimental objectives include:

- acquisition of basic neutronic data (PROFIL experiments),
- development of concepts for transmutation targets (ECRIX, CAMIX-COCHIX, MATINA) for minor actinides and fission products (ANTICORP),
- development of fuel for transmutation reactors (METAPHIX, FUTURIX-FTA).

The PROFIL-R and PROFIL-M experiments, conducted to acquire knowledge on nuclear reactions, comprise pins containing several dozen isotope specimens (actinides and fission products). During irradiation, these pins are placed in rapid spectrum (PROFIL-R) or slightly moderated spectrum (PROFIL-M). The PROFIL-R experiment reached its irradiation objective in August 2005, after 252 EFPD. The experimental pins were sent to Cadarache in late 2006. The PROFIL-M experiment was placed in the reactor in July 2006.



Experiments placed in the reactor core at the start of the 54th irradiation cycle

The MATINA 1A and MATINA 2-3 experiments were dedicated to studying the behavior of the inert matrices used as transmutation target support materials. Several materials were tested (ceramics, refractory metals). The minor actinides to be transmuted were able to be simulated by fissile phases.

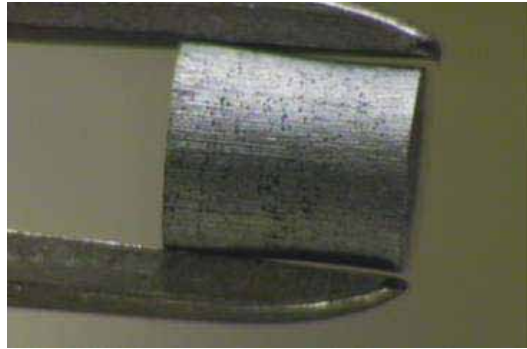
The first part of these experiments (MATINA 1A) came out of the reactor in 2004. The non-destructive testing conducted at PHENIX, and the first results from the destructive testing at Cadarache confirm the good behavior of the MgO matrix, currently considered the reference material. The second experimental part, MATINA 2-3, studies new ceramic materials and optimized concepts of actinide dispersion in the matrices. The experiment was placed in the reactor during the A6 outage in July 2006.

The ECRIX and CAMIX-COCHIX experiments test various concepts of transmutation targets (inert matrices containing particles of Americium) in slightly moderated neutronic spectra, either in the core (ECRIX-B), or in the fertile blankets (ECRIX-H, CAMIX-COCHIX). The ECRIX-B is still in the reactor after 410 EFPD. However, irradiation of ECRIX-H terminated in early 2006 after having attained a fission rate of approximately 35at%. Pins are currently still being examined in the Irradiated Elements Cell (IEC). For the CAMIX and COCHIX experiments, 2006 was primarily spent assembling the rigs at PHENIX and preparing the application for irradiation authorization with the Safety Authority.

On the subject of transmutation of long-life fission products, the ANTICORP-1 ⁹⁹Tc ingots under irradiation, has been in the reactor since 2003, with the irradiation objective of 720 EFPD. In late 2006, the transmutation rate had reached approximately 18at%.

The METAPHIX and FUTURIX-FTA experiments involve research on fuels used for the incineration of minor actinides (homogenous mode). These are international programs involving foreign partners such as ITU and CRIEPI, for METAPHIX and US-DOE, ITU and JAEA for FUTURIX-FTA. The three METAPHIX 1-2-3 rigs, each containing three experimental pins of UpuZr alloy with various levels of minor actinides and rare earths (up to 5%) entered the reactor in late 2003. The METAPHIX 1 and 2 rigs came out in August 2004 and July 2006 respectively, after 120 EFPD and 360 EFPD of irradiation, which corresponds to fission rates of 2.5 and 7 at%. Non-destructive testing of METAPHIX-2 will take place in the IEC during 2007. The experimental pins from the METAPHIX 3 rig, scheduled for unloading in 2008, had reached fission rates of approximately 7 at% at the end of 2006.

The FUTURIX-FTA experiment, whose objective is to study the different fuels containing high actinide contents (between 1.3 and 5.8 g/cm³), consists of three rigs holding 8 experimental pins in all. The fuel types under study are the metallic alloys (UPuNpAmZr, PuAmZr), the nitrides (PuAmZrN, UPuNpAmN), the CERCER (PuAmO₂+MgO) and the CERMET (PuAmO₂+Mo, PuAmZrO₂+Mo). Work in 2006 primarily concentrated on making the pellets and the pins, which were received at PHÉNIX in early September. The application for irradiation authorization is currently being processed. The objective is to place the experiments in the reactor during the first quarter of 2007.



FUTURIX-FTA – Experimental CERMET pellet

6. CONCLUSION

In conclusion, with its transmutation experiments, in conjunction with the research on separation, Phénix is currently in the process of providing the successful demonstration of the possibility of sodium-cooled fast reactors for optimized management of future nuclear waste.

In the 1980's, Phénix successfully reprocessed, first at APM, then at the Hague, the equivalent of four and one-half cores (which is approximately 25 tons of fuel). It then re-made, then re-used this fuel to demonstrate on the industrial scale, the technological feasibility of breeding. Breeding multiplies by a factor of approximately 100 the possibilities of using uranium, thus avoiding any possibility of shortage in the future.

These two demonstrations show that the sodium-cooled fast reactor is a tool for producing electricity which also entails sustainable development criteria in terms of overcoming shortages and waste optimization.

When the Phénix reactor shuts down in 2009, a series of new reactors will just be getting underway: CEFR in China, PBFR in India, BN800 in Russia and MONJU in Japan.

The sodium-cooled fast reactor increasingly appears to be the number one candidate in the category of Fourth Generation reactors.

DEVELOPING RESEARCH REACTOR COALITIONS AND CENTRES OF EXCELLENCE

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ABSTRACT

The IAEA, in line with its statute and mandatory responsibilities to support its member states in the promotion of peaceful uses of nuclear energy in concert with global nuclear non-proliferation, nuclear material security, and threat reduction objectives is well positioned to provide support for regional and international cooperation involving the research reactor community.

The IAEA is pleased to announce an initiative to form one or more coalitions of research reactor operators and stakeholders to improve the sustainability of research reactors through improved market analysis and strategic/business planning, joint marketing of services, increased contacts with prospective customers and enhanced public information. Such coalition(s) will also be designed to promulgate high standards of nuclear material security, safety, quality control/assurance and to conform with global non-proliferation trends.

1. Introduction

Research reactors continue to play a key role in the development of peaceful uses of atomic energy. They are used for a variety of purposes such as education and training, production of medical and industrial isotopes, non-destructive testing, analytical studies, modification of materials, for research in physics, biology and materials science, and in support of nuclear power programmes. The IAEA Research Reactor Data Base lists about 250 operational research reactors worldwide, many of which have been operating for more than 40 years.

Through both statistical and anecdotal evidence, it is clear that many of these reactors are underutilized, face critical issues related to sustainability, and must make important decisions concerning future operation. These challenges are occurring in the context of increased concerns over global non-proliferation and nuclear material security, due to which research reactor operators are coming under increased pressure to substantially improve physical security and convert to the use of low enriched uranium (LEU) fuel. Thus, there is a complex environment for research reactors, and one in which underutilized and therefore likely poorly funded facilities invoke particular concern.

Many research reactors are challenged to generate sufficient income to offset operational costs, often in a context of declining political and/or public support. Many research reactor operators have limited access to potential customers for their services and are not familiar with the business planning concepts needed to secure additional commercial revenues or governmental or international programme funding. This not only results in reduced income for the facilities involved, but sometimes also in research reactor services priced below full cost, preventing recovery of back-end costs and creating unsustainable market norms.

Parochial attitudes and competitive behaviour restrict information sharing, dissemination of best practices, and mutual support that could otherwise result in a coordinated approach to market development, building upon strengths of various facilities. Moreover, belief that the markets for research reactor products and services are a “zero-sum” game, with market gains by one research reactor coming at the expense of another facility, result in a general lack of openness within the research reactor community.

Yet there is evidence to suggest that the market for research reactor services is supply limited, rather than demand limited. A number of factors limit the ability of research reactors to expand their user base and to generate new sources of revenue:

- Many potential customers do not know how, or where, to contact the research reactor community, and have only limited knowledge or awareness of the range of research reactor services, equipment and locations available.
- The standards of quality control and quality assurance between research reactors are not uniform, impede business development, and may result in a lack of confidence in service reliability. As a consequence, customers need to conduct due diligence for each facility to be used, reducing the enthusiasm and financial rationale for developing additional sources of supply.
- Transport of radionuclides is becoming increasingly difficult, with examples of shipments held in customs, prevented from leaving the country of origin or from entering the customer destination, and requires specific expertise and experience to manage this issue.

In order to address the complex of issues related to sustainability, security, and non-proliferation aspects of research reactors, and to promote international and regional cooperation, the IAEA is initiating the Research Reactor Coalitions and Centres of Excellence initiative. This activity is supported by a two-year grant from the Nuclear Threat Initiative, Inc. (NTI), and by a 2007-2008 IAEA Technical Cooperation Project, “Enhancement of the Sustainability of Research Reactors and their Safe Operation Through Regional Cooperation, Networking, and Coalitions” (RER/4/029). These two activities will work in an integrated manner, along with other relevant national and regional IAEA Technical Cooperation projects and complementary IAEA regular and extra-budgetary funded programme activities in research reactor utilization, safety, security, and the fuel cycle. These activities were endorsed by the IAEA Board of Governors in its March 2007 meeting which encouraged regional cooperation and networking among research reactors.

The aim of this initiative will be to establish a pilot project involving the formation of at least one voluntary, subscription-based, self-financed coalition of research reactor operators (possibly including other participants, sponsors, etc.), which may serve as a model for the establishment of additional coalitions.

2. Concept Operations and Benefits

The principle objectives of the IAEA in initiating the Research Reactor Coalitions initiative are to promote enhanced utilization of individual facilities and at the same time support the implementation of high standards of nuclear material security and physical protection, safety, and quality assurance.

While different types of coalitions are envisaged, many potential coalitions will coordinate the marketing and sales of services from participating research reactors in order to increase the availability of such services to potential customers, and will encourage/facilitate formation of joint ventures between highly utilized facilities requiring new, lower cost, or regionally sited irradiation capacity with capable but underutilized reactors. In achieving this, it is expected that the partners will:

- Develop and peer review strategic plans of the research reactors involved, both individually and collectively,
- Share market analysis and marketing expertise to support the participating research reactors that currently do not have access to such skills, both for commercial and scientific/research activities,

- Catalogue and publicize the scientific and technical capabilities of the research reactors in the coalition,
- Develop realistic cost estimates and pricing strategies, and carry out collective procurements or negotiations with suppliers to receive cheaper prices, and
- Create economies of scale to give groups of reactors more powerful voices commercially and politically and facilitate both fuel supply and “back-end” solutions.

A coalition of this type may thus resemble joint marketing by small-scale suppliers or one of the airline alliances or similar cooperative marketing arrangements that are formed to grow the market through coordinated services, in the context of meeting high standards of quality and safety. In other ways, this type of coalition will provide some functions similar to a trade association in regard to interacting with national governments and other relevant organizations to represent the collective interests of the coalition.

Coalitions would benefit the participating research reactors, their customers, and the wider community as summarized in Table 1 and described below. They would:

- Optimize the services offered (education and training, production of isotopes, industrial irradiation services such as transmutation doping, neutron activation analysis and other analytical services for industry and government) on a geographical basis, reducing the need for international transport of radioactive materials,
- Make maximum use of expertise or equipment at a particular facilities, and perhaps enable particular facilities to specialize in services in which they a “comparative advantage”, and customers would be able to receive advice regarding the range of facilities, and locations, available from a single point of contact rather than through multiple agreements with different reactors, and
- Use the combined expertise of the participant facilities to best advise and serve their customers. This would help increase customer knowledge of, and access to, the radiation services, and support the customer with a more reliable and comprehensive customer service.

Research reactors that form a coalition would gain from the improved planning and marketing capabilities of the coalition, and sharing of best practices in operations and security. Their customers would benefit from a more homogeneous and sympathetic standard of service. Coalition participants may gain from payments made by countries or institutes that subscribe to the coalition as an alternative to operating their own reactors. Better-utilized facilities that join a coalition could gain from payments to cover professional expertise made available to the coalition.

In cases where existing, well-utilized reactors are experiencing capacity issues, contractual arrangements or joint ventures may be initiated with under-utilized reactors for irradiation services, directly benefiting the under-utilized reactor with commercial revenues and access to expertise, and the well-utilized facility with a resolution to its capacity problems.

As noted, one of the objectives of the IAEA is to contribute to the improvement of research reactor safety and nuclear material security and the physical protection of facilities. As participation in a coalition will be beneficial to the participants and therefore desirable, it provides an opportunity to define minimum standards for participation, and to make access to the coalition conditional upon those standards being maintained. It is thus expected that each coalition will:

- Encourage/incentivise best-practices on research reactor nuclear material security, safety (including application of the Code of Conduct on the Safety of Research Reactors),
- Encourage/reward/provide incentives to and provide assistance for conversion to low enriched uranium (LEU),
- Encourage adoption of a common Quality Assurance/Quality Control standards and implement a system of accreditation (e.g. through inter-comparison exercises),

- Assist with acquisition of external funding for such items as irradiation services, human resource development, including succession planning, and operational experience.

Improving utilization will result in additional commercial revenues and may help to reinforce domestic governmental support, thereby improving sustainability and assisting individual reactors to pay for operational, safety, and security improvements.

Because each coalition will be able to communicate and share best practice in all areas of reactor operation, this will reduce risks from research reactor operation, and help ensure that all appropriate international standards are fully observed. A regular technical and professional interchange would help build confidence and trust in the availability of equipment, facilities and expertise at partner reactors. In certain cases, it could be expected that smaller research facilities would find it more beneficial to have access to superior equipment and expertise at another site, via the coalition, than to maintain independent capabilities possibly not meeting the same standards.

Coalitions would therefore help promote regional and international cooperation by developing the cooperative environment prerequisite to establish centers of excellence and to rationalize research reactor activities. Other coalitions maybe formed specifically to provide shared access to scientific and experimental research, training, and irradiation services to countries without research reactors. Developing countries without a national-based research reactor could thereby access the benefits of peaceful uses of nuclear technology by participating in, and supporting, a research reactor coalition. The shared user facility would benefit by payments made for access or for shared equipment by countries or institutes that subscribe to the coalition as an alternative to operating their own reactors.

Due to the large capital costs, it is expected that future research reactors will more often be constructed as regional or international facilities instead of on a national basis. Further, any technically required research reactor operations involving HEU would eventually be concentrated at a very limited number of highly secure facilities that would also serve as shared-user centers.

The wider community would gain from overall improvements to operational safety practices and the reduced risk of nuclear accidents or incidents.

3. Project Development and Plans

Initial discussions concerning the possibility of formulating a project on Research Reactor Coalitions began on the margins of the RRFM meeting in Sofia in May 2006. A concept paper was drafted, and the IAEA requested NTI in June 2006 to provide seed funding for an initial meeting to further scope the concept.

Subsequently, the IAEA convened a Consultancy Meeting on Developing Proposals for Research Reactor Coalitions and Centres of Excellence” in Vienna from 31 August – 5 September 2006. This meeting reviewed a number of existing international arrangements involving groups of research reactors, discussed the general concept of research reactor coalitions as well as a number of potential subject areas for such work, and reviewed and revised a draft concept paper. This concept paper formed the basis of a grant request submitted by the IAEA to NTI. In October 2006, NTI’s Board approved a grant to the IAEA for a two-year project.

The IAEA views this activity as a continuation and deepening of efforts to further integrate its research reactor activities, particularly through the Cross-Cutting Coordinator for Research Reactors. As such, the NTI grant will be coordinated with other IAEA regular, Technical Cooperation, and extrabudgetary funded activities related to research reactor utilization, safety, security, spent fuel management and the fuel cycle, and non-proliferation. The IAEA aims to assist in generating and coordinating ideas, promoting concepts, providing support for meetings and expert missions. Thus, the IAEA’s role is that of a facilitator and to a smaller degree, business incubator.

Community Benefit	Reactor Operator Benefit	Customer Benefit
<i>Disseminate and Encourage Best Practices</i> • Control and Accounting • Non-proliferation • Nuclear Security/Physical Protection (including conversion to LEU) • Operational Safety • Radiation Safety	<i>Improve Sustainability</i> • Strategic Planning • Business Planning • Facilitate acquisition of new business and/or funding	<i>Better Awareness of Available Capabilities</i> • Customer less reliant on own expertise
<i>Reduce Nuclear Terrorism Risk</i> • Rationalize radioisotope supply geography • Reduce Activities Shipped • Reduce Distances Shipped • Improve nuclear material security • Improve spent fuel management	<i>Increase Market Access for Individual Reactors</i> • Some products/services via the Network • Improve utilization factors	<i>Reduced Costs and Complexity</i> • Rational matching of needs and capabilities/locations • One-stop shop
<i>Build Trust and Confidence in mutual support networks</i> • Promote Regional/International Cooperations • Improve access to the peaceful uses of nuclear technology. Precursor to Centers of Excellence • Additional resources/capabilities • Establish peer group leaders	<i>Increase Professional Opportunities</i> • Closer peer group interaction • Access to equipment and expertise at other facilities • Access to different types of irradiation facility	<i>Improve Service Level</i> • Standardized Quality Assurance • More available facilities • Improved Reliability • Back-up options

Table 1: Summary of Coalition Benefits

Between October and December 2006, the IAEA conducted informal consultations with a number of research reactor operators, commercial entities, research reactor irradiation services users, and other stakeholders. These informal discussions resulted in the development of approximately fifteen “notional proposals” covering a range of subjects for possible coalitions. A weekly conference call was held to execute an action item list designed to advance further development of the notional proposals. Several of the notional proposals were further elaborated in specific papers.

In January 2007, the IAEA held a Consultancy on Project Planning for Research Reactor Coalitions, under Technical Cooperation Project RER/4/029, which reviewed existing research reactor networking arrangements and examined the need for market studies and analyses to support specific coalitions. The meeting also reviewed and prioritized the “notional proposals” and developed a work plan. Preliminary discussions (which will continue on the margins of RRFM/Lyon) have resulted in progress on notional proposals related to:

Africa
East Asia
Europe Radiotracers
Latin America (2), and involving the following topical areas:

Research reactor planning
Production of medical and industrial radioisotopes
Fuel irradiation and testing
Neutron sciences and experimentation

The IAEA plans to issue a circular note to representatives of IAEA Member States inviting research reactor institutions and other related organizations to express interest in participating in a coalition and to provide concrete proposals to the IAEA.

Future meetings will be held in Vienna and at the sites of coalition participants in order to promote detailed discussions between potential coalition members to define specific coalition arrangements and activities. The IAEA will also provide support for administrative and other arrangements for coalition activities, and will provide expertise and assistance in the development of strategic and business plans for the coalition and the participating research reactors and also to develop public information and marketing materials.

4. Conclusion

The international research reactor community needs to be poised to meet arising societal needs, especially to support the anticipated “nuclear renaissance” to satisfy rapidly expanding global energy requirements with carbon-free electricity production and for emerging nuclear medicine technologies, but also for many other applications. This requires the operators of research reactors to be financially secure, operating under the best practices of safety, security and physical protection, consistent with non-proliferation goals, and on the basis of strengthened regional and international cooperation.

It is expected that at least one specific coalition will be announced later this year at the RERTR 2007 international meeting in Prague, Czech Republic and/or the 2007 IAEA research reactor conference in Sydney, Australia in November 2007.

REFURBISHMENT AND ACTIVITIES AT TAJOURA REACTOR

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ABSTRACT

The Tajoura Research Reactor was built in the late seventies by the former Soviet Union for Libya. Its maximum power rating is 10 MW. Its design facilitates the production of radioisotopes and the performance of material testing experiments. The reactor is provided with a critical assembly that is an exact mockup of the reactor core to test and neutronically study the different core configurations. Both of the Critical Assembly and the reactor were recently converted from the HEU fuel (Type IRT-2M) to the LEU fuel (Type IRT-4M).

1. Introduction

Tajoura Renewable Energies and Water Desalination Research Centre (REWDRC) is a national research centre, which provides a program of scientific activities in nuclear science and technology. It is located outside the city of Tajoura, 35 km east of Tripoli. The Tajoura nuclear facility is part of this center and it consists of two installations, the Tajoura Research Reactor and the Critical Facility.

The Tajoura Research Reactor is a 10 MW light water cooled and moderated beryllium reflected, pool type reactor. The reactor was designed and constructed by the former Soviet Union, as a turn key project. The construction of the reactor started in 1977; the power start-up of the reactor took place in 1983.

The reactor is intended to be used in:

1-Carrying out fundamental investigations in

- Nuclear physics
- Solid state physics
- Neutron physics
- Radiation biology
- Radiation chemistry

2- Carrying out the activation analysis of element composition of substances

3- The production of radioactive isotopes.

4- Study the behavior of structural materials directly in the process of irradiation.

The Critical Facility is a complete mockup of the Tajoura Reactor. It was commissioned at the end of 1980. It is used in reactor modeling, testing, training operators, and student education.

This paper concentrates on capabilities of the reactor, Tajoura staff practices related to maintenance and operation of the facilities, and organizational improvements to enhance the safety of the reactor.

2. Reactor irradiation positions and beam tubes

The reactor is equipped with eleven horizontal channels for neutron beams, two of them being the two ends of a through channel with a diameter of 150 mm. The largest channel is a radial channel with a diameter of 230 mm and is intended for radiation biology studies, while the rest are 100 mm diameter radial and tangential channels. During the eighties these beam tubes were utilized by physicists to study the nuclear structure of some elements, and to study the use of local materials in shielding. In the reactor core there are more than 50 vertical irradiation positions in the stationary and removable reflector. With different core configurations it is possible to introduce neutron traps at the center or in the corners of the core with very high thermal neutron flux. For sample transfer from core side to the hot cell the reactor is provided with under water taxi. The reactor is also equipped with a pneumatic rabbit system for short and intermediate half life isotopes for activation analysis measurements.

3. Organization:

REWDRC is under the Bureau of Research and Development. The general organization of the center in relation to the Reactor Section is given in Figure (1)

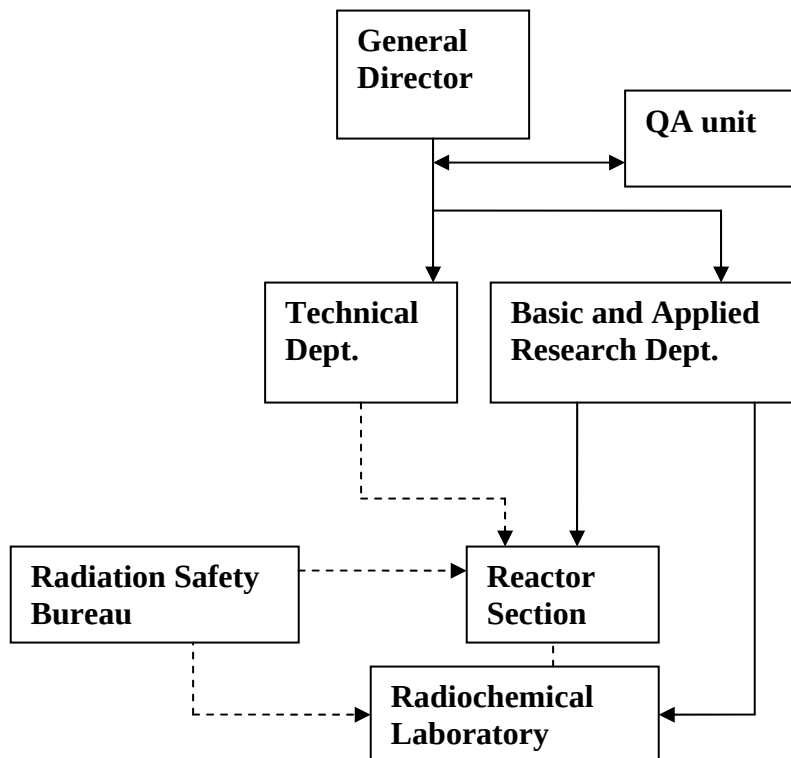


Figure 1: General outline of REWDRC Organization in relation to Reactor Section.

The Reactor Section is part of the Basic and Applied Research Department. The Technical Department provides services to the Reactor Section such as the operation and maintenance of the secondary and third circuit, air conditioning, hot water supply, and air ventilation. The Radiation Safety Office controls all radiation protection matters at the reactor section.

When the reactor was commissioned the law number 2 of the year 1982 concerning the protection against ionizing radiation was already in force. However, a dedicated law for reactor operation and utilization did not exist, and the reactor was operated under the permission of the authority of the Ministry of Atomic Energy. According to this permission the staff of the reactor had to strictly follow the rules and operation procedures set by the reactor provider (these were the rules applicable at the former Soviet Union). When the reactor was commissioned no separate safety analysis report document as it is commonly known today was provided, even though all the essential elements of a safety analysis report were included in various know how and operation manuals of the system. In the year 1997 the IAEA and in accordance to agency standards indicated to the management of the center the need to establish the safety of the reactor by preparing a safety analysis report. Since that time reactor staff started to prepare the most important part of the safety analysis report mainly the accident analysis chapter. In the year 2004 the Regulatory Body (RB) started effectively doing its work related to the Tajoura Reactor and the Critical Facility. The RB adapted the recommendations of IAEA concerning the safety and operation of reactors since conversion of the Tajoura Reactor and the Critical Facility was foreseen at that time. During the years 2005 and 2006 the accident analysis chapter for the Tajoura Reactor and the Critical Facility using the two types of fuel (HEU, LEU) was completed.

4. Reactor utilization

The utilization of the reactor suffered the most due to the hard ship which had confronted the country during the years 1985-2000. Economic hardship, sanctions and trade embargo all have contributed to the low utilization program for the reactor. The utilization was limited to the use of the reactor as an educational tool for university students, for training and retraining of reactor operators and for capacity building in the field of radiation safety, radiation chemistry, isotope production and neutron activation analysis.

In the years 1984-1986 nine different isotopes were produced in the reactor. The radiochemical laboratory at the REWDRC did the work of separating these isotopes. These isotopes were produced to gain experience and for training the personnel. However, Na^{24} was supplied to a local industry, for purposes of evaluating the homogeneity of the production process, while I^{131} (1) was ordered by local hospitals for the diagnosis and treatment of thyroids. $\text{Tc}^{99\text{m}}$ (2,3,4) was produced as a part of capacity building but its use in hospitals was not possible due to the lack of a clean room, which is necessary for producing $\text{Tc}^{99\text{m}}$ suitable for medical applications.

In the years 1987-1999 the production of I^{131} was continued to supply the local hospitals

Br^{82} (5) was also produced as part of an IAEA project to improve reactor utilization in industrial applications.

Number	Isotope	Half life	Target
1	P ³²	14.2 d	P ₂ O ₅
2	Na ²⁴	15 hrs	
3	Au ¹⁹⁸	2.7 d	Pure gold 99.99%
4	K ⁴²	12.36 hrs	K ₂ CO ₃
5	Cr ⁵¹	27.7 d	Enriched chromium metal with Cr ⁵⁰
6	Fe ⁵⁹	44.6 d	Enriched ferric oxide
7	I ¹³¹	8.1 d	TeO ₂
8	Tc ^{99m}	6 hrs	MoO ₃
9	Br ⁸²	35 hrs	KBr

Table 1: Radioactive isotopes produced in Tajoura Reactor

5. Reactor refurbishment

Due to corrosion problems, both fans of cooling tower were out of order. This fact and the lack of spare parts have contributed to the deterioration of the cooling tower which was replaced in the 1998 together with parts of the third circuit pipes.

The reactor control system included computer monitoring system which provided the monitoring of around 100 reactor parameters. The computer was also used to detect failures and provide for the operator an event log. Many of the parameters which were not measured were calculated by the computer using suitable formula. After three years the computer started to have problems due to difficulties in securing spare for the maintenance and due the rapidly changing computer technologies. It was decided to introduce a new system based on desktop computers to replace the old monitoring system. The work was done by the reactor staff. The new monitoring system is capable of monitoring more than 80 reactor parameters and can calculate some parameters which are important for safety.

The instrumentation and control system, which was provided by the supplier of the reactor, was designed and constructed in the seventies. The circuits of I& C system are of low scale integration. Its maintenance is very costly and time consuming because of the size of the system, which is huge and over dimensioned. Also no longer are spare parts available for the maintenance. It was decided to replace the system by a new system incorporating new technologies, which will reduce its size and thus the burden of its maintenance. The work expected to start on refurbishment of the control and safety systems for the reactor and for the critical facility in the near future.

6. Maintenance strategies at the facility:

Since the year 1984 when the facility was completely handed over to the Libyan side, the management has been investing all its efforts to keep both the reactor and the critical facility in an excellent technical state. This was accomplished with low inventory of spare parts and decreasing resources during the time economic hardship in late eighties and the sanctions during the nineties. Thanks are due to the maintenance program which concentrated on:

- ✓ Applying a strict control of water quality in all closed circuits to keep the conductivity in the primary circuit of the reactor below the recommended limits (<1 µs/cm) and the pH between 5.5 and 6.0.

- ✓ The conductivity in the secondary circuit was kept below 10 $\mu\text{S}/\text{cm}$ and pH between 6.0 and 8.0. In the critical facility potassium bichromate was added to its pool water as a corrosion inhibitor
- ✓ The continues maintenance of the mechanical filters responsible for air quality Control to insure the removal of fine sand particles which are a characteristic of the area.
- ✓ The operation of the primary circuit, the secondary circuit and the purification system at least twice a week when the reactor is not operated to reduce corrosion risk, and keep the circuits in working condition.
- ✓ The adoption of predictive maintenance instead of periodical maintenance without jeopardizing the safety to reduce the need for spare parts proved its effectiveness in situations through which our reactor was subjected to.

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IRRADIATION FACILITIES AT THE ADVANCED TEST REACTOR

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ABSTRACT

The Advanced Test Reactor (ATR) is one of the world's premiere test reactors for performing long term, high flux, and/or large volume irradiation test programs. The ATR is a very versatile facility with a wide variety of experimental test capabilities for providing the environment needed in an irradiation experiment. These different capabilities include passive sealed capsule experiments, instrumented and/or temperature-controlled experiments, and pressurized water loop experiment facilities. Monitoring systems have also been utilized on the exhaust gas lines from instrumented temperature-controlled experiments to monitor different parameters, such as fission gases for fuel experiments, during irradiation. The ATR irradiation positions vary in diameter from 1.6 cm (0.625 inches) to 12.7 cm (5.0 inches) over an active core length of 122 cm (48.0 inches). This paper discusses the different irradiation capabilities available and the cost/benefit issues related to each capability.

1. Introduction

The Advanced Test Reactor (ATR), a light water moderated, beryllium-reflected pressurized water reactor, located at the Idaho National Laboratory (INL) is a valuable resource available for use in developing the materials and fuels necessary to support the next generation reactors and advanced fuel cycles. The ATR has a long history of irradiation testing in support of reactor development and the INL has been designated as the United States Department of Energy's lead laboratory for nuclear energy development. The ATR reactor vessel is constructed of solid stainless steel and is located far enough away from the active core that neutron embrittlement of the vessel is not a concern. In addition, the ATR core is completely replaced every 7 to 10 years, with the last change having been completed in January 2005. These two major factors, combined with a very proactive maintenance and plant equipment replacement program, have resulted in the ATR operational life being essentially unlimited. The ATR has a maximum power of 250 MW and can provide maximum thermal neutron fluxes of $1\text{E}15$ neutrons/cm²-second and maximum fast ($E>1.0$ MeV) neutron fluxes of $5\text{E}14$ neutrons/cm²-second. This allows considerable acceleration of accumulated neutron fluence to materials and fuels over what would be seen in a typical power reactor. These fluxes combined with the 77 irradiation positions varying in diameter from 16 mm (0.625 inches) to 127 mm (5.0 inches) over an active core height of 1.2 m (48.0 inches) make ATR a very versatile and unique facility.

The ATR core cross section, shown in Figure 1, consists of 40 curved fuel elements configured in a serpentine arrangement around a 3 by 3 array of prime irradiation locations in the core termed flux traps. The flux traps derive their name from the high-intensity neutron flux that is concentrated in them due to the close proximity of the fuel and the materials used in these "traps". The ATR's unique horizontal rotating control drum system (termed outer shim control cylinders) provides stable axial/vertical flux profiles for experiments throughout each reactor operating cycle unperturbed by the typical vertically positioned control components. This stable axial flux profile, with the peak flux at the centre of the core, allows experimenters to have specimens positioned in the core to receive different known neutron fluences during the same irradiation periods over the duration of test programs requiring several years of irradiation. This system also allows the reactor to operate different sections of the core at different power levels. The ATR core is divided into five different operating lobes: the four corner lobes and the centre lobe. Each lobe of the reactor may be operated at a different power level (within specific limitations) during each reactor cycle.

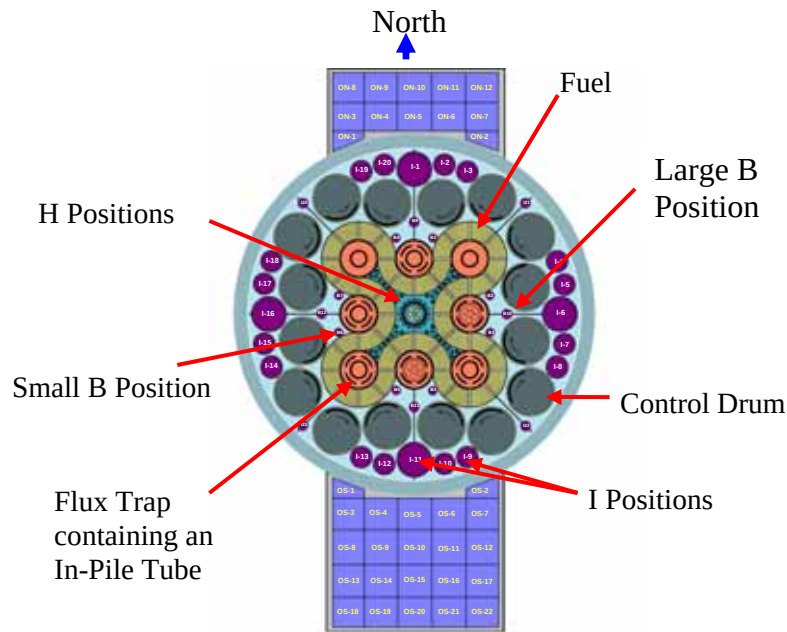


Figure 1 - ATR Core Cross Section

2. Experiment Types

Three major types of irradiation testing are employed in the ATR. The simplest and least expensive type is a static sealed capsule with only passive instrumentation. The next level of complexity in testing includes active instrumentation for measurement and/or control of specific testing parameters, typically temperature and/or pressure. The last and most complex method is the pressurized water loops that are connected to in-pile tubes located in the flux traps. Each of these irradiation types and their relative cost, schedule and operation differences are discussed in detail in the following sections.

2.1. Static Capsule Experiments

Static capsules experiments are self-contained (typically) sealed experiment encapsulations surrounding the irradiation specimens with an inert gas environment. However, occasionally the capsules are not sealed but allow the experiment specimens to be in contact with the reactor primary coolant to prevent excessive temperatures during irradiation. These capsules typically include passive instrumentation such as flux wires for neutron fluence monitoring and/or melt wires for temperature monitoring during irradiation. In addition, the temperature of a static capsule may also be controlled, within limits, by incorporating a small insulating gas jacket (filled with an inert gas) between the specimens and the outside capsule wall or pressure boundary. A suitable gas jacket width can usually be selected to provide the irradiation temperature desired by the experiment customer based upon the gamma and reaction heating characteristics of the specimens and capsule materials and proper selection of the insulating gas.

The static capsules may vary in length from several centimetres to full core height of 1.2 meters. They also may vary in diameter from 12-mm or possibly less for the small irradiation positions (or a portion of an irradiation position) to more than 120-mm for the larger irradiation positions. The capsules are typically constructed of aluminium or stainless steel, but zircaloy has also been utilized. Depending upon the contents and pressure of the capsule, a secondary containment may be included to meet the ATR safety requirements. The capsules are usually contained in an irradiation basket, which radially locates the capsules in the irradiation position and vertically positions them within the ATR core. Occasionally due to space limitations, a static capsule has been used to also serve the function of the basket, but in these cases, the capsules must fill the entire irradiation height and have a similar handling feature at the top of the capsule for installation and removal from the core.

The benefits of utilizing static capsules for irradiation testing include the ease of removal from and replacement into the reactor vessel to support specimen or capsule replacements or to avoid one of ATR's short high power cycles. This ease of removal and replacement can also be utilized to relocate fueled capsule experiments to a higher power location to compensate for fuel burn-up. This type of testing is also less expensive than the other types of irradiation testing and due to its simplicity; it requires the least amount of time to get specimens into the reactor. However, static capsule testing has less flexibility and control of operating parameters (such as specimen temperatures) during the irradiation and greater reliance is made on the design analyses since passive instrumentation can only provide snap shot values of the operating parameters during irradiation (i.e. a melt wire can provide the maximum temperature attained during an irradiation but not the amount of time or when the maximum temperature was achieved).

2.2. Instrumented Lead Experiments

The next level of complexity in testing incorporates active instrumentation for continuous monitoring and control of certain experiment parameters during irradiation. These actively monitored and controlled experiments are commonly referred to as instrumented lead experiments, deriving their name from the active instrument leads (such as thermocouples or pressure taps) that they contain. An instrumented lead experiment containment is very similar to a static capsule, with the major difference being an umbilical tube connecting the experiment to a control system outside of the reactor vessel. The umbilical tube is used to house the instrument leads (thermocouples, pressure taps, etc.) and temperature control gas lines from the irradiation position within the reactor core to the reactor vessel wall. The instrument leads and gas lines are then routed outside the reactor vessel to the control and data collection/monitoring equipment. An instrumented lead experiment may contain several vertically stacked capsules, and is specifically designed to meet the experimenter's needs. This is accomplished by selecting a suitable irradiation position, which will provide the necessary gamma and/or reaction heating as well as the total neutron fluence within the available schedule, and then designing the umbilical tube routing necessary to connect the experiment to the reactor vessel wall.

The most common parameter to be monitored and controlled in an instrumented lead experiment during irradiation is the specimen temperature. The temperature of each experiment capsule is independently controlled by varying the thermal conductivity of a gas mixture in a very small insulating gas jacket between the specimens and the experiment containment. This is accomplished by blending a conductor gas with an insulator gas. Helium is used as the conductor gas and neon is typically used as the insulator gas. However argon has also been used as an insulator gas (with helium as the conductor) when a larger temperature control band is needed and the activity from the by-product Ar-41 does not affect the experiment data collection (i.e. monitoring of the experiment temperature control exhaust gas for fission gases, etc.). During normal operation, the gases are blended automatically to control the specimen capsule temperature based upon feedback from the thermocouples. The computer controlled gas blending system permits a blend range of 98% of one gas to 2% of the other to maximize the temperature control range for the experiments.

Temperature measurements are typically taken with at least two thermocouples per capsule to provide assurance against an errant thermocouple and to also provide redundancy in the event of a thermocouple failure. The control system also provides automatic gas verification to assure the correct gas is connected to the supply ports in the system to prevent an uncontrollable temperature excursion resulting from a gas supply mix-up (i.e. insulator gas connected to a conductor gas port or vice versa). Monitoring of the temperature control exhaust gas is quite common to sense for different materials as a measure of the experiment performance or conditions. There are several options available for monitoring that have been employed on previous experiments conducted in the ATR. The most common monitoring has been for fission gases in fueled experiments to monitor fuel performance during irradiation. However, other monitors have also been utilized such as a gas chromatograph to monitor for chemical changes in an experiment cover gas due to oxidation of the specimens, and monitoring for supplemental gases to detect leakage through a test barrier during irradiation. Alarm functions are provided to call attention to circumstances such as temperature excursions or valve position errors. Helium purges to each individual specimen capsules are automatically actuated in the

unlikely event of the ability to measure or control the temperature is lost. In order to minimize response time between a gas mixture change and a change in temperature in the experiment specimens, the gas system maintains a continuous flow to the experiment through very small internal diameter tubing. Manual control capability is provided at the gas blending panels to provide a helium purge of the experiment capsules in the event of a computer failure. Data acquisition and archive are also included as part of the control system function. Real time displays of all temperatures, gas mixtures, and alarm conditions are provided at the operator control station. All data are archived to removable media, with the data being time stamped and recorded once every ten minutes to as often as once every ten seconds. The control processor will record these values in a circular first-in, first-out format for at least six months.

The benefits of performing an instrumented lead experiment are more precise monitoring and control of the experiment parameters during irradiation as well as monitoring the temperature control exhaust gas to establish specimen performance during the irradiation. However, this type of experiment has the detriments of higher total experiment costs and a longer lead time to get an experiment into the reactor than a static capsule. There are also higher costs and risks associated with removal and re-installation of an instrumented lead experiment in the reactor for specimen replacements or to avoid a short high power ATR operating cycle.

2.3. Pressurized Water Loops

Five of the ATR flux traps contain In-Pile Tubes (IPT), which are connected to pressurized water loops. The other four flux trap positions currently contain capsule irradiation facilities, and have also contained the ITV as mentioned above. An IPT is the reactor in-vessel component of a pressurized water loop, and it provides a barrier between the reactor coolant system water and the pressurized water loop coolant. Although the experiment is isolated from the reactor coolant system by the IPT, the test specimens within the IPT are still subjected to the high intensity neutron and gamma flux environment of the reactor. The IPT extends completely through the reactor vessel with closure plugs and seals at the reactor's top and bottom heads. This allows the top seals to be opened and each experiment to be independently inserted or removed. The experiments are suspended from the top closure plugs using a hanger rod. The hanger rod vertically positions the experiment within the reactor core and provides a pathway for test instrumentation. Anything from scaled-down reactor fuel rod bundles to core structural materials can be irradiated in these pressurized water loops. Each IPT is connected to a separate pressurized water loop, which allows material or fuel testing at different pressures, temperatures, flow rates, water chemistry, and neutron flux (dependent of the location within the ATR core) with only one reactor.

The loops are connected to a state-of-the-art computer control system. This system controls, monitors, and provides emergency functions and alarms for each loop. The experiment designers, though constrained by ATR's unique operating and safety requirements, are free to develop a test with specific operating conditions within the space and operating envelope created by the IPT and loop. A loop experiment can contain a variety of instrumentation including flow, temperature, fluence, pressure, differential pressure, fission product monitoring, and water chemistry. All of these parameters can be monitored by the Loop Operating Control System (LOCS) and controlled by the LOCS reactor control system, or by operator intervention. The LOCS is a state-of-the-art computer system designed specifically for the ATR loops. The system controls all aspects of loop operations (flow, pressure, and temperature) for all five loops simultaneously. This information is displayed on the Loop Operating Console and interfaces with the reactor control system. Loop Operators are stationed at the controls to operate and monitor the systems to meet the experiment sponsors requirements. Typical operations include setting, monitoring and maintaining flow rates, temperatures, pressure, and water chemistry.

There are two Powered Axial Locator Mechanism (PALM) drive units that can be connected to specially configured tests in the loop facilities so that complex transient testing can be performed. The PALM drive units move a small test section from above the reactor core region into the core region and back out again either very quickly, approximately 2 seconds, or slowly depending on test requirements. This process simulates multiple start-up and shutdown cycles of test fuels and materials.

Thousands of cycles can be simulated during a normal ATR operating cycle. The PALM drive units are also used to precisely position a test within the neutron flux of the reactor and change this position slightly as the reactor fuel burns.

The benefits of performing a pressurized water loop experiment are (as with the instrumented lead experiments) more precise monitoring and control of the experiment parameters during irradiation as well as monitoring the loop water chemistry to establish specimen performance during the irradiation. However, this type of experiment has the detriments of the highest total experiment costs and the longest lead time to get an experiment into the reactor.

2.4. New Gas Test Loop

A new Gas Test Loop (GTL) for ATR is in the conceptual design phase, and therefore concepts to be developed in later design phases of the system are being identified. The current configuration is planned for installation in one of the large flux trap positions (e.g. NE or NW) to maximize the flux rates available to experimenters. In order to achieve the high fast flux rate goals of the GTL (by minimizing the moderation effects of the coolant system on the neutron spectrum within the GTL facility), a large forced convection gas heat transfer system is needed for cooling of the GTL facility. Helium is the coolant under consideration for the large forced convection system. The existing gas testing facilities at ATR utilize either no (static capsule) or very low (lead experiments – 50 cc/min) temperature control gas flows, and therefore rely mainly on conduction but may also include radiation heat transfer mechanisms. Several irradiation positions (or MIPTs) are planned within the new GTL flux trap, and the current gas conduction/radiation heat transfer system is planned for use within the MIPTs for final temperature adjustment of the irradiation specimens. In addition to use of a flux trap position, the concept also includes fast flux boosting by including additional fuel around the outside of the test positions. A configuration has been proposed for the additional booster fuel and development and testing of the booster fuel is currently being pursued. Since the booster fuel is the main driver in the design of the GTL, the final design of the loop is dependent upon successful completion of the booster fuel testing.

3. Conclusion

The ATR has a long history in fuel and material irradiations, and will be fulfilling a critical role in the future fuel and material testing necessary to develop the next generation reactor systems and advanced fuel cycles. The capabilities and experience at the ATR, as well as the other test reactors throughout the world, will be vitally important for the development of these new systems to provide the world with clean safe energy supplies in the future.

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Current and prospective fuel test programmes in the MIR reactor

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ABSTRACT

The MIR reactor is mainly designed for testing fragments of fuel elements and fuel assemblies (FA) of different nuclear power reactor types under normal (stationary and transient) operating conditions as well as emergency ones in a certain project. At present six test loop facilities are being operated (2 PWR loops, 2 BWR loops and 2 steam coolant loops). The majority of current fuel tests is conducted for improving and upgrading the Russian PWR fuel, such as: long term tests of short-size rods with different modifications of cladding materials and fuel pellets; further irradiation of NPP refabricated and full-size fuel rods up to achieving 80 MW·d/kg U; experiments with leaking fuel rods at different burn-up and under transient conditions; continuation of the RAMP type experiments at high burn-up of fuel; in-pile tests with simulation of LOCA and RIA type accidents. Testing of the LEU research reactor fuel is conducted within the framework of the RERTR programme. Upgrading of gas cooled and steam cooled loop facilities is scheduled for testing the HTGR fuel and sub-critical water-cooled reactor, correspondingly. The present paper describes the major programs of the WWER high burn-up fuel behavior study in the MIR reactor, capabilities of the applied techniques and some results of the performed irradiation tests.

1. Introduction

The MIR reactor is a heterogeneous thermal reactor with a moderator and a reflector made of metal beryllium [1]. It has a channel-type design and is placed in the water pool. The frame of the core is made up of hexagonal beryllium blocks with width across flats of 148,5. In the central axis holes of the blocks channel bodies are installed for operating FAs (37 pcs); combined operating FA with absorber (12 pcs); experimental loop channels (11 pcs). The maximum diameter of experimental channels is up to 148 mm, height of core 1000 mm.

At present 6 loop facilities (PV-1, PVK-1, PV-2, PVK-2, PVP-1, PVP-2) are being in operation and 2 facilities (PG, PM) have not been used for the last 15 years (table 1).

No	Parameters, unit	Loop facilities							
		PV-1	PVK-1	PV -2	PVK -2	PVP-1	PVP-2	PG	PM
1.	Coolant	water	boiling water	water	boiling water	water, steam	water, steam	nitrogen, helium	heavy metal
2.	Number of test channels	2	2	2	2	1	1	1	1
3.	Maximum channel power, kW	1500	1500	1500	1500	100	2000	160	500
4.	Maximum coolant temperature, C°: -in outlet of channel, -in outlet of device	350 350	350 350	350 350	365 365	500 500	550 1100	500 1000	550 550
5.	Maximum pressure, MPa	17,0	17,0	18,0	18,0	8,5	15,0	20,0	1,7
6.	Maximum coolant flow rate through the channel, m ³ /h	16,0	16,0	13,0	13,0	0,6	10,0		5,0

Table 1. Key parameters of loop facilities

Water and boiling water high-temperature loop facilities provide necessary coolant parameters for WWER fuel testing. Lay-out of control rods of the reactor and loop facilities in the core makes it possible to perform several testing programs simultaneously at different values of neutron flux density in the loop channel (they differ by a factor of 5 to 10). A high neutron flux density (up to $\sim 5 \cdot 10^{18} \text{ m}^{-2} \cdot \text{s}^{-1}$) allows repeated irradiation of standard or experimental fuel rods from the WWER fuel assemblies up to a burnup of $\sim 80 \text{ MWd/kgU}$ and higher. The main purpose of loop testing is experimental examinations of fuel rod new modifications serviceability and reliability at different normal and accidental operating conditions. These operating conditions include in particular the following: long-term operation under nominal parameters with allowance for tolerance; daily power cycling with a fast power change (power ramping); design-basis accidents followed by heat-transfer drop (coolant loss, burn-out), positive reactivity insertion and operation with leaking fuel rods.

The presented in this paper programs and techniques for in-pile examination of the WWER fuel are aimed at obtaining experimental data that are necessary to provide conformity of the WWER fuel with licensing requirements such as: total pressure of helium and fission gas under the cladding; plastic strain of the cladding as a result of its interaction with fuel; temperature, strain and integrity of claddings in case of design-basis accident with loss of coolant (LOCA); local depth of cladding oxidation; value of fuel enthalpy under design-basis reactivity increase accident (RIA); permissible number of leaking fuel rods in the core and others.

2. Experimental techniques for WWER fuel testing in the MIR reactor

Comparison of the WWER fuel operating conditions with characteristics of the MIR water-coolant loop facilities (table 2) testify their conformity.

Parameter	WWER	MIR
Maximum LP, kW/m	≤ 44.7	Higher values are possible
Pressure, MPa	≤ 17.7	≤ 18.0
Maximum coolant temperature inlet / outlet, °C	290 / 340	325 / 350
Coolant-chemical conditions Boric acid concentration, g/kg	Ammonia-boric-potassium Up to 10	Provided Up to 10*
Coolant velocity, m/s	5.7	Provided
Burn-up, MWd/kgU	~ 55	Up to 100
Start time of fuel rod leaking	Impossible	Possible
Increase of liner power	Impossible	Possible
Intermediate control of fuel rod status	NA	Possible in the pool and shielded hot cell
Control and change of water chemistry	NA	Possible

Table 2. The WWER fuel operating conditions and characteristics of the MIR loop facilities

Several types of irradiation devices have been designed for testing of the WWER-type fuel rods [2]:

- the module type, dismountable device for testing short-size ($\leq 250 \text{ mm}$) fuel rods, up to 4 such rigs can be installed one over another in the loop channel;
- dismountable and instrumented device for testing fuel rods $\sim 1000 \text{ mm}$, containing up to 19 fuel rods;
- device for combined irradiation of non-instrumented refabricated ($\leq 1000 \text{ mm}$) and full-size fuel rods ($\leq 3500 \text{ mm}$) taken from spent NPP with WWER fuel assemblies;
- device for tests of instrumented refabricated fuel rods ($\leq 1000 \text{ mm}$) and full-size fuel rods ($\leq 3500 \text{ mm}$);
- dismountable devices for power cycling and RAMP experiments of instrumented fuel rods by displacement or rotation of the absorbing screens in the experimental channel;

- instrumented devices for simulation of RIA and LOCA conditions (fuel rod drying and overheating);
- devices and equipment for leaking fuel rods testing.

Types and characteristics of instrumentation for in-pile measurements of coolant, cladding and fuel pellet temperatures; fuel rods elongation, change of cladding diameter; gas pressure inside fuel rods, neutron flux and stem content in coolant are given in table 3.

Parameter	Transducer	Measurement range	Measuring error	Sensor dimensions, mm	
				Diameter	Length
Coolant (T_c) and cladding temperature (T_{cl})	Chromel-alumel thermocouple	up to 1100 °C	0.75%	0.5	
Fuel pellet temperature (T_f)	Chromel-alumel thermoprobe	up to 1100 °C	0.75%	1...1.5	
	W-Re thermoprobe	up to 2300 °C	~ 1.5%	1.2...2	
Cladding elongation (δL)	Liner differential inductosyn transducer (LDIT)	(0...5) mm	$\pm 30\mu\text{m}$	16	80
Diameter change (δD)	LDIT	(0...200) μm	$\pm 2\mu\text{m}$	16	80
Gas pressure inside of fuel rod (P_f)	Bellows rolling diaphragm + LDDT	(0...20) MPa	~ 1.5 %	16	80
Neutron flux (F)	Rh-, V-, Hf - direct-charge detector	$10^{15} \dots 10^{19} \text{ m}^{-2}\text{s}^{-1}$	~ 1%	2...4	50...100
Volume steam content in coolant (β)	Cable-type resistivity sensor	20...100%	10%	1.5	

Table 3. Characteristics of instrumentation for in-pile measurements

3. The program and main results of WWER fuel testing in the MIR reactor

3.1. Irradiation of refabricated and full-size WWER fuel rods

The test objective is to investigate the behavior of fuel under higher burn-up and to achieve higher burn-up for preparation of RAMP, LOCA and RIA tests (table 4).

Type of fuel rod	Number of fuel rods	Length of fuel rods, m	Initial burnup, MWd/kgU	Final burnup, MWd/kgU	Liner power, kW/m
WWER-1000	2	3.53	49...50	62...63	18...30
WWER-1000	1	0.95	49	63	19...31
WWER-440	2	2.42	61	72	17...28
WWER-440	1	0.94	60	72	19...31
WWER-1000	5	3.53	53...55	74...75	18...24
WWER-1000	3	0.4	53...58	74...78	18...24

Table 4. General data on irradiation of the WWER refabricated and full-size fuel rods

3.2. Testing under power ramping conditions

By now 14 RAMP tests with the WWER fuel rods have been performed in the MIR reactor. Experimental fuel rods of different modifications, as well as full-size and refabricated fuel rods were tested at burn-up values from ~ 10 MWd/kgU up to ~ 70 MWd/kgU. In figure 1 are illustrated the main results of experiments - range of liner power (LP) changing and state of cladding after power ramp.

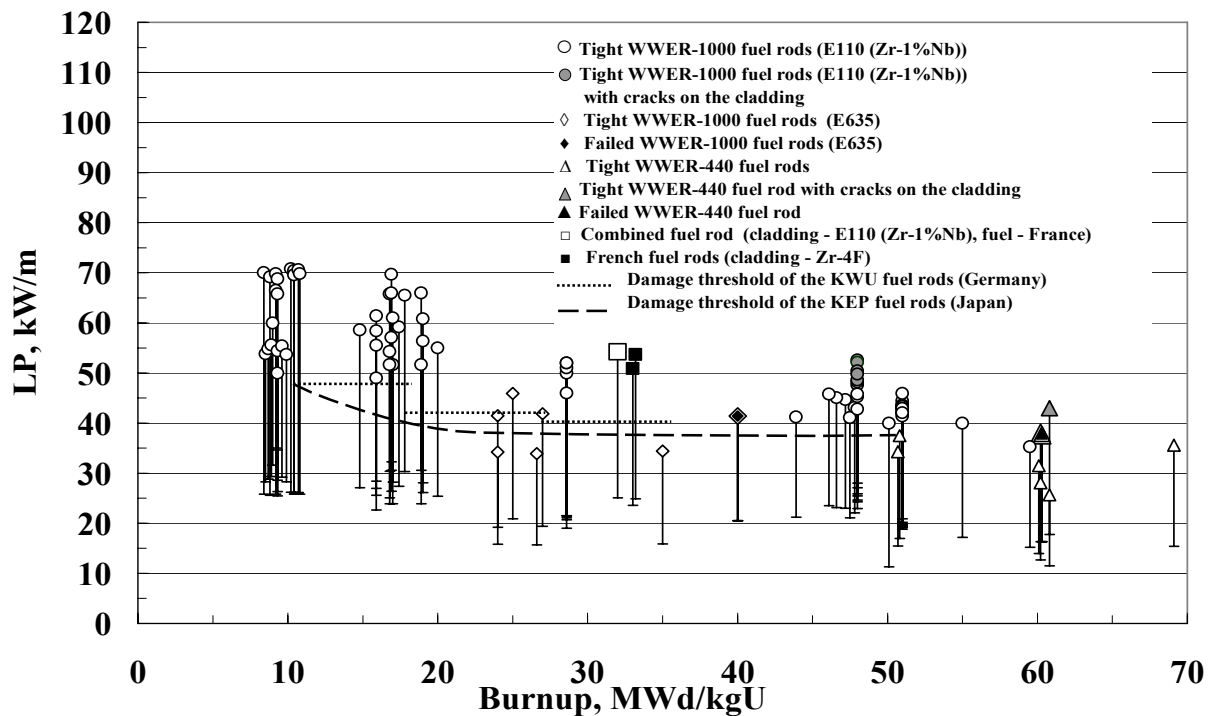


Figure 1. RAMP tests liner power amplitudes versus WWER fuel rods burn-up

In 2008 it is planned to finish RAMP experimental program for WWER-1000 fuel with high burn-up ~ 80 MWd/kgU.

3.3. Testing under power cycling conditions

The objective of testing is to obtain experimental data that characterize a change in the cladding strain, gas pressure in the free volume of a fuel rod, fuel temperature in course of daily power cycling. The fuel rod power changed within (20...30) minutes, exposure at the stable power level makes up ~ 6 hours. Data on tests are presented in table 5.

Type of fuel rod	Number of fuel rods	Instrumentation	Burnup, MWd/kgU	Initial LP, kW/m	LP increase step, kW/m	LP increase rate, kW/m/min
WWER-440	1	$P_f, \delta L, \delta D$	51	19	10	0.3
WWER-440	5	T_f	51...60	15...19	8...10	~ 0.3
WWER-440	4	T_f	52...61	18	11	~ 0.9
WWER-1000	2	T_f, L	49...50	21; 21*	9; 21*	0.6; 0.9*
WWER-1000	2	$P_f, \delta L$	49...50	21	9	0.6

Table 5. The main data of power cycling tests

Power cycling tests will be continued for WWER-1000 fuel rods with burn-up ~ 60 MWd/kgU and higher in 2007-2008.

3.4. Testing under fuel rod drying, overheating and reflooding conditions (LOCA)

A series of tests was performed with the WWER-440 and WWER-1000 multi-element fuel assembly fragments under different phases of design-basis LOCA conditions [3]. The objective of the tests is to verify or refine serviceability criteria of fuel rods and fuel assemblies, determine ultimate parameters, which allow disassembling of the core after operation under deteriorated heat transfer conditions, and to obtain data for code verification and improvement. The main parameters of experiments are given in

Experiment	Number of fresh fuel rods	Number/ burn-up, of irradiated fuel rods, MWd/kgU	Pressure in loop, MPa	Implemented temperature range, °C	Instrumentation	Fuel rod status	
						Non-failed	Failed
SL-1	18	-	12	530...950	$T_c, T_{cl}, T_f, F, \beta$	+	
SL-2	19	-	12	Up to 1200	-//-		+
SL-5	6	1/52	4.9	750...1250	-//-		+
SL-5P	6	1/49	6	700...930	-//-	+	
SL-3	19	-	4	650...730	T_c, T_{cl}, T_f, F, P_f	+	
LL-1	19	3/50	4	550...850	-//-	+	

Table 8. The main parameters of LOCA experiments

LOCA experiments will be continued for WWER-1000 fuel rods with burn-up ~ 60 MWd/kgU and higher in 2007-2008.

3.4. Testing of the WWER-1000 high burn-up fuel rods under design-basis RIA conditions

Calculation data show that the WWER-1000 reactor parameters of the design-basis RIA conditions are as follows: power ratio in impulse ~ 2 , half-width of impulse – (2...2.5)s, power rise duration ~ 1 s. The program and techniques were developed for tests performed in the MIR loop facilities to obtain experimental data on behavior of high burn-up fuel rods under the above-mentioned conditions [4]. In the MIR loop channel it is possible for high burn-up fuel to provide a rising of liner power in impulse up to ~ 4.0 times and to control power rise duration from ~ 0.5 s and more. In 2006 was started experimental program and were provided 2 experiments for WWER-1000 fuel rods with burn-up ~ 50 MWd/kgU, in 2007-2008 the program will be continued.

3.5. Leaking high burn-up fuel rods testing

Taking into account the state of the WWER fuel rods with a burn-up of above $\sim (45...50)$ MWd/kgU, new experimental data are necessary for the development and verification of computer codes, validation of safe operation criteria for WWER reactors in case of leaking fuel rods appearance, as well as for prediction of a change in their state and radiation situation. For this purpose, a testing program was developed and a series of tests is being prepared now to be performed in the MIR loop facilities with refabricated fuel rods claddings some of which have artificial defects. In 2006 first experiment was conducted, in 2007-2008 the program will be continued.

4. Conclusion

Several types of irradiation devices have been designed for testing WWER-type fuel rods under steady state parameters; daily power cycling with a fast power change (power ramping); design-basis accidents have been developed. The current fuel tests program aimed at improving the Russian operating WWER-440 and WWER-1000 fuel should be finished in the MIR reactor in 2008.

At present prospective program of fuel testing for evolutionary design of WWER with improved economics and safety (project AES-2006) is being created. The testing program of upgrading fuel AES-2006 reactors will start in 2008.

In the MIR reactor will be continued testing of the LEU research reactor fuel within the framework of the RERTR program, and in March 2007 will be started testing of 4 full-scale IRT-4 type fuel assemblies.

Upgrading of gas cooled PG-1 loop with increasing coolant outlet temperature up to 1100 °C for in-pile investigations HTGR fuel and steam cooled PVP-2 loop with increasing the pressure up to 22.5 MPa for testing fuel and constructive materials sub-critical water-cooled reactor are scheduled.

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Session II

New Projects and Upgrades

Status of the High Flux Isotope Reactor and the Reactor Scientific Upgrades Program

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ABSTRACT

This manuscript provides a summary of the status of the Scientific Upgrades Program for the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory and the plans for restart of the reactor. Pictures of guides and shield tunnels are provided along with a schedule for completion of the various upgrade projects. Information on preliminary plans for additional future upgrades to facilities at HFIR is also presented.

1. Introduction

As reported in IGORR papers in 2003 and 2005, a program was initiated in 1998 to significantly improve the neutron beam scientific capabilities for all four neutron beams at the Oak Ridge National Laboratory (ORNL) High Flux Isotope Reactor (HFIR). The upgrades to the HB-1, HB-2, and HB-3 beam lines were completed in 2005, as previously reported. This paper will focus on the status of the upgrades to the HB-4 beam line and plans for completion of the upgrades this year.

2. Description of HB-4 Beamline Upgrades

One of the major aspects of the HB-4 beam line upgrade has been the installation of supercritical hydrogen cold source in the HB-4 beam tube. The installation of this cold source has been completed and is presently undergoing testing. This paper will focus on the neutron guides, shielding, Guidehall, and new instruments that have been installed on the guides. The cold source is not addressed by this paper since there is a separate paper by Ken Morgan in the conference publication that focuses on the HFIR cold source.

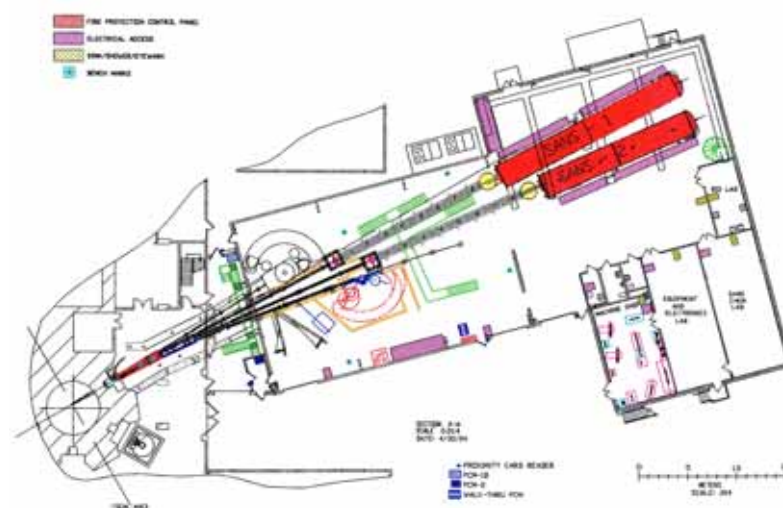


Figure 1: HB-4 Guide and Instrument Layout

Figure 1 provides a layout of the HB-4 beamline guides and instrument equipment. The system is composed of four neutron guides that transport neutrons through the reactor beam room and Transition Building to the HB-4 Guidehall. The guides have beam access locations that will support seven instruments. At this time two of the seven instruments have been installed. More detailed information on these instruments will be provided later in this paper.

2.1 HB-4 Shielding Structures

The discussion of the shielding for the HB-4 beams can be broken into three areas: Reactor Beam Room shielding, Transition Building shielding, and shielding in the Guidehall. Issues addressed in the shielding designs include personnel protection as well as instrument background issues. A very detailed commissioning plan has been developed to confirm the shielding effectiveness as part of the initial reactor startup after the cold source testing has been completed.



Figure 2: Open Shield Tunnel Door in Beam Room

The primary shielding around the guides in the reactor beam room is provided by approximately 32 inches (~81 cm) of a high density hematite concrete (~ 4/cc). A large shield door (shown in Figure 2) provides access to the neutron guides and also provides the access required to replace the beam tube as required. This door is a steel frame filled with the hematite concrete and weights over 60,000 lbs (27,000 Kg). The door is hinged to provide easy access and is locked in place when open.

The guides pass through a 32 inch (~81 cm) hematite concrete shield bulkhead between the Reactor Beam Room and the Transition Building. Since the guides are either curved or have a mirror offset prior to this bulkhead, there is no line-of-sight from the Transition Building back to the reactor beam tube. This is expected to greatly reduce the fast neutron and gamma source that the shielding must deal with in the Transition Building and the Guidehall. Shielding in the Transition Building is provided by 20 inch (~ 51 cm) thick ordinary concrete wall and roof sections. The middle wall and roof sections are removable to provide access to guides in this area.

The guides pass through a 16 inch (~41 cm) hematite concrete second shield bulkhead between the Transition Building and the Guidehall. This provides additional help in reducing background in the Guidehall. The primary shielding in the Guidehall is provided by large eight inch (~20 cm) thick wall and roof shield blocks constructed from Barytes Concrete. Each wall and roof section is removable to provide access to the guides. Ports in the west shield wall are provided for each instrument location on CG-4. The Guidehall shielding is shown in Figure 3.



Figure 3: Guidehall Shielding Layout

The blue box structures in the foreground on CG-2 and CG-3 are the shielding boxes for the CG-2 and CG-3 velocity selectors respectively. These shielding assemblies are shown with a side wall and the roof sections removed and are primarily steel frames filled with lead to deal with the significant gamma source produced in the gadolinium blades of the velocity selector.

2.2 Neutron Guides

The initial installation of the guides was initiated in January of 2005 and completed in the spring of 2006. The guides in the reactor beam room are installed inside vacuum vessels referred to as common casings to avoid having a pressure load on the guide glass where the irradiation damage on the glass will be at its highest. It is believed that these common casings can be removed and reinstalled as a unit with minimal affect on guide alignment. The common casings were removed from the Reactor Beam Room during the summer of 2006 to make room for cold source testing equipment. The plan is to reinstall the common casings in March of this year.

All neutron guides in the Transition Building and the Guidehall have been installed and their transmission will be tested as part of the initial HB-4 cold beam commissioning activities. Figure 4 shows a picture of the guides inside the beam room shield tunnel prior to the installation of the shield tunnel roof.



Figure 4: Guides installed in HB-4 Guidehall

2.3 HB-4 Guidehall and Instruments

As seen from Figure 1, we expect to have seven neutron instruments supported by the four neutron guides in the Guidehall. The show case instruments will be the two Small Angle Neutron Scattering instruments that are located on CG-2 and CG-3. These two instruments are essentially installed and awaiting neutrons. Figure 5 shows the SANS-1 and SANS-2 flight tubes in the Guidehall. Vacuum testing of the vessels has been completed and the tanks have been lined with cadmium to minimize neutron background inside the flight tube. A very large 1 m x 1m neutron detector with 5 mm x 5 mm resolution has been installed on a precision rail system in each of the SANS flight tubes. Velocity selectors on each SANS line will allow the selection of neutron beam energy for a given experiment.



Figure 5: SANS-1 and SANS-2 Flight Tubes

A Cold Triple Axis Instrument is in the process of being built for the third position on the CG-4 line. Installation of portions of this instrument is expected to begin later this summer. Present plans are to modify an existing neutron reflectometer instrument and install it at the first position on CG-4. Plans for the remaining three instrument locations in the Guidehall are in various stages of development and once neutrons are available there will be a concentrated effort to finalize those plans.

The HFIR and the Spallation Neutron Source (SNS) facility at ORNL have been recently combined under the same laboratory directorate and there is an increased emphasis to assure that new instruments proposed for either facility are optimal for that facility. Therefore, new instruments at both the SNS and the HFIR will be chosen based upon the user community need and the optimization of the instrument for either a steady state neutron beam or a pulsed neutron beam system.

3. Schedule for Completion of HB-4 Projects

The planned upgrades to the HB-4 beam line have taken considerably longer than originally planned, but are finally nearing completion. The testing of the cold source system is nearly finished and the regulatory reviews are underway. The present schedule shows the reactor restarting on March 23, 2007 with functioning cold source at HB-4. A detailed plan has been prepared to incrementally increase power level from less than 1% to full reactor power over a two day period so that data can be collected to commission all of the HB-4 shield assemblies. Expectations are that by the end of the summer the commissioning of the two SANS instruments will be completed.

4. Plans for Future Upgrades

Plans are presently in place to build a new user office and laboratory support building as part of the general upgrade of facilities at the HFIR site. Many of the buildings at the HFIR site are nearly 40 years old and the site is long overdue for additional office space and modern laboratory facilities needed to support a user program. Plans are in place to relocate staff to temporary structures to accommodate the construction of a new facility. Construction is expected to start in 2009 and take 2 to 3 years to complete. Figure 6 provides a conceptual picture of the new user facility at the HFIR.



Figure 6: Proposed New User Support Facility

In addition to the general facility upgrades of the site, discussions are presently underway with the US Department of Energy to support a project to add a second cold source and Guidehall that would be associated with the HFIR HB-2 beam. The HB-2 beam tube is the largest and most intense beam at HFIR. It has been estimated that cold beams from an HB-2 cold source would be a factor of 3 to 4 times more intense than the best in the world today. Preliminary evaluations have indicated that an HB-2 beam could illuminate 5 guides and support up to 10 new instruments. Figure 7 provides an early conceptual layout for a HB-2 Guidehall. Conceptual development of a HB-2 cold source and Guidehall system will continue over the next few years in preparation for an anticipated 2012 project start.

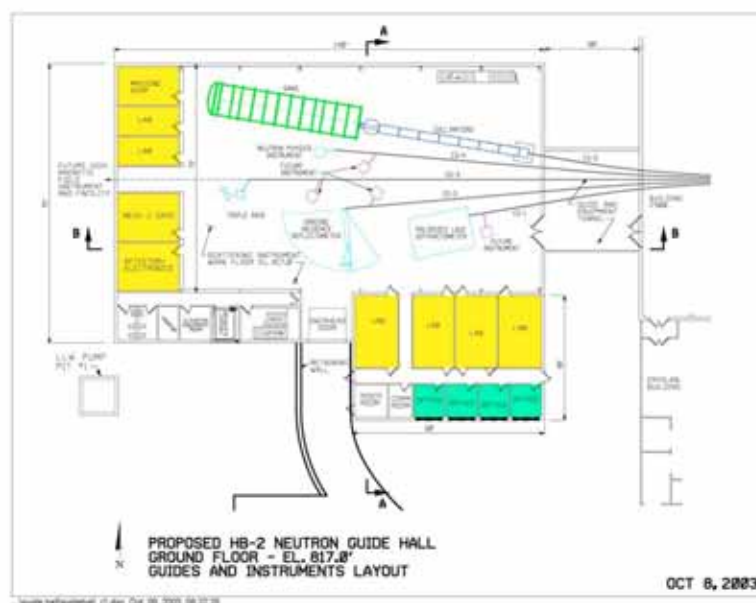


Figure 7: Early Conceptual Layout for an HB-2 Guidehall

New Moderator Chamber of the FRG-1 Cold Neutron Source for the Increase of Cold Neutron Flux

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ABSTRACT

GKSS installed a cold neutron source (CNS) at the FRG-1 in 1988. Principal component of this CNS is the moderator chamber with a discus shape. The moderator is supercritical gaseous hydrogen. In order to increase the yield of cold neutrons, a study was made for a new layout of the moderator chamber in 2003. The new fundamental design of the moderator chamber is based on a hemispherical shape, thereby increasing the cold neutron flux by approx. 60% with the use of focusing effects. The study of all relevant parameters was done by AREVA NP early 2006. The licensing procedure for fabrication and exchange of the moderator chamber took from May 2006 to the end of September including the participants of the independent experts. The set in operation program is planned in early March 2007.

1 Introduction

Long wavelength (cold) neutrons with high intensity are indispensable probe for the study of the microstructure and dynamics of condensed matter. These are necessary for its macroscopic characterization in applied as well in basic research. For this reason, around 60% of all GKSS neutron scattering instrumentation are using cold neutrons. With the CNS the number of long wavelength neutrons with wavelength > 0.4 nm were increased by a factor of more than 20. For a further increase of the important cold neutron flux, the moderator chamber of an existing spare unit should be replaced by a new one. Model of the new layout were the focusing moderator chambers of the American research reactors MURR and ORNL. These new moderator chambers resulted in gain factors between 50 to 150%. The following conditions formed the basis for the design and licensing procedure of the GKSS moderator chamber:

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

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

2 Optimisation studies

2.1 Neutronic studies for improvement of performance of moderator chamber

The Research Reactor FRG-1 is operated with a reactor core of 12 fuel elements in a 3x4 matrix arrangement. At three sides this core is surrounded by Beryllium reflector elements, the fourth side faces a block reflector of Beryllium with several holes containing the tops of the azimuthally arranged beam tubes SR6 to SR9.

The cold neutron source (CNS) is installed inside beam tube SR8 just a few millimetres outside the core outer boundary. Main parts of the CNS are a cylindrical vacuum chamber (AlMg3) arranged inside the beam tube SR8 filled with helium and a moderator chamber inside the vacuum chamber with the shape of a discus (Fig. 1).

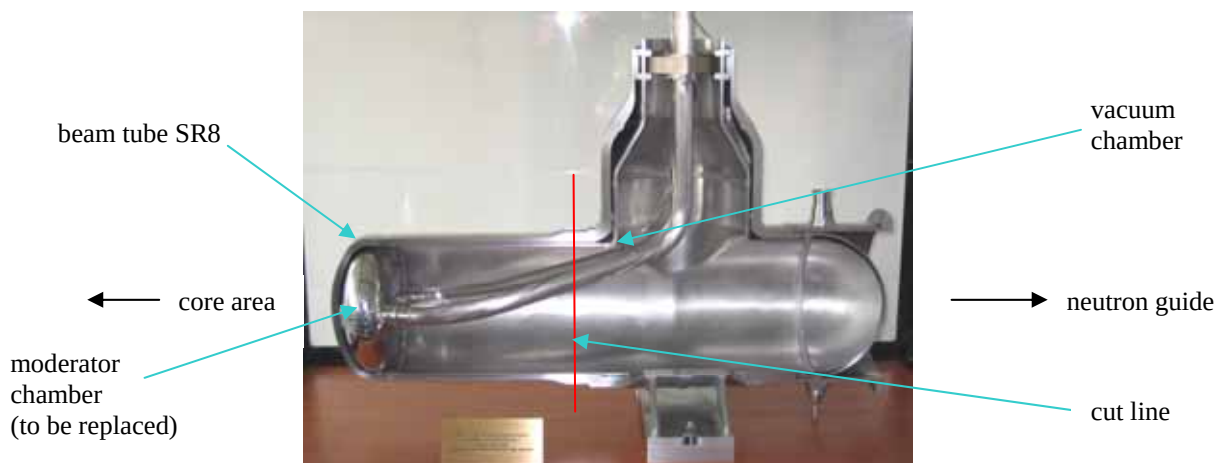


Fig 1. Inpile section of cold neutron source FRG-1, longitudinal cut through a prototype

The moderator chamber is part of a cold neutron source system operated with supercritical hydrogen at about 25 K and a pressure of 15 bar. The hydrogen serves as moderator for thermal neutrons and as coolant for the heat transport to the cryogenic helium refrigerator outside the reactor pool. The surrounding vacuum chamber provides a good thermal insulation to the beam tube and the reactor pool. The advantage of this medium at these operating conditions is to be always gaseous but with a density of about 90% of that of liquid hydrogen. For temperature control several thermocouples are attached to the moderator chamber.

In the course of the FRG-1 core compaction in 1999 the complex geometry of core, Beryllium reflector, tangential beam tubes and cold neutron source was modelled with the Monte Carlo computer code MCNP [1]. This included the detailed consideration of each single fuel plate, all structure materials, coolant, Beryllium reflector around the core and all beam tubes. An example is shown in Fig 2.

Burn up calculations were performed to get the fuel composition for an equilibrium core considering a multiple number of radial and axial burn up zones in each fuel element. The calculations showed a good agreement between calculated and measured neutron fluxes. Later in 2003 this MCNP data file was extended with a refined model of the front part of the cold neutron source, i. e. beam tube SR8, vacuum chamber and discus shaped moderator chamber thus providing the reference case for a comprehensive study of an optimisation of the geometry of the moderator chamber.

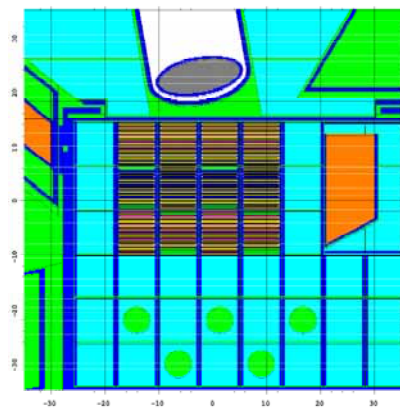


Fig 2. Cross section through core and reflector at the level of beam tube SR8

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

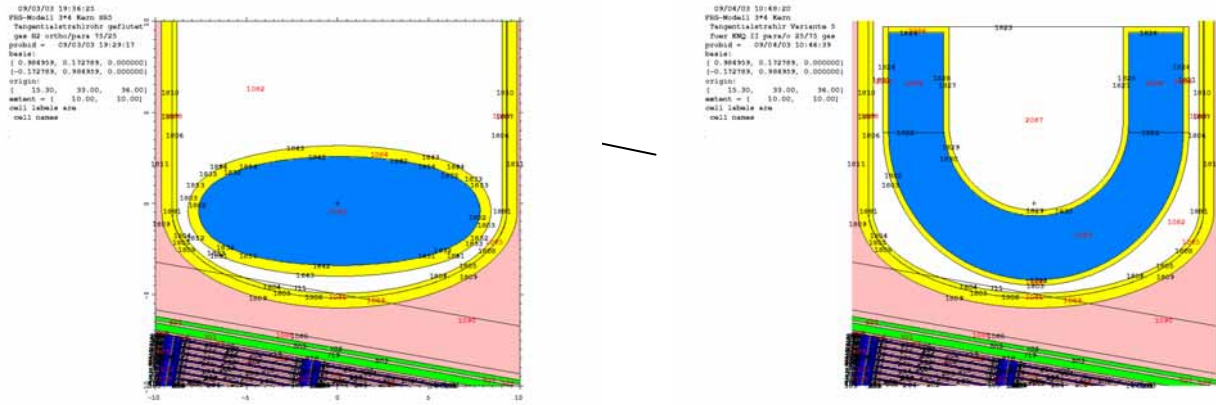


Fig 3. MCNP model of moderator chamber for reference design (left figure) and optimised design

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

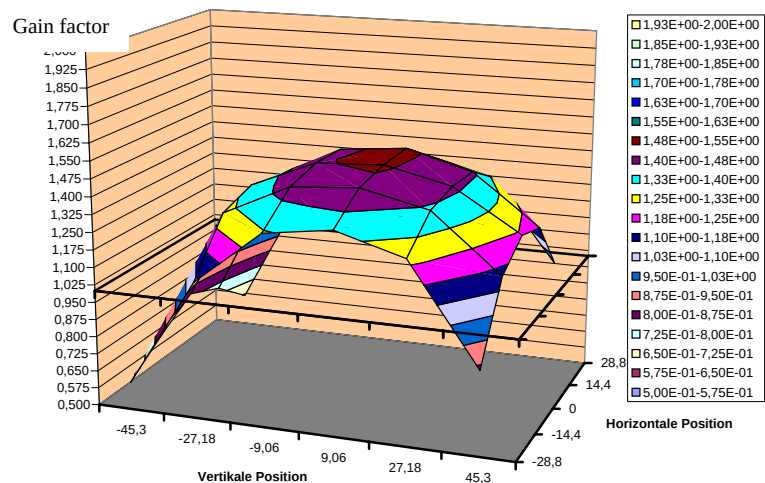


Fig 4. Gain factors in the range of 2 to 10 Å at the entrance to the neutron guide

These gain factors are energy dependent, they increase from about 1.35 at 4 Å to about 1.6 at 10 Å and to even higher values for larger wave lengths.

A further important result is the heat generation in structure material and moderator which amounts to 1625 W for the new design compared to 1355 W in the old design. This rather small increase makes it possible to operate the cold neutron source without any change of the cryo system which has a capacity of about 2000 W.

2.2 Verification of moderator chamber design

The detailed design of the moderator chamber was made with the 3d CAD tool Inventor on basis of the basic geometry resulting from the neutronic optimisation study. By means of direct data transfer to the FE code ANSYS via the existing dwg-interface the design of the moderator chamber was optimised further with respect to compliance of structure mechanical requirements at a maximum isothermal temperature of 100 °C during stand by conditions and to a minimisation of structure material. An exploded view of the optimised moderator chamber is shown in Fig. 5 together with the top of the vacuum chamber. The total mass of AlMg3 in this design amounts to 1100 g, this is only about 10% more then in the old design and assures validity of safety considerations on hypothetical CNS melt down accident scenarios for the old moderator chamber.

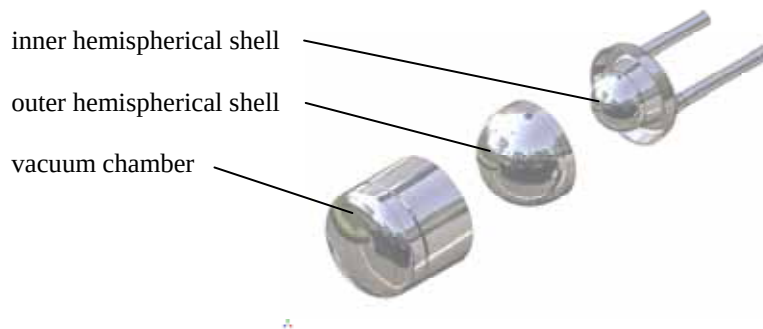


Fig 5. Exploded view of new focusing cold neutron source

For this geometry a thermodynamic study was carried out with the 3d computer code Star-CD for two operating conditions:

- normal operation with flow rate 2.0 l/s hydrogen at 25 K and 15 bar and
- stand by operation with flow rate 1.0 l/s hydrogen at 238 K (-35 °C) and 17 bar.

The total heat generation of 1625 W in structure material and in hydrogen was assigned to 7 regions of the moderator chamber. The results show a satisfactory distribution of coolant flow in the moderator chamber, low temperature rises of hydrogen and in the structure material and small azimuthal temperature differences across the spheres (Tab 1). Furthermore, the maximum temperature of 298 K is far below the temperature of 373 K (100 °C) assumed for the structure mechanical design. In Fig. 7 results are presented exemplarily for a 2d section through the model.

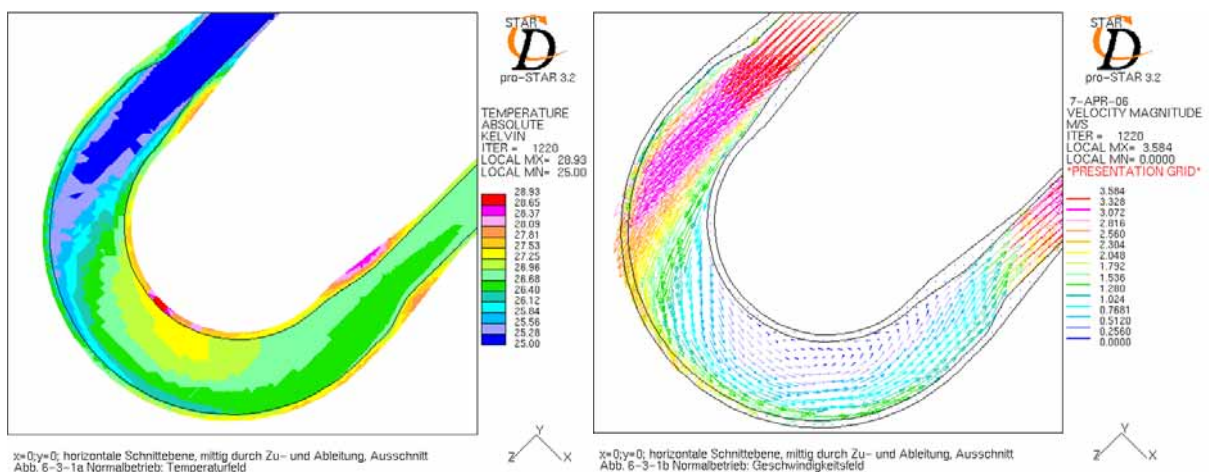


Fig 7. Temperatures (left figure) and coolant velocities in the moderator chamber at normal operation

	Min T-hydrogen	Max T-hydrogen	Min T-structure	Max T-structure
Normal operation	25	27	25	31
Stand by operation	238	271	242	298

Tab 1: Temperatures [K] calculated with 3d code Star-CD

3 Fabrication and installation

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

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

After transport of the new in-pile part to the FRG-1 the installation of the in-pile part will be made by the operation team of FRG-1 (end of February/beginning March 2007). An existing work instruction which was examined during the first installation 1988 will be applied for the installation.

4 Commissioning and validation

A part of the licensing procedure was the installation of a set in operation program. This program contains all steps from the inspection of the spare unit before beginning of the work up to the CNS operation during full reactor power. After the installation of the in-pile part in the reactor pool warm/cold leak test are accomplished, before hydrogen is filled into the plant. The operating parameters (cooling power) of the CNS with the new focusing moderator chamber are then determined by means of a heater in the helium refrigerator. The most important proof of the CNS is the determination of the operating parameters during reactor operation. For this test the reactor is operated in different power ranges. These tests are accomplished for the two operating conditions (standby operation $T = -35^{\circ}\text{C}$; normal operation $T = 25\text{ K}$). The final point of the commissioning program is intended the release of the CNS for normal operation. A measurement of the cold neutron gain factors at the experiments will serve as the confirmation of the MCNP calculations.

5. Summary

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

6. References

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Acknowledgments:

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STATUS OF MODERNIZATION AND REFURBISHMENT (M&R) ACTIVITIES OF THE IRT- RESEARCH REACTOR – SOFIA / INSTRUMENTATION AND CONTROL SYSTEM /

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ABSTRACT

The research reactor IRT-Sofia is in process of reconstruction into a reactor of low power 200 kW. Short technical description of the future IRT-200 research reactor is presented in the paper. The main fulfilled activities from the reconstruction are shown. It's written, why it is necessary to substitute the old instrumentation and control system with new one. The main subsystems and their functions of the future, new I&C system are described, according to technical project for reconstruction and technical specification.

1. Introduction

The research reactor IRT-Sofia will be reconstructed into a reactor of power 200 kW. The use of lowenriched uranium fuel, with uranium-235 enrichment below 20% (IRT-4M), is in accordance with the current norms on the security of transport and storage of nuclear and other radioactive materials which are vulnerable to theft by terrorists.

The following experimental channels are planned:

- two vertical channels in the fuel assemblies to supply fast neutron flux $3 \cdot 10^{12}$ n/cm²s;
- two vertical channels in beryllium blocks to supply thermal neutron flux $8 \cdot 10^{12}$ n/cm²s;
- seven horizontal channels outside the aluminium vessel of the reactor core with fast neutron flux $1,6 \cdot 10^{12}$ n/cm²s, and thermal neutron flux $5 \cdot 10^{11}$ n/cm²s on the core vessel;
- six vertical channels outside the aluminium vessel of the reactor core with fast neutron flux $2 \cdot 10^{12}$ n/cm²s, and thermal neutron flux $7 \cdot 10^{10}$ n/cm²s on the core vessel;
- channel for BNCT with epithermal neutron flux $0,9 \cdot 10^9$ n/cm²s.

For neutron flux and neutron spectrum measurements an assortment of neutron activation foils and threshold detectors (e.g., Au, In, Cd, Al) as well as counting devices such as gas flow proportional counters, NaI or HPGe detectors of the necessary class will be provided. For approach-to-critical experiments one or more neutron detection systems using BF₃, ¹⁰B, or a fission chamber, along with the necessary electronic equipment will be additionally provided.

Rooms (laboratories) for installation of measurement and automated systems, for radiation monitoring systems, and others according to the clients' needs are also planned. For express measurements of shortlived isotopes a pneumatic sample transfer system (rabbit system) is planned.

2. General Modernization & Refurbishment Scope

The elaboration of the Technical Project [5] and the Detailed Design for the reactor reconstruction as well as of all of the documents needed for the Safety Analysis Report is done by the INTERATOM Consortium, which consists of:

- Atomenergoproekt Ltd. Bulgaria - Chief designer
- Skoda JS a.s. Czech Republic - Chief constructor

- RRC “Kurchatov Institute”, Russian Federation - Scientific supervisor

The Technical Project, Safety Analysis Report – Revision 3 and General Plan for Partial Dismantling have been proposed for approval in Bulgarian Nuclear Regulatory Agency. The Detailed Design is in process of elaboration. All activities concerning fresh fuel conversion and spent nuclear fuel removal, financed by the US Department of Energy (DOE) in the frame of two programs: RERTR (Reduced Enrichment of Research and Test Reactors) and RRRFR (Russian Research Reactor Fuel Return), are in progress. The program RERTR is implemented for:

- return of highly enriched fresh fuel IRT-2M (HEU, 36% ^{235}U) to Russia. This was done in December 2003;
- joint studies at the Argonne National Laboratory [3,4] aiming at conversion from use of fuel, containing highly-enriched uranium IRT-2M to low-enriched uranium fuel IRT-4M (LEU, 20% ^{235}U), chosen for the reconstructed reactor IRT-200;
- delivery of fresh fuel IRT-4M from Russia, needed for the reactor start-up in 2008.

Under the RRRFR program and the contract between the INRNE and the DOE, signed in April 2005, an intensive work has been carried out for the IRT spent nuclear fuel shipment to Russia. According to the work schedule, this fuel is expected to be shipped by the end of 2007.

The term of the spent fuel shipment is crucial and determining, as far as the first, preparatory stage of the reactor reconstruction is completing by it and the second one, the construction and assembling stage is beginning.

The following other activities concerning reconstruction have been fulfilled:

- Signing of contract with “ANILS” company - Bulgaria for manufacturing, delivery and installation of Primary cooling loop, Secondary cooling loop, Water purification loops, reactor and NSF storage pool, Water make-up loop
- Signing of contract with “SKODA-JS” – Czech Republic for manufacturing, delivery and installation of reactor pool, fuel storage pool, experimental channels, reactor carrying box, reactor shroud, ejector and piping inside the reactor pool and Control rod drives
- Signing of contract under the EU PHARE program for delivery of equipment for “Radiation Monitoring System at the Nuclear Scientific and Experimental Centre with Research Reactor (IRT type) in Sofia, Bulgaria” with “SYNODYS GROUP”. All the equipment was delivered in August 2006 on site, and now some of the laboratory and dosimetry equipment are in use.

3. Modernization & Refurbishment Scope of Instrumentation and Control System

The instrumentation and control system (I&C) of the IRT-2000 research reactor was developed in the 60's of the last century in the former Soviet Union. The system is constructed according to a relay-contact scheme. Relays and contactors of 110V and 48V direct current are used in these schemes. This equipment is physically and morally old, that's why this system will be entirely substitute by a new one, which will be according to the contemporary requirements for such system. The new system design corresponds to the requirements of the acting in Republic of Bulgaria regulatory documents, as well as of the applicable for this case foreign and international recommendations and standards. This system will provide reliable control and regulation of power level above the subcritical state, at different power levels and in dynamic regimes, as well as reliable reactor shutting down in normal and accidental conditions.

The I&C system will have the following composition: complex of control and protection system equipment (CPSE complex) and information-computing system equipment (ICS equipment) [1].

3.1. CPSE complex.

The most important parameters in CPSE are the reliability and the fast response, determining the safety and failure-free operation of the reactor, so the CPSE will be built by “independent channels” principle and will satisfy the requirements for fast-execution (minimum delay) of control signal for emergency protection and of permissible probability of non-actuation of control signal for emergency protection by the requests for reactor shutdown. The system will ensure the control of emergency protection and displacement of actuators by means of duplicated channels for monitoring

and protection by power, period and process parameters. The logic of CPSE operation is majority voting according to logic 2 out of 3 signals by each of the following parameters:

- reactor power;
- period;
- temperature of water at the core inlet;
- water heating in the core;
- level of water in the reactor pool;
- pressure drop in the reactor core;
- radioactivity of water in the primary circuit pipeline;
- radioactivity of gases at the above reactor space;
- seismic activity.

The functions of the system will be:

- detection of neutron flux in all operating modes of the research reactor;
- detection of process parameters;
- generation of signals at reaching the threshold values by power and period for control of emergency protection and monitoring;
- generation of signals at reaching the threshold values by the process parameters (temperature of water at inlet to the reactor core, heating of water at the reactor core, level of water in the reactor pool, pressure drop at the core, radioactivity of water in the pipeline of primary circuit, radioactivity of gases in the above reactor space, seismic activity) for control of emergency protection and monitoring;
- generation of signals for control of actuators in the modes of reactor emergency shutdown and normal operation;
- monitoring of control rods position (detectors of the top, bottom and intermediate position of control rods are located in actuators);
- monitoring of reactivity;
- manual control of reactor power from the control panel;
- automatic power control;
- planned reactor shutdown;
- displaying and registration of information;
- determination of time of control rods insertion;
- automatic check of good condition for the equipment during operation process, including the check of good condition for communication lines, detection units of neutron flux and process parameters;
- archiving and documenting of information;
- fixation of initial cause for emergency situation occurrence;
- automated pre-start check for generation of emergency protection signals and preventive signalization;
- communication with information-computing system.

3.2. Actuators of Control and Protection System - ACPS

The actuators are divided according to their function in two types [2]:

- compensative actuators for regulation of reactor power and for reactivity compensation
- safety actuators for shut-down of reactor in case of accident

The actuators will be multi-purpose. It means that one actuator can work either as compensative or as safety. Just switch-over in control room must enable to change function of single actuators. This solution simplifies working of the operating staff. In many cases won't be necessary to remove actuators on another position in reactor core.

The main functions of the ACPS are to:

- Provide power level regulation during operation condition by absorber rod movement
- Provide reactor shutdown in case of accident by absorber rod free fall
- Provide reactivity compensation by absorber rod movement

The ACPS is composed from five following main parts:

- The drive mechanism,
- The absorber rod channel,
- The component of absorber rod,
- The supporting part of ACPS,
- The connection box,

3.3. Information-computing system

Information-computing system will provide the execution of the following main functions:

- acquisition of data on state of the reactor, its technological systems and radiation monitoring systems;
- access to the archives;
- diagnostics of hardware and software means;
- support of the unified time and assignment of the time-mark during collection of data;
- protection from unauthorized access;
- presentation of information to external users.
- ICS equipment should support functioning of the standard network interfaces RS-485, ETHERNET, and include:
 - device for archiving, analysis of archived data, documenting;
 - synchronizer (device for setting unified time);
 - personal computers in industrial implementation;
 - printers;
 - software will satisfy the requirements of the IEC 60880-1,2 – 2002 international standard:

3.4. Control rooms

There will be build two new control rooms :

- *main control room* – there will be situated control panels; units for setting emergency protection threshold values by power of neutron flux monitoring equipment; unit for setting values by power and period of automated power controller; digital displays for displaying current power and period of neutron flux values; graphical displays; keyboards; workstations; printers; archiving, diagnostic and logging hardware; etc.
- *supplementary control room* will be situated in a separate place, distant from the basic control shield. It will be seismic-proof and fire-safe and will have local ventilation system and autonomous communications.

Choosing of a company for manufacturing, delivery and installation of I&C equipment is forthcoming.

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CONCEPTUAL DESIGN OF A PRESSURIZED WATER LOOP FOR THE IRRADIATION OF 6 FUEL RODS IN THE JULES HOROWITZ REACTOR

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ABSTRACT

The Jules Horowitz Reactor will be mainly dedicated to studies of material and fuel behaviour under irradiation. This paper presents a conceptual design which could meet efficiently irradiation needs of fuel rod clusters, to support safety and performance improvement of generation 2 and 3 light water reactor fuels. It describes preliminary features of a water loop dedicated to the irradiation of 6 re-fabricated rods with pressure and temperature conditions identical to pressurized water reactors. One important feature of the concept lies in its specific sample-holder which ensures the positioning of the 6 rods in highly instrumented test channels, providing accurate monitoring of Linear Heat Generation Rate (LHGR) of each fuel rod. Another important characteristic is the possibility to load fuel rods equipped with two sensors. Moreover, solutions are proposed to minimize the LHGR differences between rods.

1. Introduction

The experimental capability of the Jules Horowitz Reactor (JHR) will be mainly dedicated to studies of material and fuel behaviour under irradiation. It will play a major role for supporting safety and performance improvements of generation 2 and 3 Light Water Reactors (LWR), and for innovations and characterizations required by Generation 4 reactor fuels and materials (1).

Among the different qualification processes of nuclear fuel, screening and comparison irradiation tests are necessary to select the most promising products (2). The aim of a selection irradiation is to choose one or a few fuel materials, among a batch of candidates, offering a good potential behaviour with respect to technical specifications. Selection can be performed thanks to irradiation results and extensive post-irradiation examination program or tests in hot cells.

First exploratory tests can be performed in a simple test device not representative of the reactor conditions and with limited on-line instrumentation. However, in a comparison phase of new products, it is relevant to irradiate, in the same flux, several rods in conditions similar to LWR.

For such an experiment, a perfect knowledge and monitoring of the local conditions is needed, especially for Linear Heat Generation Rate (LHGR). The best homogeneity of LHGR is also wished with regards to Fission Gas Release (FGR) (e.g. 5% LHGR increase may induce 40% FGR increase). In addition, to improve the understanding of fuel behaviour under irradiation, it is necessary to have in situ measurement of the main parameters e.g. fuel temperature and rod pressure.

Time history of irradiation consists generally of stable power levels with periodical power adjustments. Some samples can be unloaded during the experiment for intermediate exams, and new ones can replace them. Irradiation duration can be short if beginning-of-life phenomena are to be quantified (e.g. 3 to 6 months). But generally it is a long experiment (more than one year) and consequently the irradiation device and instrumentation shall be robust and reliable.

This paper presents a conceptual design which could meet efficiently needs for irradiations of rod clusters. It describes preliminary features of a water loop dedicated to the irradiation of 6 re-fabricated rods with pressure and temperature conditions identical to pressurized water reactors. This test device is foreseen for steady state irradiations in the periphery of the JHR's core, with the possibility to adjust the fuel rod power. The work has been initiated in the frame of FP6 European Union program "JHR-CA" which conclusions were presented in (3).

2. Main operating principle

This test device is an experimental pressurized water loop designed to LWR fuel rod testing in the JHR's reflector. Typical samples are 6 re-fabricated fresh or pre-irradiated rods with a fissile length of about 450 mm and an external diameter of 9.5 mm. It is designed for steady state irradiation but to allow fuel rod power changes, the test rig should be placed in the reactor's reflector in one specific experimental location equipped with a variable thermal neutron screen or on one of the JHR's displacement systems.

The in-pile part (Figure 1) is a Zircaloy double-wall pressure tube with a controlled gas gap. It is designed to ensure high-pressure high-temperature water containment. This pressure tube houses the sample holder, which ensures the positioning of the 6 rods in 6 independent test channels and holds the main experimental sensors. Each test channel is constituted by two stainless steel concentric tubes to provide a small gas gap filled with inert gas.

After taking the sample holder apart from the pressure tube, the loading and unloading of the fuel rods is possible from the bottom of the sample holder. Fuel rods are centred in the test channel by spring devices and locked on the bottom of the sample holder. Rod's instrumentation is connected to sample holder specific connectors. All these operations should be done in hot cell with robotic arms.

The cooling of fuel rods is based on pressurized water forced convection. After being pre-heated by electrical means in the out-of-pile circuit, water reaches the in-pile containment, and flows down into the down-comer: space between the pressure tube and the 6 double-wall channel tubes. Then, the flowrate is divided to rise into the 6 independent test channels around the fuel rods. Finally, after reaching the collector, it returns to the out-of-pile circuit.

Connected to the heads of the pressure tube and sample holder, lines to the out-of-pile equipments are composed of tubes for inlet and outlet water circulation, tubes for gas supply, safety injection, bleeding and instrumentation wires.

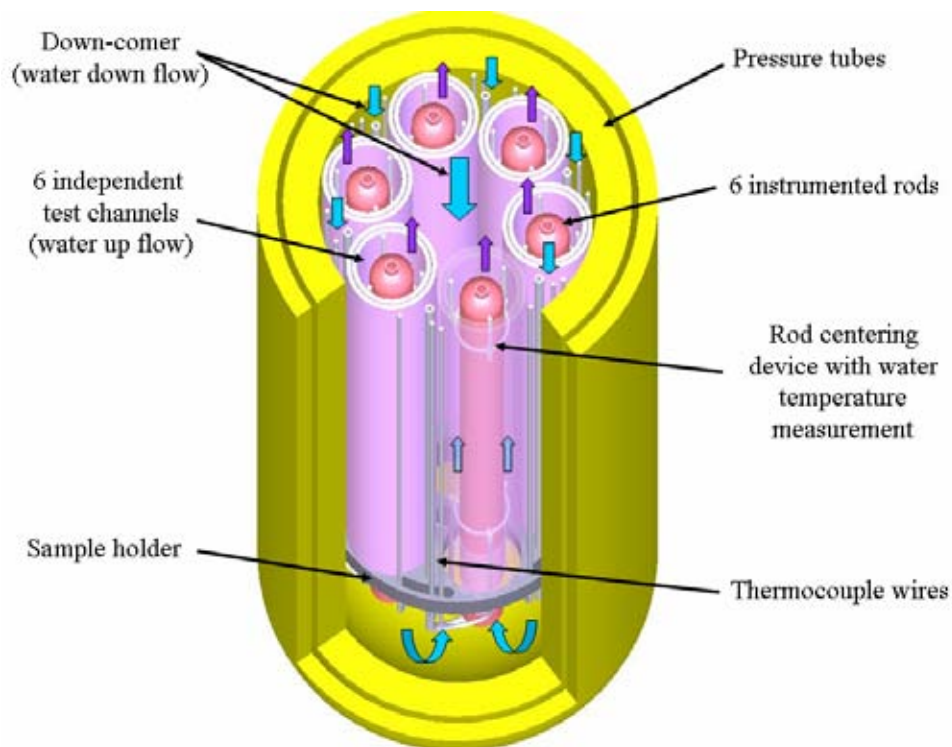


Figure 1 : Radial cut of the loop's in-pile rig

3. Main in-pile instrumentation

The on-line measurement of the linear heat generation rate is a key parameter and becomes very challenging in the case of an experiment with several rods.

The classical Self Powered Neutron Detectors (SPND) could be placed in the fissile zone close to each fuel rod to measure neutron flux, and neutronic modelling allows LHGR calculation of each fuel rod. However, this process is not very accurate, especially when irradiation history has to be taken into

account with increasing burn-up. Moreover those detectors made of materials with large thermal neutron cross section may induce local shadow effects too high for such experiments.

Thus to improve the knowledge and the monitoring of the LHGR, this conceptual design is based on the instrumentation of each test channel with thermocouples and flowmeter to be able to make a thermal balance for each fuel rod (Figure 2). The inlet and outlet thermocouples could be maintained into position thanks to rod centring devices placed respectively below and above the fissile column. To improve the thermal balance accuracy, a double wall tube filled with Xenon gas is chosen to limit the heat exchange between the test channel and the down-comer. For instance, the flowrate could be measured by a small turbine flowmeter placed in each channel well above the fuel rods. The sensors necessary for the heat balances will serve also for on-line monitoring of the in-pile water temperature and flowrate conditions. Concerning the pressure condition, transducers could be placed at the top of the in-pile part of the loop in order to have accurate measurement.

To improve the understanding of fuel behaviour under irradiation, it is necessary to have in situ measurement of some parameters. Each fuel rod could be instrumented at both ends. For instance, the central fuel temperature could be measured by a thermocouple placed in drilled pellets with a tight path in the bottom of the rod. A watertight connector will be necessary to load and unload the fuel rod with robotic arms in hot cells. The top end of the rod could be instrumented with a cable free detector based on Linear Voltage Differential Transformer (LVDT), to allow the loading and unloading of fuel rods in the sample holder. This technology could be adapted to measure either rod internal pressure, cladding length variation or fuel column displacement.

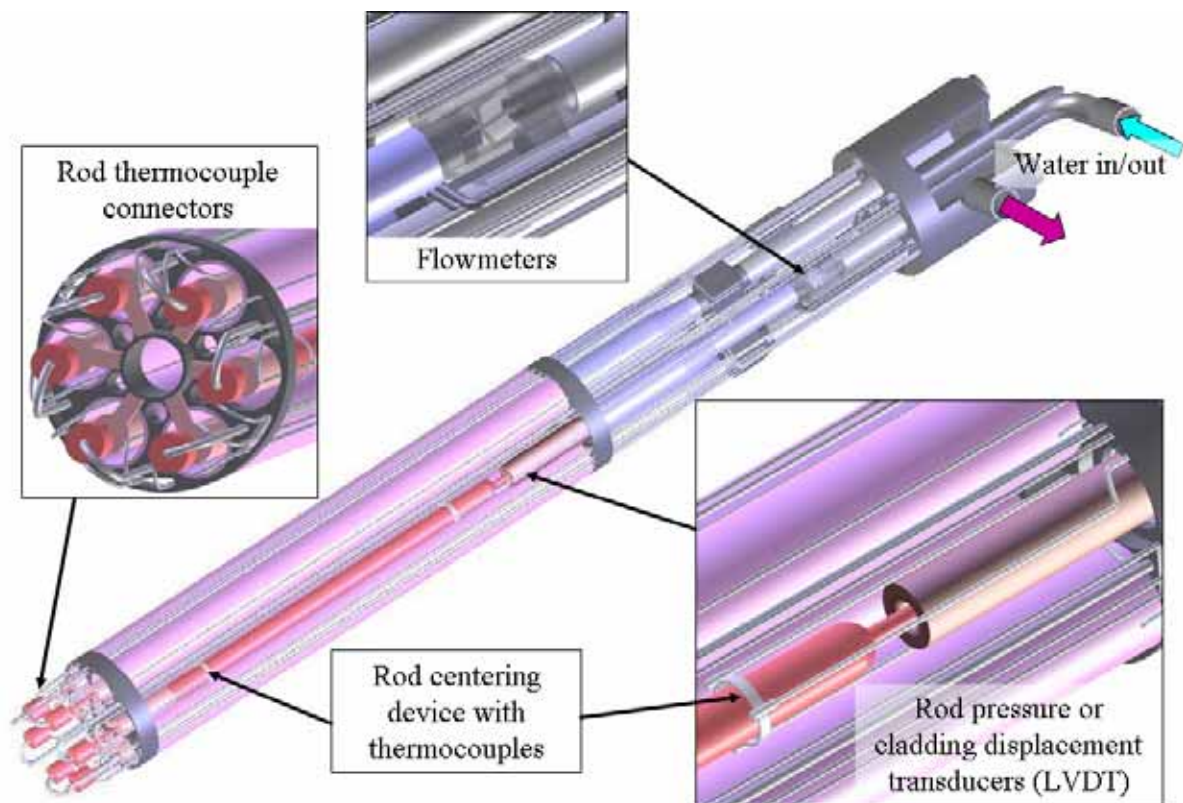


Figure 2 : Schematic view of the sample holder with the in-pile instrumentation

4. Thermal and mechanical studies

Preliminary thermal-hydraulic calculations were carried out, taking into account thermal expansion, gamma heating in the different materials (3 W/g), cosine-profile flux (max./mean=1.25 over 60 cm) and convection with the pool circuit leg of the JHR. At this stage of the design, only steady states in normal conditions are considered. To cover a large thermal-hydraulic operating range, the LHGR has been set near the hot spot values of PWR rods. Heat balances in the down-comer, and in a test channel are evaluated for a LHGR of 400 W/cm at reactor mid-plane. Flowrates were chosen to provide large temperature elevation in each test channel, in order to have accurate heat balance measurements. The main hypotheses are given on figure 3.

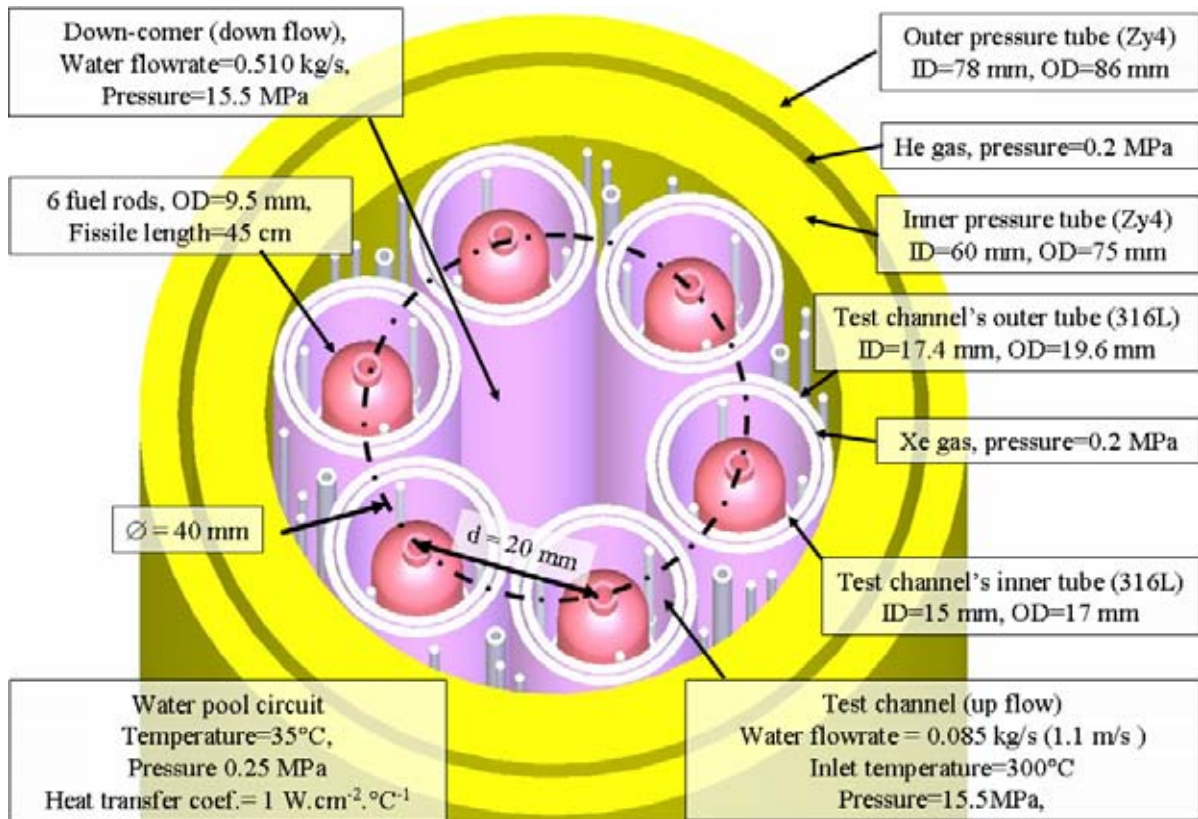


Figure 3 : Main characteristics of the loop's in-pile part

In such conditions, temperature increase in one test channel is 32°C. The outlet coolant temperature of each test channel is lower than the saturation temperature of pressurized water (344°C at 15.5 MPa) and the cladding temperature is above saturation level all along the fuel rod (Figure 4).

In the active zone, thanks to the Xenon gas gap, there is almost no heat loss from the test channel toward the down-comer and most of gamma heating in the inner tube is removed by the water flow in the test channel. Consequently, the down-comer temperature elevation is mainly due to gamma heating removal from the test channels' outer tube and from the inner pressure tube.

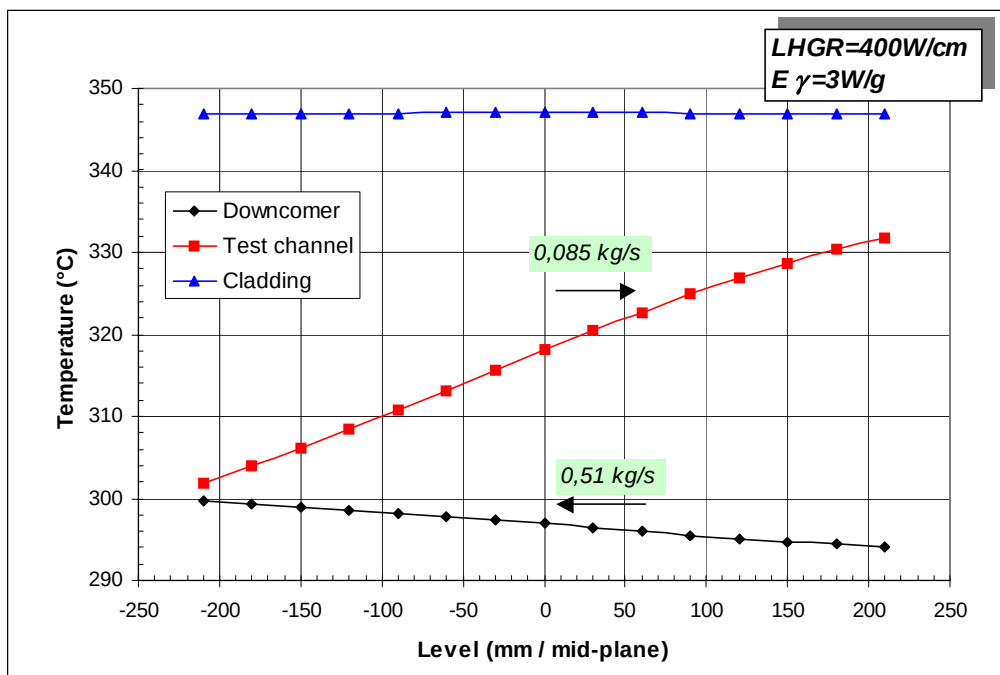


Figure 4 : Axial temperature profiles, LHGR=400 W/cm, Eγ=3 W/g

Tube	Material	Diameters (mm)	Inner temperature	Outer temperature	Inner pressure (MPa)	Outer pressure (MPa)
#1	Cold-worked 316L	15 × 17	333 °C	334°C	15.5	0.5
#2	Cold-worked 316L	17.4 × 19.6	302 °C	301°C	0.5	15.5
#3	Class1-Zircaloy 4	60 × 75	318 °C	342°C	15.5	0.2
#4	Class1-Zircaloy 4	78 × 86	74 °C	48 °C	0.2	0.25

Table 1 : Maximum temperature and pressure values of the tubes (400 W/cm, 3 W/g)

A finite element model was developed to compute the mechanical stress within tubes, taking into account the local properties versus temperature of the tubes and pressure values (Table 1).

Then, the calculated stresses were compared to the allowable stresses. This analysis does not highlight any operating problem in nominal conditions: the calculated stresses are smaller than the allowable stresses (Table 2).

Tube	Primary membrane stress (P_m)	Allowable stress (S_m)	Primary membrane + bending stress ($P_L + P_B$)	Allowable stress ($1.27 \times S_m$)
#1	104 MPa	205 MPa	117 MPa	260 MPa
#2	109 MPa	208 MPa	122 MPa	264 MPa
#3	59 MPa	61 MPa	72 MPa	77 MPa
#4	0.4 MPa	136 MPa	0.5 MPa	173 MPa

Table 2 : Mechanical calculation in nominal operating conditions (400 W/cm, 3 W/g)

5. Neutronic calculations

Neutronic simulations were carried out with the 3D TRIPOLI4 Monte Carlo code, using 10 million neutron histories. The 100 MW core is described with 34 fresh fuel assemblies of UMo7 20%-enriched 3×8 plates. Neither control system nor reloading has been taken into account. The pressurized water loop is located in a water row dedicated to displacement systems in the beryllium reflector. The experimental fuel samples are 2% enriched fresh UO_2 fuel rods. In the configurations presented in this paper the LHGR of experimental rods is given at reactor mid-plane with a statistical accuracy of $\pm 4\%$ (2σ).

Different studies were performed by varying the distance of the loop's axis from the core rack and the best homogeneity of LHGR is obtained near the core in the peak of thermal neutron flux (Figure 5). To reduce the LHGR values and minimize the difference between rods, a solution consists in keeping the test device near the core rack and adding a Nickel screen of varying thickness. In the two examples presented figures 6 and 7, the gradient is kept lower than 10 %. Indeed this solution needs to be optimized: for instance, a specific Nickel screen could be designed for a given operating point and the displacement system would be kept in order to make small changes around this operating point.

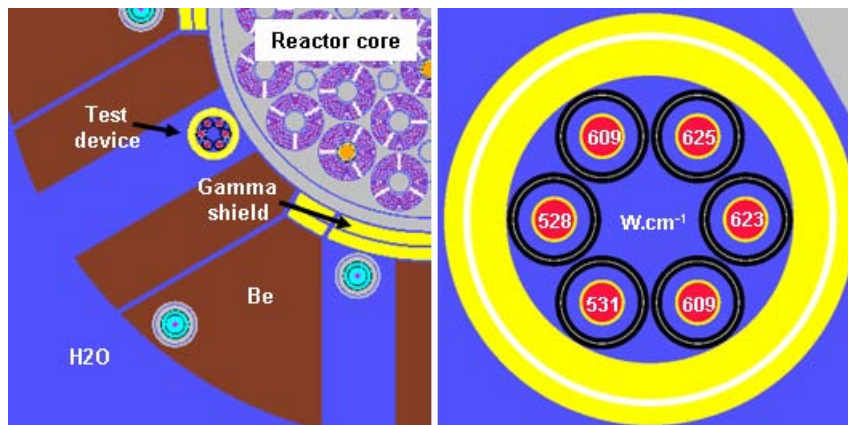


Figure 5 : Test device without specific screen – distance from the core 7 cm.

Fuel rod LHGR (core mid-plane): mean value 588 W.cm^{-1} – grad. 17 %

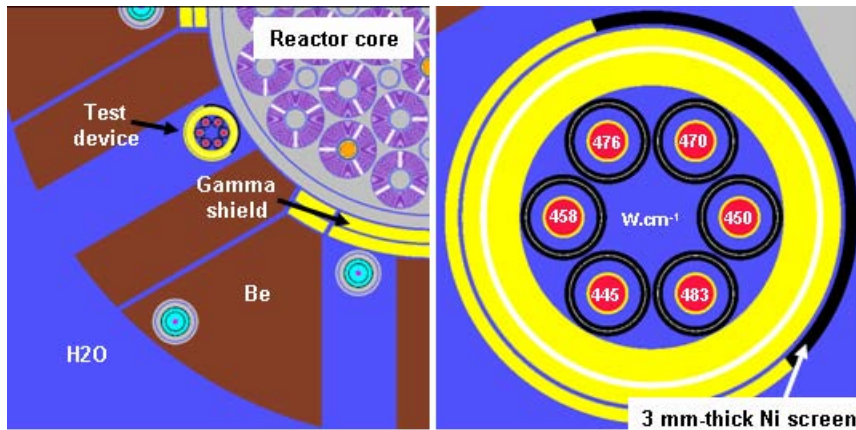


Figure 6 : Test device with a 3 mm-thick Nickel screen – distance from the core 7.4 cm.

Fuel rod LHGR (core mid-plane): mean value 464 W.cm^{-1} – grad. 8%

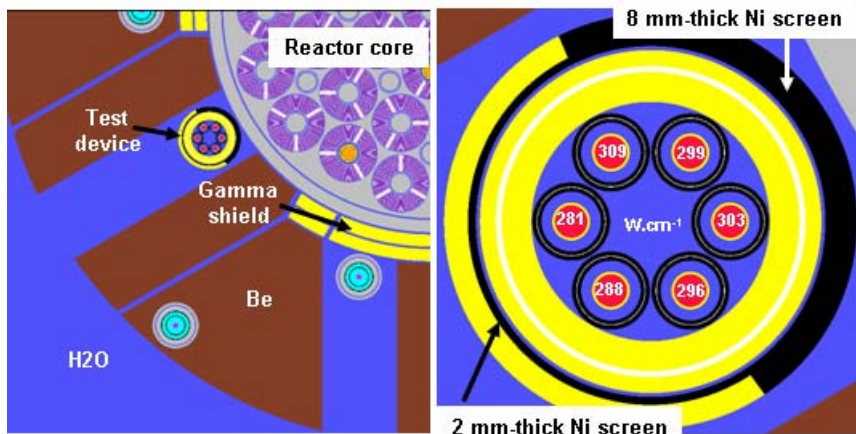


Figure 7 : Test device with an 8 mm-thick, 2 mm-thick Nickel screen – distance from the core 7.9 cm.

Fuel rod LHGR (core mid-plane): mean value 296 W.cm^{-1} – grad. 9%

6. Conclusion

The main characteristics of a water loop dedicated to the irradiation of 6 rods with pressure and temperature conditions identical to PWRs have been presented in the frame of a conceptual design study. One important feature of the concept lies in its specific sample-holder which provides 6 instrumented test channels to improve on-line monitoring of linear heat generation rate of each fuel rod by independent heat balance. Another important characteristic is the possibility to load fuel rods equipped with two sensors at both ends. Based on neutronic calculations, solutions with Nickel screens have been proposed to minimize the LHGR differences between rods.

In a next stage, the main components of the loop should be defined and all its circuit should be taken into account, not only in steady state operating conditions, but also in transient and incidental operating conditions. This proposal of loop has to be discussed with fuel R&D teams as well as end-users to confirm the interest in multi-rod irradiation experiments and thus to carry on the study and the development of such a test device.

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DEVELOPMENT OF HIGH TEMPERATURE CAPSULE FOR RIA-SIMULATING EXPERIMENT WITH HIGH BURNUP FUEL

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ABSTRACT

In order to improve fuel cycle economics and resource utilization efficiency, fuel burnup extension in LWRs has a particular importance in Japan. Behaviour of high burnup fuels during off-normal conditions, such as reactivity-initiated accident (RIA), is being studied with the Nuclear Safety Research Reactor (NSRR) of the Japan Atomic Energy Agency (JAEA). Recent RIA-simulating experiments indicate that the occurrence of fuel failure at higher burnup is closely related to the cladding embrittlement due to the hydrogen absorption. The type of fuel failure, hydride-assisted PCMI (pellet/cladding mechanical interaction) failure, may be influenced by the initial temperature of cladding, since the ductility of cladding becomes high at high temperature and the failure occurs before temperature rise of cladding due to the power burst. For the verification of the temperature effect on the fuel failure, a new capsule was developed to carry out experiments at high initial temperature.

1. Introduction

The Nuclear Safety Research Reactor (NSRR) [1] of the Japan Atomic Energy Agency (JAEA) was built for the study of light water reactor (LWR) fuel behaviour during off-normal conditions such as reactivity initiated accident (RIA). RIA-simulating experiments using fresh, i.e. unirradiated, LWR fuels were carried out since 1975. Modification [2] of experimental facility in 1989 made it possible to conduct experiments with fuels irradiated in commercial nuclear power plants. The results obtained from unirradiated fuel experiments and irradiated fuel experiments were utilized to establish the experimental data base for the evaluation of fuel integrity during off-normal conditions.

In order to improve fuel cycle economics and resource utilization efficiency, fuel burnup extension in LWRs has a particular importance in Japan. Safety of the high burnup fuels must be assessed, and behaviour of these fuels during off-normal conditions, such as reactivity-initiated accident (RIA), becomes a primary concern.

During a past decade, RIA-simulating experiments in the NSRR and the CABRI test reactor in France showed that fuel failures at higher burnup occurred at enthalpy values lower than would be expected [3], [4]. Results from the two programs indicate that the occurrence of fuel failure is strongly influenced by embrittlement due to hydrogen absorption of fuel cladding. This type of fuel failure, hydride-assisted PCMI (pellet/cladding mechanical interaction) failure, can be influenced by the initial temperature of cladding since the failure occurs before temperature rise of cladding and the hydrided cladding is brittle at low temperature. To verify the effect of initial temperature on the fuel failure, a new capsule was developed to achieve high temperature coolant condition which simulates hot zero power for PWR start-up.

This paper summarizes specifications of the NSRR, recent experimental results and the development of a new capsule. The development of a new capsule was conducted under the support of the Nuclear and Industrial Safety Agency, the Ministry of Economy, Trade and Industry.

2. Specifications of NSRR

The NSRR is a modified TRIGA-Annular Core Pulse Reactor (ACPR). Figure 1 and 2 show vertical and horizontal cross section of the NSRR, respectively. The core structure is mounted at the bottom of a 9m deep open-top water pool, and cooled by natural circulation of the pool water. The NSRR core consists of 149 driver uranium-zirconium hydride (U-ZrH) fuel/moderator elements, six regulating rods with fuel follower, three transient rods and two safety rods with fuel follower, as shown in figure 2. An experimental capsule containing test fuel rods is inserted to the experimental cavity located at the core center. The specifications of the NSRR are summarized in Table 1.

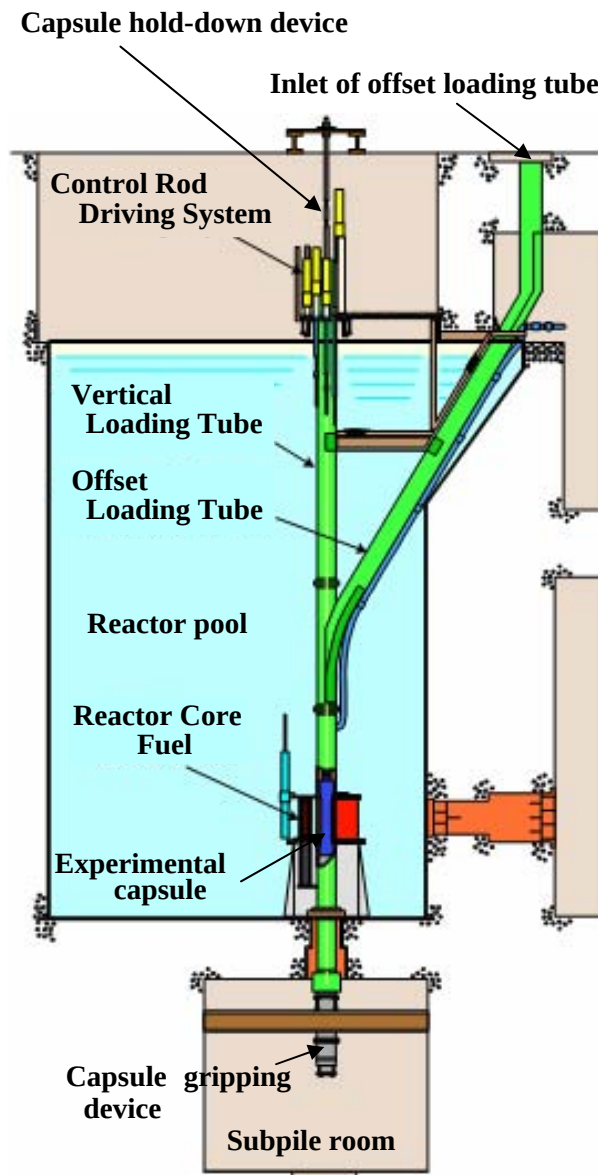


Figure 1: Vertical Cross section of the NSRR

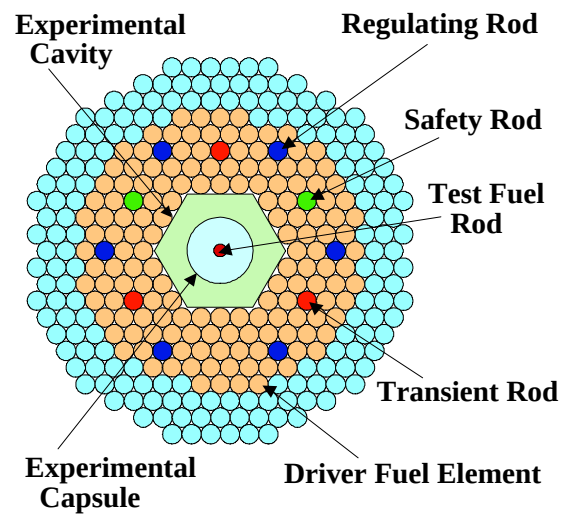


Figure 2: Horizontal Cross section of the NSRR core

Table 1: Specifications of the NSRR

Type	Swimming pool, annular core
Core	Effective height: 38 cm
	Equivalent diameter: 63 cm
	Moderator: $\text{ZrH}_{1.6}$ and H_2O
	Reflector Graphite and water
Fuel rod	Type: 12 wt%U in ZrH
	^{235}U enrichment: 20 wt%
	Shape: Cylindrical rod
	Clad material: SUS304
	Number: 149 rods
Control rod	Safety rod: 2
	Regulating rod: 6
	Transient rod: 3
Pool	Width: 3.6 m
	Length: 4.5 m
	Depth: 9 m

The operation for pulse irradiation is made by a quick withdrawal of enriched boron carbide transient rods by pressurized air. Since the hydrogen in the fuel elements is heated instantaneously by rapid power

escalation, the NSRR has a large prompt negative reactivity coefficient. This safety features provides a high self-controllability of the NSRR. Figure 3 shows typical transient power history and integrated power history. The maximum reactor power of the NSRR is 23 GW. In this case, the full width at half maximum of the power pulse is 4 ms and the total energy release reaches 130 MJ.

3. Results of NSRR experiments

About 80 experiments with pre-irradiated fuels were carried out in the NSRR. Most of them were performed under a condition of stagnant water at room temperature and atmospheric pressure. The outline of the recent results is summarized as follows.

Experiments with pre-irradiated fuels indicated that fuel failure occurred at low enthalpy before a cladding temperature rise. The failure mode is the pellet/cladding mechanical interaction (PCMI). Figure 4 shows fuel enthalpies at failure in the NSRR experiments as a function of burnup. The fuel enthalpy at failure decreases when burnup becomes high. This is because high burnup makes cladding brittle due to corrosion such as surface oxidation and hydrogen absorption. Figure 5 shows a cross section of PCMI-failed high burnup fuel. The fracture is composed of two parts, one of which is ductile in the inner region of cladding and the other is brittle in the outer region.

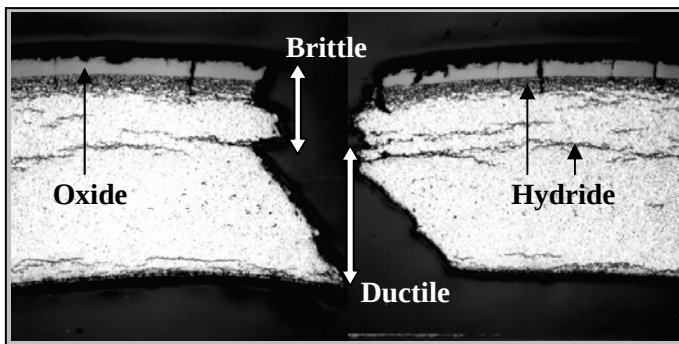


Figure 5: Cross section of PCMI-failed fuel

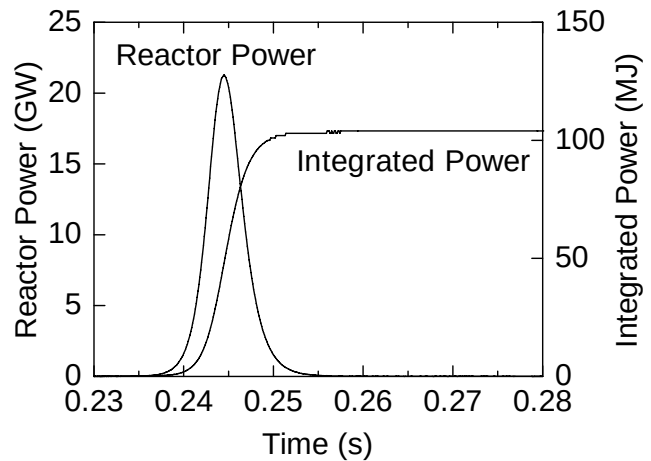


Figure 3: Typical power history of the NSRR

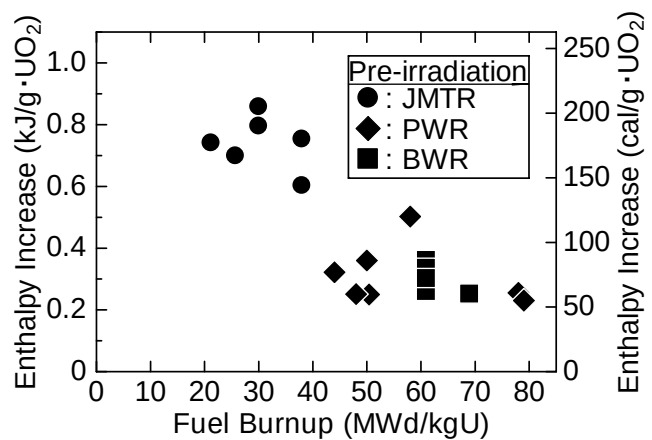


Figure 4: Enthalpy increase at fuel failure as a function of fuel burnup

The fracture is composed of two parts, one of which is ductile in the inner region of cladding and the other is brittle in the outer region. Some cracks beside the fracture are also observed in the peripheral region. The process of PCMI-failure is thought that incipient cracks through oxide layer and hydride-precipitated layer are formed in the peripheral region, hoop stress concentrates on the tip of the cracks, one of the cracks develops and penetrates the cladding, i.e. fuel failure. The cladding ductility, or the amount of hydride precipitation, is an important factor of the PCMI-failure of high burnup fuels.

4. Development of new capsule

The hydrogen solubility in the cladding increases with the cladding temperature increase. In case of high temperature, the cladding recovers the ductility due to the decrease of hydride precipitation and the threshold of fuel failure may become higher than that at low temperature. As an RIA at hot zero power condition is assumed for PWRs in Japanese safety evaluation guideline [5], it is important to verify the high temperature effect on the hydride-assisted PCMI failure. A new capsule to achieve a high temperature condition which simulates hot zero power of PWRs was developed for the NSRR experiments.

The schematic diagram and the main specifications of the capsule are shown in figure 6 and table 2, respectively. The capsule is made of stainless steel and designed to be a double sealed structure for the confinement of high burnup fuels which have high radioactivity. The assembly and disassembly of the capsule is performed by a remote handling in a cell. Test fuel rod is set up in the inner capsule and the electric heater installed in the inner capsule achieves a high temperature condition (280 °C) in a few hours. The inner capsule has a capability to endure pressure and to prevent leakage. In the pressure-resisting design of the inner capsule, destructive forces are considered. The destructive forces include pressure pulses and water hammer forces generated by the fuel failure at RIA-simulating experiments in addition to the normal pressure. Within the pressure-resisting design, the thickness of the wall was decided as thin as possible to achieve the necessary neutron flux. The inside diameter of the inner capsule is enlarged to secure enough space for coolant water as a moderator of fast neutron which passed through the stainless steel wall. The pressure suppression tank is connected to the inner capsule, through a rupture disk which breaks at a certain pressure level. Hence, this tank has a function of pressure buffer against an abnormal pressure increase in the inner capsule. The outer capsule which contains the inner capsule and the pressure suppression tank functions as the backup airtight space against a leakage from the inner capsule and tank. Allowing the insertion of the capsule to the NSRR experimental cavity through the offset loading tube, the shape of the outer capsule are designed and the outside diameter and the height of the capsule are limited to 200 mm and 1200 mm, respectively. The thickness of the wall of the outer capsule is thinned within the pressure-resisting design. In order to keep the inner capsule in high temperature, the space between the inner capsule and the outer capsule can be made to a vacuum. The instrumentations listed in table 3 can be installed to the capsule.

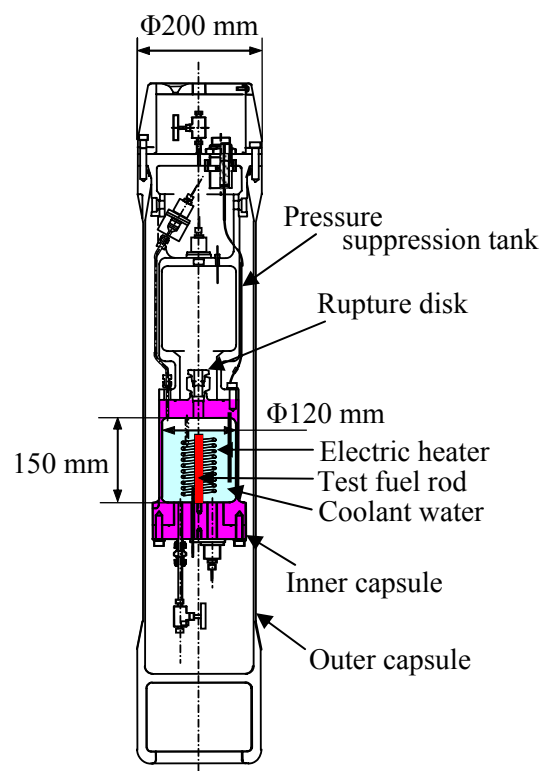


Figure 6: Schematic diagram of new capsule

The manufacture of the capsule was completed and the design requirements were confirmed by the performance tests of remote handling, leakage, temperature rising and so on. The new capsule has been developed successfully.

5. Summary

A new capsule which achieves high temperature condition simulating hot zero power of PWRs was developed successfully. Experiments with this new capsule will clarify the temperature influence on the PCMI failure limit of high burnup fuels, the cladding of which is embrittled due to hydrogen absorption.

Table 2: Specifications of new capsule

Inner capsule	Material	SUS304

	Outside diameter	Φ131 mm
	Inside diameter	Φ120 mm
	Inside length	150 mm
	Wall thickness	5.5 mm
Pressure suppression tank	Material	SUS304
	Outside diameter	Φ130 mm
	Inside diameter	Φ120 mm
	Inside length	147 mm
	Wall thickness	5 mm
Outer capsule	Material	SUS304
	Maximum outside diameter	Φ200 mm
	Wall thickness	3 mm
	Length	1200 mm

Table 3: Instrumentations of new capsule

- Cladding surface temperature - Coolant water temperature - Capsule internal pressure

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OPERATIONAL SAFETY, REGULATORY REQUIREMENTS, ADVANCES AND EXPECTED OUTCOMES IN THE GHARR-1 COMPUTERISED CONTROL SYSTEM UPGRADE PROJECT

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Abstract

A Technical Cooperation (TC) project to upgrade the micro-computerized control loop system (MCCLS) for the Ghana Research Reactor-1 (GHARR-1) facility has been approved by the International Atomic Energy Agency (IAEA). The project is designed to upgrade the computerized control system of the GHARR-1 facility in order to improve reactor operations, maintenance, safety and overall utilization of the facility. In this paper, the objectives of the project are presented. The effect of the facility upgrade or modification on the operating limits and conditions (OLC), operational safety regimes and the associated regulatory framework and requirements for the modification are also discussed. The advances achieved in the project implementation and the expected outcomes of the completed project are also presented

1.0. Introduction

The Ghana Research Reactor-1 (GHARR-1) facility is a commercial version of the Chinese-built Miniature Neutron Source Reactor (MNSR) which is similar in design and operating characteristics to the Canadian SLOWPOKE reactor. By design, it is classified as a tank-in-pool type, low power research reactor. The facility was acquired by the National Nuclear Research Institute (which is the operating organization (OO) of Ghana Atomic Energy Commission (owner) with the assistance of the International Atomic Energy Agency (IAEA) under project GHA/1/010 of the Agency's Project and Supply Agreement (PSA) protocols. Thus, the GHARR-1 facility's operations are subject to all the protocols of the Agency's PSA.

The facility was installed beginning October 1994 and achieved criticality on December 17, 1994 [1-3]. It was commissioned in March 1995 and has been safely operated, maintained, utilized and managed since in accordance with local and international safety and regulatory regimes and protocols.

The reactor is now being used mainly for training in nuclear science and technology using neutron activation analysis techniques. Various services are rendered to public and government agencies related to mining, industry, food and agriculture, nutrition, health

and environmental monitoring. Further institutional and human resource development was carried out under projects GHA/4/010 and GHA/4/011. Additional support is being derived from participation in the AFRA project RAF/4/016. The facility has made a strategic decision to enhance self-reliance and sustainability (under AFRA project RAF/4/016) by promoting income-generating activities. As a result of the development and implementation of a realistic business plan, the reactor is now being used to provide analytical services for soil mapping, mineral analysis for the mining sector, food and water analysis as well as environmental monitoring. The facility has therefore been or the utilized for research, training and commercial work since its commissioning. It has been used to generate income, which has offset its maintenance and operational costs.

However, in the course of facility operations, the micro-computerised control loop system (MCCLS) developed a couple of problems and was thus put out of service temporarily. This situation created a limitation in its utilization to satisfy the numerous clients and for teaching and education of the ever-increasing student users. For this reason, a technical proposal to modify and upgrade the GHARR-1 MCCLS for safe reactor operations and also enhance effective utilization via neutron activation analysis (NAA) was submitted to the IAEA to provide assistance within the Agency's Technical Cooperation (TC) framework. The technical proposal was accepted by the Agency in 2005 and referenced *GHA-4-012-001N/IAEA* under its Technical Cooperation (TC) assistance project [4]. It is implemented in collaboration with the Government of China. The project is presently under implementation and is expected to be completed by the close of 2007.

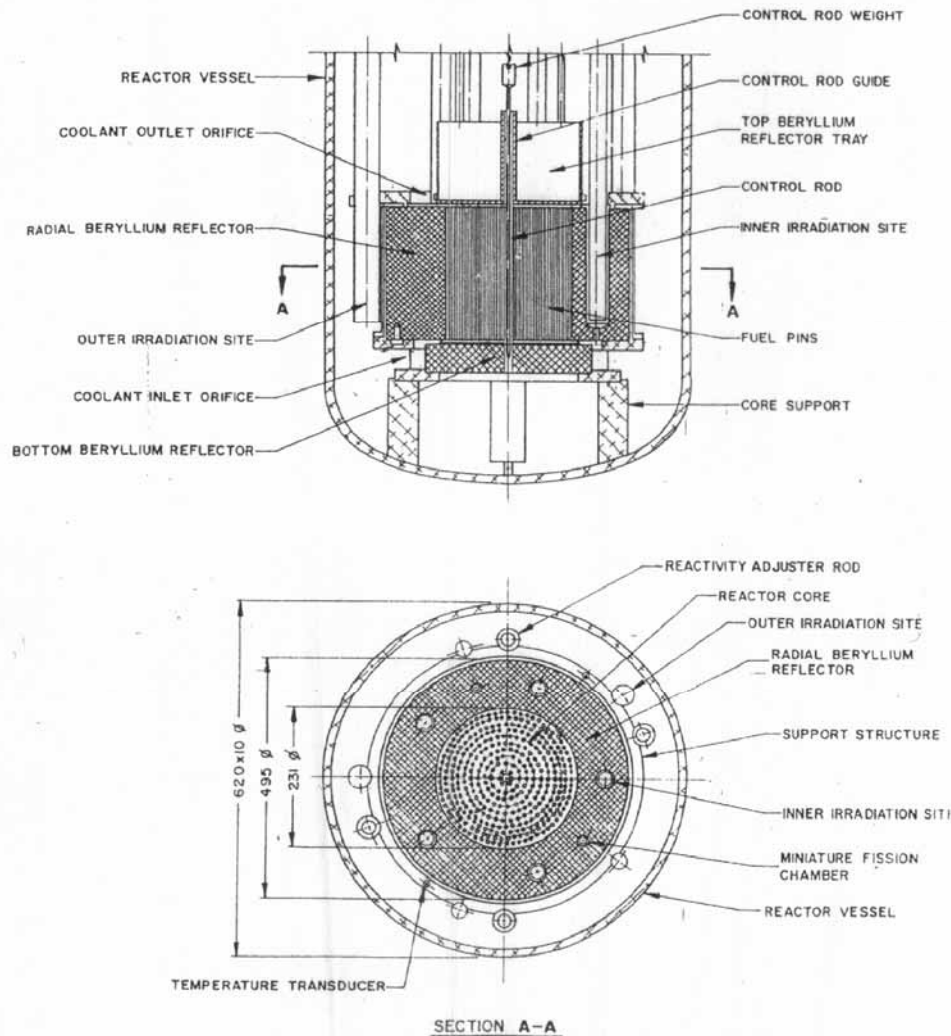
At the request of the Government of Ghana, an IAEA Reserve Fund Project was established to assist in upgrading the GHARR-1 to enhance its utilization for socio-economic development. Extra-budgetary contribution from the Government of China will provide for expenses related to equipment procurement and installation, and expert services.

The technical description of the GHARR-1 facility for which the project was designed is presented briefly following.

2. The GHARR-1 Facility

Technically, the GHARR-1 facility is operated at full power rated at 30 kW (th) with a corresponding peak thermal neutron flux of $1.0\text{E}+12 \text{ n/cm}^2\cdot\text{s}$ measured in the inner irradiation channels. Typically, MNSR reactors have large negative temperature coefficients of reactivity to boost inherent safety. The GHARR-1 core is located at the bottom of the lower section of the reactor vessel. The core consists of a fuel cage, a control guide tube and other structural components. A critical core was achieved with 344 fuel elements arranged concentrically in ten lattice zones about the central guide tube [1-3, 5]. Neutron reflection is provided on the side by beryllium annular reflectors and beneath the core by a beryllium slab a material. Cooling is by natural convection. No boiling is expected in the reactor during normal operations and under design basis accident (DBA).

The reactor is designed as a neutron source with ten irradiation channels located within and without the annular beryllium reflector [6-8]. For this reason, MNSR facilities are ideal irradiation facilities for performing neutron activation analysis. Additionally, the GHARR-1 facility is utilized for human resource development in nuclear science and technology, academic and professional teaching and training. A schematic drawing of the cross sectional view of the reactor is shown in Fig. 1.



A 3-D Monte Carlo plot of GHARR-1 core configuration showing fuel region (reactor core), channels for irradiation, fission chamber, regulating, slant and annular beryllium reflector are shown in Fig. 2 [9].

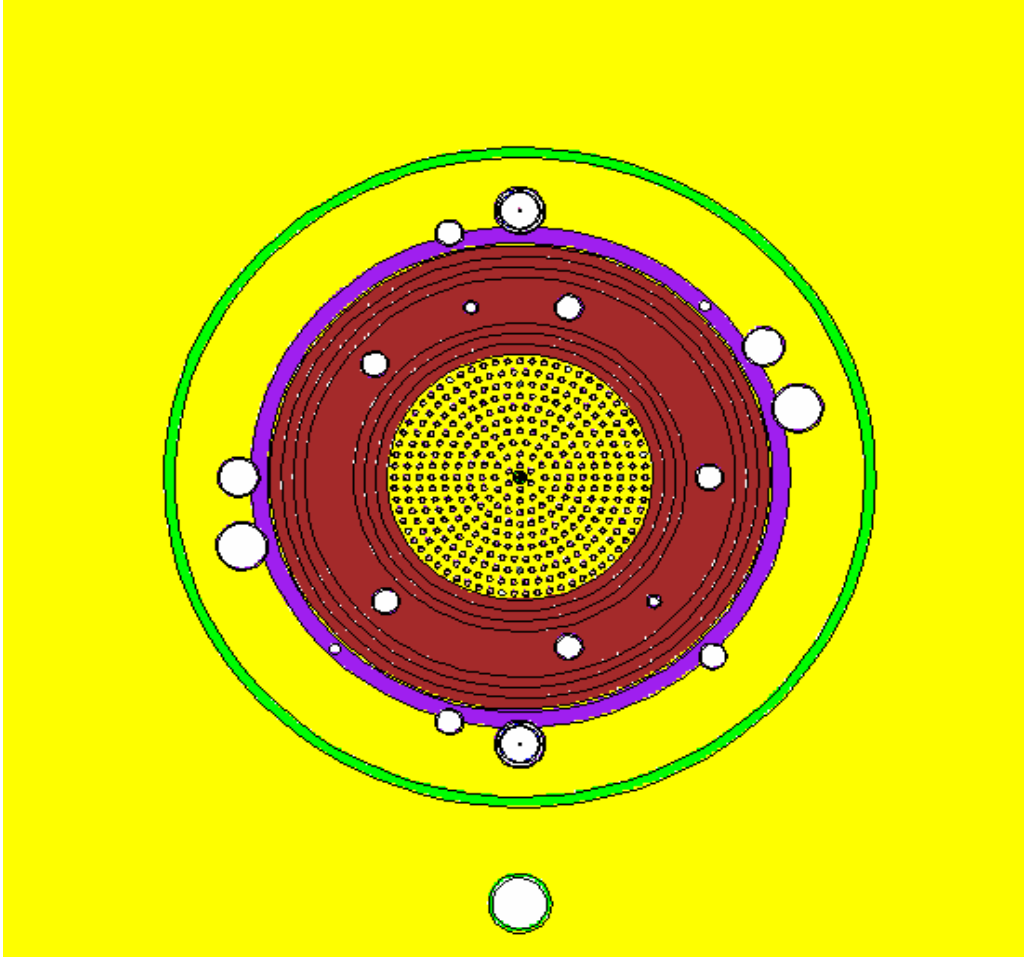


Fig. 2: MCNP5 plot of GHARR-1 core configuration showing fuel region (reactor core), irradiation channels, fission chamber, regulating and slant channels and annular beryllium reflector.

Control of GHARR-1 reactor operations are realized via the use of two operating systems: the control console and the micro-computer control loop system (MCCLS). Both systems perform complimentary roles when the other is being used for reactor control operations [1]. A brief description of the GHARR-1 control systems is presented in the next section.

3.0. GHARR-1 Reactor Control

The GHARR-1 control system includes control console and computerised control system, neutron detectors, control rod and its drive mechanism and associated electronics. The systems provide information on conditions for reactor operations [10-12]. The principal functions of the control system are; to start up the reactor in a safe and reliable manner, to maintain the reactor at a selected power, and eventually to shut down the reactor. Under-moderation of the core contributes to high negative temperature coefficient ($-0.1\text{mk}/^{\circ}\text{C}$). In addition the excess reactivity of the reactor is limited to $\rho_{\text{ex}} \leq \frac{1}{2} \beta_{\text{eff}}$ [6]. 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. For this reason, the control system is simplified.

Control is achieved either automatically or manually. The range of the reactor control mechanism for automatic control covers the range $10^8 - 10^{12}$ n/cm².s of neutron flux. The working range for NAA is limited to 10^{11} - 10^{12} n/cm².s while reactor physics experiments are performed in the range of 10^8 - 10^{12} n/cm².s. In particular, since the facility is primarily utilized for NAA, accurate control of the neutron flux is required. To achieve this, a set of control modes are adopted. The MCCLS is particularly designed to ensure a stable flux accuracy of $\pm 1.0\%$. Thus, no regulation is made by the system when a flux deviation $< \pm 0.5.0\%$ from the setting value is recorded. Quick responses and accuracy of the control systems ensure a stable flux within $\pm 0.5.0\%$ deviation.

As mentioned earlier, the IAEA TC project GHA-4-012 was approved to address the need to upgrade the GHARR-1 MCCLS which has been temporarily put out of action. The objective of the project is to upgrade the GHARR-1 to enable more efficient operation in support of human resource capacity building and activities aimed at income generation according to the business plan of the Ghana Atomic Energy Commission

The operational safety and regulatory requirements for implementation of the IAEA TC project is presented in the next section.

3.0. Operational Safety and Regulatory Requirements of Project GHA-4012

To ensure operational safety, the facility is operated, maintained and utilized in accordance and compliance with provisions and requirements of the facility's Safety Analysis Report (SAR) which is approved by the regulatory body. The facility's operational regimes also meet requirements of the regulatory body.

In fulfilment of operational safety and regulatory requirements, a safety assessment report of the proposed modification and upgrade of the GHARR-1 MCCLS was conducted by the GHARR-1 Reactor Physics, Operations, Engineering & Maintenance and Radiation Protection Groups and prepared facility management. The report was submitted for review by the GHARR-1 Reactor Safety Committee (RSC) which has responsibility of reviewing all reactor (including other instrumentation) operations and associated safety aspects of the GHARR-1 facility.

Further reviews of same were done by the IAEA to ensure that all safety aspects of the project were provided [13-16]. The safety aspects of the project implementation report included key issues such procedures approved by the national regulatory authority for modifications of the research reactor, reasons for the modifications initial safety analysis conducted for the modifications and impact of the project on the safety of the research reactor, the safety analysis, the safety analysis report, conceptual design of the modified system and verification of design criteria the training and qualification of operating and

maintenance personnel as reasons for equipment modification and upgrade, technical specifications and functions of components, and quality assurance, installation, commissioning plan for the modification

The regulatory body was duly informed of the project and the necessary safety aspects associated with it. Reports as required by the SAR were submitted for regulatory reviews and approval granted for the execution thereof.

4.0. Advances in Project Implementation

Following final reviews and acceptance of the project proposal by the RSC, regulatory body and the IAEA on the safety aspects of the project, the Agency granted approval for its implementation. A three-phase implementation plan covering purchase order for equipment, fellowship training and equipment shipping, equipment installation and commissioning activities was drawn in favour of executing the project [13-16].

The GHA-4012-001N/IAEA project implementation includes a contract for the supply of required and approved equipment for the GHARR-1 MCCLS. A new version of the MNSR computer control system called Type II will be supplied. It includes both hardware (PC, interface box unit and accessories, different types of detectors, etc) and software. In particular, an operating software which plays a decisive role in controlling precision, response time, stability and safety of the control system has been designed for the control system. Key parameters registered in the interface boxes include neutron flux (preset and measured), control rod position, reactivity, water temperature (inlet and outlet), pH and conductivities reactor and pool water systems, gamma dose distribution at different locations. In particular, the reactivities are calculated with neutron fluxes according reactor dynamic functions on real-time. The operating software collects and displays in real time these important listed reactor physics and operating parameters. It also compares the collected data with setting and limiting values defined in the operating limits and conditions (OLC) so as to determine whether regulation is required for flux stability. The software has capability for detecting motions of the control rod in conformity with given control signals. Finally, the MNSR operating software provides a record of the reactor operation including neutron integral flux, fuel burnup, duration of operation, etc.

To facilitate project implementation, the Agency has issued a procurement order to the China Nuclear Energy Industry Corporation (July, 2006) for the supply of these equipment required for the GHARR-1 MCCLS upgrade [14-15].

Another key feature in the advancement of the project is the training of two GHARR-1 operating, engineering and maintenance staff on the new MNSR MCCLS at the China Institute of Atomic Energy in China. The training will focus on the use of the new Type II MNSR MCCLS system for reactor control and operations. The training will also include troubleshooting problems, maintenance and installation procedures. It is anticipated by the middle of 2007.

Shipping of equipment for the project will be done after training at supplier's facility. This is anticipated about the end of the second half of the year. On-site equipment installation, testing of devices and reactor start-up and operation, maintenance training for other GHARR-1 Operations, Engineering and Maintenance team members is expected to be completed by the last quarter of 2007. The final stage of the implementation will include commissioning of the upgraded control system.

5.0. Expected Outcomes

With the completion of the project implementation, it is expected that the GHARR-1 facility will be operated with both the control console and the computerised control systems. The availability of the computerised system will enhance reactor operations record keeping, enhanced safe reactor operations, comparability of operating parameters for effective maintenance, stability of neutron flux readings and reliability of operating parameters, effective utilization via NAA.

An additional outcome of the GHARR-1 control system upgrade is the enhanced training of reactor of operations, engineering and maintenance staff. Flexibility of reactor operations is also anticipated.

6.0 Conclusion

The GHA-4-012-001N/IAEA project is an IAEA TC project approved for the modification and upgrading of the GHARR-1 MCCLS operating system. The successful implementation of the project will result in the deployment of a new, modified and upgraded computerised control system for the GHARR-1 facility. In particular, the deployment of this facility will enhance reactor operator training, improve access control and security in the control room. Effective utilization for NAA will also be improved since the computerized control system provides a relatively more stable neutron flux and which is most desired. An effective and realistic record keeping of reactor operations and associated reactor physics parameters will also be restored.

Acknowledgement

The authorship of this paper are most grateful to the IAEA for accepting and approving Technical Cooperation project to realize this activity. The authorship also acknowledges with gratitude the financial assistance of the Government of China and also for accepting to be a major partner in this project.

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Session III

Fuel Development

Characterization and Testing of Monolithic RERTR Fuel Plates¹

by

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CHARACTERIZATION AND TESTING OF MONOLITHIC RERTR FUEL PLATES

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ABSTRACT

Monolithic fuel plates are being developed for application in research reactors throughout the world. These fuel plates are comprised of a U-Mo alloy foil encased in aluminum alloy cladding. Three different fabrication techniques have been looked at for producing monolithic fuel plates: hot isostatic pressing (HIP), transient liquid phase bonding (TLPB), and friction stir welding (FSW). Of these three techniques, HIP and FSW are currently being emphasized. As part of the development of these fabrication techniques, fuel plates are characterized and tested to determine properties like hardness and the bond strength at the interface between the fuel and cladding. Testing of HIPed samples indicates that the foil/cladding interaction behavior depends on the Mo content in the U-Mo foil, the measured hardness values are quite different for the fuel, cladding, and interaction zone phase and Ti, Zr and Nb are the most effective diffusion barriers. For FSW samples, there is a dependence of the bond strength at the foil/cladding interface on the type of tool that is employed for performing the actual FSW process.

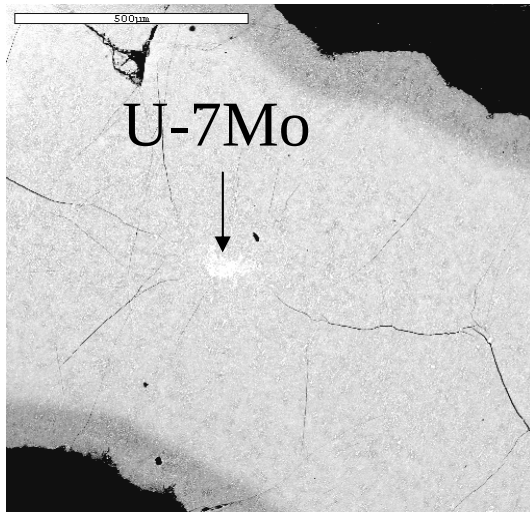
1. Introduction

Monolithic fuel plates are being developed as an LEU fuel for application in research reactors. [1]. To fabricate these fuels, three techniques have been evaluated: friction stir welding (FSW), hot isostatic pressing (HIP), and transient liquid phase bonding (TLPB), and the current emphasis is on FSW and HIP. As part of this evaluation, samples have been generated from fuel plates that were fabricated using the emphasized techniques, which were then characterized and tested. The characterization was performed using both scanning electron microscopy and optical microscopy. The testing was conducted using either hardness testing, which was performed on FSW and HIP samples, or pull testing, which was conducted only on FSW samples. The pull test was employed to investigate the bond strength at the foil/cladding interface in a monolithic fuel plate. This paper will discuss the results from the characterization and testing that was performed on various samples that were produced using FSW and HIP.

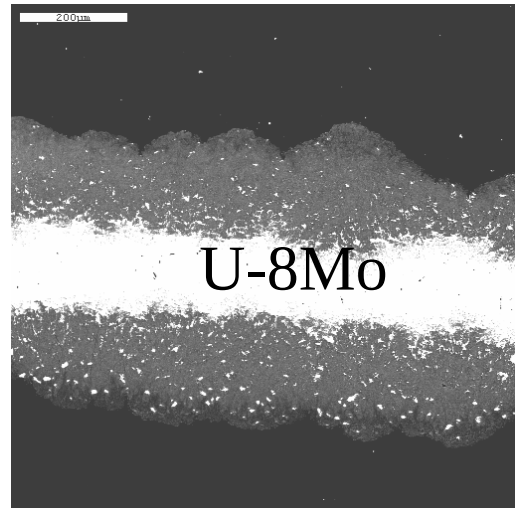
2. Results and Discussion

2.1 Hot Isostatic Pressing

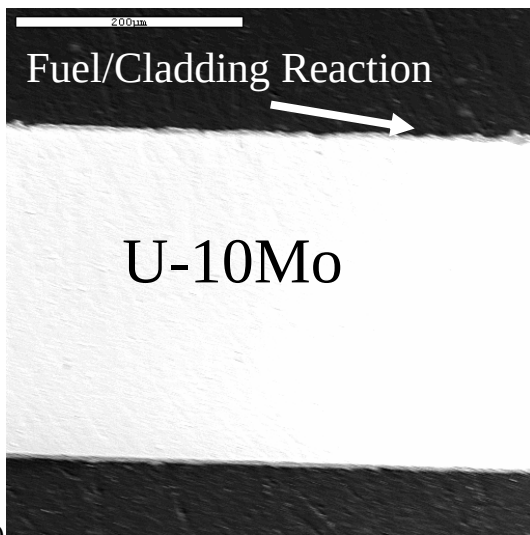
The HIP samples that were characterized were HIPed at 580°C for 90 minutes using a pressure of 15,000 psi. Figure 1 shows SEM micrographs for samples with 6061 Al cladding and U-x wt% Mo (x=7, 8, 10, and 12) fuel alloys. For the U-7Mo sample, the original fuel alloy is completely consumed. Another U-7Mo sample was run for 90 minutes at 580°C that was not completely consumed. Composition analysis of the completely consumed samples showed that the phases that formed were (U,Mo)Al₃ and (U,Mo)_{0.9}Al₄. Only a small region of original U-7Mo alloy remained at the center of the sample. For a U-7Mo sample HIPed at 580°C for half the time, the amount of reaction was less. This sample showed that the main phase to develop in the interaction zone is (U,Mo)Al₃, along with some minor phases at the reaction zone/6061 Al interface that contain component like Mg and Fe (see Figure 2). Nearest the unreacted U-7Mo is a multiphase zone that is commonly seen in diffusion couple experiments (see Figure 3) [2]. The



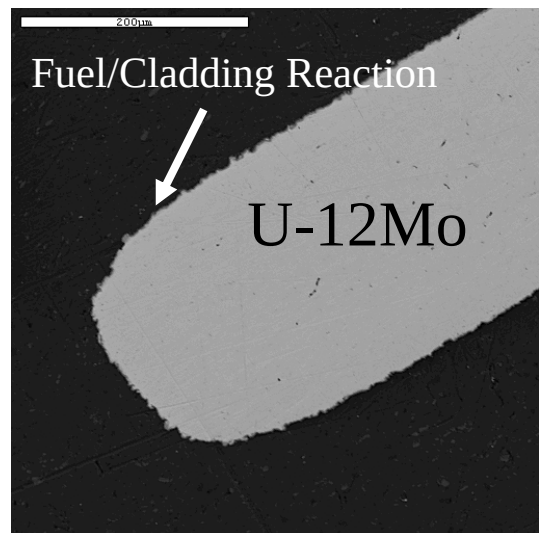
(a)



(b)

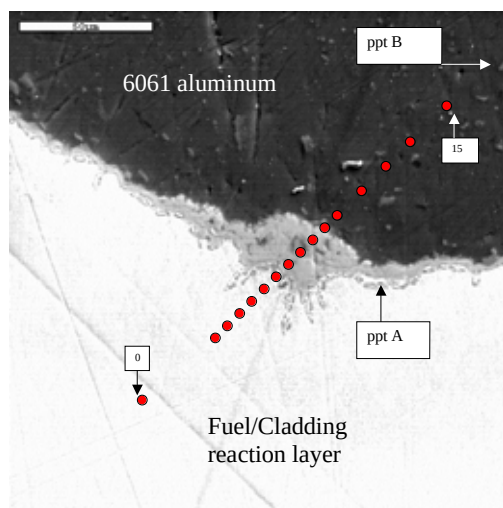


(c)



(d)

Figure 1. SEM micrographs of the (a) U-7Mo (580°C; 3 hrs), (b) U-8Mo (580°C; 1.5 hrs), (c) U-10Mo (580°C; 3 hrs), and (d) U-12Mo (580°C; 1.5 hrs) foils after HIPing. The black areas are 6061 Al cladding and the medium-contrast areas are interaction product.

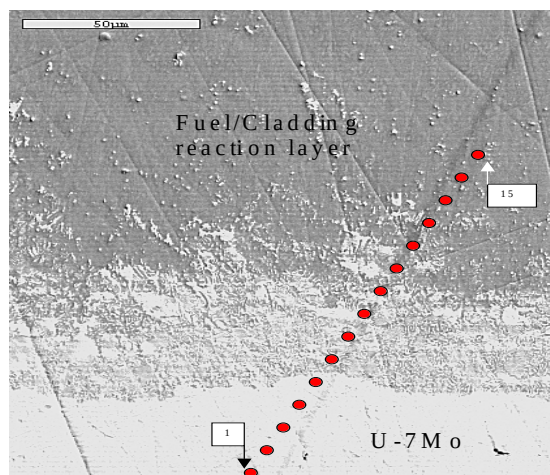


	U (at %)	Mo	Al	Mg	Si	Fe
0	20.44	2.46	75.69	2.58	-	-
1	19.14	2.57	74.71	2.22	2.09	-
2	17.33	2.30	75.95	3.42	1.57	-
3	18.79	2.86	75.66	2.09	0.68	-
4	16.50	1.86	73.66	3.35	4.95	-
5	15.33	3.90	74.53	2.46	4.32	-
6	8.19	2.94	71.54	13.55	0.44	3.33
7	6.58	0.41	79.13	2.06	-	12.61
8	3.73	0.25	90.41	4.23	-	1.84
9	5.48	-	78.86	2.23	0.29	13.51
10	-	-	97.23	3.20	-	-
11	-	0.33	92.44	3.74	-	2.42
12	-	0.35	96.62	2.00	0.82	0.25
13	-	0.11	98.09	1.6	0.34	-
14	-	0.03	98.88	1.29	0.07	-
15	0.03	-	97.31	1.3	1.39	0.13
ppt A	28.18	6.95	47.07	18.95	0.82	-
ppt B	-	0.18	92.59	1.01	2.78	3.67

(a)

(b)

Figure 2. (a) SEM micrograph of interdiffusion zone nearest the unreacted 6061 Al for U-7Mo plate HIPed at 580°C for 90 minutes at 15,000 psi. (b) Table showing compositions at various points in (a) going from 1 to 15.



	U (at %)	Mo	Al
1	80 .42	13 .73	5 .84
2	80 .35	15 .17	4 .49
3	86 .33	11 .51	2 .17
4	82 .92	10 .95	6 .13
5	81 .88	15 .02	3 .10
6	51 .69	5 .79	42 .52
7	49 .35	8 .13	42 .52
8	48 .52	9 .08	42 .39
9	43 .11	8 .45	48 .44
10	29 .80	4 .82	65 .38
11	19 .79	4 .63	75 .53
12	19 .75	3 .28	76 .96
13	20 .53	2 .11	77 .36
14	20 .31	4 .04	75 .65
15	21 .44	2 .72	75 .84

(a)

(b)

Figure 3. (a) SEM micrograph of interdiffusion zone nearest the unreacted fuel for U-7Mo plate HIPed at 580°C for 90 minutes at 15,000 psi. (b) Table showing compositions at various points in (a) going from 1 to 15.

U-8-Mo sample displays less interaction between the fuel and the cladding, but a significant amount of interaction product still forms, and a significant quantity of U-rich precipitates are distributed throughout the interaction zone. For the U-10Mo and U-12Mo samples, only a few-micron-thick layer forms at the foil/cladding interface. From looking at these different samples, it is clear that at 580°C there is a significant correlation between the Mo content of the fuel and the amount of foil/cladding interaction that occurs during HIPing. By looking at the U-Mo phase diagram [3], it can be seen that at around 562°C and at about 10.5 wt% Mo, a eutectoid is present. For the hypo-eutectoid U-7 and 8 wt% Mo alloy compositions, the alloys are in a two-phase region consisting of α -U and γ -U at 580°C. Yet, the U-10 wt% Mo alloy composition is very near the $\gamma/\alpha + \gamma$ phase field boundary and is probably γ -phase. The U-12 wt% Mo alloy is in the $\gamma + \delta$ two-phase region. When HIPing is performed at 560°C, all the alloys will be in the $\alpha + \delta$ two-phase region and how quickly the various alloys transform to these two phases will affect how quickly the alloys react with the 6061Al alloy cladding. Once α -U is present in the alloy microstructure, the amount of interaction should increase, since it has been shown that α -U reacts much more quickly with Al than does γ -U [2].

To reduce reaction between the fuel and cladding a variety of diffusion barriers (Zr, Nb, Ta, C, Si, and Ti) have been evaluated. Figure 4 shows a cross section of a sample where a Zr foil was employed as a diffusion barrier. Bonding between the Zr and the Al was good. Based on ultrasonic testing (UT), there was also bonding between the Zr and the fuel foil. Yet, during the destructive examination process the Zr de-bonded from the fuel foil. This indicates that the bond quality was relatively poor. Ti and Nb exhibited behaviors similar to that of Zr. Ni was tried as a diffusion barrier, but extensive interaction with Al to produce nickel-aluminides precludes using Ni as a diffusion barrier. For C and Mo, UT analysis indicates poor bonding between these materials and the foil and 6061 Al. Initial UT analysis suggests that Ta is a good diffusion barrier, but ongoing SEM examinations must be completed to confirm this result.

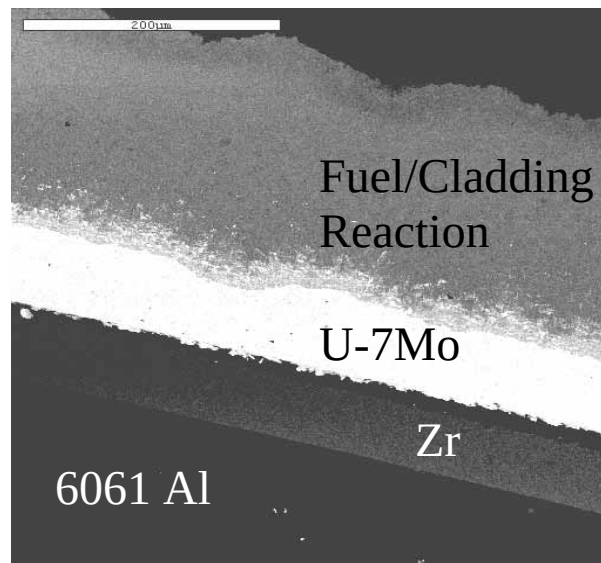


Figure 4. SEM micrograph of a U-7Mo fuel plate with a Zr liner used as a diffusion barrier that was HIPed at 580°C for 90 minutes. There is a gap between the Zr and the U-7Mo foil. Nb and Ti exhibited behavior similar to that of Zr.

Hardness values were measured for the fuel, cladding, and interaction phases for HIPed samples. For the fuel and cladding, measurements were made before and after HIPing. The initial Vickers hardness values before HIPing at 580°C for 3 hours for a 6061 T6 Al alloy and U-10Mo alloy were 107 ± 2 and 263 ± 4 , respectively. After HIPing, the 6061 Al alloy and the U-10Mo alloy registered Vickers hardness values of 47 ± 1 , and 288 ± 4 , respectively. The Vickers hardness values of the two uranium-aluminide layers that formed during HIPing were also measured. The (U,Mo)Al₃ phase had a value of 617 ± 3 , and the (U,Mo)_{0.9}Al₄ had a value of 653 ± 32 . Based on these results, it can be seen that after HIPing the 6061 Al alloy cladding is softer, while there is little change in the hardness of the U-10Mo alloy. The uranium-aluminide layers that are present after HIPing are much harder than the U-10Mo and 6061 Al cladding.

2.2 Friction Stir Welding

The strength of the bond at the foil/cladding interface is of particular interest in FSW samples. To investigate this parameter, pull tests of friction stir weld (FSW) monolithic fuel plate specimens were conducted with samples fabricated using two different materials as the tool face. A description of the pull test method and procedure may be found in [4], while a description of the FSW process may be found in [5]. The two tool facing materials are defined as Alloy A and Alloy B. In particular, Alloy A has a thermal conductivity of x , while Alloy B has a thermal conductivity of $2.5x$. Thermal conductivity is an important parameter in the FSWing of monolithic fuel plates since increased heat removal away from the weld surface allows higher loads to be applied, resulting in enhanced bonding. For example, if a low thermal conductivity material is used as the tool face, less heat will be removed from the weld face, and in order to avoid a corresponding increase in process temperature and void formation, the applied load must be decreased. Diffusion of atoms across the fuel-cladding interface that results in adequate bonding is driven by two mechanisms: 1) temperature to drive the kinetic reaction and 2) applied load to keep the interface in intimate contact with one another allowing diffusion to occur. Applied load is especially important with the FSW process, since the aluminum cladding is in a plastic state and stirs on top of the fuel foil. Lower applied loads do not result in the intimate contact across the interface therefore limiting the diffusion of atoms, even if the temperature is sufficiently high. This hypothesis is shown graphically in Figure 5, where stress-time plots for Alloy A and Alloy B are provided. Analysis of Figure 5 shows that Alloy B, the higher thermally conductive material, has superior bond strength (greater than 7 times) than that of Alloy A. Reasoning behind this observation is Alloy A, as discussed above, produces mainly a mechanical bond, while Alloy B produces a diffusion enhanced bond, similar to observations made for the HIPing process. It is important to note that the pull test was terminated for the Alloy B specimen not because the foil-clad interface failed, but because the epoxy used to bond the test specimen to the aluminum platens for pull testing failed.

4. Conclusions

Based on characterization and testing described above for monolithic fuel plates fabricated by FSW and HIP, the following conclusions can be drawn:

1. The amount of interaction that will occur during HIPing of monolithic fuel plates will depend on the Mo content of the fuel foil, the temperature and time that is used, and whether or not a diffusion barrier is applied.
2. The hardness of the 6061 Al alloy cladding will decrease during HIPing, while the U-10Mo will maintain about the same hardness. Any uranium-aluminide layers that develop will be relatively hard.

3. Based on pull tests performed on FSW samples, there is a dependence of the bond strength of the fuel plate on the type of FSW tool that is used to fabricate the plate.

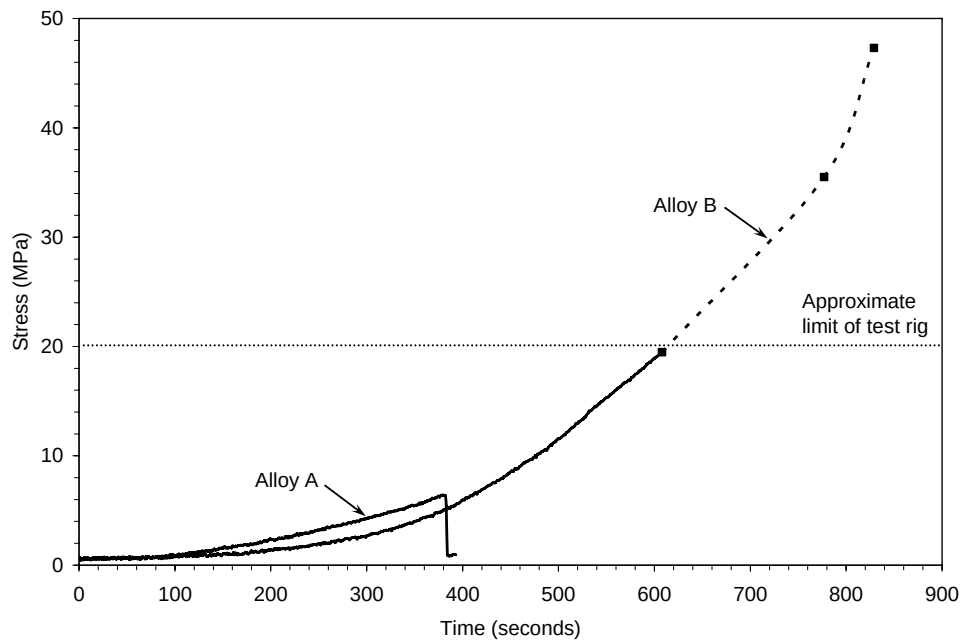


Figure 5. Stress-time plots obtained from pull test specimens fabricated employing FSW tool face materials Alloy A and Alloy B.

Acknowledgments

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2007 REPORT ON DEVELOPMENT PROGRESS ON LEU FUELS AND TARGETS IN ARGENTINA

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ABSTRACT

Since last RRFM meeting, CNEA has continued on new LEU fuel and target development activities as a part of the national policy on minimization of the use of HEU on civilian applications. Main goals are the plan to convert our RA-6 reactor from HEU to a new LEU core, to get a comprehensive understanding of U-Mo/Al compounds phase formation in dispersed and monolithic fuels, to develop possible solutions to VHD dispersed and monolithic fuels technical problems, to optimize techniques to recover U from silicide scrap samples as cold test for radiowaste separation for final conditioning of silicide spent fuels. and to improve the diffusion of LEU target and radiochemical technology for radioisotope production. Future plans include

- Completion of the RA-6 reactor conversion to LEU
- Improvement on fuel development and production facilities to implement new technologies, including NDT techniques to assess bonding quality.
- Irradiation of miniplates and full scale fuel assembly at RA-3 and plans to perform irradiation on higher power and temperature regime reactors
- Optimization of LEU target and radiochemical techniques for radioisotope production.

1. Introduction

According to the Argentina state policy to support the GTRI initiative, the non-proliferation treaty and the minimization of the use of high enrichment uranium for civilian uses, several activities and goals since the last SOFIA RRFM meeting, were carried out and achieved

- Improvements made on the RA-6 reactor conversion to LEU core.
- A comprehensive understanding of U-Mo/Al alloys interaction zone formation in dispersed fuels,
- Development of promissory solutions to VHD monolithic and dispersed fuels technical problems,
- Optimization of cold test on radiowaste separation for final conditioning of silicide spent fuels by recovering U from silicide and U-Mo monolithic fabrication scrap samples.
- Improvements on the diffusion of LEU target and radiochemical technology for radioisotope production.

2. RA-6 conversion

The RA-6 experimental reactor is a pool-type 0.5MW power in Bariloche Atomic Center, Provincia de Rio Negro, Argentina. It is dedicated to human resources formation as a part of the curricula of nuclear engineering in the Instituto Balseiro. Since 2002 it provides appropriate epithermal neutron flux for experimental BNCT (Boron Neutron Capture Therapy) techniques applied to melanoma cancer patients. It also has several NAA (neutron activation analysis) implementation and facility. At present and since its inauguration in 1983 it is working with a HEU core. The conversion project and the adhesion to the SNFFRR Acceptance Program is the result of an agreement between US NNSA DOE and CNEA. Respective contracts have been signed. Several tasks are covered by the project in progress:

- Argentina as a partner of the GTR Initiative has decided to minimize the HEU inventory through a swapping operation of HEU (Arg)-LEU (USA) materials, done on August 2006.
- CNEA's nuclear fuel reactor unit has calculated the neutronic and thermal-hydraulic new design and the PSAR report presented to the regulatory body.
- CNEA's nuclear fuel unit has designed and started the fabrication of the new U₃Si₂ fuel based LEU core and neutronic reflectors.
- CNEA's waste management unit started the construction of interim facilities to store LL and MLW .

Other tasks are foreseen for the next months:

- The removal of HEU core and condition for transportation
- Loading of the RA-6 core with the new one.
- Cask loading and transportation to exportation port

3. Very High Density fuel development

An intensive activity both on dispersed and monolithic VHD fuels are taking place

- U-Mo based monolithic miniplates using Zry-4 cladding were irradiated as a part of the RERTR 7 experiment at ATR reactor (INL-USDoE)ⁱ. One of the specimens achieved a burnup of 37% and PIE experiments showed a good behavior during irradiation. A contract between INL-DoE and -CNEA is operative for the supply and irradiation of several Zry-4 cladding –UMo and UZrNb monolithic miniplates and plates, as a part of the RERTR 9 and AFIP experiment. One of them achieved 80%
- ..
- An IAEA technical cooperation project, named ARG/4/092, related to the irradiation of a full scale fuel assembly of UMo-Al-Si dispersion in a high flux reactor is on progress.
- FSW bonding techniques: CNEA continued the development of this technology for Al cladding-monolithic metal welding. Nowadays, destructive and non destructive testing (NDT) techniques are under implementation to assess the bonding quality during production.
- An irradiation programme at RA-3 reactor is on progress. At the central part of the irradiation box where a thermal neutron flux of 2×10^{14} n/cm²/s is

achieved. The irradiation of diffusion couples to estimate the influence of irradiation on interaction zone growth is developing as planned. PIE experiments of the miniplates will start as soon as they arrive to the Hot Cell Facility. A special work will be presented during technical sessions. As a part of this program, fabrication of full scale fuel plates of UMo-Al-Si dispersion and irradiation in the RA-3 reactor are in progress.

- Codes PLACA/DPLACA designed to describe the irradiation behavior of plate-type fuels under normal operation conditions were developed. They contain about thirty interconnected and mutually dependent models. The growth of an interaction layer around fuel particles in dispersed fuels was modeled assuming diffusion of U and Al through the layer, numerically solving associated Stefan problems. Comparison of the calculation results to the experimental data showed a correct performance of the models involved and a good coupling of the ensemble. A poster will be presentedⁱⁱ

4. Characterization of phases in U-Mo/Al alloys interaction zone

- Theoretical calculation using BFS (Bozzolo, Ferrante, Smith, 1992) method for the determination of the atomic system energy as a function of its geometrical configuration were performed. This provides a virtual model of an alloy formation process. This methodology applied to UMo/Al-Si alloys predicts that the presence of Mo inhibits but does not avoid the Al diffusion and the tendency to compound formation. Mo-Si interaction inhibits Si diffusion. Also predicts Si atoms migration from Al alloys to interaction zone, the diffusion of Si in U if Mo is not present and Si inhibits Al diffusion only in presence of Mo. This methodology suggests the formation of a phase of the type $(U-Mo)-(Al,Si)_3$.
- Characterization of phases in UMo/Al alloys interaction zone: during 2006, work was related to study the effect of the consumption of Si in the Al alloy while interdiffusion is in progress. U-7w%Mo / Al-7w%Si diffusion couples, were made by FSW techniques. In addition to the results previously reported at 550°C, prolonged treatments at 340 °C were performed and analysed. Results do not show clearly that the condition of no Si in the Al alloy was reached during this last treatment. The irradiation plan continues as scheduled. PIE are intended to be started in 2007 fall.ⁱⁱⁱ
- $U(Al,Si)_3$ stabilization by Zr addition: promising results have already been obtained showing that Si addition to Al matrix is able to inhibit UAl_4 formation. It was also noticed that minor Si quantities should be required in the presence of fourth element collaboration: Zr is an already suggested candidate. Stabilization experiments in 50U49.9Al0.1Si alloys with 0, 1, 3 and 6% Zr addition (weight percentages) were performed. Heat treatments at 600°C (100h and 1000h) were undertaken and results were analyzed by x-Ray diffraction, metallographic and composition measurement techniques. Slight evidence of UAl_4 presence was still found in heat treated samples with 6%Zr content, although UAl_4 related x-Ray peaks intensities diminish with higher Zr contents. It suggests that this way enables the retard of the peritectic reaction that leads to UAl_4 formation.⁵

5. Cold test for silicide and U-Mo Zry-4 coated SNF final conditioning

- Silicide production scrap recovery: the development presented in former RERTR meeting was finished. It is an important step to have available a process to separate actinides and some radionuclides for nuclear medicine applications for final conditioning of spent fuels. U-Mo monolithic Zry-4 coated scrap recovery: first results using a chopping machine to cut, process and U recovering were satisfactory^{iv}

6. Improvement of the LEU target and radiochemical technology for Mo99 and other radioisotopes production.

It is not possible to fulfill the aim of the minimization of the use of HEU on civilian application if the production of Mo99 and other fission radioisotopes is not converted. In that sense

- CNEA has decided on 2001 to turn into LEU material for target fabrication. It was done maintaining other characteristics of the production, i.e. the alkaline chemical process. CNEA achieved successfully an adequate replacement meat.
- This LEU technology satisfies the most stringent requirements of quality for its use in nuclear medicine applications. ANSTO (Australia) and EAEA (Egypt) adopted it.
- Since September 2005, CNEA began the regular production of high quality fission I-131, a by-product of Mo-99 production, meeting international quality standards. This use recovers from the waste stream a valuable product.
- CNEA has a project in progress to optimize the Mo99 production (CiMo99/grU).
- CNEA-US DoE collaboration on LEU technologies: during Dec. 2006, USDoE monolithic targets and a process digester were tested at the RA-3 reactor and production plant. The behavior of these new targets were promissory.
- A project to recover irradiated HEU, blend it down to LEU and separate Cs137 and Sr90 is ongoing. The separation of Sr-90 will be used in Y-90 generators. Both radioisotopes are of interest for nuclear medicine applications.
- CNEA is participating in the IAEA Coordinated Research Project on developing techniques for small scale indigenous LEU Mo⁹⁹ production as an agreement holder

7. Conclusions

CNEA continued deploying an intensive activity on R&D on RERTR technologies, as a part of the national policy on GTRI partnership, non-proliferation and HEU minimization. Concerning VHD fuels, we focused our work some promissory lines for technological solutions both on dispersed and monolithic fuels in the range of 8-16 gU/cc. Concerning LEU technologies for radioisotope production we are deeply involved on its development and diffusion of the technology. Future plans include:

- Completion during 2007 the RA-6 reactor conversion to LEU
- Improvement on fuel development and production facilities to implement new technologies, including NDT techniques to assess bonding quality.
- Irradiation of miniplates and full scale fuel assemblies in RA-3 and in high flux reactor, both dispersed and monolithic fuels.
- Optimization of LEU target and radiochemical techniques for radioisotope production

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^{iv} Gauna A: personal communication

FABRICATION OF A MONOLITHIC UAl_2 ROD BY THE POWDER METALLURGY METHOD

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ABSTRACT

It is known that UAl_2 , a cubic-structured inter-metallic compound with a high melting point, has a very stable in-reactor behavior up to a high burn-up for an irradiation such as a nuclear fuel in a reactor. Monolithic UAl_2 has been reported as being fabricated by a melting method. Uranium powder, which can be obtained easily by the atomization process developed by KAERI, was used to fabricate Al_2 by using powder metallurgical technology. Uranium powder and Al powder were blended, extruded, and annealed into a UAl_2 rod in a mould. The phenomenon, where the density of UAl_2 is lower than the density of the blended and extruded rod for the U and Al powders, was taken into account. Excess volume of the mould was adjusted during the annealing process in order to avoid a cracking. Some sound UAl_2 rods were successfully fabricated by the powder metallurgical process. The best relative density of the UAl_4 formed by an annealing was at about 94%. The density increased with tighter constraints on the mould and a smaller particle size of the uranium powder. A coarse uranium powder of an average of 80 μm in diameter represented the remaining un-reacted uranium phase, which can have a harmful effect during an irradiation. On the other hand, a fine uranium powder of an average of 50 μm in diameter could achieve an almost pure UAl_2 phase without a uranium phase. The analysis by an X-ray diffraction pattern confirmed that the annealed specimens had interacted to form a UAl_2 phase. The thermal conductivities for the annealing-formed UAl_2 by the laser flash method were from 9 to 12 W/m/K, which is considered to be reasonable. In general, the above results for the annealing-formed UAl_2 rod are expected to be useful in developing certain advanced fuels.

1. Introduction

UAl_x powder dispersed in an aluminum matrix, which is made by alloying uranium and aluminum, has been widely used for the plate-type fuel elements of research and test reactors[1]. UAl_x is a mixture of two-or three uranium-aluminum inter-metallic compounds, i.e., UAl_2 , UAl_3 , and UAl_4 . The fractions of them are determined by the uranium-aluminum composition and the fabrication process parameters. It is known that U-Al alloys containing less than 35 wt % uranium are of great interest for use in research reactors, whereby the best U-Al alloy has 14-16 wt% uranium[2].

UAl_x of UAl_2 , UAl_3 , and UAl_4 has been reported to have a very good irradiation behavior. After a fuel with an alloy of U-19 wt%Al, which is the composition of UAl_2 , was irradiated up to a 60 at% burnup of a 93% enriched uranium fuel, it was observed that no fission gas bubbles appeared in the UAl_2 , UAl_3 , and UAl_4 phases and the resultant fabrication voids and cracks were found to decrease in number as well as in size by increasing the burnup rate[3].

The structures of UAl_2 and UAl_3 are isotropic just like a $MgCu_2$ face centered cubic and $AuCu_3$ Cubic structures with relatively high melting temperatures of 1590 °C and 1350 °C, respectively. However, UAl_4 is anisotropic with an orthorhombic structure with a low decomposition temperature of 731 °C. Uranium densities of UAl_2 and UAl_3 , which are 6.64 g-U/cc and 5.08 g-U/cc, are considered to be

applicable for reactor fuels. In addition, UAl_2 and UAl_3 does not have a solid solution range, while UAl_4 has a small solid solution range[4]. From the above aspects, monolithic UAl_2 is considered to be the most valuable, so an effort has been made to develop the relevant fabrication technology.

Monolithic UAl_2 has been reported as being fabricated by an arc-melting method, which is difficult when producing a rod type fuel. KAERI has developed a rotating disk atomization process to fabricate a uranium alloy powder. So the utilization of an atomized uranium powder has been pursued by using powder metallurgy technology. Relatively fine uranium powder was used. Blending, compaction, and extrusion were implemented in the same manner as the HANARO fuel fabrication process[5]. The interaction of the uranium particles and the aluminum particles was achieved in a confined mold at a high temperature for a long enough time. The phenomenon, where the density of UAl_2 is lower than the density of the blended and extruded rod for the U and Al powders, was taken into account. Therefore an excess space inside the mould was used during the annealing process in order to avoid a cracking.

In this study on SEM observation of the cross section of the UAl_2 specimen was conducted to observe the fabrication voids and cracks as well as the remaining uranium phases in the monolithic UAl_2 rod. The densities of the formed UAl_2 rods were measured by the immersion method. The presence of the UAl_2 phase was confirmed for the annealed specimen through an X-ray diffraction analysis. Thermal conductivities were measured and discussed from the viewpoint of nuclear fuel candidates. This study also was focused on an examination of the soundness of the formed UAl_2 specimen and the feasibility of a monolithic UAl_2 fuel for nuclear reactors.

2. Experimental

The uranium powder, produced by an atomization process, was blended with Al 1060 powder by a V type mixer and compacted by a hydraulic pressure. The compact was extruded in to a rod at 400°C. And then the rod was machined into specimens of 12.5 mm in diameter and 10 mm in length. Annealing treatment was implemented for the bare specimen at 1100 °C for 10 hours under an inert atmosphere of argon gas. The extruded specimen was crumbled during the heat treatment process. Therefore various kinds of graphite moulds as shown in Figure 1 were implemented. The extruded specimens made with atomized uranium powder were constrained by using graphite moulds. Annealing treatment was implemented in two steps, which was a pre-treatment at 700 °C for 10 hours in order to maintain a solid state and a final treatment for 10 hours at three different temperatures of 1100, 1200, and 1300 °C. Two different particle sizes of the uranium powder, which were an average of 50 μm and 80 μm in diameter, were used to investigate the effect of a particles' size. In order to improve the soundness of the formed UAl_2 from the viewpoints of the fabrication voids and cracks, a tight zirconium mould as shown in Fig. 2 was facilitated with a finer uranium powder of about 50 μm in particle diameter. Zircaloy mould was vacuum-sealed by an electron beam welding.

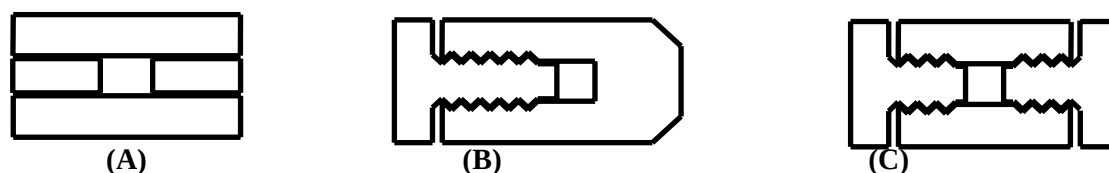


Fig.1. Graphite Mould; (A), (B), (C).

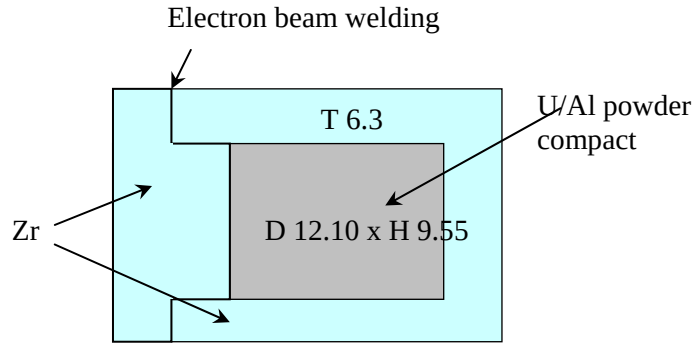


Fig 2. Zircaloy mould

Specimen was cut and polished on the surface in order to examine the microstructure of the formed UAl_2 structure more accurately. SEM observation was used to examine the fabrication voids and cracks. Phase analysis was conducted by using the X-ray diffraction method. Density for the specimen was measured by an immersion method. A specimen was machined into thermal conductivity specimen dimensions and then its' thermal conductivities were measured by the laser flash method

3. Results and Discussion

An annealing experiment for an extruded rod specimen of a blended powder compact was conducted without a mould constraining the specimen. The specimen was crumbled during an annealing. It could be attributed to the volume expansion induced from the low density phase occurrence and the Kirkendall void formation due to the considerable diffusivity difference between U and Al. Also the weak strength of the extruded rod from an insufficient bonding could also contribute to this crumbling. The specimens in the graphite moulds did not break up during an annealing. UAl_2 specimens annealed at three different temperatures for the final annealing treatment were examined by using their X-ray diffraction analysis results as shown in Fig. 3. Uranium phase was more apparent at 1100 °C than at 1200 °C and 1300 °C. Inclusions such as carbon were enhanced more at 1300 °C. The optimum annealing temperature was considered to be 1200 °C.

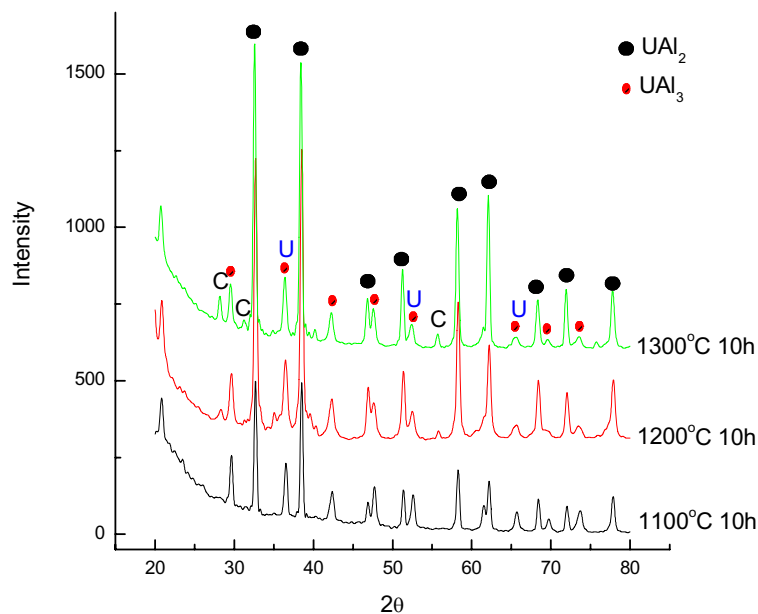


Fig. 3. X-ray diffraction patterns of the UAl_2 specimens annealed at three different temperatures for a final annealing treatment.

The measured relative densities of the annealed specimens are shown in Table 1. The densities seem to increase with a tighter constraint of the moulds because the density for a graphite mould (c), which is constrained in both directions, is higher than the other densities for the other used graphite moulds and the density for the zircaloy mould without an excess space in the mould is the highest. The densities of the UAl_2 specimens which used a fine uranium powder revealed higher densities than those which used a coarse powder. It is assumed that the finer powder has more effect during the formation of a denser UAl_2 .

Table 1. Density variation of the annealed UAl_2 specimens for different moulds and particle sizes of U powders

		Graphite Mould (A)	Graphite Mould (B)	Graphite Mould (C)	Zircaloy Mould
Density	Coarse U powder	6.09 g/cc	7.46 g/cc	7.55 g/cc	
	Fine U powder	7.24 g/cc		7.62 g/cc	7.68 g/cc
U density	Coarse U powder	4.96 g-U/cc	6.08 g-U/cc	6.14 g-U/cc	
	Fine U powder	5.89 g-U/cc		6.21 g-U/cc	6.26 g-U/cc
Relative density	Coarse U powder	74.9 %	91.7 %	92.7%	
	Fine U powder	88.9 %		93.6 %	94.4%

Typical micrographs for the cross section of the UAl_2 specimen formed by an annealing, for which a coarse uranium powder was used, are shown in Fig. 4. Un-reacted uranium phase remained in the regions distributed with a little higher fraction of the uranium powder. Also UAl_3 phases and fabrication voids were found in some aluminum rich regions.

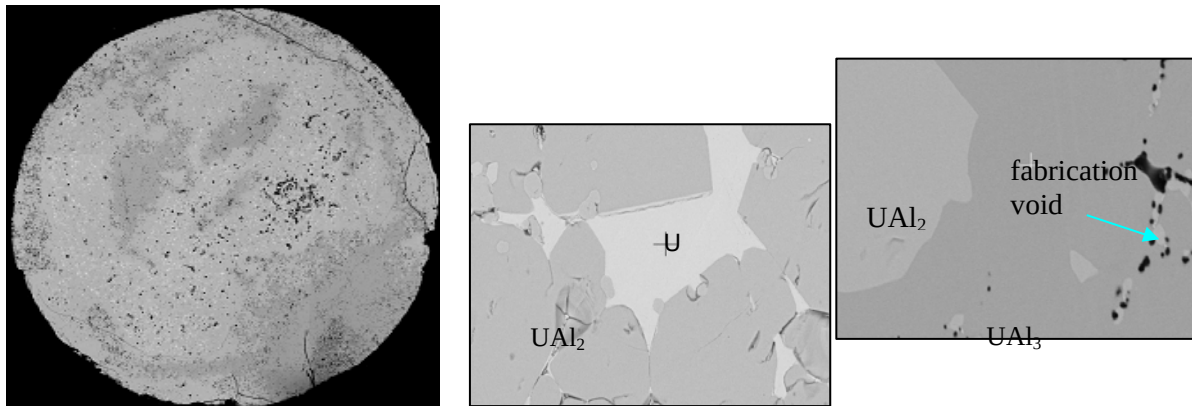


Fig. 4. Typical micrographs of the UAl_2 specimen formed by an annealing with a coarse uranium powder

Fig. 5 shows a typical micrograph of the UAl_2 specimen formed by an annealing with a fine uranium powder. Un-reacted uranium phase can not be seen in the micrographs. This phenomenon resulted from a smaller diffusion distance of the smaller uranium particles with aluminum. It is considered that 10 hours of an annealing time for the fine uranium powder of an average of 50 μm in diameter seems to be enough. Fabrication voids are also found in the UAl_2 specimen with a fine uranium powder. However, the frequency of finding fabrication voids on a typical micrograph in case of using a fine powder is lower and the size of the voids was observed to be smaller than in using a coarse powder. Presumably the fabrication voids could act as a beneficial role in compensating for a swelling during an irradiation in a reactor. The micrograph of the UAl_2 specimen formed by an annealing with a fine uranium powder using a zircaloy mould is shown in Fig. 6. According to the figure some different microstructures were observed. In the center region, the UAl_2 grains were separated and voids

occurred at the interface. On the other hand, in the periphery region the UAl_2 grains were tightly in contact with each other even though some large voids existed at the interface. It is construed that a tensile stress in the center region and a compressive stress in the periphery region prevailed. The reason for this would be that the zircaloy mould would impact more so on the periphery region thus blocking an expansion of the specimen. When this material is used for a nuclear fuel, the fabrication voids would reveal a compensation role during an irradiation. Accordingly the excess space in the zircaloy mould, which affects the fabrication voids, needs to be controlled from the viewpoint of a swelling compensation.

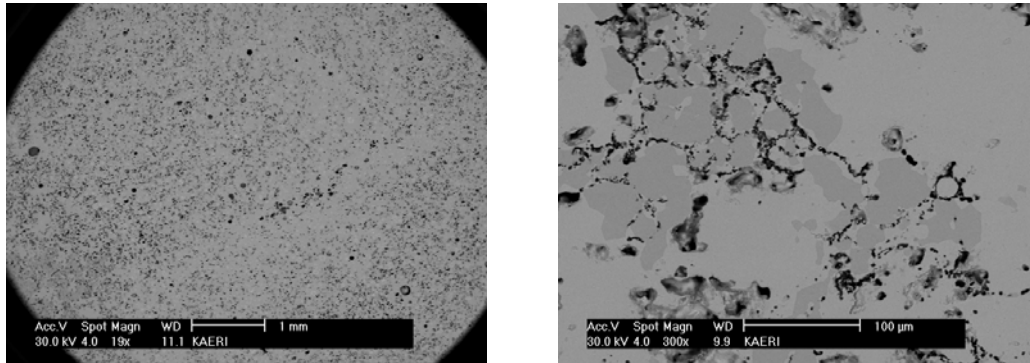


Fig. 5. Typical micrographs of the UAl_2 specimen formed by an annealing with a fine uranium powder

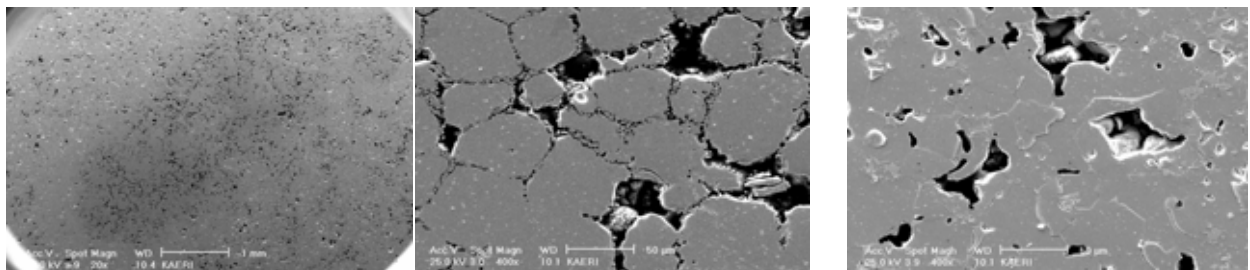


Fig. 6. Micrographs of the UAl_2 specimen annealed in the zircaloy mould; (A) whole cross-section, (B) center region, (C) periphery region

A typical X-ray diffraction result is shown in Fig. 7. Only UAl_2 peaks appeared in the X-ray diffraction results for the fine uranium powders of both moulds. In the irradiation test conducted by ANL the remaining uranium phases behaved as nucleation sites for the fission gas bubbles thus, the process parameters should be optimized to eliminate the uranium phases as much as possible. Accordingly a fine uranium powder is preferable.

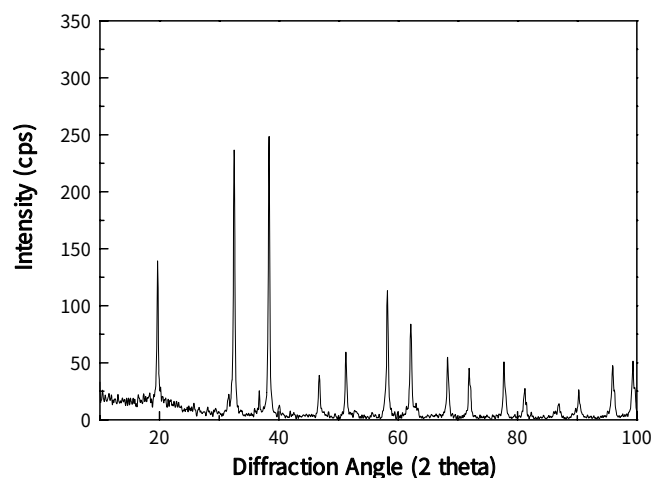


Fig. 7. A typical X-ray diffraction result for the UAl_2 specimen formed by an annealing.

Thermal conductivities measured for the UAl_2 specimen formed by an annealing with a coarse uranium powder and a graphite mould (C) are represented in Fig. 8. UAl_3 's thermal conductivity was varied from 8 to 15 W/m/K, while the annealing-formed UAl_2 revealed thermal conductivities which ranged from 9 to 12 W/m/K. These thermal conductivities are not bad for a nuclear fuel. This kind of annealing-formed UAl_2 is thus considered to be a valuable candidate for a future nuclear fuel by taking into consideration a fuel meat's geometry.

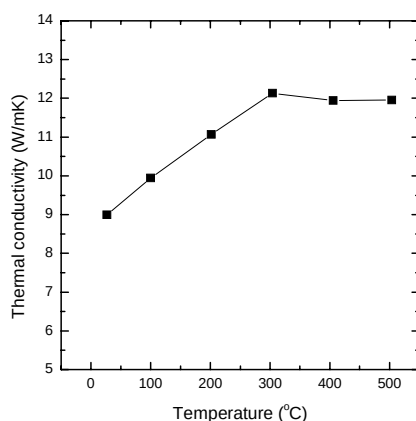


Fig. 8. Thermal conductivities measured for the UAl_2 specimen formed by an annealing at 1200 °C for 10 hours with a coarse uranium powder and a graphite mould (C)

4. Summary

Monolithic UAl_2 rod could be fabricated by the powder metallurgy method, which consisted of blending U powder and Al powder, with a compacting, extruding, moulding, and an annealing at 1200 °C for 10 hours. The maximum density of the UAl_2 specimen was 7.68 g/cc in density with 94.4% of a relative density. The density increased with tighter constraints of the mould and a smaller particle size of the uranium powder. A coarse uranium powder of an average of 80 μm in diameter remained in the un-reacted uranium phase, which has a harmful effect during an irradiation. On the other hand, a fine uranium powder of an average of 50 μm in diameter could achieve an almost pure Al_2 phase without a uranium phase. Fabrication voids seemed to be varied with the excess space of the mould. These fabrication voids could be used from the viewpoint of a compensational effect for fission gas bubbles by controlling the excess space of a mould. The x-ray diffraction results confirmed that all the annealed specimens fabricated by using the fine uranium powder were properly incorporated in the UAl_2 phase. The annealing-formed UAl_2 revealed a little lower thermal conductivity range from 9 to 12 W/m/K. These thermal conductivities are not bad for a nuclear fuel. This kind of annealing-formed UAl_2 is thus considered to be a valuable candidate for a future nuclear fuel by taking into consideration a fuel meat's geometry.

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PHYSICO-CHEMICAL ASPECTS OF MODIFIED UMO/AL INTERACTION

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ABSTRACT

This paper describes the first results of out-of-pile interdiffusion tests focused on the effects of Si added in the Al matrix on the interaction with UMo particles, highlighted by the promising results of IRIS 3 irradiation experiment. The aim of this study is to clearly identify interaction products and diffusion mechanisms. This presentation is supported by a large bibliographic study, which has guided our tests choice.

1. Introduction

The main problem we have to face to improve the in-pile behavior of the UMo/Al dispersed fuel is the instability of the interaction zone, that forms between the particles and their surrounding matrix during irradiation, and that leads in most of cases to an unacceptable pillowing of the fuel plate. Among the different solutions currently under study, there is the reduction of the interaction rate between the fuel particles and the matrix by addition of Si into the Al matrix. Many basic questions relate to the role of this new additive in the stability of the interaction zone as well as its influence on the thermal compatibility with the Al matrix and potential porosity formation. The first measurements results from IRIS 3 ^[1] and PIEs of the RERTR-6 experiment ^[2] as well as several out-of-pile metallurgical and thermodynamic analysis, encourage this solution by showing a significant reduction of the interaction layer (IL) thickness for Si-added plates.

To have a better understanding of the problem, a review of the literature concerning investigations of the UMo-AlSi interaction is summarized here. The very first results are then described in terms of IL thicknesses and microstructures.

2. Literature review

The first works on this subject are attributed to Green ^[3] and DeLuca and Sumsion ^[4] in 1957. Both determined the maximum rates of growth of U/AlSi diffusion layers, definitely lower than in the U/Al reaction, at temperature ranging from 200 to 550°C. They also identified phases formed in the IL as respectively UAl_3 (as the major phase) and a mixture of USi_3 and UAl_3 , now labelled as $U(Al,Si)_3$. The choice of Si as a moderator of the U/Al reaction has been approved by several other studies, such as ^[5] ^[6] ^[7] ^[8], not only for its ability to decrease the rate of growth of the IL but also for its ability to inhibit the growth of the deleterious compound UAl_4 . According to Thurber and Beaver ^[9], any element that forms compounds isomorphous with UAl_3 concentrates in the UAl_3 phase and stabilizes it to the exclusion of UAl_4 , known to be a non adherent, porous and easy fractured compound. Si is one of the element able to form this kind of compound and thus avoid the formation of an unwanted phase. More recently, because of the need of improvement of the UMo/Al fuel, many research programs have been conducted on the complex quaternary UMo-AlSi systems, in-pile but also out-of-pile experiments :

- thermodynamic evaluation of stabilizing element additions to UMo/Al ^[10] ^[11] ^[12],
- out-of-pile interdiffusion studies ^[13] ^[14] ^[15] ^[16] ^[17],
- heavy ion irradiation work ^[18] ^[19],
- in-pile experiments ^[1] ^[2] ^[20] ^[21] ^[22].

The main points emphasized by these studies are:

- To be efficient, stabilizing element, if added in the matrix, should form weak bonding with Al and strong bonding with U and Mo; if added to the fuel, then the opposite should be considered ^[14]. Gibbs free energy calculations can indicate the most interesting candidate.

- Modelling tools (BFS or Mediema method) predict that Si concentrates in the interdiffusion zone, thus depleting the Al-Si alloy ^[11]; atom-by-atom analysis shows that Si prefers to be surrounded by both U and Al instead of only Al ^[12], Al prefers to occupy sites far from Si atoms; with Si addition, the trend is to form interfacial multicomponent compounds.
- Si would not directly interact with UMo fuel ^[17], but preferably with UAl₃, in which Si diffusion is easier and faster than Al in UAl₃ (toward UMo); available quantity of Si has an influence on the final IL composition, a lack of Si results in an UMo decomposition.
- Under irradiation, Si addition to the matrix would reduce the fission induced high fluidity of the amorphous interaction product, root cause of the gross porosity formation ^[2].

Due mainly to the complexity of this quaternary system, a few discrepancies can be found throughout this literature, especially about the IL growth rate or the Si content required in the matrix to stabilize the IL. Even though a lot of experiments have already been conducted on the Si-solution, there is still a lack of knowledge on a few points:

- phases formed in the IL (and in the quaternary system) are not identified with accuracy,
- diffusion mechanisms occurring out-of-pile (and in-pile) are not fully determined.

Our experimental program is based on these different observations and aims to answer some of these tricky questions. First, thanks to the diffusion couple method, we would like to determine the IL thickness as a function of time, temperature, Si content and Mo content. Then in a second part we plan to identify the quaternary phases formed in the IL by means of XRD, μ -XRD and EPMA. As the program has just begun, in this paper we mainly present the first results concerning the IL morphology (shape, thickness) and give some details about the kinetics aspects of diffusion.

3. Phase diagrams

According to the Al-Si binary phase diagram, the simplest model of microstructure of aluminium-silicon alloys can be presented as a soft continuous matrix of α -Al solid solution containing hard and brittle precipitates of Si of different morphology. The maximum solubility of Si into Al reaches 1.65 wt % at 577°C, the eutectic temperature. Eutectic composition is reached at 12.6 wt % of Si.

By comparison of phase diagrams U-Si and Al-Si, it appears that Si is more attracted by U than by Al and has a strong tendency to form precipitates with U. U-Si products should then be more stable than U-Al ones. But if Mo tends to stabilize the γ phase of U, when added to the interaction product, the compound becomes less stable. That leads to compounds stabilities increasing from U(Mo,Al)₃ (slight Mo content) to UAl₃ and to USi₃ ^[11].

4. Experimental details

4.1. UMo and Al alloys

Arc melted ingots of UMo, containing 5, 7 and 10 wt % Mo, were supplied from AREVA-CERCA fuel manufacturer. Thermal annealing (900°C, 72h, secondary vacuum) followed by an helium quenching (2000°C/h) have been performed in order to homogenize the Mo content and to retain the metastable γ phase.

Aluminium alloys have been chosen with a Si content ranging from 0.11 to 12 % wt, known to be the eutectic composition. These alloys and their Si concentrations are detailed in **Table 1**.

The three following aluminium alloys (5754, 6061 and 8001) have been added to the set of experiment because of their interest as cladding materials, they are also listed in **Table 1**.

Al designation	1050	Al98-Si2	4043	4343	4045	4047	5754	6061	8001
Si (%wt)	0.11	2	5	7.4	10	12		0.6	0.17
Mg (%wt)							3	1	
Other (%wt)				0.21 Fe		0.29 Fe	0.75 Mn		0.6 Fe 1.1 Ni

Table 1. Al alloys compositions.

In 5754 alloy, there is no Si and Mg is completely soluble in α -Al at 450°C. This alloy contains small and dispersed strengthening particles of Mn and Cr in order to improve its mechanical properties.

Under irradiation, this alloy is known to harden thanks to Mg_2Si precipitation, with Si coming from Al transmutation.

In 6061, Si is already present but Mg content is almost twice higher than Si one. This results in an alloy containing Mg_2Si particles and almost no free Si. This alloy also has Cu and Cr additions. On the contrary of 5754 alloy, 6061 hardens thanks to Si precipitation under irradiation.

8001 contains Fe, Ni and also Si and Cu to a maximum level of 0.17 wt. % Fe and Si which are common impurities of Al alloys usually forms ternary compounds.

4.2. Experimental procedures

Diffusion couples are prepared with samples of approximately $2 \times 5 \times 5 \text{ mm}^3$, cut out from UMo ingots or Al foils. Both parts are mechanically polished and chemically etched before the annealing, in order to eliminate surface contamination and oxide layer. Then the two parts of the couples are placed in intimate contact and maintained thanks to a specific device at a constant clamping during the thermal treatment.

Following the kinetics considerations given by the TTT curves ^[23] of the alloys, thermal annealings were performed between 450 and 500°C for 1.25 to 3 hours, in order to avoid or limit the influence of the eutectoid transformation of UMo. Thermal treatment is performed under Ar + 5 % H_2 atmosphere.

After diffusion annealings, samples are observed with optical and electronic microscope (FEG-SEM-Philips XL 30 / EDAX EDS detector)Set of experiments

The annealing times and temperatures are summarized in **Table 2**. Each of Al alloys cited above have been used to form couples with each of other three UMo alloys. In the next set of experiments, investigations on UMo5 will be stopped because, in spite of our precautions, UMo5 is not fully retained in γ phase and decomposes during thermal treatment.

	For all Al alloys	Experiment in progress
UMo5	1.25 h - 450°C	
UMo7	2h – 450°C	1h – 550°C
UMo10	3h – 450°C	(not for 6061, 8001 and 5754)

Table 2. Annealing conditions.

5. Results

Table 3 recaps the different maximum IL thicknesses obtained through this first set of experiments, except the ones obtained for specific alloys 5754, 6061 and 8001. The main trend confirms that a Si addition in the matrix reduces, sometimes in a very large extent, the interdiffusion of the species. At this time it is not yet possible to deduce the values of the apparent activation energy and the diffusion rate, because some annealings are still in progress. However, a few remarks can be drawn from these results :

- Al alloys 5754, 6061 and 8001 give the largest IL, whatever the UMo alloy is, i.e. : 677 to 1072 μm ,
- A Si depleted zone, near from the Al/IL interface, has been observed in several samples and in most of cases IL seems to be multilayered (figure 1),
- Kirkendall effect has been widely observed ,
- UMo7 presents local decomposition, at the IL interface.

A few samples give results which are not consistent with previous ones ^[24], such as AR couple (see **Table 3**), that should have formed a more important IL and AW couple that gives a very low interaction rate. Both these data have to be verified in the next experiments set. This raises the question of reproducibility. As a matter of fact, whatever care is taken in the sample preparation, this kind of experiment necessarily includes a part of uncertainties already listed by DeLuca and Sumsion in 1957 ^[4].

So, if we take into account the credit that can be conferred to these measurements, it appears that best results are obtained with high Si content (at least 10 wt%) and UMo7 : IL thicknesses reached are approximately 50 μm .

Temperature reached : 440 ± 10 °C

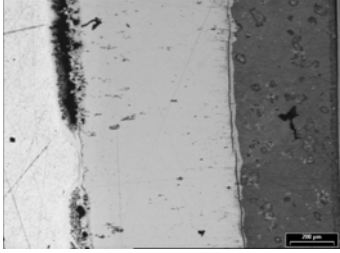
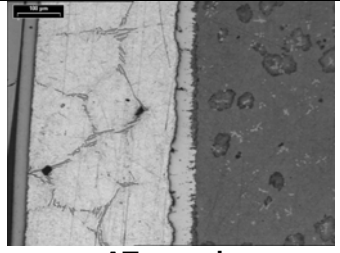
Sample Ref.	Al grade / [Si] %wt	[Mo] %wt	ILmax (μm)	
AV	1050 / 0.11	5	845	
AR	1050 / 0.11	7	50	
BK	1050 / 0.11	10	207	
AW	2	5	5	
BG	2	7	defective (oxide)	
BO	2	10	190	
AZ	4343 / 7.4	5	47	
AS	4343 / 7.4	7	202	
BN	4343 / 7.4	10	110	
AX	4045 / 10	5	60	
AT	4045 / 10	7	55	
BL	4045 / 10	10	206	
AY	4047 / 12	5	267	
AU	4047 / 12	7	48	
BM	4047 / 12	10	140	

Table 3. Maximum IL thickness measured on the diffusion couples and 2 examples of interaction (dark side is UMo).

The **Figure 1** shows the IL obtained with BN couple, that is to say an example of medium interaction.

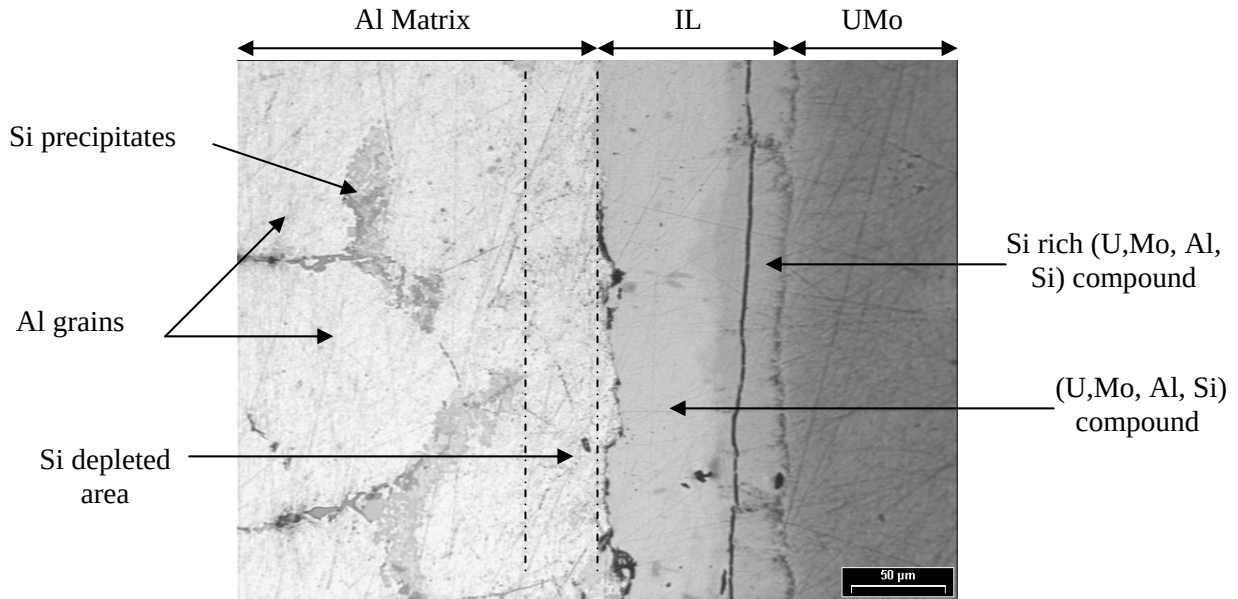


Figure 1. Diffusion couple BL : UMo10 / Al-Si 7,4 – 450°C, 3h.

6. Discussion

6.1. Al alloy evolution

Silicon content in standardized commercial cast alloys is in the range of 5 to 23 %. The properties of a specific alloy can be attributed to the individual physical properties of its main phase components (α -Al solid solution and silicon crystals) and to the volume fraction and morphology of these components. During annealing of Al around 0.7 T_m (melting temperature: 660°C, eutectic temperature: 577°C) it is probable that Al-Si matrix will evolve. As received, the alloys exhibit a very strain hardened structure. **Figure 2** shows the typical microstructure observed after diffusion annealing far enough from interface to avoid the Si depletion effect. Grain size varies from 110 μm to 220 μm in

width. Si lamellas are surrounding Al-grains (classical eutectic microstructure). The Si morphology and distribution throughout the matrix can have a major impact on the alloy properties and on its ability to react with UMo alloy during annealing.

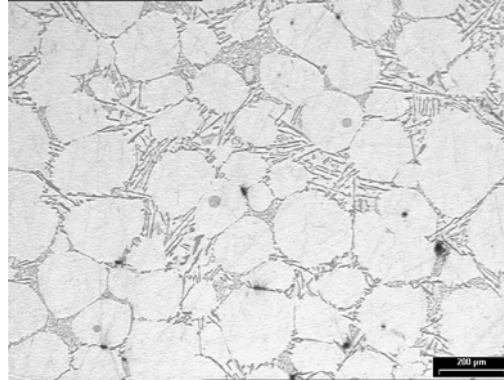


Figure 2. Al matrix (4343) after diffusion annealing, far from interface.

6.2. Remarks on IL growth

Figure 3 is the representation of data summarized in table 3. Si addition acts on the IL formation in such a way that, with our experimental conditions, when Si content exceeds 2 %, IL thickness decreases in a large extent. Concerning the first point of the UMo7 curve, it is irrelevant and should stand between 300 and 600 μm (empty squares, fig. 3) to be consistent with the previous result on the same alloys [see e.g. 24]. For UMo5 samples, thickness exhibits a global trend to increase, when Si content exceeds 2 %wt. But, one can carefully consider the last point, because, as explained above, UMo5 is not fully retained in γ -phase and its destabilisation probably forces accelerated diffusion of Al towards UMo.

Once again, all these results are preliminary results and, because of the uncertainty due to reproducibility problems and to the huge complexity of this quaternary system, need to be confirmed by new experiments.

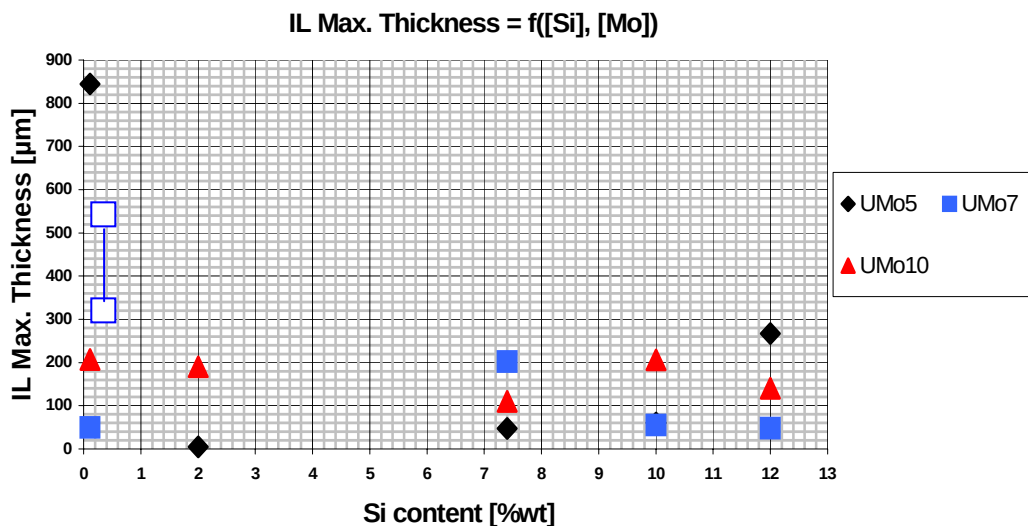


Figure 3. IL thickness as a function of Si content in the matrix.

7. Conclusion

Interdiffusion couples have been realized from 3 different UMo alloys, 6 Al-Si alloys, with Si content ranging from 0.11 to 12 %wt, and 3 Al-alloys with other additions (Fe, Ni, Mg, Mg+Si). Silicon addition seems to have an important effect on the growth rate of the IL : as a matter of fact, when Si content increases, IL thickness decreases. In the case of multilayered IL, the interaction formed near the UMo side is richer in Si than the one neighbouring the Al matrix. Last point of interest is that the Al composition and microstructure have also a significant effect on the interaction. As shown by the

first experiments Si has to be sufficiently available in the matrix to interact. If precipitated too far from interface or with other elements such as Al-Fe-Si compounds in 8001, it doesn't play its role anymore and we can assist to UMo decomposition.

8. References

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THE REACTION ZONE IN THE SYSTEM U-Mo/Al 6061 RELATED WITH THE DECOMPOSITION OF γ U-Mo.

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ABSTRACT

The interaction layer between U-7%Mo and Al 6061(0.6% Si) at 550°C during (1.5 $\pm$ 1.5) h, has been analyzed in detail. As result of the annealing, the UMo alloy showed cellular decomposition in some areas of the sample. If U-Mo remains in γ phase, a compound with the structure of $U(AlSi)_3$ was found in the IL. When decomposition occurs at the interphase with Al6061 the interaction layer became more complex and interdiffusion products are the same as reported in the case of pure Al. The interaction layers in U-7Mo-0.9Pt/Al and U-7Mo-1Zr/Al are also analyzed as examples of no decomposed or fully decomposed alloys.

1. Introduction

The UMo dispersion or monolithic fuels are being developed to be used in high flux research reactors [1]. Irradiation results have shown an important reaction between UMo and Al matrix producing a considerable swelling, accompanied in some cases, by an unacceptable porosity allocated at the interface between the interaction layer and the Al [2,3]. A modification using Al-Si alloys to control the reaction is currently being tested under irradiation with promising results [4,5]

Out- of- pile research works have been done to understand the interdiffusion between γ (U-Mo) and Al [6,7,8,9]. It is accepted that the interdiffusion of Al in γ (U-Mo) between 500°C and 600°C results in a multiband interaction layer formed by phases with the structures UAl_3 , UAl_4 , UMo_2Al_{20} . The addition of Si to Al suppressed the formation of UAl_4 [10]. Si added to Al and Zr to U-Mo were reported to reduced the growth rate of the reaction [11].

Out of pile test are time limited because of the risk of decomposition of γ (U-Mo). If this happens the interaction layer for U-Mo/ Al, is formed by UAl_3 and $U_6Mo_4Al_{43}$ [7].

In the course of the investigation of the effect of Si added to Al, particular characteristic of the IL in U-7%Mo/Al6061(0.6wt%Si) when decomposition occurs were found and are presented in this work. The interaction layers in U-7Mo-0.9Pt/Al and U-7Mo-1Zr/Al are also analyzed as examples of no decomposed or fully decomposed alloys.

2. Experimental

In this work the IL in three diffusion couples U-7wt%Mo / Al 6061, U-7wt%Mo-0.9wt%Pt / Al and U-7wt%Mo-1wt% Zr / Al were analyzed. The U-based alloys were fabricated by arc melting using depleted U (0.2 % U^{235}). The as-cast alloys were homogenized in composition by a thermal treatment of 2 h at 1000°C. Commercial Al 6061 and high purity Al were used as counterpart.

The diffusion couples were fabricated by Friction Stir Welding technique (FSW) or stainless steel clamps. Thermal treatments are detailed in Table I.

Characterization of the interaction layer (IL) was performed by Optical Microscopy (OM), Scanning electron microscopy (SEM), Energy Dispersive X-Ray Spectroscopy (EDS) and X-ray diffraction (XRD).

Samples were mechanically polished up to 1 μ m diamond paste. In some cases they were chemical etched with 1 % of HF in distilled water. To perform XRD, parallel and successive surfaces (8mm x 15mm) at a small angle with the IL interface were exposed by polishing in a specially designed abrading machine.

	Time (h)	Temp.(°C)	Fabrication technique
U-7Mo / Al 6061	(1.5 + 1.5) *	550	FSW
U-7Mo-0.9Pt / Al	2	580	SS Clamp
U-7Mo-1 Zr / Al	1.5	550	SS Clamp

* (1.5+1.5) stands for two successive thermal treatments

Table I. Diffusion anneals and fabrication technique of the couples.

3. Results

3.1. Interdiffusion between U-7wt%Mo / Al 6061 at 550 °C

Morphology

As result of the diffusion anneal, the cellular decomposition of the γ (U-Mo) alloy, into α U and γ (U-Mo) enriched in Mo, in a lamellar morphology and in isolated areas was observed. IL shows two different morphologies type 1 or 2, Figure 1a. *IL type 1* is the thin band fairly uniform of 6 μm , in average, not distinguished in the micrograph of Fig.1a and indicated as A in fig. 1b. This IL developed where U-Mo remained in γ phase. *IL type 2* are localized and much larger regions: “bumps”. A careful mechanical polishing revealed that when the decomposition of U-Mo occurred at the interface with Al 6061, bump-type IL are developed. A detailed view of them is shown in figure 1b. As it is seen, the thin band of *IL type 1* is not interrupted at the bumps but continues over it (B in fig.1b), being always the limit with Al 6061. Part of the decomposed region is also affected by interdiffusion, zone C in fig.1b.

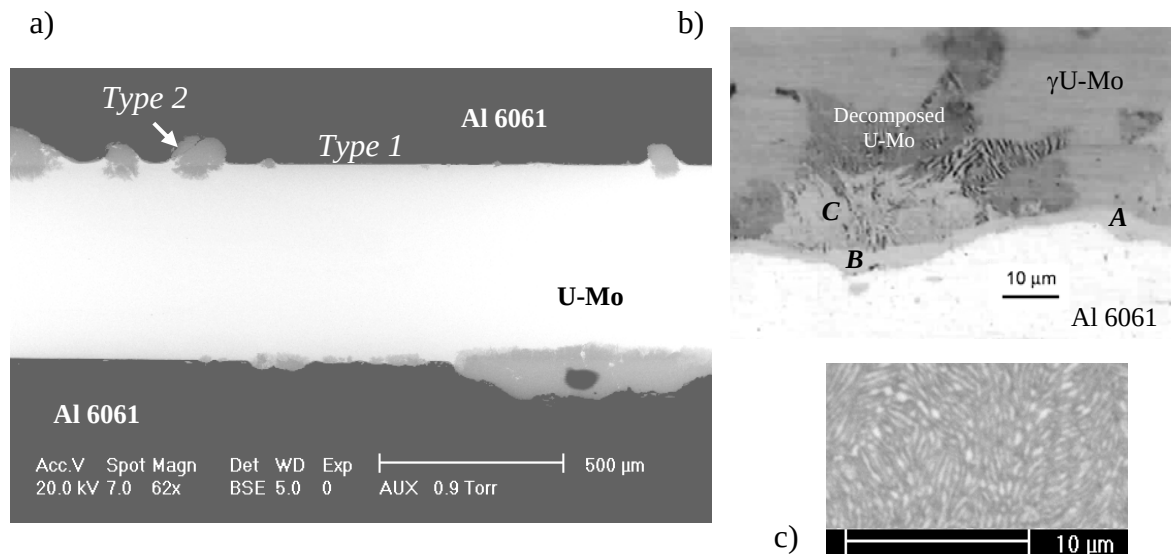


Figure 1. Sample U-7Mo/Al6061-550°C-(1.5+1.5)h. a) SEM BSE image. General view of the sample. b) OM image. Detail of the *IL type 2*. c) SEM BSE of zone C inside of the bumps.

Figure 1c is a SEM BSE image at position C inside the bump. The Z contrast revealed the presence of two phases. The morphology of this microstructure follows the morphology of the cellular decomposition.

Composition

Concentration measurements by EDS were performed along the IL in the three regions indicated in fig. 1b: **A**: Thin band between γ U-Mo and Al 6061 (*IL type 1*), **B**: This same thin band over the bumps and **C**: Inside the bumps. For each region more than 20 measurements were made. Due to the resolution of EDS technique, no variations in composition were measured in C. The most remarkable feature is the almost absence of Si in region C (less than 2 at%) while 11 at% were found in regions A and B. Table II

	U at %	Mo at%	Al at%	Si at%
A	15	3	71	11
B	13	2	74	11
C	18	2	79	<2%

Table II. Average concentrations in atomic %, in regions A, B, and C in Figure 2 b.

Structures identification

Two successive spectra were performed on this sample. Dashed lines in Fig 2a, represent the position of the two surfaces along the depth of the sample. In the first one, only the bumps (IL type 2) are intercepted. For the second one, the region A (IL type 1) also appeared.

Figure 2b shows the XRD spectra. On spectrum 1, which corresponds mainly to bumps embedded in Al, besides Al two other structures were identified: UAl_3 and $Al_{43}Mo_6U_4$. On spectrum 2, besides these ones, UAl_3 with a modified lattice parameter was also found. This lattice parameter modification is associated to the presence of Si replacing Al atoms, resulting in the phase $U(Al,Si)_3$. The value estimated for the lattice parameter is $a=4.23\text{\AA}$ which, according to Dwight [12], would correspond to $\sim 17\text{ at\% Si}$.

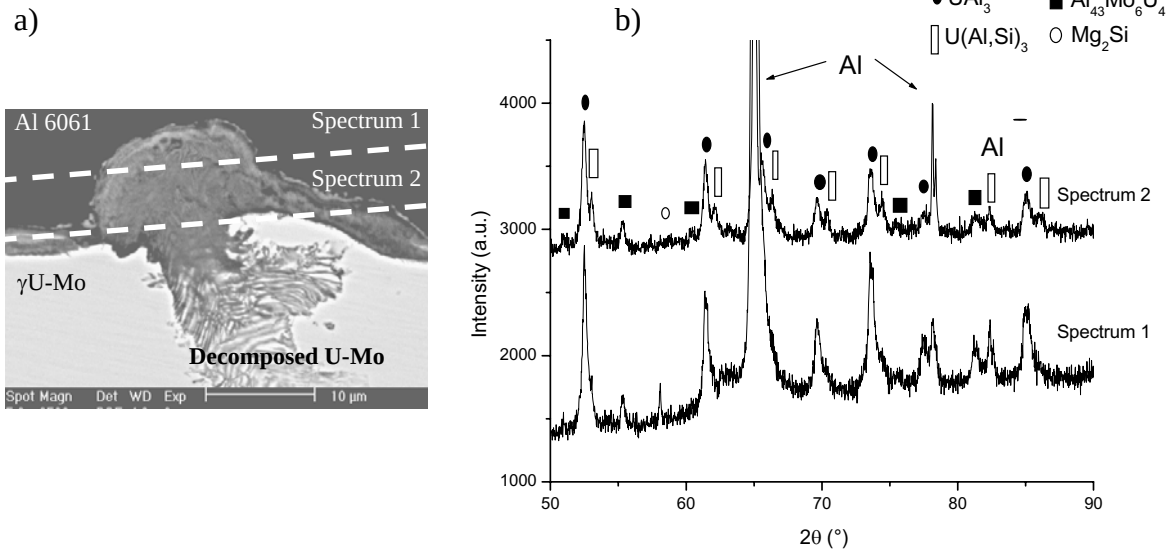


Figure 2. Sample U-7Mo/Al6061-550°C-(1.5+1.5)h. a) Position of the spectra b)XRD spectra.

3.2.Interdiffusion between U-7wt%Mo-0.9wt%Pt / Al at 580 °C

Morphology

After the thermal treatment the U-Mo-Pt alloy showed no sign of decomposition of the γ phase. Between this alloy and the pure Al an IL, of about $100\text{ }\mu\text{m}$, was identified. This IL presented smooth interfaces and a constant width. Chemical etched with 1% of HF acid in distilled water suggested the presence of more than one phase in this zone, Figure 3 a.

Structure identification

Two representative spectra are plot together in figure 3b. Identified structures were: Al, γU , $Al_{20}Mo_2U$, UAl_4 and UAl_3 . No $Al_{43}Mo_6U_4$ was identified.

The lattice parameter estimated for γU was $a=3.45\text{ }\text{\AA}$. This value is the same obtained before the thermal treatment and agree with the absence of composition of the $\gamma(U\text{-Mo-Pt})$ phase.

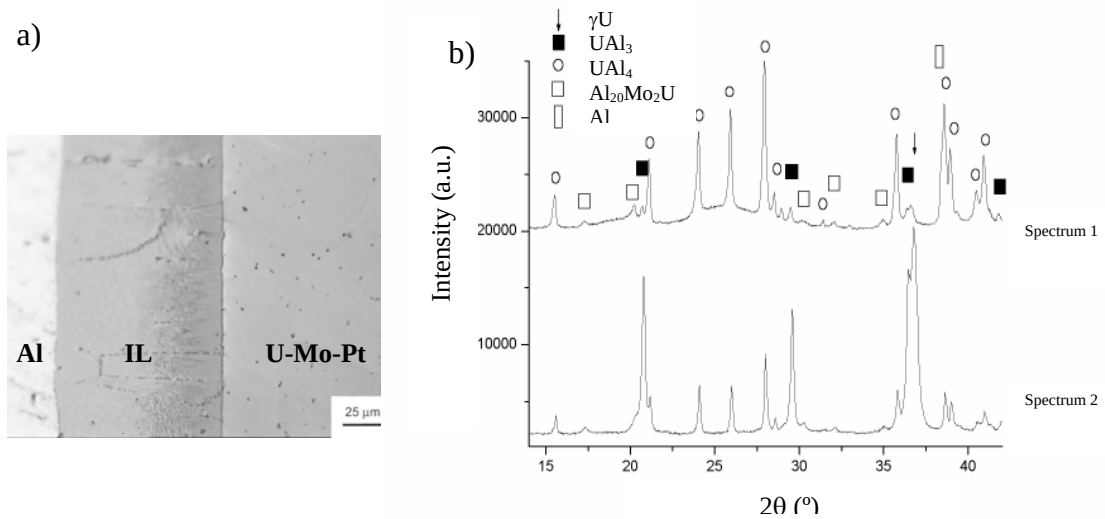


Figure 3. Sample U-7Mo-0.9Pt/Al-580°C-2h. a) OM image of the IL. b) XRD spectra

3.3. Interdiffusion between U-7wt%Mo-1wt% Zr / Al at 550 °C

Morphology

In this case, the U-Mo-Zr alloy underwent almost full decomposition. Between the U-Mo-Zr and the pure Al, an IL with irregular interfaces was identified. Figure 4 a.

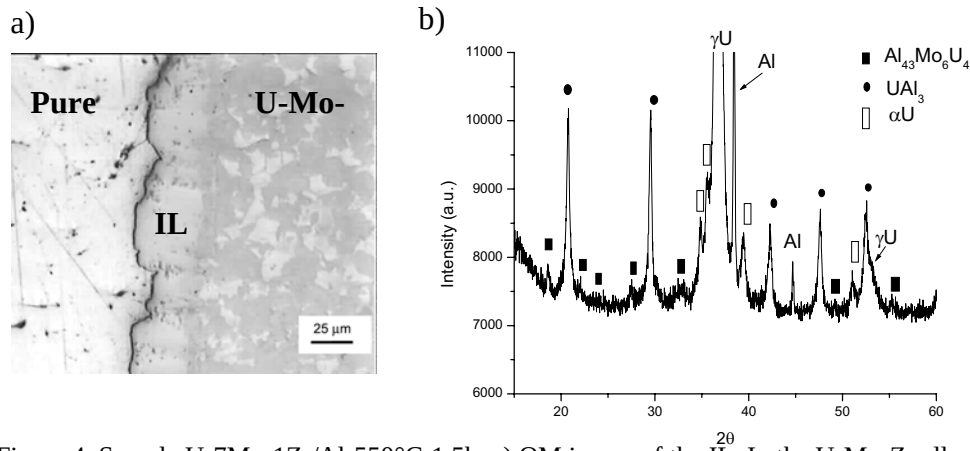


Figure 4. Sample U-7Mo-1Zr/Al-550°C-1.5h. a) OM image of the IL. In the U-Mo-Zr alloy, light gray areas and U and dark gray ones are decomposed U. b) XRD spectrum.

Structure identification

Structures corresponding to the phases Al, αU and γU were identified corresponding to the initial components of the diffusion couple. γU was indexed with a reduced lattice parameter ($a=3.41\text{\AA}$), respect to the initial alloy U-Mo-1Zr ($a=3.45\text{\AA}$). This reduction is known to be associated with enrichment in Mo due to the decomposition process. Besides these structures, other four, corresponding to the IL, were identified: UAl₃, UAl₄, Al₂₀Mo₂U and Al₄₃Mo₄U₆. For this sample it was not possible to identify all the phases in the same spectrum. This is, probably, due to the relative amount of the phases in each spectrum. In figure 4b the presence of the ternary phase Al₄₃Mo₄U₆ is clearly depicted.

4. Discussion

In this work the IL in U-7Mo/ Al 6061, U-7Mo-0.9Pt / Al and U-7Mo-1Zr/Al diffusion couples were analyzed and compared to emphasize the changes related to the decomposition of the γ U-Mo.

The detailed analysis of the morphology of the IL in sample U-7Mo/Al6061, (fig.1a and b), reveals two stages in the diffusion process. During the first one, while U-Mo alloy is in γ phase, the IL along the whole interface U-Mo/Al6061 grows as a band of planar interfaces, as has been reported for pure Al [7]. When decomposition of γ phase starts it gives rise to the second stage on which the enhancement of interdiffusion in localized sites of the sample promotes the formation of the bumps fig.b. When the percentage of decomposition is large, these bumps come in contact to each other, producing an IL with irregular interfaces in the Al side as in figure 4 a for (U-7Mo-1Zr)/Al.

Concentrations measurements on regions A, B and C of the IL (figure 1b, Table 2) showed the presence of Si only in regions A and B. This result together with XRD indicates that a phase with the structure $U(Al,Si)_3$ is formed in the IL between Al 6061 and γ U-Mo. Region C, fig.1b, corresponds to the diffusion of only Al along the α -phase of the cellular decomposition. In agree with this, XRD identified the structure of UAl_3 with no variation in the lattice parameter and the ternary phase $Al_{43}Mo_6U_4$. This is in good agreement with the two-phase microstructure shown in figure 1c. These two phases are the same identified in [7] in the IL between pure Al and a not homogenized U-7wt%Mo at 580°C which showed a high percentage of decomposition after the diffusion anneal..

These two experiments strongly suggest that the ternary phase $Al_{43}Mo_6U_4$ appears in combination with UAl_3 only when decomposition of the γ U-Mo occurs. U-Mo-Pt and U-Mo-Zr alloys prepared with the intention of delay or enhance the decomposition provide a good example of this. When Pt is added to U-Mo decomposition is retarded, phase identification showed the absence of the ternary phase $Al_{43}Mo_6U_4$. The U-7Mo-1Zr alloy fully decomposed in the diffusion couple Al/ U-7Mo-1Zr and $Al_{43}Mo_6U_4$ was clearly identified as a component of the IL. None of the ternary phases have been reported in PIE of irradiated fuel elements. The fact that γ (U-Mo) is stable under irradiation suggest that $Al_{43}Mo_6U_4$ should not be expected at all in interaction layers grown under irradiation. Though it has to be kept in mind that out of pile results may differ from the ones under irradiation because mechanism of diffusion are not quite the same.

5. Conclusions

This work has shown that as a result of the interdiffusion between Al6061 (0.6wt%Si) and U-7Mo at 550°C, a compound with the structure $U(Al,Si)_3$ is stabilized in the IL even though γ U-Mo decomposed. When this happens only Al diffuses along the lamellar microstructure giving rise to a phase with the structure of UAl_3 and the ternary $Al_{43}Mo_6U_4$. When decomposed areas are isolated a “bump” type IL is formed. If the UMo is fully decomposed an irregular IL results.

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NEUTRON POWDER DIFFRACTION OF UMo FUEL IRRADIATED TO 60 atom % ²³⁵U BURNUP

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ABSTRACT

Neutron Diffraction Analysis has been carried out on a U - 7 wt% Mo dispersion fuel core irradiated to 60 atom % ²³⁵U burnup, to examine the extent of any microstructural changes occurring in the fuels at high burnup. The results show that several crystalline phases form during irradiation, with (U,Mo)Al₃ and (U,Mo)Al₂ being predominant. Comparisons with 20 atom % burnup fuel indicates that (U,Mo)Al₃ decomposes to form (U,Mo)Al₂. There is also evidence of α U and δ U₂Mo, and a complex ternary phase UMo₂Al₂₀, but in small amounts.

1. Introduction

Atomic Energy of Canada Ltd. (AECL) is developing low-enriched (LEU < 20 % ²³⁵U) Al-UMo dispersion fuel for potential use in research and test reactors [1]. Significant efforts are currently underway around the world to develop this fuel for materials test reactors [2]. The main reasons are that the uranium loadings can be higher than in the presently qualified dispersion fuel, and that spent UMo dispersion fuel is suitable for reprocessing.

UMo fuels irradiated to 20 atom % ²³⁵U burnup provided evidence of significant swelling (~ 6 vol. %) [3]. Post-Irradiation Examinations (PIE) performed at Chalk River Laboratories (CRL) [3] and elsewhere [4-7] show evidence of an extensive reaction between the γ phase (bcc) UMo particles and the Al matrix. At low burnup, the contribution of fission product formation to the swelling behaviour is negligible, and thus swelling behaviour was attributed to chemical interaction of UMo particles with the Al matrix, and the subsequent formation of low-density reaction product(s). The crystal structure of the main reaction product in the irradiated fuel was identified to be isostructural with the uranium aluminide compound (U,Mo)Al₃ by Neutron Diffraction Analysis (NDA) at CRL [8]. (U,Mo)Al₂ and (U,Mo)Al₄ were also observed by NDA, but in small quantities.

At higher burnups, the UMo mini-elements continue to swell [9]. After 80 atom % ²³⁵U burnup, mini-elements containing particles of U - 10 wt.% Mo swelled by ~ 11 vol. % and were intact upon discharge, while the mini-element containing U - 7 wt.% Mo particles swelled by over 15 vol. % and developed a cladding defect. Further NDA experiments have been carried out on UMo fuel irradiated to 60 atom % ²³⁵U burnup to determine if the predominant reaction product originally observed at lower burnup is stable, or if further low-density reaction products evolve in the fuel core microstructure.

2. Materials

Mini-Elements were fabricated at AECL from UMo alloy powders of composition U - 7 wt.% Mo and U - 10 wt.% Mo with nominal loading of 4.5 grams U/cm³ [1]. The fuel elements were irradiated in the National Research Universal (NRU) reactor at element linear ratings up to 100 kW/m, to 60 atom % ²³⁵U burnup, and discharged from the reactor core in March 2004. Details of the irradiation conditions and PIE results are summarized elsewhere [9].

Figure 1 illustrates the fuel element cross-section obtained from the mid-plane of mini-element 59-2 (U – 7 wt.% Mo), and the microstructure of the fuel core, respectively. The specimen was prepared by machining the finned Al cladding using a mechanical lathe in the hot cells. The actual fuel core specimen used in the NDA experiments is shown in Figure 2.

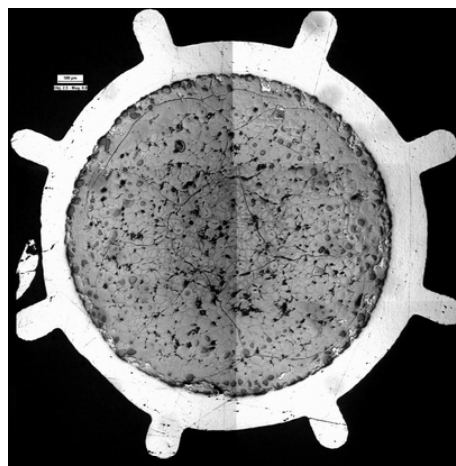


Figure 1. Composite cross-section of mini-element 59-2 (U – 7 wt.% Mo), irradiated to 60 percent burnup in NRU.



Figure 2. As-Machined fuel core specimen cut from mini-element 59-2.

3. Experiment

In order to facilitate NDA experiments on irradiated specimens, a special lead filled “castle” was previously developed with sufficient thickness of lead to permit the execution of the experiment at the NRU facility, while minimizing exposure of personnel to high external radiation fields emitted from the specimen during the experiment. Details of the castle design and its function are available elsewhere [9]. Neutron diffraction experiments are carried out at CRL on the “C2” spectrometer, which is located at the NRU reactor and is owned and operated by the Canadian Neutron Beam Centre (CNBC). The spectrometer is equipped with a banana-shaped, 800 element BF_3 position sensitive detector. As measured from the sample position to the detector, the solid angular spacing between two adjacent elements is 0.1° , so that 80° of scattering angle is measured simultaneously.

Diffraction experiments were conducted at two incident wavelengths. In the first experiment, an incident monochromatic beam of thermal neutrons was obtained from a Si single crystal oriented to scatter from the $\{531\}$ reflection ($\lambda = 0.133 \text{ nm}$) at a take-off angle of 92.7° . For the second experiment, a high resolution scan was conducted using the same Si crystal monochromator, but re-oriented to scatter from the $\{311\}$ reflection ($\lambda = 0.237 \text{ nm}$) at a take-off angle of 92.7° . In both cases, the position sensitive detector of the diffractometer was positioned to achieve nominal coverage of the Bragg peaks between 20 to 100 degrees of 2θ . Because the uranium aluminide peaks exhibit overlap, the higher resolution scans were conducted in order to more accurately resolve the intensity and shape of Bragg peaks and provide better data for quantitative analysis of the phase weight fractions via “Rietveld” analysis. Quantitative analysis of the reaction products was carried out using GSAS [10,11].

4. Results

Figure 3 shows NDA results obtained at low resolution ($\lambda = 0.133 \text{ nm}$) from the irradiated fuel core specimen 59-2. These results provide evidence that the reaction layers that form as

a result of the interaction of the aluminium matrix with the γ phase UMo particles are largely crystalline. As shown in Figure 1, except for a narrow region located at the interface between the cladding and the fuel core, the prior aluminium matrix appears to be completely consumed by the reaction between the Al and U – 7 wt.% Mo particles and replaced with reaction products.

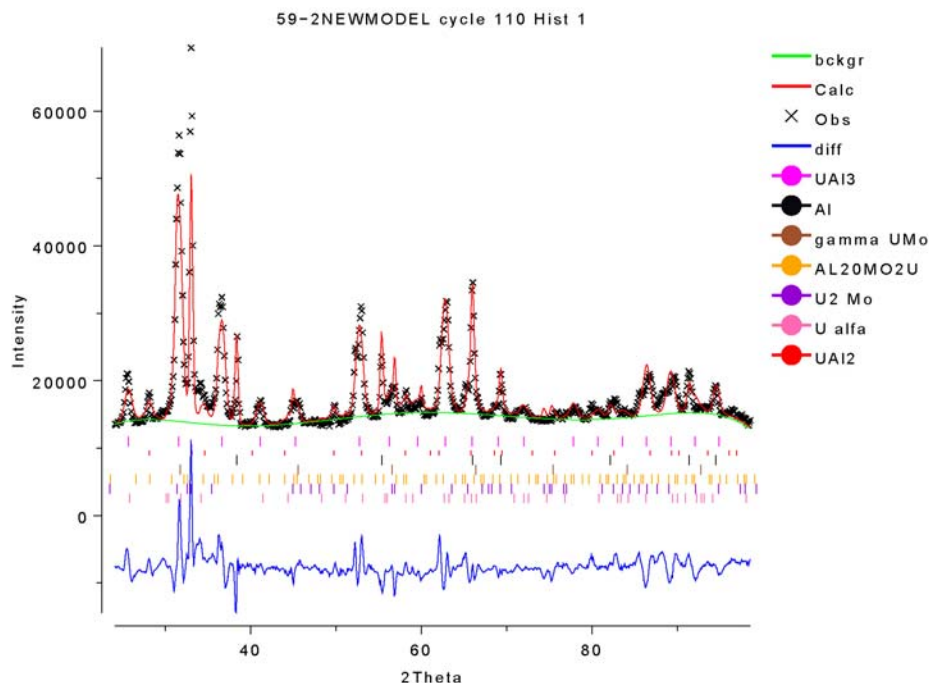


Figure 3. Neutron Diffraction Bragg peaks from specimen 59-2, low resolution scan ($\lambda = 0.133$ nm).

Table 1 lists the previous NDA results obtained from an un-irradiated specimen of U – 7 wt.% Mo core [12], the NDA results obtained from a specimen irradiated to 20 atom % burnup [8], and the new results from 60 atom % ^{235}U burnup. The quantitative analysis results are based on data collected in the low-resolution configuration of the ND experiment. At low burnup, the main crystalline reaction products were (U,Mo)Al₃ and (U,Mo)Al₄ but (U,Mo)Al₂ was not detectable in significant quantities in U – 7 wt. % Mo fuels (however, a trace amount was found in irradiated fuel cores containing U - 10 wt.% Mo). Here, the main reaction product remains (U,Mo)Al₃ (59 wt.%), but a significant fraction of (U,Mo)Al₂ (15 wt.%) is also present, but (U,Mo)Al₄ is not detected. Another important change occurring between low and high burnup is the appearance of δ U₂Mo, and evidence of UMo₂Al₂₀ and α U, but in small quantities.

Specimen ID	Al	γ UMo	UAl ₂	UAl ₃	UAl ₄	UO ₂	α U	UMo ₂ Al ₂₀	δ U ₂ Mo
Un-irradiated	27	61	n.d.	n.d.	n.d.	2	10	n.d.	n.d.
U-7 wt.% Mo 20 at % BU	11 (0.2)	37 (0.4)	n.d.	48 (0.9)	4 (0.6)	n.d.	n.d.	n.d.	n.d.
U-7 wt.% Mo 60 at % BU	14 (0.4)	2 (0.7)	15 (1.2)	59 (0.4)	n.d.	n.d.	1.5 (0.6)	1 (0.4)	7.5 (0.8)

Table 1. Quantitative Phase Analysis (wt.%) of Al-UMo fuel cores (1 σ error in bracket).

There is considerable overlap between the $\text{UMo}_2\text{Al}_{20}$ at low resolution, however, the high resolution data shows the extent of a peak “doublet” (Figure 4) showing the extent of overlap between $\text{UMo}_2\text{Al}_{20}$ and $(\text{U},\text{Mo})\text{Al}_3$ that more clearly resolves the unique existence of the ternary phase in the diffraction pattern. However, the GSAS phase analysis model assumes the same published space group of the $\text{UMo}_2\text{Al}_{20}$ ternary phase ($Fd3m$ [13]), but with a significantly smaller lattice parameter ($a = 1.42$ nm), a difference of approximately 2 % compared to the published lattice parameter of 1.4506 nm.

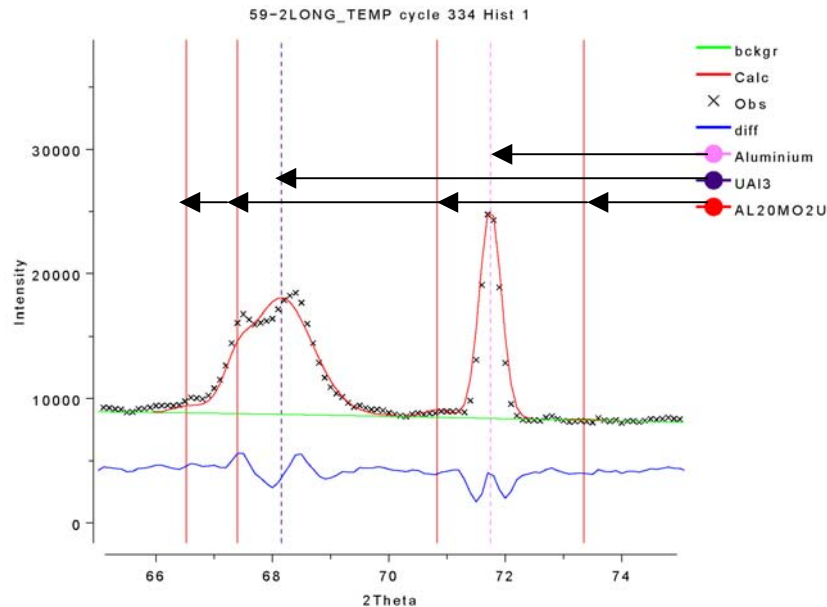


Figure 4. Neutron Diffraction Bragg peaks from specimen 59-2, high resolution scan ($\lambda = 0.237$ nm), showing location of Al (dashed), UAl_3 (dashed) and $\text{UMo}_2\text{Al}_{20}$ (solid)

It is also noteworthy that the weight fraction estimate provided above in Table 1 for $\text{UMo}_2\text{Al}_{20}$ is close to the 1σ error, which indicates that the quantity of the phase is close to the limit of detection for NDA analysis. A similar conclusion can be drawn for α U.

5. Discussion

As was the case previously [8], the Bragg peaks obtained in this experiment from the reaction product phases contrast strongly against the neutron diffraction results of irradiated uranium-silicide phases [14], which showed evidence of considerable amorphization. The data obtained by Richardson was unambiguous in that all of the Bragg peaks of the silicide phase had totally disappeared and did not form additional crystalline reaction products (binary or ternary) that were not otherwise present in the fuel prior to irradiation. This is not the case for irradiated UMo dispersion fuels.

From results presented in Table 1 the predominant crystalline phase after 60 atom % burnup is $(\text{U},\text{Mo})\text{Al}_3$. The main change that occurs in the fuel core microstructure during the transition between low burnup to high burnup is the formation of the UAl_2 Laves phase compound. This phase has not been identified by diffraction methods in any of the published out-of-pile diffusion couple experiments involving UMo – Al. However, Ryu [7] has speculated that UAl_2 may form from the decomposition of UAl_3 only when the Al supply to the Al/ $(\text{U},\text{Mo})\text{Al}_3$ interface from the Al side is less than that lost to U at the $\text{UMo}/(\text{U},\text{Mo})\text{Al}_3$ interface. This condition may never be fully satisfied in conventional binary plate-type diffusion couples. However, as evidenced by the micrographs of the fuel core (Figure 1), it is obvious that Al has been fully consumed in the core, with the exception of the region near the core/cladding interface. Although there appears to be a significant quantity of Al, it is likely

an overestimate as it has a strong extrusion texture in the core, which contributes to the uncertainty of the weight fraction estimate. This effect is not captured by the GSAS estimate of the 1σ error shown in Table 1.

At first glance, the co-existence of γ -UMo, α -U and δ -U₂Mo seems thermodynamically impossible as the phases can co-exist only at the eutectoid temperature (843 K [15]). However, the contribution to the diffraction pattern from the γ -UMo phase is likely due to un-reacted particles of UMo located close to the interface between the cladding and the core (Figure 3). The particles which have undergone the transformation from the γ -UMo phase to α -U and δ U₂Mo are likely located within the centre of the core, where the in-core temperatures and neutron fluences provide sufficient driving force to cause the gamma phase to decompose.

If UAl₂ does indeed form from the decomposition of UAl₃, excess Al is available to form additional Al rich phases in the microstructure. It is remarkable that no significant Bragg peaks of UAl₄ were observed in this experiment. As noted by Ryu [7], it is believed that (U,Mo)Al₄ exists as an amorphous compound in the irradiated fuels.

6. Summary

When irradiated, research reactor fuel cores consisting of γ UMo particles dispersed in an aluminium matrix interact chemically to form reaction products. At 60 atom % burnup the reaction products are predominantly isomorphs with UAl_x compounds ($x = 2,3$). A comparison of the NDA data obtained from UMo fuels at lower burnup (20 atom %), suggests that (U,Mo)Al₃ decomposes to form (U,Mo)Al₂. There is also evidence of transformation of the UMo particles into α -U and δ U₂Mo, and the formation of UMo₂Al₂₀, but in small quantities.

7. Acknowledgements

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IMPROVED IRRADIATION BEHAVIOR OF URANIUM-MOLYBDENUM/ALUMINUM DISPERSION FUEL

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ABSTRACT

Results of the postirradiation examination of dispersion fuel irradiated in RERTR-6 and -7 experiments are presented and analyzed. The miniplates, irradiated in the Advanced Test Reactor (ATR) to ~50% and ~100% LEU-equivalent burnup, contained up to 5 wt.% of silicon in the matrix aluminum. Metallography and plate thickness measurements showed that the addition of silicon to the Al matrix has the greatest potential to eliminate the formation of gross porosity that has occurred in previous irradiation tests of U-Mo/Al dispersion fuel. The results show that silicon additions of 2 wt.% or more drastically reduced the extent of interaction between the fuel and matrix and indeed eliminated the porosity for the irradiation conditions of the present tests.

1. Introduction

Postirradiation examination (PIE) data from the RERTR-6 experiment showed that the addition of Si to the Al matrix would not only reduce the rate of U-Mo/Al interdiffusion, but also reduce the fission-induced high fluidity of the amorphous interaction product that is thought to be the root cause of the gross porosity formation that has resulted in unstable meat swelling in previous tests in several countries [1]. As destructive postirradiation examination (PIE) data from the RERTR-7 experiment are now available, it is possible to compare the effects of the addition of Si to the Al matrix on the irradiation behavior of the dispersion miniplates for a wider range of fission rates and fission densities. Because it is difficult to separate operational parameters such as fuel temperature and fission rate (power density), further analysis of all available PIE data is necessary in order to determine the controlling factor determining the initiation and growth of the porosity.

2. Irradiation Details

RERTR-6 was a low-burnup feasibility test designed to evaluate the performance of a range of modified-matrix dispersion fuels and irradiate the first monolithic fuels [2]. The experiment was loaded into the Advanced Test Reactor (ATR) in 2005 in experimental position B-12. It remained in the ATR for three cycles for a total of 135 EFPD. Average burn-up of the 19.7% enriched fuel was 35 – 49% depending on the location in the experiment. BOL fuel temperatures were < 200°C and peak heat flux was < 220 W/cm². RERTR-7 was a high-burnup feasibility test using MEU (~58% U-235) fuels. The experiment was loaded into the ATR in 2006 and remained in the ATR for a total of 90 EFPD. Average LEU equivalent burn-up was 60 – 88% depending on the location in the experiment. BOL fuel temperatures were < 200°C and peak heat flux was < 340 W/cm². In RERTR-6 and -7 experiments, the experiment basket was positioned so that one edge of the miniplates faced the core and, hence, one edge of the miniplate ran at higher power than the other. The coolant flow through the experiment capsule was adjusted to obtain the same average cladding temperature for both tests.

3. Postirradiation Examination Data

The PIE data of the RERTR-6 test show that for the experimental conditions of the test, an appropriate amount of Si added to the Al matrix has indeed eliminated the porosity problem. In addition, the extent of interaction is significantly reduced. A comparison of microstructures for the range of Si addition, i.e., 0.2, 0.9, 2 and 4.8 wt.% is shown in Fig. 1. The interaction layer (IL) thicknesses in the plates containing 2 wt.% Si and 4.8 wt.% Si in the matrix aluminum are very similar.

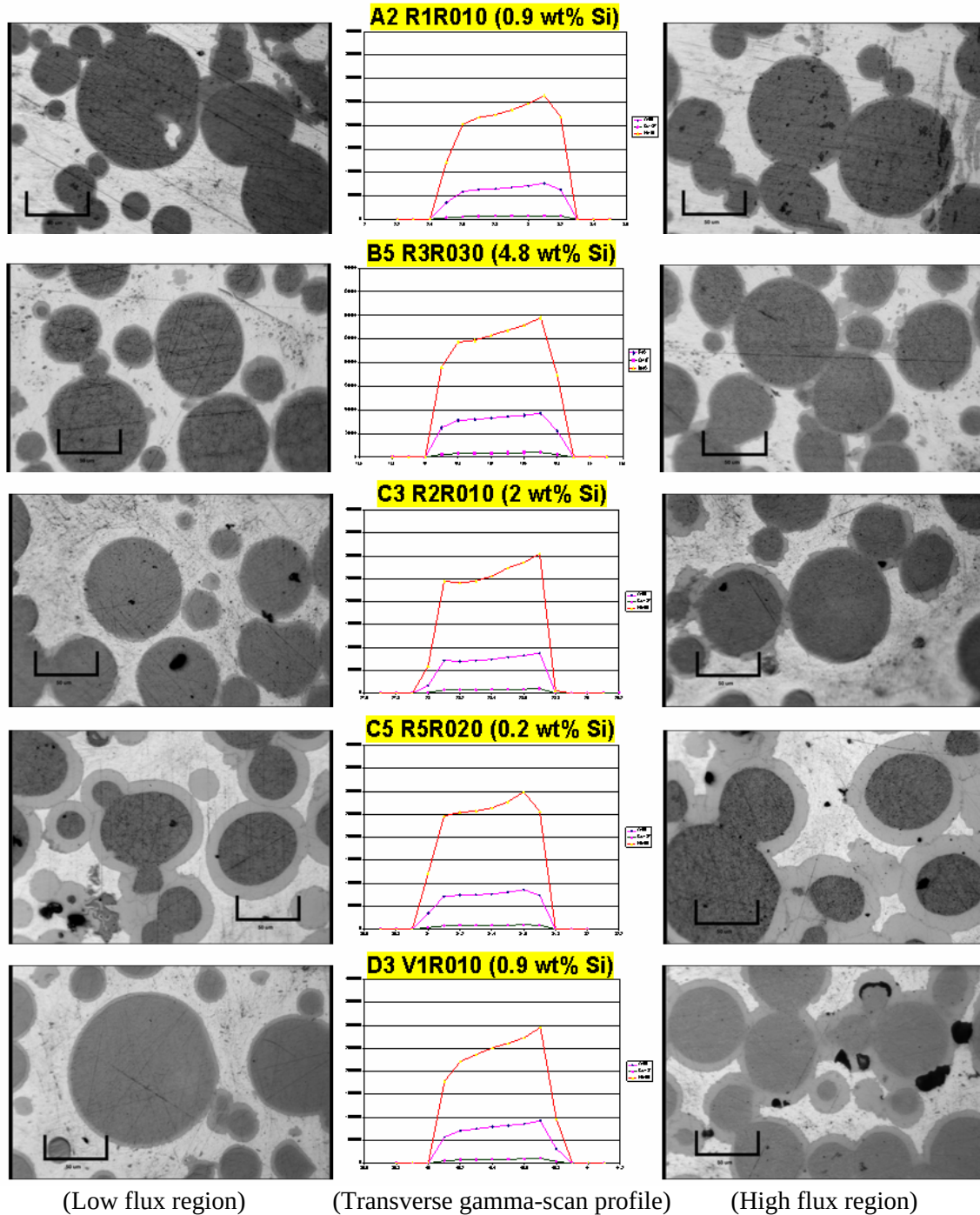


Figure 1. Comparison of fuel meat microstructures of RERTR-6 miniplates (transverse section at mid-length, BU ~50% U-235).

As shown in a comparison of a plate with 2 wt.% Si and one with only 0.2 wt.% Si in Figs. 2 and 3, the difference in meat microstructure of the two plates is striking. In the low-Si plate, there are substantial interaction and Al matrix reduction as well as gross porosity formation at the highest flux edge of the plate. The microstructure is virtually identical to that in similar plates from RERTR-4 and -5 [3]. The IL in the high-Si plate is not only much thinner, but there is no evidence of gross porosity formation in the interaction product.

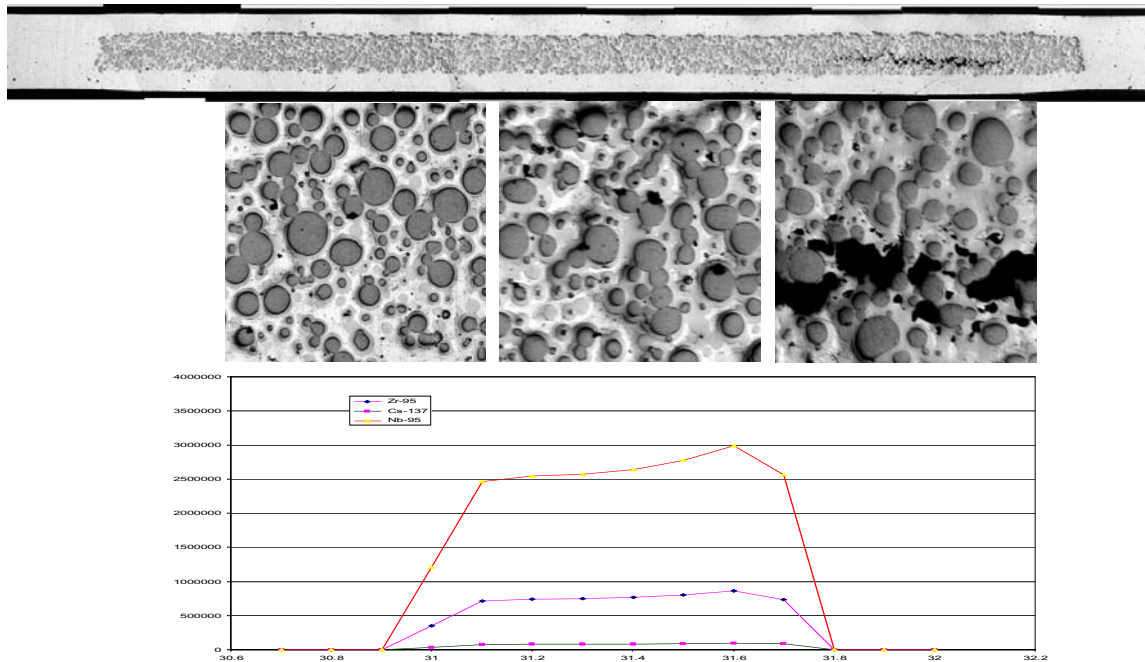


Figure 2. Fuel meat microstructure and gamma scan counts of R5R020 with 0.2 wt.% Si matrix irradiated from RERTR-6 (transverse section at mid-length).

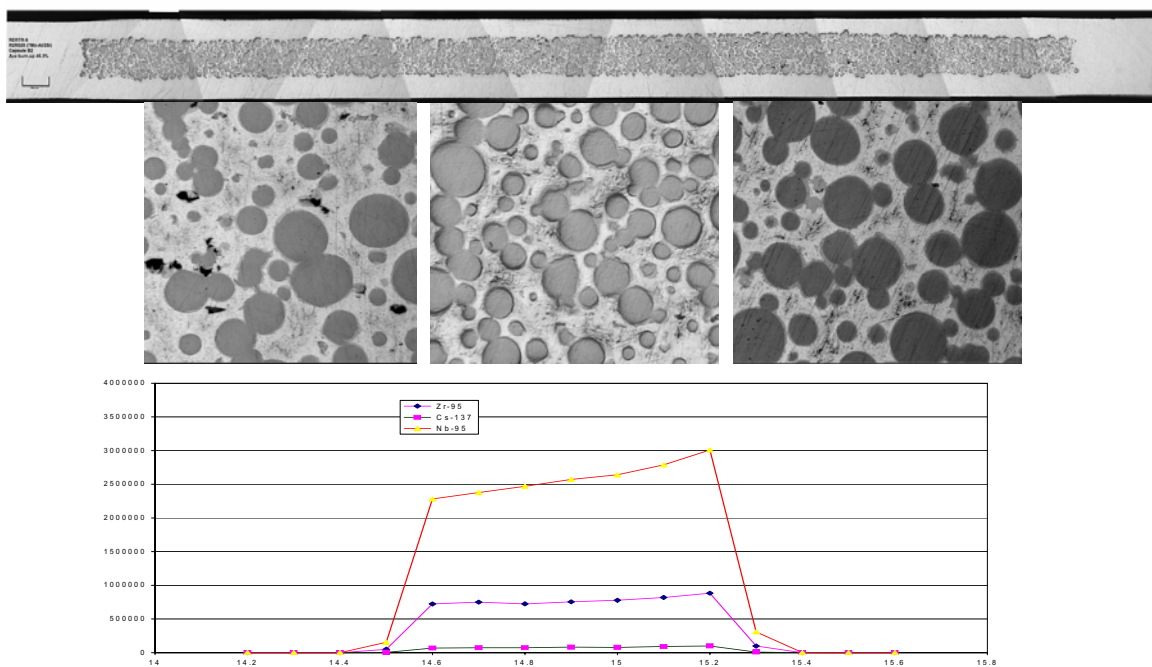


Figure 3. Fuel meat microstructure and gamma scan counts of R2R020 with 2 wt.% Si matrix irradiated from RERTR-6 (transverse section at mid-length).

4. Discussion

Because of the rather wide range of matrix Si compositions employed in the test, the data permit an evaluation of the minimum amount of Si required. This amount appears, for the present test conditions, to lie somewhere between 0.9 wt.% Si (Al6061) and 2 wt.% Si (see Fig. 1). The microstructure of the 0.9 wt.%-Si plate in position D3 captures this minimum amount. If we proceed on the assumption that a concentration of at least 5 at.% is needed in the interaction layer (IL) to “stabilize” its behavior – this level is based on the equilibrium U-Al-Si diagram and the assumption that Mo and irradiation effects do not significantly alter this – we have a situation as shown in Fig. 4.

The upper curve is obtained based on the assumption that all available Si in the matrix can accumulate in the IL, and the lower curve is based on the assumption that only Si from the fission fragment recoil zone is available to accumulate in the IL. The IL thickness at the low flux region is 2.5 μm . For both of the curves, the Si content in the IL can be maintained at the 5 at.% limit. In the high flux region, the IL is 7 μm thick, and the available Si is not sufficient to maintain the composition of the IL at the 5 at.% Si limit.

Thus, somewhere between the two extreme flux levels, lack of Si supply to the IL causes its irradiation behavior, i.e., high rate of formation and susceptibility to gross porosity formation, to revert to that of low-or no-Si conditions.

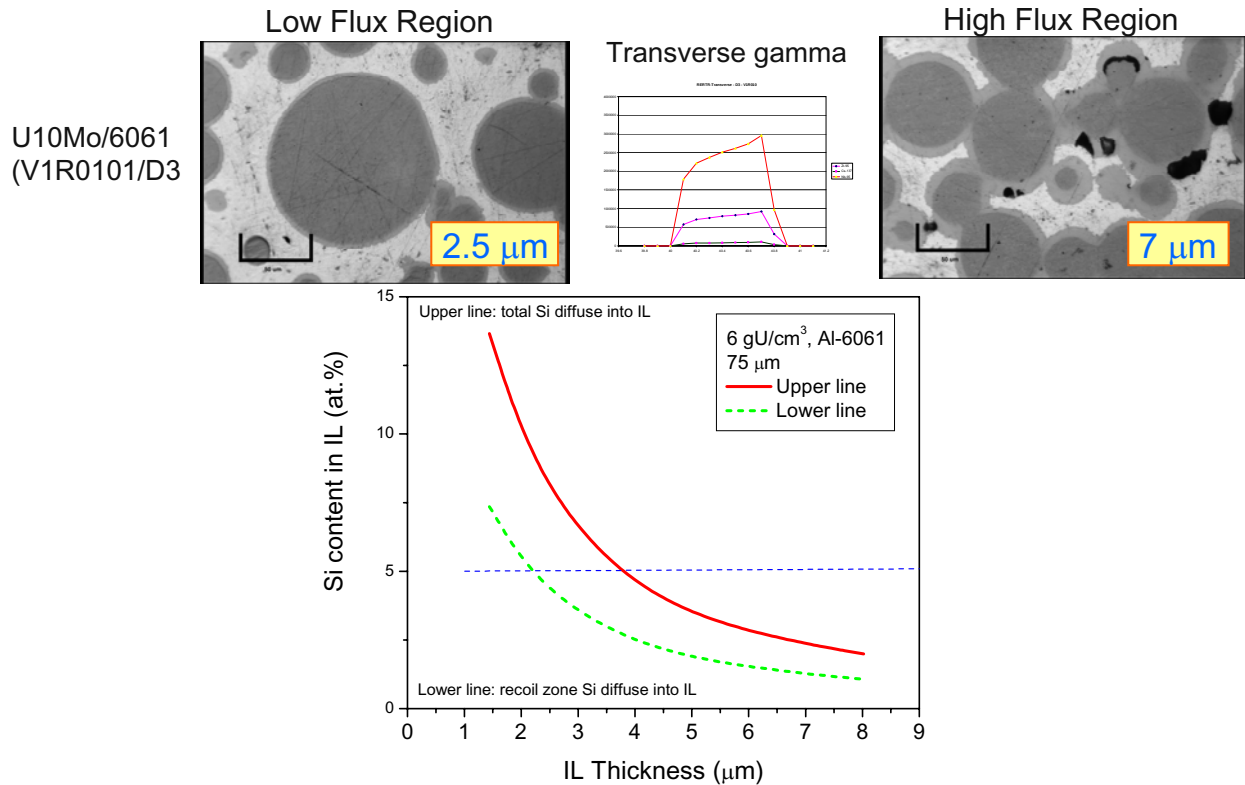


Figure 4. Estimation of Si content in the IL for two regions in miniplate V1R010. The fuel loading density is 6 gU/cm^3 , the matrix is Al6061, and the average fuel particle size is $75 \mu\text{m}$.

Choice between the upper curve (all Si in the Al matrix available) and lower curve (only Si from the fission fragment recoil zone available) is clarified by the microstructure of a high-Si (4.8 wt.%) plate shown in Fig. 5. It is clear that only the Si from the fission fragment recoil zone around the fuel particles participates in the interdiffusion process that forms the IL. Evidently the effect of fast neutron damage in the rest of the Al matrix is negligible in the process. For fuel loadings higher than 6 gUcm^{-3} , however, the recoil zones will increasingly overlap, and this should be taken into account.

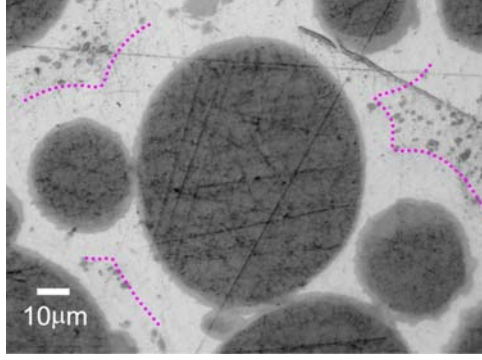


Figure 5. Distribution of remaining Si-rich precipitates in U-7Mo/4043Al (R3R030) from RERTR-6 test. The small dark particles in the bright Al matrix are the Si-rich precipitates.

The plate thickness changes are shown in Fig. 6 for U-7Mo dispersion fuel with various Si-contents in the Al matrix. These data indicate lower swelling rates in the 2 wt.%Si and 4.8 wt.%Si (4043 matrix) miniplates. The data are consistent with reported data from the full-size-plate IRIS-3 test by CEA [4]. The low-Si plate showed breakaway increases of plate thickness on the higher flux side (the side of the plate toward the core center). This might indicate that porosity formation occurred, similar to that found in RERTR-4 and -5 plates at the higher flux positions. In contrast, the high-Si plate appears to have swelled in a uniform manner.

In the plates that have no, or insufficient, Si in the Al matrix, the diffusion controlled critical amount and composition of the IL at which porosity formation occurs (as indicated by increased plate swelling in Fig. 6) are shifted to higher burnup for higher fission rates.

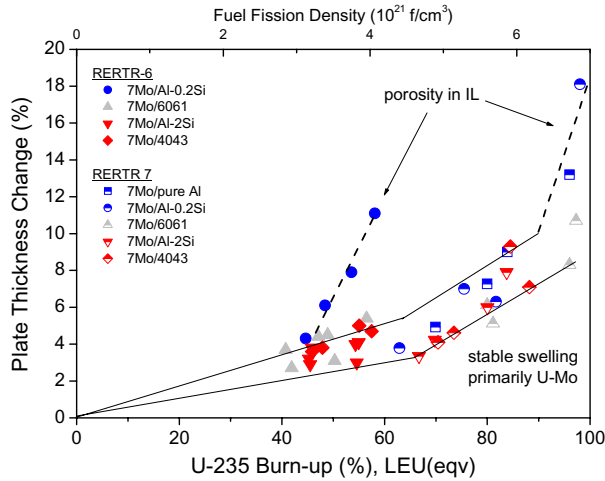


Figure 6. Plate thickness changes vs. U-235 burnup of U-Mo dispersion fuel plates irradiated in RERTR-6 and -7.

Kim et al. correlated the IL growth in U-Mo/Al-Si dispersion fuel as a function of fission rate, time and temperature such as $Y^2 = A \cdot t \cdot \dot{f}^{1/2} \exp(-Q/RT)$ [5]. If the fuel temperatures are the same for two irradiation tests with different irradiation histories, the IL thickness can be compared with $t^{1/2} \cdot \dot{f}^{1/4}$. Because one edge of the miniplates facing the core ran at a higher fission rate than the other in RERTR-6 and -7 experiments, the effect of fission rate can be validated by comparing the measured IL thickness and the calculated values of $t^{1/2} \cdot \dot{f}^{1/4}$ for each experiment. Figure 7(a) shows the distribution of measured intensities of Cs-137, Nb-95, Zr-95 from gamma-scanning along the plate transverse (width) direction. The fission rate histories of RERTR-6 and -7 were calculated based on the gamma-scan results as shown in Fig. 7(b). Table 1 shows that the calculated $t^{1/2} \cdot \dot{f}^{1/4}$ values for the low-flux region and high flux region of U-7Mo/Al-2Si plates from RERTR-6 and -7 are in accord with the measured IL thicknesses. The micrographs of these plates are shown in Fig. 8.

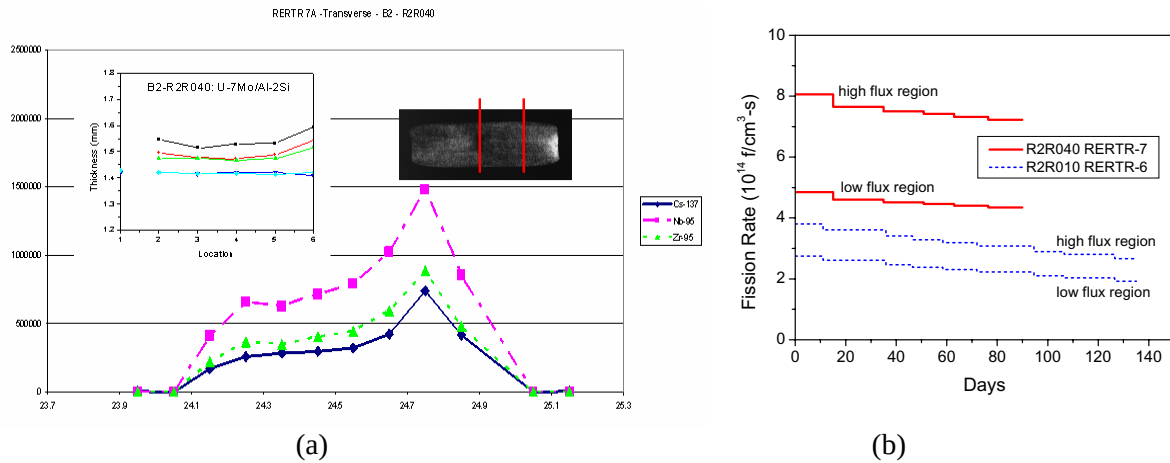


Figure 7. (a) Transverse plate thicknesses and gamma-scan result of the U-7Mo/Al-2Si dispersion fuel (R2R040) irradiated in the RERTR-7 test. (b) Fission rate histories of high and low flux regions in U-7Mo/Al-2Si dispersion fuel plates irradiated in RERTR-6 and -7 tests.

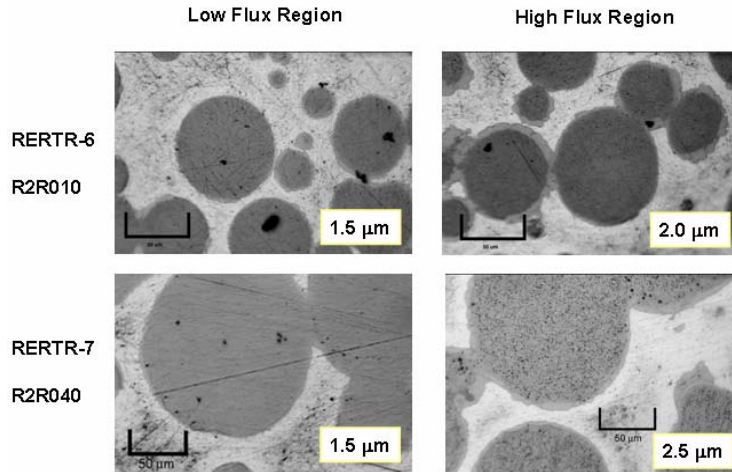


Figure 8. Post-irradiation micrographs of the low flux region and high flux region of U-7Mo/Al-2Si dispersion fuel irradiated in RERTR-6 and -7 tests.

Table 1. Comparison of measured IL thickness and $t^{1/2} \cdot \dot{f}^{1/4}$.

Tests	Flux	EFPD	Average fission rate (f/cm ³ -sec)	Measured IL thickness (μm)	$t^{1/2} \cdot \dot{f}^{1/4}$
RERTR-6	High	135	3.3E14	2.0	5.0E4
	Low	135	2.4E14	1.5	4.6E4
RERTR-7	High	90	7.5E14	2.5	5.0E4
	Low	90	4.5E14	1.5	4.4E4

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POST IRRADIATION EXAMINATION OF MONOLITHIC MINI-FUEL PLATES FROM RERTR-6 & 7

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ABSTRACT

Successful qualification of the monolithic fuel is required for the conversion of the high performance research reactors and significant effort is being devoted to its development. The RERTR-6 experiment was designed to irradiate the first monolithic fuel mini-plates at moderate power density to moderate burn-up. The follow-on experiment, RERTR-7, aimed to irradiate monolithic fuel mini-plates at very high power density to high burn-up. It contained monolithic mini-plates fabricated by friction stir welding (FSW) and transient liquid phase bonding (TLPB). In this paper the post-irradiation examination results of monolithic mini-plates from RERTR 6 & 7 are presented.

1. Introduction

After an initial trial of the monolithic fuel concept in RERTR-4 and in view of the requirements for conversion of the high power reactors, an aggressive program to fabricate, irradiate and qualify the monolithic fuel has been developed and pursued.

The first irradiation of monolithic mini-plates was conducted in RERTR-6. Post irradiation examination (PIE) showed that the concept was feasible and yielded encouraging results but indicated that the interface between the foil and the cladding was an area that was likely to present problems [1]. Evidence of incipient porosity formation in the interaction layer and delamination between the foil and the cladding was observed. Now that some of the PIE results from RERTR-7 are available, it is possible to begin evaluating the behaviour of the monolithic mini-plates at much higher fission rates and fission densities. RERTR-7 also included plates with increased molybdenum content and plates fabricated by transient liquid phase bonding (TLPB).

2. Irradiation Details

The irradiation details of RERTR-6 have been reported previously [1]. RERTR-7 was an aggressive test designed to evaluate the performance of a range of modified matrix dispersion fuels and monolithic fuels. To achieve the necessary surface heat fluxes characteristic of the high performance research reactors, the enrichment was increased to 58% ²³⁵U. The experiment was loaded into the Advanced Test Reactor (ATR) in November 2005 in experimental position B-11. It remained in the ATR for two cycles for a total of 90 EFPD. Average burn-up of the 58%-enriched fuel was 21-30% depending on the location in the experiment. When converted to an LEU equivalent, to facilitate comparison with RERTR-6 data, average burn-up was 62-89%. BOL fuel temperatures were < 225°C and the peak heat flux was 325W/cm². The composition, fabrication method and position of each monolithic plate within the experiment basket are shown in Figure 1. The RERTR-7 experiment consisted of four capsules, A-D, each containing eight plates. The power histories of plates from RERTR-4, 5, 6 and 7 are shown in Figure 2. The experiment basket is positioned so that one edge of the mini-plates faces the core. As a result a strong neutron flux gradient exists across the width of the plates. The higher enrichment increases the contribution of self shielding and as a consequence the high flux edge ran at least two times the power of the low flux edge. Figure 2 shows the difference in fission rate and burn-up between the core edge and outer edge of mini-plate L1F140.

A				A8-H1F020
B	B5-L1F0IL		B7-L1F140	B8-MZ25
C	C1-H1F030	C2-L1T020	C3-L1F110	C4-MZ50
	C5-L1F120	C6-H1T020		
D				
		D6-L1F160	D7-L2F040*	

Legend:

H1F- FSW-12Mo (250µm foil)

H1T – TLPB-12Mo (250µm foil)

L1F – FSW-10Mo (250µm foil)

L1T – TLPB-10Mo (250µm foil)

L2F* – TLPB-10Mo (500µm foil)

MZ – Roll 7Mo/Zr clad. (250 & 500 µm foil)

* this plate should have been labelled L2T, not L2F

Figure 1. Schematic diagram of the RERTR-7 experiment showing location, composition and foil thickness of the monolithic mini-plates.

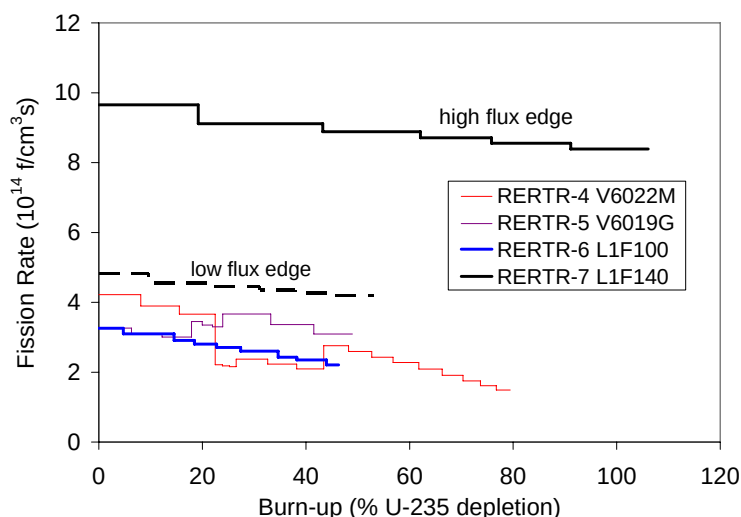


Figure 2. Power history of selected mini-plates from RERTR-4, 5, 6 & 7.

3. PIE RERTR-6

The PIE results from RERTR-6 provided the first insight into the irradiation behaviour of monolithic fuel. Given the propensity of the UMo fuel and Al matrix to react extensively in dispersion fuels there is considerable interest in the reaction layer, developed during fabrication and/or irradiation between the foil and the clad in the monolithic mini-plates. Of equal interest is the bond achieved by the joining process. Considerable effort is being devoted to the question of ‘what is an acceptable bond’ and how to determine, it as reported by Burkes at this meeting [2].

In RERTR-6 the reaction layer was generally thin, uniform and adherent. The extent of reaction was less in the U-10Mo mini-plates (~ 3 µm) than the U-7Mo (4-6 µm) as shown in Figure 3. There were departures from this behaviour and some examples are provided below. In all cases the bond between the fuel and clad was retained during the irradiation.

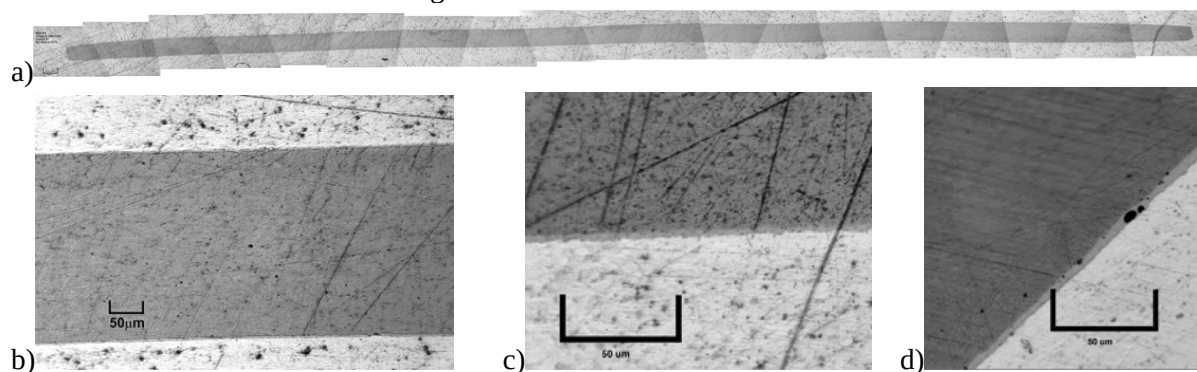


Figure 3. Optical micrographs of L1F040 (U10Mo) a), b) & c), reaction layer ~ 3 µm, and d) N1F030 (U7Mo), reaction layer ~6 µm. The dark phase is the fuel; the light phase is the cladding.

In Figure 3d), indications of the phenomenon that beset the U-Mo dispersion fuels are seen. Small voids are apparent in the reaction layer that are consistent with the voids that form in the reaction layer of the dispersion fuels. They have a viscous-like appearance and the voids close together appear to be linking up to form larger voids. Close inspection also shows the voids initiating at the reaction/cladding interface which is consistent with the fission gas bubbles seen in the dispersion fuels. Since the burn-up is moderate at $\sim 50\%$ ^{235}U , only the earliest signs of the phenomenon might be expected.

Figure 4 shows a transverse section of a foil which revealed that the foil does not appear to be flat within the clad. This was the result of using a $\frac{1}{4}$ " diameter pin and the 5-6 passes required to cover the plate surface. This was addressed by using a larger capacity mill that accommodated a $\frac{1}{2}$ " pin and reduced the number of passes required.

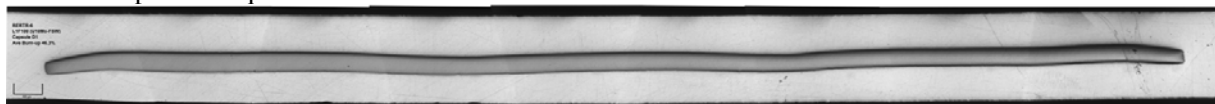


Figure 4. Transverse cross section of L1F100, U-10Mo fabricated by friction stir welding using a $\frac{1}{4}$ " diameter pin.

A mini-plate containing a 500- μm -thick foil delaminated upon sectioning in preparation for metallography. Plate thickness measurements prior to sectioning showed that the core edge of the mini-plate was marginally thicker than the outer edge. Examination of the post-sectioning montage shows a significant bulge in the midpoint of the plate as shown in Figure 5. Although not shown, the reaction layer was equivalent on both sides of the foil. If not in contact during the irradiation then the reaction layer would not have formed.

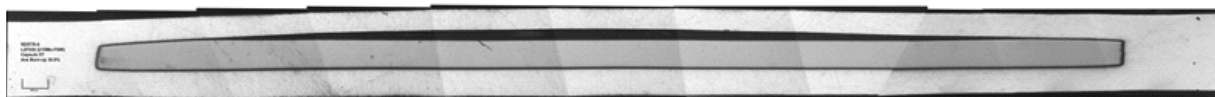


Figure 5. L2F030 (U-10Mo), 500 μm thick foil. Transverse cross section of the mini-plate which shows a delamination between the fuel and the cladding.

4. PIE RERTR-7

Preliminary PIE results from RERTR-7 reveal the same type of porosity that required a redesign of the dispersion fuel affects the irradiation performance of the friction-stir-welded monolithic-fuel. In Figure 6, porosity has formed in the reaction layer on both sides of the foil and the cladding. On one side the cladding has delaminated from the reaction layer. This most likely occurred during plate cutting for metallography after the irradiation was complete, otherwise the reaction layer would not be of a similar thickness to the other side of the foil. Nonetheless, it demonstrates the tenuous nature of the bond. Another FSW plate, not reported here, L1F120, displayed the same result.

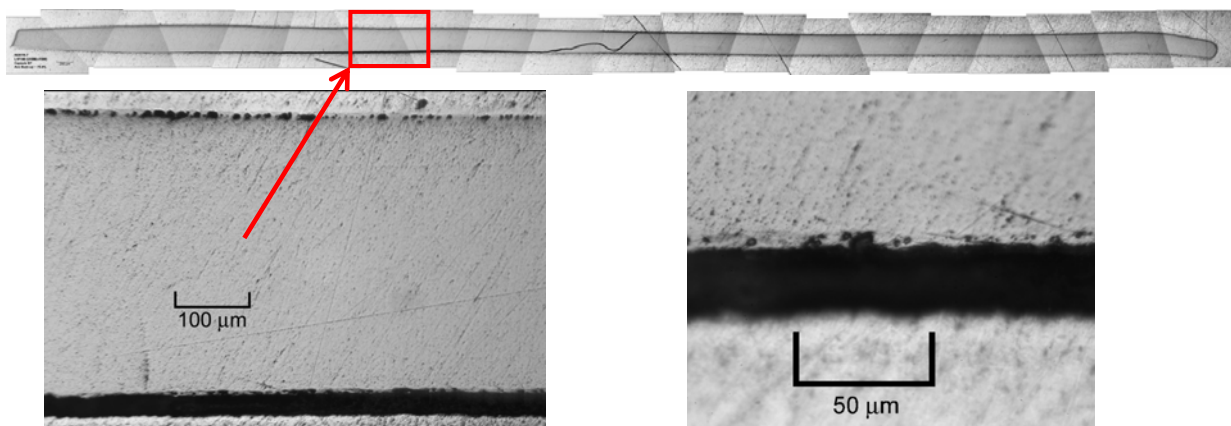


Figure 6. Transverse cross section and higher magnification optical micrographs of L1F140, U-10Mo, FSW showing porosity formation and delamination.

The onset of the porosity seems to correlate well with the current understanding of the behaviour of the dispersion fuels [3]. The cladding material is aluminium alloy 6061 and contains ~ 0.8 wt% silicon. At

reaction layer thicknesses of less than $\sim 4\ \mu\text{m}$ the layer has been shown to be stable and the results observed in RERTR-6 seem to support this. However at reaction layer thicknesses of greater than $4\ \mu\text{m}$, as observed for RERTR-7 ($\sim 6\ \mu\text{m}$) the layer is unstable. It is assumed that there is insufficient silicon available to maintain a dense, stable reaction layer and porosity formation results.

The morphology of the porosity appears different than that seen in the dispersion fuel. It does not clearly nucleate at the cladding-reaction interface and appears to occupy the full reaction layer thickness. It is however, a planar interface and residual stresses from fabrication of the plate coupled with the stresses from the differential thermal expansion of the foil and the cladding may also be contributing to the evolution of the pores.

RERTR-7 contained the first monolithic mini-plates fabricated by transient liquid phase bonding (TLPB) [4]. The aluminium-silicon eutectic that bonds the cladding to the foil during fabrication can be seen in

Figure 7. It has a non-uniform thickness of $10\text{--}40\ \mu\text{m}$ but appears to provide an effective interdiffusion barrier layer with no evidence of any porosity or subsequent delamination. The fabrication technique was not sufficiently developed and excess silicon built up in the rail area. This resulted in the formation of silicon rich brittle precipitates that led to delamination of the cladding as reported in Section 5. It did however, show that a high silicon content layer, thicker than the recoil distance in the cladding and



adequately bonded, may be effective in addressing the porosity and delamination illustrated in Figure 6.

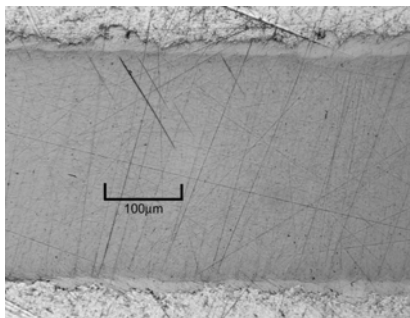


Figure 7. Transverse cross section and higher magnification optical micrographs of H1T020, U-12Mo, TLPB.

Figure 8 illustrates the swelling of the monolithic foils in RERTR-6 and RERTR-7. The strong flux gradient in RERTR-7 compared to RERTR-6 is evident in the peak swelling at the high flux edge of the foil. The reduced fuel swelling of the U-12Mo TLPB plates compared to the U-10Mo plates may be related to pre-irradiation reaction and reduced foil thickness rather than an effect of composition alone.

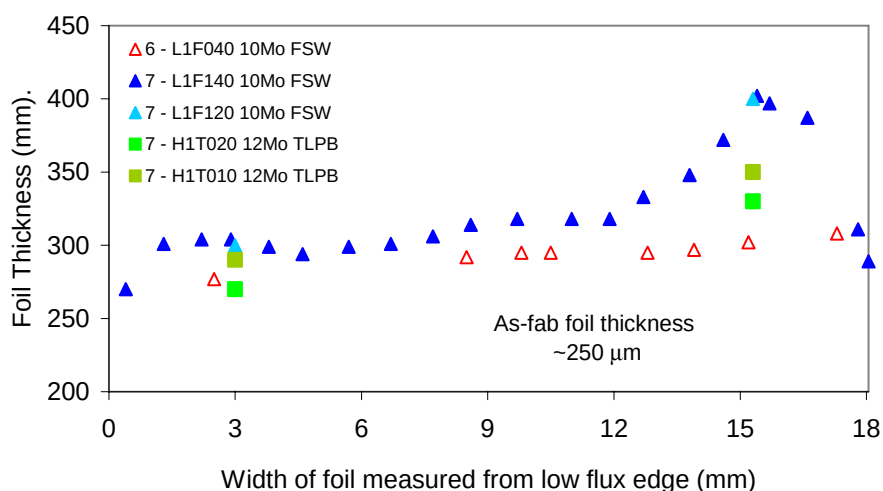


Figure 8. Plot of foil thickness across width of foil illustrating strong flux gradient and benefit of increased molybdenum content.

The observations of delamination in RERTR-7 suggest that the bond strength between the foil and the cladding in FSW plates is poorer than achieved by TLPB. This observation correlates with the initial results of pre-irradiation bond strength [2] reported at this meeting. As a remedy to this problem, a new FSW tool material has been identified that permits increased load and subsequently better bonding. Mini-plates fabricated by this method are scheduled for inclusion in RERTR-9.

5. RERTR-7: Plate Failure

During the RERTR-7 experiment, elevated levels of activity were detected in the Advanced Test Reactor primary coolant circuit after ~ 80 days. Sip testing of individual capsules at the end of the cycle revealed that at least one plate from the C capsule was leaching fission products [5].

Detailed examination of each plate from the C capsule under periscope in the Hot Fuel Examination Facility (HFEF) revealed that plate L1T020 had split open along 60% of the high flux edge (i.e. the edge closest to and facing the core) and across ~70% of the top of the plate as shown in Figure 9. The split exposed the fuel foil to direct contact with the primary coolant and resulted in corrosion of the fuel. There is visible staining of the plate surface as the fuel and fission products were eroded and flushed from the plate. Approximately 50% of the fuel alloy mass was lost to erosion as shown by the longitudinal gamma scan and plate thickness data (

Figure 10). Further evidence of this is found in the boehmite oxide layer on the plate surface. There is a distinct transition at the midpoint of the plate from a grey color indicating a lower surface temperature to a white color where the surface temperature was hotter and presumably the foil still in contact with the cladding. The mass of the plate in the as-fabricated condition was ~14.5g and post-failure the mass was ~11.1g. Since the initial alloy mass was ~6.3g, the loss of ~3.4g of fuel alloy represents ~53%. This correlates well with the gamma scan data. Detailed metallography is scheduled to examine the remains of the plate.

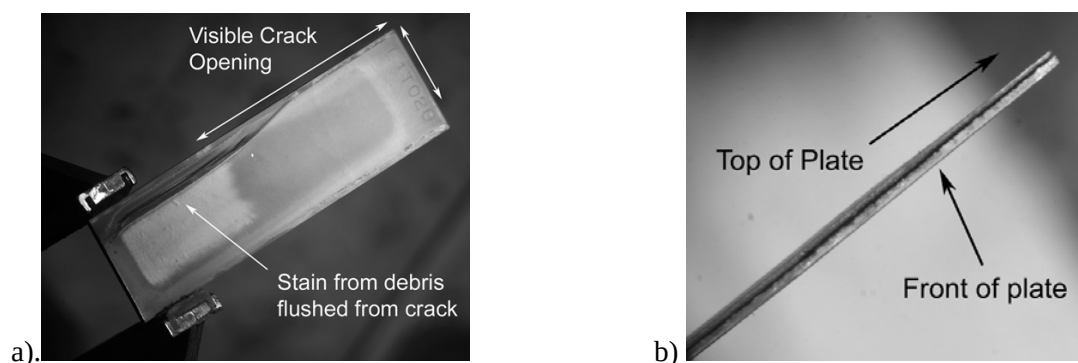


Figure 9. Photographs of plate L1T020, U-10Mo fabricated by transient liquid phase bonding.

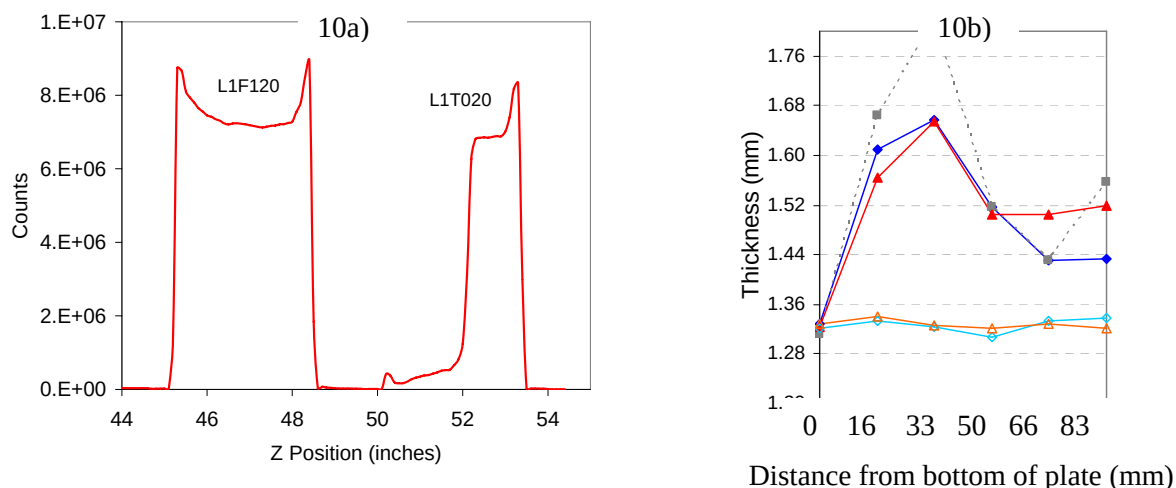


Figure 10.a) Longitudinal gamma scan (gross) showing the difference between a sound monolithic plate and the failed plate. b) Longitudinal thickness traces of L1T020 pre- and post-irradiation. Both graphs illustrate the loss of material in the upper half of the fuel zone.

The probable cause of the failure of L1T020 became apparent when the second TLPB plate in the C capsule, H1T020, was destructively examined. The data for this plate showed no loss of plate mass, normal gamma scans, and plate swelling that was consistent with other plates in the capsule. Metallography did reveal that plate failure was imminent as shown in Figure 11, which shows a transverse section through the mid-plane of the plate. The high flux edge of the foil has undergone significant swelling (35-45%) due to the accumulation of fission gas. The swelling has provided the driving force for the bond line to open and a crack to develop. The bond line can be traced to the edge of the plate. Examination of the bond line in an SEM has shown the presence of silicon rich precipitates which are believed to weaken the strength of the bond.

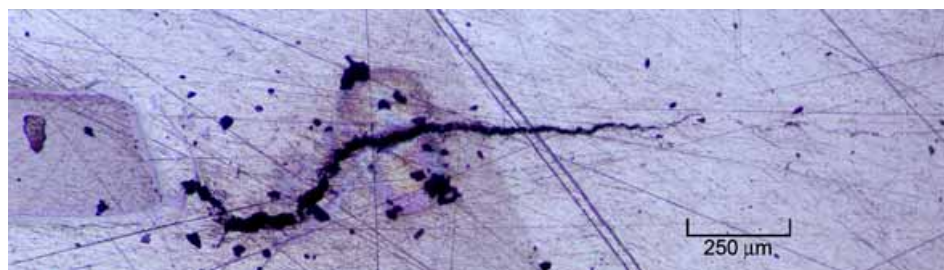


Figure 11. Transverse section through the mid-plane of H1T020. A crack starts at the edge of the foil and follows the bond line.

6. Summary

The PIE of RERTR-7 indicates that porosity formation is occurring at the interface between the foil and the cladding in FSW plates. No porosity was observed in TLPB plates. The observations of delamination of FSW plates correlate with the initial mechanical testing results. A number of developments are being pursued on the fabrication front to address some of the observations and should be implemented in RERTR-9.

7. Acknowledgements

The authors gratefully acknowledge the efforts of the Hot Fuel Examination Facility (HFEF) staff at the Idaho National Laboratory, in particular Jim Wiest, Paul Lind, Clay Brower and Julie Jacobs.

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COMPREHENSIVE OVERVIEW ON IRIS PROGRAM: IRRADIATION TESTS AND PIE ON HIGH DENSITY UMo/Al DISPERSION FUEL

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ABSTRACT

This paper presents the IRIS program results, on high density UMo/Al dispersion fuel. Different UMo powders (ground or atomized) and matrix compositions (pure Al or added with Si) have been tested in full size plates irradiated in the OSIRIS reactor. Very good results were obtained on ground powder based plates, up to high power level and burn-up. The positive effect of Si is evident for atomized fuel. First mechanical calculations were realized to simulate the meat tearing. Then analyses and post irradiation examinations permit us to highlight the likely beneficial parameters of the ground powders.

1. Introduction

The IRIS program has been launched in 1999 in the framework of the French FUMOG group¹ [1], to develop and qualify the LEU UMo fuel. It currently involves 4 full-sized experiments performed on UMo/Al dispersed fuel: IRIS1 [2], IRIS2 [3], IRIS3 [4] and IRIS-TUM [5], in chronological order. The IRIS 2 experiment, confirmed by the FUTURE [6] irradiation also carried out in this framework, evidenced the poor UMo (atomized)/Al fuel behaviour, with an unacceptable pillowing of the fuel plate. In 2003, this limit has been worldwide accepted and afterwards, many studies have been launched to solve this drawback by:

- modification of the UMo particles: ternary alloy (Zr, Ti) [7, 8, 9], type of powder [10],
- modification of the Al matrix: use of additives [7, 8, 11, 12, 13, 14, 15], stabilized porosity or replacement of Al [16],
- change in the UMo/Al interface: coating material, diffusion barrier [10, 17].

The IRIS program proceeded with the IRIS3 and IRIS-TUM experiments. The former has been defined by the CEA, with a specific collaboration with AREVA-CERCA, for the manufacturing aspects. The IRIS-TUM experiment has been defined in the framework of a collaboration between TUM, CEA and AREVA-CERCA. It has been realized for TUM in the framework of a CERCA/TUM contract, for the plate manufacturing, and in a CEA/TUM contract, concerning irradiation test [5]. Both irradiations aim at testing the addition of silicon in Al matrix, either with usual atomized powder (IRIS3) or ground UMo powder (IRIS-TUM). The choice of ground UMo was promoted by IRIS1 good behaviour.

This paper gives an overview on the four irradiations already achieved in the IRIS program. The major parameters governing the UMo/Al in-pile behaviour are discussed. Proposals are finally highlighted.

2. Description of the IRIS experiments

The IRIS program (from the name of an OSIRIS irradiation device) has been carried out in the OSIRIS reactor. All the plates are full sized, manufactured by AREVA-CERCA, with a classical hot rolling process.

The main differences in the manufacturing parameters are (Table1):

- the type of UMo powder, atomized or ground. The main differences are highlighted in § 4.2,
- the type of matrix, either pure Al or added with silicon up to 2.1 wt %.

¹ a collaboration between CEA, COGEMA, CERCA, FRAMATOME and TECHNICATOME ended in 2003

Note that the enrichment for the IRIS-TUM plates is higher (49.5 % wt ^{235}U). This enrichment increases burn-up and shorten the irradiation time to reach it.

The irradiation parameters are described hereafter:

- The higher maximum heat flux is twice the lower, i.e. 250-260 W/cm² for IRIS-TUM irradiation and 123-145 W/cm² for IRIS1 one,
- As a consequence, the fission rate covers a large range, from 1.5 to 4.4 10¹⁴ f/cm³/s.

	Experiment	IRIS1	IRIS2	IRIS3		IRIS-TUM	
Manufacturing data	Year	2000-2001	2002	2005-2006		2005-2007	
	Number of full-sized plates	3	4	4		4	
	Type of UMo powder	ground	atomized	atomized		ground	
	Mo content in UMo alloy (wt %)	7.6 or 8.7	7.6	7.2		8.1	
	Enrichment (wt% ^{235}U)	19.8	19.8	19.8		49.5	
	Si content in Al matrix (wt %)	/	/	0.3	2.1	0	2.1
	Matrix type	A5	A5	'Al99.7Si0.3'	'Al97.9Si2.1'	A5	'Al97.9Si2.1'
	Fuel loading (gU/cc)	7.9-8.3	8.2-8.3	7.8-8.0		8.4	
	As-fabricated meat porosity (%)	11-13	1-2	0.8-2.4		8-9	
	Cladding material	AG3NE	AG3NE	AG3NE		AlFeNi	
Irradiation data	Status of experiment	Complete	Stopped	Complete Stopped (0.3 Si)		Complete (2 plates) In progress (2 plates)	
	Location in OSIRIS reactor	17	52	14		11+17	
	Max heat flux at BOL (W/cm ²)	123-145	238	196		250-260	
	Max. cladding temperature at BOL (°C)	68-73	93	83		103	
	Number of irradiation cycles	10	4	7		7	
	Total duration (full power days)	241	57.9	130.6		133.2	
	Burn-up: plate average at EOL (% ^{235}U)	46.9	32.5	48.8		55.9 ^(a) (LEU equivalent)	
	Burn-up: average at MFP at EOL (% ^{235}U)	54.0	39	56.5		64.4 ^(a) (LEU equivalent)	
	Burn-up: local max at MFP at EOL (% ^{235}U)	67.5	39.7	58.8		74 ^(a) (LEU equivalent)	
	Fission density: plate average (f/cm ³ UMo)	3.2*10 ²¹	2.2*10 ²¹	3.4*10 ²¹		4.2*10 ²¹ (a)	
	Fission density: maximal at MFP (f/cm ³ UMo)	4.6*10 ²¹	2.7*10 ²¹	4.1*10 ²¹		5.6*10 ²¹ (a)	
	Average Fission rate (f/(cm ³ .s))	1.5*10 ¹⁴	4.4*10 ¹⁴	3.0*10 ¹⁴		3.6*10 ¹⁴ (a)	

^(a) to be confirmed

Table 1 : Manufacturing and irradiation data for the IRIS program irradiations of full-sized plates

The plate thicknesses are measured before and after each cycle, along 5 longitudinal lines (250 points each) and one transverse (~120 points) at the maximal flux plane (MFP). The evolution of the plate is thus easy to follow. For safety reason, the thickness increase acceptable limit is 250 µm. Above this value, the irradiation is immediately stopped.

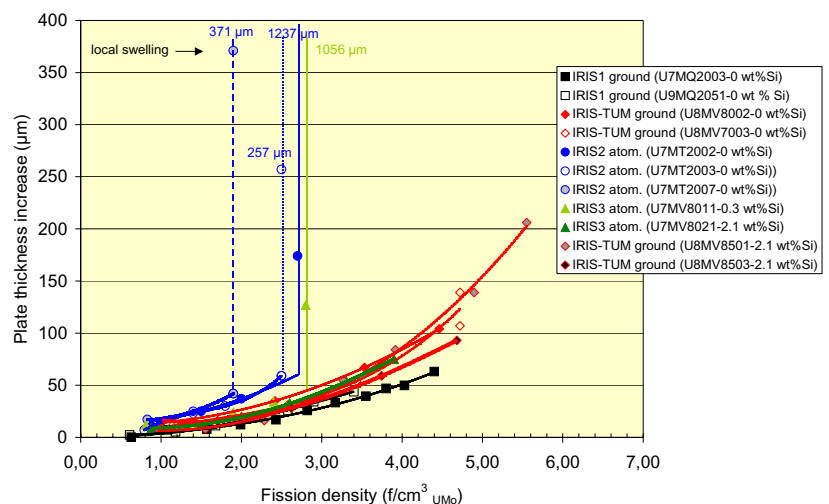
Destructive PIE have been performed for IRIS1 and IRIS2 plates. They are planned in 2007 for both IRIS3 and IRIS-TUM plates. They consist in optical microscopy, SEM, EPMA and XRD analysis.

3. Results

Figure1 shows the plate thickness increase of all the IRIS plates as a function of fission density. It underlines the great difference in behaviour between the atomized UMo based fuel and the plates made of ground particles. For the former, at a critical fission density, the local plate thickness increases dramatically, of several hundreds of microns and even up to twice the initial thickness, at the maximal flux plane. In this case, a large cavity inside the meat results in a tear of the fuel (IRIS2), while plates made of ground UMo keep their integrity (IRIS1, IRIS-TUM).

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

The silicon addition to Al matrix drastically decreases the swelling of the plates containing atomized powder (IRIS3). Indeed, the fission density above which abnormal thickness increase occurs ('critical fission density'), is far higher (Table 2). But



the Si content has to be high enough, otherwise the fuel behaviour is not improved [4]. The silicon positive effect on atomized powders has also been demonstrated [11, 12] on RERTR irradiated mini-plates, with a substantial reduction of UMo/Al interdiffusion. However, first results on RERTR-7 revealed a probable performance limit for fuel with 2% silicon in Al matrix in high flux conditions [12]. Higher Si content is then recommended.

Moreover, considering the recent results on IRIS-TUM, the Si effect seems to be lower in ground UMo fuel. Plates swelling is already limited for free Si meat.

	IRIS1	IRIS2	IRIS3		IRIS-TUM	
Si wt% in Al	0	0	0.3	2.1	0	2.1
Critical fission density (10^{21} f/cm ³)	>4.6	2	2.9	>4.1	>4.7	>6.4

Table 2 : Critical fission density.

IRIS-TUM and IRIS3 plates have not yet been examined. But according to past IRIS1 and IRIS2 PIEs, fuel meat can be described as follows:

1. Fission gas bubbles in the fissile particles. These bubbles logically increase with burn-up, both in size and density.
2. A chemical reaction between the UMo alloy and the Al matrix inducing the formation of an interaction layer (IL) at the UMo/Al interface. Its characteristics can be summarized as:
 - a. Enhanced by irradiation and inter-diffusion from Al into UMo (and IL) and UMo into Al.
 - b. a few microns thickness (5-6 μm in IRIS1 and 7-10 μm for IRIS2) and a rather large volume fraction in the meat, whatever the UMo powder type, $\sim 36\text{-}42\%$ ⁽²⁾ for IRIS1 and $\sim 45\text{-}49\%$ ⁽²⁾ for IRIS2, though a less lower fission density for IRIS2 experiment. In fact, there is some gradient of these IL dimensions inside the meat, relevant to heat flux. But the apparent volume of UMo is quite uniform due to both IL growth (UMo consumption) and fission gas formation (UMo swelling). However in high IL content areas, the UMo particles are smaller than initial ones. The residual Al is very low, of about 5 to 10 %.
 - c. the IL has been characterized as an Al-rich phase. Its average composition can be formulated as 'UMoAl_x' (but it is probably a mixture of phase) with Mo/(U+Mo) close to its initial value and $x=\text{Al}/(\text{U}+\text{Mo})$ in the range of 4.6 to 5.8 for IRIS2 and 6 to 8 for IRIS1, carried out at lower temperature. Note that for the FUTURE experiment, irradiated at higher temperature, x is lower (3.3 to 4.7) [6]. This tendency is also confirmed by the UMUS irradiation ³ performed at high temperature on ground powder fuel plates (x is ~ 3 , for a calculated fuel temperature of $\sim 225^\circ\text{C}$) [18].
 - d. no crystalline structure was determined, neither for IRIS1, nor for IRIS2. No existing ternary (UMo₂Al₂₀ and U₆Mo₄Al₄₃) nor binary (UAl₂, UAl₃, UAl₄) phases were identified. This IL was assumed to be amorphous [19]. A crystalline structure, the UAl₃ phase, was evidenced by neutron diffraction only for high temperature experiment (300-400°C), performed with a fuel rod [20].
 - e. presence of fission products and fission gas in the IL. But bubbles precipitate inside the IRIS1 IL, while no or very few bubbles in IRIS2 one.
3. Fission products accumulation, mainly fission gases (Xe) (but also Zr, Nd, Cs...) at the IL/Al interface (Figure 2.c and 2.e) for both type of fuel. Such an accumulation has been previously observed in U₃Si₂/Al fuel [21]. In the case of IL connection, fission gases accumulate at the IL/IL interface. The porosities developed at the IL/Al interface, are filled with fission gases. Their size and number increase, along the cross section fuel, in relevance with heat flux. The role of Mo is probably determinant in the large cavities formation, since such behaviour has not been evidenced in UAl_x/Al neither in U₃Si₂/Al fuels, even at high irradiation conditions [22].

² determined by image analysis at the maximal flux plane

³ the UMUS fresh fuel is comparable to the IRIS1 one. Irradiation is characterized by a peak heat flux of 275 W/cm², a maximal fission density of $4.5 \cdot 10^{21}$ f/cm³ UMo, and a fuel temperature calculated to be higher than 225°C (due to a dramatic cladding oxidation). Almost all the Al matrix is consumed at MFP.

We can assume, according to the fuel plate thickness increase (see Figure 1), that neither IRIS-TUM nor IRIS3 (with 2.1 wt% Si) plates developed large interconnected cavity.

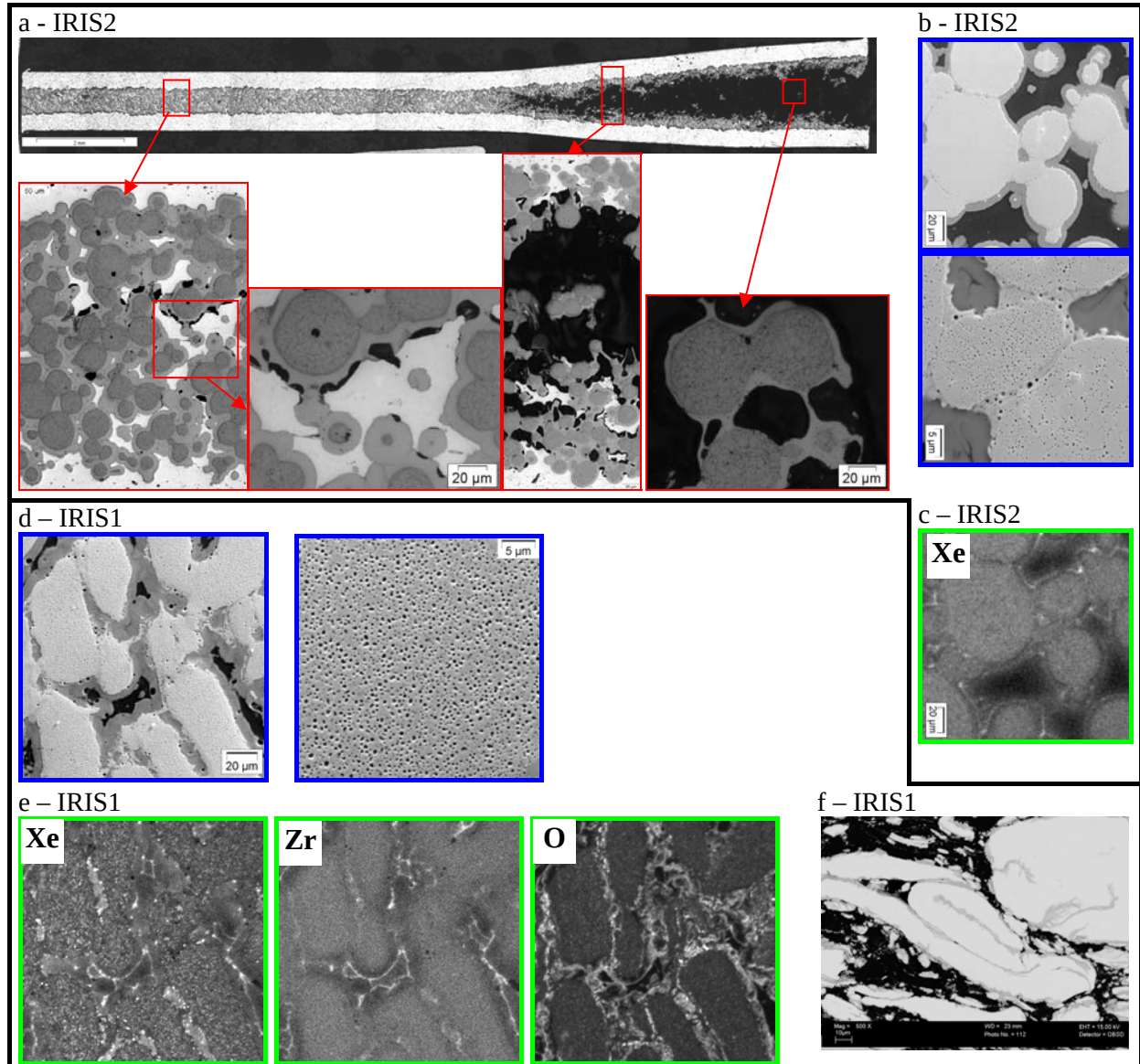


Figure 2: Metallographic examinations on IRIS2 at MFP (a) SEM images on IRIS2 (b), X-ray maps of IRIS2 plate (c), SEM images on IRIS1 (d), X-ray maps of IRIS1 plate (e), IRIS1 fresh fuel (f).

4. Discussion

In this section, we first propose a mechanism illustrating the abnormal swelling, when it occurs, and then try to underline some differences in material parameters likely to affect the meat behaviour.

4.1 Breakaway swelling phenomenology

4.1.1 Physico-chemical aspects

The thermodynamic instability between UMo and the Al matrix induces the formation of an interaction layer. Its growth and the fissions production imply a permanent evolution of the IL/Al interface. For comparable irradiation levels, the IL growth kinetics depends on the UMo particles and the matrix features. The fission products implanted in the Al matrix are progressively reached and carried away by this moving interface, where they accumulate. Fission products diffusion mechanism could also participate to enrich this interface. This accumulation certainly damages the interface mechanical strength. Among the fission products, we find rare fission gases, high yielded formed in fission and mainly composed of large Xe atoms. Due to their limited solubility, the gas atoms generally diffuse until they are trapped by radiation damages (vacancies clusters or dislocation loops), or precipitate at grain boundaries, voids or pre-existing pores. At the IL/Al interface, they lead to the

formation of gas-filled bubbles, giving evidence of high enough gas concentration. The factors promoting or not bubbles formation are still unclear. But the bubbles morphology and orientation indicate a low energy surface between IL and Al (Figure 2.a). The diffusion of even more gas atoms and vacancies induces bubbles growth, their possible interlinkage resulting in a large swelling of meat. This phenomenon has much to do with poor thermo-mechanical properties of the IL itself in these zones during irradiation.

Considering PIE room temperature microhardness measurements, the IL seems to be harder than the surrounding Al or UMo. But, the large swelling areas illustrated by Figure 2.a clearly indicate a high viscosity of the IL during irradiation; hence acceleration of gas mobility and bubbles coalescence (permanently fed during irradiation process).

4.1.2 First modelling of plate deformation

This part deals with mechanical simulations performed on a simplified structure (Figure 3.a), considering an initial cavity (a 'penny shape' of R radius) formed by the phenomenon previously described. The structure represents a pressurized cavity (radius R), with an intact circular surrounding area. Due to the axi-symmetry of this system, the calculations are performed on half plate thickness. These calculations are the very first and only the cladding is considered. They aim at estimating (order of magnitude) the temperature and/or the pressure gradients necessary for the plate (e.g. cladding) to swell, typically of several hundreds of microns.

Two mechanisms are studied:

- ✓ An immediate deformation due to a high pressurisation of cavity driving to plastic deformation. Note that local thermal buckling due to a thermal gradient seems improbable, since a high temperature should be necessary,
- ✓ A delayed deformation due to creep.

The following material properties are considered (those of the cladding):

$E = 67000 \text{ MPa}$, $\nu = 0,33$, $\alpha = 24,3 \cdot 10^{-6} \text{ K}^{-1}$, $\sigma_Y = 80 \text{ MPa}$, $h = 750 \text{ MPa}$, $\sigma_R = 176 \text{ MPa}$

For the creep law we have used: $\dot{\epsilon}_{vp} = 7,7 \cdot 10^{-54} \sigma^{6,12}$ (σ in Pa and $\dot{\epsilon}_{vp}$ in s^{-1}).

Firstly, considering a pressurized cavity of radius R, the elastic solution of the problem is given by, for

maximal stress: $\sigma = \frac{3}{4} p \left(\frac{R}{e} \right)^2$. The maximal pressure is then: $P_{\max} = \frac{4}{3} \sigma_R \left(\frac{e}{R} \right)^2$ with σ_R the ultimate

stress in traction. To have a lateral stable cavity this equation has to be verified. For example, a pressure of 150 bars should be necessary to observe a radial increase of a 5 mm diameter cavity; respectively 40 bars pressure for a 1 cm diameter cavity. For this case, the plate deformation thus induced would reach only 50 μm , much lower than experimental observations. The plasticity modelling is thus not sufficient to induce plate tearing.

Considering creep mechanism, we obtain the plate deformation evolution represented in Figure 3.b and c. The deformation would reach $\sim 500 \mu\text{m}$ in 20 days (480 hours i.e. around a cycle), for a ~ 15 bars pressure gradient. Fewer days would be necessary for a higher pressure. The calculated values are relevant to those experimentally observed. Due to the material properties considered, this approach is certainly conservative.

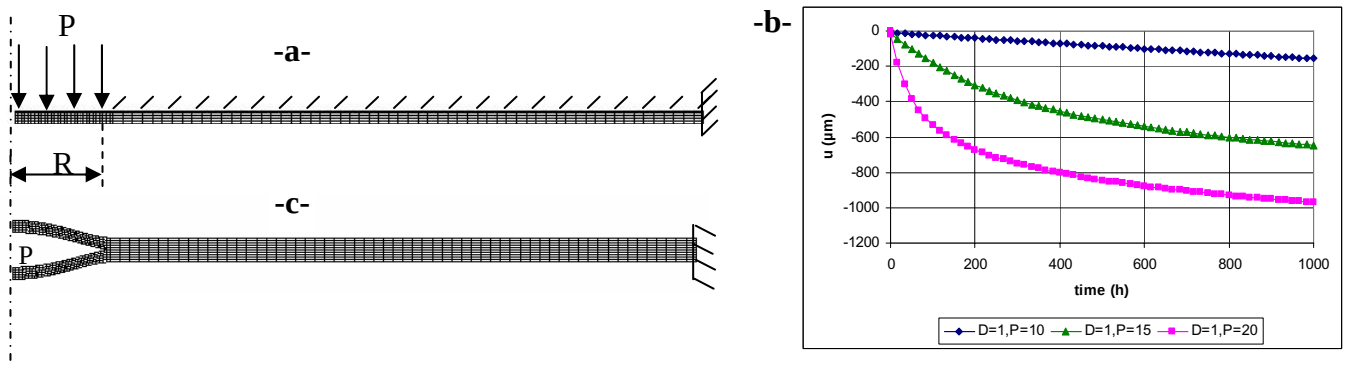


Figure 3: simplified structure (half thickness plate, P=cavity pressure applied on the cladding) (a), Plate deformation evolution, considering creep mechanism, for a 1 cm diameter initial cavity and for different pressures (10, 15 and 20 bars) (b), Example after 1000 hours for P=20 bars (c).

With these assessments, creep is thus a possible mechanism inducing plate deformation (plate geometry). This deformation probably proceeds simultaneously with the radial extension of cavity. Our conclusion and the orders of magnitude obtained are relevant to past Russian computational analysis [23].

Nevertheless, during plate cooling (stop reactor), an additional buckling mechanism due to cladding compression could increase the plate deformation. This phenomenon has not yet been assessed.

4.2 The UMo particles type: its role and the resulting parameters

As previously discussed, the observed breakaway swelling of plate meat (in plate geometry) is a consequence of the fission products (mainly gases) accumulation and evolution at the IL/Al interface. Considering the non destructive and destructive PIEs of IRIS plates, it seems that the irradiation levels (fission rate and cladding temperature) and the amount of interaction layer are necessary but not sufficient conditions for abnormal swelling to occur. At high enough gas concentration, the IL properties seem to be the driving force. Additional examinations are needed to better understand the in-pile behaviour. Nevertheless, atomized and ground UMo based plates differ in several parameters likely to affect fuel behaviour:

- ✓ Composition is probably the main one. Such a poor behaviour has never been observed in UAl_x/Al , nor in $\text{U}_3\text{Si}_2/\text{Al}$ systems, though an IL is also formed. This underlines the crucial role of Mo (but not sufficient) in the IL instability, though the difficulty to differentiate its influence in the IRIS fuels behaviour. The $\text{Al}/(\text{U}+\text{Mo})$ ratio (x) is probably not a major parameter. Indeed, measured by EPMA, it seems to decrease with the irradiation temperature whatever the fuel. In the UMUS experiment, this ratio is the lower measured ($x \sim 3$) [18], without any large voids formation, in spite of high temperatures. This point has to be checked with the IRIS-TUM examinations, where a low ratio is expected but with limited bubbles formation at the IL/Al interface. In fact the key difference probably stands in the higher oxygen content found in the ground particles fuels, typically a few thousand ppm on ground UMo (IRIS1), compared to the composition of IRIS2 UMo powder with an oxygen content ten times as low. The oxygen is introduced during grinding process as an oxide layer (UO_2) surrounding irregularly the UMo particles (Figure 2.f). After irradiation, EPMA X-Rays maps (Figure 2.e) show the presence of this oxygen all over the IL. It apparently has a positive effect on the mechanical properties of the IL (which never looks like a plastic phase) in hot irradiation conditions. Silicon addition to aluminium also seems to improve the mechanical properties of the IL formed, in case of atomized powders. Moreover, composition likely governs other under irradiation properties: gas retention, yield stress ...
- ✓ Morphology/granulometry: the shape of ground particles is very irregular (and the size distribution is larger). This could improve both meat cohesion and cohesion between the meat and the cladding, thus improving the plate mechanical behaviour. Another consequence of this morphology is a residual porosity in Al matrix of about 10 vol. %, against 1-2 vol. % for spherical atomized powder. This porosity could act as a reserve for fission gases. In irradiated fuel, the residual porosity is very low and certainly has disappeared. Additional examinations and measurements should be useful for a better comparison between fresh and irradiated fuels.
- ✓ Microstructure: for atomized particles, fission gases are retained in small bubbles mainly located at UMo grain boundaries (Figure 2.b), characterized by Mo depletion. It is to be feared that, at higher burn-up and temperature, this microstructure facilitates diffusion path. Inside ground UMo, the fission gas is uniformly distributed in the grain volume (Figure 2.d). However, the role of this parameter is not clearly determined. At last, note that grinding process induces high concentration of defects, as a possible advantage to trap gas atoms.
- ✓ Structure: although Mo has been chosen to stabilize the UMo γ -phase, the plates made of ground powder contain a high amount of α -phase. Just traces are detected in atomized powder. After irradiation, there is evidence of its transformation in γ phase. This difference is just mentioned without any evidence of its impact.

These differences are maybe not exhaustive.

4.3 Further UMo LEU fuel development

Considering the previous results and discussion, the main objective of UMo LEU development should consist in reducing the interaction layer (kinetics slow down) and above all improving the IL properties yet obtained. All the means increasing the fission gases trapping would also improve fuel performance.

The use of alloy element, either in UMo or in Al, is one of the approaches. Our studies are in progress concerning ternary alloy, UMoTi [9], and Si influence in Al [15]. The reactivity of both system is currently studied, UMoTi/Al and UMo/AlSi. The recent IRIS3 irradiation carried out on atomized UMo was successful for 2.1 wt % Si addition in Al matrix [4]. The role of silicon is largely studied, based on out-of-pile and in-pile works [8]. Recent RERTR6 and RERTR7 results encourage its use for atomized powders. But a fuel performance limit already appears. Moreover, reprocessing aspect should be treated. Ti effect has been tested in RERTR7B, whose PIE are waited for [16].

UMo plates made of ground particles were successfully irradiated at high burn-up and heat flux. We assess a major role of oxide presence. But grinding process is not currently used in industrial UMo fuel fabrication. We developed a thermochemical treatment [24, 25] to control the oxide formation around atomized UMo particles. Out-of-pile studies attested that it prevents IL formation and that oxide way could be promising as regard to the in-pile behaviour. But we think that under irradiation its role would result in IL modification rather than in IL prevention (unless a large oxide thickness). A future French irradiation, IRIS4, is foreseen in OSIRIS reactor, at the end of 2007. The objective is to test and discriminate the influence of an oxide layer coating atomized UMo particles.

The development of UMo monolithic is still in progress [26], as an objective to decrease the interaction product.

5. Conclusions

The IRIS program gave a better understanding on the UMo fuel in-pile behaviour. Even if irradiation conditions are important parameters, the type of powders seems to be the major one. Good results have been obtained with ground powders, without any abnormal swelling at high burn up and temperature, though the plates made of atomized powders revealed a limit in performance. Some differences in characteristics between ground and atomized powders are discussed. But additional analyses are needed to properly differentiate them. A first mechanical modelling is proposed. It should be completed to identify the more likely mechanism in the plate abnormal swelling. Few investigations are in progress either to limit the interaction layer formation or to modify its properties.

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NOVEL TRENDS IN FUEL AND MATRIX ALLOYING TO REDUCE INTERACTION

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ABSTRACT

Novel approaches to fuel and Al matrix alloying are discussed that are aimed to reduce interaction between UMo fuel and matrix. A concept of U-Mo fuel alloying has been elaborated that consists in producing an alloy of a two phase structure, viz., basic γ (U-Mo) and intermetallic phases. The intermetallic phase shall have the maximal density of uranium, a low molybdenum content, be well compatible with aluminium, highly irradiation resistant and precipitate along grain boundaries. The latter condition allows fuel particle production via grinding. Using a thermodynamic approach and phase diagrams, optimized systems were selected that meet those requirements. The first results of developments are presented. Also considered are methods of fabricating alloyed fuel by powder metallurgy, i. e. by sintering powders of initial components followed by grinding resultant ingots.

Novel trend in alloying a matrix to decrease diffusionability of aluminium (alloying with high temperature resistant additives) is under consideration. It consists in applying aluminium alloys SAP (sintered aluminium powder) as a matrix. They contain 6-19% Al_2O_3 as intergranular thin interlayers. Aluminium insoluble oxide interlayers in a structure have to serve as diffusion barriers against diffusion of aluminium into U-Mo fuel. Preliminary technological studies and compatibility tests demonstrated the potential of this approach.

Also under consideration is feasibility of combining methods of protection against interaction to produce a qualitatively novel effect

1. Introduction

It is known that the basic reason for delaying the program of transition from HEU to LEU fuel of research high flux reactors is interaction between UMo fuel and aluminium matrix [1]. Presently the basic ways of resolving this problem have been contemplated. They are alloying of fuel, alloying of a matrix, creating protective barriers, application of monolithic UMo fuel.

In terms of technology fuel and matrix alloying is the most optimal version since the process of fuel fabrication becomes complicated to a less extent. The first results that were acquired although evidence a lower interaction do not allow the complete alleviation of the problem.

This paper deals with qualitatively novel approaches to fuel and matrix alloying that might help to resolve the interaction problem.

2. Alloying of fuel

Investigations demonstrate that elements of the γ -(U-Mo) phase solid solution such as Zr, Nb, Ti do not essentially decrease the interaction with aluminium [2]. But they might be considered to be extra microadditives to a main alloying element. The assessment of the tolerable content of alloying elements shows that to keep the uranium content of alloys at the level of 14–15 g/cm³ they have not to exceed 2% on the average.

A concept of UMo fuel alloying has been elaborated that lies in producing a two phase structure alloy, viz., the basic γ -(U-Mo) phase and an intermetallic phase [2]. The intermetallic phase shall have the maximal content of uranium, a low molybdenum content, be well compatible with aluminium, be highly irradiation resistant and precipitate along grain boundaries. The latter condition allows fuel particle production via grinding.

Using a casting method a group of alloys were fabricated with additives of Si (0.5 – 3.0%) as well as Si + Al (1.2% in total) and various Mo contents within 6 to 9% and examined metallographically. The microstructures of some alloys are shown in Fig. 1.

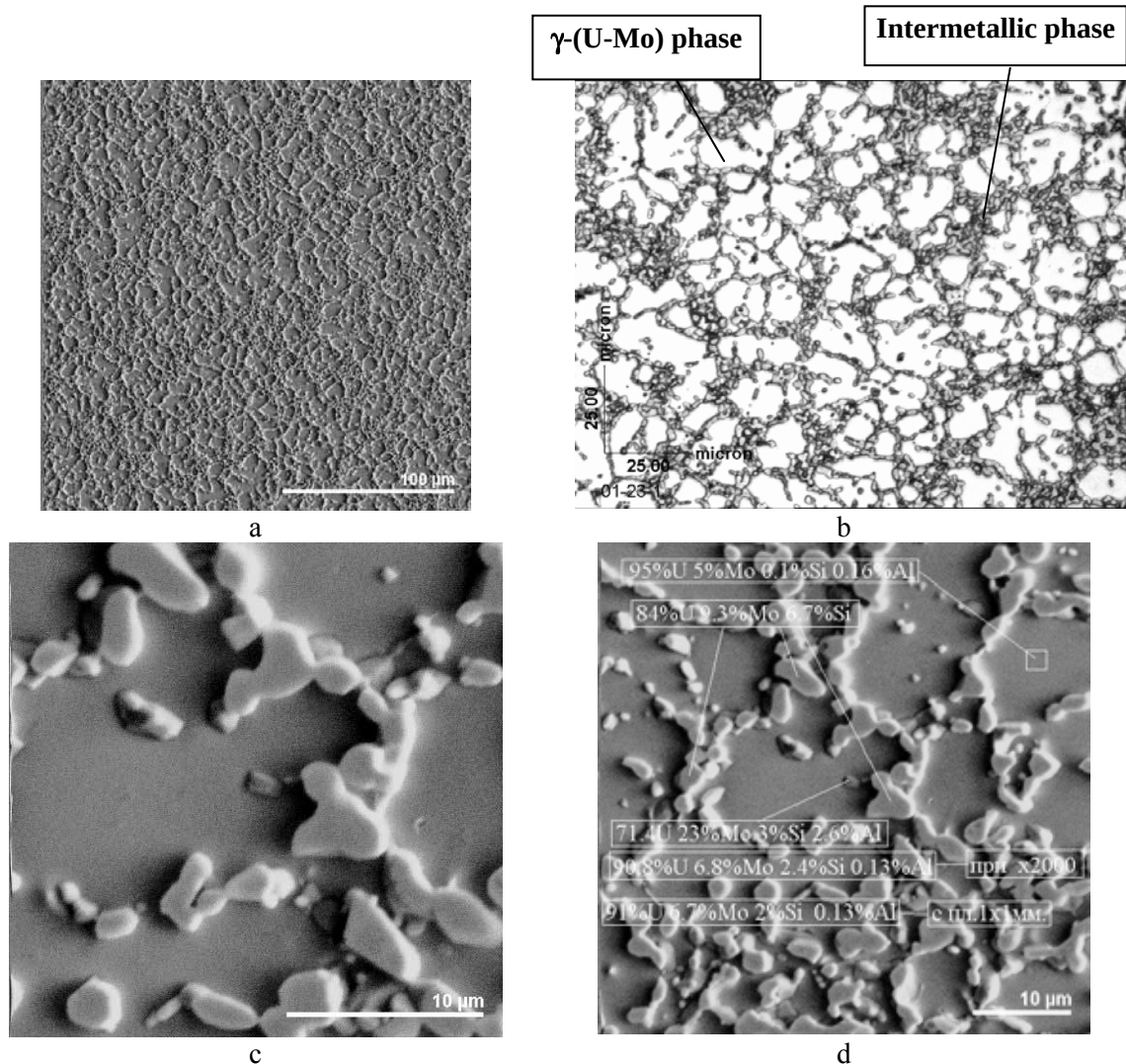


Fig.1. Structures of alloyed UMo two-phase alloy; a, b – U + 6.2Mo + 1.2Si;
c, d – U + 6.8Mo + 1.0Si + 0.2Al

After grinding the weakly interacting with Al intermetallic phase is primarily available at the fuel particle surfaces, forming some kind of a protection barrier. Figure 2 (a, b) illustrates structures of fuel compositions with Al matrix in the original condition and after anneal at 600°C for 6 hours. Fuel was prepared via grinding. It is well seen from Fig. 2b that the intermetallic phase weakly interacting with aluminium is primarily available at the fuel particle surfaces, that is why the interaction is of a nodular mode. The acquired results indicate the feasibility of diminishing the interaction through this method.

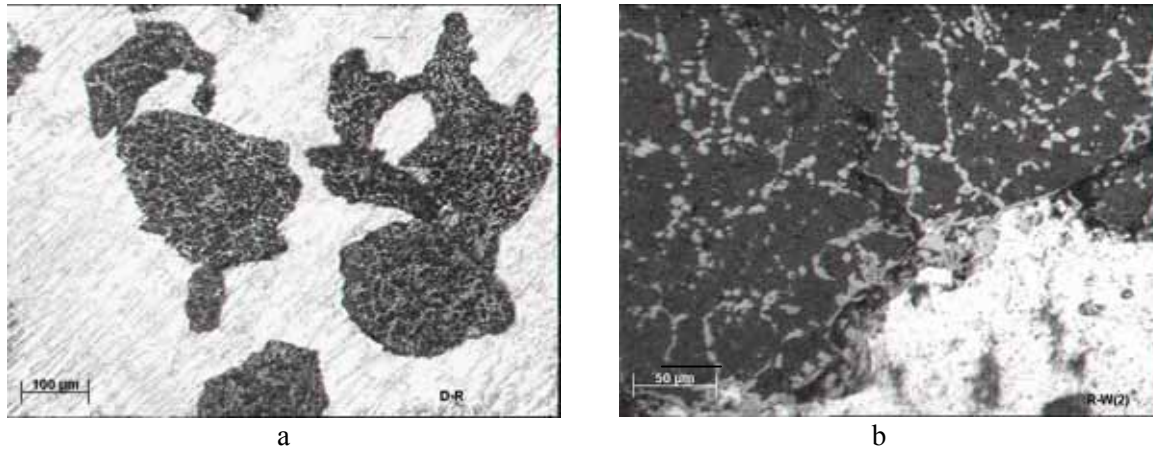


Fig.2. Microstructure of fabricated fuel composition with UMo alloy, a – (U+6.2Mo+1.2Si) + Al as fabricated; b – (U+6.8Mo+1.0Si+0.2Al) + Al - 6 hours annealed at 600 °C

The main challenge in alloying UMo fuel is a decrease in the γ -(U-Mo) phase stability at the expense of a reduced molybdenum content when alloying elements, specifically, aluminium are added. The same effect in less extent takes place when alloying with silicon. Some molybdenum leaves the γ -U phase for the intermetallic phase, depleting in this way the γ phase solid solution in Mo, which shall decrease its stability. As it might be seen from figure 1d molybdenum is present in the intermetallic phase of uranium with silicon which is undesirable. To prevent this a thermodynamic analysis using phase diagrams was implemented.

As a rule, the alloying of UMo alloy with elements that form intermetallic compounds with uranium leads to the undesired type of the phase diagram with ternary intermetallic phases that comprise molybdenum, e.g., U-Mo-Al, U-Mo-Si, U-Mo-Sn etc systems (see Fig. 3a) [3]. For molybdenum not to enter an intermetallic phase but to remain in a solid solution with uranium the type of the phase diagram (Fig. 3b) has to be realized when the phase triangle is restricted by U-U₂Mo-UX phases where UX is a binary intermetallic compound of uranium and an alloying element. This version is feasible in the U-Mo-C system (Fig. 3b) in which theoretically ternary compounds must not form close to the uranium angle of the phase diagram. The confirmation of this might be seen in Fig. 4 that illustrates the distribution of the elements of the U-9Mo alloy with a carbon impurity in the characteristic radiation of U, Mo and C. The unavailability of molybdenum in carbide phase inclusions along the grain boundaries of the γ -solid solution of U-Mo is clearly seen.

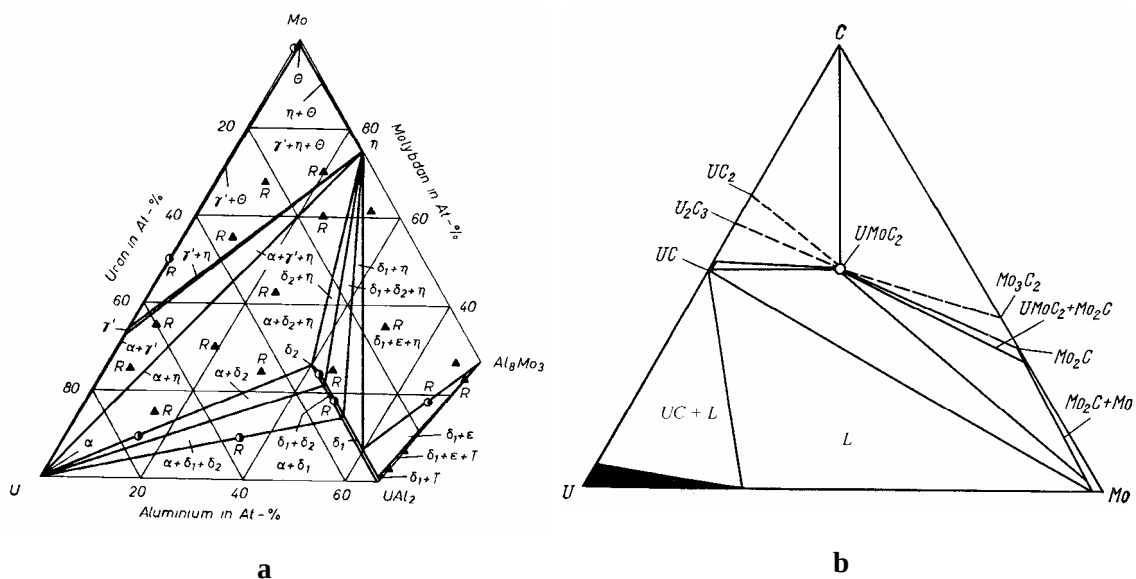


Fig. 3. Types of phase diagrams of UMo alloy with elements that form intermetallic compounds with uranium, a – system with ternary compounds (U-Mo-Al) [3], b – isothermal section of U-Mo-C phase diagram at 1500 °C

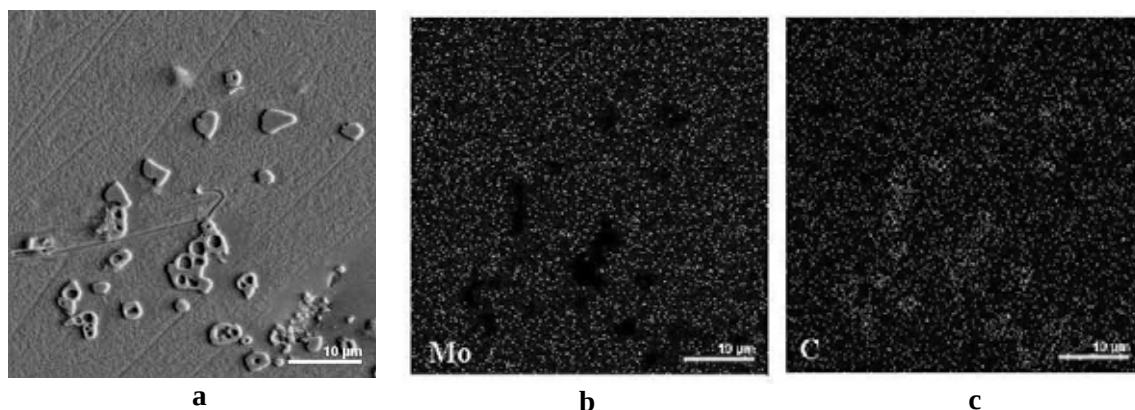


Fig.4. Distribution of elements of U-6.5Mo alloy with carbon impurity in characteristic X-ray radiation a – secondary electrons, b – Mo, c - C

The other way to avoid the Mo content reduction in the γ (U-Mo) phase upon alloying is to create a non-equilibrium alloy structure. This is put in practice when fabricating alloys by the powder metallurgy method, viz., sintering initial component powders, followed by grinding resultant ingots. The main advantages of the latter method are the following:

- feasibility of producing a thermodynamically non-equilibrium structures of fuel alloy (inert to Al interaction), that provides a decreased emergence of Mo from γ -phase;
- a qualitatively different composition of a powder surface layer after grinding that simplifies protective thermochemical treatment, particularly, oxidation;
- produced alloys are adequately brittle and readily ground.

Alloys were prepared from powders of uranium, molybdenum, silicon and other elements. The process dynamics was studied by a differential thermal analysis. During sintering uranium and silicon reacted exothermally. The resultant alloys are brittle and easily ground. Figure 5 presents microstructures of some alloys produced by the powder metallurgy method. It might be seen that the produced structure differs from the equilibrium one (Fig. 1) and has a great number of inclusions of brittle phases. The powder metallurgy process is promising, however, needs thorough investigations and final developments.

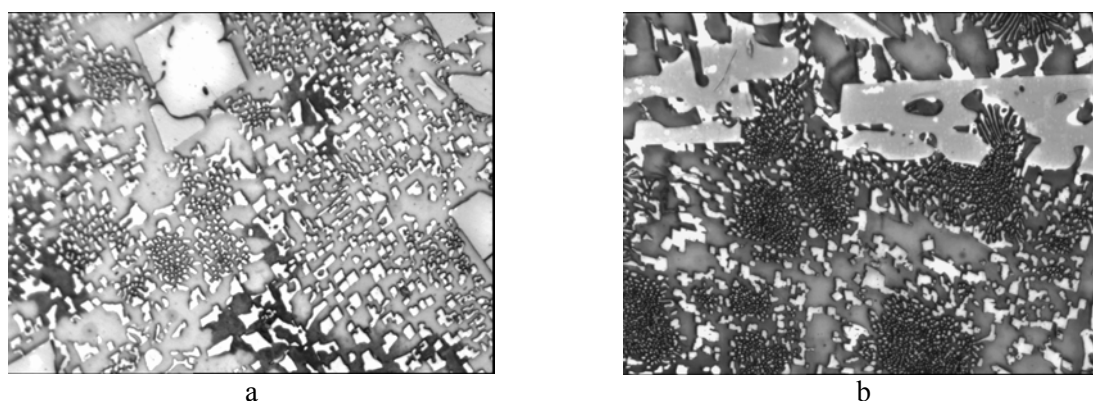


Fig. 5. Microstructure of alloys produced by powder metallurgy method, a – (U+6.5Mo+1.4Si); b – (U+6.5Mo+1.1Si+0.6Fe)

3. Alloying of matrix

Three trends of matrix alloying are under consideration now:

- alloying with magnesium;
- alloying with silicon to obtain an optimal structure of an interaction layer;
- alloying with high temperature resistant additives to decrease diffusionability of aluminium.

In our view the third direction of alloying – the alloying of a matrix to decrease the diffusionability of aluminium – is a qualitatively novel one and most promising. At the same time the main problem is related to the technology and method of introducing alloying additives, since they are little soluble in aluminium, and in the aluminium structure they are present as intermetallic solid phases, which involves difficulties in the deformation processes during fuel element fabrication. That is why, aluminium alloys of the SAP (sintered aluminium powder) type in the form of particles 0.05-0.2 mm in size were used as a matrix [4]. They contain 6-19% Al_2O_3 as intergranular thin layers. Thanks to this fact they are capable of straining and sintering while retaining their high thermal conductivity since Al is the base of alloys. SAP granules are produced via milling and conglomerating a fine aluminium powder $\sim 1\text{ }\mu\text{m}$ in size in ball mills. The SAP particle structure is schematically shown in Fig. 6a. Aluminium insoluble oxide coats have to serve as diffusion barriers against diffusion of aluminium into UMo fuel.

Specimens of SAP matrix fuel rods clad in CAV-1 alloy were fabricated (Fig. 6b). During 4 hour anneal of the specimens at 630°C granules of UMo fuel present in a cladding fully reacted (Fig. 6 c, d), while on granules available in the SAP matrix in the fuel rod centre, the interaction layer was significantly less (Fig. 6e) [4]. Thus, the promising character and technologic feasibility of applying SAP type alloys as matrices were demonstrated.

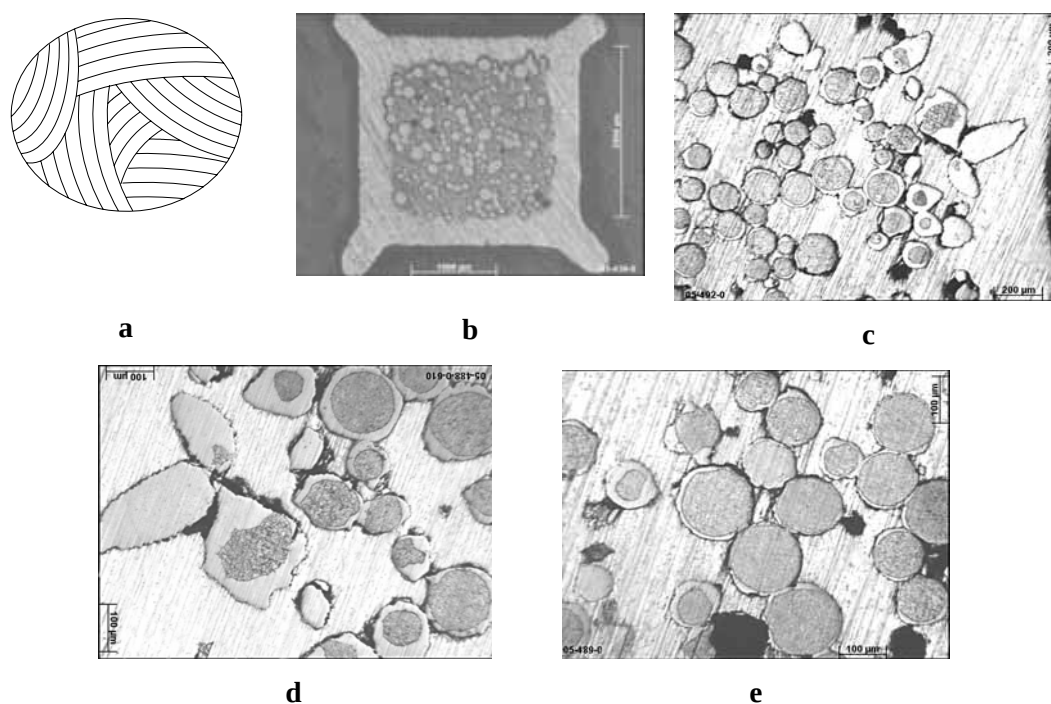


Fig. 6. Structure of SAP particles (a) and structures of U9Mo + SAP fuel compositions, b – as fabricated, c, d, e - as 4 h annealed at 630°C , d - fuel granules pressed into aluminium cladding, e - fuel granules in fuel element centre in SAP matrix [4]

4. Combination of methods

To reinforce the effect of reducing the fuel – matrix interaction combined methods might be applied, specifically, the alloying of matrix and fuel. A qualitatively novel effect may be received with the use of alloyed fuel produced by grinding in combination with a subsequent surface treatment. At the surface of ground alloyed uranium-molybdenum fuel a second phase will be mainly present. It has other physical and chemical properties and the thermal-chemical treatment of a surface will result in qualitatively new coats. Particularly in this case high quality oxide coats are feasible.

5. Conclusion

Potentialities of U-Mo fuel alloying to reduce interaction have been analyzed. A concept of U-Mo fuel alloying has been elaborated that consists in producing an alloy of the two phase structure, viz., basic

γ (U-Mo) and intermetallic phases. The intermetallic phase shall have the maximal density of uranium, a low molybdenum content, be well compatible with aluminium, highly irradiation resistant and precipitate along grain boundaries. The latter condition allows fuel particle production via grinding. After grinding the weakly interacting with Al intermetallic phase is primarily available at the fuel particle surfaces, forming some kind of a protection barrier. The first results of developments are presented.

The novel way of U-Mo fuel alloying by powder metallurgy, i. e. sintering powders of initial components followed by grinding resultant ingots is being developed. Besides producing non-equilibrium structures (inert to Al interaction) fabricated ingots are adequately brittle, therefore, readily ground and the surface of the milled powder easily oxidizes, forming protective barriers against interaction. The first studies in this direction are presented.

A novel trend in the alloying of a matrix to decrease the diffusionability of aluminium (alloying with high temperature resistant additives) is under consideration now. It consists in applying alloys of the SAP (sintered aluminium powder) type as a matrix. They contain 6-19% Al_2O_3 as intergranular thin interlayers. Aluminium insoluble oxide interlayers in a structure serve as diffusion barriers against diffusion of aluminium into U-Mo fuel. Preliminary technological studies and compatibility tests demonstrated the potentiality of this approach.

6. References

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POST-IRRADIATION EXAMINATION OF ALFENI CLADDED U_3Si_2 FUEL PLATES IRRADIATED UNDER SEVERE CONDITIONS

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ABSTRACT

As manager of the Réacteur Jules Horowitz (RJH) project, the CEA has initiated a study on the performance of full size U_3Si_2 fuel plates irradiated under relatively severe, but well defined conditions. Three U_3Si_2 fuel plates with a loading of $4.8 \text{ gU}_{\text{tot}}/\text{cm}^3 (\pm 20 \% {}^{235}\text{U})$, were irradiated in the BR2 reactor of Belgian Nuclear Research Centre (SCK•CEN). The fuel plates have an AlFeNi cladding instead of the (for CERCA) more common AG3-NET material. The objective of the irradiation was to reach an average fuel burnup of 55% ${}^{235}\text{U}$ while maintaining a maximum temperature of the outer cladding surface of 140°C . After the irradiation, the fuel plates were submitted to an extensive post-irradiation campaign at the hot cell laboratory of SCK•CEN. The PIE shows that the fuel plate performance was excellent as no detrimental defects have been found.

1 Introduction

SCK•CEN has been involved in the international development of low enriched Research Reactor fuels for several years now, particularly through the collaboration with the CEA in their fuel qualification and licensing efforts for the RJH. Over the past years and also at the current moment, post-irradiation examination (PIE) campaigns have been and are being carried out in the framework of irradiations of novel fuel plate designs that are performed in the BR2 reactor [1, 2].

The PIE campaign described in this paper involves 3 U_3Si_2 fuel plates with AlFeNi cladding, in which the attention is mostly devoted to the alternative cladding material. The objective of the irradiation was to reach an average fuel burnup of 55% ${}^{235}\text{U}$ with a maximum temperature of the outer cladding surface of 140°C while keeping it below 150°C at all times.

2 Fuel plates and irradiation history

Three U_3Si_2 fuel plates, manufactured by CERCA, were used to constitute the outer cylinder of a standard BR2 element. The U_3Si_2 fuel plates were given a fissile material density of $4.8 \text{ gU}_{\text{tot}}/\text{cm}^3$ and an uranium enrichment of 19.9 % ${}^{235}\text{U}$. The meat of the fuel plates consists of U_3Si_2 particles dispersed in a pure aluminum matrix. The cladding of the fuel plates is an Al alloy with 1 % Fe, 1 % Ni and 1 % Mg, commonly called AlFeNi.

The fuel element was incorporated in the BR2 reactor during three irradiation cycles (respectively 22.2, 22.5 and 26.1 operating days). The results of thermal hydraulic calculations [3] show that the mean heat flux in the axial hot plane for each of the plates was respectively 239, 245 and $245 \text{ W}/\text{cm}^2$ reaching maximum values of over $400 \text{ W}/\text{cm}^2$. At their EOL the fuel plates have reached an average

burnup of 54 % ^{235}U (1.3×10^{21} fissions/cm³) with a peak burnup of respectively 87, 86 and 79 % ^{235}U . From the temperature calculations, it was found that the external cladding surface temperature was kept within the range of 120 to 140°C. It should be noted that the thickness of the outer cladding oxide layer developed during irradiation has a major influence on the temperature calculations and therefore leads to an uncertainty on the temperature values of about 10%.

3 Post-irradiation campaign

3.1 Non-destructive testing

The fuel element was transferred to the hot cell near the BR2 reactor pool for non-destructive testing (NDT) and dismantling. After visual examination and dimensional inspection of the intact fuel element, the fuel plates were detached from the assembly. Visual examination of the individual fuel plates, shows a coloring, clearly distinguishing the meat zone from the cladding (Fig. 1). The darker color is the result of an oxide layer covering the meat section. It is noticed that the coloring is not homogenous, indicating that the oxide layer thickness varies. This inhomogeneity of the oxide layer thickness corresponds with the flux distribution and as such also the temperature distribution over the plate. Plate thickness profile measurements showed no severe deformation of any of the plates.

All three individual plates were transferred to the Laboratory for High and Medium Activity at the SCK•CEN site. The PIE was focused on the plate which had received a mean heat flux in the axial hot plane of 245 W/cm² (with a maximum of 300 w/cm2 at the beginning of cycle 2) and had reached a burnup of 86%.

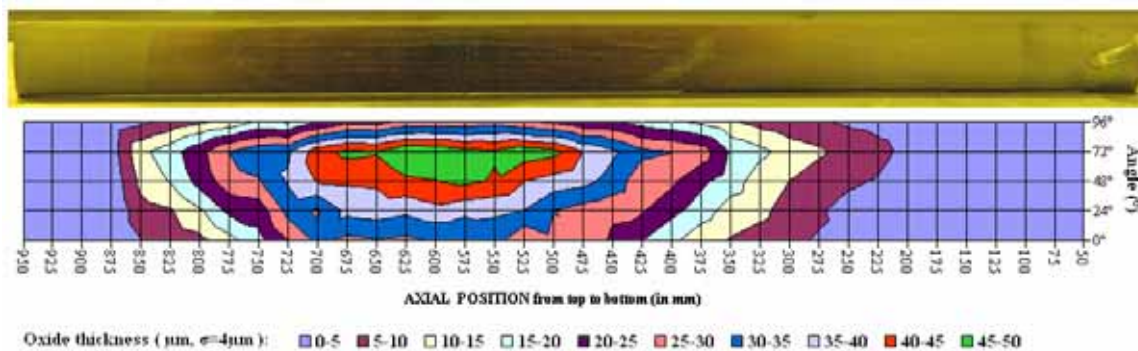


Fig. 1 The top image shows the plate itself, while the graph at the bottom shows the distribution of the oxide thickness.

The oxide layer thickness is measured using Eddy Current probe. The resulting graph shows a plateau at a maximum of 50 μm, indicating that at these positions the oxide layer has spalled off.

In Fig.1 the oxide layer thickness 2D distribution is shown next to an image of the fuel plate itself. During the NDT, it was observed that the oxide layer detached easily, maybe because of its evolution in the dry hot cell atmosphere. Some of those detached grains were collected and measured using X-Ray Diffraction (XRD), showing boehmite ($\text{AlO}(\text{OH})$) and bayerite ($\text{Al}(\text{OH})_3$) as major constituents.

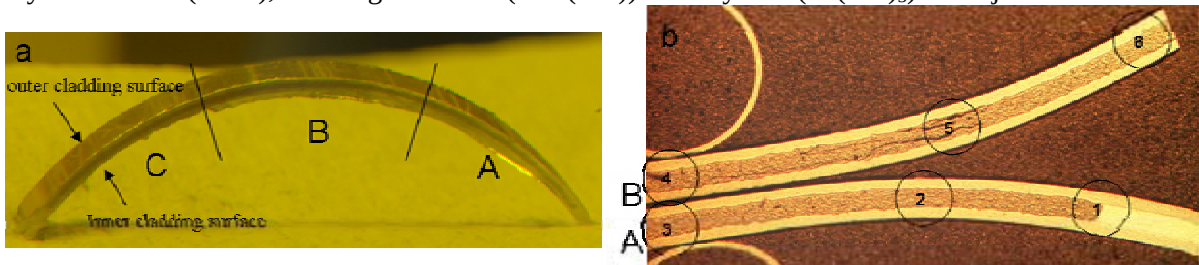


Fig. 2 A slice of the fuel plate (a) is cut in three pieces. Two samples (A and B) are embedded in the same mount (b).

A slice of the fuel plate (Fig.2a) was cut at a position that is characterized by a maximum count rate in the gross gamma spectrum (measured in the axial direction). This area also contains the thickest oxide layer on the outer surface of the cladding. Next, the slice was cut into three samples. To ensure that also the area corresponding to the maximum count rate in the perpendicular direction is going to be examined, samples A and B were imbedded in the same mount (Fig.2b).

The samples are embedded in an epoxy resin in such a way that the complete section of the fuel (meat and cladding) can be observed. The mount is polished with SiC paper of successively finer grain size, finishing on cloth with diamond paste of 3 μm and 1 μm . The samples have been examined in as-polished condition with optical microscopy (OM), scanning electron microscopy (SEM) and electronprobe microanalysis (EPMA), at several positions as indicated in Fig.2b.

3.2 Optical Microscopy

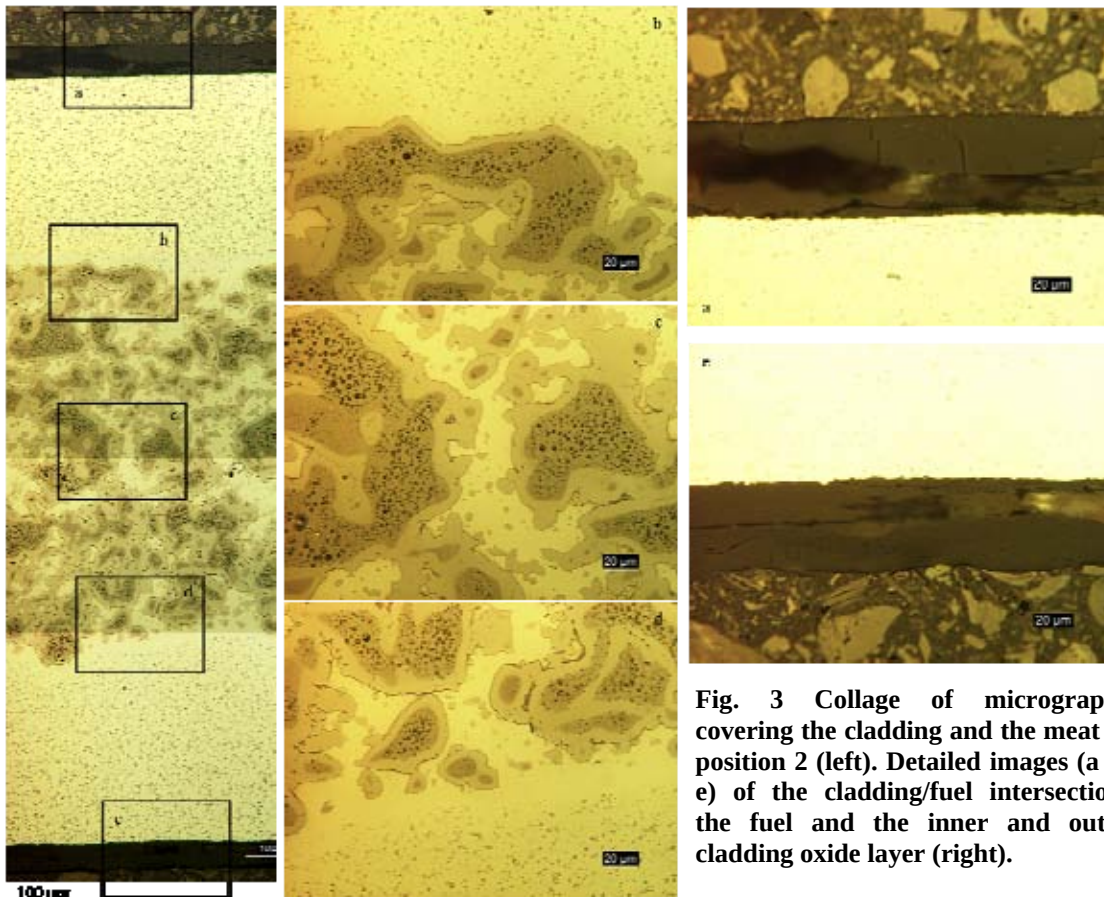


Fig. 3 Collage of micrographs covering the cladding and the meat at position 2 (left). Detailed images (a to e) of the cladding/fuel intersection, the fuel and the inner and outer cladding oxide layer (right).

The collage of micrographs in Fig. 3 (left) covers the complete plate cross section. A thick ($\pm 35 \mu\text{m}$) oxide layer at the inner and outer surface (Fig. 3a,e) of the plate can be observed. The oxide layer has cracked in both the directions perpendicular and parallel to the surface. This could explain the observation that the oxide layer detaches easily, outside of the reactor. In the cladding numerous, typical Fe-Ni rich precipitates can be seen. Fig.3b,d show the excellent bonding from the AlFeNi cladding to the pure Al fuel matrix. In the meat (Fig. 3b,c,d), the classical interaction layer between the U_3Si_2 particles and the Al matrix has formed during irradiation. Inside the U_3Si_2 particles, fission gas bubbles with different sizes can be observed.

Vickers microhardness measurements of the cladding, U_3Si_2 fuel particles, Al matrix and the Al-U-Si interaction layer were performed. For each of the measured items the values do not differ significantly as function of the position on the sample. The mean values found are 10HV72 for the AlFeNi cladding, 10HV399 for the Al-U-Si interaction layer, 10HV153 for the Al matrix and 100HV290 for the U_3Si_2 particles.

3.3 Scanning electron microscopy

The SE images (Fig.5a,e) clearly show the cracking of the oxide layer. Furthermore, it is seen that the AlFeNi precipitates of the cladding are also present in the oxide layer but only in the small layer close to the cladding (Fig.4). The images of the meat (Fig.5b,c,d) more clearly reveal

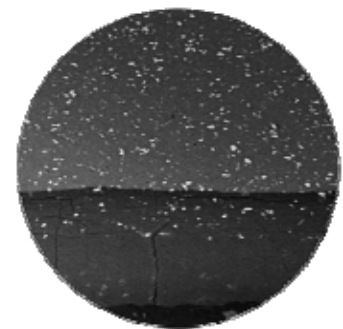


Fig. 4 BSE of interface oxide layer – AlFeNi cladding

fission gas related bubbles of varying diameters. The bubbles are mostly located inside the fuel particle, but a few bubbles can be observed at the interface between the fuel and the Al-U-Si interaction layer or inside the interaction layer. It should be noted that in some regions inside the fuel particles, no bubbles are visible.

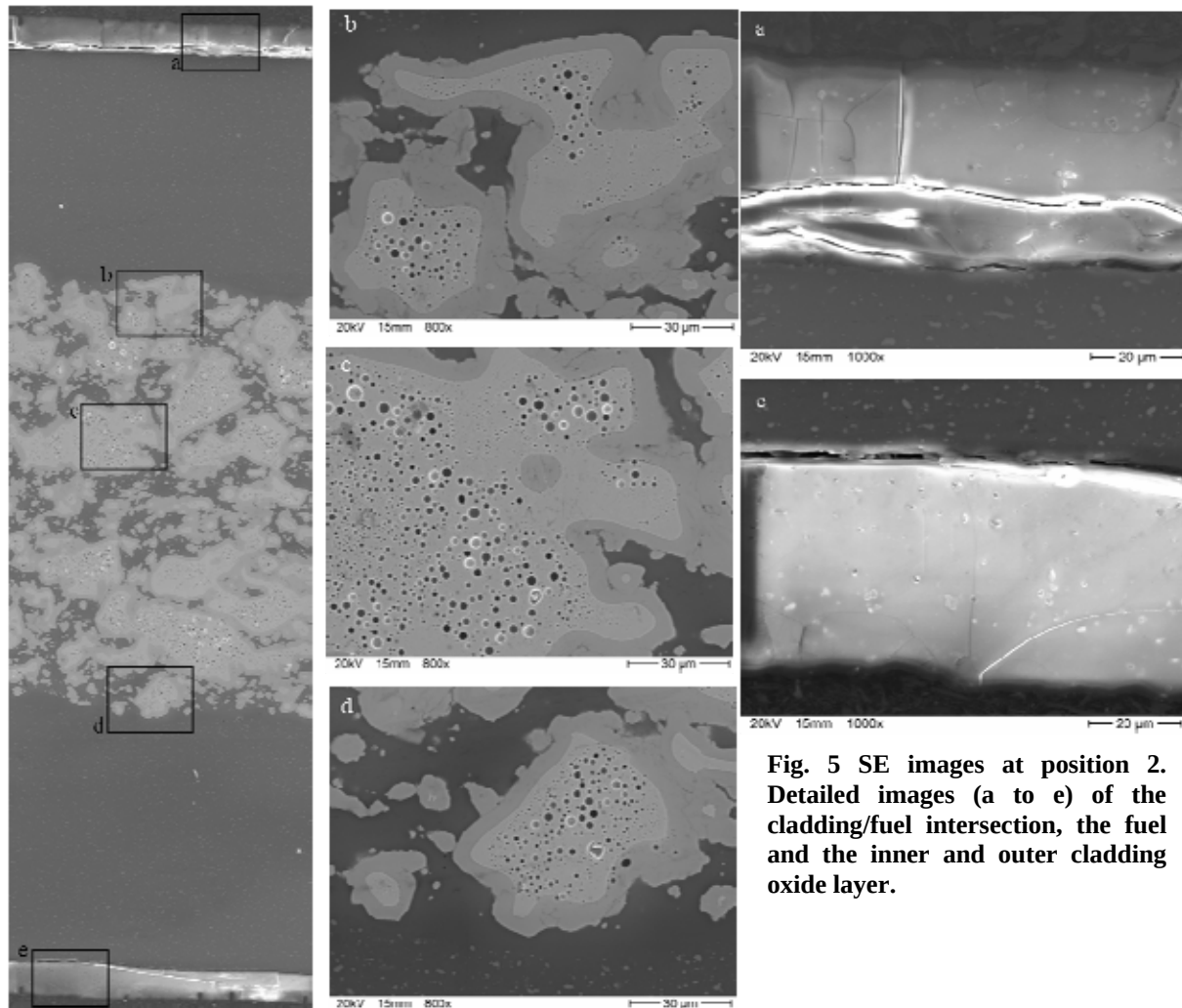


Fig. 5 SE images at position 2. Detailed images (a to e) of the cladding/fuel intersection, the fuel and the inner and outer cladding oxide layer.

The volumetric fraction of each class of bubble diameter was calculated as a function of the Feret diameter (Fig.6). This distribution is monomodal which would indicate that continuous bubble growth has occurred. However, detailed analysis of the SEM images show that the sizes of the bubbles depend on the composition of the material they are generated in. In the SE image in Fig.7a, zone 1 contains numerous small bubbles with a diameter of 100 - 300 nm, while in zone 2 larger bubbles with diameters up to 4 μm are observed. The backscattered electron image (Fig.7b) clearly shows that the darker zone 2 is of a denser structure than zone 1 and will differ in composition. The observed difference in composition of the fuel particles is confirmed qualitatively by EPMA measurements.

Due to the fabrication conditions, i.e. hyperstoichiometric Si charges (7.6 wt% Si), and some characterisation performed on fresh fuel comparable to the U_3Si_2 plates before irradiation, the most probable phases are U_3Si_2 , USi and possibly some USi_3 . Zone 1 would then be USi, while zone 2 the denser U_3Si_2 . Further quantitative examinations are under progress to confirm those expectations. By reference [4, 5, 6] the bubble sizes in zone 2 are somewhat larger than previously observed for U_3Si_2 for this burnup, which is probably related to the relatively severe irradiation conditions.

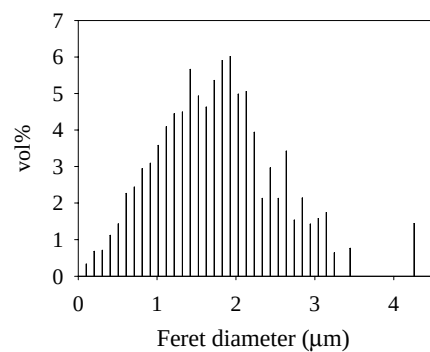


Fig. 6 Volumetric distribution of fission gas related bubbles.

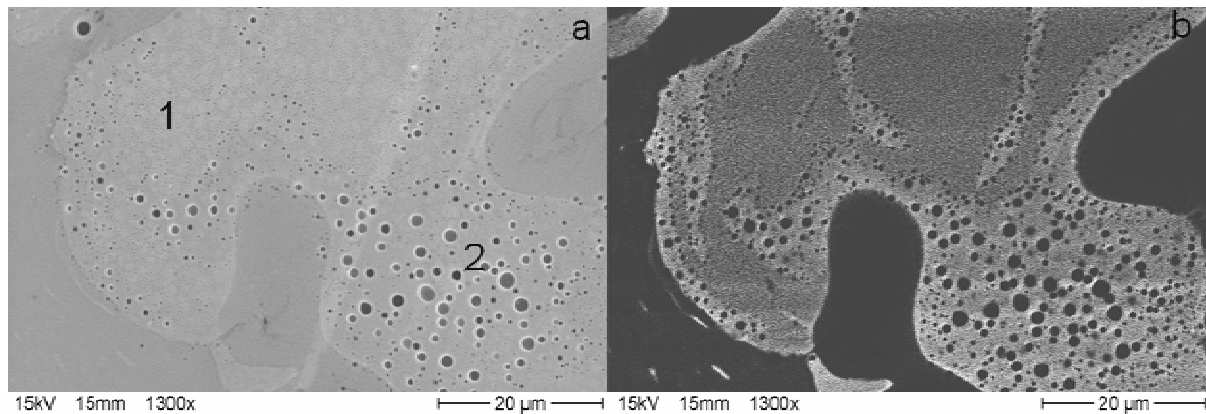


Fig. 7 Secondary electron (a) and backscattered electron (b) image of a fuel particle.

The thickness of the inner and outer oxide layer is measured (Fig.8a) on the SE images (recorded at the positions indicated in Fig.2b) and clearly shows the stagnation in layer thickness at $\pm 50 \mu\text{m}$ as observed in the NDT. Further image analysis (Fig.8b) show that the surface fraction of the fuel particles decreases from the plate side (pos 1) towards the middle of the fuel plate (pos 6). The even larger reduction of Al matrix in combination with the growth of the Al-U-Si layer gives a good indication that the interaction layer mainly grows at the expense of Al matrix. The thickness of the interaction layer (measured in the middle of the fuel section at each position) follows the increase in surface fraction.

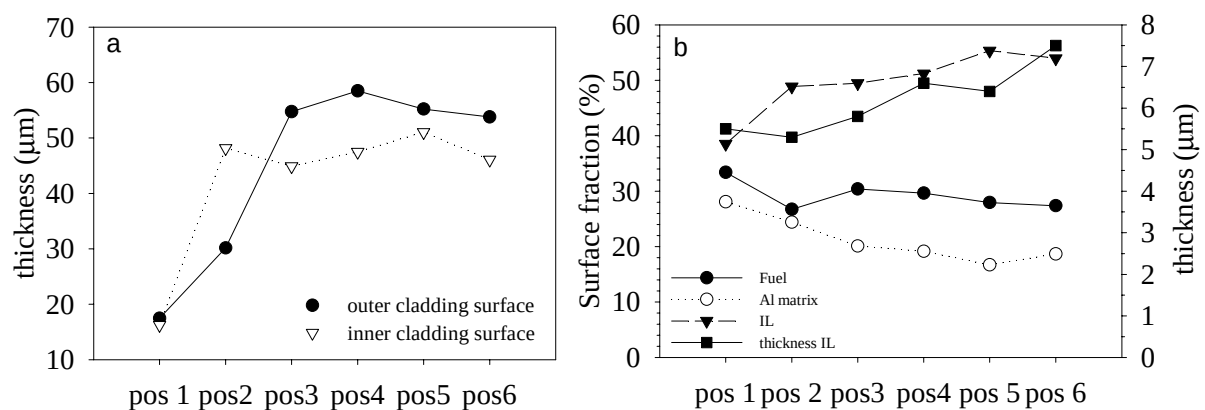


Fig. 8 (a) Thickness of the oxide layer on the outer and inner cladding surface, measured at several positions. (b) Surface fraction occupied by the fuel, matrix and inter action layer and the thickness of the interaction layer measured at several positions.

4 References

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PROGRAM of the TRIAL OF LEAD TEST ASSEMBLIES IN WWR-K REACTOR*

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ABSTRACT

Neutron-physical and thermohydraulic characteristics of the core with the low-enriched (19.7 %) eight- and five-tube FAs are presented. For trial of lead test assemblies (LTA) a program is developed with substantiation of trial representative ness over basic controlled parameters (the LTA power, the peak flux, the fuel element wall temperature). The determined characteristics of the reactor core in course of forthcoming trial show that the trial isn't an obstacle for other irradiation work. Выполнен The trial safety analysis is performed for cases of core arrangements, for potential reactivity accident or radiation accident (e.g., in case of LTA seal failure).

1. Introduction

In-reactor trial of a batch of lead test assemblies (LTA) is the last and the most important pre-fabrication stage. As a rule, such trials are carried out in loop reactors, where close-to-natural FA operational conditions can be created. However, this way is too expensive; so the trial in the pool-type reactor is evaluated through appropriate calculations. Main requirements imposed on in-reactor trial of LTAs are as follows: the number of LTAs is to be not less than three; the trial parameters (the LTA power, the peak thermal flux, burnup, etc.) must be close, to the highest extent, to those related to the FA operation conditions; control of specified parameters of the trial is to be provided.

The main problem of the future trial in the WWR-K core is a fact that the LTA power at operational mode is twice as much as that of the ordinary fuel assembly (FA). Due to this, in view of increasing LTA power, the WWR-K reactor core configuration and composition are assumed to be changed.

2. The low-enriched core neutron and thermo physical characteristics

As a results of comparing of the neutron-physical characteristics of the core versions differing in fuel composition and design of fuel elements (FE) and FAs [1-2], eight-tube FA-1 and five tube FA-2 with the $2.8\text{-g}\cdot\text{cm}^{-3}$ uranium density in $\text{UO}_2\text{-Al}$ fuel meat are recognized as the optimum for the WWR-K reactor. The thickness of the meat and the FE are 0.7 and 1.6 mm respectively. All FEs in the FA-1, at the except of the central one, are of hexagonal cross section. The FA face-to-face size is 66.3 mm, the core grid spacing is 68.3 mm. The FA-2 design (five hexagonal FEs) makes it possible to use existing executive components of the WWR-K CPS. Mass of uranium-235 in FA-1 and FA-2 are 252.6 and 201.9 g respectively.

The initial core with 16 FA-1, 10 FA-2 and waterside reflector is shown in figure 1. $K_{\text{eff}}=1.0778$. Neutron-physical characteristics are obtained with the computer code MCU-REA [3].

Efficiency of all groups of CPS control rods (CR), with its interference taken into account, for initial core is presented in Table 1. It is assumed that a couple of shim elements is fixed to one drive. Table demonstrates that safety is assured for the core with maximum excess reactivity, judging on efficiency of shim elements (KO) and automate rod (AP).

Via calculation, a succession has been determined for gradual transformation of the core with water

* Work was carried out with financial support from NTI.

reflector, formed by hollow displacers, to the core with beryllium one, formed by hexagonal cross-section beryllium blocks with face-to-face size 66.3 mm. Gradual replacement of displacers is produced, basing on efficiency of shim elements (KO), as excess reactivity reduces. Operational core comprises 16 FA-1, 10 FA-2 and 52 beryllium blocks. Complete replacement of water reflector by beryllium one covers 365 days of reactor operation. To this time in three FAs (averaged over FA) burn up level (50%) is reached.

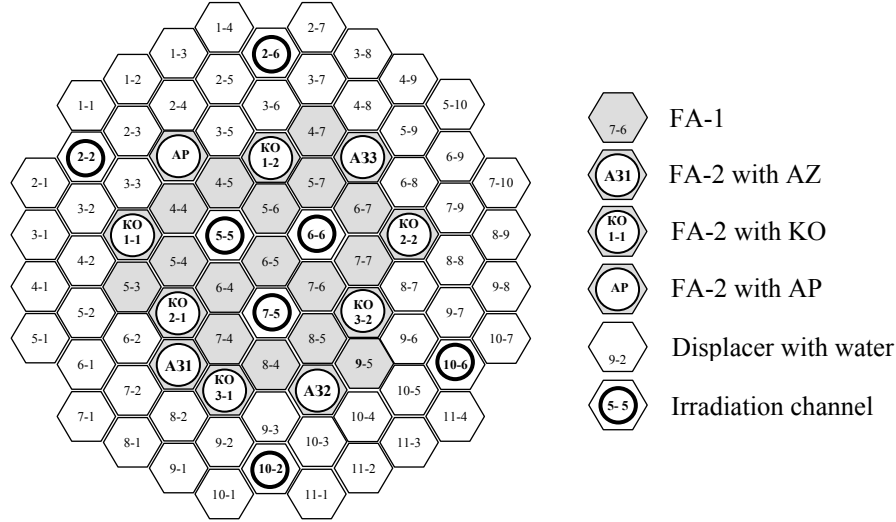


Fig. 1. Initial core

Control Rod (CR) Immersed to the Core	K_{eff}	Excess Reactivity, %($\Delta k/k$)	CR Worth, %($\Delta k/k$)
NO	1.0778(3)	7.21	-
All KO and AP	0.9830(2)	-1.73	9.00
Two KO1	1.0525(5)	4.99	2.20
Two KO2	1.0497(5)	4.73	2.48
Two KO3	1.0511(5)	4.86	2.35
All AZ	1.0476(3)	4.54	2.67
AZ1	-	-	0.94
AZ2	-	-	0.71
AZ3	-	-	0.81
AP	1.0694(5)	6.49	0.70

Tab 1: CPS CR efficiencies

Values of the power generated in all FAs of the core have been calculated. The hottest FA is in the core cell 6-5; its power in the reactor operation cycle is 360 kW.

Neutron flux density in irradiation channels has been calculated for initial and operational core options. Table 2 shows that the thermal neutron flux density in all irradiation channels is getting greater when water reflector is changed by beryllium one.

Core cell	5-5	6-6	7-5	2-2	10-2
"16+10"	$2.1 \cdot 10^{14} / 3.8 \cdot 10^{13}$	$2.1 \cdot 10^{14} / 3.7 \cdot 10^{13}$	$2.2 \cdot 10^{14} / 4.1 \cdot 10^{13}$	$5.4 \cdot 10^{13} / 4.0 \cdot 10^{12}$	$5.7 \cdot 10^{13} / 3.8 \cdot 10^{12}$
"16+10"+52Be	$2.2 \cdot 10^{14} / 3.8 \cdot 10^{13}$	$2.2 \cdot 10^{14} / 3.6 \cdot 10^{13}$	$2.2 \cdot 10^{14} / 3.7 \cdot 10^{13}$	$9.2 \cdot 10^{13} / 3.4 \cdot 10^{12}$	$1.0 \cdot 10^{14} / 4.7 \cdot 10^{12}$

Tab 2: Neutron flux density in the core irradiation channels ($E_n \leq 0.4$ eV / $E_n \geq 1.15$ MeV)

The performed thermohydraulic calculation has shown to thermotechnical reliability of the core [2].

3. The WWR-K reactor core arrangements to forthcoming trial

To achieve operational parameters of LTAs, in the center of the core beryllium displacer is installed with through a hole of a diameter 140.5 mm, forming a neutron trap where thermal neutron flux density $\approx 2.0 \cdot 10^{14} \text{ cm}^{-2} \cdot \text{s}^{-1}$. Three LTAs are fixed in the displacer by means of auxiliary grid and side

beryllium insertions which provide assured gap of ~2 mm for coolant flow. 20 VVR-C-type FAs are removed from the core periphery, in order to increase the specific generated power in LTA at the 6-MW power level. To compensate reactivity loss, 23 hexagonal beryllium blocks of the 65.3-mm face-to-face size are installed in the core periphery. Cartogram of the core and LTA layout is shown in Figure 2. Balance of reactivity has been calculated for the core arranged to trial. Results of calculation, including values of the CPS CR efficiency, are given in Table 3.

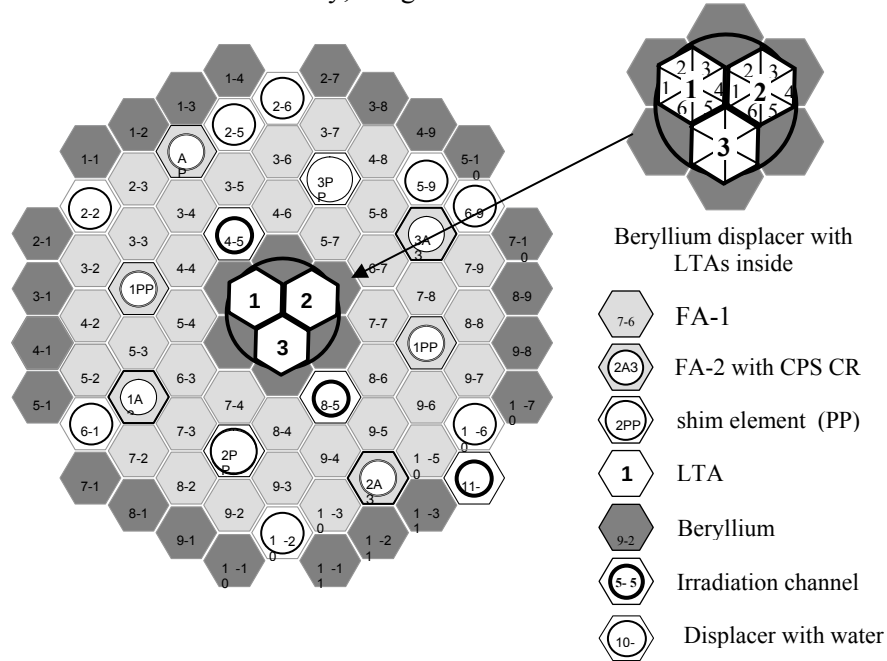


Fig. 3. The WWR-K reactor core cartogram of trial

Changes in the core	Core composition	Excess reactivity, $\%(\Delta k/k)$	Reactivity change, $\%(\Delta k/k)$
Initial core	76 FA	4.83	-
20 peripheral FAs are replaced by 23 beryllium block	56FA; 23 Be	4.97	+0.14
6 FAs from the core center are replaced by Be displacer to house LTAs	50FA; 23 Be ;Be dspl	-2.77	-7.74
CR Worth, $\%(\Delta k/k)$	1PP: 3.03%; 2PP: 2.16%; 3PP: 1.93%		
	$\Sigma PP+AP^*$: 8,8 ; ΣAZ^* : 3,3		
LTA1 is installed	50FA; 23Be; Be displ; one LTA	0.73	+3.5
LTA2 is installed	50FA; 23Be; Be displ; two LTA	5.1	+4.37
CR Worth, $\%(\Delta k/k)$	1PP: 3.23%; 2PP: 2.62%; 3PP: 1.94%		
	$\Sigma PP+AP^*$: 8.7% ; ΣAZ^* : 3.3%		
8 FAs are removed, LTA3 is installed	42FA; 23Be; Be displ; two LTA	1.97	-3.13
	42FA; 23Be; Be displ; three LTA	5.0	3.03
CR Worth, $\%(\Delta k/k)$	1PP: 2.23 %; 2PP: 1.89%; 3PP: 1.46%		
	$\Sigma PP+AP^*$: 6.6%; ΣAZ^* : 2.2%		

Tab 3: Components of reactivity balance

As the installed central beryllium displacer changes considerably the pattern of power distribution over the core, the generated power has been calculated in all FAs and LTAs. The hottest is the FA in cell 4–4: 192 kW. It has been taken into consideration that the core operated for 380 days prior insertion of LTAs (all reloads of FAs as well as loads and unloads of irradiation devices were taken into consideration).

4. The trial program and parameters

Program of trial covers the following: arrangement of the core and LTA loading; gradual approach to the LTA design values of power; control of trial parameters during transient and stationary periods; visual examination of FA exterior in case achieving the $\approx 40\%$ burnup; prolongation of irradiation - to achieve the 60-% burn up, followed by visual examination of exterior.

In course of irradiation, two LTAs will be turned azimuthally by 60° in course of an operation cycle (≈ 20 days): the third LTA will stay in fixed position - to take into account azimuthally irregularity to be occurred in FAs neighboring with beryllium reflector.

In course of trial, measuring system will record the following parameters: relative change in the neutron flux density in two LTAs; reactor thermal power; the core inlet and outlet coolant temperature; the coolant flow rate; the water activity in the primary circuit. The trial conditions are given in Table 4.

Parameter	Value
The coolant outlet pressure, MPa	0.135
The LTA power, kW	354
The coolant inlet temperature, $^\circ\text{C}$	$30 \div 45$
The coolant flow rate in LTA, m^3s^{-1}	15,1-17,4
The clad maximum temperature, $^\circ\text{C}$ (35 and 45°C)*	76; 84
The peak thermal flux density, kW/m^2	427
Trial time, day	555

* coolant inlet core temperature

Tab 4: Trial conditions

5. Characteristics of LTAs in trial

The calculated values of the power generated in three LTA comprise: 353 kW in LTA1, 342 kW in LTA2 and 354 kW in LTA3. Table 5 shows values of the specific power generated in the meats of all FEs in each of six faces of LTA1; the azimuthally irregularity factor comprises 1,23.

FE №	Sector 1	Sector 2	Sector 3	Sector 4	Sector 5	Sector 6
1	0.7585	0.8992	0.8303	0.5848	0.5893	0.7430
2	0.8209	0.9411	0.8733	0.6786	0.6891	0.8034
3	0.7739	0.8683	0.8144	0.6906	0.6791	0.7764
4	0.7625	0.8124	0.7954	0.7016	0.6881	0.7545
5	0.7315	0.7490	0.7350	0.6472	0.6707	0.7081
6	0.7295	0.7520	0.7360	0.6986	0.6861	0.7051
7	0.7680	0.7670	0.7665	0.7350	0.7171	0.7350
8	0.7720	0.7445	0.7410	0.7435	0.7071	0.7350

Tab 5: Generated power density in LTA 1
(shim elements (KO) and automate rod (AP) are immersed to the core mid line)

The LTA1 hottest face is divided into 11 sections over the height, and the generated power height irregularity factor is found to be 1.33. So, the peak specific power generated in the FE meat comprises 1.177 kW/cm^3 . The performed thermal calculation has shown that maximum temperatures of the FE wall and coolant at the core outlet don't exceed 90°C and 65°C respectively. Peak thermal flux reaches 427 kW/m^2 , the ONB temperature is 122°C , and ONB margin is 2.1.

In order to achieve average burned in LTA of $\sim 60\%$, about 555 effective days are required. The LTA power and burned versus time of reactor operation are shown in Figure 3. Also, for comparison, similar dependences are given for FA occupied cell 6-5 in cartogram of the core with low enriched fuel, presented in Figure 1.

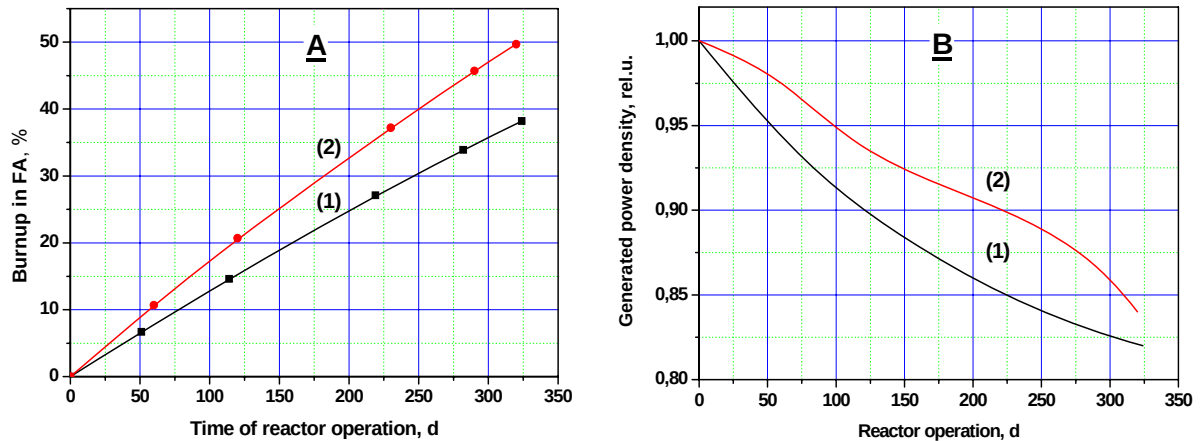


Fig. 3. Burned (A) and generated power density (B) versus reactor operation time (1) in LTA1; (2) in FA occupied cell 6-5 in Fig.1

6. Safety calculations

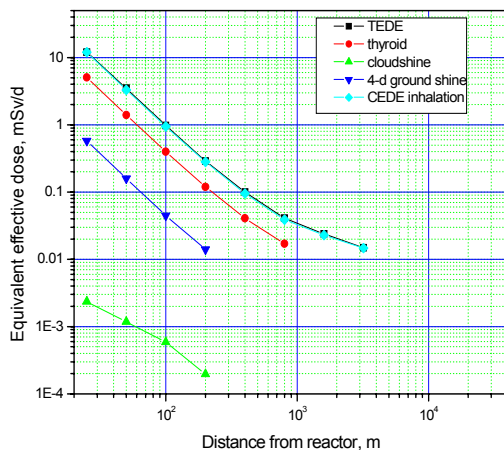


Fig.4. Dose versus distance

are neglected, the 1-minute release is of puff character. Fission products with half-life from 14 minutes to several years are taken into account. Figure shows that accident consequences fall only within reactor buffer area (1 km of the closest settlement).

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Session IV

Optimisation and Research Reactor Utilisation

VIBRATION MONITORING AND DIAGNOSIS AT THE IEAR1 NUCLEAR RESEARCH REACTOR

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ABSTRACT

Because IPEN's IEA-R1 reactor is old and it has been operating more and more to radioisotope production, an improvement program for its facilities was needed. Among others actions, a Continuous Vibration Monitoring System (CVMS) for the reactor rotating machines was installed. The present study objective was to establish a vibration monitoring strategy for the IEA-R1 reactor primary pumps, and to verify to what extent the strategy can be implemented by the CVMS. Three information sources were used to analyze the adequacy of some widely known vibration analysis techniques: historical data from the primary pumps monitoring, the experimental results from a mechanical defects simulation device and the specialized literature. The results showed that, although the CVMS seems to be enough for the fault detection of the selected defects, it is advisable to use an additional analysis for an earlier detection and a more precise diagnosis. A monitoring strategy was also established.

1. Introduction

The IPEN IEA-R1 Reactor reached its first criticality in 1957 and it is one of the first research reactors to operate in the world. At the beginning its main goal was to develop researches, in addition of personnel training to be certified as reactor operators. However, since 1996 the IPEN has decided to use the reactor more and more for industrial applications, specially for radioisotopes production. In this way, it was necessary to license the reactor to operate at its maximum design power, 5MW, and to begin a program for its modernization and ageing management for its safety operation, increasing production availability and for extending its useful life. It was needed, among others actions, an improvement of the installation maintenance activities. It was in this context that it was installed a Continuous Vibration Monitoring System (CVMS) for the reactor: first due to a safety necessity, in order to reduce the LOCA probability as it is explained in the installation Safety Analysis Report, and then to increase the reactor availability by the early detection of mechanical problems.

The objective of this work was to create a diagnostic and vibration monitoring strategy for the hydraulic pumps of the IPEN IEA-R1 Reactor primary cooling loop. This strategy includes the transducers position and orientation confirmation, the measurements periodicity establishment, and the determination of the defects to be observed. Also, the vibration signal analysis techniques to be used as well as the alarm limits should be defined. We determined the applicability, for our specific case, of those tools and techniques that are already consecrated by the industry and that are the most adaptable to the conditions and operation needs of our installation, specially of those that could be implemented by the CVMS. The results of this work should make possible an appropriated CVMS configuration, through the appropriate configuration for analysis and the correspondent alarm limits. In a complementary setup, the utilization of a Parallel Vibration Analysis and Acquisition System (PVAAS) was also implemented, in order to add techniques that aren't available in the CVMS and that could improve the information available to the primary pumps monitoring.

2. Methodology

When vibration monitoring is concerned, we have available today a vast informative literature: books (1) and manuals with a wide theoretical and practical basis, some standards (2) that represent the

accumulated industrial experience about the relationship between the vibration severity and the monitored machine conditions, as well as a reasonable quantity of papers (3,4) that prove and establish the use of all known tools in this field. However, it is also known that the application of this knowledge in actual situations is not a trivial activity, of simple transposition of the theory to the case in question. The way and the degree of efficiency in the use of different known vibration monitoring techniques depends a lot on the real situation: the correct application of the techniques will depend a lot on the historical experience with the vibration behavior of the machine to be monitored.

The essential part of this work focused on the historical verification of the functionality of some vibration analysis techniques that seemed to be more appropriated to the real situation of the plant and of the available monitoring equipments. As the CVMS does not give access to the raw data and its analysis have a closed format, it was necessary to utilize the PVAAS, that was used also to try others complementary tools. Besides that and since the pumps are part of the reactor protection system and due to the impossibility to test mechanical defects on them, it was decided to utilize a Mechanical Defect Simulation Machine (MDSM) to perform experiments that allow for a wider and more detailed analysis of the probable defects of the primary pumps. The MDSM also served to confirm the results with the primary pumps and to verify the Matlab programmed algorithms used in PVAAS for the analysis.

2.1 The primary pumps

The primary pump is a centrifugal pump of 100Hp and a rotation speed of 1780rpm. The Fig. 1 below shows its scheme and the CVMS accelerometers positions: the transducer direction is horizontal and radial at A1, vertical at A3, A4 and A5 and axial at A2 and A6.

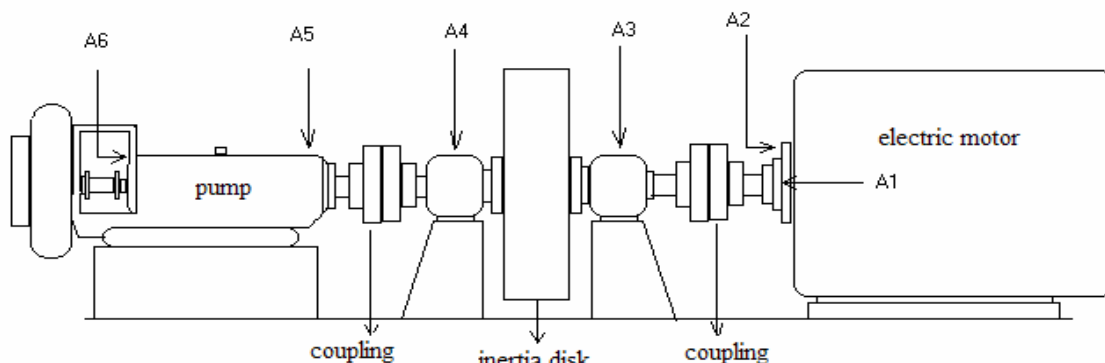


Figure 1. The IEA-R1 primary loop pump

2.2 The Continuous Vibration Monitoring System (CVMS)

The CVMS (5) is composed by a vibration monitor, which is a modular system that does all the handling of the signals from the accelerometers and the events, and by a UNIX workstation that contains the monitoring softwares and the data base. The workstation is also used to configure and control all the monitoring system, besides doing the user interface. All the equipment is mounted in one of the Reactor control room racks.

The system softwares allow the monitoring of the instrumented components, the establishment of up to two alarm levels for each measurement, the data processing and the results storage. The vibration measurements that are allowed by the current software are the peak, the RMS, the crest factor and the signal spectrum.

2.3 The Parallel Vibration Analysis and Acquisition System (PVAAS)

The primary pumps vibration signals used by the PVAAS were acquired through the CVMS fixed accelerometers. They are sent independently to a conditioner, where they are amplified and filtered, following to an acquisition board, where they are digitalized, and finally they are stored in a lap-top by

mean of the LabView acquisition software. All signal analysis was carried out by algorithms written in the Matlab platform.

2.4 The Mechanical Defect Simulation Machine (MDSM)

A MDSM (6) is a tool for studying the vibration signatures of the rotating machines mechanical defects and has a great assembly versatility that depends on the kind of the defect that is simulated. We can see, in the Fig.2, the experimental bench with its signal acquisition equipment.

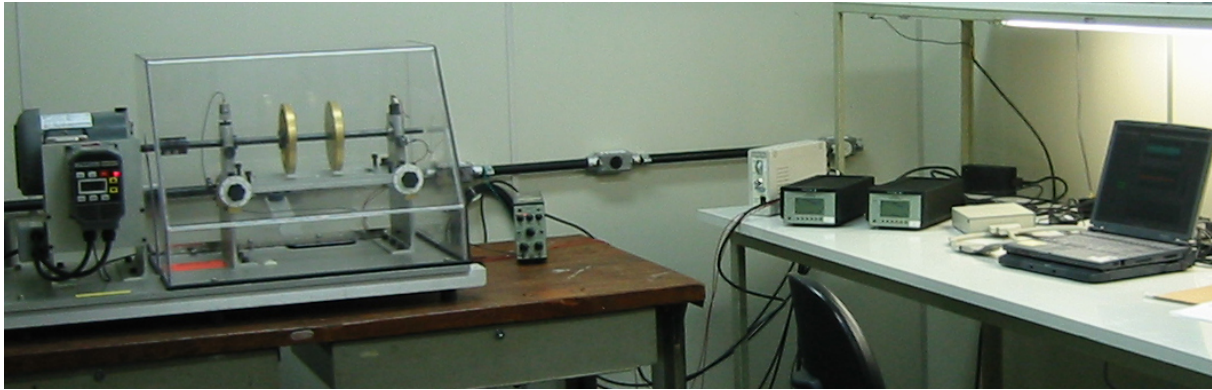


Figure 2. MDSM experimental set up.

2.5 The techniques efficiency and their application

The performance evaluation of the vibration analysis techniques was done by using the R index: $R = V_d / V_b$, where V_d is the measured value of a vibration parameter in a known defect condition and V_b is the reference value of this parameter in a condition without defect. It is considered as a vibration parameter any measure of a value that is calculated from a vibration signal by applying a specific analysis technique. The value of V_b in the primary pumps was taken right after a reform or repair by the average of a historical sequence of the parameter measurements while the signal is low and stable. In the MDSM, V_b was taken by the average of eight values, where each one corresponds to a different assembly without defect.

It was considered as an *indicator* (of defect) the parameter for which $R \geq 1,6$. The 1,6 factor was taken based on the ISO10816 standard (2), that defines that a change of 60% in the vibration severity of a machine means that there is a significant change in its working condition.

It was chosen the following group of parameters to be studied: root mean squared of the signal for low frequencies, from 10Hz to 1kHz, in velocity (RMS_v); RMS of the signal and peak of the spectrum for high frequencies, from 1kHz to 10kHz, in acceleration (RMS_a e $Peak_a$); the amplitude of the first four shaft speed harmonics and inter-harmonics in the velocity spectrum (1^o_h , 2^o_h , 3^o_h , 4^o_h and 1^o_i , 2^o_i , 3^o_i , 4^o_i); the amplitude of the first four harmonics of the rolling bearings defects frequencies – inner race defect (fi), outer race defect (fo), cage defect (fc) and rolling balls defect (fb) for the velocity spectrum as well as for the envelope spectrum, for which the harmonic order is indicated inside parentheses and the spectrum type is indicated by an index ('e' for envelope and 'v' for velocity); for example, $fo_e(2)$ means the second harmonic amplitude of the outer race defect frequency in the envelope spectrum. Besides this, it were also studied the amplitude of the first four harmonics of the blade pass frequency (fp) in the A5 and A6 positions on the primary pumps. In the beginning, others parameters were also considered, like the crest factor, the cepstrum and the skewness. However, they did not have a better performance than the others tried parameters, despite of their higher complexity. Therefore, they were not included in the results.

The chosen defects were mechanical imbalance, coupling misalignment, mechanical looseness and the rolling bearing faults (localized defects and lubrication problems). All these defects were simulated and studied in the MDSM, in different configurations and degrees of severity.

3. The results

All the vibration signals of the primary pumps composing the data base have been collected during 30 months, once a month, with a sample frequency of 30kHz, low pass filter of 10kHz, high pass filter of 1Hz and a duration of 10s each. It has been done a historical comparison between all the events records and the results of the vibration analysis of all the collected data. The results of all historical comparison are summarized in the following Table1.

Table 1. Indicators on primary pumps.

mechanical imbalance	coupling misalignment	mechanical looseness	bearing outer race defect	bearing inner race defect	bearing balls defect	bearing lubrication problem
1°h, RMS _v	2°h, 4°h RMS _v	1°h, 2°i, 3°i 4°i, fp(2) RMS _v	fo _v (1,2,3) RMS _v	fi _v (1,2,3,4) fi _e (1,2,3,4), RMS _a Peak _a , RMS _v	fb _e (1,2,3,4)	1°h, 2°h, 3°h, 4°h fo _v (1,2,3), fo _e (2,3,4), RMS _v

The Table 2 presents the main MDSM results, where the “common indicators” represent the parameters that were indicators in all tested cases for the defect in question, while the “best directions” and the “best parameters” were those with the higher values of R.

Table 2. Main results with MDSM.

defect	common indicators	best direction	best parameters
mechanical imbalance	1°h	H	1°h
coupling misalignment			2°h
mechanical looseness		H	2°h, 3°h, 4°h
bearing outer race	fo _e (1,2,4), Peak _a	H	fo _e (1,2,4)
bearing inner race	fi _e (1,2,3,4), fi _v (1,2,4), RMS _a , Peak _a , RMS _v	H	fi _e (1,2,3,4)
bearing balls	fb _e (1,2,3,4), fb _v (3,4), RMS _a , Peak _a	V	fb _e (1,2,3,4)
bearing cage	RMS _a , Peak _a	H	Peak _a
bearing lubrication	fo _v (1)	V	fo _v (1)

The envelope spectrum was more efficient than the velocity spectrum in all tested rolling bearings defects, except for the cage defects and the lubrication problems.

3.1 Monitoring strategy and diagnosis

The defects considered for monitoring of the primary pumps – mechanical imbalance, coupling misalignment, mechanical looseness and rolling bearings faults – have been chosen due to the mechanical characteristics of the pumps, safety and operational requirements, and the present industrial experience. The safety requirements that implicated in the acquisition of CVMS may be more seriously jeopardized by mechanical looseness and rolling bearings defects.

The positions of the CVMS accelerometers on the primary pumps were confirmed both by the conformity with the literature, that indicates that the accelerometers should be positioned as closer as possible to the bearings, and by the MDSM results, that indicate that the radial directions are the best for the detection of the considered defects. The two axial accelerometers on the primary pump are also in conformity with the literature, that states that the imbalance of the hanging rotors may be more easily detectable in this direction, as well as in conformity with the pumps results, that presented an indicator in the axial direction for looseness in the bearing box. The axial positions from two of the CVMS accelerometers are also in agreement with the MDSM results, which pointed that a conjugated imbalance may be better detected in the axial direction.

The results with the primary pumps point out that the continuous monitoring by CVMS, in addition to a monthly measurement of their vibration signals for a more detailed analysis using the PVAAS, has been enough to assure their safe operation.

The vibration parameters for the primary pumps monitoring, presented on following Table 3, were defined based on those which were indicators on the pumps themselves and, as a complement, from those that were “indicators” or showed some correlation with the defect ($R>1$) in the MDSM experiments.

Table 3. Vibration parameters for the primary pumps monitoring.

mechanical imbalance	coupling misalignment	mechanical looseness	rolling bearing localized defects (rbld)	lubrication problems in rolling bearings
1 ^o h, 2 ^o h, RMSv	2 ^o h, 4 ^o h, RMSv	1 ^o h, 2 ^o h, 3 ^o h, 4 ^o h, 2 ^o i, 3 ^o i, 4 ^o i, RMSv, fp(1,2,3,4)	the first four harmonics of fo _v , fi _v , fb _v , fc _v , fo _e , fi _e , fb _e , fc _e RMS _a , Peak _a , RMSv	all the parameters of rbld plus 1 ^o h, 2 ^o h, 3 ^o h, 4 ^o h

The CVMS alarm limits for the primary pumps were established based on the ISO10816 standard, that evaluates the working condition of the rotating machines by their vibration signals severity. The R=2 and R=3,2 values were used as the relation between the two alarm levels and the reference value for each considered vibration measurement, because the standard uses these same values to establish the relation between the alert and danger state level and the normality state level of the equipment, for the type of the machine being monitored.

4) Conclusions

The obtained results showed that CVMS has been able to detect all mechanical defects of the primary pumps that could jeopardize the reactor safety. The bibliographical research and the MDSM results indicate that the use of some additional analysis to the ones done by CVMS, such as the envelope analysis, increases the safety and speed on the detection of mechanical defects on those equipments. However, the results also reveal that the detection of mechanical defects by vibration monitoring is a complex activity and many parameters must be observed simultaneously to make possible their discrimination.

The experience provided by this work points to the high value of the knowledge that can be obtained through a machine such as MDSM, both to elaborate a monitoring strategy and to train people that want to dedicate themselves to the activity of vibration monitoring. The possibility to make tests of defects in a concrete rotating machine adds information and even a feeling that cannot be acquired in any kind of literature. Another relevant observation refers to the importance of having direct access to the raw signal and to the capability of doing its analysis in a more flexible way than that allowed by most of monitoring systems available in the market.

Three years have passed since the conclusion of this work and the implementation of CVMS new configuration. And after that, the vibration monitoring of the primary pumps has been done with more efficiency, safety and confidence by the reactor operation staff.

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NEW SILICON IRRADIATION RIG DESIGN FOR OPAL REACTOR

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ABSTRACT

Described is an overview of NTD silicon processing facilities in ANSTO's OPAL reactor. A suite of six (6) high capacity Neutron Transmutation Doping (NTD) silicon rigs have been developed and installed. Optimum quality of the arrays is achieved through rotation of the silicon in the rigs to ensure even neutron fluence. The innovative design features a simple combination of water bearing and drive/rotation utilising reactor pool water. A hydrostatic water bearing ensures that there is no physical contact between the single moving component and static rig housing during operation. This design overcomes many of the existing reliability and maintenance issues involved with typical mechanical drive systems. The resulting layout leaves the pool area clear of obstructions which might obscure vision and hinder target handling for operators. Ingot handling systems are also provided to ensure the safe and efficient transfer of silicon between poolside and irradiation positions.

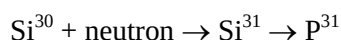
1 Introduction

OPAL achieved full power of 20MW in November 2006 and is currently being commissioned. Its official opening is scheduled for April 2007. OPAL replaces HIFAR which was finally shutdown for decommissioning on 30th January 2007 after 49 years of successful operations. HIFAR, having produced high quality, high conductivity silicon for the computer industry for many years, has allowed a unique opportunity to redevelop new silicon neutron transmutation doping (NTD) facilities for OPAL. With high volume and high quality NTD facilities set as a priority from the point of conceptual design, the new facilities could be truly integrated and optimised for performance whilst meeting all design criteria and user needs.

This paper briefly describes the unique design features of the silicon irradiation rigs and the associated monitoring and handling equipment necessary for a production process.

2 Silicon Neutron Transmutation Doping (NTD)

The silicon NTD process is used to produce a high quality semiconductor, N-type silicon, with precisely defined resistivity of less than 5% variation, thereby increasing the efficiency of the silicon in conducting electricity, an essential characteristic for the electronics industry. Depending on the fluence of neutrons absorbed, the transmutation process converts a proportion of silicon atoms to phosphorus atoms in the silicon single crystal:



Six NTD irradiation positions are situated vertically in the heavy water reflector vessel, which surrounds the reactor core, as shown in Figure 1. Each irradiation position has the capacity for a 600mm high array of mono-crystalline silicon ingots. The nominal diameters of ingots that can be irradiated are 4" (100mm), 5" (127mm), 6" (152mm) and 8" (203mm) depending on the irradiation positions and canisters used.

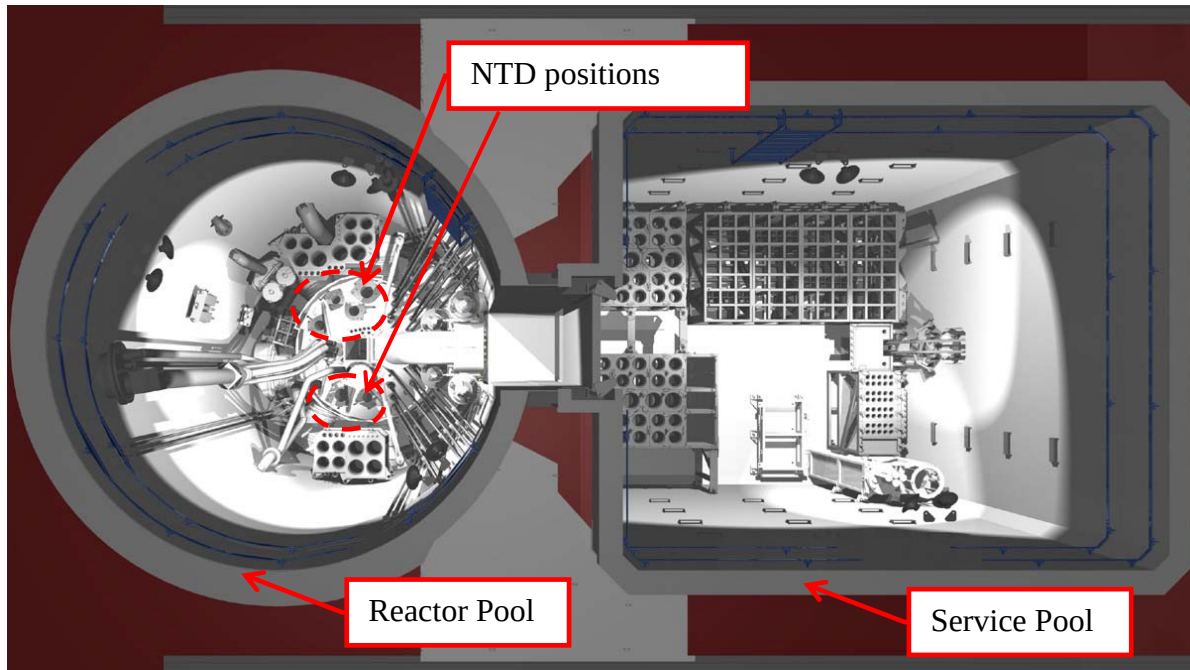


Figure 1. OPAL Reactor Pool Layout – Plan View

Semiconductor quality is ensured by a high thermal to fast neutron flux ratio and high axial and radial neutron flux uniformity over the silicon target. Radial uniformity of irradiation is assured by continuous rotation of the ingot in the rig and also by the placement of the NTD facility in a region with a low gradient of thermal neutron flux. Axial uniformity is achieved by the use of a flux-flattening device consisting of concentric bands of aluminium and stainless steel. By displacing the light water surrounding the target array, minimum perturbation of flux is ensured. Neutron flux is continuously monitored by self powered neutron detectors (SPND) embedded in the flux flattening sleeve. Induction type displacement transducers are used to monitor and verify constant rotation of the silicon. It can also be visually monitored via a camera from the reactor control room. Data from both sensors are transmitted to a central control and monitoring system by which irradiation parameters are logged.[1]

3 Silicon Rig Design Features

The primary aim of the irradiation rig is to ensure an even flux distribution over the entire silicon array. The innovative rig design features a combination of water bearing and rotation drive, as shown in Figures 2 and 3. Using reactor pool water as the working fluid, the hydrostatic water bearing system ensures no physical contact between the moving parts during operation thereby minimising or eliminating mechanical wear and tear. Bypass flow from the water bearing and a tangential impulse jet combine to provide the rotation force on the hybrid hydraulic turbine. This ‘non-contact’ configuration overcomes reliability and maintenance issues involved with typical mechanical drive systems that utilise electric motors, overhead drive shafts, couplings, gearboxes and mechanical bearings within the reactor pool. Such mechanisms not only clutter the pool space but also inherently suffer from wear, which is exacerbated by the inability to adequately lubricate contacting components.

The new drive system requires only hydraulic piping conveniently routed to the rigs, to which the instrumentation conduits are attached. The resulting layout leaves the pool area clear of obstructions which might obscure vision and hinder target handling for operators. Pump banks and control valves are located remotely in a dedicated plant room allowing easy access and online maintenance.

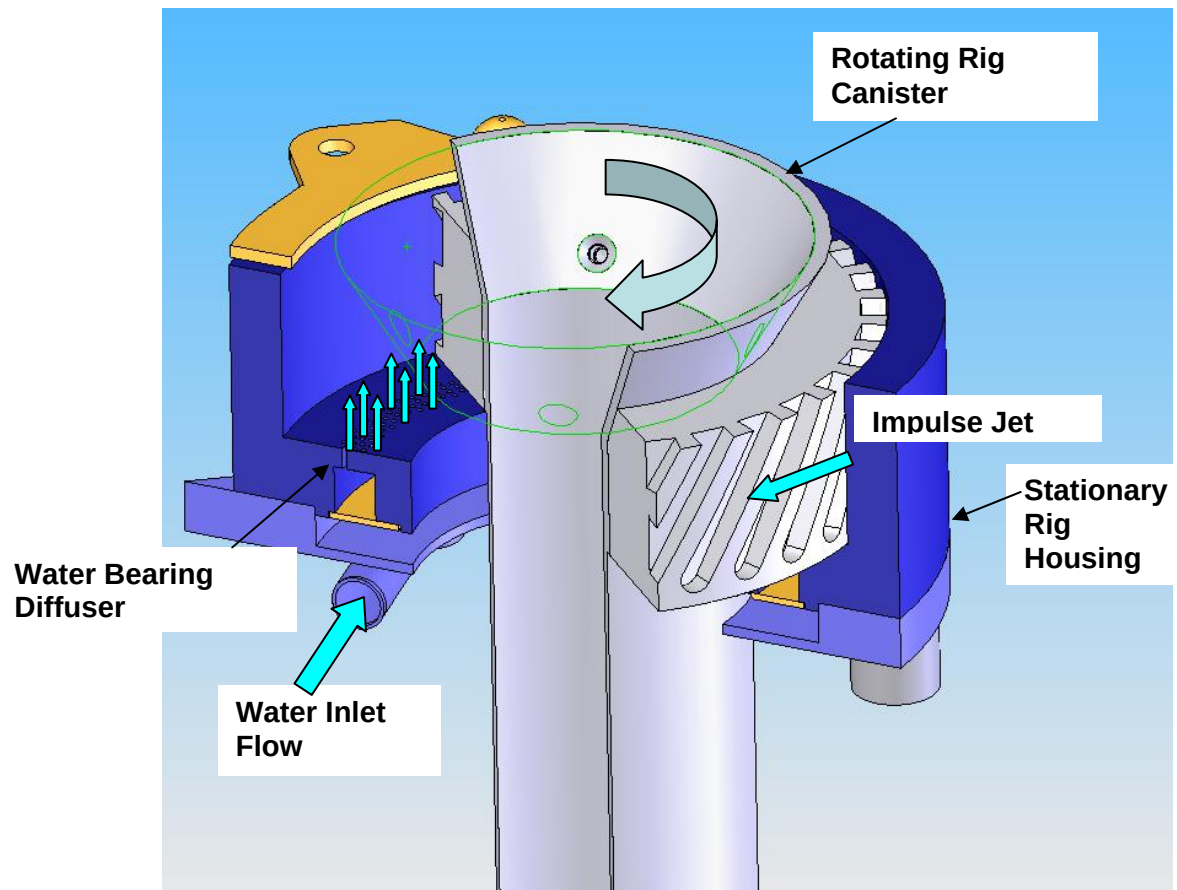


Figure 2. OPAL Silicon NTD Rig Concept - Water drive design
NB, this silicon rig design is the subject of an application for design registration in Australia and potentially other countries.



Figure 3. OPAL Silicon NTD Rig

An important feature incorporated in the design is the ability to perform vertical axis assembly. This ensures the rig can be removed for maintenance by a technician stationed on the pool top operations bridge over 15 metres above. Pool top manipulations using standard tools have been tested and proven. This enables the main consumable ‘in pile’ items, namely the SPND and rotation sensor, to be replaced remotely from the pool top.

‘End of life’ (40 years) disposal of material has been considered in the design. Construction materials, such as aluminium (1100) and titanium, that produce minimum half-life radioisotopes on irradiation were utilised where possible. For this reason the use of stainless steel in irradiated components was minimised. For components requiring mechanical strength or impact resistance aluminium (6061) was selected and a hard anodised surface treatment applied. This was particularly important for the mating surfaces of the turbine and housing, which come into contact if hydraulic inadvertently stopped. Such selection of materials and low wear characteristics of the drive unit is anticipated to significantly reduce the quantity of activated waste generated over the service life of the system.

The new design has been prototyped and bench-tested for 12 months to verify and validate the design prior to final production and installation. Commissioning tests confirming satisfactory operation of the hydraulic system were completed in late 2006 allowing irradiation trials to commence in 2007.

4 Ingot Handling System

A safe and efficient means of transferring large numbers of silicon arrays was developed. Open canisters of several sizes are used to carry and protect ingots within the pool. Despite the use of low alloy 1100 grade aluminium canisters, they will become activated after irradiation and so they must remain shielded by the pool water. For this reason silicon ingots are transferred between the activated canisters to the poolside preparation table by a dedicated loading machine. Human factors considered important to operators for this ‘Loading Station’ were; radiation dose, manual handling and ergonomics. Other critical design parameters were; prevention of damage to ingots, throughput capacity, reliability and versatility to handle all required ingot sizes. For simplicity and reliability manual pneumatic control and mechanical interlock devices were utilised in preference to electronic control of this loading station.

A dedicated lifting tool, attached to a hoist, is used to move canisters within the reactor and service pools. With the largest loaded silicon canister exceeding 80 kg the damage caused by mishandling could be significant. The drop of such a load in the reactor pool is considered a ‘design basis’ event, with physical protection added to guard safety critical items, though this is certainly an event to be avoided. Interlocks in this control also guard against inadvertent release of load and the maintenance of a minimum depth of shielding water covering activated canisters. The release mechanism utilises the pool water once again as a working fluid ensuring no air is introduced to the region of high radiation flux surrounding the reactor core. Graphite piston seals were developed to ensure maximum radiation resistance and extended service life over polymer seals. Extensive testing was performed on this lifting tool to ensure high levels of safety, reliability and maintainability. Included in the test program was activation of the release mechanism for 10,000 cycles, load tests and simulated tests of adverse operation. It was through these tests that failure modes identified in design reviews could be eliminated or reduced to an acceptable probability.

5 Conclusion

The new OPAL silicon rigs have been designed to optimise safety, reliability and capacity, and successfully installed and tested. The new rigs will help ANSTO increase productivity and continue to produce high quality semiconductor silicon for the world NTD market.

6 Acknowledgements

Development of the OPAL silicon NTD facilities has been undertaken by ANSTO Technical Services & Facilities Management (<http://home.ansto.gov.au/ansto/eng.html>). Additional support was provided from other ANSTO divisions in close collaboration with INVAP. This paper is the work of all those involved in the project. Special credit goes to Michael Deura who led the project through the conceptual design and Alec Kimber who completed the successful commissioning.

7 References

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MEASUREMENT OF VOID FRACTION IN HYDROGEN MODERATOR USED FOR MODERATOR CELL OF HANARO COLD NEUTRON SOURCE

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ABSTRACT

The void fraction in the hydrogen moderator used for the moderator cell of the HANARO cold neutron source was measured by using a gamma densitometer technique. A mock-up of the HANARO CNS facility with an electrical heating system as the heat source instead of radiations was constructed. The attenuation of the 59.5 keV gamma-rays from Am-241 through the hydrogen medium was measured by using an HPGe detector. The void fractions were measured for stable thermo-siphon loops with several heat loads applied to the moderator cell. The geometrical distribution of the void fraction was determined. The void fraction measured at a heat load of 720 W had values of 8~41% depending on the height from the bottom of the moderator cell. The void fraction determined at the expected nuclear heating power for this moderator cell was about 20%. These measurements will be very useful for the design and operation of the HANARO cold neutron source.

1. Introduction

The design and installation of a cold neutron source (CNS) facility for HANARO, a 30 MW research reactor, is in progress. The in-pool assembly consists of a two phase thermo-siphon loop and a vacuum chamber. Liquid hydrogen has been selected as the moderator. In order to validate the assembly design concept, a thermo-siphon mock-up test is being conducted. A mock-up of the HANARO CNS facility with an electrical heating system as the heat source instead of radiations has been constructed. The void fraction in the hydrogen moderator is one of the important parameters for the operation of the moderator cell in a cold neutron facility since it affects the moderation capability and the stability of a cold neutron source [1]. Therefore, one of the major purposes of this mock-up test was to provide information related to the void fraction in a two-phase hydrogen moderator to confirm the performance of the HANARO cold neutron facility.

Measurements of the void fraction are of considerable technical importance for two-phase systems in many industrial applications such as nuclear reactors, chemical processing plants and the oil industry [2]. Various methods have been proposed for the measurement of a void fraction, and among these methods, the most widely used ones are the volumetric, electrical, ultrasonic, and radiation attenuation techniques. The radiation attenuation technique, especially, the gamma densitometer technique is widely applied for the measurement of the void fraction of various systems including a cryogenic two-phase system because it is non-intrusive, and in general, quite reliable and easy to apply [3,4].

In this work, we measured the void fraction in the hydrogen moderator in the moderator cell of the mock-up for the HANARO in-pool assembly by using a gamma densitometer technique. We designed and installed a densitometer by using an HPGe detector and an Am-241 gamma-ray source, and measured the void fraction and its distribution in the moderator cell. These measurements will be very useful for a characterization of the HANARO cold neutron facility.

2. Void fraction measurement by gamma densitometer

The gamma-ray attenuation technique is based on the fact that the intensity of a collimated gamma-ray beam decreases exponentially as it passes through matter [2,4]. For a two-phase flow, a void fraction can be obtained easily by

$$\alpha = \frac{\ln(I_\alpha / I_L)}{\ln(I_G / I_L)}, \quad (1)$$

where, I_α is the gamma-ray intensity measured by a detector for an arbitrary void fraction, I_G and I_L are the intensities measured for a single-phase vapour and a single-phase liquid, respectively. When the signal processing system of a gamma densitometer is operated in the count mode, the measured intensity can be represented by the counting rates of the gamma-ray detector.

3. Experimental setup

The mock-up test facility for the cold neutron source of HANARO is composed of an in-pool assembly (IPA) connected to a hydrogen buffer tank, a vacuum system, and a helium refrigeration system. The IPA consists of a vacuum vessel, a moderator cell, a transfer tube, a heat exchanger and the related piping. Additionally, several sensors and heaters are installed in the IPA for the test.

Figure 1 shows a layout of the moderator cell of the HANARO CNS. It is made of 6061 aluminium alloy, and its inner diameter and wall thickness are 130.0 mm and 1.0 mm, respectively. The inner shell is open at the bottom. The moderator cell is enclosed with a cylindrical vacuum chamber made of 6061 aluminium alloy. The thickness of the vacuum chamber is 5 mm, and it is 7 mm away from the moderator cell. In order to simulate the nuclear heating load applied to the aluminium alloy case, several line heaters are wound round the moderator cell and the inner shell. And also, rod heaters are installed in the liquid hydrogen zone.

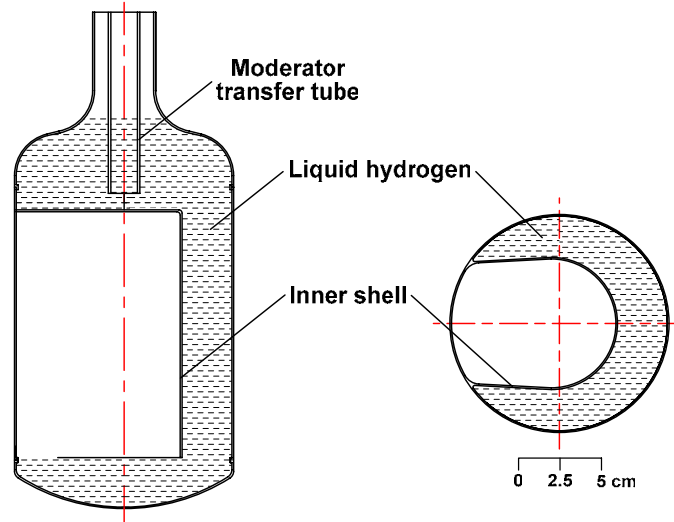


Fig. 1. Layout of the moderator cell of the HANARO cold neutron source.

Figure 2 shows a schematic diagram of the experimental setup to measure the void fraction. The selection of the gamma-ray source depends on the characteristics of the test section such as the pipe material, detection sensitivity, and shielding considerations. Cs-137 isotope is usually utilized in a conventional gamma densitometer for a water flow system. 662 keV gamma-rays from the Cs-137 isotope pass through the pipe wall materials easily, and the uncertainties due to the counting statistics can be reduced. They are quite sensitive to the water content unless the test section is small. However, it is very difficult to detect a void fraction change in the hydrogen medium with a transmission length of about 10 cm by using the gamma-rays from the Cs-137 isotope. The smaller the gamma-ray energy

is, the better the sensitivity for a void fraction change is. Therefore, in this work, 59.5 keV gamma-rays from the Am-241 isotope were utilized as the testing gamma-rays.

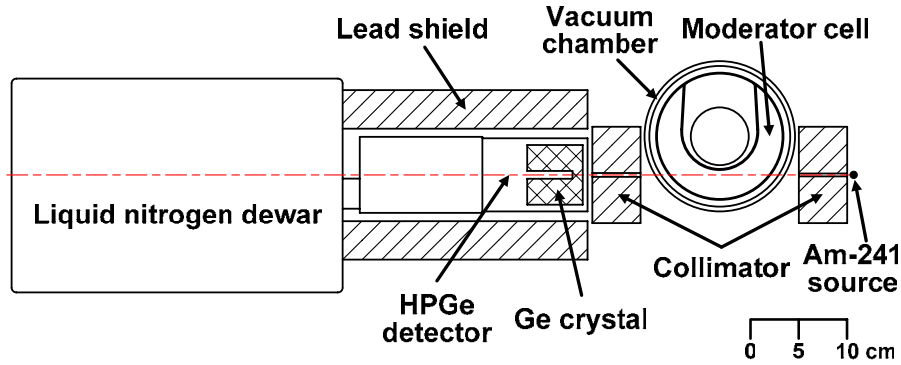


Fig. 2. Schematic diagram of the experimental setup to measure the void fraction.

In order to detect transmitted gamma-rays, NaI(Tl) scintillation detectors are commonly used due to their good detection efficiency. But, their poor energy resolution causes the gamma-ray peak to be broad. When 59.5 keV gamma-rays are being detected by using an NaI(Tl) detector, the full energy peak of the 59.5 keV gamma-rays is affected by the closely spaced background peaks such as the X-ray peaks originating from the lead shielding. Therefore, we used an HPGe detector with greater superiority for the energy resolution to reduce an uncertainty due to the peak area determination. In the analysis for the 59.5 keV gamma-ray peak area, a straight-line shaped background was assumed and subtracted.

The detector used in this work is a 171 cm³ coaxial HPGe detector with a detection efficiency of 40% of the 3"×3" NaI(Tl) detector. The gamma-ray spectroscopy system was set-up, and the shaping time of the amplifier was set at 6 μsec. The diameter of the source side collimator is 3 mm, and that of the detector side one is 4 mm.

The counting rates of the transmitted gamma-rays were measured at several electric powers applied to the heater installed in the moderator cell. The longitudinal void fraction distributions inside the moderator cell were determined. The pathlength of the gamma-rays transmitted through the test section of the hydrogen medium was 102.5 mm.

4. Results

From the thermodynamic calculation, the non-nuclear heating rate for this moderator cell was estimated to be about 3 W, which is negligible in comparison with the nuclear heating power [5]. If there is no heat load intrusion from the outside of the moderator cell, the hydrogen in the moderator cell would be maintained as a single-phase liquid. Therefore, the gamma-ray count rate for this single-phase liquid was determined by using measurements without an electrical heating power.

Figure 3 shows the longitudinal distribution of the relative count rates of the 59.5 keV gamma-rays measured by changing the height of the test section from the bottom of the moderator cell. From the figure, it is confirmed that the measured count rates below about 100 mm are smaller than those above this level. This originates from the fact that the local void fractions in the lower and upper regions of the moderator cell are affected by a heater's position [6]. In this experiment, the bottom of the rod heater was located at a height of 105 mm. The count rate value which was averaged for each region was deduced, and the void fraction at each region was determined by using the averaged count rates for the two-phase hydrogen medium and the count rate measurements for the single-phase hydrogen vapour and liquid. The overall averaged void fractions for the moderator cell were determined by considering the void fraction determined for each region and the volume inside the moderator cell occupied by each region.

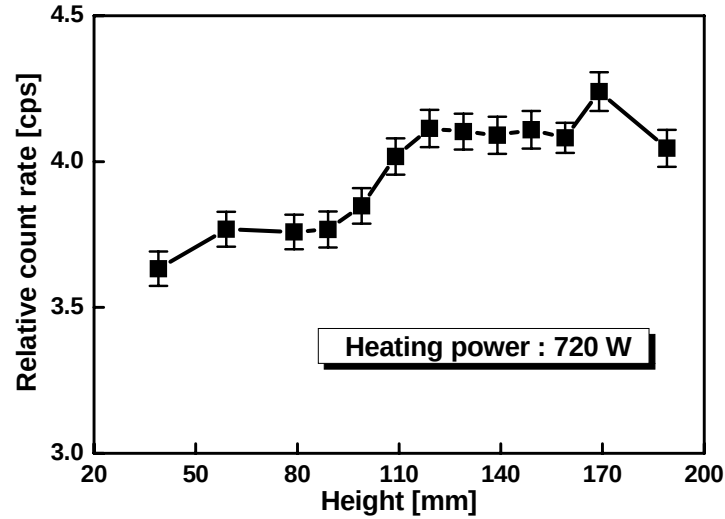


Fig. 3. Longitudinal distribution of the relative count rates of the 59.5 keV gamma-rays transmitted through the test section of the moderator cell.

Table 1 represents the void fractions determined by using the gamma densitometer used in this study at several heating powers. When the heating power is 720 W, the variation of the local void fraction is quite large, which is from 7.8 to 41%. From these results, the average void fraction is deduced to be 28.7%.

Table 1. Determined void fractions by using the gamma densitometer at several heating powers.

Heating power [W]	Void fraction		
	Lower region	Upper region	Volumetric weighted average
0	0.000±0.042	0.000±0.043	0.000±0.030
203	0.018±0.040	0.110±0.037	0.076±0.028
425	0.065±0.040	0.318±0.038	0.225±0.028
721.4	0.078±0.039	0.410±0.034	0.287±0.026
Gas	1.000±0.036	1.000±0.036	1.000±0.027

Figure 4 represents the result of the void fraction determination for the entire hydrogen medium inside the moderator cell. For the basic design procedure of the cold neutron research facility in HANARO, the heat load for the in-pile assembly was determined, and the nuclear heating rate for the moderator cell was estimated to be about 470 W. As shown in the figure, we assumed a linear relationship between the void fraction and the applied heating power. The void fraction at a heating power of 470 W was determined to be about 20% by using a fitting line for the measurements with the heating power.

As shown in table 1 and figure 4, the uncertainty in the void fraction determination by using a gamma densitometer is 2~3% in terms of the void fraction unit for a hydrogen medium of about 10 cm. This uncertainty value is quite big for the case of a small void fraction of less than 10%. The uncertainty of the determined void fraction is closely dependent on the uncertainty in the determination of the count rates of the transmitted gamma-rays through the test section. And, the count rate determination is directly related to the gamma-ray peak area. Therefore, in order to reduce the uncertainty of the determined void fraction, the gamma-ray peak area should be increased. With no

change for the detection system, reducing the uncertainty of the measurements by half requires an increase of the detection time by a factor of 4. Otherwise, the use of a gamma-ray source with a bigger activity or a detector with a bigger efficiency would be useful for reducing the uncertainty. At any rate, when the void fraction for the hydrogen medium is near 20%, the uncertainty in the void fraction determination by using a gamma densitometer is relatively small, and it can be regarded as an acceptable level.

Conclusively, the gamma-densitometer technique can be very useful for the measurement of a void fraction in a cryogenic liquid such as hydrogen which is used in a cold neutron source system.

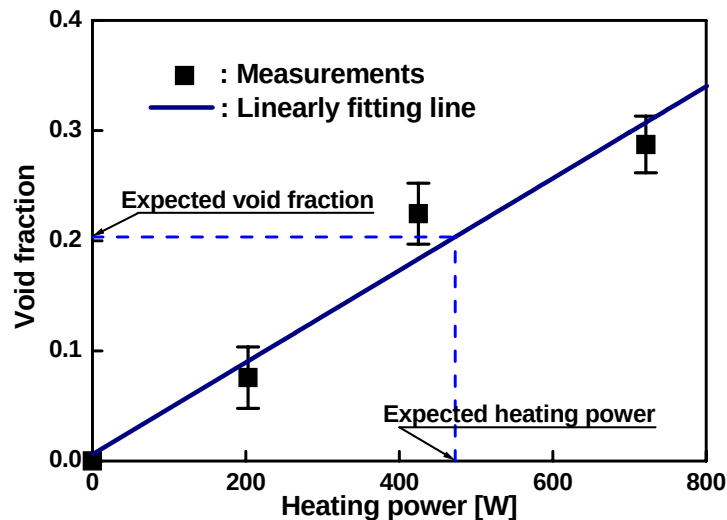


Fig. 4. Determined void fraction of the moderator cell.

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OPTIMISATION OF THE POOLSIDE FACILITY FOR NEUTRON DOPING OF SILICON IN HIGH FLUX MATERIALS TESTING REACTOR BR2

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ABSTRACT

The paper contains a description of the optimisation procedure performed during a design study of the poolside facility at the High Flux Materials Testing Reactor BR2 for neutron doping of large silicon crystals. Analysis of different moderator materials to maintain thermal neutron flux distributions in silicon crystals and the influence of geometrical design of the facility on neutron flux distributions and on nuclear heating were performed during the optimisation. Special attention in the design was paid for the feedback reactivity effect in the BR2 reactor core and to nuclear heating in moderator. A description of several variants of the poolside facility for irradiation of silicon crystals of the diameter 15-20 cm in the reactor pool of the BR2 is included in the present paper.

1. Introduction

When the High Flux Materials Testing Reactor BR2 first ventured into the Neutron Transmutation Doping (NTD) silicon business in 1992, the demand for 4 and 5-inch diameter irradiations was about equal. By 2003, this trend had completely shifted towards 5-inch whilst latterly a requirement for 6-inch is rapidly becoming the norm.

By the end of 2000, demand for NTD-Silicon production in BR2 began to exceed the available capacity of its existing silicon irradiation facility (SIDONIE). The commercial revenue generated for SCK•CEN by this device is considered to be an important contribution towards BR2's operating costs. However, this source of income is entirely dependant on the availability of only one 5-inch production facility which cannot be easily modified to meet today's increasing demand for 6-inch capacity nor can it be adapted to accommodate 8-inch diameter silicon irradiations!

More recently, customers have begun to anticipate their production requirements for 8-inch irradiations to meet the automotive industries fast growing demand for Insulated Gate Bipolar Transistors (IGBTs). World leading car manufacturers consider the production of these key electronic devices from 8-inch silicon to be an important part of their overall strategy which is aimed at reducing the price of the next generation of super fuel efficient Hybrid Electric Vehicles (HEV).

Although BR2 has horizontal beam tubes that could easily accommodate 6-inch and possibly 8-inch diameter silicon crystals, they do not have an adequate production capability to be cost effective and to meet the current demand!

Therefore, a scheme has been conceived for a pool-side facility (PSF) that is not constrained by the geometry of BR2's existing reactor pressure vessel (RPV) through-holes or channels. The largest being only 200-mm in diameter of which one is already occupied by SIDONIE.

2 Primary Objectives of the Study

Throughout the conceptual evolution of this project, the 'keystone' of its technical analysis has been the determination of the neutronic conditions that will exist within the proposed variants for a PSF. This has been carried out by BR2's Reactor Physics Group using a full-scale 3-D Model of the BR2 reactor to use as input data for the MCNP-4C Monte Carlo code. In particular, the information generated during optimisation has been used to determine:

- the most feasible scheme configuration.
- the optimum material for use as a neutron moderator
- the integrated (perturbed) thermal neutron flux density in each of the irradiation channels. Clearly, this is key data for establishing the NTD-Silicon production capacity of the PSF and hence its financial feasibility
- the spatial and axial distribution of the thermal neutron flux in the Si crystals. This is also critical information that is needed for assessing the technical feasibility of the scheme in terms of its capability to achieve a homogeneous resistivity profile and thereby keep Axial Resistivity Gradients (ARG) within acceptable limits

3 PSF Configuration Options

Focusing on the poolside position of the irradiation device, several proposals of the conceptual design were analysed.

In the Carrousel design it was planned to have 4 irradiation quadrants separated by Cd screens in order to reduce uncontrolled irradiation in sectors located at a large distance from the core. In this scheme, silicon crystals during irradiation were located at different distances from the reactor core and the thermal neutron fluxes in the silicon are different by one order of magnitude. Even if this fact is not taken into account, the dimension of such a device is too large to be installed in the available space in the pool of BR2.

In the parallel-channel type PSF (Fig.1), silicon crystals are located in the box filled with a moderator in order to mitigate neutron absorption by light water in the reactor pool. The moderator box outside the reactor core mainly prevents the loss of thermal neutrons in the pool water and partially improves the thermal fluxes by additional slowing down of fast and epi-thermal neutrons escaping from the reactor core. All channels have a different neutron fluence because of radial and azimuth dependencies of the thermal flux density. This shortcoming can be diminished by positioning the channels in the form of an arc (Fig.2).

One of the most important requirements of the PSF is to facilitate the production of a uniform distribution of ^{31}P within each of the silicon crystals. This can be achieved by a special procedure involving the physical positioning of the silicon inside the irradiation device. For this purpose, the flux profile with respect to its vertical axis must be at least symmetrical relative to a fixed position (inside the irradiation channel).

4 PSF Moderator Options

Absorption of the thermal neutrons in the light water in the reactor pool is so that makes practically impossible to irradiate large crystals of 20 cm, diameter. Neutron absorption can be considerably diminished by using a container filled with a moderator with a low neutron absorption cross-section. Several candidate materials for a suitable moderator were considered: heavy water; helium-4; graphite; beryllium; aluminium.

The largest value of the mean neutron flux density in silicon crystals is in the container filled with heavy water, while the minimum flux is in the container filled with Aluminium. The thermal fluxes in silicon targets in the container filled with the inert gas He-4 are located between the maximum and the minimum values. The fluxes in the container filled with graphite and Be materials are slightly lower than that in the heavy water container. Examples of the thermal neutron flux distributions versus the material of neutron moderators is presented in Fig.1.

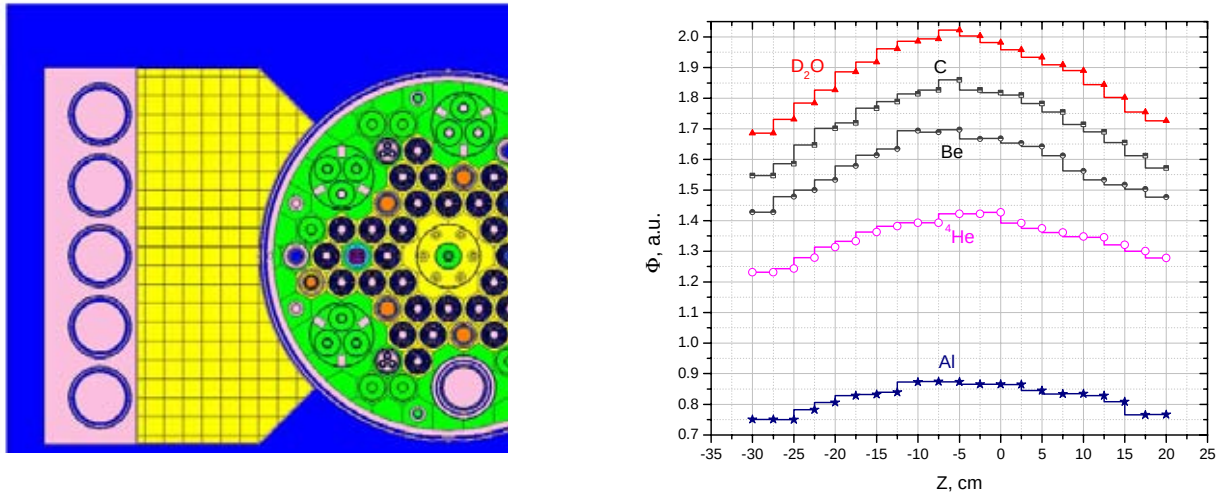


Fig.1. Schematic picture of parallel-channel proposal for Neutron Transmutation Doping silicon. The figure in the right shows a dependence of axial distributions of the mean thermal neutron flux density (in arbitrary units) in the silicon crystal versus the type of material of moderator.

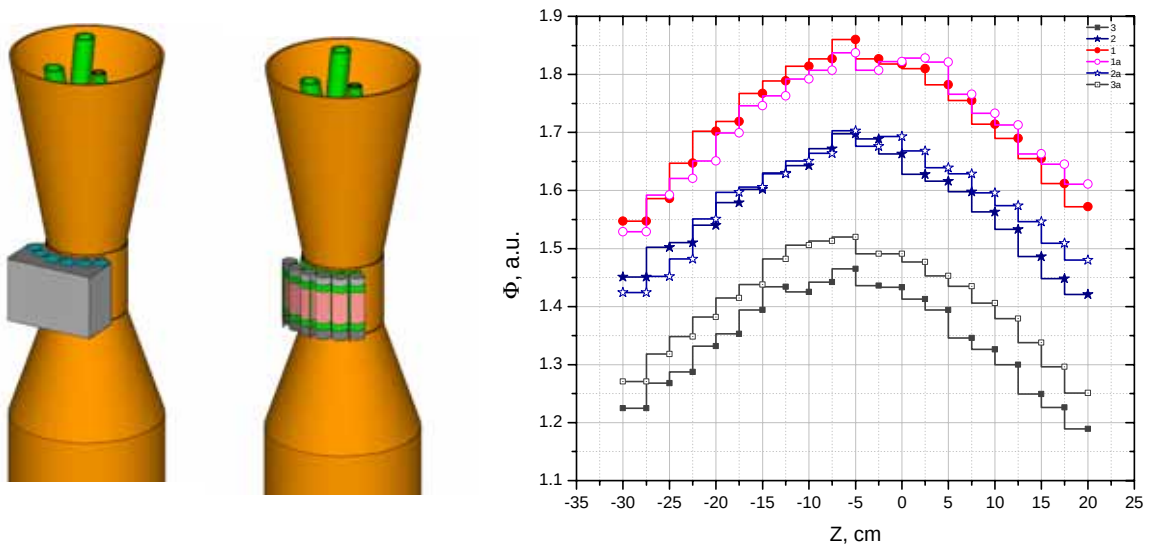


Fig.2 Schematic view of the PSF irradiation device for Neutron Transmutation Doping silicon. The figure on the right shows an example of the dependence of axial distributions of the thermal neutron flux versus the position of silicon crystals in the moderator box.

Each of the moderator variants has advantages and disadvantages:

- heavy water is very expensive and there is a problem of tritium activation during irradiation, moreover it is necessary to maintain a low concentration of light water in the case of leakage of heavy water from the container;
- a container filled with He is a self-floating structure in the reactor pool and therefore this causes problems with buoyancy;
- a graphite moderator under a long duration irradiation at low temperature accumulates Wigner energy which must regularly be removed by a special annealing procedure;
- beryllium is a hazardous material to machine and requires a specialised manufacturing facility;
- Aluminium is the simplest alternative, but the neutron fluxes are more than 2 times lower than in the graphite moderator.

It is also important for the PSF to have a highly thermal neutron environment and to keep fast neutron damage to the silicon lattice as low as possible. The position of the silicon during irradiation relative to the reactor core and the choice of moderator for the PSF can have a significant influence over the fraction of fast neutrons present. The cadmium ratio was estimated by calculations using the ratio of reaction rates for Co foil ($E_{Co}=1$ eV)

$$R_{Cd} = \frac{\int_0^{20} \sigma_{Co}(E) \Phi(E) dE}{\int_{E_{Co}}^{20} \sigma_{Co}(E) \Phi(E) dE},$$

Depending on the moderator type, the ratio within the silicon crystals varies from 25 to 100. The maximum and minimum ratios were found to be in the heavy water moderator and Al container respectively.

5 Further Considerations Regarding the Choice of Moderator

Whatever moderator is used for the PSF, it is important to understand what influence it may have over reactivity within the reactor core. For this purpose, changes of reactivity were calculated while taking into account the presence of the new facility in the reactor pool, as well as for the variation in temperature and density of the moderator. In the case of a graphite moderator, its presence near the RPV introduces an additional reactivity of about +5 pcm, which is comparable with the statistical error of Monte Carlo calculation. This change of reactivity is small because of the presence of a thick Be matrix between the fuel elements and the RPV. In this case an additional reflector does not change neutron reflection. The dependence of reactivity on the temperature, on the density and dimensions of the POSEIDON facility is also small and the reactivity changes ordinary by 10 pcm.

Detailed analysis of the dependence of the thermal neutron flux density versus the change of the temperature of the thermal neutron spectrum in the moderator, the variations of the moderator density, the change of the void fraction inside the container, the presence of admixture of the light water, variation of the plugs structure. Depending on the type of the technological variations the thermal neutron flux can vary by 10%.

An important problem with graphite as a moderator when irradiated at a low temperature is the accumulation of Wigner energy induced by fast neutrons. The risk of an uncontrolled release of accumulated hidden energy is low if the graphite is irradiated at a relatively high temperature. Another source of thermal energy is photons:

- the energy deposition induced by prompt photons produced in capture reactions;
- the energy deposition from delayed photons generated by fission products.

The energy deposition from prompt photons is calculated using the MCNP-4c code[1]. The energy spectrum and intensity of the delayed photons in the BR2 fuel elements were obtained from the SCALE-4.4a code [2] and used in the MCNP model for the transport calculations of photons from fission products. The heating from the β -decay reaction was calculated using the intensity of neutron capture reaction in structural elements.

The PSF is primarily designed to irradiate silicon crystals of 8-inch diameters. However, when irradiating 6-inch diameter crystals, the free space is occupied by filler containing the same material as the moderator in the box. The mean non-uniformity factor of the thermal flux distribution inside the silicon crystals which was calculated for 8-inches and 6-inches silicon crystals is spread in the region from 0.98-1.07. However, the statistical errors of these results are comparable with the observed effect.

6 Conclusion

In principal, the scheme adopted (POSEIDON) consists of six vertical, parallel-channels for the large volume production of 8-inch diameter NTD-Silicon. These form a concentric arc around the reactor core in a position that is immediately outside the RPV. The channels are located within a neutron moderator which displaces the reactor pool water from around them. Thereby, much of the thermal and epithermal neutron flux leakage from the reactor in this region is preserved for irradiating the silicon. The channels can be easily adapted with sleeves to provide for the bulk production of 6-inch diameter NTD-Silicon also.

Located outside the reactor pressure vessel and operating quite independently of all BR2's critical systems, POSIDON is currently configured to meet a very large proportion of the semiconductor industries foreseeable demand for 6 and 8-inches diameter NTD-Silicon production. This concept provides flexibility which allows it to be relatively easily reconfigured to accommodate even larger sizes of crystals or even a combination of 6-inch, 8-inch and possibly 12-inch diameters.

POSEIDON provides a very financially economical and convenient method for producing NTD-Silicon in BR2 with key attributes that can be summarised by:

- a very large volume of production capacity capable of meeting demand for the foreseeable future
- the flexibility to irradiate 6-inch, 8-inch and possibly 12-inch diameter silicon
- an irradiation environment with a cadmium ration that can be characterised as highly desirable

Due to these benefits, SCK•CEN has secured long-term agreements with all of the major customers for NTD-Silicon in Japan.

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MATERIALS SURVEILLANCE PROGRAM FOR THE OPAL RESEARCH REACTOR

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ABSTRACT

The OPAL research reactor has recently achieved full power and will commence normal operation in the early part of 2007. One aspect of the design of OPAL has been the inclusion of a surveillance program for the materials used in the reactor core regions. These materials are exposed to a high neutron flux and their properties, such as tensile strength, fracture toughness and physical dimensions (through radiation-induced growth), are expected to change through the life of the reactor. Estimates of these changes have been obtained from literature data and have been accommodated in the design. However, data at the operating temperature of OPAL is limited. In order to guarantee safe operation of these materials, a surveillance program was developed during the detailed design phase of the project. The program involves the placement of miniature samples in high flux regions close to the reactor core. These samples will be removed at intervals and will be subjected to extensive mechanical testing to determine any changes compared with samples in the unirradiated condition. Additional samples will be sectioned from other high-fluence components that will be removed well before the 40 year design life.

1. Introduction

The OPAL nuclear research reactor has recently commenced full power operation. It was constructed as a replacement for HIFAR, which had given 49 years of safe and reliable operation but which had become unable to provide the range of neutron based experiments and production facilities that are required to take ANSTO and the Australian nuclear industry into the 21st century. As part of the specification for OPAL a program of sample irradiations forming part of a surveillance program was made a requirement of the design. The surveillance program that has eventuated is a combination of the original INVAP proposal plus extensive modifications which have arisen as part of the detailed engineering and construction phases of the project.

The objective of the surveillance program is to monitor the core reactor materials to ensure that their mechanical properties are sufficient to ensure safe and reliable long-term operation of OPAL. The effects of greatest interest are the changes that neutron irradiation makes on the tensile and fracture properties of zirconium alloys and the possible effects of corrosion on all reactor materials. The objectives of the surveillance program will be met by placing a range of specimens in locations throughout the reactor pool (RPO) and which will be examined periodically during the 40 year design life of the reactor. This paper describes the methodology of the program, the test samples, their locations and the tests to which they will be subjected. Additional information will be provided on other specialised inspection procedures.

2. OPAL Research Reactor

OPAL is a 20 MW pool-type research reactor, where the core is located at the bottom of a 13 m deep pool of demineralised water. A schematic view of the OPAL reactor pool and internal structures is given in Fig 1. This shows the main components and the location of the key items of interest in the surveillance program. The reactor and service pool liners are stainless steel structures located inside a massive, high-density concrete block; the concrete providing both shielding and structural support.

The reflector vessel (RVE) surrounds the core, providing both the core boundary (and path for the primary cooling water) and the volume of heavy water (D_2O) that forms the reflector. It is a complex fully welded structure, with many through and re-entrant tubes providing access for irradiation facilities and beam tubes. Fig 1 shows the location of the RVE in the pool. It is a 2.6 m diameter cylinder with the core in a square-sectioned central region. Also shown in Fig 1 are the cold neutron source (CNS) and the associated vacuum containment (VC). The VC provides the barrier between the CNS moderator vessel (containing liquid D_2 at ~ 18 K) and the core; an important safety feature of the design. The VC is 3 m long Zr-2.5Nb tube with the aluminium moderator vessel located inside.

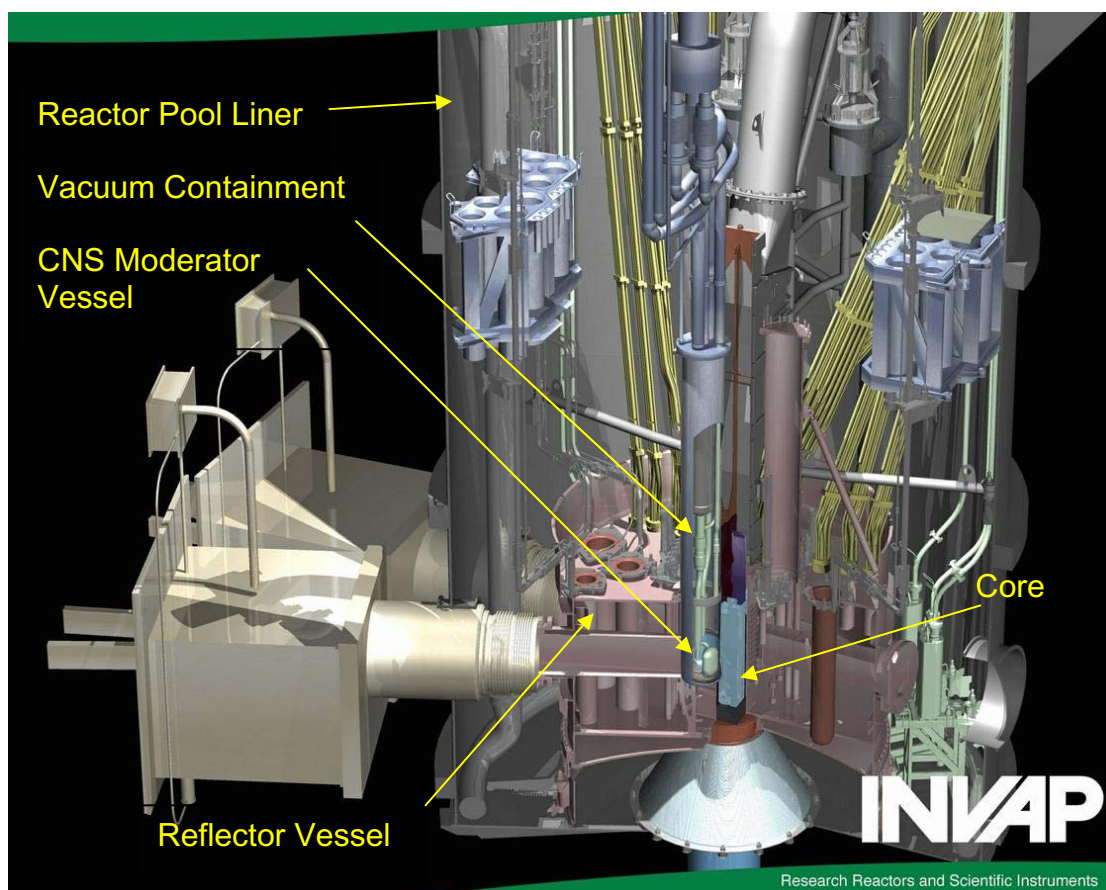


Fig 1. A schematic of the OPAL research reactor. Components of interest in the surveillance program are the reactor pool liner and the reflector vessel (around the core). Also visible are the CNS moderator vessel at the bottom of the vacuum containment.

3. Surveillance Program Methodology

The surveillance program is based on that described in ASTM E 185. This standard provides guidance for the setting up and operation of a surveillance program for light-water power reactors. The basic recommendations of this standard have been included in the surveillance program, however, changes have been made in a number of areas to reflect the lower level of risk associated with a research reactor compared with that of a power reactor.

The objectives of the program are to monitor the changes in materials properties of components essential for the safe operation of the reactor. These changes include corrosion (principally the pool liner material), growth in zirconium alloys and changes in the tensile and fracture properties of all materials exposed to radiation. Each of these effects is considered below.

Corrosion resistance is clearly an important materials property for components designed to operate for 40 years or more. All the materials selected for OPAL have excellent corrosion resistance under the reactor's operating conditions and problems are not expected. However, samples of dissimilar metal couples (Zircaloy-4/aluminium and 304L/aluminium) have been included in the program.

Irradiation by neutrons causes an increase in tensile properties. The tensile properties of all the materials used will increase and samples of all materials are included in the program.

Fracture toughness is expected to decrease for all the materials used. While a small decrease is acceptable, significant changes may invalidate the structural assessment performed in the design phase. A number of sample types will be used to obtain fracture data. Obtaining valid fracture toughness information for reasonably ductile materials is difficult, principally because of the sample size requirements for a valid assessment of toughness. A great deal of work has been undertaken over the past decade internationally on developing and validating small sample test methods that provide realistic values of toughness. A number of these will be used in the OPAL surveillance program, including the small punch (SP) test, the compact tension (CT) test and the sub-size Charpy test. These are described in the next section.

3.1 Sample Types

Simple “dog-bone” tensile samples are used extensively as they are the simplest and can be made small without seriously affecting the validity of the results. They will provide information on the tensile properties and some information on ductility. However, they cannot provide information on fracture toughness. Three other sample types are used; small punch (SP), compact tension (CT) and the quarter-size Charpy. All these will provide information on fracture properties. The three main sample types used in the surveillance program are shown in Fig 2. The advantage of the SP sample is clear from a weight and activity perspective, being much smaller.

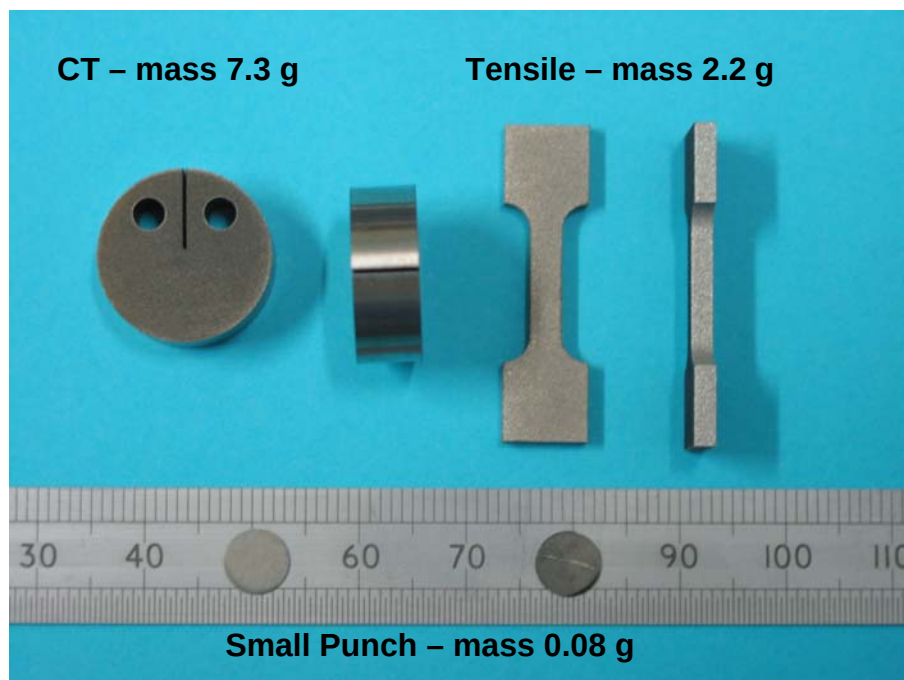


Fig 2. Photograph of the sample types used in the surveillance program. Of note is the small mass of the SP samples; important when testing radioactive materials.

The SP test is a recent development and is at the stage of significant international application in a wide range of industries; details of the test can be found elsewhere [2]. In brief, the technique uses small disks of the material (in this work 6 mm diameter and 0.5 mm thick), through which a hardened steel

ball is pushed. The load as a function of deflection is recorded. A measure of fracture toughness is gained through the use of a finite element model that uses the strain energy density at the point of initial fracture appearance. ANSTO has extensive experience in this technique and has developed the analysis to the point that very good correlations between absorbed energy and fracture toughness have been achieved.

The compact tension (CT) test-piece is one of the smallest that can be used to give a valid evaluation of fracture toughness. Well-known standards (ASTM) are followed for the testing. The advantage with this sample type is that it will be relatively easy to manufacture from the control rods when they are retired. “Relatively” in this case is used advisedly since the control rods will be highly radioactive on their removal from OPAL. However, in-cell machining facilities (using electro-discharge machining) will be used to prepare the samples.

3.2 Material Types

The materials of interest in the OPAL surveillance program are: Zircaloy-4 (used for the reflector vessel), Zr-2.5Nb (used for the cold neutron source vacuum containment), AlMg5 (used for the cold neutron source moderator vessel), Al6061 (used for the core grid and many in-pool components) and 304L stainless steel (used for the lower grid, hold down bolts for the upper grid, fuel clamps and reactor and service pool).

There is little data available for Zircaloy-4 at research reactor temperatures (power reactors generally operating at much higher temperatures). This is the reason why the material is included in the surveillance program. Zr-2.5Nb is also used in both power and research reactors and there is a similar shortage of data for this material in research reactor applications. The CNS VC was manufactured in Russia and the choice of the material was due to their experience in its use and its improved tensile properties over those of Zircaloy-4.

AlMg5 has been used in other cold neutron sources but its use is recent and limited, so there is little historical data. The moderator vessel is expected to be replaced after 10 years but the material has been included in the program to generate more information on its use. Al6061 has been used extensively in research reactors and is included in the program for completeness rather than for the generation of new information.

304L stainless steel is another material that is included in the program for completeness only. It is used extensively in power reactors and there is a great deal of information available on its performance under neutron irradiation.

3.3 Surveillance Sample Location

The surveillance samples are located in several locations. The corrosion coupons are located in seven sites within the RPO. The samples are located at three heights within the RPO and at locations that have a variety of water flow conditions. The bulk of the surveillance samples are located in the fission counter (FC) location in the reflector vessel. This group is in the surveillance sample holder that formed the original part of the program (Fig 3a). Dosimeter materials are included in the surveillance rig. These are placed in two of the small screw-top pots shown in Fig 3b. With the development of the program further samples were added. Many SP samples were included and examples are shown in Fig 3b. These will be removed for testing at 5 and 10 years of operation. This other group is located in a specially designed basket in the lowest position of one of the Iridium irradiation positions (Ir-2). Further samples will be extracted from retired control rods when they are removed after about 10 – 12 years of full power operation.

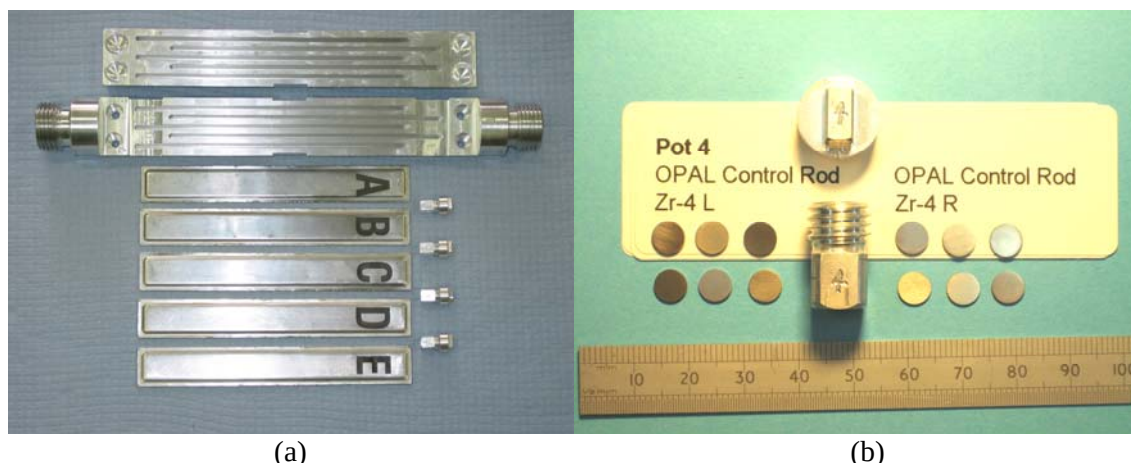


Fig 3. Photographs showing (a) the surveillance rig prior to final assembly and (b) the SP samples that fit inside the dosimeter pots. The five capsules (A to E) each contain 12 tensile samples, and are ~200 mm long.

4 Examination Procedures

The main surveillance samples will be removed from OPAL after 5, 10, 20, 30 and 40 years of full power operation. They will be subjected to a number of mechanical tests and dimensional measurements. All tests will be performed with the samples in a hot cell. The facilities for in-cell activities are currently being planned and will be available well before the first samples are ready for testing.

4.1 Corrosion Samples

The corrosion coupons will be examined initially after 1 year and then at intervals based on the rate of material loss determined at that time. Sample masses and dimensions were recorded prior to insertion into the reactor pool.

4.2 Samples from Retired Reactor Components

While the design life of the OPAL reactor is 40 years, there are a number of components that will be replaced before this time. In some cases it will be possible to include these components in the surveillance program and use their materials to manufacture test samples. The prime components in this group are the control rods. These sit in the centre of the core and at the positions of highest fast flux. In 12 years (the period after which they will be replaced) they will receive a fluence higher than that received by the RVE in 40 years of operation. The control rods will be sectioned into discs and then prepared as CT specimens for testing. Data from the OPAL Reactor Control and Monitoring System will allow the fluence variation along the rods to be determined. Fracture data will therefore be available for a wide range of fluences; important for determining lifetimes.

The frame which surrounds the control plates has also been designed with a weld that is positioned so that quarter-size Charpy samples can be removed. Both these and the previous CT samples will need to be prepared in a hot-cell because of the radioactivity levels. The facilities and techniques are to be developed well before the first control rods are removed. The control rods and plates are shown in Fig 4. The control rods have also been marked so that growth can be determined when they are removed. Fiducial marks were scored on the rod surface and the distance between them measured using a travelling microscope. The distance (~200 mm) will be re-measured when the rods are retired from service.

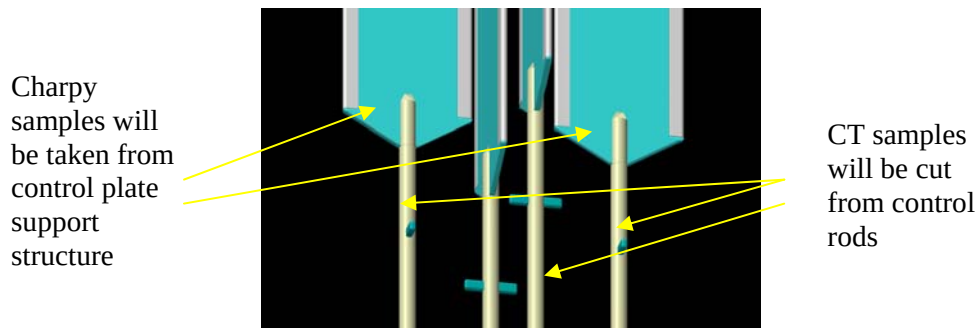


Fig 4. Schematic of the control plates and control rods (there is also a fifth cruciform shaped control plate in the centre of the four plates indicated above). The control plates are hafnium surrounded by a framework of zirconium alloy. They will be replaced after 12 years of operation and will be used to prepare additional samples for testing.

5 Conclusions

The core components of the OPAL reactor have been designed to last at least 40 years. In order for this to be achieved the design included materials that are resistant to corrosion and that have a predictable response to neutron radiation. The monitoring of changes to material properties and in-service degradation has been included in an extensive surveillance program that includes all materials used in the areas important to safety; such as the core, the reflector vessel and the pool liners.

The surveillance program schedules the removal of test coupons from the reactor core at regular intervals. These samples will have (in most cases) received a fluence greater than the component they represent, and so will enable advanced warning of any unexpected deterioration of material properties. A great deal of work has to be performed in the next 5 years in preparing for the removal of the active test samples. Mechanical testing and sample preparation facilities for use in hot cells are currently being designed and will be ready before the first samples are removed in about the year 2011.

The information on material property changes provided by the surveillance program, along with the normal day-to-day in-service inspection activities undertaken by the OPAL reactor operations and maintenance staff, will ensure that OPAL is operated safely and with high availability throughout its operational life.

6 References

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Session V

**Innovative Methods in Research Reactor
Physics**

MCNPX 2.6.C vs. MCNPX & ORIGEN-S: State of the Art for Reactor Core Management

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ABSTRACT

This paper discusses the application of the Monte Carlo burnup code MCNPX 2.6.C for the criticality and depletion reactor core analysis of the Material Testing Research Reactor BR2 in Mol, Belgium. A comparison with the developed at the BR2 reactor department combined MCNP&ORIGEN-S fuel depletion method is presented. The accuracy of the both methods, the consumption of the calculation time, the depletion capabilities, the advantages and disadvantages of use of the both methods are discussed. Validation of MCNPX 2.6.C is performed on the reactivity measurements at the Reactor BR2.

1. Introduction

In this paper we discuss the application of the Monte Carlo burnup code MCNPX 2.6.C [1] for the reactor core management of major reactor systems. The code MCNPX 2.6 is an extended version of the code MCNPX 2.5.0 and includes new depletion/burnup capabilities. At the present time MCNPX 2.6 is under active development and the latest versions of the code (A,B,C,...) are available to beta testers under a Beta Test Agreement. The code is tested mostly on simple reactor core models, represented by a single or few fuel assemblies. In this paper we present results of testing of the code on the whole core of complex heterogeneous system – such as the core of the Material Testing Reactor BR2 in SCK•CEN, in Mol, Belgium. A comparison with the combined MCNP&ORIGEN-S method [2] for reactor core physics analysis is presented. Validation of the code MCNPX 2.6.C is performed on the reactivity measurements at Reactor BR2. The detailed full scale 3-D heterogeneous geometry model of the reactor BR2 is used in the calculations.

2. MCNP&ORIGEN-S depletion methodology

The depletion calculations are performed using the burn up code ORIGEN-S [3]. It is a module of the SCALE system, which can be used in combination with other modules of the SCALE or it can be used as a stand-alone module as it is in the presented here calculations. The ORIGEN-S nuclear data libraries contain cross sections and fission yields for LWR. MCNPX is used for calculations of the continuous energy reaction rates and fluxes, which are converted into one – group constants. The MCNP calculated effective microscopic cross sections $\langle \sigma \rangle_{eff}$ for the main actinides, dominant and some non dominant fission products of the HEU fuel, weighted in the spectrum of the needed fuel region j , are used to update the existing cross sections for the LWR reactor in the ORIGEN-S libraries (see Table I). The input for ORIGEN-S can be the fission power or the neutron flux, calculated by MCNP in the spatial cells where the burnup calculations are needed. ORIGEN-S evaluates the evolution of the isotopic fuel densities for the desired number depletion time steps. The isotopic fuel composition for a given time step is introduced back into the MCNP model and distributed in the core using the detailed 3–D power peaking factors, which are earlier evaluated with MCNP [2]. The comparison of the depletion methodologies by MCNPX 2.6.C and MCNPX&ORIGEN-S method is schematically presented at Fig. 1.

Table I. MCNPX calculation of effective thermal microscopic cross sections in typical fuel channel of the reactor BR2, which are used to update the existing cross sections in the ORIGEN-S libraries for

LWR. The cross sections data from the files ENDF/B-V,VI are used in the calculations of $\langle \sigma \rangle_{\text{eff}}$ [barn] by MCNPX.

Nuclide	$\langle \sigma \rangle_{\text{therm}}^{\text{ORIGEN-S}}$	$\langle \sigma \rangle_{\text{therm}}^{\text{MCNPX}}$	Nuclide	$\langle \sigma \rangle_{\text{therm}}^{\text{ORIGEN-S}}$	$\langle \sigma \rangle_{\text{therm}}^{\text{MCNPX}}$
$^{235}\text{U} (n,\gamma)$	98	68	$^{103}\text{Rh} (n,\gamma)$	150	113
$^{235}\text{U} (n,f)$	520	400	$^{105}\text{Rh} (n,\gamma)$	1.8E+04	1.2E+04
$^{238}\text{U} (n,\gamma)$	2.73	2	$^{135}\text{Xe} (n,\gamma)$	3.6E+06	2.2E+06
$^{238}\text{U} (n,f)$	0	8E-06	$^{147}\text{Pm} (n,\gamma)$	235	127
$^{237}\text{Np} (n,\gamma)$	170	153	$^{149}\text{Sm} (n,\gamma)$	4.15E+04	5.5E+04
$^{237}\text{Np} (n,f)$	0.019	0.013	$^{150}\text{Sm} (n,\gamma)$	102	72
$^{239}\text{Pu} (n,\gamma)$	632	360	$^{151}\text{Sm} (n,\gamma)$	1.5E+03	8.3E+03
$^{239}\text{Pu} (n,f)$	1520	750	$^{152}\text{Sm} (n,\gamma)$	210	150

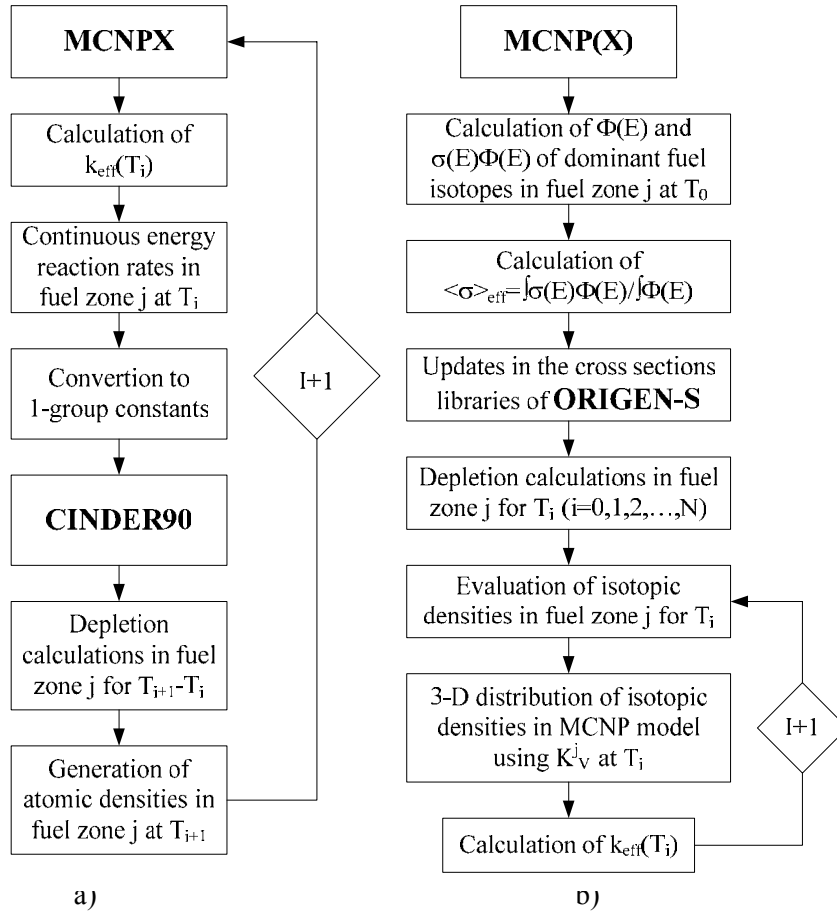


Figure 1. Comparison of depletion methodologies by: a) MCNPX 2.6 and b) MCNP&ORIGEN-S.

3. MCNPX 2.6 depletion methodology

The depletion/burnup capability in MCNPX is based on the 1 – D burnup code CINDER90 [1] and Monte Burns [1]. The MCNPX depletion process internally links the steady – state flux calculations in MCNPX with the isotopic depletion calculations in CINDER90. MCNPX runs a steady – state calculation to determine the effective multiplication factor k_{eff} , 63 – group fluxes and continuous energy reaction rates for (n,gamma), (n,f), (n,2n), (n,3n), (n,alpha) and (n,p), which are converted into one – group constants and used by CINDER90 to carry out the depletion calculations and to generate new number densities for the next time step. MCNPX takes those new number densities for the corresponding fuel cells and generates another set of fluxes, reaction rates. The process is automatic and repeats itself for each time step until the requested final time step. The calculated MCNPX 63 – energy group fluxes in combination with the inherent 63 – group cross sections of CINDER90 are used to determine the rest of the interaction rates, which are not calculated by MCNPX.

4. Testing of MCNPX 2.6.C on the Research Reactor BR2

The burnup code MCNPX 2.6.C is tested on the Reactor BR2. Depletion and eigenvalue calculations are performed for the full scale 3-D heterogeneous geometry model of the reactor, which describes the real reactor core of BR2 in a form of a twisted hyperboloidal bundle [2]. The evolutions of the macroscopic, effective microscopic cross sections and atomic densities are evaluated using CINDER90 and compared with ORIGEN-S.

4.1. Evolution of macroscopic cross sections by MCNPX 2.6

The evolutions of the macroscopic cross sections of the main actinides, burnable absorbers, and fission products were performed for the fresh fuel elements, which have been irradiated during 5 operating cycles with shutdowns ~ 20 days and without core reloading (see Fig. 2). It was obtained that the major contributions into the negative reactivity of the core give the burnable absorbers ^{10}B and ^{149}Sm in the fresh fuel. For the burnt fuel the main contributors into the negative reactivity are ^{149}Sm and ^{10}B at BOC and ^{135}Xe during the cycle and at EOC, which is in accordance with the results in Table II [2]. Among the non dominant F.P. the most contributions come from ^{103}Rh , ^{147}Pm , ^{151}Sm , ^{152}Sm . All other non dominant isotopes have been evaluated, but their Σ are less than 0.0005 cm^{-1} .

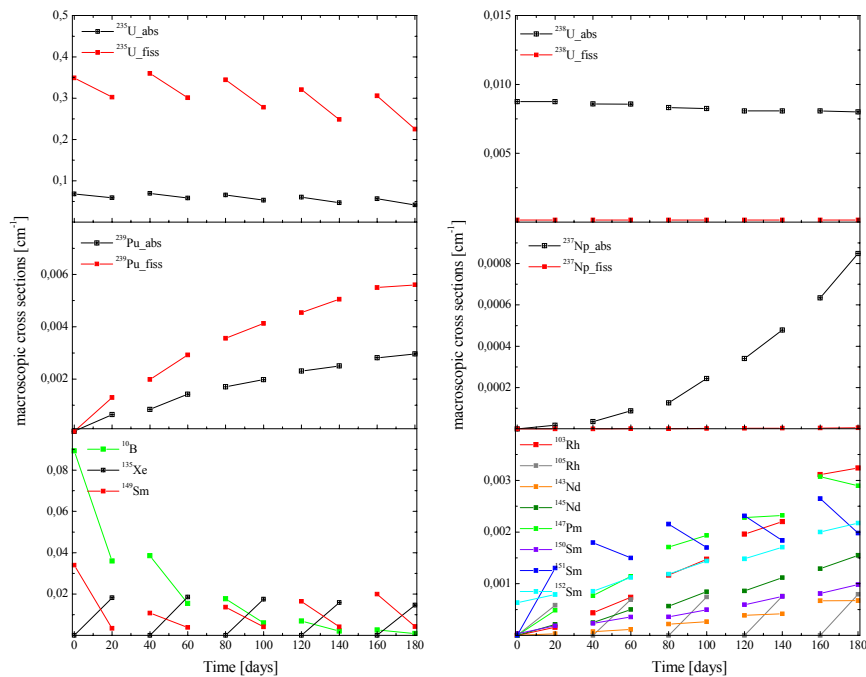


Figure 2. Evolution by MCNPX 2.6.C (CINDER90) of the macroscopic cross sections of the main actinides, burnable absorbers and dominant F.P. during 5 operating cycles (shown are data only for BOC and EOC).

4.2. Comparison of the atomic densities by CINDER90 and ORIGEN-S

The final goal of this study was to compare the depletion and criticality capabilities of the new burn up Monte Carlo code MCNPX 2.6.C with those of the combined MCNP&ORIGEN-S method. The evolutions of the isotopic fuel densities of the HEU fuel are evaluated by CINDER90 and by ORIGEN-S and compared at Fig. 3. As can be seen and as it was expected the evolutions of the masses of major fissile actinides and dominant F.P. are similar, because the both methods use the calculated reaction rates by the same Monte Carlo method, i.e. MCNP, which are further introduced into the one – group depletion equation.

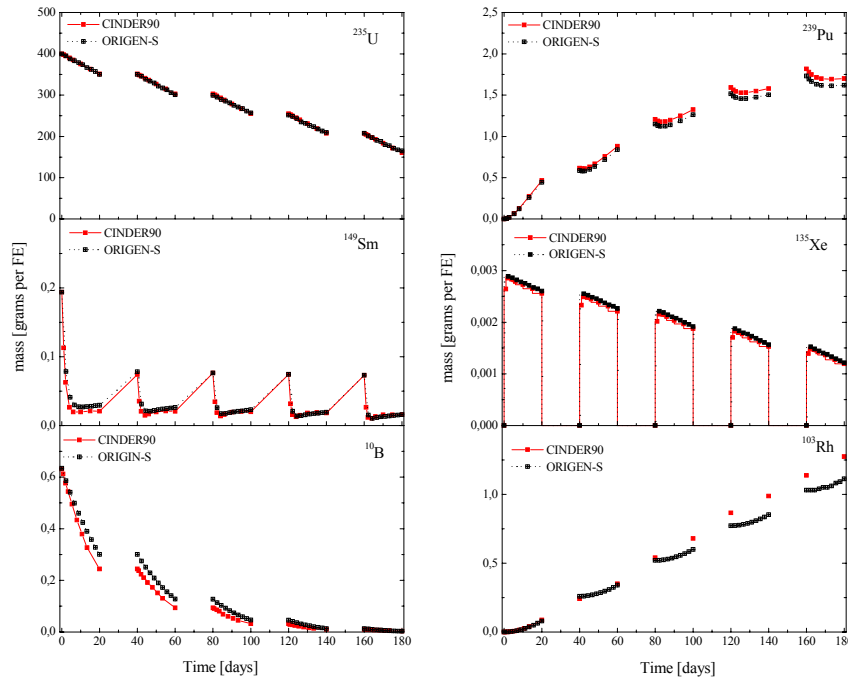


Figure 3. Time evolution of fissile isotopes and dominant F.P. in HEU (90% ^{235}U) fuel

4.3. Criticality calculations by MCNPX 2.6 and MCNP&ORIGEN-S

Finally, k_{eff} and the reactivity evolutions during an operation cycle have been evaluated by the both methods and validated on the reactivity measurements of the reactor BR2. The calculation procedure by MCNPX 2.6.C includes automatic calculation of k_{eff} and nuclide inventory by CINDER90 at each depletion time step. The reactivity calculations by the combined MCNP&ORIGEN-S method are performed in the following manner: the effective microscopic cross sections for major fissile actinides, dominant and some of the non dominant F.P. are evaluated with MCNP at BOC and used by ORIGEN-S to evaluate the isotopic fuel densities for all needed depletion time steps. After that the isotopic fuel composition for a given depletion step is introduced back into the MCNP model and distributed in the core using the preliminary calculated with MCNP 3 – D power peaking factors [2]. The k_{eff} is calculated for the relevant time step (see Fig. 4a). The calculations by both methods can be performed at the same critical position of CR bank at BOC. After that the reactivity value for each time step (Fig. 4b) together with the calculated earlier differential CR worth (Fig. 4c) are used to evaluate the positions Sh of the CR bank during the operating cycle (Fig. 4d). The comparison of the reactivity evolutions and CR motion during the cycle determined by the both methods has shown an acceptable agreement with the experimental values (see Fig. 4b, 4d).

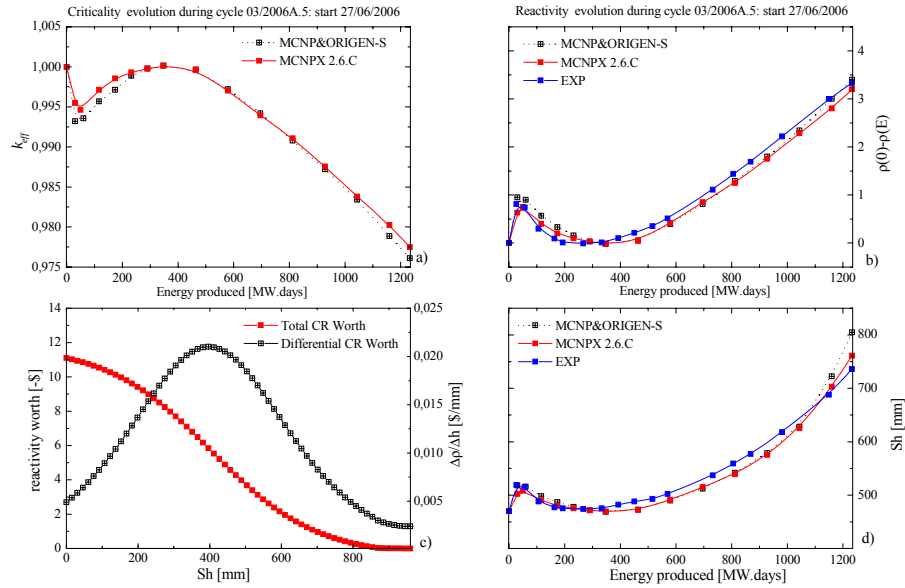


Figure 4. a) Evolution of k_{eff} ; b) Reactivity evolution; c) Total and Differential CR Worth; d) Motion of the Control Rods bank during cycle 03/2006A.5 of BR2.

5. Conclusions

The capabilities for depletion and criticality reactor core analysis of the new burnup Monte Carlo code MCNPX 2.6.C are compared with those of the combined MCNPX&ORIGEN-S method for the reactor BR2. The both methods use the same Monte Carlo code, which is linked with a 1 – D depletion code: CINDER90 in MCNPX 2.6 and ORIGEN-S in the MCNP(X)&ORIGEN-S method. In the both methods the reaction rates are calculated by MCNP(X) and the one – group constants are introduced into the depletion equation. The difference is that in MCNPX 2.6.C the whole process is automatic and the steady – state flux calculations by MCNPX in the requested fuel region are internally linked with the depletion calculations by CINDER90. Therefore the reaction rates are updated for each time step in the requested fuel region during the irradiation period. In the MCNP&ORIGEN-S method the reaction rates are calculated by MCNP once – at BOC and introduced into ORIGEN-S, which performs the depletion calculations for all desired time steps. Then the isotopic fuel composition for a given time step is introduced back into the MCNP geometry model and distributed in the core using the calculated earlier 3 – D power peaking factors, and the k_{eff} is evaluated. The same procedure is repeated for each time step, so that the different depletion steps can be calculated independently and simultaneously, which saves a lot of computational time. The number of the fuel depletion zones used in the MCNP&ORIGEN-S method is about 4000. Although the number of the fuel cells, in which the material can be burnt is unlimited in the latest version MCNPX 2.6.C, in practice, for a complex heterogeneous system, the number of the spatial fuel zones, which can be depleted is still limited by the allowed computer memory. The accuracy of the criticality calculations by MCNPX 2.6.C is still lower in comparison with the validated on many experimental results MCNP&ORIGEN-S method. However, the development of the code MCNPX 2.6 is a dynamic process, and each higher version of the code is an improved variant of the previous one....

6. References

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SIMULATION OF IRRADIATION OF A BUNDLE OF MOX FUEL RODS IN THE OMICO EXPERIMENT IN BR2

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ABSTRACT

A description of the calculation method which was applied for the simulation of the irradiation history of an assembly of different MOX fuel rods in the BR2 is given in the paper. The Monte Carlo simulation of irradiation experiment (OMICO) consisting of 16 MOX rods of different fuel compositions and assembled into two separated bundles one over other and loaded into one of the in-pile sections of the PWR simulation loop in BR2 was performed using the BR2 model with an interface module linking the input data for the MCNP and SCALE codes. The results of a direct Monte Carlo simulation are compared with the results of online thermal balance measurements of the power distribution in the IPS1 in-pile section comprising the OMICO bundles. In most cases the difference between the experimental and calculated values is less than 7-10%.

1. Introduction

The experimental program in the Belgian High Flux Materials Testing Reactor BR2 includes irradiation of different materials and of new types of nuclear fuel, radio-isotopes production, neutron transmutation doping of Si, et. al. The monitoring of irradiation conditions in IPS channels of BR2 includes on-line measurement of the thermal power using the thermal balance method implemented in the data acquisition system BIDASSE. This method permits to perform on-line measurement of absolute values of the deposited heating energy in the IPS channels. However, the detailed distribution of the power density inside the channel can be obtained using only the preliminary calculated relative distributions of power in all structural elements in the channel. Supplementary control of neutron fluxes inside the channel using self-powered neutron detectors was developed by L.Vermeeren and incorporated into BIDASSE system. The absolute values of the calculated power and temperature distributions in the IPS channels strongly depend on the reactor core load and on the reactor power history during the operation cycle. The maximum allowed deviation from the requested power density in the irradiated fuel rods is 10%, our goal is to limit this deviation to less than 5%. A post irradiation examination of the fission products distribution in the rods is used to reconstruct 'a posteriori' the distribution of fission events density in the fuel rods and consequently of the energy deposition.

In the present paper we focus on the approach used to simulate and predict power and burn-up distributions in different complex fuel assemblies. In particular, we consider a bundle of different MOX fuel rods grouped into 2 segments one over other and containing 16 fuel rods of different fuel compositions in the OMICO experiment. One group of the fuel rods was equipped with detectors placed inside fuel rods to allow online measurements of temperature and pressure.

The general view on the 3-D MCNP model used in the calculations is shown in Fig.1. Orientation of the fuel rods OMICO in the channel IPS1 is shown in left Fig.1. In the right Fig.1 the position of the IPS1 channel is marked by red-blue colours in the reactor core cut.

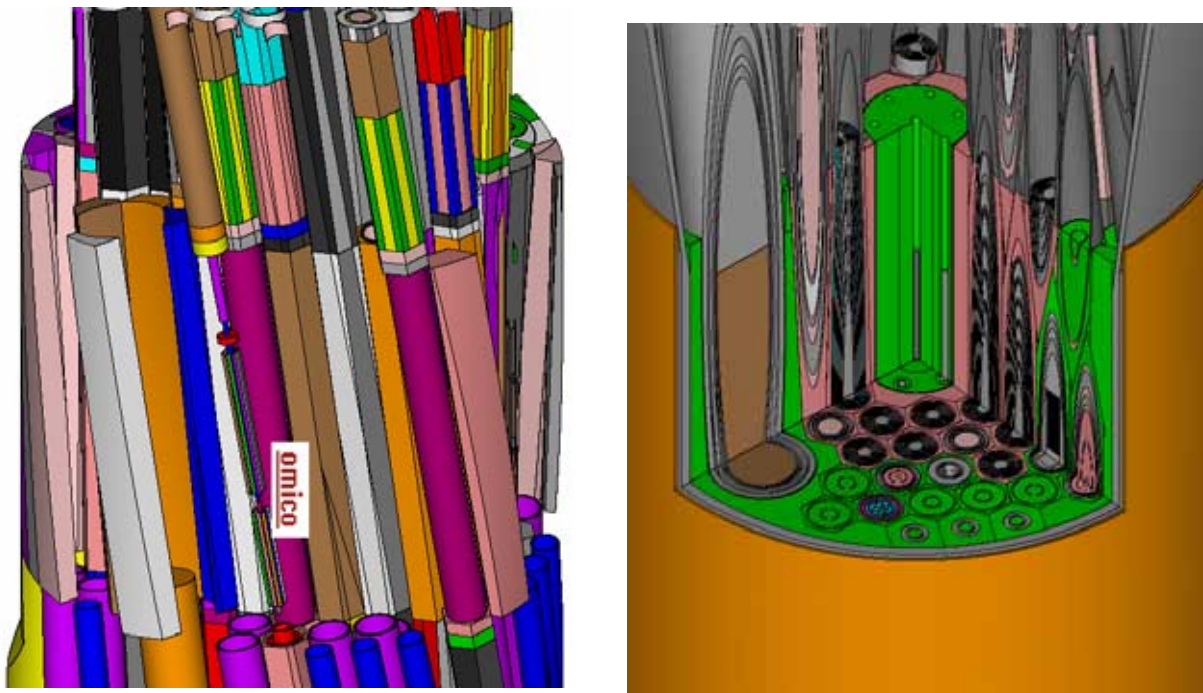


Fig 1. General view of inclined channels and a reactor core cut in MCNP 3-D model of the BR2. Two fuel bundles grouping of 8 MOX fuel rods each are shown in the left figure, while IPS1 channel (marked by red-blue colours) can be seen in the right figure of the BR2 core.

Each bundle includes of experimental fuel rods of different type, different initial enrichment and different fuel composition. Moreover, each bundle contains fuel pellets of different geometrical form: annular or solid fuel pellets. During the 7 irradiation cycles the environment near the position of the experiment was changed several times to satisfy all requested irradiation conditions in BR2. Examples of channels arrangement in different irradiation cycles near the position of the IPS1 channel are shown in Fig.2. In the last two cycles one bundle of MOX rods in IPS1 was replaced by dummy steel rods, while the irradiation of second bundle continued.

2. Calculation of fuel burn-up distribution in fuel rods

Accurate prediction of the fuel burn-up distribution and of the change of fissile nuclides concentration in the fuel element is important for maintaining the requested power and irradiation conditions in the tested fuel elements. However, it is difficult to perform a direct calculation of the detailed fuel burn-up distributions using the Monte Carlo codes in small parts of the fuel element irradiated in different positions in the reactor core. In practice the mean fuel burn-up in the fuel elements is calculated using average neutron fluxes inside fuel elements (sometimes in small numbers of fuel zones). This simplified approach permits to predict accurately an irradiation history of the fuel elements in the reactor core. However, detailed information about the spatial distribution of the fuel burn-up in the fuel element is not provided in this way.

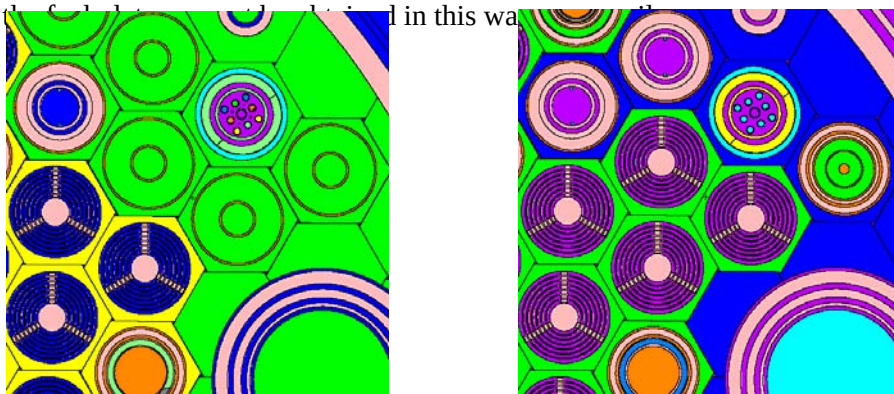


Fig.2 Examples of channels arrangement in different irradiation cycles near the IPS1 channel containing OMICO rods.

The fuel burn-up history and the change of fuel composition in the local fuel zone can be calculated using the mean values of the burn-up in the fuel element (rod, plate) and using the distributions of power peaking factor. The regular meshes of registration cells, $\{v\}_n$ $\{n=1,N\}$, are created in fuel elements for this purpose.

The dependence of the fuel burn-up, $\beta(v,T)$ expressed as the ratio of burned fissile atoms in the registration cell $\{v\}_n$ to their initial concentration, versus the energy released in fuel zone at the end of fuel cycle T is determined as

$$\beta(v,T) = C_v \frac{\int_0^T P(v,t) dt}{M(v)} 100\%, \quad C_v = \frac{A_U}{N_A E_{eff}} \alpha_v, \quad \alpha_v = \frac{\langle \sigma_f + \sigma_c \rangle_v}{\langle \sigma_f \rangle_v}$$

where A_U is the atomic mass number of the fissile element, N_A is the Avogadro constant, E_{eff} is the effective fission energy, $M(v)$ is the weight of the fuel in the cell $\{v\}$ in the beginning of the irradiation period and $P(v,t)$ is the power at the time t . Writing similar expression for the mean burn-up in the whole fuel element, we can find the dependence of the local burn-up $\beta_v(T_N)$ versus the mean burn-up, $\bar{\beta}(T_N)$, in the fuel rod after the N^{th} irradiation cycle, and versus the change of the specific power peaking factors $k_v(T_i)$ during the irradiation.

After N irradiation cycles (duration of N cycles is T_N) the local fuel burn-up, $\beta_v(T_N)$, after T_N irradiation time in each fuel zone $\{v\}$ can be calculated

$$\beta_v(T_N) = \beta_v(T_1) + \sum_{i=2}^N (\bar{\beta}(T_i) - \bar{\beta}(T_{i-1})) \times k_v(T_i) = \sum_{i=1}^{N-1} \bar{\beta}(T_i) [k_v(T_i) - k_v(T_{i+1})] + \bar{\beta}(T_N) k_v(T_N),$$

The mean fuel burn-up in the whole fuel rod (element) can be calculated for the mean operation power using the SCALE or ORIGEN codes. The dependence of the nuclide composition versus the energy deposition (or equivalently on the fuel burn-up) in the fuel rod (element) can be calculated only once and be kept in the form of a burn-up data bank. These data are used each time when it is necessary to obtain the fuel composition for the local burn-up in the registration mesh.

In the approach presented here it is not necessary to solve the burn-up equation in each registration zone. The only what is necessary is to calculate the detailed distribution of the power peaking factors on the registration mesh. After that the distributions of the power peaking factors, k_v , are used to obtain the distributions of the fuel burn-up in registration cells. The nuclide composition in the registration cell for the obtained fuel burn-up can be taken from the burn-up data bank containing the dependence of the fuel composition on the fuel burn-up.

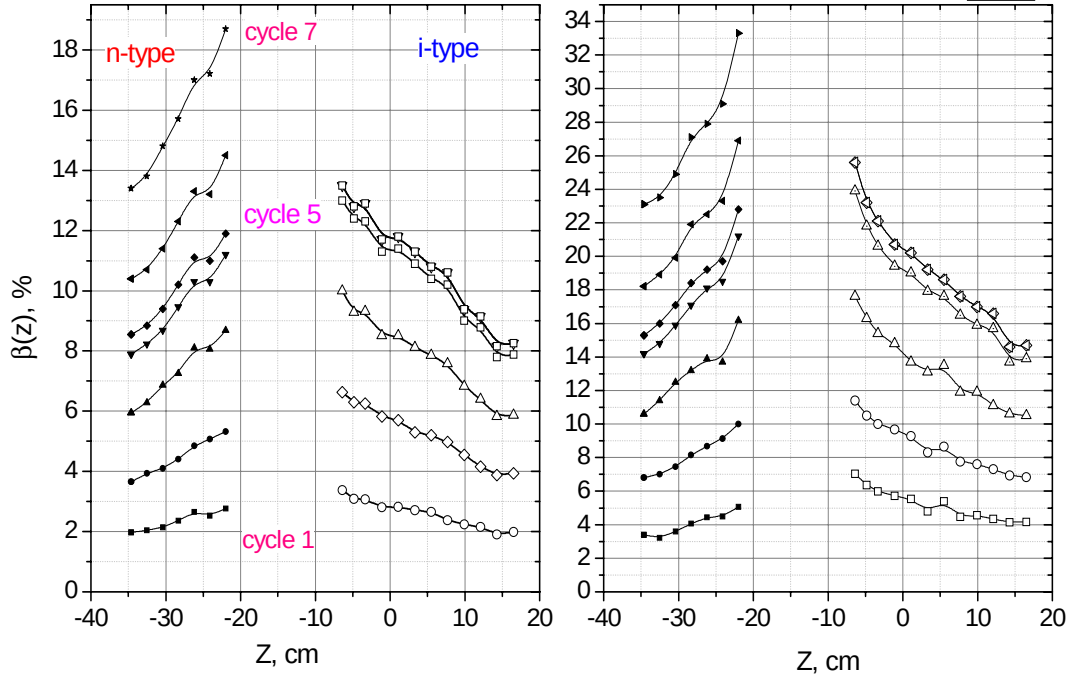


Fig.3 Example of spatial distributions of fuel burn-up in different fuel rods obtained for the scenario of several irradiation cycles.

The distributions of fuel burn-up in all 16 OMICO rods in 7 irradiation cycles was calculated using the described approach in automatic interface module linking input data for MCNP and SCALE codes. Because of a strong axial non-uniformity of neutron fluxes in the BR2 core and due to segmentation of rods, the distributions of the fuel burn-up are very non-uniform, see example in Fig.3. In the 'i'- type of fuel rods the local burn-up can vary from 13% to 34%, and in the 'n'-type can vary in the range of 8% - 26% (relative the initial concentration of fissile nuclide).

3 Comparison of calculated power history in the IPS1

Nuclear heating induced by prompt and delayed photons in structural materials of IPS1 channel amounts in average to about 20% of the total heating power in the channel comprising MOX fuel rods. The energy deposition from prompt photons produced in fission and neutron capture reactions in the reactor core was calculated using the MCNP model of BR2. The fraction of an energy deposited by delayed photons was estimated in a separate step. The spectrum and the intensity of the delayed photons from fission products were calculated using the SCALE-4.4a code, for example, in SAS2H (depletion analysis) module. The obtained spectrum of delayed photons was used as an external source of photons distributed in fuel elements in the BR2 core. Additional MCNP calculation of photon transport was performed with a new external source of delayed photons. The deposited heating energy E_{γ}^{SCALE} induced by the delayed photons was normalised per total intensity I_{γ} of delayed photons produced in BR2 fuel elements

$$E_{\gamma}^{SCALE} = n_{\gamma} \bar{E}_{\gamma} = \frac{I_{\gamma} \bar{E}_{\gamma} E_{fiss} 1.6 \times 10^{-19}}{P_{FE}} \text{ MeV / fiss}$$

where the effective fission energy is equal to $E_{fiss}=200$ MeV/fission, n_{γ} is the number of delayed photons, and $\bar{E}_{\gamma}$ is the mean energy of photons. Normalizing the energy released with delayed photons in ^{235}U to $E_{\gamma} = 7.2 \pm 1.1$ MeV/fiss [1] we can obtain the intensity of delayed photons I_{γ} or the mean energy $\bar{E}_{\gamma}$ of delayed photons. These values were used to normalize the calculated heating energy in

IPS1 for the case with the external source of ‘delayed photons’ in MCNP code. The calculated heating energy in the IPS1 as was estimated is equal to 20.4-22.1 kW at the nominal power of BR2 reactor. The comparison of the calculated power with the direct measurements became possible when measurements of the total energy induced by gammas in the IPS1 channel were performed by the thermal balance method. The measured energy was determined from several sets of measurements and equal to 20.7-21.2 kW. The difference between the calculated and measured gamma deposition power in the channel is less than 6%.

The total heating energy in the IPS1 with OMICO MOX rods in 7 irradiation cycles was calculated taking into account the axial profile of the fuel burn-up in all 16 rods. Comparison of power calculations with the on-line measured power in the IPS1 in each cycle was possible due to BIDASS system in BR2. The comparison of calculated and measured total power is included in Table 1. The difference in average is less than -5%, while for the first cycle is about 12%.

Table 1. Comparison of the calculated and measured thermal power in the IPS1 channel containing 16 MOX rods.

cycle	Time	BR2 reactor	IPS1 channel		
			Calculated power (C), kW	Measured power (M), thermal balance, kW	Difference (1-C/M), %
1	BOC	46	80	91	-12
	EOC	52	90	103	-13
2	BOC	61	80	86	-7
	EOC	60	89	89	+0
3	BOC	56	73	77	-5
	EOC	56	82	81	+2
4	BOC	57	72	74	-3
	EOC	57	77	82	-6
5	BOC	60	71	70	+2
	EOC	60	76	73	+4
6	BOC	58	36	38	-4
	EOC	58	37	35	+6
7	BOC	58	50	52	-4
	EOC	58	47	53	-11
mean					-4

4. Conclusion

In the present paper a simple approach for the calculation of detailed distribution of fuel burn-up was applied to bundles of different MOX fuel rods. The calculated power in the irradiation channel containing MOX rods was compared with the results of on-line measurements of the total power. The accuracy of calculations for most irradiation cycles is less than -10% and in average is about -4%. Preliminary comparison of the calculated number of fission reactions in the rods after the 1st irradiation cycle has revealed a small systematic deviation from the measured values.

5. References

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DETERMINING MTR RIA LIMITS USING EXPERIMENTAL DATA

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ABSTRACT

Reactivity initiated accidents (RIAs) are a category of events required for research reactor safety analysis. A subset of this is unprotected RIAs, in which mechanical systems or human intervention are not credited in the response of the system.

MTR-type reactors are self-limiting up to a reactivity insertion limit beyond which fuel damage occurs. This characteristic was studied in the BORAX and SPERT full-scale reactor tests. This paper describes a parametric analysis of the experimental data and a methodology for determining these limits from this data set for any MTR-type reactor.

This approach was used in the recent McMaster Nuclear Reactor (MNR) Safety Analysis Report update. A conservative step reactivity insertion limit of 11 mk was determined for the MNR LEU Reference Core, based on an irradiated-fuel-blistering safety criterion. An associated stability limit of 21 mk was also estimated.

1. Introduction

Reactivity initiated accidents (RIAs) are a category of events required for research reactor safety analysis. A subset of this is unprotected RIAs, in which mechanical systems or human intervention are not credited in the response of the system.

MTR-type (*i.e.*, light-water cooled and moderated, plate fuel) reactors are strongly self-limiting up to a reactivity insertion limit beyond which fuel damage occurs. This characteristic was studied in the BORAX (Boiling Water Reactor Experiments) and SPERT (Special Power Excursion Reactor Tests) reactor tests of the 1950s and 1960s in the USA for HEU cores and was found to be effective, reliable, and highly predictable [1,2]. Low enrichment oxide cores were also studied which qualitatively demonstrate the additional effect of fuel temperature feedback [3]. Together, the BORAX-I and SPERT MTR-type cores represent a range of various system parameters including: core size, fuel plate spacing and loading, operating temperature, pressure, and coolant flow.

Power and temperature transients were studied for both step (*i.e.*, fast) and ramp (*i.e.*, slow) reactivity insertions of varying magnitudes including the range associated with fuel damage and up to the point of core disassembly. In terms of safety analysis, examples of fast reactivity insertion accidents are expulsion or fast removal of absorber rods, fast sample movement, or fuel drop during fuelling operations. Examples of slow reactivity insertion accidents are the start-up transient (uncontrolled motor withdrawal of absorber rods), slow sample movement, or a leaking irradiation vessel.

This paper presents a methodology for determining RIA limits for any MTR-type reactor from this experimental data. The safety criterion adopted in this study is the onset of fuel damage associated with a fuel cladding temperature. This is considered a safety limit since breaching the fuel plate cladding represents a compromise of the first level of containment for radiation release. Limits are found by determining the maximum reactivity insertion which does not achieve or exceed this temperature.

The approach presented herein uses correlations in the reactor test data accounting for differences in important system parameters. A semi-empirical approach is used to quantify parametric dependencies

on core size, power distribution, void coefficient, and initial degree of subcooling. An extension is also presented with respect to LEU fuel.

This approach provides an extension to PSA analysis of events with probabilities of occurrence of less than one in one million years (typical PSA cutoff for analysis). It also presents an alternative to simulation-based studies where quantitative accuracy is lacking due to modelling limitations [4]. These limitations appear mainly in transient boiling and hydraulic behaviour which limits the accuracy of such studies to that of the initial power pulse. The peak fuel temperatures occur in the post-power-burst stage of the transient and depend on hydraulic aspects of the reactor core. Insight is also provided into further validation and improved use of simulation tools which have been benchmarked against SPERT HEU data for the initial stage of the transient response.

This work further quantifies and extends previous use of experimental data in the context of research reactor safety analysis [5-9]. It has been incorporated in the most recent version of the McMaster Nuclear Reactor (MNR) Safety Analysis Report [10].

2. Transient Characteristics

A stylized self-limited power excursion for an HEU MTR-type reactor core is shown in Figure 1.

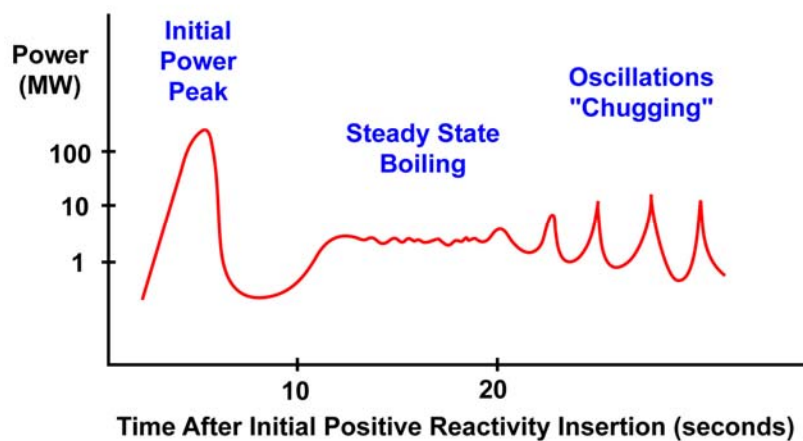


Figure 1: Self-Limited MTR-type Reactor Power Excursion

A convenient index of the transient is the reciprocal asymptotic or minimum reactor period of the initial power burst. For reactivity insertions in the range of interest (short period range), this period is on the order of 35 msec or shorter. It is found that for an HEU MTR-type system, power initially increases exponentially until feedback mechanisms can become effective. If these are fast and large enough the power increase is arrested, returning the system to a new equilibrium, or semi-stable state, which can involve steady-state coolant boiling, low amplitude power oscillations, or large amplitude power oscillations (“chugging”) in which the coolant cyclically is expelled from and refills the core.

Ramp insertions of reactivity produce similar excursions to the step insertions of reactivity, but with a reduced, or absent initial power pulse. This reduction depends on the rate of reactivity insertion.

The wealth of data from the BORAX and SPERT reactor tests show that the self-limiting response of an MTR-type reactor is highly predictable and consistent for a wide range of system parameters such as plate spacing, fuel loading, and core size. This is illustrated in the correlated data plots of maximum power (P_{max}), energy generated to the time of maximum power (E_{tm}), and the maximum fuel plate surface temperature rise (ΔT_{max}) for the different test cores as functions of the transient reciprocal period (α_o). The parameters P_{max} , E_{tm} , and ΔT_{max} are all indicators of proximity and approach to the onset of fuel damage. The maximum power data for the BORAX-I and SPERT HEU test cores is shown in Figure 2.

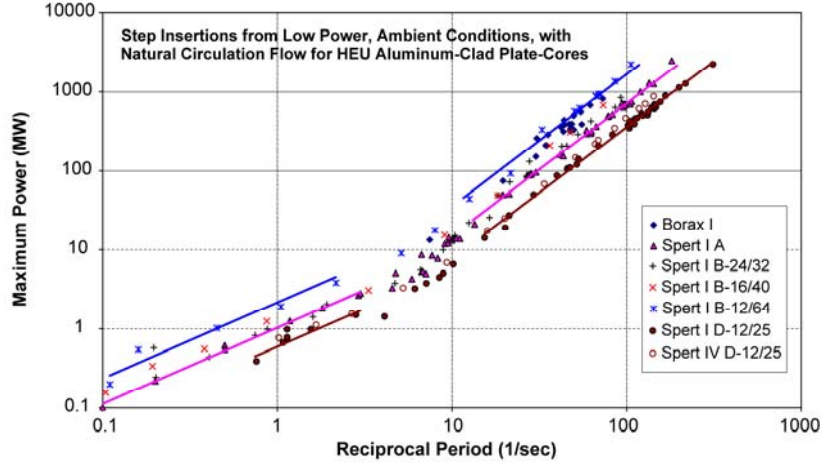


Figure 2: Peak Power Experimental Data

For an HEU MTR-type core the self-limiting response in the short period range is governed primarily by coolant voiding producing negative reactivity feedback [11]. In this sense the self-limiting response depends on the voiding characteristics of the core which in turn depend on nuclear characteristics (*e.g.*, void coefficient), heat transfer characteristics (*e.g.*, thermal resistance and heat transport) of fuel and coolant, and initial conditions of the system (*e.g.*, pressure, temperature). For LEU fuel the self-limiting behaviour is further strengthened by negative fuel temperature feedback.

Various degrees of fuel damage have been observed for periods less than 10 msec ($\alpha_o > 100 \text{ sec}^{-1}$). These various types and degrees of fuel damage are directly tied to maximum fuel plate temperatures and progress in severity with shortening reactor period and increasing maximum temperature.

3. Data Analysis

3.1 Description of the Data

The step-insertion power and energy response of an HEU MTR-type core is described well by the simple lumped parameter analytical model [12,13]:

$$\frac{P'(t)}{P(t)} = \alpha_o - w[E(t-t_d)]^n \quad (1)$$

Where $P(t)$ is the power at time t , $E(t-t_d)$ is the energy generated to time $(t-t_d)$, t_d is a delay time for thermal feedback, α_o is the reactor period, w is a shutdown reactivity coefficient, and n is the exponent of the energy dependence. The prime superscript indicates the derivative with respect to time. This is referred to as the Shutdown Model and provides the following expression for the peak power in the initial power burst:

$$P_{max} = \frac{\alpha_o^{(n+1)/n}}{w^{1/n}} e^{(\alpha_o t_d - 1/n)} \quad (2)$$

The product $(\alpha_o t_d)$ is found to be approximately constant over the short period range of transients [14]. A similar expression is found for the energy generation to the time of peak power during a transient.

Curve fitting was performed on the P_{max} , E_{tm} , and ΔT_{max} vs. α_o data using the above expressions and an exponential function relating the maximum temperature change and the reciprocal period, *i.e.*,

$$\begin{aligned}
P_{max} &= b_1 \alpha_o^{m_1} \\
E_{tm} &= b_2 \alpha_o^{m_2} \\
\Delta T_{max} &= b_3 e^{\alpha_o m_3}
\end{aligned} \tag{3}$$

The uncertainties in the test data have been conservatively estimated at 5% for the power, energy and temperature measurements and 10% for the period from information in the original technical reports. Details of this assessment are found in Reference [15].

3.2 Variation with Core Size

It is found that differences in different sets of the test data can be expressed in terms of differences in the system feedback parameter. The trends in the power and energy data are found to be consistent in the maximum temperature change data when the power and energy are related to the maximum temperature change in terms of peak power and energy density, *i.e.*,

$$\begin{aligned}
\Delta T_{max} &\propto P_{max}/V_f \times PPF \\
&\propto E_{tm}/V_f \times PPF
\end{aligned} \tag{4}$$

V_f is the fuel meat volume of the core, and PPF is the overall power peaking factor. This relation allows transformations applied to power and energy data to be similarly applied to temperature test data provided core size and power distribution differences are taken into account.

3.3 Variation with Void Coefficient

The test data from ambient initial temperature conditions was correlated to a shutdown coefficient expressed in terms of the uniform void coefficient of the core and the average channel volume for each test core. It takes the form:

$$w = K_c \left(\frac{-C_{void} V_c}{\ell} \right) \tag{5}$$

w is the shutdown coefficient in equation (1), K_c is a constant of proportionality, C_{void} is the uniform void coefficient of reactivity (in reactivity per unit void volume), V_c is the volume of a representative coolant channel within the void distribution, and ℓ is the prompt neutron lifetime. This relation was derived heuristically, based on the concept of boundary layer voiding of the coolant, fast full-channel expansion of the steam to force the remaining coolant from the core, and uniform core voiding. It differs from earlier work based on only part of the data set which did not include the volume component [16] and was found to not be suitable for the entire set of test cores. The scaling then takes the form:

$$\begin{aligned}
P_{max,j} &= \left(\frac{w_j}{w_i} \right)^v P_{max,i} \\
v &= 0.726 \pm 0.063
\end{aligned} \tag{6}$$

The parameter v has been found empirically from fitting the test data. The power test data is shown in Figure 3 both before and after scaling with the void-shutdown coefficient.

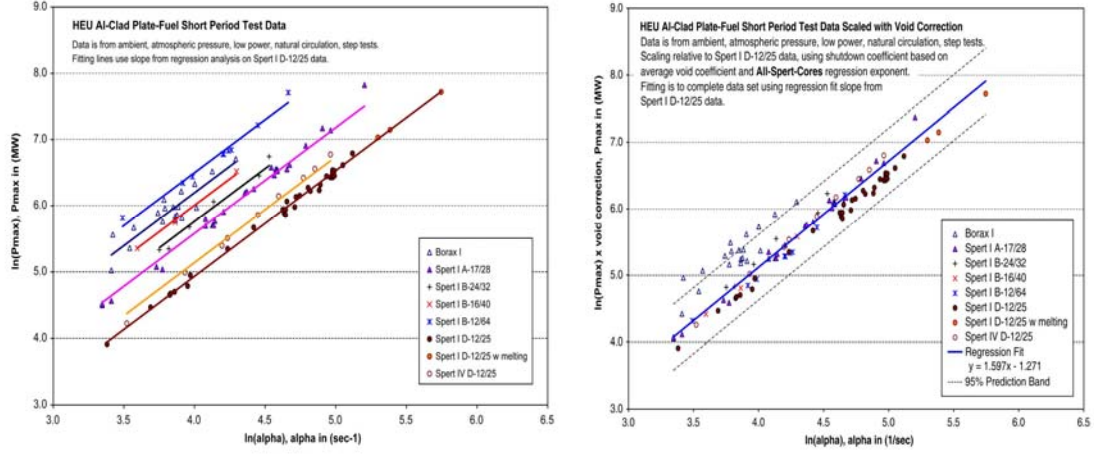


Figure 3: Power Test Data Before and After Scaling with the Void-Shutdown Coefficient

This scaling was performed using measured void coefficients for the SPERT cores and a simulation-based value for the BORAX-I core. Values of V_c and ℓ were taken from the literature and are summarized in Reference [15]. The same scaling is applicable to the energy and temperature data provided the latter is also scaled to core size and power distribution.

3.4 Variation with Subcooling

Subcooling is defined as the difference in temperature between the saturation temperature of the coolant and the actual temperature of the coolant, *i.e.*,

$$T_{sub} \equiv T_{saturation}^{coolant} - T_{initial}^{coolant} \quad (7)$$

A larger degree of subcooling means a lower initial coolant temperature. As subcooling is increased more energy must be transferred from the fuel to the coolant, translating into a longer delay time before the self-limiting void feedback can take effect. The result is larger energy generation, peak power, and fuel plate temperature rises. The BORAX-I subcooling test data were correlated and provide a relation between change in subcooling and change in maximum temperature rise:

$$\Delta T_{max}(T_{sub,i}) = \frac{(1 + s \times T_{sub,i})}{(1 + s \times T_{sub,j})} \times \Delta T_{max}(T_{sub,j}) \quad (8)$$

$$s = 0.0424 \pm 0.0071 \quad / ^\circ C$$

This relation is consistent with trends observed in the SPERT subcooling test series and is an extension to work previously reported on the BORAX-I data [17].

3.5 Variation with Doppler Coefficient

As demonstrated by the tests on LEU oxide rod fuel as part of the SPERT Project the large magnitude and prompt in nature Doppler feedback associated with LEU fuel provides a second major contribution to the self-limiting nature of a reactor core. This characteristic has been studied previously using the PARET kinetics code and the IAEA 10MW Benchmark Reactor problem [4,18]. The ratio of limiting step insertion of reactivity from these simulation results is found to be 1.18 for an instantaneous reactivity insertion, increasing significantly to 2.29 when the duration of the reactivity insertion is lengthened to 0.5 seconds. This provides a first estimate for an extension of the HEU experimental data to LEU fuel.

4. Reactivity Limits

The resulting methodology incorporates the parametric dependencies outlined above. This involves scaling the experimental test data to account for differences in system parameters between the test cores and a generic MTR-type core of interest. The methodology is based on relations found from the step-insertion test data but can be applied to both ramp-insertions of reactivity and also to the longer-term stability limits for the system based on equivalence arguments. Stability limits can be found by the additional credit of the temperature defect from initial conditions to those associated with steady state boiling. Ramp insertion limits can be found by equating the minimum period produced in the ramp-initiated transient to the step-insertion asymptotic period. The steps in the methodology are as follows:

- Start with BORAX/SPERT ΔT_{max} vs. α_o test data
- Scale for difference in subcooling
- Scale for difference in core size
- Scale for difference in power peaking
- Scale for difference in void coefficient
- Convert limiting period to limiting reactivity
- Scale for application to LEU fuel
- Adjust to associated ramp insertion or stability limit

Mathematically the HEU data scaling is expressed as:

$$\Delta T_{max}^{TC} = \Delta T_{max}^i \times \frac{V_f^i}{V_f^{TC}} \times \frac{PPF^{TC}}{PPF^i} \times \left(\frac{w^i}{w^{TC}} \right)^v \times \frac{(1 + s \times T_{sub}^{TC})}{(1 + s \times T_{sub}^i)} \quad (9)$$

The superscripts TC and i refer to the test core and the system of interest respectively. For stability limits the system of interest temperature coefficient of reactivity is required while for application to ramp-reactivity insertions the equivalence relationship is needed. Delayed neutron characteristics are also required to convert period to reactivity. The entire methodology may also be applied in reverse to determine maximum power, energy generation to peak power and maximum fuel temperature rise from limiting reactivity values.

This analysis approach was used in the recent MNR Safety Analysis Report update [10]. A conservative step reactivity insertion limit of 11 mk was determined for the MNR LEU Reference Core, based on an irradiated-fuel-blistering safety criterion. An associated stability limit of 21 mk was also estimated. The data scaling and associated system conditions have at all times been kept conservative in the analysis to account for both uncertainties in the test data and in the statistics and empirical parameter factors.

5. Conclusions

A quantitative methodology for determining RIA limits for unprotected transients from the HEU experimental data has been developed. For HEU fuel the primary parameters are found to be core size, power distribution, channel-volume-based void reactivity feedback, and initial subcooling of the reactor. Extensions to LEU fuel are included and based on existing simulation results.

Step insertion and stability reactivity limits are estimated for the MNR LEU Reference Core as 11 mk and 21 mk respectively and has been incorporated into the MNR safety case. Further details of the analysis can be found in Reference [15].

This analysis approach provides extensions to PSA methods for rare events and an alternative to simulation-based studies which are limited in their accuracy. Information gained via this study is also valuable in conjunction with simulation work.

6. References

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SAFETY ANALYSIS OF RESEARCH REACTORS WITH BEST ESTIMATE COMPUTATIONAL TOOLS

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ABSTRACT

Best Estimate computer codes have been, so far, developed for safety analysis of nuclear power plants and were extensively validated against a large set of separate effects and integral test facilities experimental data relevant to such kind of reactors. With the sustained development in computer technology, the possibilities of code capabilities have been enlarged substantially. Consequently, advanced safety evaluations and design optimizations that were not possible a few years ago can now be performed. According to the IAEA Research Reactor Database (RRDB) 651 research reactors have been constructed around the world for civilian applications. On the basis of the RRDB, 284 research reactors are currently in operation, 258 are shut down and 109 have been decommissioned. The purpose of the present paper is to provide an overview of the accident analysis technology applied to the research reactor, with emphasis given to the capabilities of computational tools.

1. Introduction

An established international expertise in relation to computational tools, procedures for their application including best-estimate methods supported by uncertainty evaluation and comprehensive experimental database exists within the safety technology of NPP. The importance of transferring NPP safety technology tools and methods to research reactor (RR) safety technology has been noted in recent IAEA activities. However, the ranges of parameters of interest to RR are different from those for NPP: this is namely true for fuel composition, system pressure, adopted materials and overall system geometric configuration. The large variety of research reactors prevented so far the achievement of systematic and detailed lists of initiating events based upon qualified PSA (Probabilistic Safety Assessment) studies with results endorsed by the international community. However, bounding and generalized lists of events are available from IAEA documents and can be considered for deeper studies in the area.

An established technology exists for development, qualification and application of system thermal hydraulics codes suitable to be adopted for accident analysis in research reactors. This derives from NPP technology. The applicability of system codes like RELAP5, COBRA and MARS to the research reactor needs has been confirmed from recent IAEA. Definitely, system codes are mature for application to transient analysis in research reactors. However, code limitations have been found in predicting pressure drops as a function of mass flux at low values of mass flux when nucleate boiling occurs. The importance of the Whittle & Forgan experiments shall be mentioned, as well as the dependence of results from the nodding (cell subdivision) adopted by the code users. Several code user choices, including time step may have a significant effect upon prediction, thus confirming the need for detailed code user guidelines. Furthermore, code validation must be demonstrated for the range of parameters of interest to research reactors.

The crucial role of uncertainty in research reactor technology has been emphasized, a) for the design, with main reference to the prediction of the nominal steady-state conditions and, b) for the safety, with main reference to the prediction of the time evolution of significant safety parameters. It has been found that suitable-mature methods exist, but the spread of these methods and procedures within the community of scientists working in research reactor technology is limited.

2. Topics of interest for accident analysis in RR and current status

A list of topics relevant to the deterministic accident analysis in research reactors is provided below.

- *Postulated Initiating Events*. The identification of accident scenarios, typically derived by considering probability of occurrences and severity of consequences, constitutes the first step needed for performing deterministic safety analyses.

- *Acceptance criteria*. The availability of ‘thresholds of acceptability’ for consequences of accident, as a function of the probability of occurrence of the event, constitute the second requirement for deterministic safety analyses: namely results of the analyses shall be compared with ‘limiting values’. Acceptance criteria are imposed by national authorities and are not connected with the deterministic safety analysis.

- *Experimental database*. Computational tools are used to perform deterministic analyses (see below) and experimental data are needed to demonstrate the quality of those tools. Experimental data can also be used directly to improve the design and the performance of research reactors.

- *Qualification of system codes*. System codes, widely used within the safety technology of Nuclear Power Plants (NPP), can be used for the deterministic analysis of accidents in Research Reactors (RR). The application must be based upon the evaluation of the complexity of the transient: in a number of cases owing to the ‘simple’ configuration of RR compared with NPP, simpler tools including analytical-hand calculations should be used. However, for the cases when system codes are needed, proper demonstrations of qualification must be provided.

- *Uncertainty in research reactor technology*. The role of uncertainty has been considered at two levels: a) design of research reactors, mostly addressed to the calculation of nominal steady state operating conditions, and b) evaluating the results from best-estimated predictions performed by thermal-hydraulic system codes, mostly addressed to the calculation of transient scenarios. Origins and impacts of uncertainty within both the frameworks are discussed in [1]. Methods and procedures to deal with uncertainty are presented in the same reference.

3. Example from Code Qualification Process

With widespread use of research reactor, there is a real need to get more realistic simulations of the phenomena involved during steady state and transient conditions, and eventually the identification of design/safety requirements that can be relaxed or enhanced. Several attempts were performed to assess the applicability of Best Estimate codes to RR operating conditions [2]. Relevant assessments were applied against the following cases.

4. The IAEA 10 MW Benchmark Problem

The IAEA Benchmark is based upon one of the SPERT series test reactors. A standard-quality RELAP5 nodalization has been developed and applied [2], [3] (Figs 1, 2). The reactor pool above the core zone is modelled in order to adequately simulate the natural convection process. The benchmark problem consists in analysing ‘controlled’ (or ‘protected’) transients in MTR Highly Enriched Uranium core (HEU) and Low Enriched Uranium (LEU) cores. The boundary conditions for the analysis are demanding from the thermal-hydraulic point of view. The Natural Convection Valve (NCV), as modelled in the RELAP5 nodalization, allows a flow reversal and the establishment of passive decay heat removal process by natural circulation flow.

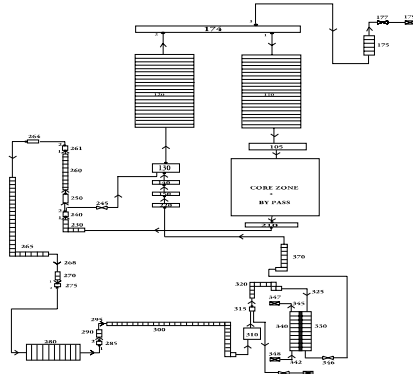


Fig 1 Nodalization for RELAP5

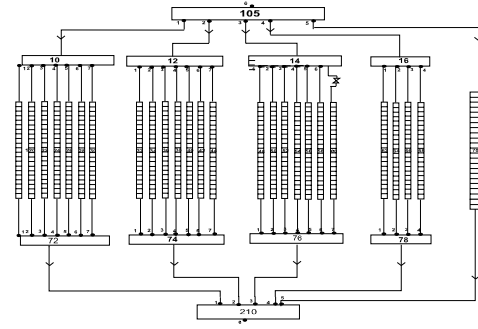


Fig 2 RELAP5 Core Nodalization

The Fast RIA (FRIA) transients are initiated by a super prompt ramp positive reactivity addition of $1.5/0.5$ s in the HEU and LEU cores. The Slow RIA (SRIA) consists in a continuous insertion of 9ϕ /s in the HEU core and 10ϕ /s in the LEU core. The reactor is assumed to be at an initial operating power of 1W and with full downward cooling flow (not as the benchmark specifications which consider initial upward flow). The safety system is activated when the core power exceeds 12 MW, by inducing a negative reactivity of -10 in 0.5 sec within a response delay time of 0.025 s. The flow decay is modelled as an exponential ($\exp(-t/T)$) decrease with a period T equal to 1 s and 25 s for the Fast LOFA (FLOFA) and the Slow LOFA (SLOFA) case, respectively. The LOFA transients are initiated at a nominal core power of 12 MW and full core downward cooling flow conditions. The reactor scrams when the flow decay is reduced by 15%, with a response delay time of 0.2 s. Representative results are given in Figs. 3 to 6, where comparison is made, when applicable, with reference results obtained by the RR devoted codes PARET and RETRAC.

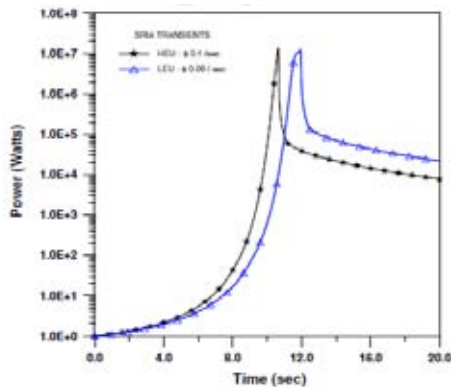


Fig 3. Core power during SRIA

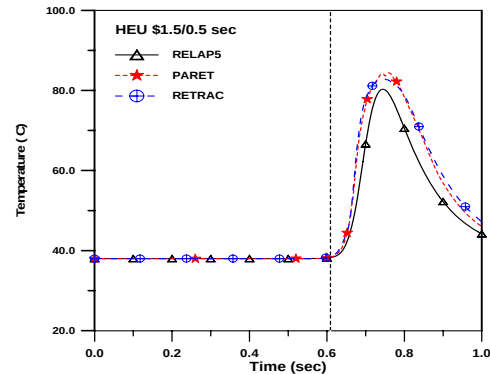


Fig 4. Outlet fluid temperature during

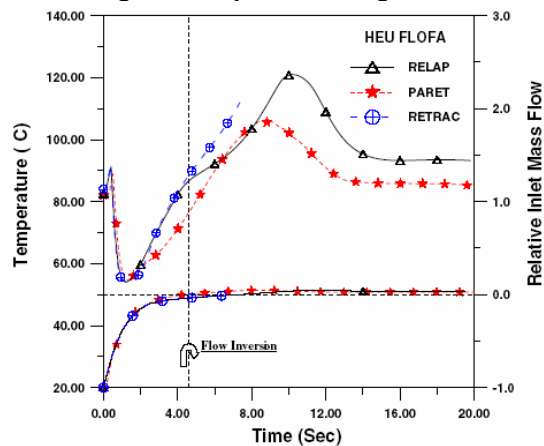


Fig 5. Clad surface temperature and relative mass flow rate during SLOFA transient

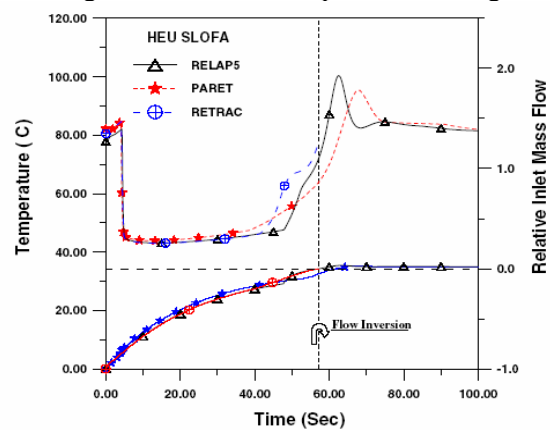


Fig 6. Clad surface temperature and relative inlet mass flow rate during FLOFA

Relevant experimental data (Whittle & Forgan experiments) and Relap5 calculation results for typical RR conditions are compared in Fig. 7, [4].

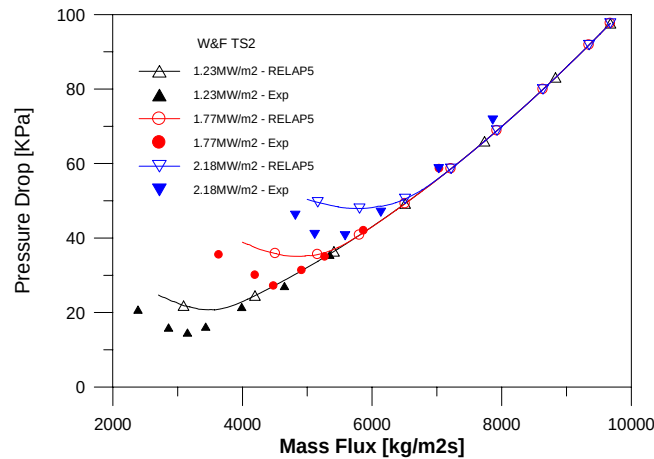


Fig 7. Pressure Drop Characteristic curve for W&F TS2

5. FRJ 2 Research Reactor

The Relap5 code has been applied to the safety evaluation of the FRJ2, 23 MW RR, installed at Juelich research centre. A global view of the reactor is given in Fig 8 [5]. The ‘oblique’ Control Rods (CR) can be observed: in case of ‘free’ CR drop, a negative insertion of reactivity is followed by a reactivity increase (Fig 9).

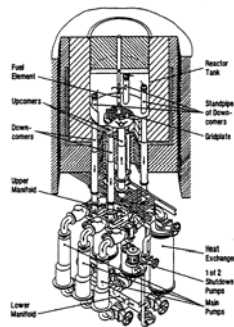


Fig 8. Global view of the cooling system of the FRJ-2

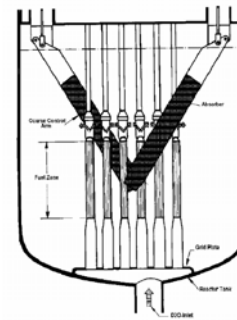


Fig 9. FRJ-2 core and CCA arrangement inside the reactor tank

A detailed Relap5 nodalisation, based on CATHENA one, has been developed including more than 500 hydraulic nodes and about 4000 meshes for conduction heat transfer, Figs 10 to 12. The considered transient is a CR drop without scram originated in the complex geometry of FRJ2 reactor. A result of the analysis by RELAP5 and CATHENA code of the RIA w/o scram in the FRJ2 reactor is given in Fig 13.

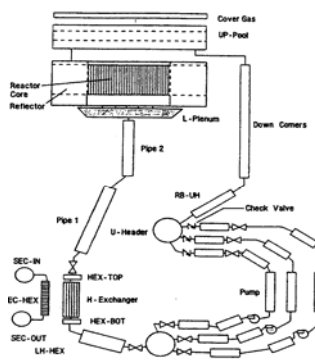


Fig 10. Nodalization for CATHENA

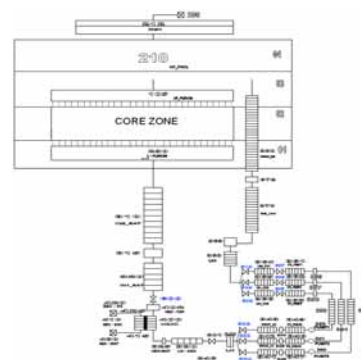


Fig 11. Nodalization for RELAP5

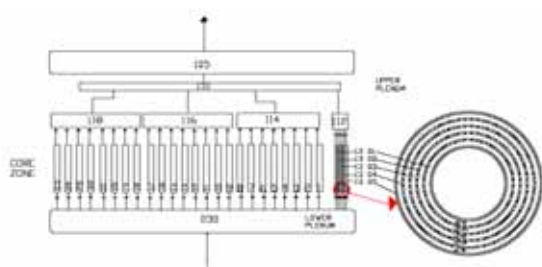


Fig 12 RELAP5 Core Nodalization

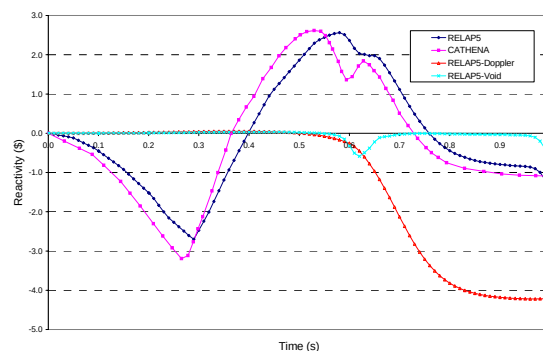


Fig 13 Core Reactivity

6. Conclusion

The demonstration of applicability of qualified best-estimate system codes to RR accident analysis constitutes the key message from this paper: a proper accident analysis technology should be developed for RR that could benefit of the experience available from NPP, considering that the risk level and the cost associated with RR are orders of magnitude lower. Recommendations are provided hereafter distinguishing between potential RR system thermal-hydraulic code users and decision makers in the area.

Recommendations to the users of computational tools are:

- To consider experimental data and to perform code-to-experiment comparison before any code application to prediction relevant to the RR design or safety analysis.
- To demonstrate that any code adopted for design and safety is qualified.
- To consider that any best-estimate code, even though supported by the use of the optimised procedures, produces results that are affected by an unknown error, i.e. uncertainty.

Recommendations to decision makers focus on establishing an international understanding in the area:

- To plan “benchmark” exercises in conditions where neutron kinetics and natural circulation are relevant.
- To promote the use of PSA techniques, establishing detailed PIE (postulated initiating events) lists.
- To make an effort to establish ‘validation-matrices’ for computational tools.
- To plan suitable training in the area of RR accident analysis.
- To consider innovative techniques including of CFD (Computational Fluid Dynamics) and coupled three-dimensional neutron kinetics codes and thermal-hydraulic system codes.

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KINETIC PARAMETERS CALCULATION AND MEASUREMENTS DURING THE OPAL COMMISSIONING

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ABSTRACT

During the Commissioning Stage of the OPAL Research Reactor (Australia) the Prompt Neutron Decay constant was measured by analysing the inherent fluctuations that occur in the neutron population. The ratio of the variance to the mean number of counts as a function of counting time is used to determine the α parameter. This technique is also called Feynman- α Method.

The CITVAP and MCNP codes were used to calculate the prompt neutron decay constant for the first core configuration. By means of two different MCNP calculations, one considering prompt fission neutrons only and another with total fission neutrons; the effective delayed neutron fraction is estimated.

The experimental method, the measured value as well as the numerical assessment are presented in this paper. A good agreement was obtained between measurements and calculations and a comparison is presented in the paper.

1. Introduction

The OPAL Research Reactor is a multi-purpose open-pool type reactor. The nominal fission power of the reactor is 20 MW. The core is located inside a chimney, surrounded by heavy water contained in the Reflector Vessel. The whole assembly is at the bottom of the Reactor Pool, which is full of demineralized light water acting as coolant and moderator and biological shielding.

Several irradiation facilities are located around the reactor core. Three types of neutron sources: a cold neutron source with two tangential beams and several neutron guides, a thermal neutron source with two beams and several neutron guides, and a space reserved for a future hot neutron source with a beam.

During the design stage, core calculations were performed to obtain the effective delayed neutron fraction (β_{eff}), the neutron lifetime (Λ) and its ratio, the prompt neutron decay constant (α), which is used in the Safety Analysis Report.

The CONDOR - CITVAP codes (references [1] and [2]) and MCNP code (reference [3]) were used to calculate the prompt neutron decay constant for the first core configuration.

Both the delayed neutron fraction and the neutron lifetime were assessed with the CITVAP code using microscopic cross-section libraries.

Carrying out an MCNP calculation with an external neutron source allows the estimation of the core neutron population as a function of time. Through the analysis of the neutron population, the prompt neutron decay constant (α) is obtained.

The effective delayed neutron fraction is estimated by means of two different MCNP calculations, one considering prompt fission neutrons only and another one with total fission neutrons.

2. Description of the calculation lines

There are two calculation lines used to estimate the kinetic parameters. The calculation lines are the CONDOR – CITVAP line and the MCNP line.

2.1. CONDOR - CITVAP calculation line

The CONDOR code for neutronic calculations is used to calculate fuel cells, fuel-rod clusters, as well as fuel plates with slab geometry or 2D geometry. Flux distribution within the region of interest is obtained through the collision probability method or the Heterogeneous Response Method in a multi-group scheme with various types of boundary conditions.

The CITVAP code used to carry out the core design of the OPAL reactor is a new version of the CITATION-II code, developed by INVAP's Nuclear Engineering Division. The code was developed to improve CITATION-II performance. In addition, programming modifications were performed for its implementation on personal computers.

The code solves 1, 2 or 3-dimensional multi-group diffusion equations in rectangular or cylindrical geometry. Spatial discretization can also be achieved with triangular or hexagonal meshes. Nuclear data can be provided as microscopic or macroscopic cross section libraries.

CITVAP performs flux and adjoint flux calculations in order to assess the prompt neutron lifetime and delayed neutron fraction. Energy spectra with a higher number of energy groups (especially in the fast group region) are used to take into account the difference between the delayed and prompt neutrons.

2.2. MCNP calculation line

The MCNP code is a well known Monte Carlo code that was used to design the irradiation facilities and to verify some neutron parameters through an independent calculation method. This is a Monte Carlo transport code for neutron and gamma calculations using ENDF/B-VI cross sections in any order and performs 3-D calculations.

3. Kinetic Parameters Calculation using MCNP

The effective delayed neutron fraction was calculated with MCNP under the following considerations: The nuclear delayed fraction is the ratio between the delayed neutrons and the total neutron generated by fission, that is:

$$\beta = \frac{\nu_d}{\nu_t} \quad [1]$$

To obtain the effective delayed fraction is necessary to weight the delayed and total neutron fractions by the number of neutrons that can be produced in the next generation.

$$\beta_{eff} = \frac{\nu_d N_d}{\nu_t N_t} \quad [2]$$

Where N_d is the number of neutrons that will be produced in the next generation by each delayed neutron.

$$N_d = \frac{\nu_t FR_d}{\nu_d} \quad [3]$$

And N_t is the number of neutrons that will be produced in the next generation by each total neutron

$$N_t = \frac{\nu_t FR_t}{\nu_t} \quad [4]$$

FR_d and FR_t are the fission rate produced by delayed neutron and total neutrons, respectively.

Replacing N_d and N_t in equation [2], the expression for β_{eff} is obtained.

$$\beta_{eff} = \frac{FR_d}{FR_t} \quad [5]$$

This equation [5] is also presented in reference [4].

The β_{eff} was also calculated as the difference in the core reactivity when total neutrons are used and when only prompt neutrons is used.

To calculate the prompt neutron decay constant with MCNP it was simulated the Rossi- α experiment, reference [5]. A full detail model of the core and reflector vessel was used to carry out the calculations with an external neutron source. The neutron population in the core (meat material) as a function of time was analysed. The prompt neutron decay constant (α) is obtained fitting the neutron population as a function of time with an exponential function.

From the α parameter and the core reactivity, the neutron lifetime can be obtained.

4. Prompt Neutron Decay Constant Measurement

During the OPAL commissioning, the α constant was measured by analysing the inherent fluctuations that occur in the neutron population. The ratio of the variance to the mean number of counts as a function of counting time is used to determine de α parameter. This technique is also called Feynman- α Method, reference [6].

According to reference [6], the equation [6] relates the variance to mean ratio $V(t)$ with the α parameter:

$$V(t) = \frac{N \sum_{i=1}^N C_i^2 - \left(\sum_{i=1}^N C_i \right)^2}{N \sum_{i=1}^N C_i} = 1 + \frac{\epsilon \chi}{(\beta_{\text{eff}} - \rho)^2} \left[1 - \frac{(1 - e^{-\alpha t})}{\alpha t} \right] \quad [6]$$

Where:

N: number of time intervals of length t analysed.

C_i : number of counts recorded during the time interval i of length t.

ϵ : absolute detector efficiency [Counts/Fission].

$\chi = \frac{\nu(\nu-1)}{\nu^2}$, where ν is the number of neutrons released during the fission. For the thermal fission of the Uranium-235 $\chi=0.795$.

β_{eff} : effective delayed neutron fraction.

ρ : core reactivity.

α : prompt neutron decay constant, $\alpha = \frac{\beta_{\text{eff}} - \rho}{\Lambda}$. When the reactor is critical, $\alpha = \frac{\beta_{\text{eff}}}{\Lambda}$.

Λ : neutron lifetime.

To obtain α , the counting of a Fission Counter (FC) detector placed in the core is recorded in files for different core reactivities (all of them lower than zero). The information recorded in each file is the time interval between two successive counts.

The recorded file is analysed numerically. The software divide the measuring time in time intervals of length t and obtain the number of counts (C) recorded during each time interval (i) of length t. For that time interval of length t, $V(t)$ is evaluated as the left side of equation [6].

$V(t)$ is evaluated for different time length from 50 μs to 0.05 s with steps of 50 μs , that is, there are 1000 evaluations of the variance to mean ratio.

Figure 1 shows the variance to mean ratio $V(t)$ for different core reactivities.

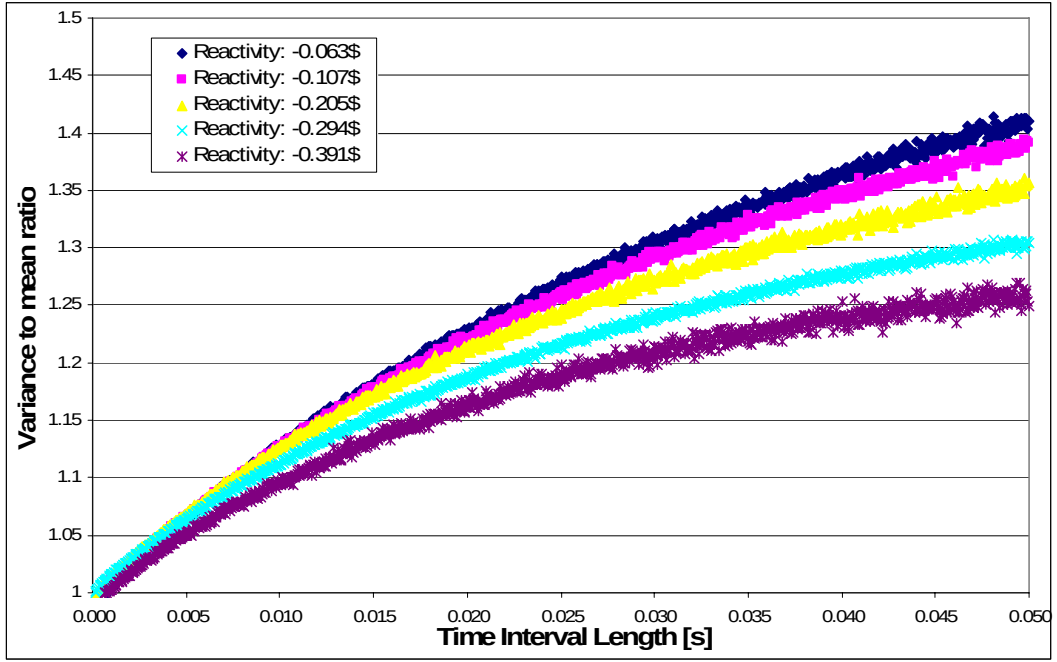


Figure 1: $V(t)$ when the time interval length vary from 50 μ s to 0.05 s with steps of 50 μ s.

By fitting $V(t)$ with the right side of equation [6] it is possible to obtain α for that core reactivity.

By repeating the FC counts recording process for different core reactivities the plot of α as a function of the core reactivity (ρ) is obtained, as shown in Figure 2.

The plot of α as function of the core reactivity (ρ) is adjusted by a linear fitting and the value of α when $\rho=0$ is $\beta_{\text{eff}}/\Lambda$.

For accurate experimental results it is essential that the detector efficiency ε be as high as possible, reference [7].

5. Results

The kinetic parameters of the OPAL Research Reactor were calculated with CITVAP and MCNP for the full core configuration, i.e., 16 Fuel Assemblies (FA).

Table 1 shows the calculated parameters β_{eff} and Λ for the 16 FA core.

	CITVAP	MCNP
β_{eff} [pcm]	768	769.5
Λ [μ s]	171	171.6
α [1/s]	44.9	44.8

Table 1: β_{eff} and Λ for the first 16 FA core.

The β_{eff} for the first 16 FA core was also calculated by MCNP as the difference in the core reactivity when total neutrons are used and when only prompt neutrons given a value of 769.6 pcm.

Due to the fact that the measurements of kinetic parameters require high absolute detector efficiency, during the Commissioning of the OPAL reactor, the neutron decay constant was measured for the 15 FA core configuration, replacing one central FA by an FC detector.

To measure the α parameter, the Feynman- α method was used. The α parameter value was measured for different subcritical levels. The $\alpha(\rho)$ function was adjusted by a linear fitting and the value of α when $\rho=0$, i.e., the $\beta_{\text{eff}}/\Lambda$ value, was obtained.

This experiment was also compared with the results obtained with MCNP simulating the Rossi- α experiment. The α parameter was obtained for different subcritical levels and adjusted by a linear fitting.

Figure 2 shows the comparison between measured and calculated values. From this figure it is worth noticing that the measured value for the α parameter, when $\rho=0$, is 38.1 1/s while the calculated value is 37.2 1/s.

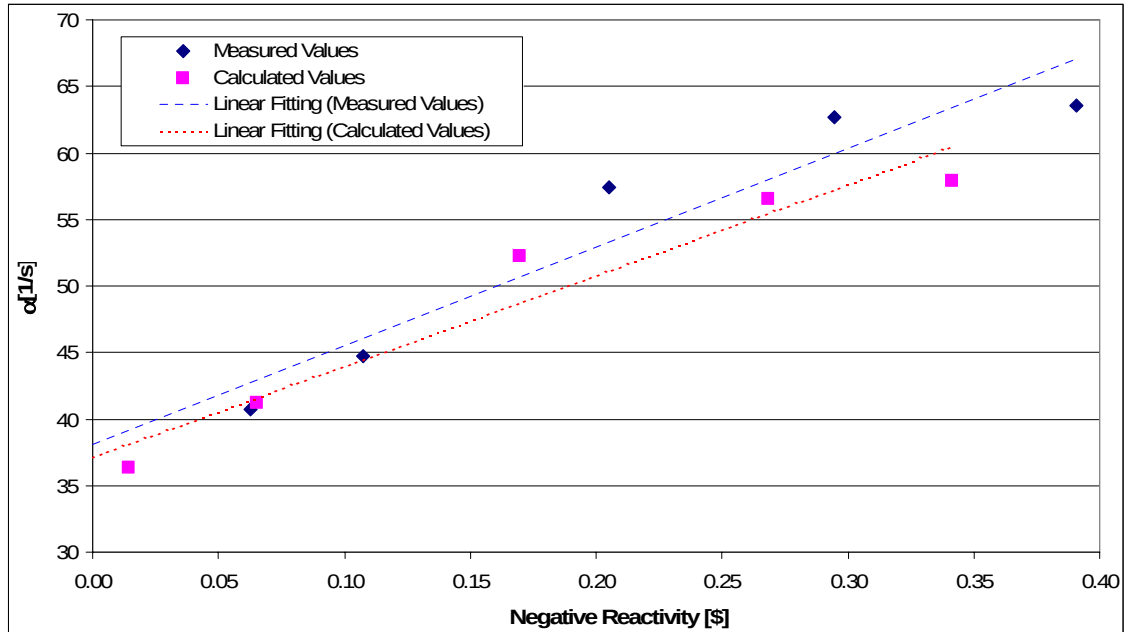


Figure 2: Parameter α as function of the core reactivity.

6. Conclusions

The MCNP code was used to obtain the kinetic parameters β_{eff} and Λ . There is a good agreement between the MCNP calculated values and the values obtained by the traditional calculation line for this parameters (CITVAP code).

The α parameter was measured with the Feynman- α method using the plant instrumentation (Fission Counter detector). There was good agreement between the measured α value and the MCNP calculated value following the Rossi- α experiment.

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DETERMINATION OF SAFARI-1 NEUTRON FLUXES BY MCNPX MODELLING OF FOIL EXPERIMENTS

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Abstract

Necsa (South African Nuclear Energy Corporation Limited) is planning an experiment to test the performance of the Pebble Bed Modular Reactor (PBMR) fuel particles at specified burn-up conditions by placing it in the SAFARI-1 core. In order to simulate the irradiations required to reach these burnups, accurate neutron fluxes are needed. To verify the SAFARI-1 core model, which is used for flux calculations, a previous cobalt and nickel foil irradiation experiment was modelled using the Monte Carlo transport code MCNPX. Neutron fluxes and reaction rates were calculated and compared with measured activity values. Results are shown and discussed.

1. Introduction

Necsa's reactor, SAFARI-1, is used for material testing applications and for the production of radioisotopes. The Radiation and Reactor Theory Group gives support to the reactor through modelling of different applications to determine radiation safety safeguards or engineering requirements. One such an application is the planned experiment to test the performance of the Pebble Bed Modular Reactor (PBMR) fuel particles at specified burn-up conditions attained by irradiating them in the SAFARI-1 core.

In order to reach the required burn-ups it is vital to know the relevant neutron fluxes and the gamma heating in the irradiation rig that will be used. Due to restrictions in the design of the irradiation rig, it is not possible to determine these parameters through measurement inside the rig. By modelling the experiment, neutron fluxes and gamma heating can be obtained both inside the irradiation rig and at locations outside, where measurements are possible. Thus the outside values can be compared to measurements and hence inside values can be estimated with greater accuracy.

In preparation for this experiment a geometrically detailed MCNPX [1] model of the SAFARI-1 core was developed that is capable of representing every single moment in a reactor cycle in terms of isotopic inventory. An interface code, OSMINT [2], manages the transfer of material data from the 3D nodal depletion code OSCAR-3 [3] to an MCNPX input template. The resulting model greatly improves previous approximations where the core was modelled as a homogenised mixture of uranium, water and aluminium [4].

The objective of this work is to verify the applicability of this SAFARI-1 core model. For the verification, an earlier cobalt and nickel foil irradiation experiment was modelled using the MCNPX core model, neutron fluxes and activities calculated and compared with the measurements.

2. Modelling with MCNPX

An MCNPX input deck of the reactor at a thermal power of 20 MW was constructed, using the OSCAR-3 program and the OSMINT interface program, representing the status of the core at the time of the foil irradiations. The exact geometry of the irradiation rig (foils and the foil holders) was obtained from engineering drawings and defined as a separate object inside the core (see Fig 1.).

The calculation was run as a KCODE source problem for criticality calculations on MCNPX v2.5. The ENDF VI (60c.) cross-section data set was used for all the radioisotopes when available; when absent, the alternate older ENDF V (50c) was used. Using 25000 k-effective cycles, the foils were tallied with F4 tallies for neutron flux and reaction rate.

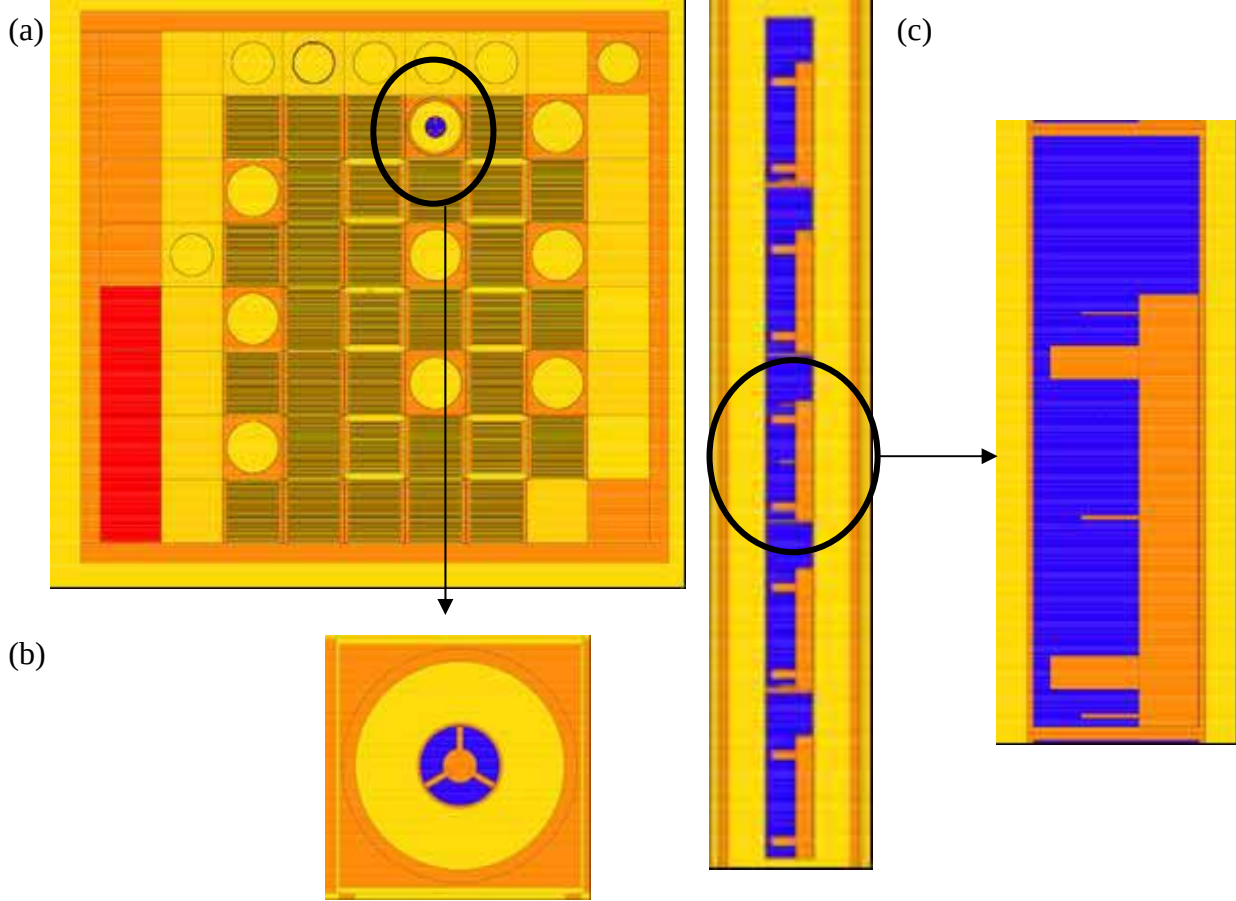


Fig 1. An illustration of the SAFARI-1 core model, with (a) the flux monitoring irradiation rig inside, (b) close up of the rig, (c) vertical view of the irradiation rig that consists of 5 foil holders and (d) close up of one foil holder with three foils.

3. Results and Discussion

Using the calculated reaction rates and fluxes, the activities (A_i) of all the foils were determined by

$$A_i = n\sigma\phi(1 - e^{-\lambda t}) , \quad 1.$$

with n the number of atoms, σ the absorption cross section, ϕ the neutron flux and the bracket term a decay correction factor. These and the measured values are tabulated in Table 1.

Good agreement is seen between the measured and calculational data sets (a graphical comparison between the sets of data is depicted in Fig 2 and Fig 3). For the nickel foils only two points do not show agreement. It is unsure whether it can be attributed to statistics or to modelling. However, a possibility for errors occurs when doing modelling, as only a snapshot of the actual experiment is investigated and not the total experiment. The core depletion process can therefore have an effect on the results. From the results it is concluded that the core model shows promise as a tool to aid the PBMR irradiation experiment.

Position from core centre line	Co foil activity (Bq)		Ni foil activity (Bq)	
	Measured	Calculated	Measured	Calculated
219.9	7.86E+02	8.88E+02	2.86E+05	2.75E+05
188.9	9.32E+02	1.02E+03	3.54E+05	3.71E+05
158.9	1.13E+03	1.18E+03	4.18E+05	4.41E+05
128.4	1.35E+03	1.45E+03	4.84E+05	4.98E+05
97.4	1.57E+03	1.76E+03	5.97E+05	6.57E+05
67.4	1.75E+03	1.94E+03	6.17E+05	6.11E+05
36.9	1.80E+03	2.14E+03	7.52E+05	7.48E+05
6	1.91E+03	2.19E+03	8.12E+05	6.46E+05
-24.1	2.13E+03	2.31E+03	7.96E+05	7.89E+05
-54.6	2.20E+03	2.35E+03	8.37E+05	7.98E+05
-85.6	2.24E+03	2.33E+03	8.35E+05	9.09E+05
-115.6	2.18E+03	2.24E+03	8.10E+05	8.41E+05
-146.1	2.07E+03	2.25E+03	7.96E+05	7.74E+05
-177.1	1.91E+03	1.98E+03	7.48E+05	5.66E+05
-207.1	1.86E+03	2.05E+03	6.56E+05	6.19E+05

Table 1: The measured and calculated activities for the cobalt and nickel foils per position in the SAFARI-1 core. The measurement error and the statistical error are both 10%.

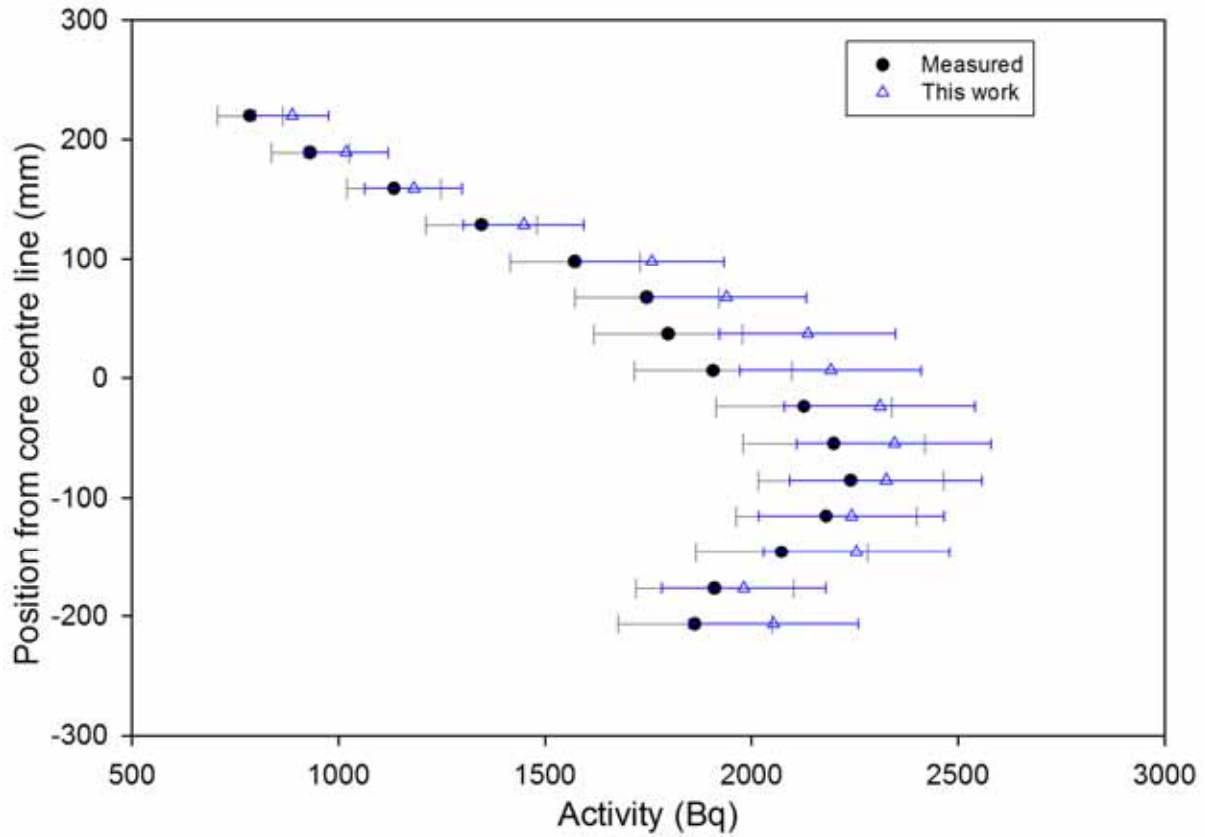


Fig 2. The measured activities compared with the calculated activities of this work for the cobalt foils as a function of position in the SAFARI-1 core.

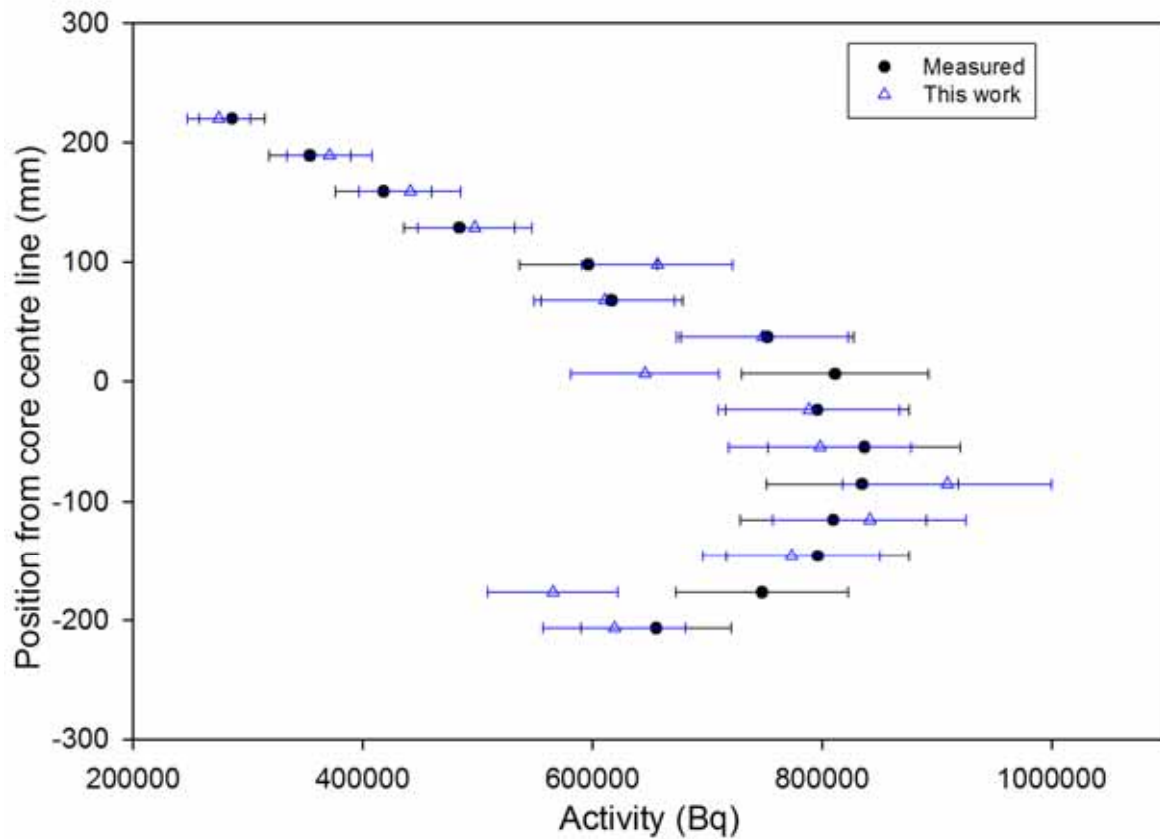


Fig 3. The measured activities compared with the calculated activities of this work for the nickel foils as a function of position in the SAFARI-1 core.

4. References

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SOPHISTICATED MCNP CALCULATION OF THE FLUX MAP OF FRJ-2 USING A FULLY NODALIZED MODEL

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ABSTRACT

After 44 years of operation for material research, the FRJ-2 research reactor was finally shut down on May, 2nd 2006. The running decommissioning activities which are realized according to a well defined plan, intend to place the facility in a condition that provides for the safety of the general public, decommissioning staff and the environment.

In this respect the amount of radioactive waste and radio nuclide inventory are of particular interest which are calculated on the basis of neutron flux. For the determination of the flux map the Monte-Carlo-Code MCNP was employed due to its modeling capability for high performance computing. The requirement on detailed nodalization and precise neutronic calculations results from the complex reactor construction consisting of different structures absorber layers, penetrations, beam tubes and holes. The work shows that it is possible to determine the flux distribution with a high statistical precision even for large reactor systems. Accordingly the thermal n-flux depends on the location of the components as well as on the geometry of the surrounding structures. The average thermal flux in the core, the graphite reflector, steel tank and in the inner and outer layer of the biological shield amounts to 2.0×10^{14} , 1.7×10^{13} , 3.2×10^{10} , 8.7×10^7 and 6.0×10^5 n/cm²s respectively. Due to the streaming and scattering effects the flux around the openings is significantly higher than in the undisturbed region of the surrounding structures. The results of the calculations were produced with a reasonable computing time using the high performance computer system JUMP by employing variance reduction method.

1. Introduction

For the solution of neutron transport equation in complex systems different numerical methods are employed which are of deterministic or statistical character. In the first category mainly using the discrete ordinate method or diffusion approximation the numerical solution is performed by the definition of finite differences for a discretized geometry. These methods are associated with shortcomings in the representation of real geometry that may result in some uncertainties. Additional numerical uncertainties and errors may result from the use of the condensed nuclear data in few energy groups [1]. By comparison, the Monte-Carlo codes enable the user of providing the adequate flexibility in the numerical representation of complex geometries and energy domain.

Due to the complex geometry of FRJ-2 consisting of inhomogeneous structures and configuration the MCNP Monte-Carlo-code seemed to be the optimum method for precise neutronic analysis. The MCNP code is also extensively used worldwide in nuclear engineering to perform complex criticality studies and neutron and particle transport calculations [2]. It is capable of treating any 3-dimensional configuration of materials in geometric cells of complex form using pointwise continuous-energy cross sections existing for a variety of reactions [3]. The numerical models and features of the code have been extensively validated on the basis of comprehensive benchmark tests and experiments [4].

The decision for the application of MCNP was made due to the fact that a geometrical model of the reactor was existing from the sophisticated core physics investigations [5,6]. Due to the large dimension of the reactor block and limited penetration length, the model was modified by homogenizing the core region and detailed nodalization of the outer regions of the reactor. Variance reduction technique using cell importance map was applied in order to decrease uncertainties in the geometrical regions of the model, where only few neutron tracks are sampled due to large distances and shielding effects.

2. Description of FRJ-2

The FRJ-2 is a DIDO-class tank-type research reactor cooled and moderated by heavy water. The core consists of 25 tubular MTR fuel elements arranged in five rows of fuel elements. The active part of the tubular fuel elements is formed by four concentric tubes having a wall thickness of 1.5 mm and a length of 0.61 m. The reactor has been equipped with two independent and diverse shut-down systems the CCAs and the RSRs. The six CCAs are raised and lowered by angular movement around a pivot, whereas the three RSRs are shot in by pneumatic actuators.

The core is accommodated within an aluminium tank 2 m in diameter and 3.2 m in height. The tank is surrounded by a graphite reflector of 0.6 m thickness enclosed within a double-walled steel tank surrounding a lead zone as the thermal shielding with a thickness of 10 cm in which cooling tubes are installed for cooling purposes. Outside of this structure a boral layer is used to prevent neutron leakage and protect the surrounding concrete shielding (biological shield). The biological shield is a concrete blanket between the steel tank and the outer casing of the reactor. Three different kinds of heavy concrete were used in the construction. The first layer surrounding the boral layer is made of baryt concrete with boron additive.

The vertical shield of the aluminum tank and the graphite reflector is made of three layers of different thicknesses and materials namely stainless steel (1.9 cm), cadmium layer, lead (10 cm), a mild steel layer (3.8 cm) and a massive iron shot concrete in the top having a height of 70 cm. The whole tank shield has couple of holes foreseen for loading of fuel elements, which are tightened during the operation by using plugs of the same construction like the surrounding structures. Accordingly the whole reactor contains all round an absorber layer (boral) used between the thermal and biological shield. For experimental purposes a part of the boral layer and outer biological shield has been replaced by a thermal column consisting of the graphite in the peripheral zone and of steel and graphite structures at the outer region.

Due to the design of the reactor for neutron beam experiments, a high number of vertical and horizontal holes and channels have been provided in all structures and parts which penetrate from outer surfaces of the reactor block.

3. MCNP Model

The precision of the MCNP calculations is mainly determined by the modeling details of the geometrical material cells and by the uncertainties of the nuclear data as well as by the number of histories in the simulation run. The MCNP Model of the FRJ-2 is a complete 3-dimensional full-scale model with a high level of geometric fidelity representing a detailed geometrical nodalization like shown in Fig 1. Because of high interest in neutron flux in the structures outside the reactor aluminum tank the whole active core consisting of the fuel elements, part of the CCA and heavy water has been homogenized and represented as a unitary cell. This approach for the central core was applied to limit the modeling effort and achieve the necessary precision in the neutron flux calculations in the outer regions within a reasonable computing time.

For the homogenization aim, the material composition in the individual fuel elements was distributed in a cylindrical cell of the core size in accordance with the fraction of individual volumes. By this way the density of all nuclide existing in the core were determined. The core as a homogenized cell 41 cm in diameter and 60 cm high is positioned in the center of the reactor model. A second cell was generated for the zone above the core in which all aluminum tubes (for holding the fuel elements) as well as the upper structures of the CCA are homogenized.

The lower region of the core down to the bottom of the tank accommodating the grid plate, unfueled ends of the fuel elements including the aluminium structures were modelled in accordance with the design. In the whole geometric model, the cell boundaries were specified by 1st and 2nd degree surfaces with appropriate transformation in accordance with the position of the cells. The beam tubes of various diameters and lengths were modelled in detail and integrated into the entire model in accordance with the design and construction documents. Some high dimensioned material cells like the graphite reflector, thermal and biological shield, respectively, as well as the top shield with individual structures were additionally nodalized for generating tallies for a more detailed neutron flux distribution.

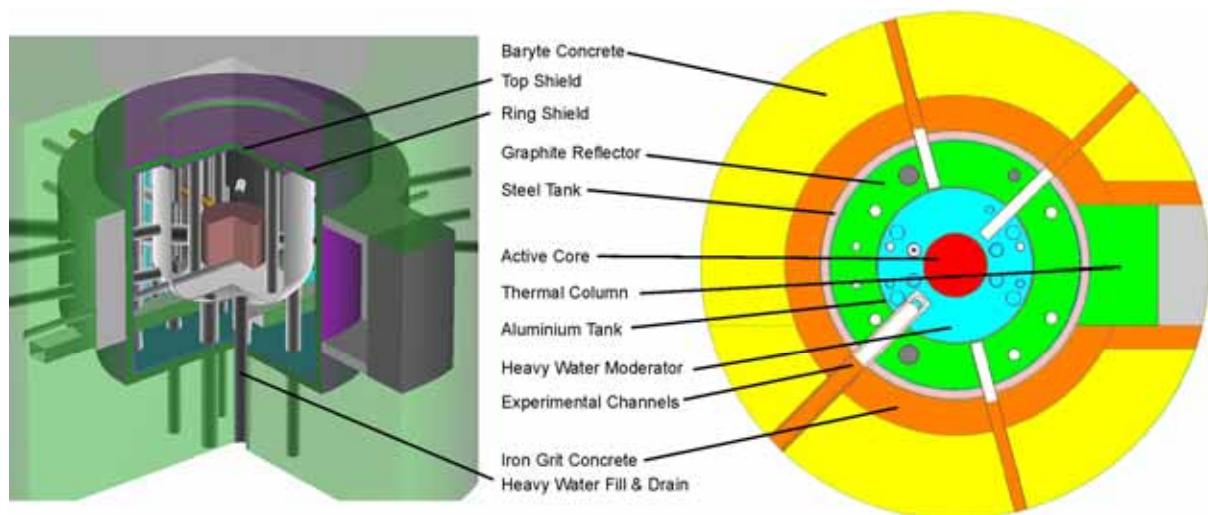


Fig 1: The MCNP model of FRJ-2 for neutron flux calculation

Due to the shielding effects and large dimension of the model, small number of neutrons is tracked in the outer regions resulting in a high statistical relative errors and low reliability on the calculated neutron flux. To improve the performance of the simulations, the variance reduction technique has been employed by the adjustment of cell importances. Using the optimized cell importances, number of neutron history was continuously increased to achieve a sufficient number of neutron tracks for a performed sampling in all cells including the outer regions. The simulations have been performed on the JUMP supercomputer operated at the Research Centre Jülich. JUMP (JUelich Multi Processor) is an IBM p690-Cluster consisting of 1312 processors in 41 nodes with 5.2 Terabytes Memory and a theoretical computation performance of 8.9 Teraflops/s.

4. Results of neutronic calculations

Due to the complexity of the geometrical model of the reactor and the large number of particle histories, the computing time was reduced significantly by the application of the parallel version of MCNP5. The simulations were carried out in steps of 5 million histories. After each step results were checked with regard to statistical errors and standard deviation and returned to MCNP by using the restart capability of the code. In total 50 Mio. histories were simulated resulting in a standard deviation (neutron flux) of less than 1 % around the graphite reflector and steel tank and up to 10 % in the inner parts of the biological shielding behind the absorber blanket. The results of a standard run in an output file are given for one source neutron in each cell of the whole model. To calculate the neutron flux for the real operating condition of the reactor the result had to be divided by the volume of the respective tally cell and multiplied with the total neutron generation rate (1.55×10^{18} n/s). This conversion factor is determined according to the nominal reactor power at which the reactor has been running during the last phase of operation before final shut down (20 MW).

To check the accuracy of the results, the neutron flux in different zones of the core and surrounding parts as well as the neutron spectrum at the boundary surface of the active core were compared with the calculations separately carried out on the basis of a detailed core model. Due to the fact that the effect of the homogeneity of the core decreases with the distance from the core, a good agreement could be found for the regions outside the aluminium tank, which are of particular interest for further analysis and determination of radioactivity inventory. The results of the final calculation are summarized in in Fig. 2 for the whole reactor block. Accordingly the neutron flux in the inner parts of the reactor, the aluminium tank, the grid plate and in the experimental channels surrounded by heavy water is significantly higher than in the outer structures. The average flux from the core to the outer structures is decreased n many orders of magnitude.

The thermal neutron flux in the steel tank with its different layers experiences a decrease in the direction of the outer regions. In this structure, the flux is ranged between 9×10^9 n/cm²s (inside the

boral layer) and 1.2×10^{10} n/cm²s in the outer steel casing. Due to the simulation of high number of histories the relative error of the flux calculation is less than 1 % (1σ). The highest flux in the heavy concrete of the biological shield behind the steel tank is found to be 8.7×10^7 n/cm²s. For some parts of the biological shield the flux decreases with the increase of distance to a level, where the statistical error exceeds 15% due to the low number of neutron tracks. For these few zones (outer boundary zones) the thermal neutron flux was determined by an analytical attenuation calculation using the neutron flux distribution in the surrounding region.

The neutron flux in the structures containing beam tubes and channels is significantly influenced by neutron streaming through the holes. The result is an increase of the flux in the backside structure (Fig 3) around the channel. This effect is clearly demonstrated in Fig. 3 representing the flux pattern in the steel tank over the whole surface. The streaming effect causes a flux profile around the opening which gradually decreases. According to the results the background flux is reached at a distance of three times the geometrical radius. The simulation reveals the same phenomenon behind an absorber layer with distributed cut-outs.

In the top shield, the thermal neutron flux is about 2×10^{11} n/cm²s for the lower stainless steel plate, which falls in the upper parts of the heavy concrete under 10^5 n/cm²s. The relative error of the simulation remains limited to less 10 % in the lower components of the top shield. For the upper parts of the top shield consisting of iron shot concrete it was possible to increase the precision of the simulation by the modification of the cell importances as a variance reduction method. The thermal flux in the lower layer of the concrete shield amounts to 5.8×10^7 n/cm²s, which falls three orders of magnitude. Due to lack of absorber layer in front of the thermal column higher values for the neutron flux are obtained in this structure as well as in the fill and drain Ducts. Using the same approach for certain cells in the outer biological shield, a considerable calculation performance expressed in computational time and precision could be achieved. In view of the decommissioning process of FRJ-2 the existing distribution of the flux allows the determination of the detailed activity inventory in the individual components and structures as well as the radiation dose.

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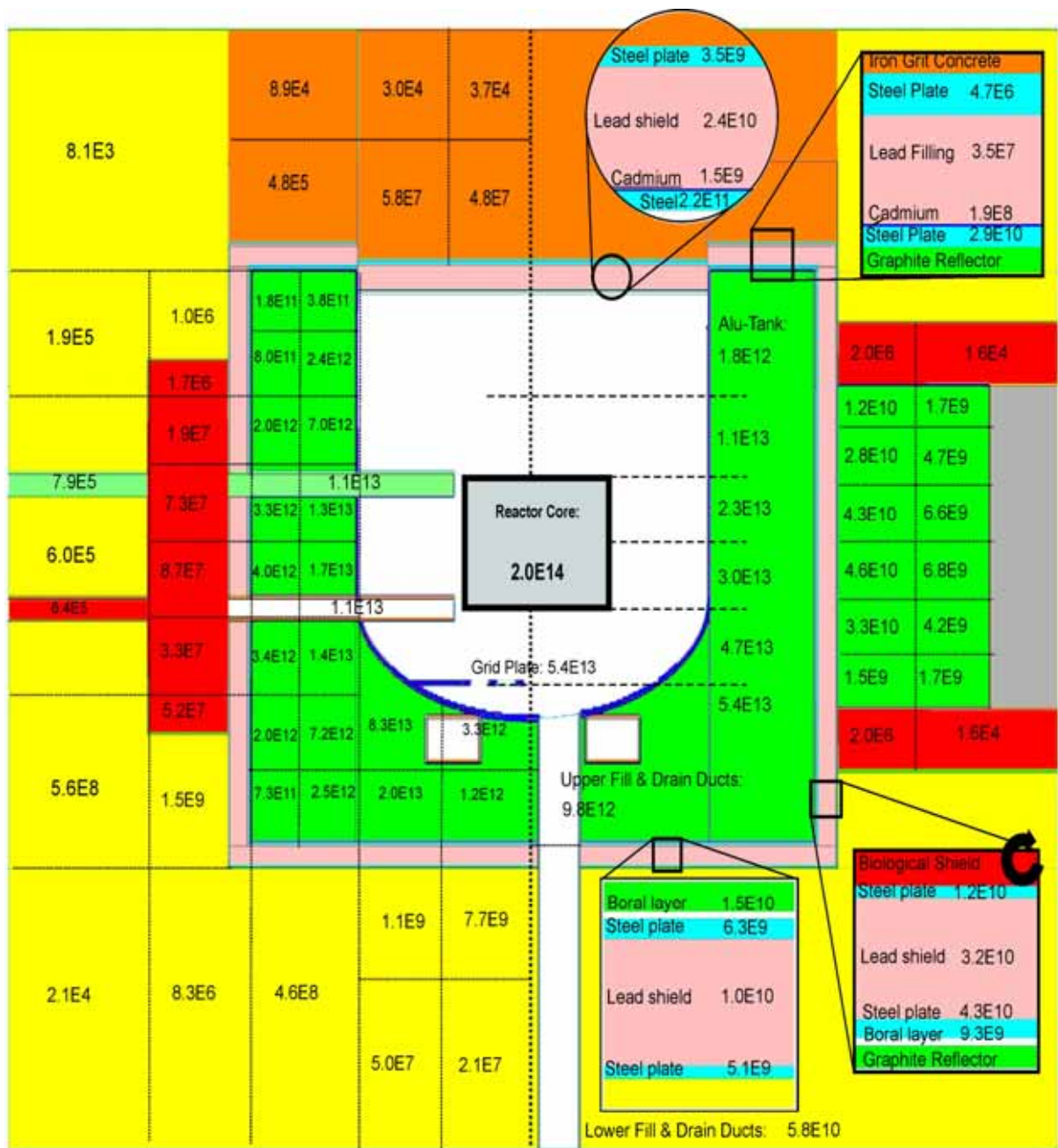


Fig 2: Results of the FRJ-2 neutron flux calculations using MCNP [n/cm²s]

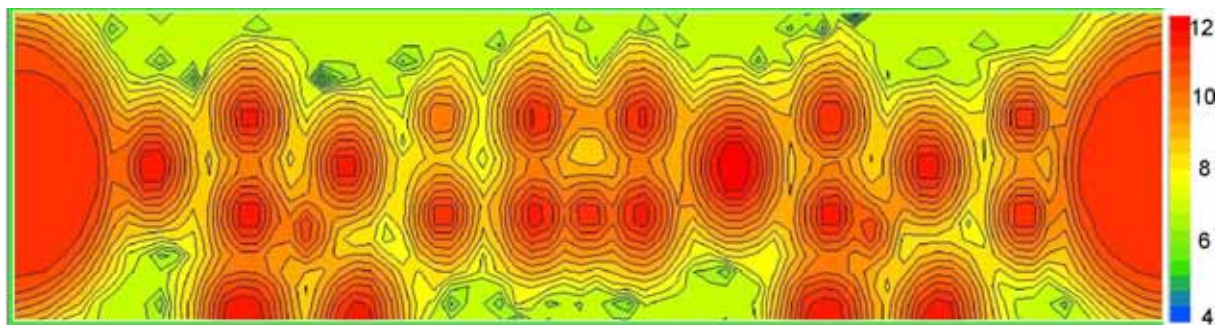


Fig 3: The distribution of neutron flux in the thermal shield behind the borol layer [log(n/cm²s)]



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 ⁹⁹Mo 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/cm³, 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 $3\text{--}3.5 \times 10^{14} \text{ n/cm}^2\text{s}$;
- 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 obsolescence 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:

- 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.

K	J	I	H	G	F	E	D	C	B	A	
D	D	D	D	D	D	D	D	D	D	D	12
R	R	R	R	R	R	R	R	R	R	D	11
R	R	R	CR3		CR7	F58	R	R	R	D	10
R	R	F51	L38		L42	F52	F53	R	D	D	9
R	R	CR4	L41	CR2	L9	L44		R	D	D	8
R	R	L48	L24		L5	L46	L44	R	D	D	7
R	R	L32	L4	L2	L45			R	D	D	6
R	R	CR1	L48	CR3	L42	L47	L45	R	D	D	5
R	R	F56	F57	L48	L39	L1	F58	R	D	D	4
R	R	R	CR4	F55	CR5	F48	R	R	D	D	3
R	R	R	R	R	R	R	R	R	D	D	2
D	D	D	D	D	D	D	D	D	D	D	1

Fig 1. The full converted core configuration

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 617°C and clad 114°C 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).

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 286°C in comparison to HEU fuel, where power peak is 550 MW at a fuel temperature of 810°C. 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

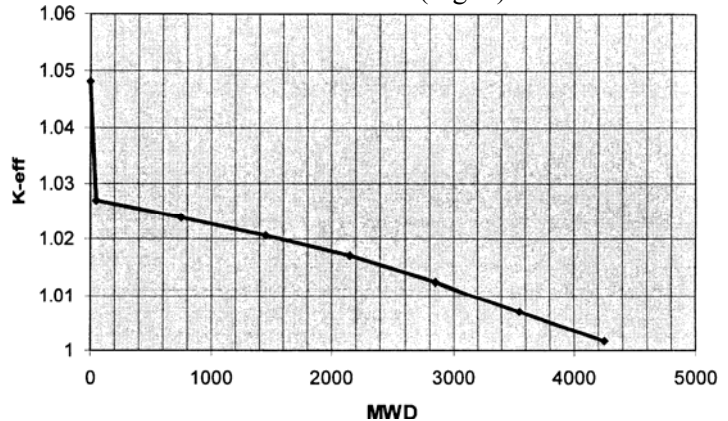


Fig 2. First LEU core lifetime

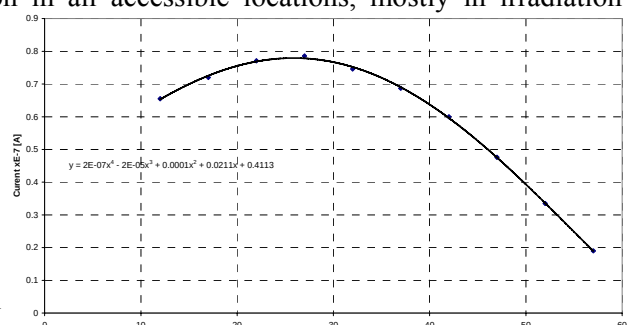
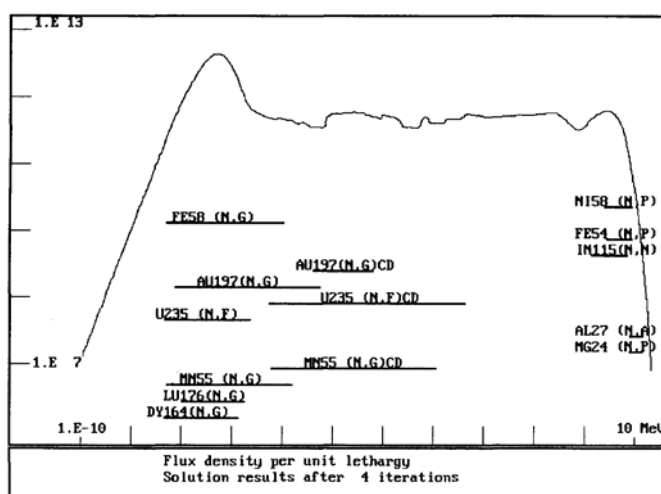


Fig 3 Axial thermal neutron flux distribution, P=1MW

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



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

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 than 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.

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.

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.

- 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.

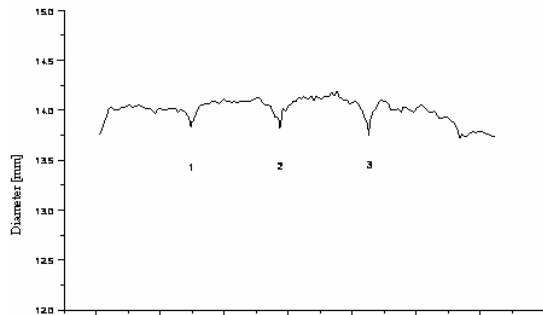


Fig 5 Distribution of values of the mean diameter along the fuel rod - LEU

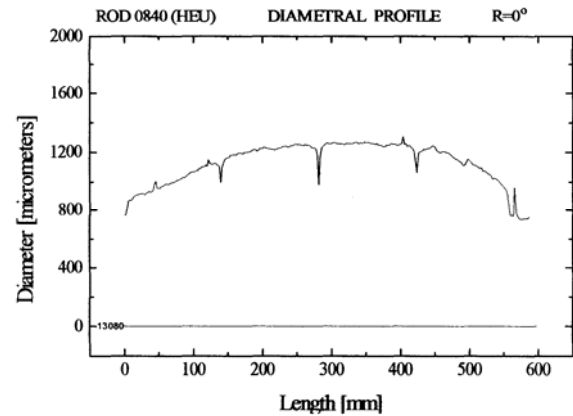
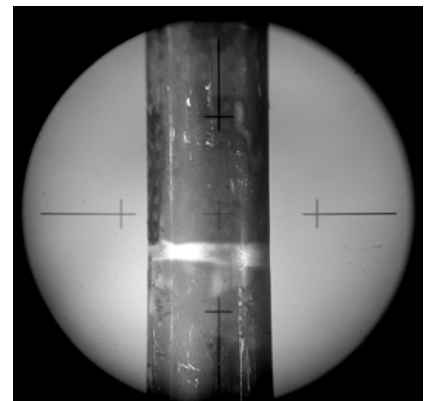


Fig 6 Distribution of values of the mean diameter along the fuel rod - HEU

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 for LEU and HEU fuel elements. For long term measurement and comparison the fission product Cs-137 was selected to avoid the uncertainty produced by accumulation and decay of other fission products. Figs. 8 and 9 show similarities and differences produced by manufacturing technology.

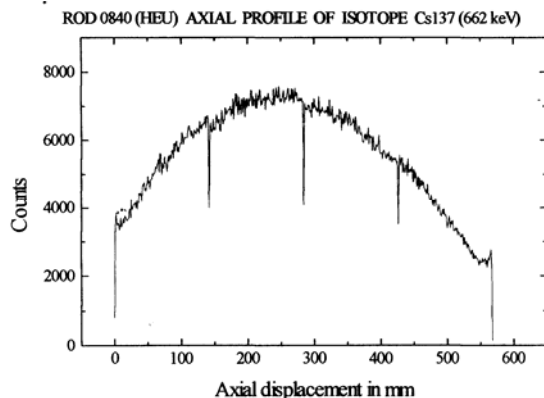


Fig 8. Axial profile of gamma activity for Cs137

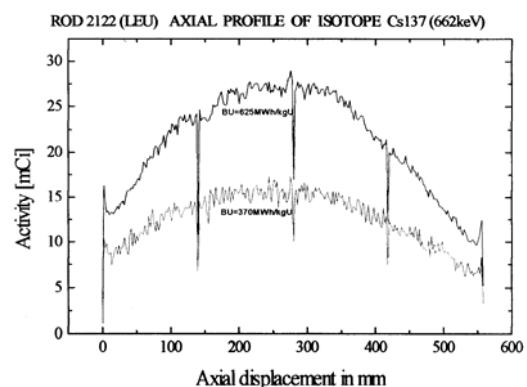


Fig 9. Axial profile of gamma activity for Cs137

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 1 μm and 5 μm in

LEU fuel, producing a quasi-continuous uranium matrix. Above 330°C 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).

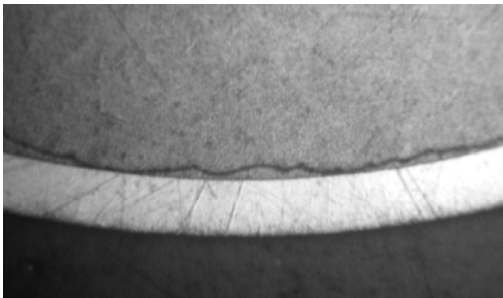


Fig10. Fuel circumferential cracks located at clad

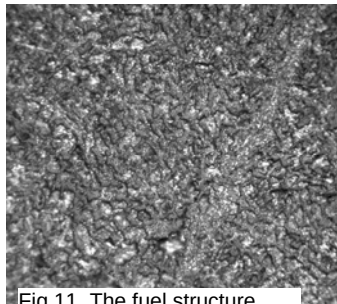


Fig 11. The fuel structure

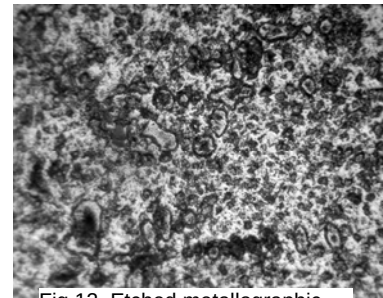


Fig 12. Etched metallographic

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.

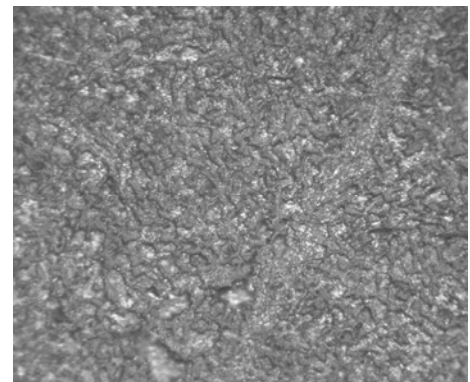


Fig 13. Fuel structure (X100)

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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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 UO₂ 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 UO₂ fuel, different results were obtained [1,5]. In a previous work of some colleges [1] a UO₂ fuel with 5.45 g content of ²³⁵U/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 UO₂ were considered : the UO₂ 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 core 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).

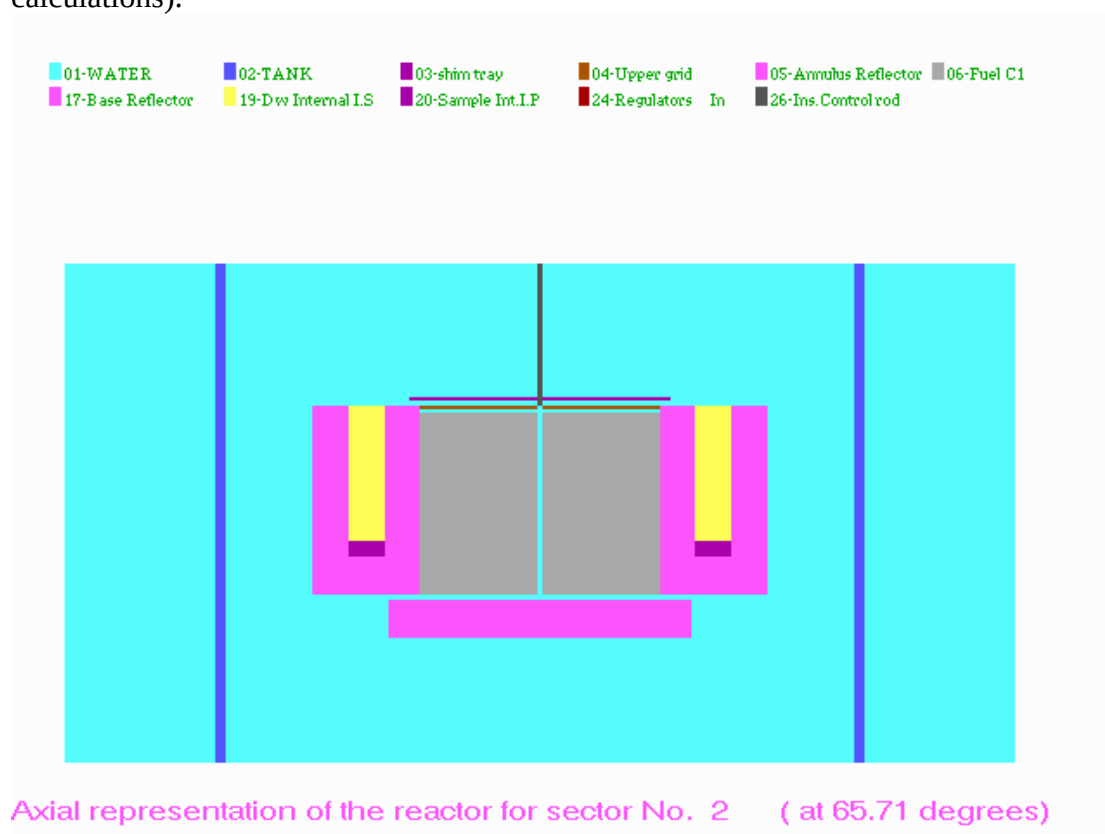


Fig. 1 . The new model of the Syrian MNSR used for the core calculations.

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-Al ₄ -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 $\sim 1.10^{12}$ n/cm².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 UO₂-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 UO₂-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 UO₂ pellets cladde 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

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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.

SAFARI-1 POWER HISTORY

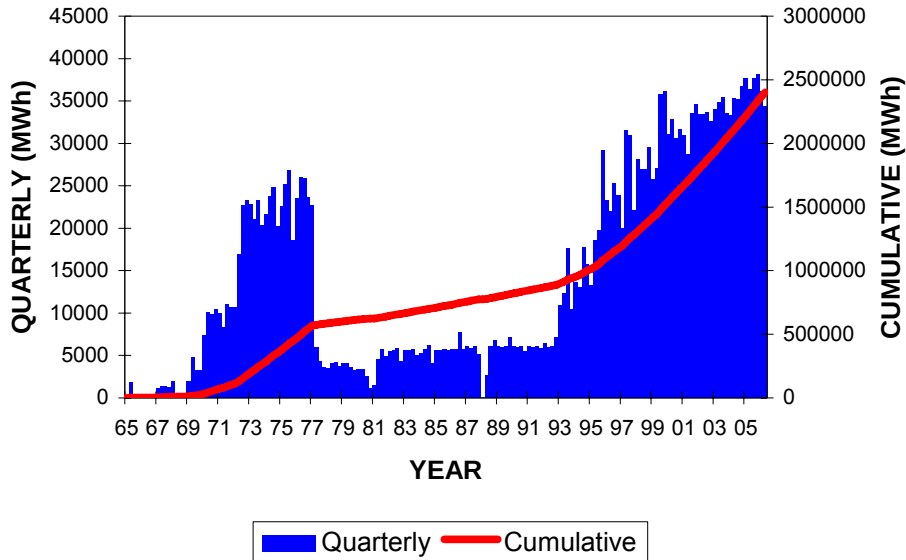


Figure 1: SAFARI-1 Power History (MWh/quarter and Cumulative)

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/cm³. 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.

Flat Plates



Curved Plates



Ring-shaped Plates



Tubular Plates



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.

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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 M³O 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 LEU¹, 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 M³O-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 $k(\text{eff})_{\text{ini}}$ 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 $k(\text{eff})_{\text{ini}}$ 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 $k(\text{eff})_{\text{ini}}$ [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 $k(\text{eff})_{\text{ini}}(x_i)$ 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 ($f(x_i)=k(\text{eff})_{\text{ini}}(x_i)$).

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 $k(\text{eff})_{\text{ini}}$ on increasing meat thickness.⁴ Increasing the active height of the fuel element, directly raises $k(\text{eff})_{\text{ini}}$. 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 $k(\text{eff})_{\text{ini}}$ significantly [17,18].⁵

² See [11, 14] for more details on this discussion.

³ As a faster alternative to the MCNP part of M³O 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 $T_{\text{cycLEU1}} \approx 50$ 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 $k(\text{eff})_{\text{ini}}$ [17,18] without changes to the outer fuel element geometry. However, studying the parameter space only to increase $k(\text{eff})_{\text{ini}}$ 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, $k(\text{eff})_{\text{ini}}$ increases from 1.169 to 1.172, and flux loss increases by additional 0.5%.)

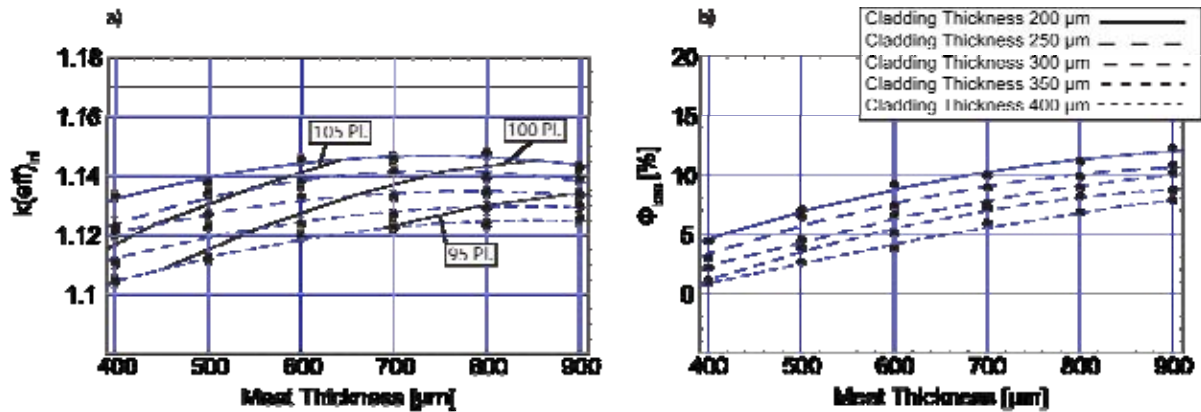


Fig. 1 Calculation results for different meat and cladding thicknesses (cooling channel thickness 3.0 mm, active height 70 cm, LEU). Each point represents a M³O calculation with a particular fuel element geometry of FRM-II. The plate number differs from core to core. a) Initial reactivity for different 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 $k(\text{eff})_{\text{ini}}$ 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 $k(\text{eff})_{\text{ini}}$. 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 $k(\text{eff})_{\text{ini}}$ (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 $k(\text{eff})_{\text{ini}}$ and flux: for wider cooling channels $k(\text{eff})_{\text{ini}}$ 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.

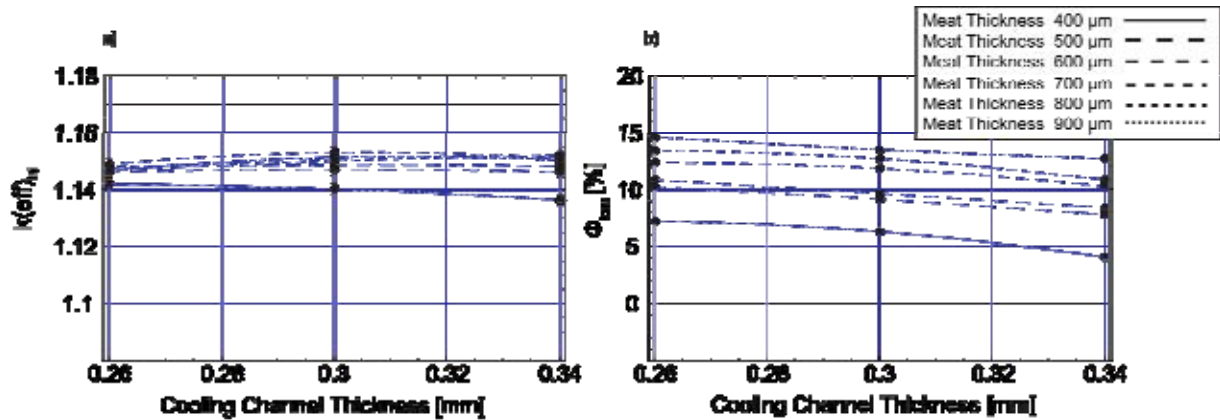


Fig. 2 Results for different meat and cooling channel thicknesses (active height 70 cm, 150 μm Zr-cladding, FRM-II, LEU). Each point represents a M^3O calculation with a particular fuel element geometry. The plate number differs from core to core. a) Initial reactivity for the different core dimensions. 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 $k(\text{eff})_{\text{ini}}$ 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 $k(\text{eff})_{\text{ini}}$ 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.

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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 safely 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.

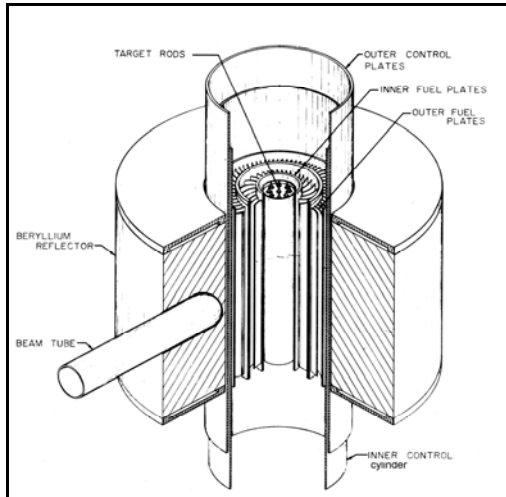
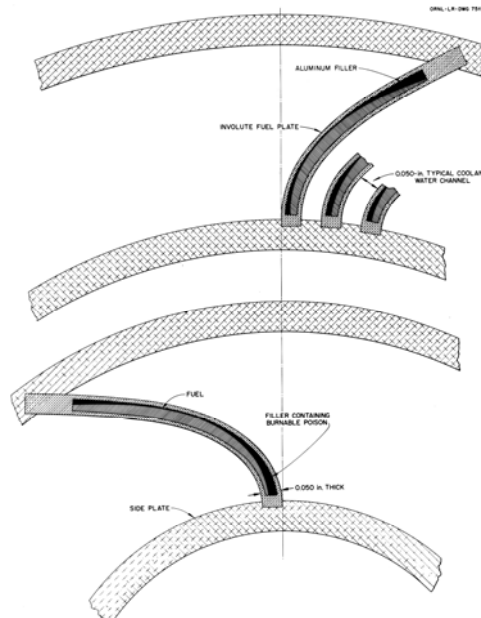


Figure 2. The core of HFIR, showing the inner (IFE) and outer (OFE) fuel elements.



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 calculation) 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

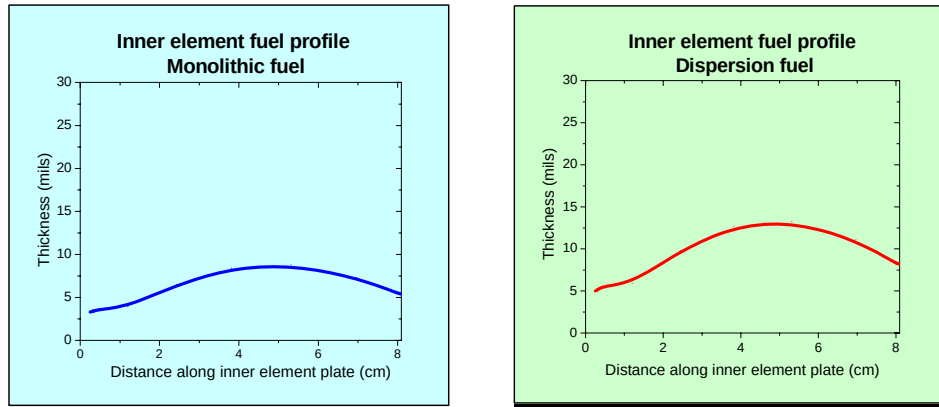
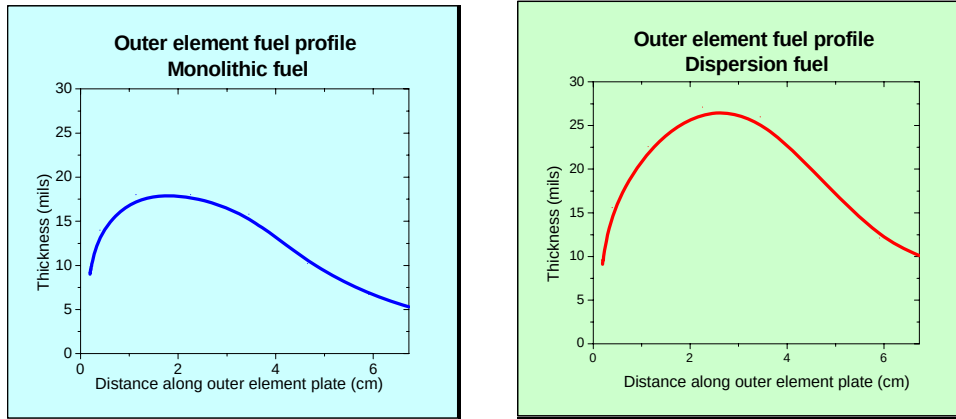


Fig. 5. Outer element LEU fuel profiles



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.

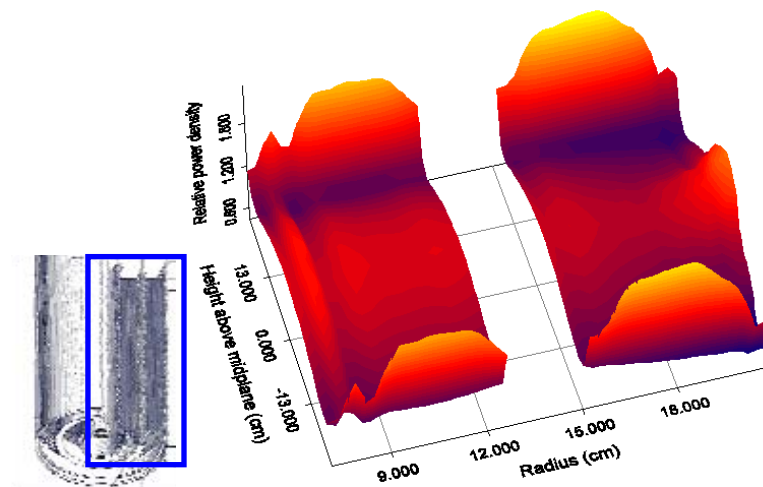
Table 1. Selected LEU performance parameters

Parameter		HEU	LEU monolith	LEU dispersion
Peak thermal flux in reflector ($\text{n/cm}^2\text{s}$)	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 ($\text{n/cm}^2\text{s}$)	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 UO₂ 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 UO₂ 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 UO₂ 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 ($1/2 \beta_{\text{eff}}$) 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/ $^{\circ}\text{C}$) to boost its inherent safety properties [3]. Additionally, the excess reactivity of the reactor is limited to $\rho_{\text{ex}} \leq 1/2 \beta_{\text{eff}}$ [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 losses and conserve neutron economy as shown in the cross-sectional view (Fig.1).

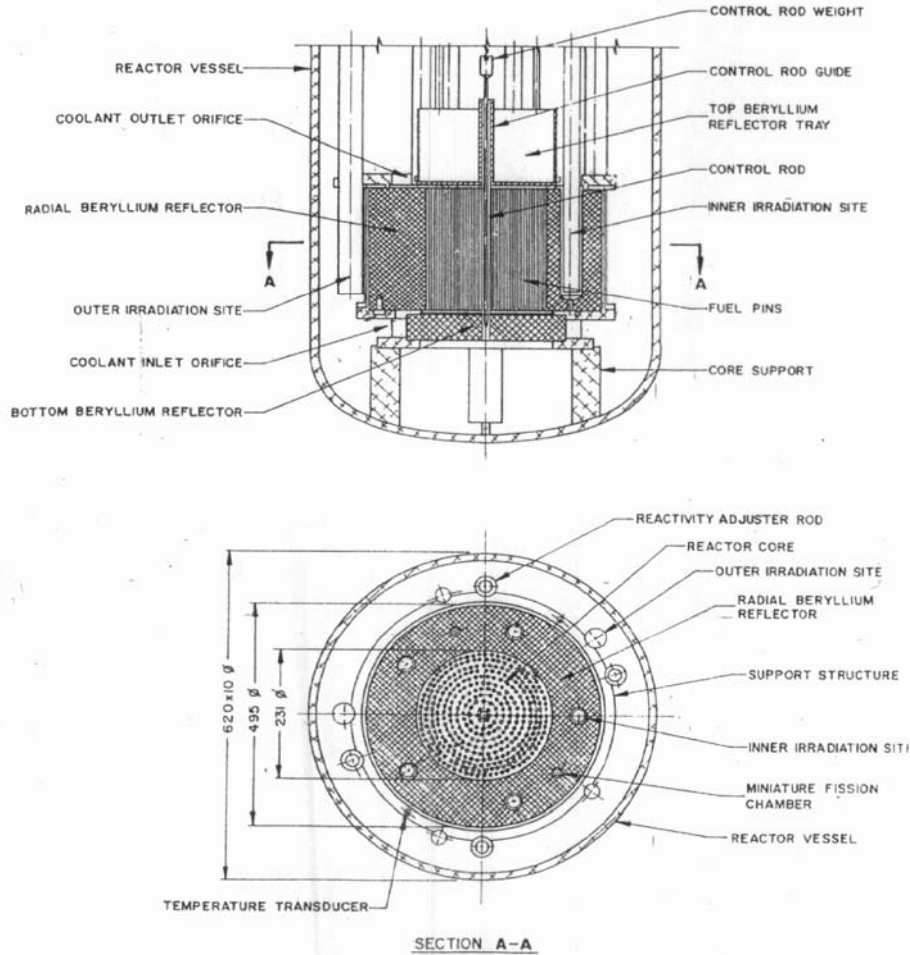


Fig.1: Cross section through the GHARR-1 reactor

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.

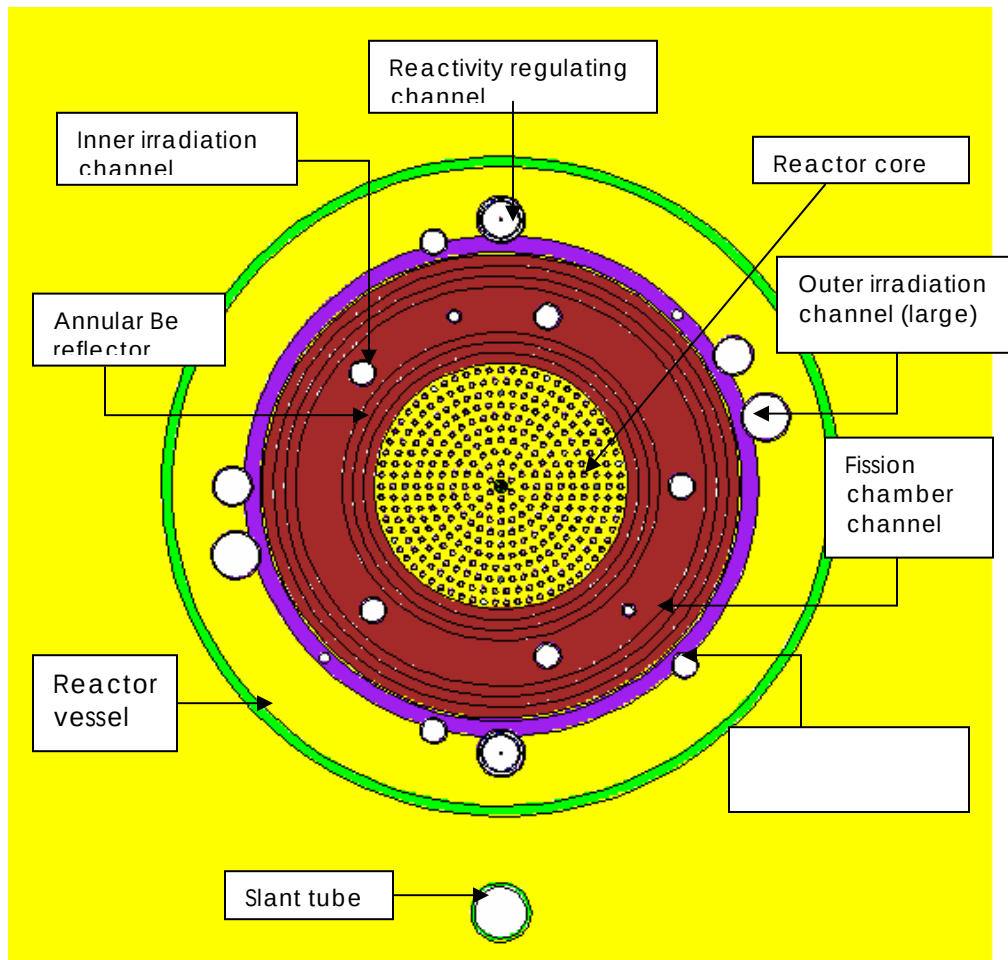


Fig. 2: MCNP5 plot of GHARR-1 core configuration showing fuel region (reactor core), channels for irradiation, fission chamber, regulating, slant and annular beryllium reflector

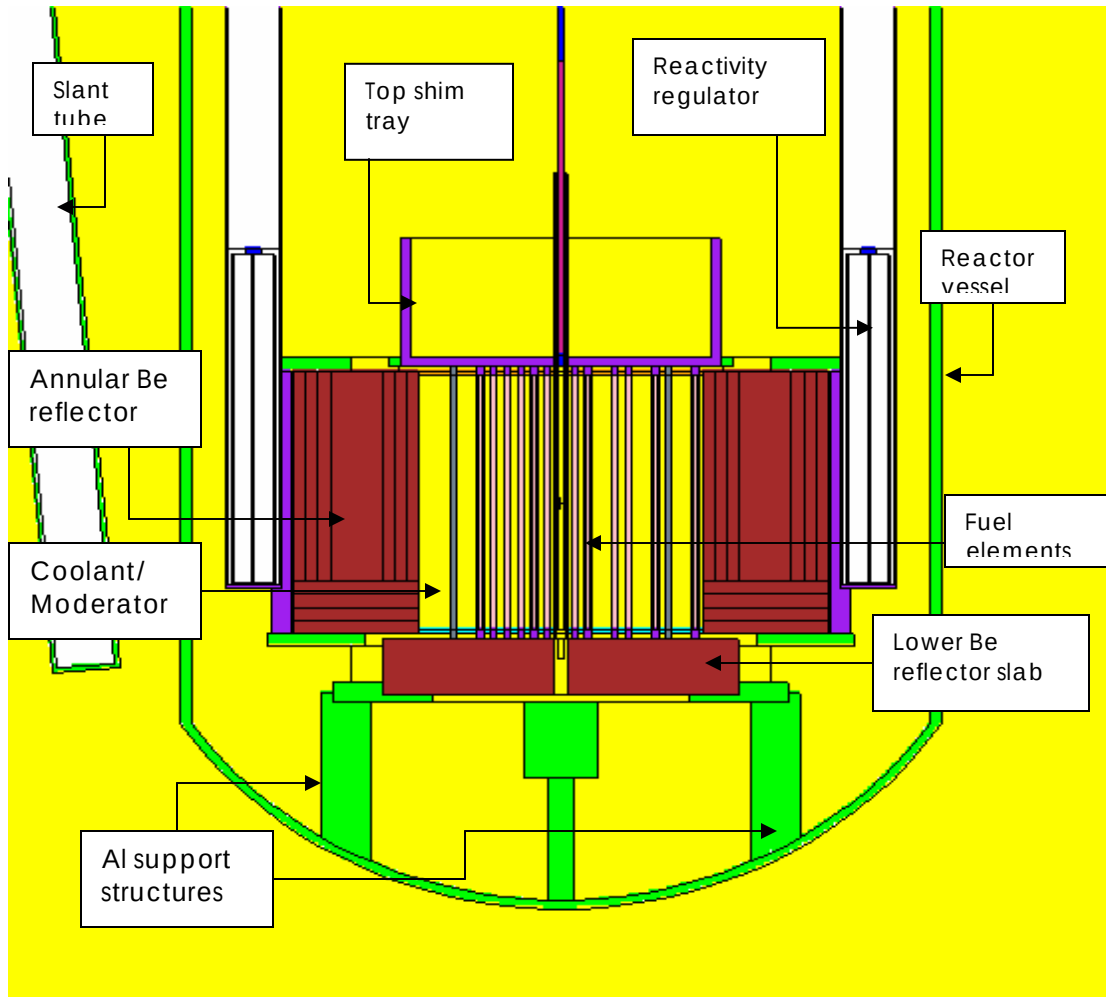


Fig.3: MCNP plot of vertical cross section of GHARR-1 reactor (control rod in full withdrawn position) showing structural supports, reactor vessel, etc.

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 UO₂ 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)

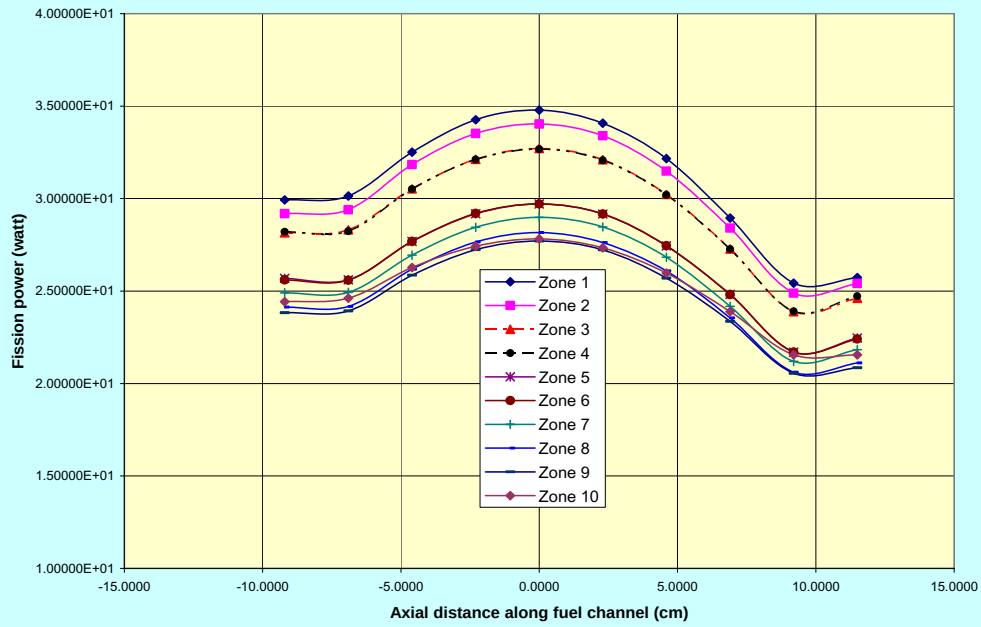
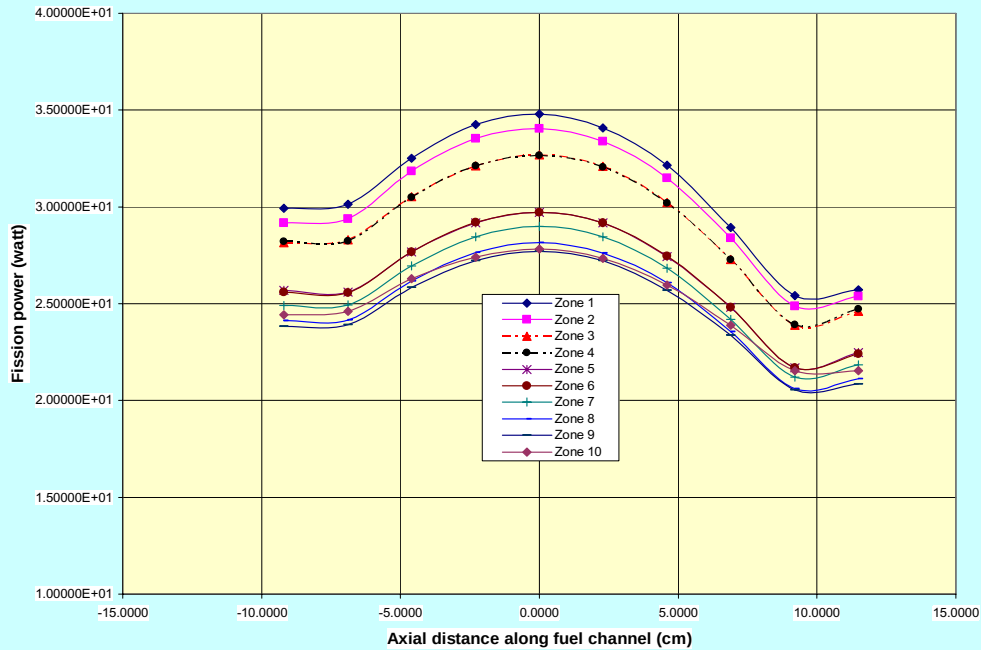
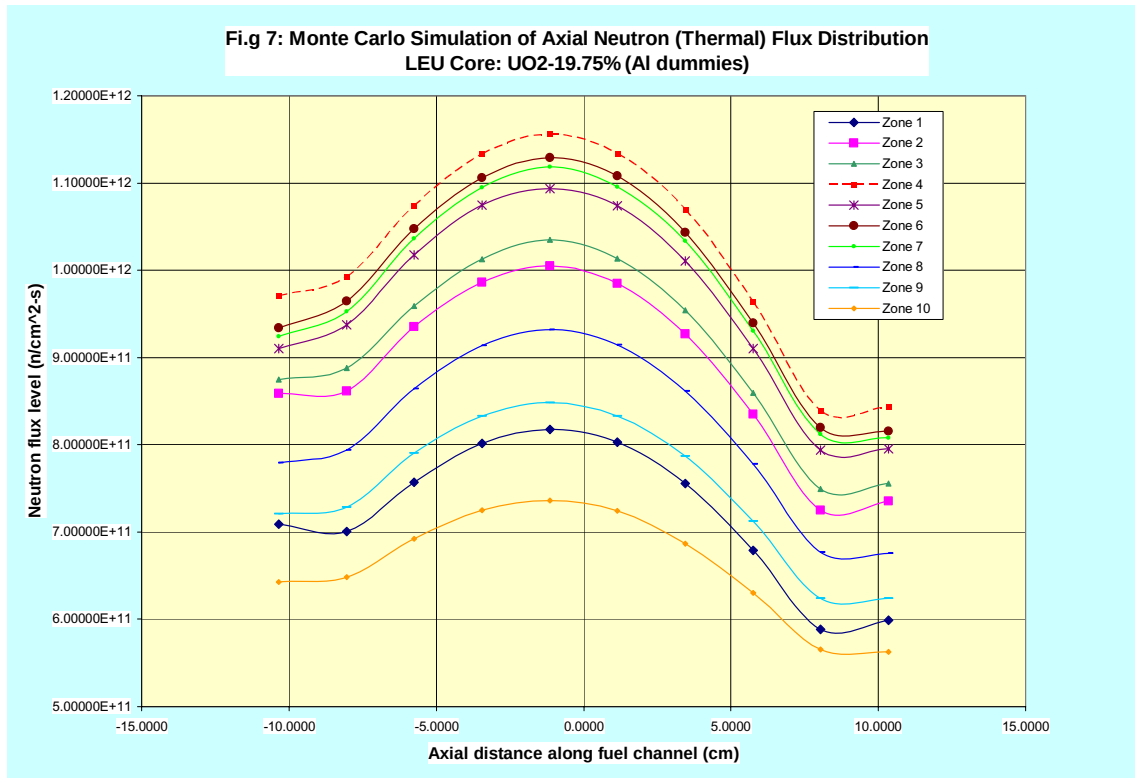
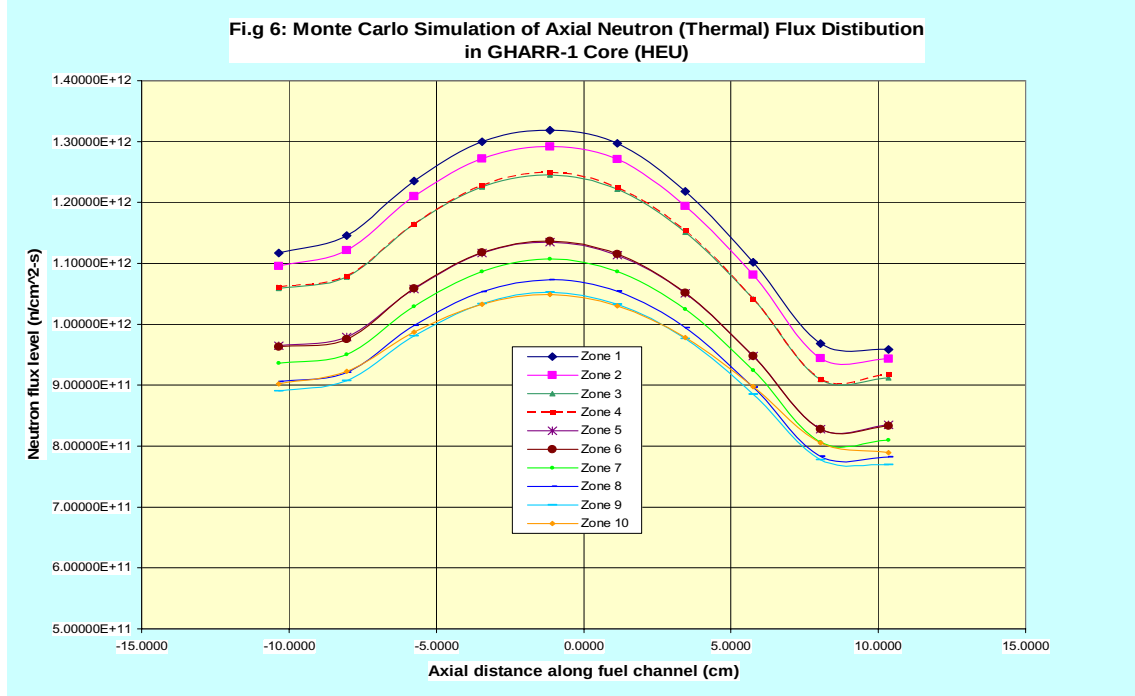


Fig. 5: Monte Carlo Simulation of Axial Variation of Fission Power Deposited in GHARR-1 Core (UO₂ LEU fuel: 19.75% + Al dummies)



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.



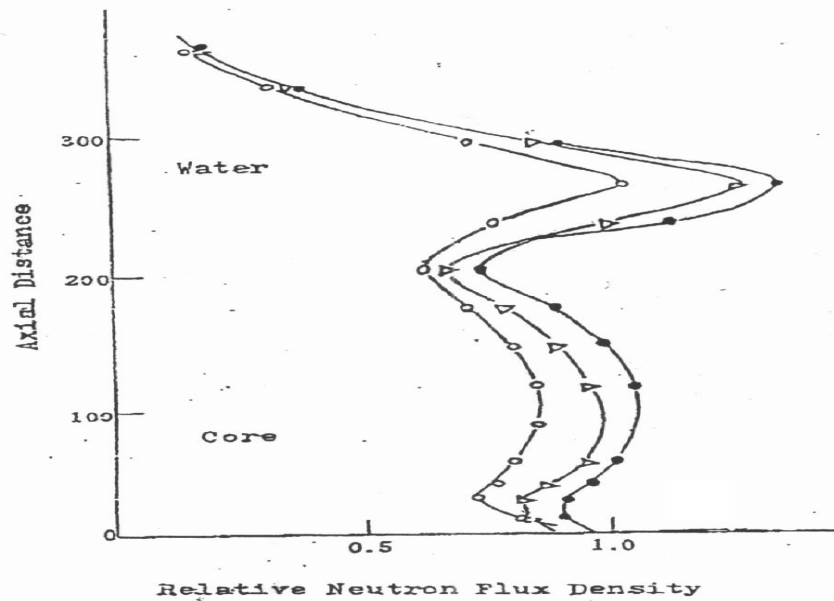


Fig.8: Relative axial flux density (experimental)

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 $5.0\text{E}+11 \text{ n/cm}^2\cdot\text{s}$. 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 $1.0\text{E}+12 \text{ n/cm}^2\cdot\text{s}$ 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.

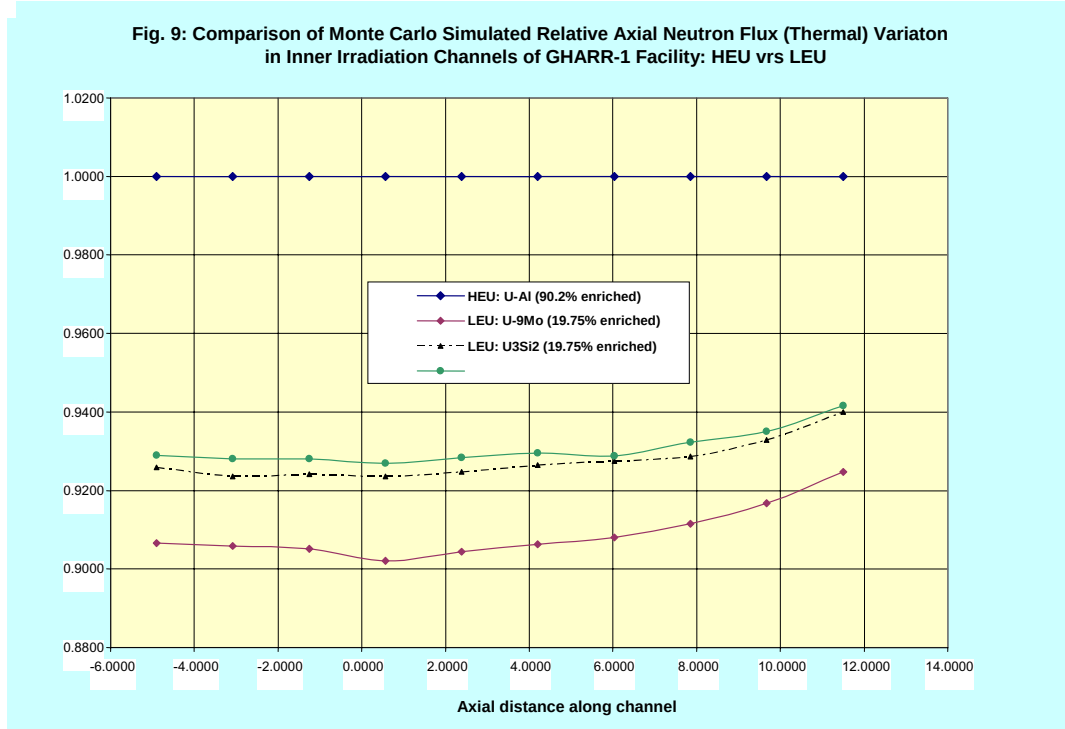


Table 2: Kinetic parameters of GHARR-1

	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	8.0800E-03
LEU fuel: UO_2 , 12.6% 344 FE, 6 Al dummies	1.00454	0.99621	8.3240E-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.B.M is the one to carry out replacement and inspection on the planned time. T.B.M is subdivided into the periodic maintenance (P.E.M) and the age based maintenance (A.M). P.E.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.

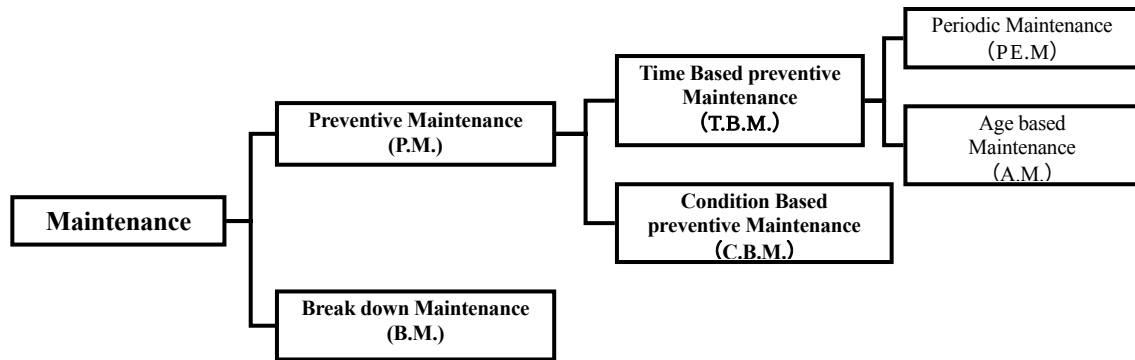


Fig.1 System of maintenance

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 $1900\text{W}/\text{m}^2 \cdot \text{K}$.

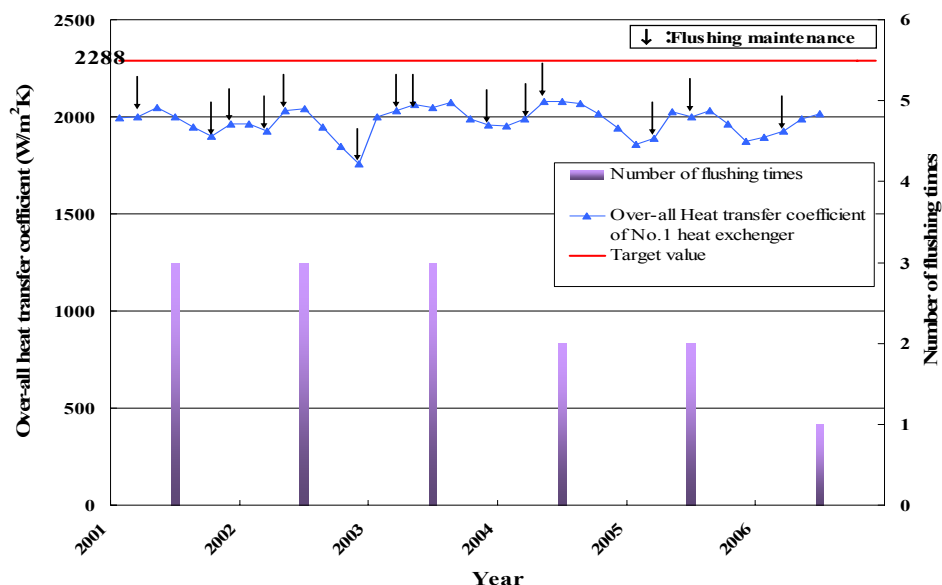


Fig.2 Relation between flushing maintenance and Over-all Heat Transfer Coefficient

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 50 μ m) 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 60 μ m 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.

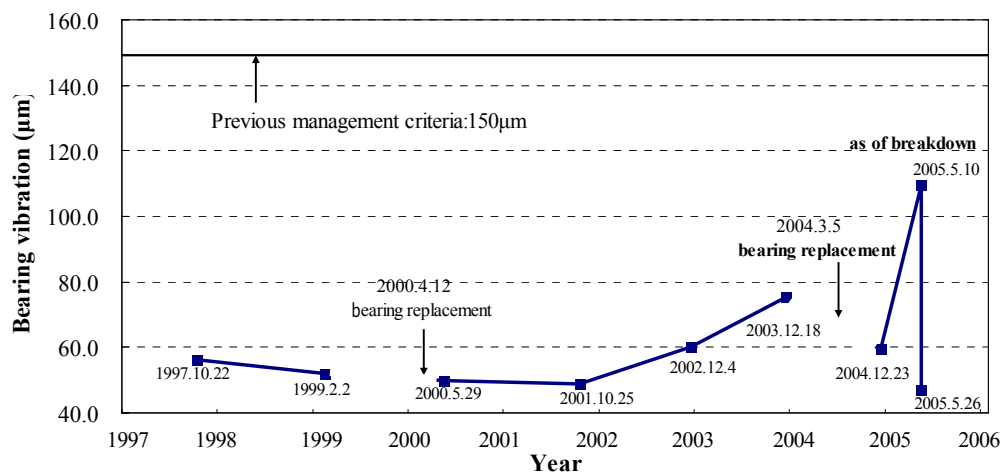


Fig.3 Vibration data of air blower

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 D₂O and cooled with H₂O. The core consists of a single fuel element and one central control rod. The D₂O filled moderator tank containing the fuel element is situated in a pool with H₂O. 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 D₂O 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 $\leq 5 \mu\text{Sv/h}$, 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 $\leq 5 \mu\text{Sv/h}$ (locally $\leq 10 \mu\text{Sv/h}$), 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 $\leq 1 \text{ mSv/h}$, 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.

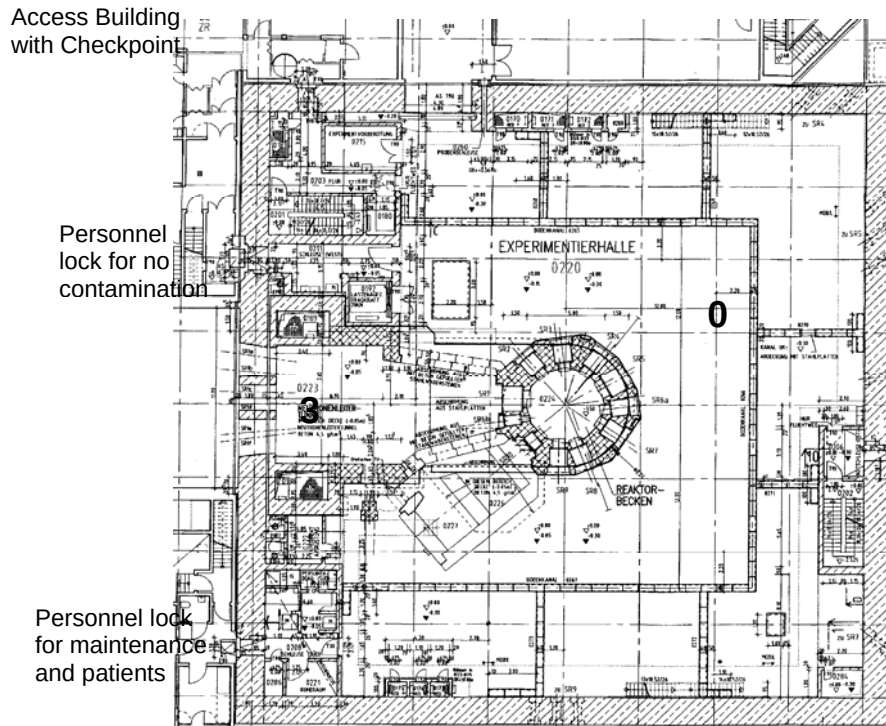


Fig. 1 Experimentation Hall

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, $\rho \geq 2.1 \text{ g/cm}^3$ - in general

- Heavy (hematite) concrete, $\rho \geq 3.6 \text{ g/cm}^3$; 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, $\rho \geq 4.5 \text{ g/cm}^3$; 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.

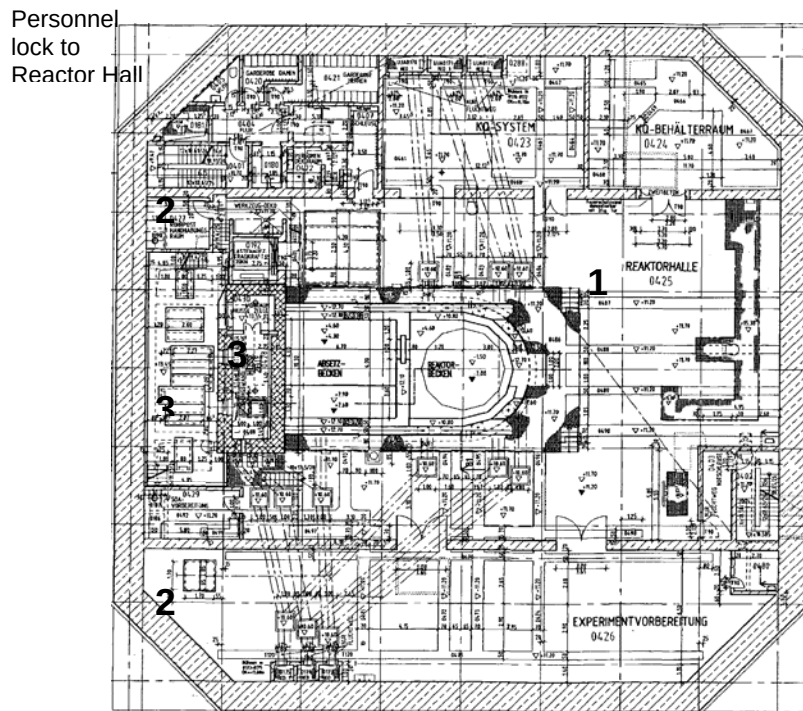


Fig. 2 Reactor Hall

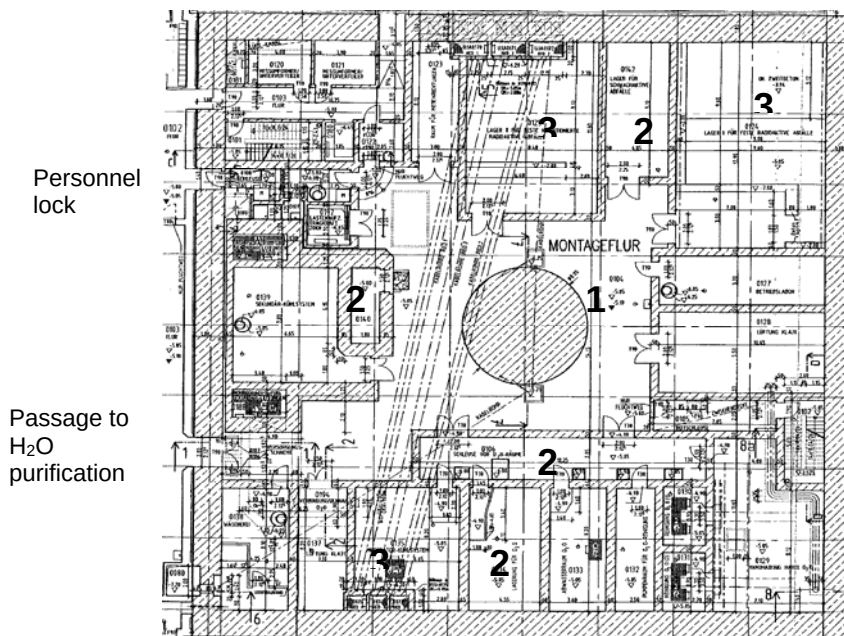


Fig. 3 Cellar of Reactor Building

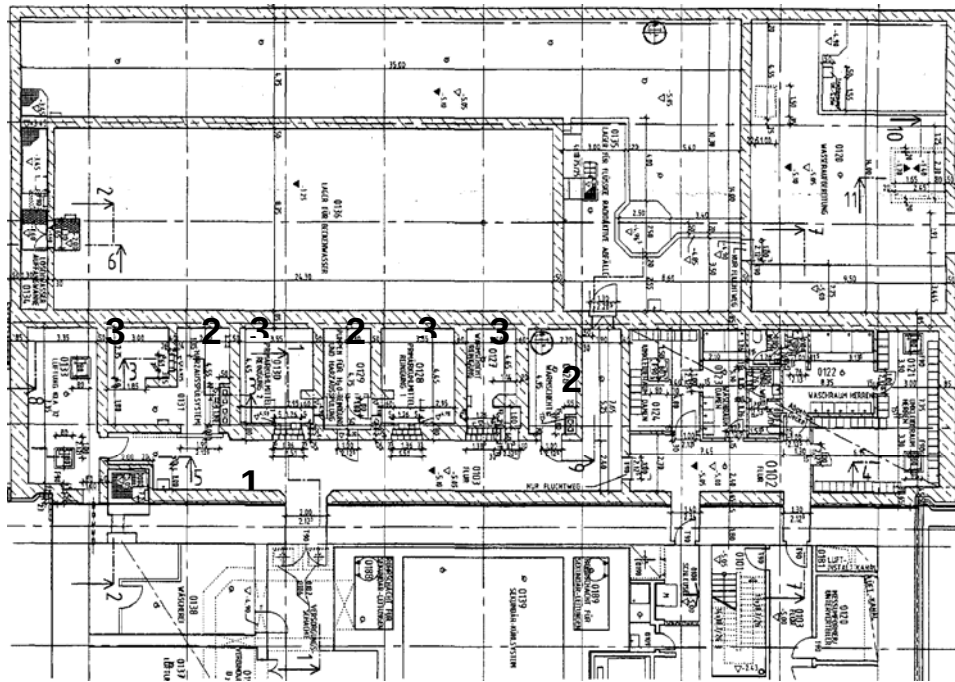


Fig. 4 Cellar for H₂O Purification and Liquid Waste

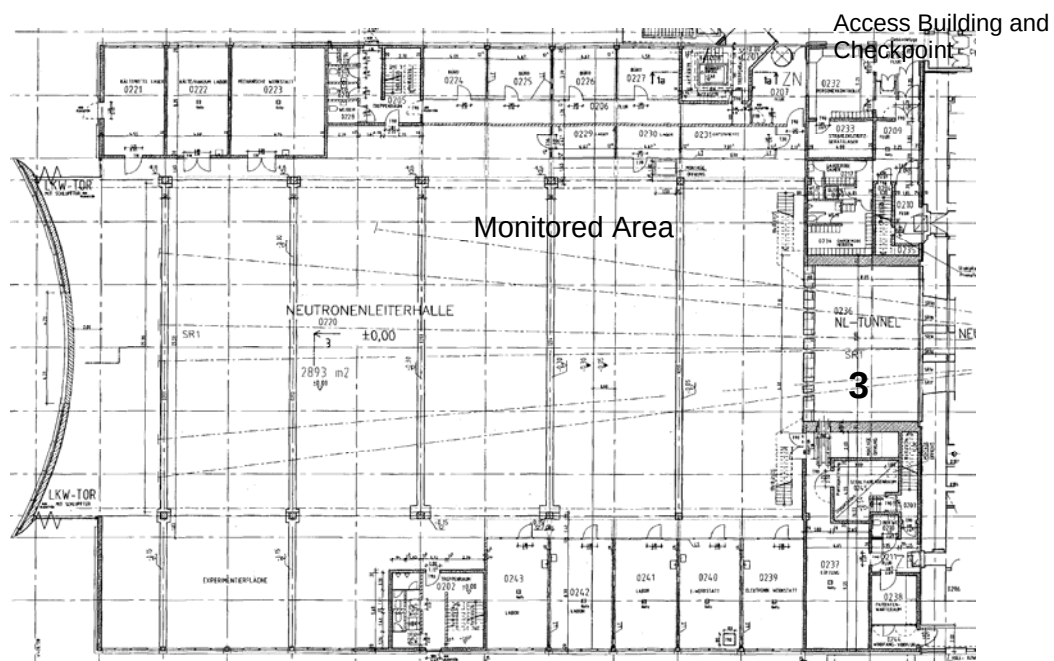


Fig. 5 Neutron Guide Hall

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-foot 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 ^3H 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).

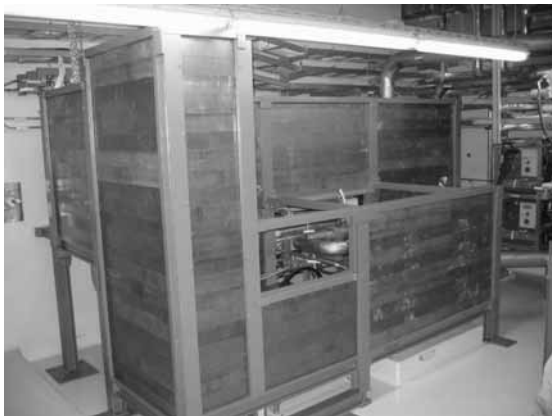


Fig. 6 Lead shielding around the D_2O cooler and the associated pump and valves

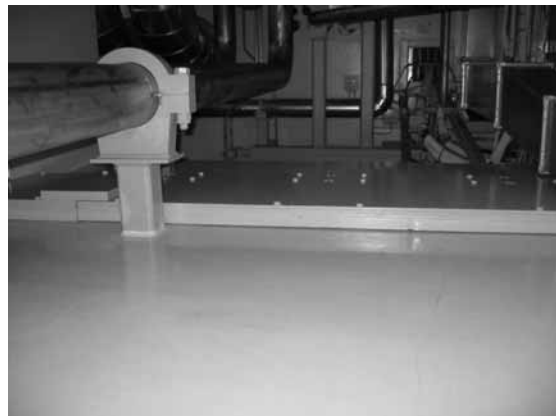


Fig. 7 Additional steel protection of a concrete gap in the floor of the tunnel for D_2O piping

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 ($MDA = 2,75 \cdot 10^{-4} \text{ Bq/cm}^2$ 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	$5,60 \cdot 10^{-04}$
Zn-65	244 d	$4,10 \cdot 10^{-04}$
Mn-54	312 d	$2,80 \cdot 10^{-04}$
Co-58	70,8 d	$3,50 \cdot 10^{-04}$
Cs-134	2,0 y	$< 9,33 \cdot 10^{-05}$
Cs-137	30,2 y	$1,50 \cdot 10^{-03}$

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 m³ 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/m³ of ¹³⁸Cs was detected.

2.7 Measurement of aerosol sample collected in the off-gas channel of the reactor.

The measurement was performed sampling 50 m³ of air on a active-carbon and absolute filter (porosity 4,5 µm) and only natural radioisotopes were detected (MDA for ¹³⁸Cs = 7,5 10⁻² Bq/m³; MDA for ¹³⁷Cs = 5,80 10⁻⁴ Bq/m³).

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	[λ _{Xe} N _{Xe} (1h) = 5.97 10 ⁻² R]	
–	direct production (Y= 4.81 %)	λ _{Xe} N _{Xe} (1h) = 4.56 10 ⁻² R
–	production from ¹³⁸ I (Y = 1.49 %)	λ _{Xe} N _{Xe} (1h) = 1.41 10 ⁻² R
¹³⁵ Xe	[λ _{Xe} N _{Xe} (1h) = 3.79 10 ⁻⁴ R]	
–	direct production (Y= 0.0785 %)	λ _{Xe} N _{Xe} (1h) = 5.70 10 ⁻⁵ R
–	production from ¹³⁵ I (Y = 6.28 %)	λ _{Xe} N _{Xe} (1h) = 2.36 10 ⁻⁴ R
–	production from ^{135m} Xe (Y = 0.178 %)	λ _{Xe} N _{Xe} (1h) = 8.57 10 ⁻⁵ R
⁸⁸ Kr	[λ _{Kr} N _{Kr} (1h) = 7.69 10 ⁻³ R]	
–	direct production (Y= 3.55 %)	
⁸⁷ Kr	[λ _{Kr} N _{Kr} (1h) = 1.075 10 ⁻² R]	
–	direct production (Y= 2.56 %)	
^{85m} Kr	[λ _{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 $\varnothing$ 25,4 mm) with a funnel terminal ($\varnothing$ 55 mm)
- a hydraulic pump OMAN mod. ALM25 in SST (max flow 43 lt/min – Prevalence 12 m)
- a rubber tube for water circulation ($\varnothing$ 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

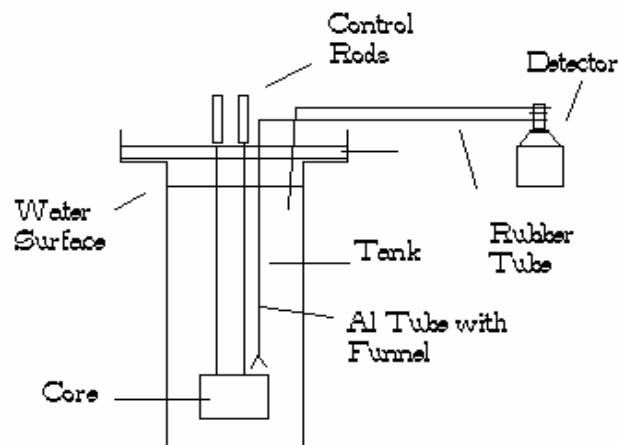


Fig. 1 - Sampling and detection apparatus lay-out

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.

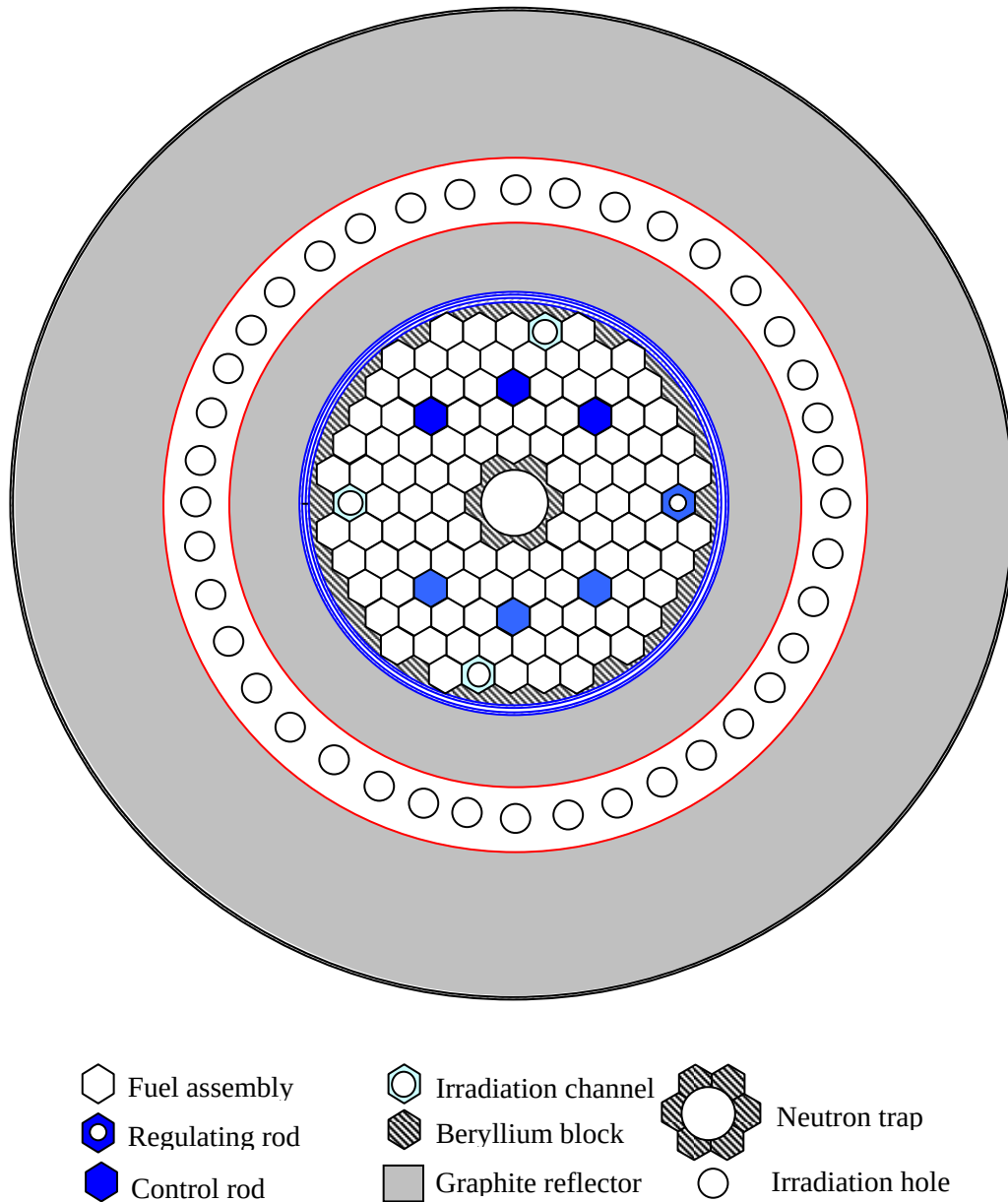


Figure 1. Core configuration with the 104 fuel assemblies

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%).

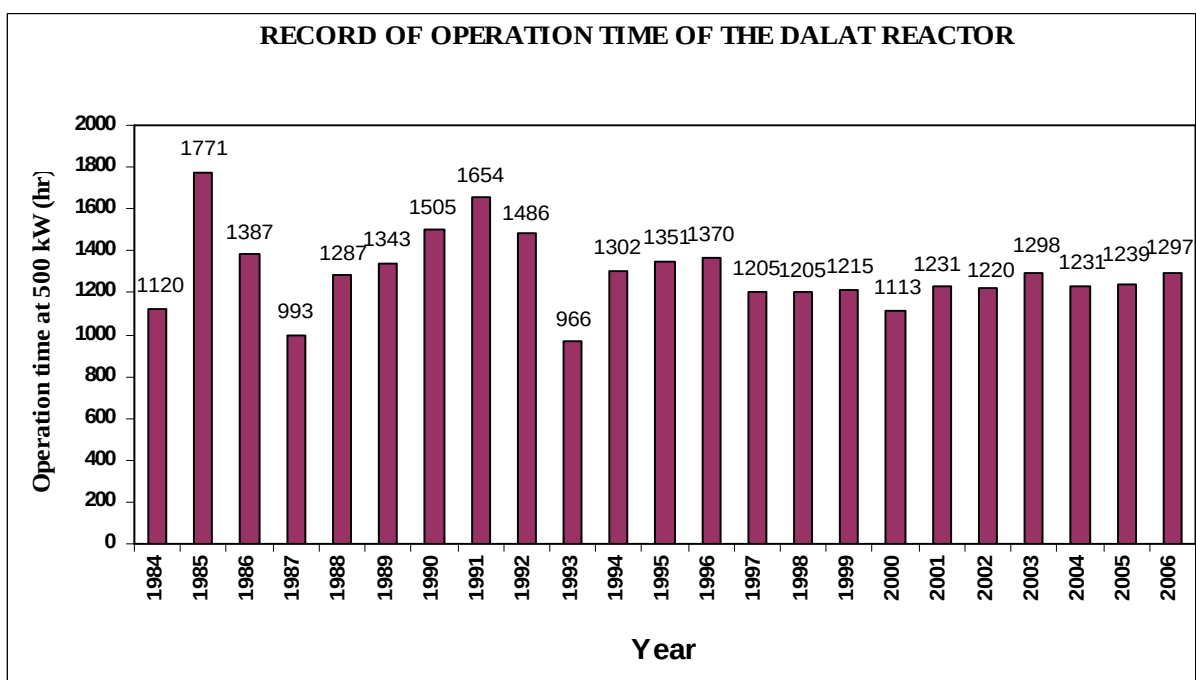


Figure 2. Record of operation time of the Dalat reactor

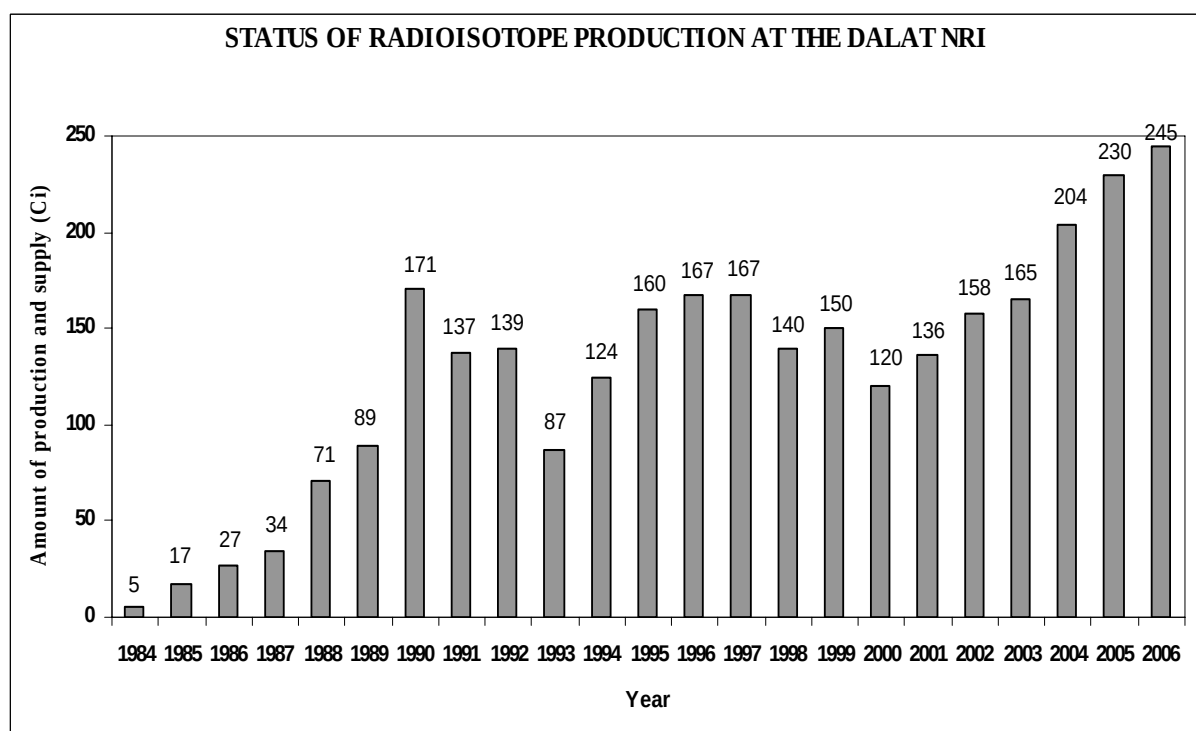


Figure 3. Status of radioactive production at the Dalat Nuclear Research Institute

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
<u>Dry Irradiation Channels</u>								
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
<u>Wet Irradiation Channel</u>								
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.

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Session VII

Fuel Back-end Management

THE JASON REACTOR: FROM CORE REMOVAL TO FUEL REPROCESSING

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ABSTRACT

The 10 kW JASON Argonaut reactor was operated at the Royal Naval College, Greenwich, London, between 1962 and 1996. After initial cooling in the core, the MTR type fuel (80% enriched ^{235}U) was dry stored on site before transport in 1998 to BNFL, Sellafield for interim wet storage. Arrangements for reprocessing of the fuel at AREVA NC, La Hague are now in progress and this paper will describe various aspects of the storage, transfer and monitoring, including criticality calculations using MCNP, prior to its possible final transport to France for reprocessing.

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1. Introduction

The JASON 10 kW Argonaut reactor was used to educate and train military and civilian personnel involved in the naval nuclear submarine propulsion programme. It was situated in a reactor hall within King William Building, a Grade 1 listed building within the Old Royal Naval College at Greenwich, which itself is a Scheduled Ancient Monument having World Heritage Site status. The reactor was first taken critical at the College in November 1962, having been previously operated by the Hawker Siddley Nuclear Power Corporation at Langley from February 1959. The decision to decommission Jason was taken in 1996, following the ministerial decision that the Royal Naval College would pass to non-defence use by the millennium.

Following the final reactor shut-down, the decommissioning programme began with the setting up of the management, safety and project teams, obtaining the various heritage and planning approvals for the works and gaining regulatory approval of the nuclear safety cases. The first physical stage in the programme, called Post-Operational Clear Out (POCO), involved disabling the reactor and removing particular operational equipment and was completed by June 1998. The rest of the programme consisted of fuel removal, reactor dismantling, waste removal, site survey and clean up and the final radiological clearance of the site by the Environment Agency, which was achieved on 4 November 1999. A full description of the JASON decommissioning has been given previously [1,2].

Criticality calculations have been performed for the JASON fuel in the reactor hall fuel store and in the UNIFETCH flask using the general purpose Monte Carlo code MCNP4c2. Each model included as much geometrical detail as was available. The most recent ENDF/B-VI cross-section data was used. In addition, MCNP was used for dose rate calculations. The MCNP source term was created from the fission product inventory calculated by AEA Technology in 1998 from the JASON power history modified to account for radioactive decay. It should be noted that due to the very low power operation (25 MWh over 36 years) and nine years of storage, the activity in the fuel is very low. The dose rate at one metre from a single fuel box in air is approximately 15 $\mu\text{Sv/h}$ now.

2. Storage on Site and Fuel Transfer to Sellafield

Fuel removal from the core was completed under the initial POCO phase and the fuel was stored in dry fuel pits within the main reactor hall to allow for radioactive decay of the short lived fission products. In total eight fuel pits were used to store a maximum of 16 fuel boxes and MCNP calculations indicated k_{eff} of 0.23456 and 0.38909 respectively for normal dry storage and worst case flooding of the pits. With respect to fuel burnup the following data was generated at the time of decommissioning [3]:

- The standard core used in JASON used 10 fuel boxes of 16 plates and 1 box of 12 plates. Calculations performed at RNC give the “start of life” fuel inventory in the core as 1980.98 g of U-235, 0.0274 g of U-234 and 495.245 g of U-238.
- Total mass of Uranium in the core at start of life is therefore 2476.2524 g.
- The JASON reactor power history gives the total integrated power between 1961 and shutdown in 1997 as 25.13 MWh. (The reactor had been operated by Hawker Siddley Nuclear Power Company Ltd. for a total of 1.4 MWh before 1961).
- Fuel inventory and activation level calculations for the standard core and power history were performed by AEA Technology using the codes WIMS, TRAIL and FISPIN in 1997 as part of the decommissioning.
- At shutdown in 1997, these calculations give the total mass of Uranium in the core as approximately 2480 g (calculation rounding), and the total mass of Plutonium in the core as 0.0142 g.

The overall fuel removal phase consisted of several interrelated off-site and on-site preparatory activities, including the procurement, preparation and modification of a UKAEA UNIFETCH fuel transfer flask. Modifications were made off-site to interface the flask to the existing fuel removal transfer equipment held at the College. Extensive trials were also carried out using a dummy fuel module to test the modified fuel handling equipment.

The other major off-site activities included the design, manufacture, test and installation of a range of custom made and proprietary equipment needed to install the UNIFETCH flask in the reactor hall, transfer the Jason reactor fuel modules, remove the flask and subsequently transport it by road to British Nuclear Fuels (BNFL), Sellafield (now BNGSL). After this equipment had been installed and tested, non-active UNIFETCH flask commissioning trials were carried out. The internal fuel transfer route was then proven using a dummy fuel element, witnessed as required by the regulator and the Jason Reactor Safety Committee. Final approvals were obtained

by 11 September 1998. The fuel was then transferred into the UNIFETCH over a two-day period and it was subsequently transported by road to BNFL, Sellafield, on 16 September 1998. The open UNIFETCH flask showing the fuel baskets is shown in Fig. 1 and the UNIFETCH during transport is shown in Fig 2.



Fig 1. Open UNIFETCH Flask showing the baskets containing JASON Fuel



Fig 2. UNIFETCH Flask during transportation

Criticality calculations using MCNP indicated k_{eff} of 0.03046 and 0.67800 respectively for normal dry storage and worst case flooding of the flask.

3. Interim Storage at Sellafield

After transport to BNFL Sellafield the JASON fuel remained in dry storage in the UNIFETCH flask until the end of November 1998 after which each fuel box was transferred into individual storage locations within a designated storage rack in a suitable pond. Each rack was fitted with a

steel sleeve to accommodate the fuel within the rack. All fuel modules were inspected for corrosion and damage to provide benchmarks for future inspections. Criticality calculations were carried out by BNFL using the MONK code and the JASON fuel store fell within the safety criterion for the pond.

The pond chemistry in the JASON store at Sellafield must continue to be carefully controlled to allow for interim storage of JASON fuel. The pH levels of the JASON pond at various intervals between 10/08/1998 and 21/12/2000 are shown in Fig 3. During this time the pH varied but still remained managed within the acceptable band of 5.3 to 9.0. Peak conductivity levels were measured at 8.1 $\mu\text{S}/\text{cm}$. While this is above the level of 1 $\mu\text{S}/\text{cm}$ that suppresses corrosion, it is much lower than the levels that result in rapid corrosion (100 – 600 $\mu\text{S}/\text{cm}$). It was therefore concluded that while corrosion may not be eliminated, the rate of corrosion will be acceptably low.

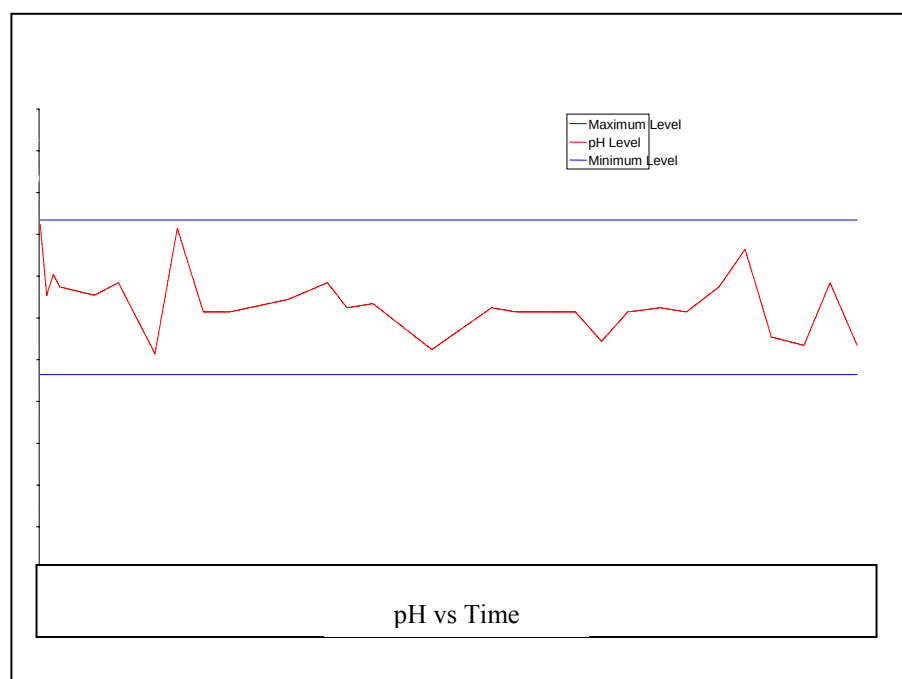


Fig 3. pH in the JASON Fuel Pond –August 1998 to December 2000 [6]

In addition to routine sampling, the JASON fuel has been visually examined using CCTV to monitor any potential corrosion. Fig. 4 shows screen captures from a video inspection of the JASON fuel made in 2001. Fig. 4-C shows a healthy fuel box with no visual corrosion. Fig. 4-B shows a structurally intact fuel box but with pit and crevice nodules on the surface of the lower aluminium location cone. Since the fuel boxes showed no signs of corrosion prior to storage, the difference in corrosion rates could be the result of localised poor chemistry or galvanic corrosion where some of the aluminium cladding has been in contact with the steel sleeves. Additionally, some fuel boxes had been in longer contact with water through more frequent use in the reactor at Greenwich. Consequently some minor corrosion/damage may have been sustained which could have made them more susceptible to corrosion later.

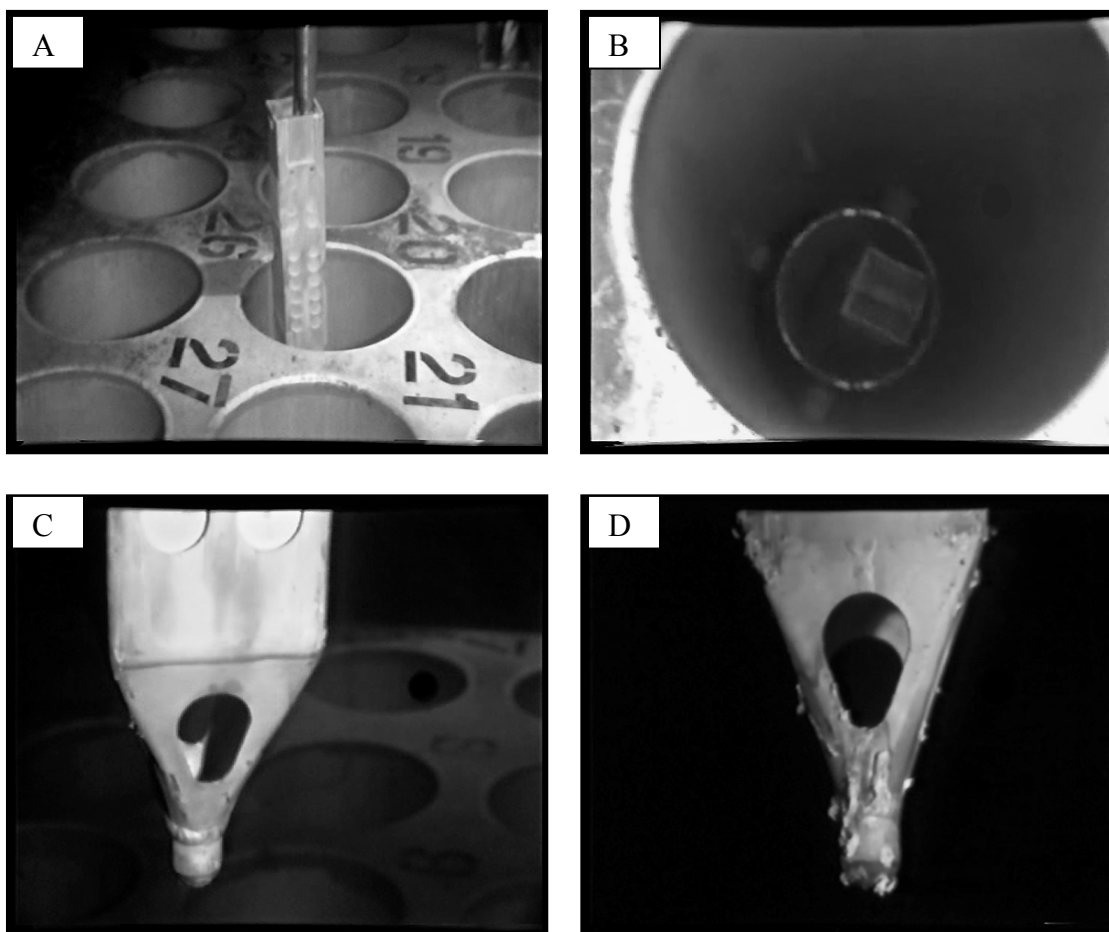


Fig 4. JASON Fuel in the BNGSL Fuel Pond [6]

4. Planning for Transportation to AREVA NC La Hague for reprocessing

The JASON fuel is now being considered for reprocessing at AREVA NC La Hague site, France. It should be noted that the fuel was originally intended to be reprocessed at UKAEA Dounreay before closure of the MTR reprocessing facility.

The various industrial operations that may be carried out during reprocessing are shown in Fig. 5. At present, it is assumed that the JASON fuels will be transported, possibly with another consignment from the UK, in a TN MTR 52 cask designed and operated by TN International (a subsidiary of AREVA NC). The basket inside the cask can hold up to a maximum of 52 MTR used fuel elements, which will be sufficient for this operation [4,5]. TN International will be responsible for the transport except for the nuclear responsibility which will be supported by UK authorities and by AREVA NC in France. TN International will witness and provide guidance to loading operations at selected UK sites in order to guarantee the right application of the TN MTR 52 operating instruction manual (already transmitted to UK's MTR project team) especially for the dryness operations, as well as supplying the necessary equipment and tools to the UK sites.

Once the cask reaches La Hague site, it is planned that the used fuel elements and the AREVA NC Boxes will be unloaded into a single temporary storage basket which capacity is 16 x 4 used fuels. AREVA NC La Hague plant has a long feedback experience in the reception and

unloading of the TN MTR transport cask at the La Hague site, and the AREVA NC Box is fully compatible with the unloading equipment.

The interim storage basket is then placed in one of the interim wet storage pool of La Hague plant. AREVA NC will be responsible for maintaining the integrity of the fuel during this interim storage phase.

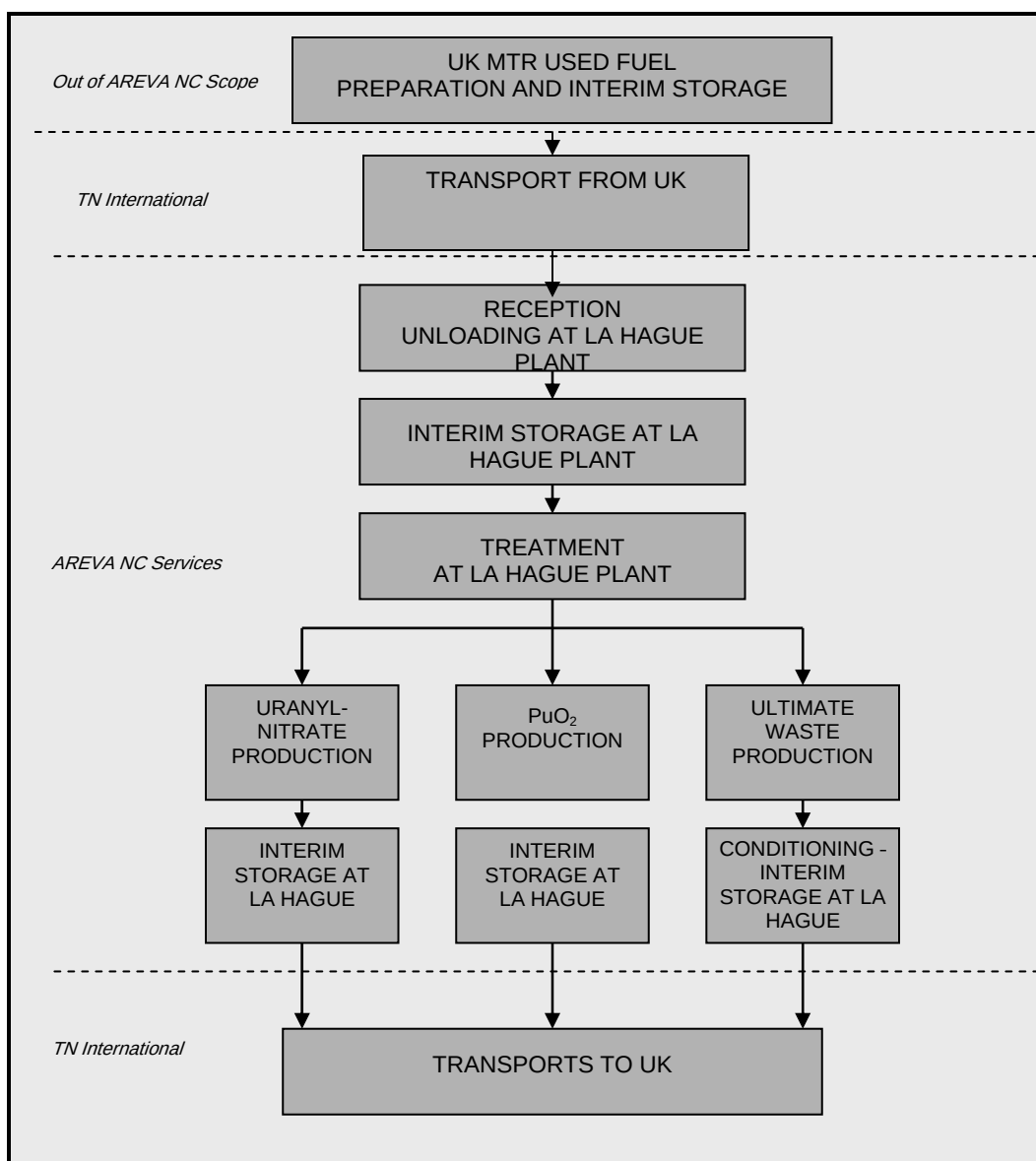


Fig. 5 Scope of Work for Reprocessing of JASON Fuel

At a suitable time, the JASON fuels will be removed from interim storage for treatment. The treatment consists doing nitric acid dissolution batches according to appropriated operating conditions.

The reprocessing operations will comply with EURATOM accountancies rules from fissile material management point of view, therefore different consignments (JASON and other UK

MTR fuel) will be dealt with separately. Other rules state that for dissolution the number of batches required depends on the amount of aluminium present (limited to 140 kg of Al per batch). After dissolution, the liquid solution will be sent to the Accountability Unit for nuclear material balance.

Next, the solution will be mixed with UOx solution for chemical treatment, including; fissile material partition and purification, fission product concentration and vitrification. At the outlet of the reprocessing line the ^{235}U concentration will be less than 1%.

The radioactive waste resulting from the processing of these used fuels will be conditioned into a suitable package for return to UK.

5. Summary

Following decommissioning of the JASON reactor the MTR type fuel has been in interim storage at BNG Sellafield. Review is now in progress for the possible reprocessing of the fuel at the AREVA NC La Hague site. The earliest provisional time scales for this back end of the JASON fuel cycle are shown in Fig. 6.

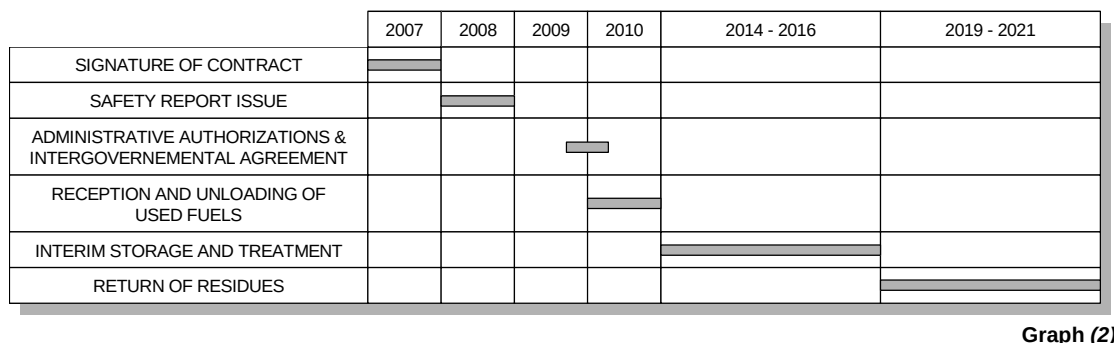


Fig. 6. Time Scales for the Back End of the JASON Fuel Cycle

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Research and Test Reactor Fuel Treatment at AREVA NC la Hague

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ABSTRACT

The La Hague plant has a long successful history of treatment power reactor spent fuel. To accommodate a new contract, the process needed to be adapted to fuel from Research Test Reactors (RTR) taking into account the dimensional characteristics, the chemical composition and the high uranium enrichment of RTR fuel. Characteristics that are very different from those of power plant spent fuel.

Several treatment options resulted from the R&D performed by CEA and conceptual studies and preliminary design proposed by AREVA/SGN. In addition, SGN benefited from previous experience processing RTR fuel near Marcoule. The chosen approach for the detailed design consists of dissolving the RTR fuel in a dissolution pit in one line of dissolver facility (T1B). The process was designed for small RTR fuel elements composed of aluminium and uranium alloy.

“Taxi” baskets containing the RTR fuel elements are transferred from the fuel storage pools, to the feed cell in T1B. The fuel elements are loaded into canisters and transferred to a rack in the general maintenance cell. The RTR fuel canisters are picked up one by one and the fuel element is dropped into a transfer tube leading to the dissolution pit.

The RTR dissolution process is fundamentally different from the continuous power reactor fuel dissolution process and is done in batches. At the end of a batch, the solution is drained from the dissolver and diluted with UOX solutions from line T1A.

The resulting UOX/RTR mixture complies with the specifications for downstream processing operations, including high level waste vitrification, where aluminium is incorporated into glass in accordance with specified limits.

The innovative nature of the process, demanded by the special characteristics of the RTR fuel, required a major qualification program. The main objectives of the qualification program were to validate the dissolver pit concept, to verify basic RTR process data for the T1B production environment and to acquire data showing control of the process.

The first active batch dissolution was successfully performed from June 9th to June 25th in 2005. About 2 tons of Research and Test Reactor fuel coming from Australian and Belgium have been reprocessed by the end of 2006.

Feed-back has been collected to improve the process and get in-depth knowledge about RTR fuel treatment. The feed back is also useful to enlarge the scope of RTR treatment to uranium/silicon or uranium/molybdenum based fuels.

The paper will describe the process implemented in the La Hague plant and the qualification program, as well as the main results from the first year RTR treatment.

1 INTRODUCTION

Research and testing reactors (RTR) are used for nuclear applications in many fields, including medicine, with boron neutron capture therapy, industry, with gauges, detectors, and other devices, research and education with irradiators and calibration sources. All these applications generate radioactive waste. Used Nuclear Fuel (UNF) back-end management has experienced many stops and starts in the past ten years.

Until the end of 1988, US obligated materials were subject to the "Off Site Fuels Policy" which required spent fuel to be returned to the United States and to be reprocessed there. Since this policy terminated on the 31st of December 1988, research reactors operators were then required to implement other management solutions.

At the same time, the Reduced Enrichment for Research of Test Reactors (RERTR) Program was leading to a new Low Enrichment Uranium (LEU) fuel to replace High Enrichment Uranium (HEU) fuel.

Since new LEU fuel was not as easy to reprocess as HEU fuel (the LEU fuel is made of silicide, whereas the HEU fuel is made of aluminide), a new US spent fuel return policy was introduced in early 1996 for all research reactors converted (or that had agreed to be converted) to LEU, and for reactors operating with HEU for which no suitable LEU was available. This policy covers all the spent fuels discharged.

At the end of this return program, each operator will again be fully responsible for its spent fuel. For the ultimate back-end management, there are three options [1] :

- ① Interim storage
- ② Direct disposal
- ③ Treatment-conditioning by treatment

The interim storage option does not constitute a reliable solution, while some research reactor operators have been confronted with corrosion and material degradation problems in existing facilities. Extended storage of RTR fuel would obviously require extensive R&D programs, as well as new facilities designed for long term storage. Most significantly, this option does not provide a definitive solution.

The direct disposal option entails several unsolved difficulties. First, one has to ensure that the enriched uranium content will not lead to criticality hazards through long term processes as selective leaching. Moreover, RTR spent fuel is generally considered as unstable (high corrosion rate, hydrogen build-up) under the conditions of a geological repository. It requires watertight and durable conditioning on a geological time-scale, for which no satisfactory solution has yet been found. Finally, direct disposal remains a "virtual" solution that has never been implemented.

The treatment option avoids the above difficulties because it produces residues which are suitable for direct disposal. The 40 years of experience gained at AREVA NC's treatment site in La Hague demonstrates the industrial expertise achieved in commercial treatment.

2 PRINCIPLES OF TREATMENT : THE EXAMPLE OF LA HAGUE

Treatment has two main objectives:

- ① Recover the recyclable materials (mainly uranium and small quantities of plutonium),
- ② Generate final waste according to their potential hazards, in order to dispose them safely for the environment.

Treatment at the La Hague complex, in the UP2-800 and UP3 plants, uses the PUREX process, including the following steps (see flow sheet in figure 1) :

- *Transport* of fuel to the plant and *cooling* in storage ponds. This cooling, or "deactivation", decreases the radioactivity of the fission products substantially.
- *Shearing and dissolution* of the fuel, followed by *clarification* of the liquor generated:

The first treatment operation consists of stripping the fuel rod to prepare it for chemical attack.

The process employed for zircalloy cladding rods is cutting them into pieces with a shearing machine. At La Hague, the shearing machine is placed above a continuous dissolver, and the rod pieces fall into a perforated basket, which, placed in the dissolver, allows the selective dissolution of the oxide in nitric acid without attacking the hulls. At the end of the operation, the hulls are removed and sent in a workshop for compacting and conditioning as a solid Medium Active Level Waste (MALW).

Depending on the fuel type, it is possible that some insoluble products may remain after dissolution. This is the case of oxide fuels for which the insoluble particles are made of cladding residues and metallic inclusions. These solids, which would hamper further purification steps, are removed from the solution by centrifugation. A centrifugal clarifier has been selected for La Hague because it provides good efficiency with high throughput.

➤ *Uranium and plutonium splitting and purification by a liquid-liquid extraction process :*

Basically, extraction consists in transferring a solute from one liquid phase to another one that is not miscible with the first. This operation enables separation of salts whose suitability for extraction by a given solvent is different.

For the extraction of uranium and plutonium, tri-butyl phosphate diluted in hydrocarbons has been universally adopted. In the extraction operation, most of the fission products and actinides, except U and Pu, remain in the aqueous phase. Scrubbing by nitric acid improves the separation by stripping most of the fission products entrained by the solvent.

Several extraction cycles of the clarified liquor, in pulsed columns, mixer-settler banks, or centrifugal extractors are necessary to meet the end-product specifications. At the end of these cycles, different kinds of solutions are generated:

- a solution containing specifically the uranium
- a solution containing specifically the plutonium
- raffinates containing the fission products and the minor actinides
- the solvent, which is regenerated by a treatment with sodium carbonate followed by caustic soda, and then recycled.

➤ *Final conversion of uranium and plutonium to end-products :*

The uranium solution is concentrated by evaporation, stored, and eventually converted to UF₆ for a new isotopic enrichment.

In the same way, the plutonium is precipitated as an oxalate salt by the addition of oxalic acid. This salt is then filtered, dried and calcinated to form the PuO₂ oxide that is used to make the MOX fuel. The mother liquor is concentrated and recycled.

➤ *Management and treatment of process waste :*

The process waste comprises :

- The hulls, produced during shearing and dissolution operations, which are compacted in a canister, and intended for final disposal.
- The High Activity (HA) liquid waste made up of solutions containing :
 - the insoluble particles from the clarification (fines),
 - the fission products and minor actinides separated during the extraction process,
 - the concentrates generated by evaporation of the aqueous acidic process sewage in an acid recovery unit. Acid generated in this unit is recycled in the process, and distillates, with very low activity, are discharged into the sea.
- The various streams, except the fines, are concentrated and generate the HALW concentrates which are stored in large vessels fitted with cooling and pulsation devices. The concentrates are then mixed with the fines and treated in a vitrification facility to form

- a glass matrix with high resistance to leaching. Today, this matrix appears to be the most suitable and safer packaging for long term storage.
- The gases, which are collected according to type and level of activity, washed and treated on specific traps to recover elements such as iodine, and then filtered through high efficiency filters before discharge through a stack.

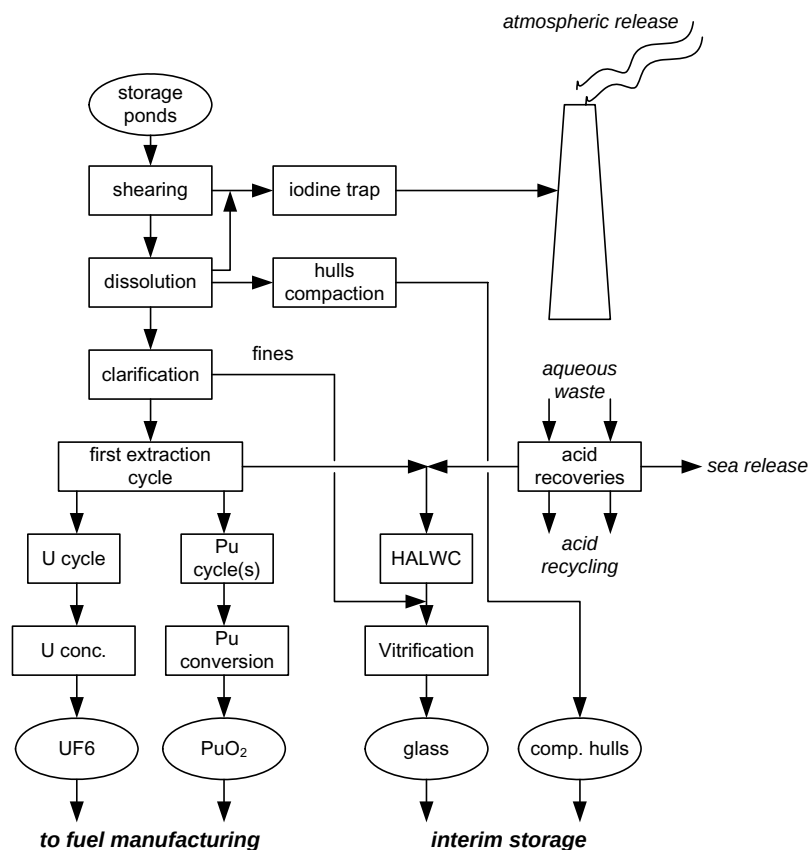


Figure 1 : LWR process scheme at La Hague

Treatment plants such as La Hague are designed to be operated for very long periods. During their lifetime, they will have to reprocess fuel with changing characteristics, although they were primarily designed for fuel from light water reactors. That can include LWR with an increased burn-up, but also different fuels such as RTR which are highly diverse in terms of weight, shape, and composition, and therefore require high flexibility of back-end services. Fortunately, in La Hague, design adaptations can be realised thanks to evolutionary plant design and the significant experience in performing modifications even in the active part of the plant.

3 COMPARISON OF POWER PLANT FUEL AND RTR FUEL

The characteristics of RTR and power plant spent fuel elements are very different as shown in the following table:

Characteristics	Power plant UOX fuel Typical value	RTR fuel Typical value
Length	4 m	1 m
Weight	500 kg	5 kg
Cladding	Stainless steel + zirconium	Al alloy
Enrichment 235U	4%	95%
Burn-up	< 45 MWd/tHM	14 to 680 MWd/tHM

Table 1 : Comparison of power plant and RTR fuel elements characteristics.

The different characteristics of those fuel elements demanded adaptations of the process :

- handling operations to fit with the small dimensions of RTR fuel elements,
- dissolution chemistry to deal with the cladding that is soluble in nitric acid,
- criticality to adapt to the high enrichment of RTR fuel elements.

4 RTR PROCESS DESCRIPTION

4.1 General description

The chosen approach consists of dissolving the RTR fuel element in a dissolution pit in the existing dissolver. The dissolution pit specially design for RTR fuel dissolution is placed in the dissolver instead of an air-lift not needed for RTR process configuration (it aims at recycling hulls in UOX process in the perforated baskets).

The fuel elements are transferred in "taxi" baskets from the pool storage till the maintenance cell of shearing – dissolution facility, which was the most suitable place for loading of fuel elements. The baskets are the one-used to store BWR spent fuel. They have been adapted to the shape and the high enrichment of RTR spent fuel. The fuel elements are stored in a rack in the cell maintenance until they are loaded one by one in the dissolution pit.

Unlike power plant fuel elements, the RTR fuel elements are completely dissolved in nitric acid. The solution of dissolution is circulating between three existing tanks to optimize the volume of each batch of dissolution. The required number of fuel elements in a batch is assessed to reach a suitable final concentration of aluminium to manage the risk of precipitation of aluminium nitrate. During the dissolution, the couple acidity / aluminium concentration is controlled to limit risks of corrosion and precipitation of aluminium nitrate. At the end of a batch the solution is drained from the dissolver and diluted with UOX solution from another dissolution unit of the plant.

The resulting UOX/RTR mixture complies with the qualification for back-end operations at UP3: particularly the uranium enrichment and the plutonium spectrum.

For the chemical elements that were not present (or in traces in) UOX dissolution solution (such as Al), it was checked that they were not changing the efficiency of the process and that their presence was compatible with the products specifications.

4.2 Qualification program

4.2.1 Necessity of qualification program

The RTR treatment affects every process in use at UP3, from the storage fuel to vitrification, including the analytic laboratories and waste processing.

The special characteristics of RTR demanded particular demonstration in each field of studies as shown in the following examples:

Mechanical field: Gravity feeding of the dissolution pit from the maintenance cell had to be demonstrated for a long term use,

Lay-out : The implementation of the pit in the dissolver was a challenge. The space available was just a little bigger than the dimension of the pit.

Chemical field : The risk of cristallisation of aluminium had to be managed properly

Safety, criticality: the management of the risk of criticality in the dissolution pit had to be demonstrated, taking into account the high uranium enrichment.

4.2.2 Steps of the qualification program

The main objectives of the qualification program were to validate dissolver pit concepts, to verify the basic RTR process data for the T1 production environment, and to acquire data demonstrating control of the process.

A major test program was conducted in two phases at the SGN development and testing laboratory (HRB):

- the development testing phase involved dissolving simulated fuel elements in full-scale mock-up of a fully-instrumented pit made out of glass,
- the qualification testing phase involved dissolving simulated fuel in a pilot unit consisting of a full-scale pit and dissolver in an environment as representative as possible of T1 environment.

Those testing phases allows :

- to improve the dissolution pit concept for instance by developing a device allowing the follow-up of fuel loading and dissolution,
- to ensure the efficiency of the process for instance to validate the chemical kinetics of dissolution,
- to manage risks such as the risk of dispersion of fuel material out of the dissolution pit.

5 INDUSTRIAL FEED-BACK OF RTR TREATMENT

Thanks to a good qualification of the process and a training program, the first active dissolution of RTR fuel elements unfolds successfully in June 2005. About 2 tons of RTR UAI fuel have been processed by the end of the year 2006.

In addition of the usual control of the process, special analysis and observations were made during the first batches to ensure the efficiency of the process and gain knowledge on RTR processing.

5.1. Insoluble material

Special analysis have been made to ensure the total dissolution of RTR fuel elements. The results are satisfying as:

- The amounts of residual suspension material in RTR solution are low and are comparable to those in clarified UOX solution.
- At that time no significant amounts of deposit have been observed in the dissolving vessels.

5.2. Corrosion

As expected because of higher nitric acid and temperature, the corrosion rate of stainless steel equipments placed in the dissolving unit is higher than the corrosion during UOX dissolution. Therefore, a special program of measurement is implemented to check the acceptability of corrosion rate. The first measurement of dissolution pit confirms a significant life time for this equipment which has been designed and implanted to be easily replaced.

5.3. Dissolution rate

It appeared that the dissolution rate is higher than expected allowing more flexibility in operation. Thanks to the gain in operating skills for both mechanical and analytic operations, the global rate of dissolution operations allows important margins towards the back-end of the process.

Improvements have been made to gain in operating efficiency and reduce the amount of effluents generated. The dissolution process has been successfully performed on two types of RTR UAI fuels (ANSTO, BR2). AREVA NC will keep on gaining knowledge on RTR UAI fuels treatment in the coming years.

5.4. Back-end operation

The global rate of treatment is limited by the amount of aluminium compatible with high active glass wastes. Research and developments are going on to increase the amount of aluminium coming from RTR dissolution in the glass by decreasing the amount of aluminium oxide coming from non active glass used in vitrification process. The industrialisation of the R&D program is expected in 2008.

6 PERSPECTIVES

The RTR treatment is limited by now to RTR fuel composed of uranium aluminium alloy and small enough to fit with the dissolution pit size. Studies are going on to adapt the process to :

bigger UAI fuel,

- o Studies are consisting in finding a way to cut the fuel elements taking into account the criticality risks

USi fuel (fuel made of aluminium alloy mixed with U₃Si₂ particles)

- o the process is adaptable to USi fuel but some issues are still under progress to manage properly the siliceous component in the process and to validate a sufficient treatment rate,

UMo fuel (fuel made of aluminium alloy mixed with UMo particles)

- o UMo Fuel development is still under progress at the same time promising research and development about its treatment is made.

5 CONCLUSION

Treatment RTR fuel is an important issue for Research and Testing Reactors Operators. AREVA NC la Hague plant is now equipped to reprocess some RTR UAI fuel and has begun actual operation since June 2005 with success. This new functionality required studies and qualification to accommodate the different characteristics of the RTR fuel elements in comparison with power plant reactor fuel elements and is now entering in an industrial maturity phase.

Studies are going on to extend the type of RTR fuel that could be reprocessed and to increase the capacity of treatment.

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Reprocessing of Research Reactor Spent Nuclear Fuel at the PA "Mayak"

Paper

The production association "Mayak" was founded in 1948 within the framework of Soviet Union defense program. It's situated in the Ural, in Chelyabinsk region.

Nowadays PA "Mayak" is a big complex of industrial plants and production departments. The enterprise management is consolidated.

PA "Mayak" has several kinds of activity. The main of them are reactors, radio-chemical, chemical-metallurgical and radioisotope productions.

The first Russian reprocessing facility, known as RT-1, was started on the radio-chemical plant base in 1977. Nowadays RT-1 remains the single reprocessing plant in Russia.

The fact is that at present there is a full-scale reprocessing in France, Great Britain and Russia. All these reprocessing facilities use similar technological processes, such as: water-pool storage of spent nuclear fuel (SNF), shearing, nitric acid dissolution, extraction by means of Purex-process, vitrification of High Level Wastes (HLW) and others. However, each enterprise has its own technological features, sometimes very different from others.

The main characteristic property of RT-1 is a broad spectrum of reprocessing spent nuclear fuel.

The following spent fuel types are reprocessed here:

- SNF of PWR reactors (WWER-440) and FB reactor (BN-600);
- SNF of transport ship reactors;
- Production reactors SNF;
- Research reactor spent nuclear fuel.

As it was mentioned above, the world-known technological processes are used at RT-1. But there are some distinctive features.

There are the following distinctive features of RT-1 technology:

1 Universality of the three technological lines which allows not only reprocessing of various SNF kinds, but also to implement the combined reprocessing of some types of them.

2 Extraction of neptunium during SNF reprocessing which is used to implement its separate storage and for radioisotope production.

3 Target enrichment of recycled uranium is achieved by mixture of uranium from reprocessing SNF of various kinds.

4 Extraction of various elements (such as cesium, strontium, promethium and etc.) which are used for radioisotope production.

Basic technological processes for research reactor SNF are:

Shipment of SNF. Shipment of research SNF is accomplished by rail in transport packages such as TUK-19, TUK-32, TUK-128.



TUK-19

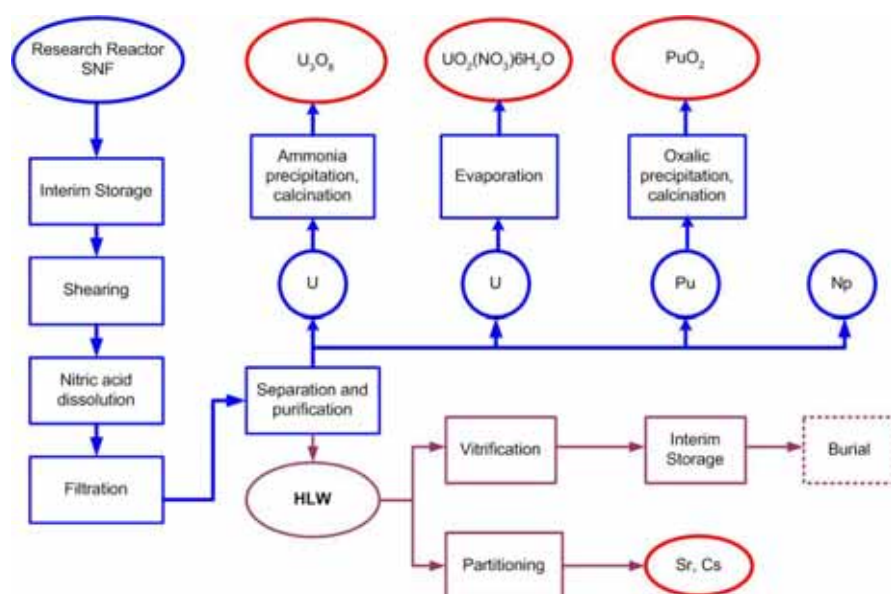


TUK-128

"Skoda" VPVR/M transport packages will be used in future for shipment of spent fuel from the Czech Republic. All above mentioned transport packages have Russian certificates for package design and may be validated in other countries in accordance with safety rules of TS-R-1 (IAEA).

Interim storage of SNF. Transport packages delivered to the plant are unloaded in a hot cell by "dry" way. SNF is placed into the water pool. The layer of water of more than 3m high guarantees reliable biological safety. As a rule, the duration of interim water-pool storage of research reactor SNF is up to 2 years.

Reprocessing. The first stage of reprocessing is shearing of spent nuclear assemblies into pieces of less than 60 mm. After that pieces are dissolved by nitric acid in a cycling dissolver. After filtration the solution of spent fuel is separated by Purex-process into uranium, plutonium, neptunium solutions, and liquid radwastes. Multistage mix-settle extractors with air mixing are used for Purex-process.



Technological scheme

Target products of reprocessing are:

- hexahydrate of uranyl nitrate obtained by evaporation of nitric acid uranium solution;
- triuranium octaoxide obtained by means of ammonia precipitation and calcination;
- dioxide of plutonium by oxalic precipitation and calcination.

All recycled uranium is supplied for nuclear fuel production.

Extracted plutonium is placed into a special storage.

Besides above-mentioned products the technological process provides extraction of neptunium and iodine for their isolation.

Krypton (Kr-85), strontium (Sr-90), cesium (Cs-137), americium (Am-241), promethium (Pr-147) and other radionuclides are extracted for radioisotope production.

Safety of radioactive wastes management is one of the most important tasks of RT-1 activity.

Since 1987 vitrification facility has been used in RT-1. The main task is immobilization of high- and middle-level liquid wastes into phosphate glass.

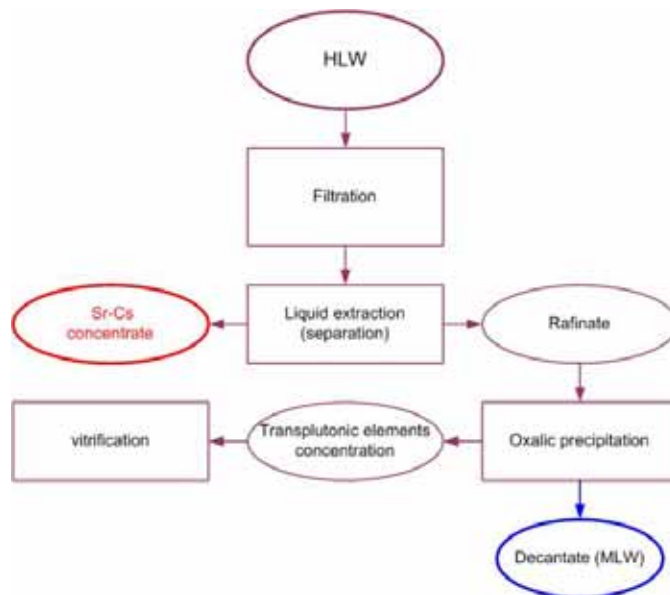
The main technological unit of this facility is a furnace with direct electric heat. Its production capacity is 500 liters per hour of reprocessed solution. This is the continuous action furnace with a working volume of 5,5m³. The results of vitrification facility operation are shown in the next picture.

№ Furnace	Period	Volume of immobilized HLW, m³	Glass quantity, t	Total activity, x 10⁶ Ku
1	1988-1989	1 215	161,2	3,864
2	1991-1997	11 474	2 190,8	282,728
3	2001-2006	7 974	1 818,7	171,957
4	2007-	—	—	—
Total	—	20 663	4 170,7	458,549

The total quantity of immobilized liquid wastes is 20663 cubic meters. At present the 4th furnace is being put into operation.

The first world full-scale partitioning facility was founded at RT-1 in 1996. The main reason of it was that a lot of high level wastes with complicated composition were accumulated. Direct effective vitrification of such wastes was impossible.

Partitioning technology uses processes of strontium and cesium extraction by cobalt dicarbollide as extractant and oxalic precipitation for extraction of transplutonic elements and rare earths. Nominal capacity is 180 liters of liquid wastes per hour. Nowadays partitioning is used for strontium and cesium extraction from high level wastes for the radioisotope production.



Partitioning scheme

Besides the main production activity PA "Mayak" solves problems related to past military activity. Such as:

- treatment of accumulated liquid high-level wastes;
- closing of the technological reservoir named "Karachay";
- decommissioning of old facilities and constructions.

Research reactor SNF reprocessing directly. More than 20 years the RT-1 has been receiving nuclear fuel for processing from research reactors located on the territory of the former USSR, nowadays - Russia. At present on a regular basis SNF is received only from Russian research reactors.

Before disintegration of the USSR SNF of research reactors was taken from Latvia, Uzbekistan, Ukraine, Kazakhstan.

The range of SNF to be processed includes mainly fuel compositions on the basis of aluminium and magnesium. But it can be extended if necessary. For example, the technological process can be adapted to the reprocessing fuel on the basis of metal uranium from "Vinca" institute (Serbia).

It's necessary to say about the problem of shipment and reprocessing of damaged or leaking spent nuclear fuel. Some institutes have a big quantity of leaking fuel because of its long water storage. There are two tasks of safety in this case. The first: safety ensuring during shipment, and the second: safety during interim storage at the reprocessing plant.

The most reliable way to ensure safety at all stages is hermetical canning of damaged (leaking) spent fuel before shipment. It is the most reliable way but rather difficult technically.

Shipment and reprocessing of leaking spent nuclear fuel without hermetical canning will be possible if two safety reports are prepared: for the shipment stage and for the

stage of management at the reprocessing plant. The first is needed for certification of the shipment. The main problem at the reprocessing plant is interim water storage of damaged (leaking) fuel. Based on the safety report, time of the interim storage will be limited if it is necessary.

Shipment and reprocessing of physically damaged fuel may be done only in cans. For example, canning must be provided for SNF from Vinca institute.

Under the auspices of the "Agreement between the Government of USA and the Government of the Russian Federation concerning cooperation for the transfer of Russian-produced research reactor nuclear fuel to the Russian Federation" in the first half of 2006 transportation of research reactor SNF from Uzbekistan was made. At the end of 2006 this fuel was reprocessed. As a result more than 60kg of high-enriched uranium was transformed into the low-enriched category and was prepared for use at nuclear power plants.

The Czech Republic and Latvia are the next countries for shipment of SNF in the framework of the Agreement.

The U. S. Department of Energy / Idaho National Laboratory's Research Reactor Spent Nuclear Fuel Acceptance Program

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I. PROGRAM OVERVIEW

The Department of Energy's Idaho Operations Office (DOE-ID) has successfully implemented a management program that is responsible for the safe and cost – effective transportation and storage of TRIGA spent nuclear fuel at the Idaho National Laboratory (INL).

In May 1995, a Record of Decision (ROD) on the Environmental Impact Statement (EIS) for Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory (now known as the Idaho National Laboratory) Environmental Restoration and Waste Management Program was published. Based on that Record of Decision, the United States Department of Energy, in consultation with the Department of Navy, adopted a policy regarding the management of existing and reasonably foreseeable inventories of spent nuclear fuel through the year 2035. The spent nuclear fuel inventory covered by this policy is generated from many different sources: DOE reactors, other government agency and university research reactors, and foreign research reactors. The policy consisted of a Department-wide decision to regionalize spent nuclear fuel management by fuel type at three DOE sites, with the INL being responsible for several spent fuel inventories, including all TRIGA research reactor spent fuel. The timing of the transport of the spent fuel between the respective sites is prioritized and scheduled based on the needs of the shipping site, fuel condition, facility availability, safety, safeguards and security concerns, budget and cost considerations, and transport logistics.

This Record of Decision was amended in late February 1996 to reflect requirements identified within an October 16, 1995 Settlement Agreement among DOE, the State of Idaho and the Department of Navy pertaining to spent nuclear fuel shipments into and out of the State of Idaho. In essence, shipments of spent nuclear fuel into the State of Idaho are restricted, and tied to completion of various INL environmental restoration and radioactive waste management activities that are important to the State of Idaho.

Specific to foreign research reactor spent nuclear fuel, and in support of the 1995 Programmatic Spent Nuclear Fuel ROD, DOE then published a Record of Decision in May 2006 to implement a new foreign research reactor (FRR) spent fuel acceptance policy as identified within the Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel Environmental Impact Statement. This ROD supported the DOE

regionalized spent fuel management policy while providing additional information regarding the shipping of FRR spent nuclear fuel containing uranium enriched in the United States back to the United States for spent fuel management. This ROD specified that the FRR facilities were required to stop irradiating their fuels by May, 2006 and ship it to a U.S. DOE facility by May, 2009. However, the Record of Decision was amended in 2005 to extend the Foreign Research Reactor Spent Fuel Acceptance Program to 2016, and 2019 respectively.

Table 1 provides a listing of all of the facilities included within these programs, while Figure 1 provides a map identifying the countries with TRIGA foreign research reactor facilities that are eligible to participate.

II. INL RESEARCH REACTOR SPENT FUEL RECEIPT PREPARATIONS

During the first ten years of the Department's Spent Nuclear Fuel (SNF) Acceptance Program, DOE-ID has supported the DOE FRR program with 6 shipping campaigns, involving 8 different countries, with 15 casks containing approximately 1500 TRIGA spent fuel assemblies; and the U.S domestic program with 7 shipping campaigns, involving 7 different research reactor facilities, with 14 casks containing approximately 700 spent fuel assemblies. All shipments have been safely received and stored at the Idaho National Laboratory (INL). Each shipment, and fuel type received, has gone through a rigorous pre-shipment preparation process that includes fuel characterization and cask shipping data, in support of criticality and facility specific safety reviews, culminating in an "authorization to ship" from the DOE-ID.

I.A. Receipt Preparations

DOE-ID, in conjunction with the INL's Environmental Management contractor, CWI, has developed a disciplined process for completing the required activities to approve the safe receipt and storage of SNF at the INL. Figure 2 depicts the flow of the basic process for prospective program participants. Two years in advance of the planned fuel shipment to the INL, an agreement on the terms of the shipment is reached. The process then proceeds in two parallel paths, an administrative path and a technical path. The administrative path is represented by the activities outlined in black on the left side of Figure 2 and involves the formalizing of agreements, schedules and specific terms of the shipment. The areas outlined in red, on the right hand side of the figure, deal with the specific INL activities that provide the technical bases to support the safe receipt and storage of the SNF.

I.A.1 Spent Nuclear Fuel Characterization

The first, and most important, step in the receipt preparations process is to secure an accurate characterization of the fuel. Characterization activities are segregated into three groups, which, ideally, are worked in parallel and are completed at least nine months prior to fuel handling at the INL.

- First, the characterization of fuel data required within the Appendix A attachment to the contract; the data forms are known as the Fuel and Packaging - Required Shippers Data (F & P RSD). The fuel data is to be included in Section III of the F&P RSD form.

- Second, a team of DOE/INL personnel visit the reactor facility to assess the fuel and reactor operating history and operating condition.
- And third, the final fuel characterization activity involves a review of the cask design/certification parameters, and how the fuel is to be handled. This packaging information and data is also required within Appendix A of the contract, and is included as Section IV of the F&P RSD form.

The following paragraphs provide a more detailed description of these tasks.

Data Collection

Fuel data is collected and documented per the guidance provided within the Appendix A, which is, by contract, a part of DOE's agreement with the reactor facility. The reactor operator provides the Appendix A information to the INL for use in validating compliance with INL facility operations safety and authorization basis. The INL uses the reference documents, such as drawings, fuel fabrication reports, reactor operating logs, facility safety analysis reports, and others, to review the submitted data. The Appendix A is approved when all of the INL review comments have been resolved in the comment resolution cycle. Accuracy and timeliness are important factors during this cycle and are essential for the success and cost effective execution of each shipment.

Thoroughness and accuracy in preparation of the Appendix A is important for several reasons. First, the technical information provided, including drawings and other reference material, is used by the INL as the basis for safety and operational reviews to ensure safe receipt and storage of the fuel in the existing dry storage facility. Secondly, this fuel data provides the basis for the cask vendor to verify and/or modify the cask license certificate for the transport of a particular fuel. Inaccurate data may delay the cask certification process, with the potential for adverse schedule impacts. Finally, thorough and accurate Appendix A data ensures that any fuels will be properly characterized for ultimate disposition in a future permanent repository.

Timely submittal of the Appendix A document is also very important. Ideally, the final Appendix A should be approved at least 6 months prior to scheduled fuel loading. Historical trends indicate that about 3 to 5 months are required for the initial INL review of the Appendix A fuel data, and involves critical site resources and communications with the research reactor operators. The initial document should therefore be submitted approximately one year in advance of the scheduled fuel loading. Early finalization of the Appendix A will allow ample time for the INL to complete its safety bases and operational reviews and implement any new facility modifications, process changes, or special training of fuel handling personnel that may be required to safely receive, unload, and store the fuel. If cask license reviews and revisions are required for transport of a particular fuel, additional time may be required. Cask vendors, foreign government competent authority representatives, and the U. S. Nuclear Regulatory Commission (NRC) have taken the position that cask license reviews may not begin until the Appendix A document has been finalized. Depending on the extent of the evaluations needed to review license submittals, the U. S. NRC and Department of Transportation approval process could range from 8 weeks to 12 months. Therefore, late submittals of Appendix A's have the potential to result in significant delays or cancellation of shipments because of licensing issues.

Inspection/Assessment Visits

A team of DOE and INL representatives may visit the reactor facility for those facilities that are preparing a shipment. These visits are scheduled to occur 12 to 18 months in advance of the intended INL receipt date in order to initiate the exchange of technical information and to identify and resolve early concerns. Contracts between DOE-ID and the reactor facility are finalized, clear understanding of all INL receipt requirements is ensured, and preliminary fuel shipment logistics are identified during these visits. If necessary, the INL will also inspect the fuel at this time for structural integrity, evidence of corrosion, ease of handling, fuel cropping or canning needs, and any other indicators that could possibly affect receipt, handling and storage at the INL dry storage facility. (The INL has remote video inspection and recording capabilities that have been used in the past for specific fuel inspections, which could be used, as needed, for future inspections as well). The facility assessments also cover a review of the radiological and/or industrial work activities to help ensure a safe work environment. The visits provide an excellent opportunity for the reactor operator and the INL representatives to discuss the Appendix A Fuel and Packaging RSD forms and review the reactor operating history to support the timely resolution of issues.

Cask and Fuel Handling Assessment

Once the cask to be used for the shipment is chosen, the INL will initiate an independent review of the various documents that describe the cask, its licensed contents, and its handling. The cask's physical dimensions and handling methods are reviewed against the capabilities of the INL receipt facility. Areas of concern are either resolved by modifying the INL equipment, or are brought to the attention of the research reactor and cask vendor for mutual discussion and resolution. The fuel data compiled in the Appendix A document is compared to the licensed contents specified within the cask Certificate of Compliance to determine if any license revision is required. Ongoing communications provide the feedback mechanism to discuss any potential discrepancies. The cask vendor is also very much involved in working through any problems.

The fuel handling assessment includes: the internal cask "basket" or "shipping can", which will contain the fuel within the cask, the cask and basket loading configuration, and any specific cask or fuel handling tools that will be used during cask unloading and storage activities at the INL. Often, assistance from the reactor facility is needed to properly determine the correct handling tools. Any equipment that will be used for handling or storage at the INL will also require design and fabrication reviews by INL quality assurance personnel to ensure safe handling of the fuel within the INL receipt facility.

I.A.2. Pre-shipment INL facility activities

Once the Appendix A fuel data is finalized, fuel inspections are complete, and the cask/fuel loading and shipping configurations have been determined, the information is passed on to the INL receipt facility safety analysis and operations staffs. The facility safety personnel perform the necessary evaluations to ensure that the fuel can be received, unloaded and stored without the possibility of a criticality incident or an "un-reviewed safety question". The operations teams ensure all fuel handling facilities, procedures and training have been adapted to the specific fuel receipt and that the fuel storage location has been properly designated.

Upon completion of the INL criticality and facility safety analysis evaluations, (which are conducted in parallel) the facility will identify that it is ready to receive and store the shipment of TRIGA spent nuclear fuel and that:

- The facility criticality and safety bases will not be compromised;
- Cask handling issues, including any facility modifications, have been resolved and implemented.
- If a specific cask does not provide them, a set of spare tools is staged to minimize delays in the unloading if the SNF; and,
- All receipt, unloading, and storing procedures are implemented, and all facility operators and supervisors are trained on these procedures.

I.B. Authorization to ship

All of the information collected and reviewed during the fuel characterization phases, and the subsequent INL pre-shipment activities provides the technical basis for DOE-ID to provide to the research reactor facility an “Authorization to Ship” letter, allowing the shipment process (loading and transporting) to commence. The INL process to achieve this technical justification was established to ensure the safe and cost effective receipt and storage of spent nuclear fuel.

III. CONCLUSION

This receipt preparation process provides a good foundation for the success of a shipment, for both the research reactor facility and for the DOE Spent Nuclear Fuel Acceptance Program. The preparations for the safe and cost effective shipment of spent nuclear fuel to the INL start well in advance of the actual receipt date. Much effort at the INL is spent executing the technical and operational reviews, analyses, and evaluations that support the INL receipt process. For these reasons it is important to maintain a disciplined approach and schedule to ensure all pre-shipment preparation activities are initiated and completed in a timely manner, with accurate data.

Table 1
INL Potential Shippers List

Domestic Shippers:

University Shippers:

Cornell University*
 Kansas State University (KSU)
 North Carolina State University (NCSU)
 Oregon State University (OSU)
 Pennsylvania State University (PSU)
 Reed College
 University of Arizona (UA)
 University at Buffalo, State University of New York (SUNY)*
 University of California-Davis (UC-Davis), formerly McClellan Air Force Base reactor)
 University of California-Irvine (UC-Irvine)
 University of Illinois (UI)*
 University of Maryland (UM)
 University of Texas (UT) at Austin
 University of Texas A&M*
 University of Utah (UU)
 University of Wisconsin (UW)
 Washington State University (WSU)

Non-University Shippers:

Aerotest, Aerotest Research & Radiobiology TRIGA Reactor (ARRR)
 Armed Forces Radiobiology Research Institute (AFRRI)
 Argonne National Laboratory-East (ANL-E)
 Argonne National Laboratory-West (ANL-W) (now known as the Materials and Fuels Complex)
 Babcock & Wilcox (B&W), Lynchburg, Virginia
 DOW Chemical
 General Atomics (GA)*
 Hanford (HR)
 Fort. St. Vrain*
 Oak Ridge (OR)*
 Sandia National Laboratory (SNL)
 Savannah River Site (SRS)
 Veterans Administration (VA)*
 United States Geological Service (USGS)
 West Valley (WV)*

International Shippers (Foreign Research Reactors):

<i>High-income-economy</i>	Austria	Germany*	Japan*	Taiwan
<i>countries -</i>	Finland	Italy*	Slovenia*	United Kingdom (England)*
 <i>Other-than-high</i>	Bangladesh	Indonesia*	Mexico	South Korea*
<i>income economy</i>	Brazil	Malaysia	Philippines	Thailand
<i>countries -</i>	Democratic Rep.	Romania*	Turkey	
	of Congo			

*Identifies facilities that have made shipments to the INL

Countries with Spent Nuclear Fuel Eligible for Shipment to the INL



FIGURE 1

INL SNF RECEIPT PREPARATATION FLOW DIAGRAM

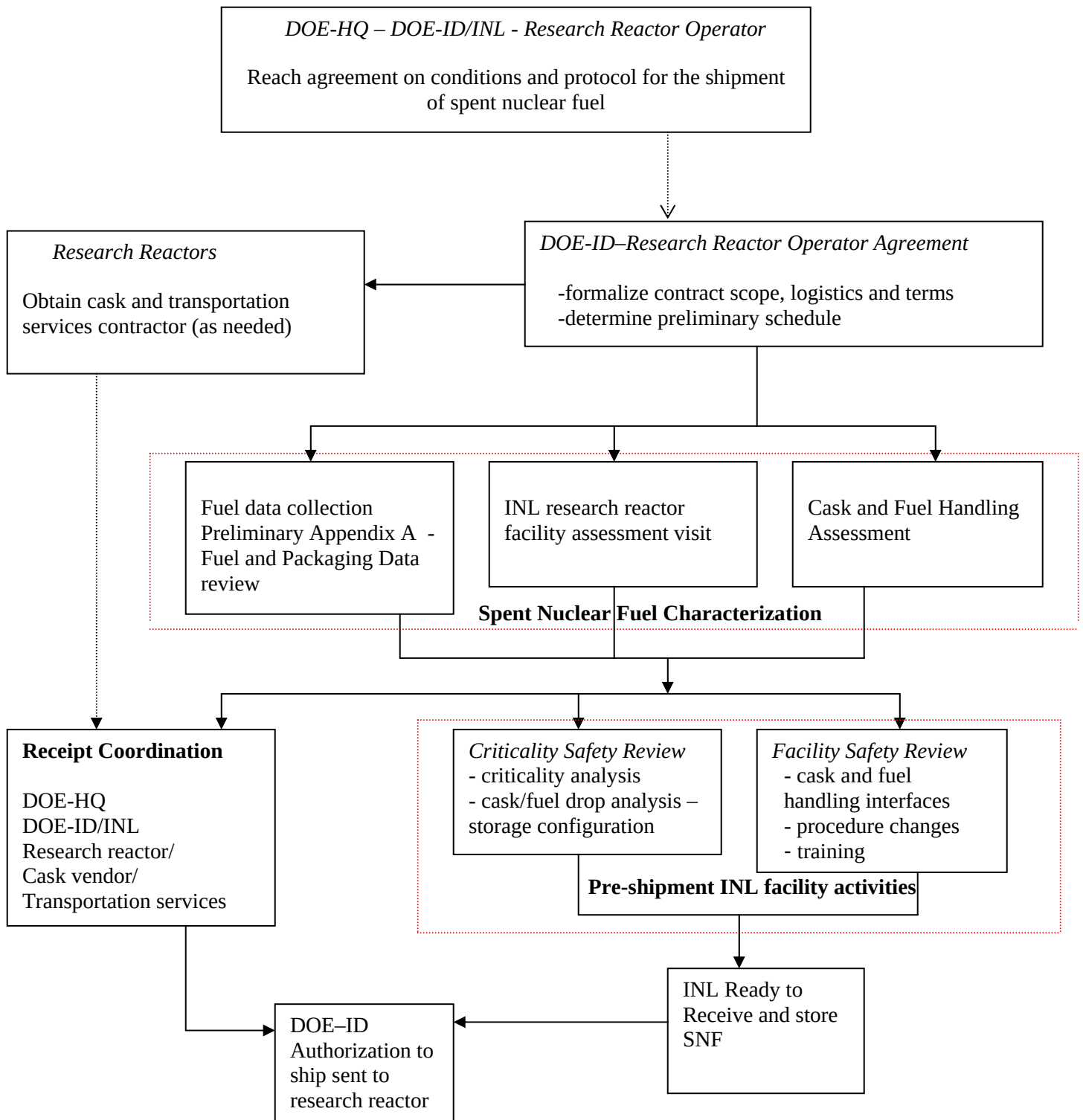


Figure 2

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INVESTIGATIONS TO THE BEHAVIOUR OF RESEARCH REACTOR FUEL ELEMENTS IN REPOSITORY RELEVANT AQUATIC PHASES

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ABSTRACT

At the Institute of Energy Research the behaviour of UAl_x -Al and U_3Si_2 -Al dispersed fuel elements in repository relevant aquatic phases is under investigation. As repository relevant aquatic phases a salt brine ($MgCl_2$ -rich brine 2 solution), granite water (Äspö-type) and clay pore water (Mont-Terri-type) are used. Both fuel types have similar corrosion rates in these repository relevant aquatic phases. Due to the high amount of chloride the highest corrosion rates were obtained in salt brine. The corrosion products obtained by a complete corrosion of an UAl_x -Al fuel sample in the salt brine were separated and treated under various geochemical conditions. Mobile and immobile radionuclides were determined. Furthermore one identified crystalline phase component, a Mg-Al-Cl-hydrotalcite was synthesized and sorption investigations with different repository relevant radionuclides were performed in brine 2 solution. Selenium in the anionic form as selenite and cationic radionuclide species as the trivalent americium, the trivalent europium and the tetravalent thorium were shown to be retarded.

1. Introduction

At present three research reactors, the FRM-II-reactor (München), the FRG-I-reactor (Geesthacht) and the BER-II-reactor (Berlin) with a thermal output higher than 5 MW are in operation in Germany. In these reactors dispersed U_3Si_2 -Al-fuel is used. In general, three possible back-end options for irradiated research reactor fuel elements exist in Germany [1]: first, irradiated research reactor fuel elements of USA origin can be sent back when these fuels were taken out of the reactor not later than May 2006. This affects basically the dispersed metallic UAl_x -Al-fuel, which was used in the FRJ-II-reactor (Jülich) till May 2006. Secondly, reprocessing in Great Britain or France is possible and it will be performed with respect to economical reasons. Thirdly, dry interim storage and later on, direct disposal in deep geological formations has been taken into account too. As deep geological formations salt, granite and clay are considered. In granite and clay formations pore water is always present whereas in a salt repository a water ingress and subsequent formation of salt brines is considered as accident scenario.

At the Institute of Energy Research (Safety Research and Reactor Technology, ISR) investigations to the behaviour of research reactor fuel elements in these repository relevant aquatic phases are performed. In November 2003 a third project, funded by the federal Ministry of Economics and Technology had started and three main work packages were established:

- Leaching tests with non-irradiated and irradiated dispersed UAl_x -Al- and U_3Si_2 -Al-fuel types in granite water (Äspö-type), clay pore water (Mont-Terri-type) and in $MgCl_2$ -rich salt brine (brine 2) in order to study their corrosion behaviour
- Treatment of the secondary phases, formed by corrosion of an dispersed UAl_x -Al-fuel type in brine 2 under various geochemical conditions in order to distinguish between mobile and immobile radionuclides
- Synthesis of an identified crystalline phase, a Mg-Al-Cl-hydrotalcite, in order to study the retardation of repository relevant radionuclides.

The results from these work packages are presented in this paper [2].

2. Experimental set-up

Leaching experiments with the non-irradiated $\text{UAl}_x\text{-Al}$ - and $\text{U}_3\text{Si}_2\text{-Al}$ -fuel elements samples were performed in glass autoclaves at 90°C in 400 ml of MgCl_2 brine solution, in clay pore water (Mont-Terri-type) and in granite water (Äspö-type) in presence of 10 g FeCl_2 (iron is the basic material of the fuel container) under anaerobic conditions. The dimension of the $\text{UAl}_x\text{-Al}$ -samples was 28 x 20 x 1.5 mm with an amount of 0.120 g U total (U-235-enrichment: 89%). The dimension of the $\text{U}_3\text{Si}_2\text{-Al}$ -sample was 40 x 20 x 1.36 mm with an amount of 1.44 g U total (U-235-enrichment: 0.21%). Similar leaching experiments were performed with irradiated $\text{UAl}_x\text{-Al}$ - and $\text{U}_3\text{Si}_2\text{-Al}$ -fuel elements samples. The dimension of the $\text{UAl}_x\text{-Al}$ -sample was 40 x 20 x 1.36 mm with an amount of 0.25 g U total (U-235-enrichment: 80%). The dimension of the U_3Si_2 sample was 40 x 20 x 1.36 mm with an amount of 1.704 g U total (U-235-enrichment: 19,75%).

In order to distinguish between mobile and immobile radionuclides, trapped by the corrosion products, the geochemical conditions were varied. Corrosion products used for these studies were obtained by complete corrosion of an irradiated $\text{UAl}_x\text{-Al}$ -fuel element sample (40 x 20 x 1.36 mm) in MgCl_2 -rich brine in the presence of iron under anaerobic conditions at 90 °C. In order to study the influence of dilution, aliquots were treated with “fresh brine 2” solution. The influence of the ionic strength was studied by the use of different concentrated brine 2 solutions. The fraction of inventory was determined by complete dissolution of some aliquots in 8 M HNO_3 solution. All samples obtained were analysed radiometrically.

The experimental set-up for the syntheses of the Mg-Al-Cl-hydrotalcite is described in [4]. The radioactive selenium, americium, europium and thorium solutions used were prepared from standard stock solutions. Sorption experiments were performed according to the batch-technique. All samples were stored in glass tubes with occasional shaking under argon-atmosphere for two days. Then all samples were filtered (450 nm) and the pH was measured. Aliquots of the solutions were analysed radiometrically. The solids were washed, dried and analysed by XRD and FT-IR. Blank experiments were performed too, indicating that the sorption on the glass walls was negligible.

3. Results and Discussion

Topic 1

First, the corrosion behaviour of non-irradiated dispersed $\text{UAl}_x\text{-Al}$ - and $\text{U}_3\text{Si}_2\text{-Al}$ -fuel elements specimens in repository relevant solutions was investigated. In Figure 1 and Figure 2 the hydrogen production for both fuel-types in brine 2 and in clay pore water (Mont-Terri-type) are presented. The functions of the hydrogen formation in granite water (Äspö-type) are comparable to the clay pore water.

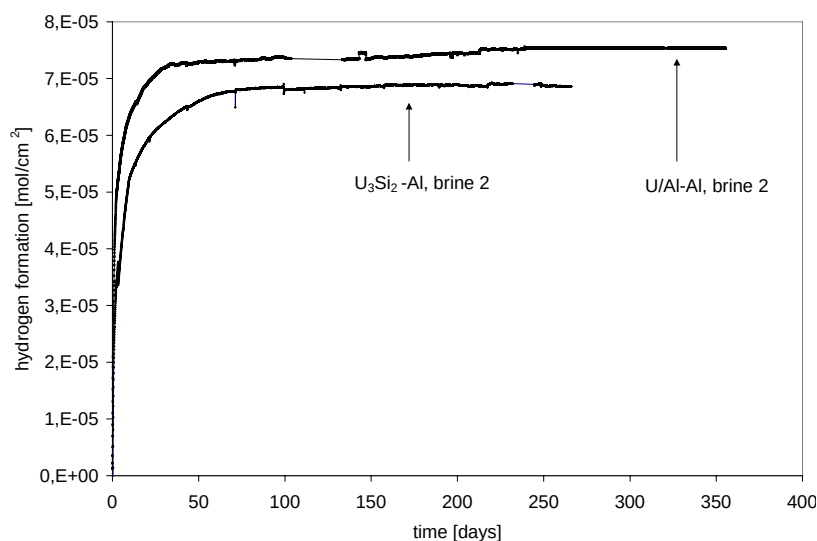


Fig. 1 Hydrogen formation in brine 2 solution

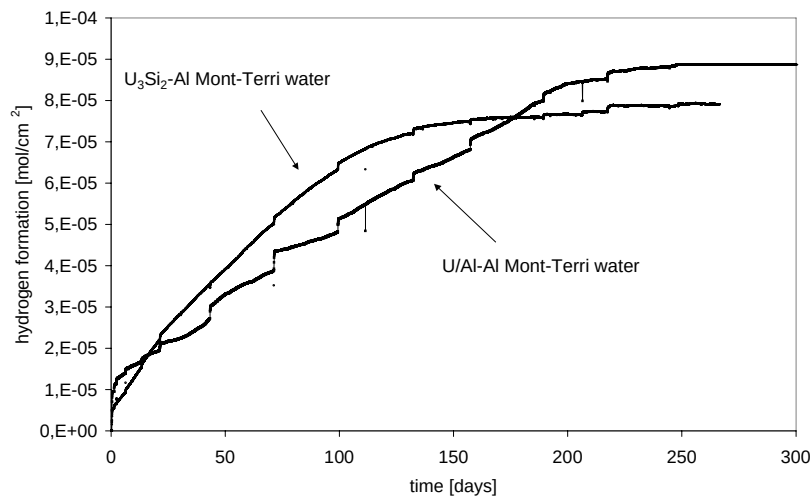


Fig. 2 Hydrogen formation in Mont-Terri water

In brine 2 the hydrogen formation is terminated after 100 days. This indicates the complete corrosion of the fuel. In granite and clay pore water the corrosion processes are terminated after 250 days. Nevertheless in view of the time periods of final disposal the assumption can be drawn, that in these repository relevant aquatic phases both fuel types corroded instantaneously. The detected maximal hydrogen formation normalized to the fuel surface was between $7.0E-05$ and $8.5 E-05$ mol/cm². Furthermore in brine 2 higher corrosion rates were obtained. In brine 2 more chloride is present which causes a low pH value. Under these conditions the aluminium dissolution and the formation of local elements, which accelerate the corrosion, are favoured. The matrix elements, aluminium, uranium, and silicon were detected only in the secondary phases formed. Similar experiments were performed with irradiated fuel elements. First results will be obtained in the next months.

Topic 2

The secondary phases obtained by complete anaerobic corrosion of an irradiated UAl_x -Al-fuel element in brine 2 were treated under various geochemical conditions particularly under aerobic conditions.

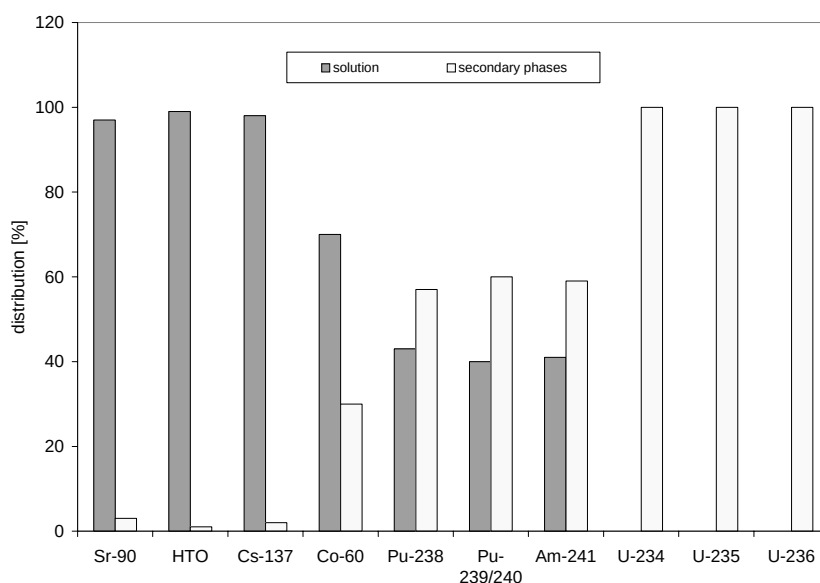


Fig. 3 Radionuclide distribution between solution and secondary phases after complete corrosion of an UAl_x -Al fuel element sample in $MgCl_2$ -rich brine

First the radionuclide inventory was determined and a radionuclide distribution between secondary phases and solution was obtained (Figure 3). The main mobile radionuclides are Cs, Sr, and Tritium, while the radionuclides Pu, Am, and U are immobile.

Then aliquots of the secondary phases were taken and treated with “fresh” brine 2 in order to study the dilution effects. Other aliquots were treated with different concentrated brine 2 solutions in order to study the influence of the ionic strength. The results indicate clearly, that under aerobic conditions Am and U must be regarded mobile as well. This can be explained mainly by the change to aerobic conditions. Oxygen causes the oxidation of U-IV to U-VI and these U-VI components are lightly soluble. The higher solubility of Am-III can be explained by the formation of lightly soluble Am-III-carbonyl-complexes. The behaviour of Pu is different. It seems that Pu already formed polymeric oxy or hydroxy-complexes which are nearly insoluble.

Topic 3

In view of the actinides the secondary phases have a significant influence on radionuclide retardation. After the complete corrosion of an UAl_x -Al-fuel element sample in brine 2 a Mg-Al-Cl-hydrotalcite was identified as a crystalline phase-component in the secondary phases formed [3]. A synthesis was performed and the obtained solid was analysed. From these results the formula of the Mg-Al-Cl-hydrotalcite can be derived as: $Mg_3Al(OH)_8Cl_{0.88} (CO_3^{2-})_{0.063} \cdot 2.4 \cdot H_2O$. Due to the structure of this Mg-Al-Cl-hydrotalcite an anion exchange in the interlayer should be possible. We performed sorption experiments according to the batch technique and for the monovalent anionic species, iodine [4] and pertechnetate no retardation in brine 2 solution was achieved, but for the divalent anionic specie selenite, a retardation was gained.

For cationic species two reactions are feasible: sorption and incorporation. Results from sorption experiments did show that the trivalent cationic species, Am and Eu had a similar sorption behaviour. Then the sorption data were fitted to the Dubinin-Radushkevich (D-R) equation [5] and the mean energy of sorption was calculated. The mean energy of sorption is the free energy change when one mole of ion is transferred to the surface of the solid from infinity in the solution. The energy calculated for Am was 11.8 kJ/mol and for Eu a value of 11.18 kJ/mol was determined. Both values were in the range for ion exchange reactions, i.e., 8-16 kJ/mole. For thorium, a tetravalent cationic specie, the D-R-Plots with two different hydrotalcite concentrations are shown in Figure 4.

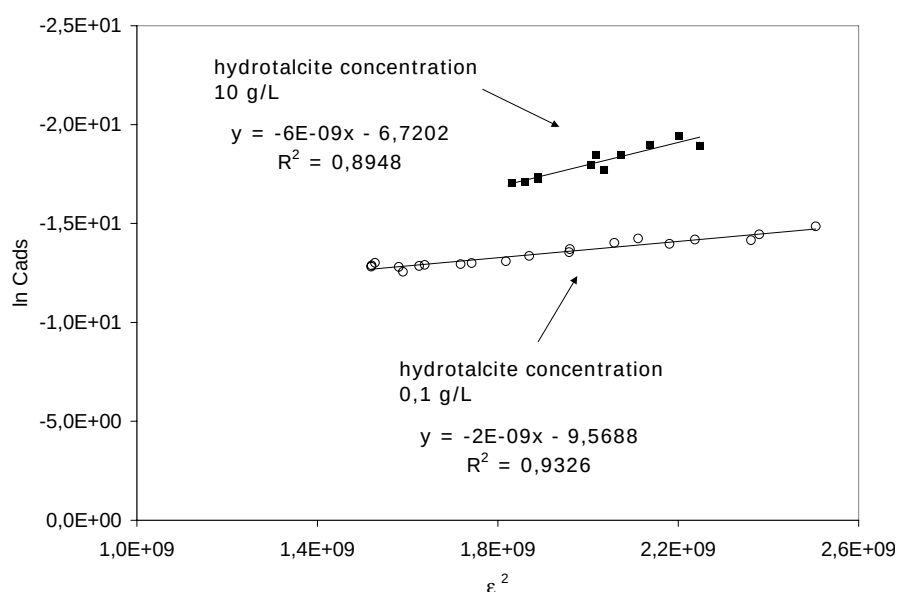


Fig.4 D-R-plots for the sorption of thorium onto the Mg-Al-Cl-hydrotalcite

For a hydrotalcite concentration of 10 g/L the mean energy of sorption was 9.13 kJ/mol and a value of 15.8 kJ/mol was obtained for the hydrotalcite concentration of 0.1 g/L. In both cases sorption occurs via an ion exchange mechanism.

4. Conclusions and outlook

From the results of the present work the following main points can be summarised:

- Metallic UAl_x -Al dispersed fuel as well as the U_3Si_2 -Al dispersed fuel corrodes rapidly in salt brine (brine 2), in granite water (Äspö-type) and in clay pore water (Mont-Terri-type) instantaneously. The matrix elements aluminium, uranium and silicon are quantitatively found in the secondary phases formed by corrosion. The hydrogen formation created in these repository relevant aquatic phases was determined to be in the range of $7.0\text{E-}05$ and $8.5\text{E-}05$ mol/cm^2 .
- Secondary phases, formed by corrosion of an irradiated UAl_x -Al-fuel element in salt brine, were separated and treated under various geochemical conditions. Mobile radionuclides are Cs, Sr and Tritium. The remobilisation of uranium and americium is caused by the aerobic conditions which accelerate the formation of soluble U-VI compounds and soluble Am-III-complexes. The actinide Pu is not affected and we assume the formation of polymeric oxy or hydroxy-complexes which are slightly soluble.
- In the leaching experiments using salt brine, one crystalline phase component of the corrosion products, a Mg-Al-Cl-hydrotalcite was synthesised and it was shown that selenium, americium, europium and thorium were retained by means of ion exchange reactions in brine 2 solution. Under final disposal conditions in a salt repository these elements will be retarded by the Mg-Al-Cl-hydrotalcite. No retardation was given for iodine and pertechnetate in brine 2 solution. Towards the hydrotalcite the affinity of chloride as competition anion is higher.

In our future work leaching experiments with irradiated fuel elements in hot cell facilities will be continued and after complete corrosion the radionuclide distribution between solution and secondary phases will be determined for both fuel types in the three repository relevant aquatic phases. Secondary phases obtained after complete corrosion of the non-irradiated dispersed fuel elements will be analysed for elemental composition. Furthermore the identification of all present crystalline phases will be aspired. Besides sorption, radionuclide retardation may be achieved by incorporation. The incorporation of various radionuclides in the lattice structure of the Mg-Al-Cl-hydrotalcite will be another main topic of our future work, because this radionuclide retardation can be regarded as an irreversible process.

5. Acknowledgment

This work is funded by the BMWi (Förderkennzeichen: 02E9803).

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CORROSION OF SPENT RESEARCH REACTOR FUEL: THE ROLE OF SETTLED SOLIDS.

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ABSTRACT

Reactor components or fuels that have remained immersed in a research reactor pool or in a spent fuel storage basin often reveal solids on their surfaces. The sources of these solids are many and include air borne dust, corrosion products and precipitated salts. Results of the IAEA coordinated research project (CRP) "Corrosion of Al-clad Spent Fuel in Water" and the Regional Project for Latin America (RLA) revealed that settled solids contribute to corrosion of Al coupons. Further studies carried out at different sites worldwide to determine the rates of settling of solids, their composition and influence on corrosion of aluminium alloys revealed that the solids consisted mainly of oxides of aluminium, iron, silicon and calcium. Other constituents in the settled solids were site-specific. Short term laboratory tests in which solids with specific composition were positioned on aluminium substrates in relatively pure water revealed their role in initiating pitting and/or crevice type of corrosion.

1. Introduction

Reactor components or fuel assemblies that have remained immersed in a research reactor pool or in a spent fuel storage basin for a reasonable length of time have loose solids on their surfaces. The quantity and nature of these solids vary. These solids settle on all surfaces inside pools or basins. Most of these solids eventually settle at the bottom of the pool or basin and are the main constituent of sludge. The origin or sources of these solids are mainly air borne dust, corrosion products and precipitated salts. Airborne dust in the reactor hall or in the spent fuel basin (SFB) hall settles on reactor pool or SFB surfaces. The dust (fine solid particles) on the pool/basin surface floats until wetted by the water. Surface skimmers in reactor pools remove most of the floating dust. However, depending on properties of the solid, mainly density, it settles at the bottom of the pool or on any surface that it encounters as it descends through the pool or basin water. Solids settle faster in stagnant regions of the pool. Fine solid particles also have a tendency to agglomerate to form larger particles which settle faster. This tendency varies with the composition and density of the particles. The corrosion products on metallic surfaces in contact with flowing water are easily detached and carried in the circulating water, often returning, depending on its size, to the reactor pool or basin. Another source for solids is precipitated solids. Reactor hall operations such as: (a) movement of cranes; (b) opening of doors; (c) immersion of inadequately dusted components in the reactor pool; (d) shifting of immersed components are other sources of settled solids. In away-from-reactor fuel storage facilities the above mentioned scenes play out to a greater extent.

Experimental work carried out in the IAEA coordinated research project (CRP) “Corrosion of Aluminium-clad spent fuel in water” and the Regional Project for Latin America (RLA) included the exposure of horizontal and vertical coupons of aluminium (Al) alloys to spent fuel basins in many countries. [1] Comparison of the corrosion behaviour of these coupons revealed that horizontal coupons corroded considerably more than vertical coupons. The top surfaces of horizontal coupons corroded more than the bottom surfaces of the same coupon. These observations lent proof to the dominant role of settled solids on the corrosion of Al coupons and by extension on the corrosion of Al-clad spent research reactor fuel. [2] Further studies were undertaken to determine the: (a) amount of settled solids and the rate of settling of these solids at the various sites; (b) composition and/or constituents of the solids; (c) role of specific settled solids on the nature and extent of corrosion of Al alloys. This paper presents details of these studies.

2. Methods and materials

In this study, a stainless steel or aluminium recipient (henceforth referred to as the collector) was used to collect settled solids at 11 different sites where the two IAEA projects were being carried out. This collector was positioned in the vicinity of racks with corrosion test coupons. The collectors remained in the research reactor (RR) or SFB for periods ranging from 4-6 months. After withdrawal of the collector, the water was filtered; the solids collected on a filter paper were dried at 100° C for 24 hours and weighed. The amount of solids and the rates of settling were determined. Representative specimens of the solids were examined in the SEM and/or analysed using one or more of the following techniques: (a) energy dispersive spectroscopy (EDS), (b) x-ray diffraction (XRD), (c) x-ray fluorescence (XRF), (d) mass spectrometry. Fig. 1 shows a collector and settled solids in one of the collectors.



Fig. 1. (a) AISI 304 stainless steel collector used at CChen, Chile. (b) Settled solids in a collector

3. Results and discussion

The rates of sedimentation of the solids at different sites and their composition and/or constituents are shown in Table 1. The solids consisted mainly of oxides of aluminium, iron, silicon and calcium. This indicated that the sources of the solids were construction debris, atmospheric dust and corrosion products of aluminium alloy and steel components in the pool or basin. Other constituents in the settled solids were site-specific.

Test site	Rate of sedimentation ($\mu\text{g}/\text{cm}^2/\text{month}$)	Composition or constituent of settled solid
RECH-1, Chile.	58.6	Al; SiO_2 (quartz); SiO_2 (cristobalite) $(\text{Ca}, \text{Na})(\text{Si}, \text{Al})_4\text{O}_8$ (anortite)
RECH-2, Chile.	1.5	SiO_2 (quartz)
CDTN, Brazil.	18.1	Amorphous CaCO_3 (calcite); Fe_3O_4 (magnetite), SiO_2 (quartz); $\text{CaMg}(\text{CO}_3)_2$ (dolomite) $\alpha\text{-Al}_2\text{O}_3 \cdot 3\text{H}_2\text{O}$ (gibbsite), Fe_2O_3 (hematite) $\text{Mg}_3\text{Si}_4\text{O}_{10} \cdot (\text{OH})_2$ (talc)
ININ, México.	17.666	Iron oxides (hematite, magnetite), aluminosilicates, sodium and calcium carbonates
România.		Fe, Zn and Cd
SF pool, Thailand.	195	Fe, Al, SiO_2 , Cu, Ba; Light particles of plant tissue, bio-mass floating in water with some fine dust.
Reactor pool, Thailand.	141	Fine light brown dust. Main component is iron.
Cooling pool, Kazakhstan.	36.000	Deionizing column resin
Water storage, Kazakhstan.	12.000	Deionizing column resin
RA3 Decay Pool, Argentina	19.5	Silicon oxide and combined oxides of Si, Al and Fe
IPEN, Brazil.	178.2	56.79 Al_2O_3 , 21.04 SiO_2 , 14.93 Fe_2O_3 , 2.35 CaO , 1.6 Cr_2O_3 , 0.76 TiO_2 , 0.6 NiO (wt%)

Tab. 1. Rates of sedimentation of solids and their composition or constituents.

3.1. Effect of settled solids on corrosion of aluminium alloys

Aluminium and its alloys normally corrode in aqueous solutions with low or high pH. Pitting is the main form of corrosion and aluminium alloys are prone to pitting corrosion in solutions with not only chloride ions but also other anions like sulphates and nitrates. Aluminium alloys do not generally pit in environments totally free of aggressive ions. Nevertheless, Al alloys have been observed to pit in very low conductivity water in the presence of solids on the surface. This solid particle-induced corrosion of aluminium could be due to one or more of several reasons: 1) The alkaline nature of the solid or leaching of alkaline products from the solid; 2) Formation of crevices under the solid and consequent crevice corrosion in the presence of aggressive ions; 3) The solids if conducting, become the cathode for the cathodic reaction, result in localized pH increase and consequent metal dissolution.

Reason (1) is considered to be operative if the solid is a concrete particle. Fig 2a reveals the surface of an AA 6061 alloy with a concrete particle and exposed for 40 days to nuclear grade demineralised water. [3] Aluminium alloys are also susceptible to crevice corrosion, which occurs in the presence of aggressive ions. [4] In pure water, this form of corrosion sometimes occurs in the presence of aggressive ions originating from the sediment itself. Figs 2b and 2c show corrosion of AA 6061 alloy with hematite and magnetite particles on the surface. These figures reveal that the crevice corrosion mechanism could be operating at the microscopic level. The alloy surface around the hematite particle is corroded, where as the surface around the magnetite particle is unaffected. [5]

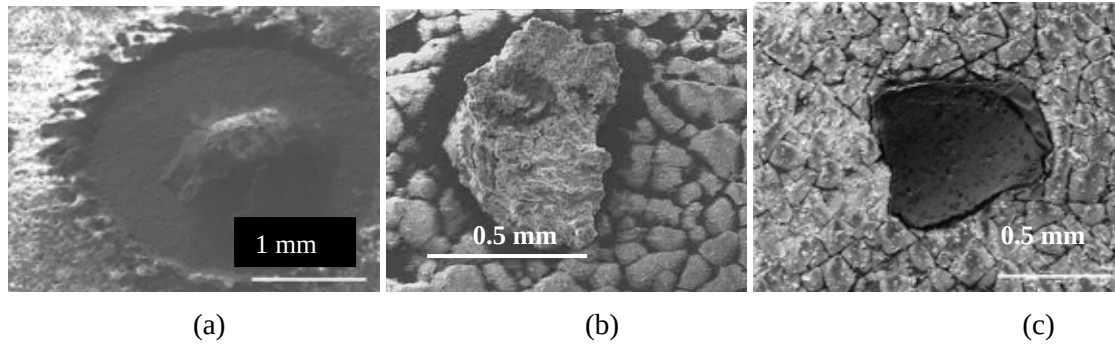


Fig. 2: Corrosion of AA 6061 alloy exposed for 40 days to nuclear grade demineralised water: (a) with a concrete particle; (b) with a hematite particle; (c) with a magnetite particle.

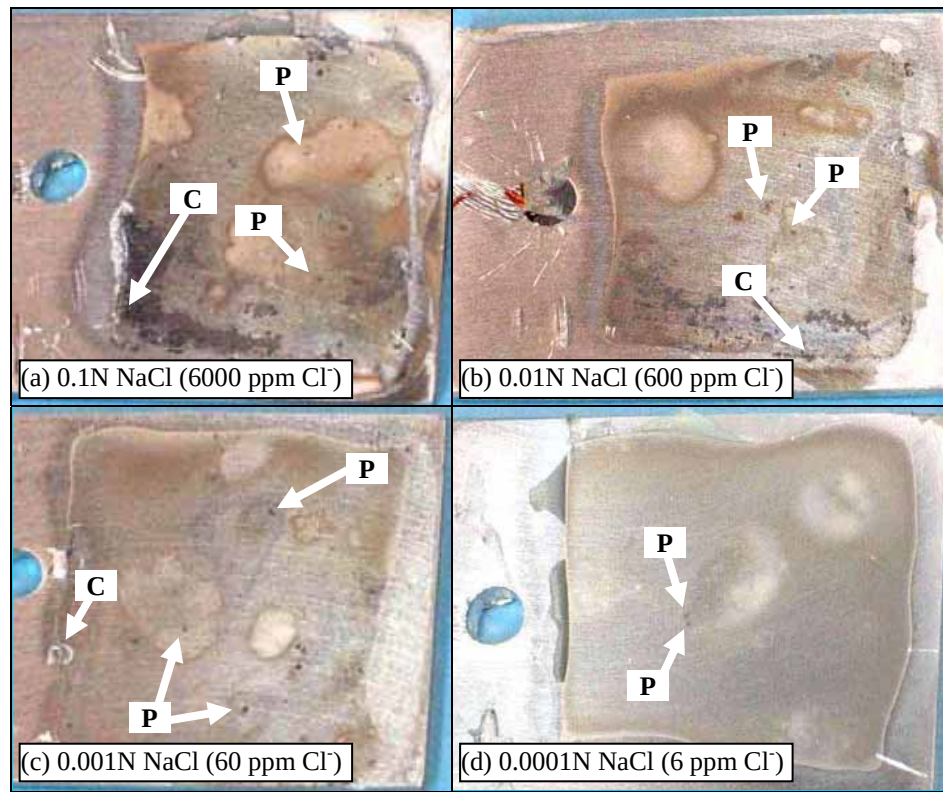


Fig. 3: Appearance of AA 6061 sample surfaces with hematite sediments after 7 days of exposure to NaCl solutions of different concentrations. Pitting is indicated with 'P' and crevice corrosion with a 'C'. The stains produced by the sediments are still visible.

The sediment induced pitting produced in the laboratory resembles, to some extent, observations made on Al surfaces exposed to SFB waters. However the pits were not as deep as those observed on certain spent fuels stored for extended periods. To evaluate the short term influence of sediments on the corrosion of Al, immersion tests were carried out in waters similar to that in SFBs. In these tests, particles of hematite, magnetite and glass were deposited on aluminium surfaces exposed to sodium chloride solutions of various concentrations.

3.2. Effect of short term exposure of deposited solid particles on Al alloy corrosion

The Al surface with hematite particles that was exposed for 60 days to a solution with 40 ppm of chlorides did not reveal pits but some stains. However, a similar surface coupled to stainless steel revealed pits after only 7 days. The results also indicated the marked effect of chloride ion concentration as shown in Fig 3. In waters with high chloride ion concentrations, both pitting and crevice corrosion was observed. Pits formed both under the deposits and on the free surface. In the solution with 0.001N NaCl, very few pits formed at regions away from sediments. In 0.0001N NaCl all the pits were under the particles. The Al surface with magnetite particles and exposed for 7 days to 0.001N NaCl (60 ppm chloride) revealed pits only around particles that had transformed from magnetite to hematite. Glass pieces were used to simulate inert particles. The Al surface with glass pieces was exposed for 7 days to 0.001N NaCl and it revealed pits that could be observed through the glass.

4. General discussion

Although these tests were of short duration, the results demonstrate that spent fuel elements that are stored for extended periods can undergo these forms of environment assisted degradation. It is quite probable that corrosion occurs in what is considered to be innocuous environments and sediments could trigger the onset of pitting. The results also demonstrate the deleterious synergistic effect of galvanic contacts and sediments. These conditions facilitate the onset of corrosion processes, which in more dilute environments are prone to occur beneath the deposits. Similar conditions are encountered in some spent fuel storage sites in which fuel bundles are positioned inside steel structures and without proper electrical isolation.

5. Recommendations

The composition of the settled solids at a specific site helps indicate the possible source. Knowing the source, steps can be taken to reduce or eliminate it. The actions are usually site-specific. However, some general recommendations can be made to reduce or eliminate airborne dust from settling on RR pool or SFB water surface. These include:

1. Increase in efficiency and/or frequency of water circulation through a filter.
2. Use of a skimmer and filter system.
3. Increase in water flow in the vicinity of stored spent fuel.
4. Use of stainless steel doors or lids for away-from-reactor storage basins.
5. Use of adequate corrosion protection schemes, if plain carbon steel doors or lids are used.
6. Installation of double doors, in case of a dusty atmosphere just outside a SFB.
7. Vacuuming of all surfaces in reactor hall and inside the pool, in the case of in-reactor basins.
8. Improved air circulation and pumping of filtered air into the reactor hall.
9. Reduction in water turbulence in the SFB.

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INTERNATIONAL TOPICAL MEETING ON RESEARCH REACTOR FUEL MANAGEMENT (RRFM) - UNITED STATES FOREIGN RESEARCH REACTOR (FRR) SPENT NUCLEAR FUEL (SNF) ACCEPTANCE PROGRAM: 2007 UPDATE

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Abstract

The Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel, adopted by The United States Department of Energy (DOE), in consultation with the Department of State (DOS) in May 1996, has been extended to expire May 12, 2016, providing an additional 10 years to return fuel to the U. S. This paper provides a brief update on the program, now transitioned to the National Nuclear Security Administration (NNSA), and discusses program initiatives and future activities. The goal of the program continues to be recovery of nuclear materials, which could otherwise be used in weapons, while assisting other countries to enjoy the benefits of nuclear technology. The NNSA is seeking feedback from research reactor (RR) operators to help us understand ways to include eligible RRs who have not yet participated in the program.

1. Introduction

This paper presents the Foreign Research Reactor (FRR) Spent Nuclear Fuel (SNF) Acceptance Program, (the Acceptance Program). After an initial discussion of program history, contract extension and compliance are discussed. Planning issues are then set out to incorporate lessons learned from recent shipments in order to help FRRs understand issues which may assist in achieving their objective of proper disposition of SNF. The final discussion topic is DOE efforts to advance the goals of the Acceptance Program, with a conclusion that the Acceptance Program wants to work with FRRs to plan for shipment of their eligible spent fuel as early as possible.

2. Acceptance Program Metrics

The Acceptance Program, now in the eleventh year of implementation, has completed 37 shipments to date, safely and successfully, and another is expected to be completed soon. Twenty-seven countries have participated so far, returning a total of 7,620 spent nuclear fuel elements to the United States for management at Department of Energy (DOE) sites in South Carolina and Idaho, pending final disposition in a geologic repository. Twenty nine (29) of the 37 shipments contained aluminum-based spent nuclear fuel from research reactors and were placed into storage at the Savannah River Site (SRS) in South Carolina. One shipment was forwarded on to the Y-12 National Security Complex, since the fuel was only slightly irradiated and eligible for receipt at that facility. The remaining seven (7) shipments were placed into storage at the Idaho National Laboratory (INL). The most recent shipment

was completed without incident, arriving at SRS on February 6, 2007. During the remaining calendar year (January - December 2007), the program is planning to receive as many as five shipments of SNF from various locations.

3. Contractual Requirements

3.1 Contract Extensions

DOE believes that all contract extensions, required to support reactor conversion and continued operation after May 2006, have now been signed. Other research reactors which have already converted to LEU fuel will need a contract extension to authorize shipments wanted after May, 2009. DOE intends to modify these contracts with priority given to those who are scheduled to ship in the near future. Reactor Operators in this situation are strongly encouraged to coordinate with the Acceptance Program office to negotiate the extension of the FRR-DOE contract to authorize continued Acceptance Program participation.

3.2 Contract Implementation

DOE enters into a contract with each of the customers who return SNF to the United States. It is very important that the contracting parties clearly understand all of the provisions in the contract. Contract requirements are usually described in detail prior to the first shipment. As time passes and personnel change, some understanding may be lost. Further discussions on contract requirements can always be addressed to the Acceptance Program office. Compliance with all contract requirements must be maintained. One important article which has recently been misunderstood covers compliance with government regulations concerning public disclosure of any shipping plans or shipment information, or the individual details comprising such plans or information. Compliance with this article is an important obligation to support security for any shipment activity. During a recent shipment, a press release was made after the ship reached international waters on the way to the United States. DOE believes this is an unwarranted violation of the contract which made the security of the shipment more vulnerable. This premature release of information also violated the United States Nuclear Regulatory Commission regulations under which the shipments are authorized. Further, The Convention on the Physical Protection of Nuclear Material entered into by states which support the Acceptance Program requires that each state protect the confidentiality of this information. Our ability to continue this program depends on our customers following the agreed process.

4. Focus on Advance Planning

The FRR SNF Acceptance Program focuses on the planning and implementation of these shipments of research reactor spent fuel to the United States in support of worldwide nuclear nonproliferation efforts, while assisting other countries to enjoy the benefits of nuclear technology. Along with shipment logistics, the DOE Office of Global Threat Reduction (GTR) continues to address many other issues of importance to the program.

4.1 Shipment Scheduling

The most critical barrier to smooth operation associated with the program remains early scheduling and coordination of planned shipments. It is always important that NNSA clearly understands each Reactor Operator's intentions so that our planning can be well integrated and supported to meet the Reactor Operator's needs. It is also important to submit the required fuel data as early as possible in order to allow the receiving site adequate time to perform necessary reviews and prepare for receipt and storage.

Early availability of this data is also important for use in verifying transport package license requirements or submitting for a license amendment. Budget limitations could challenge implementation of shipping plans while NNSA and the Department of Energy receiving facilities also face increasing challenges in preparing to receive material, particularly when shipping plans are not well known. The GTR Acceptance Program staff will be happy to answer questions about scheduling or clarify what type of information is needed to facilitate receipt of fuel.

As requested by many FRRs the program was extended to allow additional time for further development of LEU fuels and planning for back end solutions in the fuel cycle. The change was made to benefit the FRRs that needed justifiable relief. Some other FRRs are now taking advantage of these benefits by extending their shipping schedules to defer costs. These delays are hurting DOE's ability to continue normal planning and to maintain adequate resources for the receipt facility. The FRRs are strongly encouraged to continue shipping as early as possible and maintain original schedules and plans as closely as possible. Deferring shipments when spent fuel is available for shipping could result in changes designed to improve shipping decisions.

4.2 End –User Assurances

Some countries require the issuance of an End-Use or Dual-Use Undertaking in order to obtain an export license. In the past, DOE provided that document to the reactor operator when requested. DOE no longer provides that document. However, assurances are already provided to those countries through the Agreements for Cooperation between each country and the United States when one exists or other avenues. The U.S. Department of State can validate those assurances to the participating country as necessary. We recommended that these requirements be identified and resolved by the reactor operators as early as possible to ensure this political process is completed without shipment delays.

4.3 Insurance Issues

One issue has been noted to be a problem for reactor operators in high-income economy countries who participate in joint shipments. Nuclear liability insurance associated with the ocean transport has the potential to adversely affect the total cost of shipping. This is because the shippers are sometimes required to have overlapping insurance coverage and also may have different requirements for minimum coverage. It is important for reactor operators to plan early for the required coverage and how to provide coverage in the least expensive manner. Consideration should be given for reactor operators entering into a joint shipment to coordinate in obtaining their nuclear liability insurance with the same pool or under a joint contract, where possible, in order to mitigate overlapping insurance costs. It is also important to be conscious of this potential problem and budget for any added cost that cannot be mitigated.

4.4 Cask License Review

The Acceptance Program enjoys a very good working relationship with Nuclear Regulatory Commission (NRC) staff and wishes to take every measure possible to respect this relationship by ensuring that cask applications are timely and complete. DOE has been meeting periodically with NRC to discuss planned shipments and forecasted support required to meet the needs of the Acceptance Program and our customers. However, because there are limited resources for review of cask licenses, it is necessary for our customers to provide adequate time in the preparation process, scheduling for early application for review and approval of cask licenses.

5. Efforts to Improve and Accelerate

The Acceptance Program has now passed its approximate midpoint. More than ever before, DOE and reactor operators need to work together to schedule shipments as soon as possible, to optimize shipment efficiency over the remaining years of the program. Countries interested in participating in the Acceptance Program should express their interest as soon as possible so that fuel and facility assessments can be scheduled and shipments may be entered in the long-term shipment forecast. New and current Acceptance Program participants should also coordinate with DOE approximately 18 - 24 months in advance to ensure DOE can meet the Reactor Operator's plans and needs. Accelerated schedules are possible if there are no significant issues over past shipments. However, decreasing resources and coordination requirements with other agencies such as the Nuclear Regulatory Commission and Department of Transportation have the potential to limit DOE's capability to support these accelerated schedules. Specifically, the Acceptance Program may not be able to accommodate a large number of requests at the end of the program, particularly from geographically isolated regions.

5.1 Reorganization

The Office of Global Threat Reduction has reorganized in order to better use available resources and align the offices within three global regions and three cross-cutting program pillars. The regions include The Office of North & South American Threat Reduction (NA-211), Office of European & African Threat Reduction (NA-212), Office of Former Soviet Union and Asian Threat Reduction (NA-213). The organizational program pillars include Convert, Protect, and Remove. The FRR SNF Acceptance program, as a Remove function, is located under the Office of FSU and Asian Threat Reduction. Although the program is managed under the Office of FSU and Asian Threat Reduction, the program operates globally across all regions. The program Technical Lead will continue to implement the program and will be the primary point-of-contact for this program. Regional Country Officers will assist in program coordination and shipment implementation. This reorganization should be essentially transparent to the reactor operator and other supporting shipment participants.

5.2 Material Disposition

The DOE Environmental Management (DOE-EM) organization that used to manage the FRR SNF Acceptance Program is making strides to further disposition the repatriated spent nuclear fuel. The DOE-EM organization is considering continuing with the DOE Programmatic Spent Nuclear Fuel Environmental Impact Statement [1] and associated Record of Decision [2]. This decision included transporting fuel to place all aluminum clad spent fuel at the SRS and stainless steel fuel such as TRIGA fuel at INL. This allows for a potential decision to further treat the aluminum clad fuel in the H-Canyon facilities at SRS for disposition as waste in the same fashion as other high level waste material within the DOE complex. Any decision to further treat the material would be subject to further evaluation under the National Environmental Policy Act.

5.3 Potential Fee Changes

NNSA continues to evaluate ways to accelerate repatriation activities. Therefore, fees may change in the future and/or other changes may be implemented, if DOE believes the changes will positively influence program goals. DOE is also continuing to try to keep the reactor operator's cost to participate in the Acceptance Program low as possible. Any suggestions of methods to accelerate repatriation of SNF, especially Highly Enriched Uranium (HEU), would be welcomed and given all due consideration.

5.4 Coordination with Other Programs

A primary goal of the Acceptance Program is to support worldwide nonproliferation efforts by disposition of HEU which contains uranium enriched in the United States. Integral to this process is the U.S. assistance offered in helping reactor operators convert their cores to low enriched uranium (LEU) as the reduced enrichment fuels become qualified and available. In addition, DOE plays a strategic role in ensuring a supply of enriched uranium for fuel fabrication. In the Acceptance Program, the primary goal is intertwined with the missions of the Reduced Enrichment for Research and Test Reactors (RERTR) Program and the Enriched Uranium Operations group from DOE's Y-12 National Nuclear Security Complex in Oak Ridge, Tennessee. DOE Acceptance Program staff remain committed to working with staff in these other program offices within DOE and to do whatever is possible to assist in smooth transitions of core enrichment level and a steady supply of fuel.

6. Conclusion

The United States remains committed to supporting worldwide nonproliferation goals while assisting other countries to enjoy the benefits of nuclear technology such as those for which this program was designed. The programmatic goal is to accept eligible fuel sooner rather than later. Reactor operators are strongly encouraged to work closely with technical points-of-contact in order to ensure shipping schedules are accurate and achievable. The GTR staff hopes to work with all remaining eligible research reactors to plan for shipments of their eligible spent fuel as early as possible. NNSA continues to support research reactor operators' needs and would be happy to meet any interested parties to discuss the program.

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AUSTRALIAN NUCLEAR SCIENCE AND TECHNOLOGY ORGANISATION (ANSTO) AND NAC INTERNATIONAL

AREA: FUEL BACK-END MANAGEMENT

PREPARATION AND PERFORMANCE OF THE LARGEST EVER SHIPMENT OF IRRADIATED HEU FUEL ELEMENTS UNDER THE FRR PROGRAM: THE 2006 ANSTO HIFAR SPENT FUEL TRANSPORT FROM SYDNEY TO THE UNITED STATES

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ABSTRACT

Just over six years after signing the contract between ANSTO and INVAP for the design and construction of a new 20 MW multipurpose research reactor, Australia's new research reactor, OPAL, went critical for the first time on August 12, 2006. The HIFAR reactor continued to operate in parallel - until officially shut down on 30 January 2007 in order to ensure continuity of operation during the transition.

The management of spent fuel remains a very important aspect of the operation of research reactors for ANSTO. For disposition of UK-origin spent fuel arising from the operation of the HIFAR reactor, ANSTO initially elected to ship the irradiated fuel assemblies to the UKAEA in Dounreay, Scotland. With the closure of Dounreay, alternatives were evaluated and reprocessing of the spent fuel at the La Hague reprocessing plant was selected as the option for the disposition of ANSTO's UK origin spent fuel. Between 1999 and 2004, a total of 1288 fuel assemblies were sent in four shipments to the La Hague reprocessing plant. For the remaining HIFAR fuel assemblies containing U.S. origin uranium, ANSTO decided to exercise its option to return the fuel assemblies to the U.S. under the Foreign Research Reactor Spent Nuclear Fuel (FRR SNF) Acceptance program. This paper will describe the organization and performance of the latest and largest shipment of HEU fuel elements under the FRR program from the HIFAR reactor in Sydney to the U.S. DOE's Savannah River Site (SRS) in South Carolina.

PAPER

Introduction: Background HIFAR

HIFAR was Australia's only multi-purpose research reactor for the past 50-years and has operated safely and effectively over this time. Over the past five years ANSTO has built a replacement research reactor – OPAL – which is now ready to take over from HIFAR in supplying neutrons for research, for industry, and for making nuclear medicine. There is, therefore, no further need for HIFAR and, accordingly, it was shut down on 30 January 2007 and will now undergo a staged decommissioning process.

HIFAR was originally built to test materials for use in future power reactors. The idea was to test materials in HIFAR by subjecting them to an intense neutron flux; a relatively quick assessment could be made of their suitability for use in a power reactor. With the decision not to pursue a power reactor program in Australia, there was a gradual change in how the reactor is used over the years.

The construction of HIFAR commenced in February 1956; it first went critical on 26 January 1958, and routine full power operation commenced in January 1960.

HIFAR is a copy of the DIDO reactor in the United Kingdom, which was, in turn, modelled after the CP5 reactor built near Chicago.

	CP5	DIDO	HIFAR
First went critical	February 1954	November 1956	26 January 1958
Thermal power	5 MW	Originally 10 MW Finally 25.5 MW Shutdown 3/90	10 MW
Peak thermal neutron flux (n/cm ² /s)	10 ¹⁴	2.3 x 10 ¹⁴	1.4 x 10 ¹⁴

The HIFAR reactor had 25 fuel elements containing enriched uranium encased in aluminium. Every four weeks, three of these fuel elements were replaced and are deemed “spent fuel”. HIFAR uses around 37 fuel elements annually.

HIFAR's spent fuel elements are each 600 mm long and 100 mm in diameter.

Some fuel used in HIFAR was manufactured using uranium enriched in the U. S. Such fuel, when spent, can be repatriated to the U. S. under the Foreign Research Reactor (FRR) acceptance program, where the U.S. Government takes ownership of it. The U.S. Government is then responsible for the safe storage and disposition of this material. No waste arising from the storage or handling of this spent fuel is returned to Australia from the U. S.

ANSTO also operated an Argonaut type 100 kW reactor (MOATA) from 1961 to 1995 when it was finally shut down. All of the fuel plates (HEU) have been removed, re-configured into shipment assemblies and returned to the US with the current 2006 spent fuel shipment.

Since commencement of the HIFAR operations there have been eight spent fuel shipments overseas totalling 2,142 Spent Fuel Elements (SFEs), as summarised below:

- 1963 Dounreay (UK) 150 SFEs
- 1996 Dounreay (UK) 114 SFEs
- 1998 US (Savannah River Site) 240 SFEs
- 1999 COGEMA (France) 308 SFEs
- 2001 COGEMA (France) 360 SFEs
- 2003 COGEMA (France) 344 SFEs
- 2004 COGEMA (France) 276 SFEs
- 2006 US (Savannah River Site) 330 SFEs

Shipment Preparation

The initial discussions between ANSTO and NAC International (US company specialized in nuclear materials transport, spent fuel management technology and nuclear fuel cycle consulting) took place in 2004. By mid 2005, an initial contract was in place. NAC would provide three NAC-LWT casks capable of transporting 126 fuel elements, as well as designing, licensing and manufacturing two new sets of baskets. In mid 2006, the contract was amended to include four additional casks (corresponding to a total of 204 fuel elements in the same shipment campaign) and also to integrate the transportation services for the seven empty and loaded casks to and from the Port of Sydney, Australia. The shipment consisted of five NAC-LWT casks containing 42 fuel elements each and two TN 7/2 casks each containing 60 fuel elements each representing a total of 330 fuel elements. This was the largest shipment of HEU fuel elements removed from one single facility under the FRR program to date. While the TN 7/2 casks owned by Nuclear Cargo Services had been used during numerous occasions at ANSTO, the NAC-LWT cask had not previously been used at the ANSTO facility.

Licensing and Basket manufacturing effort

Three types of fuel elements were part of the shipment (Type 1, Type 2 and Type 3).

The Type 1 and Type 2 fuel types are cylindrical elements while the Type 3 fuel consists of flat plates.

The TN 7/2 casks were used to ship the Type 1 fuel elements for which they were already licensed. The NAC-LWT casks were used to transport the three different fuel types. The NAC-LWT cask was already licensed for the Mark IV fuel, as NAC had previously shipped such fuel from the Danish research reactor of Risø. However, the NAC-LWT casks had not been licensed as an authorized package for the Type 2 fuel elements. Additionally, the Type 2 fuel elements could not fit into the existing DIDO baskets as the fuel is 4.5 mm (0.18”) larger in diameter than the Type 1 fuel. Consequently, NAC obtained a license for and manufactured a new basket type to accommodate the Type 2 fuel elements. In order to maintain a maximum capacity of 42

fuel elements per cask, NAC designed a new basket with seven cells (six baskets/cask). This new basket design, with the same capacity as the original DIDO basket, was possible in part because of the low heat load of the fuel which had been cooling for more than 20 years.

Regulatory approval was also necessary for the Type 3 fuel plates, which were re-assembled by ANSTO from their original Reactor Fuel Element configuration of 12 plates with 15mm spacers into 14 Shipment Elements with spacers of 1.5mm to reduce the K effective to below 0.5 prior to packaging.

A request to amend the NAC-LWT certificate of compliance was submitted by NAC to the U.S. Nuclear Regulatory Commission (NRC) in April 2006. Upon completion of its review, the NRC issued a revised NAC-LWT certificate of compliance in August 2006. In order to apply to the Australian Maritime Safety Authority (AMSA) and the Australian Radiation Protection and Nuclear Safety Agency (ARPANSA) for an Australian validation of the package, NAC received a U.S. Department of Transportation (DOT) certificate in August 2006. The Australian validations were issued in October 2006.

Project Preparation

A project kick-off meeting was held between ANSTO and NAC at the Lucas Height site in July 2006 to initiate the project. During this meeting, the ANSTO and NAC project teams assessed the site layout and infrastructure, selected a cask loading method, reviewed the cask documentation and agreed on a project schedule.

All maintenance of the NAC-LWT casks, packing, and preparations for shipment of all support equipment and parts were performed at NAC's storage and maintenance facility located at Wampum, Pennsylvania.

The two German-based TN 7/2 casks were already at Savannah River Site (SRS), which simplified their inclusion in the outbound shipment. Recently, operators of liner ships have become more reluctant to accept radioactive cargo. Consequently, it was decided to ship all seven casks and three additional 20' ISO containers of equipment together on one ship in order to facilitate the booking acceptance for the cargo using a liner-ship.

Site Preparation

Organizations utilizing the German TN 7/2 or the French TN-MTR casks will be well aware of the preparations required before and after loading the casks with spent fuel whilst submerged in a pond. The NAC-LWT cask presents a completely different scenario; however it is one that is easy to operate despite the larger volume of support equipment. Loading the baskets in the pond and retrieving them into the Dry Transfer System (DTS) is no different from the principles employed of withdrawing fuel from a pond using a specified cask and depositing it in dry storage.

Given that not many operators of research reactors have pools sufficiently deep to receive the NAC-LWT cask nor, indeed, buildings of sufficient headroom and crane capacity (23 tones) to conduct this exercise inside, this is best done outside by deploying a 100 tones mobile crane.

Preparatory steps:

- Availability of a suitable working area for all the containers which has sufficient space to employ a side loading device to lift the container from the ground on to a transporting trailer.
- Availability of a reinforced concreted area to receive each cask at the loading area which is flat, level and of sufficient area to receive the container.
- The base plate area for the NAC-LWT to be mounted is flat and level concrete-base.

At ANSTO all seven casks were received, inspected, surveyed and unloaded in the presence of NCS and NAC technicians.



Cask Receipt/Inspection/Unloading Area



Cask Survey prior to setting up casks



NAC-LWT Cask Loading Area



TN 7/2 Cask Loading Preparation Area

The full complement of ten ISO containers, including the seven casks and three ancillary equipment containers, were received in a dedicated area where they underwent comprehensive smear swipe testing to check for potential contamination.

Cask Loading Operation

As stated above, the TN 7/2 casks had previously been used at ANSTO. Consequently, the same wet loading method was repeated. It took a total of ten working days to load and test the two TN 7/2 casks.

The reactor pool at ANSTO is too shallow to permit a wet loading of the tall NAC-LWT cask. It was therefore decided at an early stage to perform the cask loading operations using the NAC Dry Transfer System (DTS) with the pool adapter (shielded tube placed across the top of the pool). This concept was approved by the Australian Regulatory agency (ARPANSA). The DTS is a transfer cask used to move one loaded basket from the pool into the cask. Each cask accommodates six baskets.

The photos below provide a pictorial overview of the cask loading operation. This method was new to ANSTO. However, it is a well-proven process which has been used in more than 20 facilities worldwide.



NAC provided technical guidance to ANSTO during the cask loading operation which took place from the beginning of October until mid-November, 2006. The time taken to load and test the five NAC-LWT casks was less than four weeks.

Shipment

The convoy of seven loaded casks with attendant three containers of equipment was transported under high security to a nearby port. After a long maritime voyage, the casks were delivered safely to the DOE site at Savannah River in South Carolina.

Conclusion

This shipment is the largest spent nuclear fuel shipment ever performed from a single facility under the FRR acceptance program (330 fuel elements in total representing approximately 30kg of U235). It was a very comprehensive project including design, licensing, basket manufacturing, usage of seven casks, technical assistance for cask loading operations and transportation activities. Adding to the complexity of the project was the numerous entities involved in all the different tasks and jurisdictions. The successful shipment provides a useful precedent for the shipment of the remaining HIFAR spent fuel will be shipped to SRS in the coming years.

THE EXPERIENCE OF SHIPPING SPENT NUCLEAR FUEL FROM UZBEKISTAN TO THE RUSSIAN FEDERATION

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ABSTRACT

In April 2006 the last of four shipments of spent nuclear fuel left the Institute of Nuclear Physics outside of Tashkent, Uzbekistan and traveled to the Mayak site in the Russian Federation. The shipment marked the completion of the first campaign under the National Nuclear Security Administration's Russian Research Reactor Fuel Return (RRRFR) Program to return highly enriched spent nuclear fuel to its country of origin. In total, 252 spent fuel assemblies containing over 63 kg of highly enriched uranium were returned. The project proved to be an excellent example of cooperation as four countries, Uzbekistan, Russia, Kazakhstan and the United States, were involved in its planning and implementation. This paper will describe the shipment process from planning to completion with emphasis placed on the critical activities. Specifically the paper will discuss: the activities performed to prepare for the shipments; the roles and responsibilities of each country; the shipment details; the lessons learned; and the future plans of the Institute and the RRRFR program.

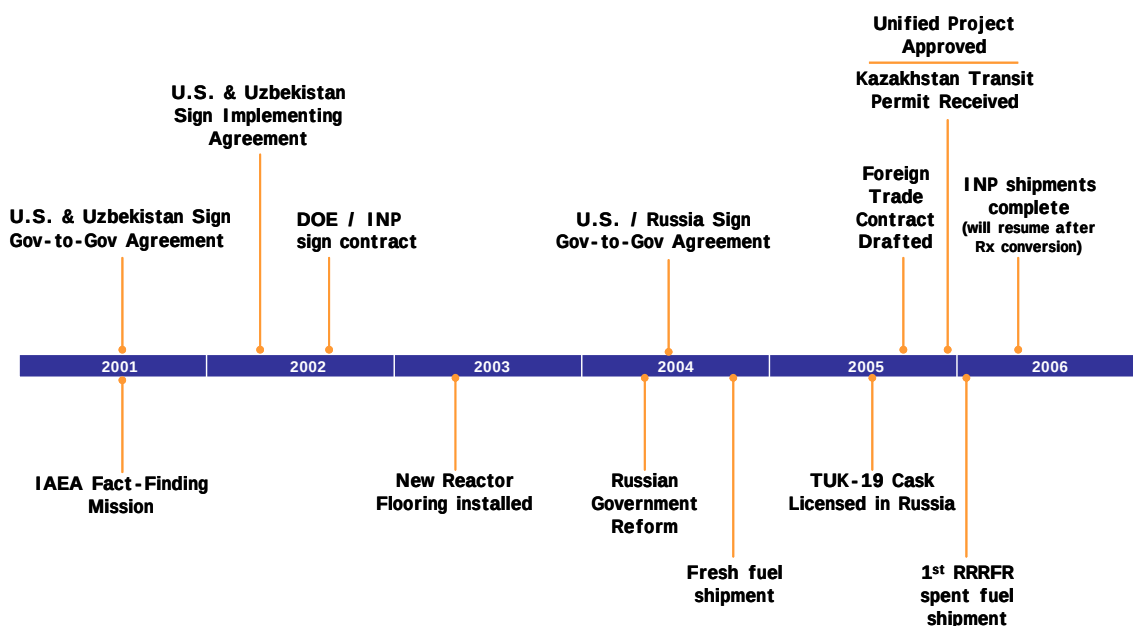
Introduction

The research water-water reactor of the Institute of Nuclear Physics of Uzbekistan Academy of Sciences (Tashkent, Uzbekistan) is operating since 1959. From the beginning till 1977 the reactor was working at thermal power 2 MWt using high enriched fuel. In 1977-1978 after upgrading the reactor started to operate at power 10 MWt which helped to increase significantly

the flux of neutrons and, as result, to start mass production of isotopes for needs of medicine, science and technique. In former Soviet Union the Institute of Nuclear Physics (INP) was one of the leading facilities producing reactor isotopes at the commercial level. Nowadays the INP is producing isotopes like I-125, I-131, P-32, P-33, S-35, Au-198, Ir-192, Sr-89, Re/W- and Tc-99M generators and other ready to use preparations for internal needs as well as for export. In addition, the reactor is used for neutron activation analysis, material sciences, ennobling the precious and semiprecious stones and for studies in nuclear physics.

For the last fifteen years reactor was operating, in average, more than 5000 hours a year using from 1978 till 1997 the 90%-enrichment fuel and then, from the middle of 1997 till the present time, the 36%-enrichment Russian made IRT-3M type fuel. Such an intensive use of reactor, in turn, created a problem of handling, in particular, of spent fuel the storage facility for which is limited. The last shipment of spent fuel back to Russia was done in 1991 and due to known reasons (decrees of Russian government in 90's on not accepting highly radioactive waste, in particular) the INP was not able to send spent fuel and had to keep it building two new additional storage facilities. In result, by the beginning of 2006 more than 300 spent fuel assemblies were stored at facility and, because of limited space, the problem of further operating of reactor became serious.

However, in January 2006, Uzbekistan became the first country in fifteen years to return spent nuclear fuel (SNF) to the Russian Federation and the first under the Russian Research Reactor Fuel Return (RRRFR) Program. The RRRFR Program was created in 1999 from a tri-partite initiative between the Russian Federation, United States, and the International Atomic Energy Agency (IAEA) to return Russian-origin research reactor fuel containing high enriched uranium (HEU) from countries of the former Soviet Union. After the signing of the Implementing Agreement between the Government of Uzbekistan and the United States Department of Energy (DOE) in March 2002, the Institute of Nuclear Physics (INP) began the initial planning for the return of SNF from its WWR-SM research reactor¹. Progress of the project was slow at first until the signing of the, *'Agreement Between the Government of the United States of America and the Government of the Russian Federation Concerning Cooperation for the Transfer of Russian-Produced Research Reactor Nuclear Fuel to the Russian Federation'* in May 2004, which gave the project the legal basis to proceed. The project experienced frequent challenges due to the fact that many of the laws, regulations, and procedures had changed dramatically since the last shipments of spent fuel in 1991. Through persistence and commitment to support the



goals of the Global Threat Reduction Initiative (GTRI) and the RRRFR Program, the Russian Federation, Uzbekistan, Kazakhstan and the United States successfully coordinated the completion of the required preparation activities and shipped 252 SNF assemblies containing 63 kg of HEU. The timeline below highlights the major milestones of the project.

This paper will not attempt to discuss all of the activities completed over the past two years but will focus on providing the details of the critical preparation activities and the actual shipment. The critical preparation activities were: Government-to-Government Agreements; Unified Project; TUK-19 cask licensing; Kazakhstan transit requirements; and facility preparations. The organizations who were involved will also be identified along with a brief description of their

¹ Initial criticality was reached in 1959 using EK-10 fuel assemblies. The reactor operated at 10MW with 90% enriched IRT-3M fuels from 1971 to 1997. Since conversion in 1997, the reactor has been using 36% IRT-3M fuel assemblies.

responsibilities. The details of the shipments (i.e. cask loading, logistics) will be described followed by the lessons learned and future plans of the reactor.

Critical Preparation Activities

Before the critical preparation activities are highlighted and discussed, it is important to identify the major organizations who were involved with the shipments and their roles and responsibilities. Table 1 includes the information on these organizations.

Table 1: Project Team

Organization	Country	Description and Responsibilities
NNSA	United States	National Nuclear Security Administration – division of the DOE that manages and funds the RRRFR Program.
INP	Uzbekistan	Primary contractor with NNSA and primary contractor with Mayak. INP provided project management and was responsible for all of the activities within Uzbekistan.
Rosatom	Russian Federation	Federal Atomic Energy Agency – responsible for regulating the import of research reactor fuel.
Mayak	Russian Federation	Prime contractor with INP. Mayak was the shipper of record, provided the shipping containers and rail cars, unloaded the fuel, and is responsible for reprocessing and interim storage.
Techsnabexport (TENEX)	Russian Federation	TENEX – one of the two companies in the Russian Federation authorized by the Russian Government to import spent nuclear fuel. TENEX was subcontracted by Mayak to complete the Unified Project and authorize the import of the SNF.
VNIPIET and VNIIEF	Russian Federation	Subcontracted by TENEX to perform the safety analyses, prepare the required documentation, and obtain the licenses for the TUK-19.
KATEP	Kazakhstan	Company authorized to manage spent fuel shipments in Kazakhstan. KATEP coordinated all activities for the transit of the spent fuel.
KAEC	Kazakhstan	Kazakhstan Atomic Energy Committee – nuclear regulator for Kazakhstan. Approved the transit permits and cask license.

Government-to-Government Agreements (GTGA)

The first critical activity and major prerequisite for the preparation activities was the establishment of the government-to-government agreements between Uzbekistan and the United States and Uzbekistan and the Russian Federation. Two GTGAs provided the legal framework between the U.S. and Uzbekistan. They were: the “US/Uzbekistan Non-Proliferation

Agreement” signed in June 2001; and the “DOE/Ministry of Foreign Affairs (MFA) Non-Proliferation Agreement” signed in March 2002. The first agreement provided liability protection and tax exemption for non-proliferation activities and the second delegated the DOE and the MFA as agents with permission to enter into a contract for the spent fuel return. The Governments of Uzbekistan and the Russian Federation agreed upon and signed an agreement (1997) on the peaceful use of atomic energy in which the focus was the dedication of the management of spent fuel. This GTGA was important because it served as the legal basis for the importation of spent nuclear fuel and the basis for the development of the Unified Project (to be discussed later), which was another critical path activity. With the government-to-government agreement in place, INP was permitted to contract with Mayak for the return of the spent fuel. This contract called the Foreign Trade Contract and discussed later in this paper, specifically addressed the following issues:

1. The scope of services to be provided by the Russian Federation. Services included temporary storage of SNF, SNF processing, interim storage, and radioactive waste return.
2. The contract defined the owner of the SNF after their importation and the owner of the reprocessing products after SNF reprocessing.
3. The contract included confirmations from Uzbekistan regarding the acceptance of the radioactive waste after a period of twenty years and assurances that all requirements are and will be met for the safe transportation of the SNF. The issue of the return of the radioactive waste after reprocessing is important because if the country decides to have the waste remain in the Russian Federation, additional costs (sometimes substantial) would result.

For future shipments of spent fuel, Rosatom has stated that the legal issues above should be included as part of the government-to-government agreement with the Russian Federation. The negotiation and approval of the government-to-government agreements are a lengthy evolution. Experience dictates that at least one year should be allotted for planning purposes.

Unified Project

A Unified Project was required for the importation of spent nuclear fuel into the Russian Federation per Russian Law [1]. The Unified Project was basically an overall assessment of the radiological, economical, social, and environmental impacts to the Russian Federation, particularly the areas surrounding the Mayak Plant (Chelyabinsk Region) [2]. The various elements that are included in the Unified Project are briefly discussed in the following paragraphs.

The first part of the Unified Project included the documents that make up the Special Ecological Programs (SEPs). The SEPs are used to rehabilitate the radioactive contaminated areas of the territory surrounding Mayak and are financed by the receipt of the SNF from foreign customers. In this case, the programs provided support to activities associated with the V-9 Industrial Water Basin and the development of systems for dosimetry, radiometry, and spectrometry monitoring. The SEPs went through a vigorous review process, with reviews by Rosatom, the Ministry of Economic Development and Commerce, and Medbiokestrem, culminating in a State

Environmental Expert Review (SEER) by Rostekhnadzor. A positive outcome from Rostekhnadzor meant that the SEPs could be included in the final Unified Project package.

The second part of the Unified Project was the draft Foreign Trade Contract for the processing and storage of the SNF. The draft Foreign Trade Contract contained:

1. The number of SNF assemblies to be shipped
2. The scope and cost of the services provided
3. Confirmation of the decision by the originating country to accept the return of the high level waste
4. Total project cost
5. Durations of temporary and interim storage

The third part comprised of a set of documents that substantiate an overall radiation risk reduction and environmental safety increase as a result of the Unified Project implementation. These documents also address the storage durations and hazards associated with the products of the reprocessing activities. An additional document titled ‘Assessment of Environmental Impact (AEI)’, not required at the time of the Uzbekistan shipment Unified Project, has been recommended by the SEER to be included in future Unified Projects.

The fourth and final part of the Unified Project was the set of materials used to discuss the SNF importation project with the community members and public organizations in the areas affected by the shipments. In this case, the records of discussions included people of the Chelyabinsk Region, city of Ozersk, and Mayak employees.

Once all of the required documents were collected into the final Unified Project package, it was submitted to Rostekhnadzor for the State Ecological Expert Review. Positive results were transmitted to Rosatom, the Foreign Trade Contract was signed, and the Russian Government issued the declaration authorizing the importation of the SNF from Uzbekistan. Based on this experience, a duration of 15 months is recommended to develop and obtain SEER approval of the Unified Project.

TUK-19 Cask Licensing

In the Russian Federation, casks transporting radiological materials must be licensed for both design and transportation [3]. The design license for the TUK-19 (right) had been allowed to expire in 2000 due to its inactivity. The transportation license, which is shipment specific and issued for each shipment campaign, required development. The transportation license included information such as: duration of the shipment; actual radioactive content; mode of transport; emergency card information; and proposed shipment



category to name a few. Both the design and transportation licenses were analyzed and prepared by VNIPIET in less than five months with approval by Rosatom following shortly thereafter.

The Kazakhstan license for the design and transportation of the TUK-19 cask was issued by the Competent Authority, Kazakhstan Atomic Energy Committee (KAEC) of the Ministry of Energy and Mineral Resources (MEMR). The approval process involved reviews from both independent and state experts. The TUK-19 license in the Russian Federation was issued according to the regulatory guidelines developed in Russia, not the IAEA TS-R-1 guidelines adopted by the KAEC. Therefore, KAEC requested that VNIPIET prepare a comparative analysis that confirmed the compliance of the TUK-19 safety analysis to the IAEA TS-R-1 guidelines. KATEP was chosen to coordinate all of the TUK-19 licensing activities and the license was issued by the KAEC in less than four months.

The TUK-19 license validation in Uzbekistan was issued by the State Inspectorate on Safety in Industry Mining using the Russian license. This activity was completed within two months.

Kazakhstan Transit Requirements

As with the TUK-19 cask license, KATEP coordinated all of the transit activities for this shipment. This included the following main activities:

- Development of ‘Assessment of Radiation Impact of SNF Transit to Environment and Population (EIA)’ and receiving the State ecological conclusion.
- Purchase of the required insurance (obligatory and voluntary) policies for the SNF transit.
- Development and approval of the SNF Transit Program.
- Obtaining the permission to transit through Kazakhstan
- Signing of the contracts for: rail transportation; physical protection; emergency preparedness; and customs.

The transit program was quite extensive and included provisions for liability, route selection, security, physical protection, and emergency preparedness. All competent authority approvals were coordinated by KATEP and received in less than three months.

Facility Preparations

A number of facility and equipment enhancements were completed to support the loading and shipping of the TUK-19 casks. The major activities completed were:

- New reactor hall flooring was installed to increase safety and help prevent the spread of contamination.
- New reactor hall lighting and remote cameras were installed to improve the conditions for fuel and cask handling. Previously, the crane operator used mirrors and visual cues to assist with the alignment of the basket and cask. The new remote cameras improved the loading operations (right), making loading quicker and safer.



- A backup generator was installed to provide emergency power if electrical power was lost during the loading operations.
- New transport racks were fabricated to secure the TUK-19 casks to the trucks during transport from the reactor to the rail yard.
- New trucks were procured to ensure the safe transport of the SNF and to reduce the number of road transports.
- Additional radiological monitoring and communications equipment was purchased.
- A self-releasing grapple was designed, fabricated and used to load the basket containing fuel assemblies into the cask.

The reactor staff and support organizations received extensive training on the operations and procedures of every aspect of the fuel shipments. Many practice exercises were performed on: fuel loading; cask handling and loading; cask preparation; criticality and personnel safety; radiological safety; and security.

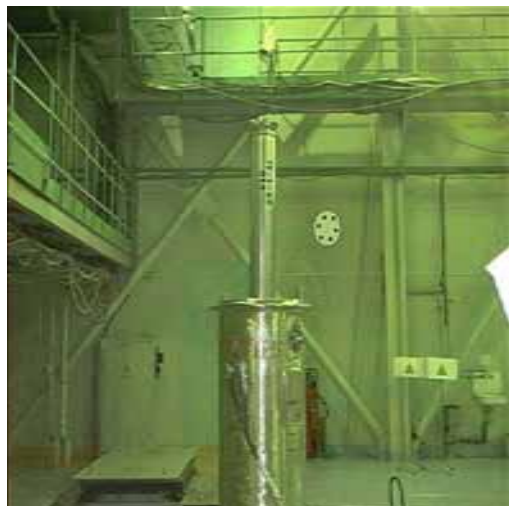
Shipment Details

The shipment consisted of the transport of 252 IRT-3M spent fuel assemblies enriched to 36% and 90% ^{235}U . The IRT-3M assembly is shown in the left picture below. Several months prior to the shipment, the fuel assemblies were inspected by Mayak experts. All of the spent fuel assemblies met the acceptance criteria [4] for shipment and receipt with none requiring encapsulation. The TUK-19 cask was chosen because it was designed for Russian research reactor fuel and for use in the Russian designed reactors. The TUK-19 has the capacity to hold 4 IRT-3M assemblies and a total of 16 casks were available for each shipment. The casks were transported to Mayak by rail in 2 TK-5 railcars. Each TK-5 railcar holds 8 TUK-19 casks and has a roof that can be opened for loading and unloading operations (below, right). With a maximum of 64 IRT-3M fuel assemblies transported in each shipment, four shipments were needed to return the 252 spent fuel assemblies to Mayak.



The shipment process was identical for each shipment. The TK-5 railcars transported the empty TUK-19 casks from Mayak, through Kazakhstan to a rail yard near INP. The casks were off-loaded and transported to the reactor hall and staged for loading. The casks were allowed to acclimate for 24 hours before opening. Detailed cask loading plans were prepared in advance to

ensure that none of the cask license contents limits (i.e. decay heat, activity, cooling time) were exceeded. IAEA inspectors were present and verified the presence of ^{137}Cs in 100% of the spent



fuel assemblies. Each of the measurements was taken during the basket loading process and did not significantly affect the loading process. Once the four spent fuel assemblies were placed in the basket, the basket was remotely raised out of the spent fuel pool by the overhead crane and allowed to drip dry (removal of most of the water) for 15 minutes. After the drying period, the basket was placed into the cask. The cameras at this point proved to be a tremendous improvement to historical loading operations. Due to the in-air loading the grapple was designed to self-release once the basket was fully lowered in the cask. This grapple worked flawlessly during all 64 cask loading operations. One reactor operator along with two radiological protection operators entered the reactor

hall carefully monitoring the radiation levels. The operators were able to approach the cask and connect the crane hook to the cask lid. The cask lid was then installed on the cask and secured with two bolts prior to movement to its assigned storage spot. The remaining bolts were installed and torqued and the cask prepared for hermetic seal testing. A helium detector was used to confirm a proper seal per the TUK-19 handling instructions. The time to load a TUK-19 cask averaged less than one hour per cask.

On the day of the shipment, the TUK-19 casks were transported to the rail yard and loaded into the TK-5 railcars while under constant security surveillance. Final surveys were conducted and the shipment left by a dedicated train at the predetermined time specified by the authorities. The transit time from Tashkent to Mayak was less than four days and the total turnaround time to return the empty casks was approximately three weeks. All four shipments were completed in less than four months, four months ahead of the baseline schedule. There were no incidences reported during the loading of the casks at INP and unloading of the fuel at Mayak.

Future Plans

As reported earlier, the reactor currently operates with 36% enriched IRT-3M fuel assemblies. The Government of Uzbekistan and INP have decided to convert the reactor to use low enriched fuel, specifically, 19.7% IRT-4M fuel assemblies. Reactor conversion analyses with assistance from Argonne National Laboratory under the Reduced Enrichment for Research and Test Reactors (RERTR) program have been performed and final reactor parameters are being reviewed. There is a possibility that the reactor's power level may be increased to approximately 11 or 12 MW in order to preserve previous flux values. Other projects such as the refurbishment of the secondary cooling loop piping and the reactor control system are planned with their completion contingent upon funding support.

Lessons Learned

Because this was the first shipment of research reactor spent fuel to the Russian Federation, a significant amount of information was learned that would apply to future shipments from other RRRFR Program countries. At the time this paper was written, the Uzbekistan shipment project team and the IAEA were finalizing the plans for a lessons learned workshop to be held in early October. The focus of this workshop is to transfer knowledge and information regarding the necessary technical and administrative preparations to institutions planning future shipments. Ultimately, this information will be collected and published in a guideline document and intended for use as a reference tool. Sample passport and receipt documents, agreements, and contracts will be shared at this workshop. This experience demonstrated the need to identify all of the legal and technical requirements as soon as possible. It is also recommended that a dedicated project manager or team be appointed due to the significant work load. Lastly, it is important to have Mayak experts characterize and inspect the fuel to be shipped well in advance of the shipment to help reduce delays associated with suspect or deformed fuel assemblies.

Conclusion

The completion of the four shipments of spent HEU fuel from the Institute of Nuclear Physics of the Uzbekistan Academy of Sciences to the Russian Federation was a tremendous accomplishment for the Russian Research Reactor Fuel Return Program and the Global Threat Reduction Initiative. It marked the first return of spent research reactor fuel to the Russian Federation in over fifteen years and the first under the RRRFR program. Much was learned during the preparations phase, with many of the challenges requiring the development of new procedures to meet the updated regulations. In the end, the project proved to be an excellent example of international cooperation between the Russian Federation, Uzbekistan, Kazakhstan, and the United States in the area of nonproliferation.

The authors would like to extend their appreciation to Rosatom, the Governments of Uzbekistan and Kazakhstan, and the U.S. Department of Energy for their support of these spent fuel shipments and the RRRFR Program. Special thanks are also expressed to the team of experts from INP, Mayak, KATEP, TENEX, INL, SRS, the IAEA and others for their professionalism and excellent work.

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POSSIBILITY OF A PARTIAL HEU-LEU TRIGA FUEL SHIPMENT

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ABSTRACT

The TRIGA reactor Vienna operates since March 1962 and was initially started up with 66 TRIGA fuel elements type 102 (Al-cladding, LEU). In the following years additional spare fuel elements of type 104 (SST cladding, LEU) were purchased which were added to the core. Therefore the core was composed of two types of fuel elements. Later on in 1974 fuel elements type 110 (SST cladding 70% enriched) were acquired and placed into the core resulting in a complete mixed core with 3 different fuel element types. As the reactor power of 250 kW is rather low only a very few fuel elements had to be removed from the core in the past 45 years, mainly because of mechanical damage during fuel handling. In fact in all those years only 8 fuel elements are removed permanently from the core and stored in a dry spent fuel storage pit. This paper describes MCNP modelling of the TRIGA Vienna core if FLIP elements are replaced by standard fuel elements

1. Introduction

As a result of the US spent fuel return program the HEU fuel elements are the main target of DOE. For the TRIGA reactor Vienna it is however necessary to continue operation as long as possible. Recently possible solutions were worked out and will be discussed in more detail in this paper:

1. Austria could possibly obtain 15 fresh TRIGA elements from the TRIGA Slovenia and could thus remove the HEU from the operating core.
2. Further a fuel shipment is planned from Romania in 2008 which could be used for the Austrian HEU and the 8 LEU spent fuel elements if these fuel elements are shipped with proper timing from Vienna to Pitest. It all depends on careful planning and excellent cooperation with the involved parties.

2. The fuel history

The 250 kW TRIGA Mark-II reactor operates since March 1962 at the Atominstitut, Vienna, Austria. Its main tasks are nuclear education and training in the fields of neutron- and solid state physics, nuclear technology, reactor safety, radiochemistry, radiation protection, dosimetry, low temperature physics and fusion research. Criticality of a typical TRIGA Mark II reactor is usually achieved with about 57 standard TRIGA fuel elements (about 2 kg ^{235}U). To allow higher power operation (100 kW) more fuel elements and several graphite reflector elements are usually added in the outer ring of the core. This results in an operating core of about 66 fuel elements and a core excess reactivity of about 2 \$ depending on specific license of such a reactor. Since the first criticality further fuel elements were purchased and added to the core. During the past decades the fuel supplier (General Atomics) has changed the fuel specifications several times. Since the development of the original TRIGA type reactors in the mid-fifties, several new types of TRIGA fuel elements have been developed. A significant change in the design was replacing the aluminium cladding material to stainless steel (SST).

Therefore, many TRIGA reactors are operated with a mixed core using different fuel elements in the core. Among the SST elements several different types of fuel exist. Most widely used fuel elements are TRIGA elements with <20% enrichment (Standard SST) and in some cases fuel elements with approximately 70% enrichment (called FLIP = **F**uel **L**ifetime **I**mprovement **P**rogram). Details about these different fuel types are given in Table 1. Recently, General Atomics has designed and certified a new low enriched fuel with high uranium loading (as high as 20 wt-%) to replace the FLIP fuel. Gradually, FLIP research reactors are planned to be converted to low enriched uranium (LEU) fuel due to worldwide nuclear proliferation concerns. This new LEU fuel has higher uranium loading to give a long fuel lifetime without high enrichment. Additionally, General Atomics has produced a TRIGA fuel with a smaller diameter for higher (>3MW) power operations.

Fuel element type	102	104	110 (FLIP)
Fuel moderator material	U-ZrH _{1.0}	U-ZrH _{1.65}	U-ZrH _{1.65}
Uranium content (wt%)	8.5	8.5	8.5
Enrichment (%)	20	20	70
Average ²³⁵ U content (g)	38	38	136
Burnable poison	SmO ₃ -disk	Mo-disk	Erbium 1.6 wt%
Diameter of fuel meat	35.8 mm	36.3 mm	36.3 mm
Length of fuel meat	35.6 mm	38.1 mm	38.1 mm
Graphite reflector length	10.2 mm	8.73 mm	8.81 mm
Cladding material	Al-1100F	304 SS	304 SS
Cladding thickness	0.76 mm	0.51 mm	0.51 mm

Tab 1: Specifications of fuel elements used in the core of the TRIGA reactor Vienna

At present nine Flip elements are installed in the Vienna TRIGA core. Eight of them are since 1972 in the C ring of the core, one is near the neutron source of the reactor in the F- ring. Since the elements are in the core more than 34 years, table 2 gives more detailed information about the nine high enriched elements.

Fuel Number	Fuel Typ	Burn up (MWd)	Uranium weight (g)	U-235 weight (g)
7301	Typ 110	5,3	194	135
7302	Typ 110	4,3	195	136
7303	Typ 110	5,3	194	135
7304	Typ 110	4,3	194	135
7305	Typ 110	4,3	194	135
7306	Typ 110	3,5	194	135
7307	Typ 110	3,98	194	135
7308	Typ 110	3.4	194	135
7309	Typ 110	3.3	194	135

Tab 2: History of the Flip elements at the TRIGA reactor Vienna

3. Present Situation

During the past RRFM meetings the possibilities to remove and return these nine FLIP elements were discussed with NNSA and DOE representatives. Presently the situation is as follows:

The TRIGA reactor Vienna is used intensively not only by the Vienna University of Technology and other Austrian universities for education and training purposes but also by the IAEA because it is now the only nuclear facility close to the IAEA headquarters. The previous 10 MW ASTRA reactor had been ultimately shut down in summer 1999, thus remaining the only nuclear facility in Austria. Removing these nine FLIP elements from the TRIGA core would seriously endanger the future operation of this reactor. Although some standard TRIGA fuel elements are on stock which will not be of sufficient worth to extend the reactor operation life till May 2016.

Contacts have been established with the Josef Stefan Institute/Slovenia as information was obtained that some fresh TRIGA fuel elements are available, however the latest information received was that they will be offered to CERCA in a compensation deal. Otherwise the plan existed to obtain about 9-10 fuel elements from Slovenia and to remove the FLIP elements from the TRIGA core for interim storage until a reasonable shipment to the US is available such as it is planned from Pitest/Romania in 2008. However it seems that the CERCA deal is not yet settled as in February 2007 information was received from IJS that all fuel is still in Ljubljana.

4. On-going Activities

Since fall 2006 Monte Carlo calculations started at the Atominsitute incorporated in one Diploma work and one PhD work to calculate the flux density changes when replacing the 9 FLIP element but standard 104 TRIGA fuel elements. As the Vienna TRIGA core is mixture of old Al-clad fuel elements (Type 102), SST clad elements from various periods and FLIP elements these calculations are rather complicated especially as it turned out that even among one series of fuel elements considerable variations in composition (i.e. H-content of FLIP elements) are present. The Monte Carlo method is used to simulate a statistical process such as interaction of nuclear particles with material and is particularly useful for complex problems which need more approximations (homogenisation and multi group cross-section treatment, etc.) when modelled by deterministic computer codes.

To model the TRIGA MARK II reactor, Vienna, the general purpose 3-D Monte Carlo N-Particle code MCNP was chosen because of its generalized geometry, continuous energy cross sections modelling capability and long history of use in reactor calculations. All fresh fuel, control rod, and other elements (e.g., neutron source, graphite dummy elements) models were prepared in order to be able to simulate any possible fresh core.

4.1 Methodology

A full core MCNP model up to the tank wall was created for the TRIGA Mark II Vienna using detailed core geometry and isotopic composition of each element in the core materials (Ref). All three types of fuel elements (102, 104, 110) control elements, the surrounding reflector and beam tubes were modelled in detail. In this frame work, two main cases were considered for simulation. In first case, core incorporates only two types of fuel i.e. 102 and

104. Three control rods i.e. regulating, safety and transient rod were kept fixed at position 500 (full out) to obtain the total excess reactivity of the core.

In second case, the nine 104-type fuel elements at positions C1, C2, C4, C5, C7, C8, C9, C11 and F27 were replaced by FLIP elements (as used in the actual core). All other conditions i.e. control rod and other elements position were kept unchanged. The top (XY) and vertical (YZ) view of the core simulation is in Fig.1.

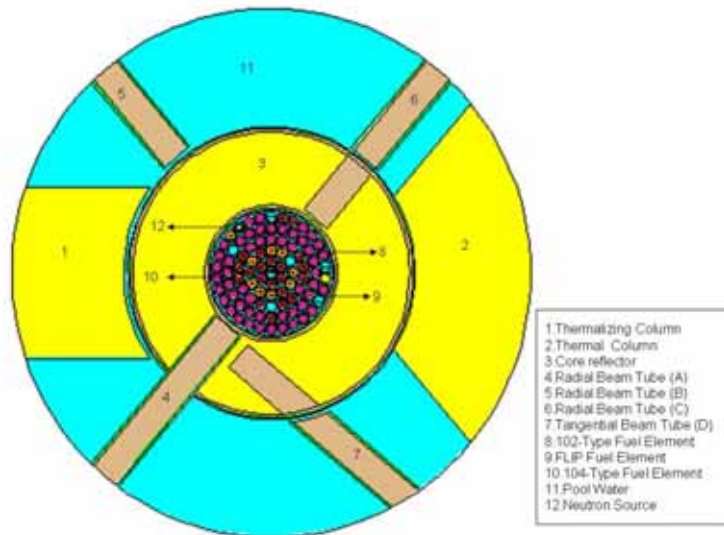


Fig 1. Horizontal cross section of the TRIGA Mark II core in Vienna

Using flux tallies, the radial and axial distribution of the fluxes were plotted for both case described above and are shown in Figs. 2 and 3 at the end of the paper.

In this Monte Carlo simulation, with the help of KCODE card, the effective neutron multiplication factors for both cases have been calculated. The simulation results give us more reliable information about the rod worth compared to the experimental results.

4.2 Reactivity Worth Difference between 102, 104 and 110 fuel elements.

Generally speaking there are three important characteristics for a fuel element in MCNP if the geometry is constant and well known: U235 mass, H content and poison mass. Concerning the poison the influence is small due to the low power and all results will be related to a non-Xenon-poisoned core, the other two factors are significantly different between fuel elements and the overall average can differ from the values given in the supplier's data sheets. While the U235 mass varies for about $\pm 0.5\text{g}$ (at an average of 38g per fuel element) the core average value can easily be obtained using the data from the shipping documents, however this is not the case with the H content. The H/Zr ratio is given for some of the elements and i.e. varies from 1.57 to 1.63 for the 9 FLIP elements. Even relative minor shifts in the ratio have a significant effect on keff simulations: In example changing the H-content from 1.6 to 1.65 (+0.0484 wt%) for the 104 elements (22 out of 82) results in a keff change of +1.12%.

While operating the reactor at a power level of 10W and replacing a 104 rod with a 110 rod at core position C1 produces a difference in control rod position and gives therefore an estimate of the reactivity worth difference between the two rods. In this case the difference is about 0.267 \$ per rod or a total of 2.4 \$ for the whole core (rel. error about 5-10% due to the

accuracy of the rod calibration curve). The simulated values are 0.262 \$ and 2.36 \$ respectively. If replacing a 104 rod with a 102 fuel element in position F2 with similar burn-up the reactivity change is extremely small with -0,02\$ and in the order of magnitude of the measurement precision.

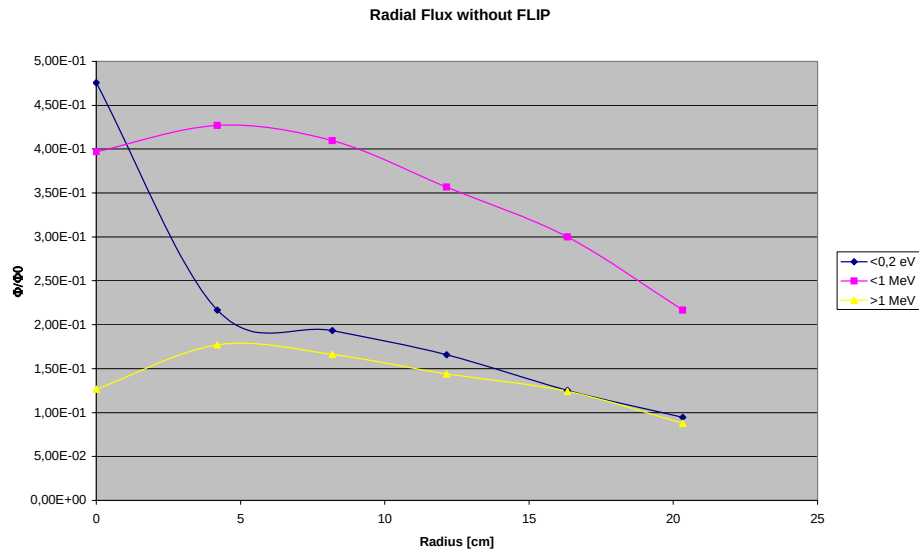


Fig. 2: Relative energy dependent radial flux without FLIP integrated over the fuel meat height from the core centre to the outer ring (Normalisation: Total flux at $r=0$:=1).

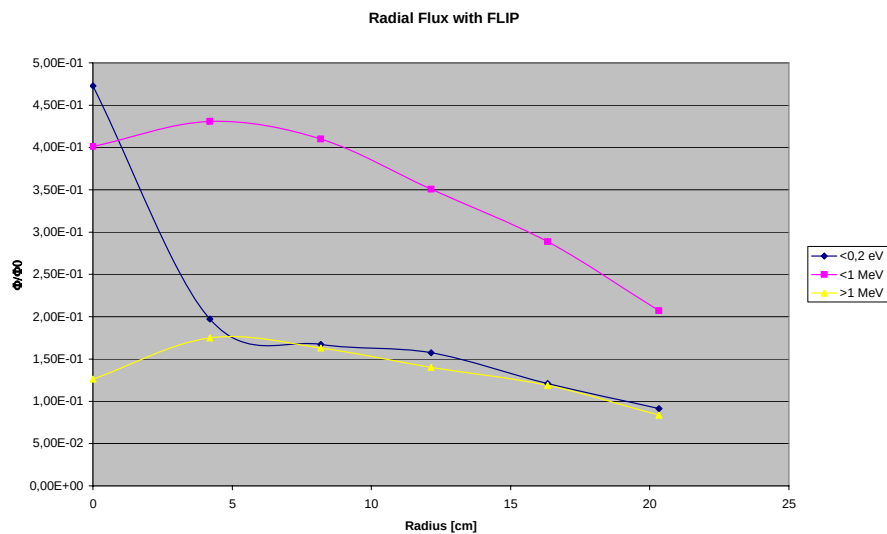


Fig. 3: Relative energy dependent radial flux with FLIP integrated over the fuel meat height from the core centre to the outer ring (Normalisation: Total flux at $r=0$:=1). Note the relatively lower thermal flux around the FLIP elements ($r=5-10$ cm) compared to the 104 elements at the same position in the previous diagram.

5. Summary and conclusions

Calculations show that there is no major difference in core performance between a TRIGA core with 9 FLIP elements and TRIGA core using 20% enriched standard fuel elements except in long time burn-up behaviour. The main problem with the TRIGA reactor Vienna reactor is that there are only 8 spent fuel elements stored and a partial fuel shipment of these stored elements together with 9 FLIP elements is extremely expensive, further when initiating such an effort a high risk of a governmental decision to shut down the reactor completely is imminent.



Poster Session

THE NEW AREA RADIATION MONITORING SYSTEM OF THE TRIGA NUCLEAR RESEARCH REACTOR FACILITY OF THE UNIVERSITY OF PAVIA

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The area radiation monitoring system of the TRIGA nuclear research reactor facility of the University of Pavia has been renewed after 30 years of operation. The new system is based on a commercial micro-computer and an home-made software developed on Lab-View platform. The system collects the data sampled by six β - γ dose-rate proportional counters, a free-air ionization chamber and a weather station through a serial data bus line RS232. Collected data are displayed through a desktop PC in the reactor control room and are also accessible, for a restricted number of users, through internet using the TCP/IP protocol. The software allows the operator to access the data, to modify parameters and perform tests remotely, by means of any common web browser.

For an improved safety level, data are stored both in the micro-computer and in the desktop PC, both accessible remotely. The system is provided by output relays that activate automatically the nuclear alarm detection system of the facility when pre-set levels of environmental radiation dose-rate are exceeded. A watchdog, integrated into the micro-computer, tests the whole data acquisition system regularly in order to prevent possible software or hardware failures.

1. Introduction

The purpose of the new radiation monitoring system is to perform a continuous acquisition of the radiation level throughout the facility. According with the requirements of the Radiation Protection Officer of the reactor facility, the following parameters are constantly monitored:

- β - γ Dose-rate inside the reactor room and inside the air outlet chimney
- β - γ Dose-rate of the water of the reactor primary cooling circuit
- Activity of the air particulate inside the reactor room
- Environmental data collected inside and outside the building (i.e. wind speed and direction, pressure, inside and outside temperatures)

2. General description

The system is made of a network of different instruments coupled, through a serial bus line, with a data acquisition station.

The detection system is configured as follows (see Fig.1):

- three Berthold LB111 proportional counters equipped with two β - γ dose-rate probes each, placed as follows: four probes mounted inside the reactor room at each cardinal point position and at the same level, one probe mounted on the reactor room ceiling, exactly over the reactor open tank and one probe inside the air outlet chimney;
- one proportional counter Berthold LB111 with one β - γ dose-rate probe for the monitoring of the water of the reactor primary cooling circuit;
- a free air Ionization Chamber for the monitoring of the activity of the air particulate inside the reactor room;
- one weather station Peet Brothers placed on the roof of the building, connected to Peet Brothers U2000 instrument (placed in the reactor control room) for the acquisition of environmental data;
- one National Instruments Compact Field Point (NI CFP 2020), placed in the reactor control room, for radiation monitoring data acquisition and analysis;

- one personal computer (Host PC), placed in the reactor control room, for radiation monitoring data display (throughout a graphic interface) and storage.

In order to get the best reliability and to ensure a continuous operation, the system power supply is served by an Uninterruptible Power Supply (UPS). When a power failure or abnormality occurs, the UPS will effectively switch from utility power to its own power source instantaneously.

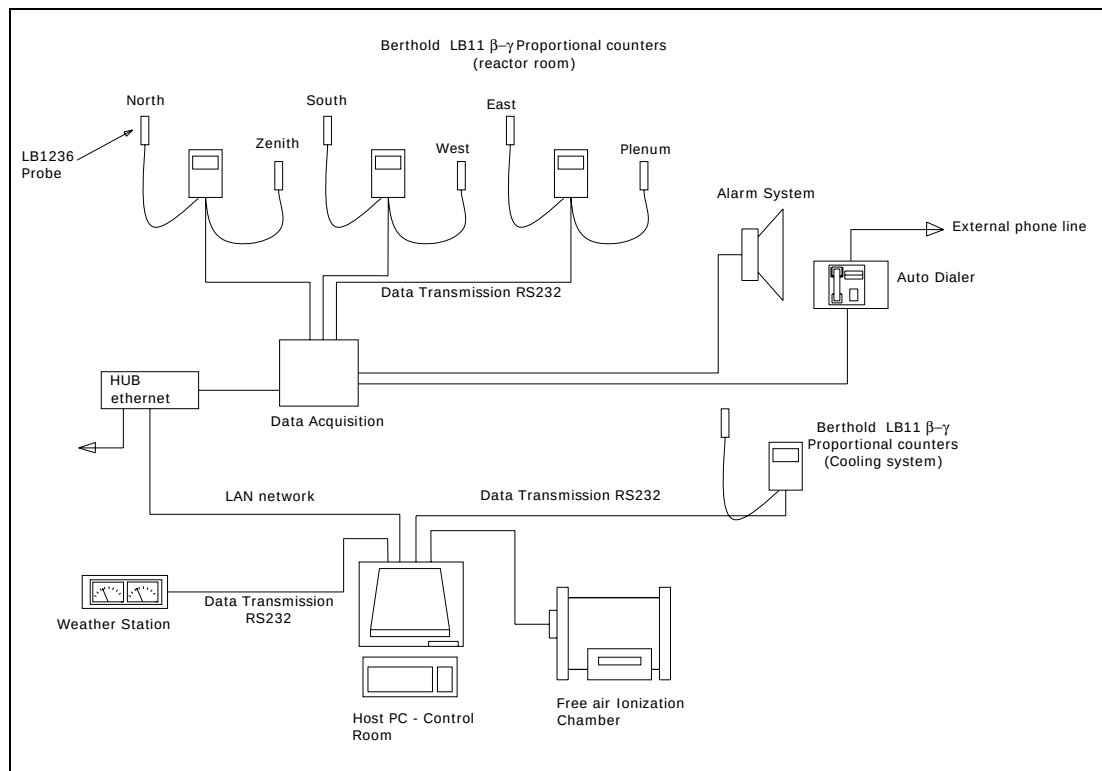


Fig. 1 – System lay-out

3. Instruments description

Berthold LB111 Micro-Gamma proportional counter

The LB111 Micro Gamma is a versatile instrument to measure local doses and doserates in one or two channels. It can be software-programmed for manifold applications, and contains all facilities for data storage and data transmission. The instrument can be employed as a stand-alone system with local result and alarm indication. Several LB111 systems may be connected to one central data station inside one premise via a local network with leased line, e.g. RS232 or RS485 bus system.

National Instruments Compact Field Point (CFP) data acquisition system

National Instruments Compact Field Point® is a programmable automation controller composed of I/O modules and intelligent communication interface. Compact Field Point® network communication interfaces and automatically publishes measurements with an Ethernet network. Is it possible to access I/O points nearby or miles away on the network using the same simple read/write software framework. The instrument can acquire data from serial lines RS232, RS485 or from analog I/O banks. It can also operate a series of output relays trough an output module.

4. Data acquisition and analysis software

Data acquisition and storage is performed in two different ways:

- the six proportional counters that monitor the dose-rate inside the reactor room transmit the data to the acquisition system CFP, via a RS232 serial line. The CFP decodes and stores data on a local flash memory and then re-transmit them, via Ethernet, to the Host PC for the visualization and for the backup storage on a local hard disk. CFP can activate output relays for the alarm system intervention;

- all the other parameters monitored (i.e. weather data, primary reactor cooling circuit water activity, air particulate activity) are acquired, processed, displayed and stored directly by the Host PC.

The home-made software developed on LabVIEW platform operate contemporary on two different levels, sharing data but maintaining a physical independence: a part of the software is resident on the CFP and is dedicated to the acquisition of the dose-rate monitoring data inside the reactor room, while the other part of the software is resident on the Host PC and performs the acquisition of all the other parameters.

This kind of configuration was chosen in order to grant redundancy and better reliability for the acquisition and storage of dose-rate monitoring data inside the reactor room.

The program installed on the CFP contains an automatic start-up file and provides the following tasks:

- acquire the dose-rate monitoring data inside the reactor room
- decode the transfer patterns
- compare the dose-rate with pre-set alarm thresholds
- operate the output relays
- transmit data to the Host PC via TCP/IP protocol
- storage data every 60 s on an removable flash memory

The tasks chain, except for the storage on the flash memory, has an execution time of 2 seconds. In the case of software failure, the program automatically saves an error report with the position of the failure in the execution chain.

In order to get an accurate data log, data from CFP are saved also on the Host PC every 60 seconds.

The software running on the Host PC collects, saves, and displays all the data, those coming from the CFP and the others acquired directly. The program code has been split in several “while loop”, one for each operation required, with an execution time of 1 second. The PC displays data both as numeric values and graphs (i.e. bar graphs, gauges, history charts) as shown in Fig.2.



Fig. 2- Host PC Front panel

From the main menu is possible to access the history charts of all data stored as shown in Fig.4. Is it also possible to access the program remotely via TCP/IP, using any common web browser. For security reasons though, only authorized clients can access the program and the data files.

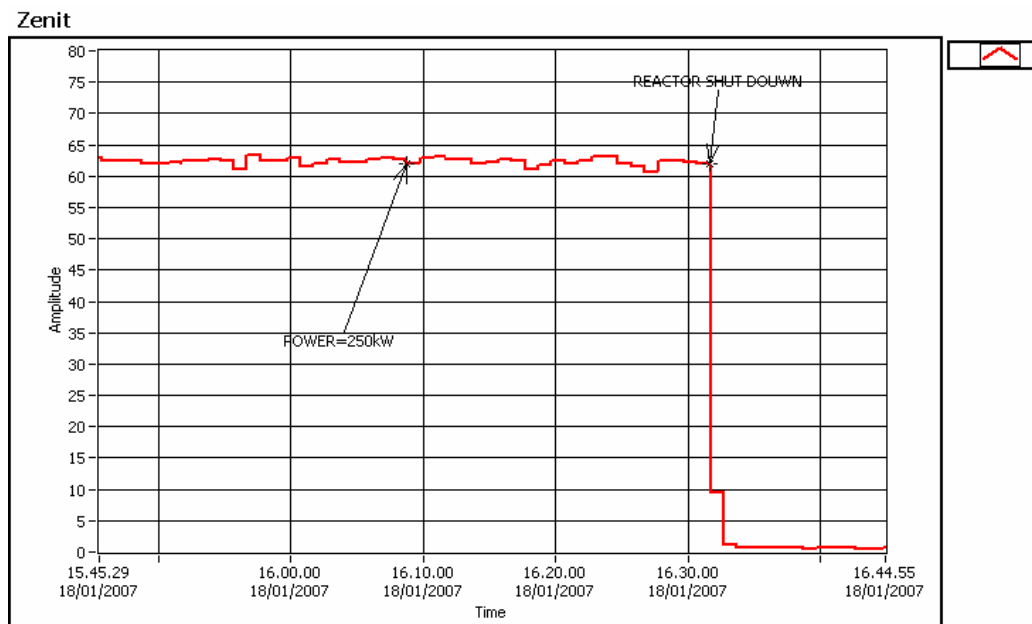


Fig.4 – Historical dose-rate data (Amplitude = $\mu\text{Sv/h}$)

The system recognizes and operate for two different pre-set levels of alarms:

- *Investigation Level Alarm*: as soon as the measured dose-rate in one of the counters reaches the first pre-set alarm threshold, both acoustical and light signals on the local counter (through a yellow signal lamp and a buzzer – see Fig.5) and on the front panel of the Host PC (buzzer and a blinking yellow led) are activated;
- *Intervention Level Alarm*: as soon as the measured dose-rate in one of the counters reaches the second pre-set alarm threshold, both acoustical and light signals on the local counter (through a red signal lamp and a buzzer– see Fig.5) and on the front panel of the Host PC (buzzer and a blinking red led) are activated. In this case CFP operates two output relays that activate the general alarm horn of the facility and the telephone dialler respectively. The dialler alert on the mobile phones the emergency personnel by means of a pre-registered message.

The alarm-activation channels do not depend on the PC. Radiation counters failures or program crashes are promptly notified to the personnel on call via the auto-dialler

CFP has also a watchdog which is a computer hardware timing device that triggers a system reset if the main program, due to some fault condition, such as a hang, neglects to regularly service the watchdog. The watchdog goal is to bring the system back from the hung state into normal operation. If an error occurs, the watchdog activates the Intervention Level Alarm relay.



Fig.5 – LB111 Proportional Counter

5. Data communication

Both CFP and Host PC access the instruments, via the bus line, to request data or to set parameters. Data transfer is always initiated and executed by the Host PC or by the CFP and the request flag is stored in the battery-buffered memory of the instruments with the FIFO content. Every instrument has its own COM port, in order to avoid possible failures of the whole system due to a problem on the communication line. The communication protocol is the RS232 standard, and, after several test, the following parameters have been set in order to get the best performance:

Transfer mode asynchronous	
Full duplex	
ASCII type	
Start bit	1
Stop bit	1
Parity	No
Baudrate	1200
Handshaking	RTS/CTS
Signal GND	

Tab. 1- Serial interface RS232 parameters setting

The wind direction and speed are displayed on a polar graph collecting data from the last 12 hours. This representation is very useful to follow diffusion of a radioactive cloud in the case of a nuclear emergency.

Monthly, a file with the report of the average temperatures and atmospheric pressures of every single day is sent by e-mails to the Health Physics Service of the reactor.

Whenever the CFP or Host PC memories usage reaches the 80% of its capacity, the program sends an alert e-mail to the maintenance manager of the system.

6. Conclusion

The new radiation monitoring system has been calibrated and tested for three months at the reactor facility of the University of Pavia showing to be very reliable and accurate. The new system will be implemented as substitute of the old one within a couple of months.

7. References

- National Instruments Compact Field Point (CFP) 2020 User Manual
- Berthold LB111 Micro-Gamma proportional counter Operating Manual
- LabVIEW Real-Time Module 8.0 Manuals

EVALUATION OF THE FEASIBILITY OF APPLYING CERMET FUEL PINS ON BASIS OF URANIUM ENRICHED UP TO 20% ^{235}U TO UPRATED POWER RESEARCH REACTORS

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ABSTRACT

The paper deals with the feasibility of employing fuel pins with a cermet fuel composition (UO_2 particulates in an Al-based alloy) clad with the zirconium-based Zr+1%Nb alloy to elevated power research reactors.

The fuel pin production process enables to place uranium dioxide particulates, up to 65 % by volume, into small diameter claddings and to attain in fuel composition the volume densities up to 6.4 g U/cm^3 of uranium. The impregnation of the charged UO_2 particulates with the liquid matrix material in the subsequent course of fuel pin fabrication enables to ensure high thermal conductivity of the cermet fuel composition and zero thermal resistance at the fuel-cladding interface

The computations (made, by way of example, with reference to the IR-8 research reactor) have shown that the replacement of tubular fuel elements with high-enriched uranium, by fuel pins whose cermet fuel is enriched not more than to 20 % is possible in principle, with the main thermal-hydraulics and neutronics characteristics of the reactor being conserved.

1. Introduction

Russian water-water research reactors (RR) are operated now with use of tubular cermet fuel elements with a cermet fuel (UO_2 in an aluminium-alloy matrix), fabricated by means of the extrusion technique. Alongside with the extended energy conduction surface as an advantage, these tubular fuel elements possess several disadvantages:

- diversity as to the shapes and sizes of fuel elements;
- spatial unevenness of energy release across the section of a fuel assembly (FA) and a high unevenness of longitudinal distribution of energy;
- the extrusion method of fuel element fabrication limits the volume content of UO_2 fuel particulates to 29 %, and the volume concentration of uranium to 2.5 gU/cm^3 [1]. This limitation requires high enrichment with uranium, coming up to 90 %.

The above mentioned disadvantages can be eliminated in case of the transfer from tubular elements to fuel pins. With that end in view it is essential to ensure the extended energy conduction surface; it can be made at the expense of lowering the fuel pin diameter.

2. Cermet fuel pins

The process of fabricating cermet fuel pins developed in Russia [2] is similar to that of fabricating fuel pins for the Bilibino NPP that had been mastered at the Elektrostal Engineering Plant, Russia.

In a few words, the process of fabricating such a fuel pin looks as follows: a cladding made of the zirconium-based (Zr+1%Nb) alloy is filled (up to 65% of its volume) with UO_2 particulates, after which the charge is impregnated with a liquid aluminium-based alloy as the matrix material (whose volume fraction is 35% or somewhat higher). This process ensures the metallurgical bondage between the fuel and its cladding. The measured values of thermal resistance at the fuel-cladding interface, R_k , are of an order of $10^{-6} \div 10^{-5} \text{ m}^2 \cdot \text{degree/W}$.

The fabrication process like this guarantees the metallurgical bondage between the fuel and its cladding. Moreover, the examination carried out by far has shown that the fuel of this kind possesses sufficiently high thermal conductivity, as can be seen in Table 1.

Table 1. The values of the thermal conductivity coefficients of cermet fuel (UO₂+aluminium-based alloy) as a function of temperature

Temperature, °C	200	300	400	500	Percent composition of cermet fuel
Thermal conductivity coefficient, λ , W/m·degree	37,4	35,9	34,3	32,8	60 vol% UO ₂ + aluminium-based alloy

This fabrication process has been mastered for cermet fuel pins 5 mm in diameter whose fuel composition can contain up to 6.4 g uranium per 1 cm³. It is expected that the enrichment of their fuel with ²³⁵U, X_5 , can be lowered to ≤ 20 %.

Nevertheless, the transfer from the present tubular fuel elements to the suggested fuel pins will bring about a certain reduction in the specific heat removal surface, and a certain increase in the loading of ²³⁵U, which will result in some changes of physical and thermo-hydraulic characteristics of a reactor. The question whether such changes are acceptable is being studied now, by way of example, with reference to the IR-8 research reactor (Kurchatov Institute, Moscow), where a rather considerable extension of the heat removal surface per volume unit of the reactor core, $F_{hr}/V_{core}=525$ m²/m³, has already been achieved. With reference to the IR-8 reactor we have carried out a comparative computational analysis of physical and thermo-hydraulic characteristics as applied to the transfer from tubular fuel elements to fuel pins. However, it is yet of a preliminary character as it has been carried out without taking into account both the control and protection system of specifically that reactor, and its structural distinctions.

3. Results of thermo-hydraulic and physical computations

Table 2 presents thermo-hydraulic characteristics of a standard FA of the IR-8 reactor, and a FA with fuel pins.

Table 2

Characteristics and units	Values	
	Standard, tubular	Pin type, Ø 5 mm
Type of fuel elements		
Number of fuel elements in a FA, pcs.	8	121
Spacing of FAs (in a square lattice), mm	71.5	71.5
Spacing of fuel elements in a square lattice, mm	–	6.5
Surface of heat removal in a FA, m ²	1.45	1.1
Open flow area in a FA, m ²	0.0030	0.00274
Maximum thermal load, kW/m ²	740	976
Temperature drop from a fuel element to water, °C	22	24.4
Maximum temperature of a fuel element surface, °C	72	74.4
Thermal load ruling out the possibility of surface boiling of water, not more than, kW/m ²	1746	2072
Marginal load capacity up to surface boiling	2.36	2.12

It follows from the data of Table 2 that in the view of thermo-hydraulic characteristics the version of a FA with 121 fuel pins for utilization in the IR-8 reactor in exchange for its standard FAs seems to be accessible.

The comparative physical computations have been carried out both for a standard FA of the IR-8 reactor with eight tubular fuel elements and for a FA with 121 fuel pins. The fuel pins were considered as loaded with a cermet fuel (30 %UO₂+70 vol%Al) on basis of uranium enriched with $X_5 \approx 21$ wt% of ²³⁵U. The results of these computations are presented in Figure 1 and in Table 3.

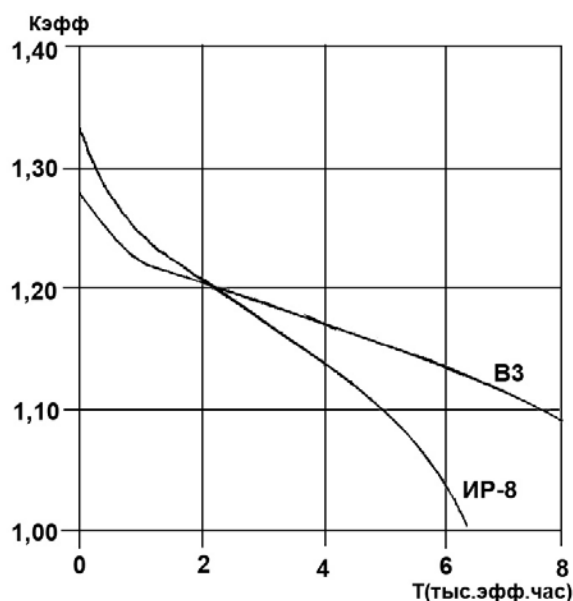


Fig. 1. The temperature dependence of K_{eff} in cases of tubular and pin fuel elements as a function of duration of life (in thousands of operational hours)

Table 3. Results of reactor physics computations relating to the versions of the reactor core considered

Variant	Number of FA, pcs	Number of fuel elements per FA, pcs	X_5 , %; composition of fuel	$K_{\text{eff}}(0)$	T , eff days	$\Phi_{\text{core}}^{\text{max}} (10^{14} \text{ n/cm}^2 \cdot \text{s})$	
						thermal, $E_n < 1 \text{ eV}$	fast, $E_n > 0.1 \text{ MeV}$
Pin type fuel elements, $\varnothing 5 \text{ mm (B3)}$	16	121	21 30% $\text{UO}_2 + 70\% \text{ Al}$	1,2766	370	0,93	3,66
Standard FA of the IR-8 reactor	16	8	90 U-Al	1,3315	280	1,7	3,31

As can be seen in the figure and in the table, the values of $K_{\text{eff}}(0)$ for fuel pins in the beginning of the life cycle are somewhat lower than for the standard fuel elements of the IR-8 reactor. But owing to a greater load of ^{235}U and a contribution from plutonium into the fission of nuclei the slope of curves becomes more gentle, and the core life time turns out to be longer.

If the standard life time of the IR-8 reactor is regarded as acceptable also for cermet fuel pins, and the volume fraction of UO_2 particulates in cermet fuel is increased up to 60÷65%, it will become possible to reduce the degree of enrichment with ^{235}U even to values of $\leq 15\%$.

4. On the efficiency of pin-type fuel elements

Computations of the stressed-strained state and efficiency of a cermet fuel pin under the operating conditions of the IR-8 reactor have been also carried out. The following admissions have been made for these computations:

Cladding material	(Zr+1%Nb) alloy
Enrichment with ^{235}U in UO_2	20% of ^{235}U
Volume fraction of UO_2 in the fuel composition	60% $\text{UO}_2 + 40\% \text{ Al}$
Burnup of uranium at the end of life (EOL), %	80% ^{235}U , or 16% h.a.
Swelling rate of UO_2 , $\Delta V/V$, up to $B \approx 3\%$ h.a.	0
at $B \geq 3\%$ h.a.	0,59 % per 1 % h.a.
Core life time T , eff. hours	$6 \cdot 10^3$

These computations have shown that the swelling particulates of UO_2 cause the strain of the fuel pin cladding at a rate of $\dot{\epsilon}_\theta = 1,95 \cdot 10^{-6}$ 1/h. Such a rate of loading of the cladding is compensated by its irradiation creep, see [4], at a stress of $\sigma_\theta \approx 200$ MPa, which is considerably less than the yield stress of the (Zr+1%Nb) alloy ($\sigma_T \geq 450$ MPa at $\Phi \approx 8 \cdot 10^{21}$ n/cm², $E > 0.1$ MeV). At the end of life (EOL) the cladding will increase its diameter by $\Delta d \approx 0.04$ mm, which will have practically no influence on the thermo-hydraulic characteristics of the reactor core.

5. Conclusion

As has been shown, it is possible in principle to replace tubular fuel elements with HEU in Russian research reactors by fuel pins with cermet fuel whose uranium is enriched less than to 20%. At that the thermo-hydraulic and neutronics characteristics of reactors, as well as the performance of its fuel elements, will remain acceptable.

The use of cermet fuel pins in research reactors will enable:

- to replace a broad spectrum of tubular fuel elements, as to their shapes and dimensions, by the unified fuel pin of the same diameter;
- to perform a transfer from the fuel elements on basis of HEU to that on basis of LEU (with less than 20% ²³⁵U);
- to use the fuel elements with and without gadolinia as a burnable poison in their fuel composition to compensate the most part of their initial reactivity excess, which will enable to simplify the control and protection system, to improve the economy and the safety of the reactor.

6. References

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SOURCE OF RADIONUCLIDES IN PRIMARY CIRCUIT WATER OF LVR-15 REACTOR

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ABSTRACT

The early detection of a damaged fuel assembly in reactor core is one of the most important aspects for safety operation of the reactor and radiation protection of the research reactor staff. The indication of damaged fuel assembly can be based on evaluation of the primary circuit water activity. Radionuclides in the water are produced by activation of stable nuclides and by fission of fissile nuclides, mainly ^{235}U . From comparison of theoretical results made by ORIGEN 2.1 code and measured values of volume activities of fission products and ^{239}Np (activation product of ^{238}U) the enrichment of the irradiated uranium has been estimated in LVR-15 research reactor. Finally, on the basis of the known enrichment it can be concluded if the source of fission products is mainly natural uranium (e.g. from demineralized water) or uranium from fuel assemblies (contamination of fuel cladding or damage of fuel assembly).

1. Introduction

The LVR-15 reactor is a light water research type reactor, which is situated in Nuclear Research Institute, Rez near Prague. At present the IRT-2M fuel of Russian production with enrichment of 36 % is used. In the reactor core there are usually from 28 to 32 fuel assemblies with the total mass about 5 kg of ^{235}U . Reactor is cooled by demineralized water. The maximum thermal power is 10 MW and the reactor is operated in 21-days irradiation cycles, with 8 to 10 cycles per year.

The indication of damaged fuel assembly can be made e.g. from gas effluents. This paper deals with the method based on the primary circuit water activity measurement. The measurement using gamma spectrometry method has been performed regularly (weekly) since 1996, measurement of actinides using alpha spectrometry method in evaporated samples of primary circuit water since 2005.

2. Methods of measurement

The samples of primary circuit water are taken every week (on Thursday) and the spectrometric measurement is made 4 days later (on Monday).

Originally the Marinelli beakers were used for the gamma activity measurement. Since 1998 the same 0,5 l PET bottle has been used both for taking of the sample and measurement (no trapping of radioisotopes on the walls of the bottle used for taking of the sample and on the walls of the Marinelli beaker, minimum handling with the radioactive water).

A gamma spectrometric assembly (Canberra) with an HPGe detector with relative efficiency of 18 % and FWHM=1.8 keV for energy of 1332 keV is used for gamma activity measurements (Fig. 1). The detector is placed in a shielding box with 5 cm thick lead walls. For calibration special radionuclide standards of 0,5 l bottle has been used. The minimum detectable volume activity is 10 Bq/l for ^{137}Cs and measuring time of 3600 s, which is sufficient for the measurements. The analysis is made for the library of about 100 radionuclides but only 26 radionuclides is used for standard evaluation.

For alpha measurement the water sample is evaporated on cup made from Al foil. The same water sample previously used for gamma measurement is used for evaporation (Fig. 2). An alpha spectrometric assembly (Canberra) with a PIPS (Passivated Implanted Planar Silicon) detector with FWHM = 20 keV for energy of 5486 keV is used for alpha activity measurements. Each sample is measured in vacuum chamber for 3 days.



Fig. 1. Gamma activity measurement



Fig. 2. Evaporation of water sample

3. Time dependence of fission product activities

In Fig. 3 the time dependence of ^{131}I and ^{137}Cs volume activity is given. Increased values in 1996 are due to a damaged fuel assembly in the reactor core, low values in 1996 due to reactor shutdown and cleaning during a maintenance period. Missing data in 2002 are related to flood near the reactor building.

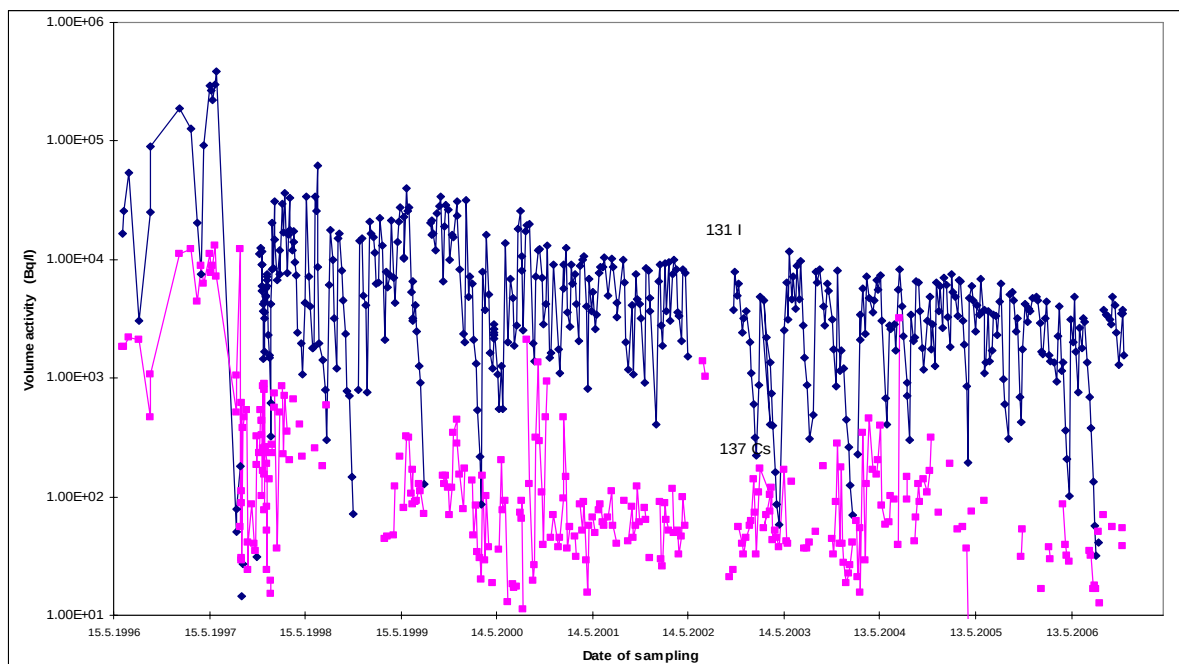


Fig. 3. Time dependence of volume activity of ^{131}I and ^{137}Cs in primary circuit water

4. Analysis of radionuclide sources

In Table 1 the example of volume activities and volume mass of source element values primary circuit water sample is given. The sample was taken on 28. 4. 2005 during the reactor cycle started at 13. 4. 2005.

Table 1. The example of volume activities and volume mass of source element values

Radionuclide	Half life	Reaction	Volume activity	Volume mass of source element (^{235}U or ^{238}U)
	(days)		(Bq/l)	($\mu\text{g/l}$)
^{95}Zr	64.03	$^{235}\text{U}(\text{n},\text{f})^{95}\text{Zr}$	1.99E+03	0.346
^{95}Nb	35.15	$^{235}\text{U}(\text{n},\text{f})^{95}\text{Nb}$	3.17E+03	0.761
^{99}Mo	2.78	$^{235}\text{U}(\text{n},\text{f})^{99}\text{Mo}$	5.57E+03	0.468
^{103}Ru	39.5	$^{235}\text{U}(\text{n},\text{f})^{103}\text{Ru}$	6.19E+02	0.183
^{131}I	8.05	$^{235}\text{U}(\text{n},\text{f})^{131}\text{I}$	5.85E+03	1.339
^{132}Te	3.24	$^{235}\text{U}(\text{n},\text{f})^{132}\text{Te}$	7.31E+02	0.088
^{133}I	0.867	$^{235}\text{U}(\text{n},\text{f})^{133}\text{I}$	2.62E+04	1.92
^{133}Xe	5.30	$^{235}\text{U}(\text{n},\text{f})^{133}\text{Xe}$	5.63E+03	0.492
^{137}Cs	11000	$^{235}\text{U}(\text{n},\text{f})^{137}\text{Cs}$	7.45E+01	1.268
^{140}Ba	12.75	$^{235}\text{U}(\text{n},\text{f})^{140}\text{Ba}$	9.56E+03	1.162
^{140}La	1.678	$^{235}\text{U}(\text{n},\text{f})^{140}\text{La}$	1.57E+04	2.07
^{141}Ce	32.5	$^{235}\text{U}(\text{n},\text{f})^{141}\text{Ce}$	1.78E+04	2.706
^{143}Ce	1.375	$^{235}\text{U}(\text{n},\text{f})^{143}\text{Ce}$	1.63E+04	1.358
^{144}Ce	284.3	$^{235}\text{U}(\text{n},\text{f})^{144}\text{Ce}$	1.04E+03	0.601
^{24}Na	0.623	$^{23}\text{Na}(\text{n},\gamma)^{24}\text{Na}$	4.75E+06	
^{41}Ar	0.076	$^{40}\text{Ar}(\text{n},\gamma)^{41}\text{Ar}$	4.33E+05	
^{46}Sc	83.9	$^{45}\text{Sc}(\text{n},\gamma)^{46}\text{Sc}$	2.85E+04	
^{51}Cr	27.7	$^{50}\text{Cr}(\text{n},\gamma)^{51}\text{Cr}$	8.56E+05	
^{54}Mn	312.3	$^{54}\text{Fe}(\text{n},\text{p})^{54}\text{Mn}$	4.54E+03	
^{56}Mn	0.108	$^{55}\text{Mn}(\text{n},\gamma)^{56}\text{Mn}$	1.26E+05	
^{58}Co	70.86	$^{58}\text{Ni}(\text{n},\text{p})^{58}\text{Co}$	2.63E+03	
^{59}Fe	44.50	$^{58}\text{Fe}(\text{n},\gamma)^{59}\text{Fe}$	2.96E+04	
^{60}Co	1921	$^{59}\text{Co}(\text{n},\gamma)^{60}\text{Co}$	3.39E+04	
^{65}Zn	243.9	$^{64}\text{Zn}(\text{n},\gamma)^{65}\text{Zn}$	1.42E+04	
$^{110\text{m}}\text{Ag}$	255	$^{109}\text{Ag}(\text{n},\gamma)^{110\text{m}}\text{Ag}$	3.44E+02	
^{113}Sn	115.1	$^{112}\text{Sn}(\text{n},\gamma)^{113}\text{Sn}$	1.14E+03	
^{122}Sb	2.8	$^{121}\text{Sb}(\text{n},\gamma)^{122}\text{Sb}$	3.73E+04	
^{124}Sb	60.2	$^{123}\text{Sb}(\text{n},\gamma)^{124}\text{Sb}$	2.11E+04	
^{125}Sb	1007	$^{124}\text{Sn}(\text{n},\gamma)^{125}\text{Sn}(\beta^-)^{125}\text{Sb}$	1.87E+02	
^{152}Eu	4944	$^{151}\text{Eu}(\text{n},\gamma)^{152}\text{Eu}$	4.44E+02	
^{181}Hf	42.39	$^{180}\text{Hf}(\text{n},\gamma)^{181}\text{Hf}$	9.50E+02	
^{187}W	0.988	$^{186}\text{W}(\text{n},\gamma)^{187}\text{W}$	7.70E+04	
^{239}Np	2.36	$^{238}\text{U}(\text{n},\gamma)^{239}\text{U}(\beta^-)^{239}\text{Np}$	8.82E+03	1.703

The theoretical values of volume mass of source nuclides ^{235}U and ^{238}U were calculated from individual radionuclide activities by ORIGEN 2.1 code. The calculation was made for the following conditions: the effective fluence rate is $2.1 \times 10^{16} \text{ m}^{-2} \text{ s}^{-1}$ and all fission products and ^{239}Np are released into the water.

The mean values of volume mass of ^{235}U and ^{238}U under these conditions calculated for 3 samples are 1.09 $\mu\text{g/l}$ and 1.70 $\mu\text{g/l}$ respectively, which correspond to enrichment of 39 %.

To estimate the total mass of ^{235}U in primary circuit water with volume of 40 m^3 , two cases can be considered:

- All this ^{235}U is homogeneously dissolved in primary circuit water (effective fluence rate of $2.1 \times 10^{16}\text{ m}^{-2}\text{s}^{-1}$). For this case the total mass would be 44 mg.
- All this ^{235}U is situated in the reactor core (effective fluence rate of $1.7 \times 10^{18}\text{ m}^{-2}\text{s}^{-1}$) and all fission products are dissolved in primary circuit water. For this case the total mass would be 0.54 mg.

5. Alpha activity measurement

Alpha measurement is performed for shorter time, since 2005. Example of the results is in the Table 2.

Table 2. The example of alpha radionuclide activities in the sample taken on 5. 11. 2005

Radionuclide	Volume Activity (Bq/l)	Volume activity error (%)
^{238}Pu	0.034	10.95
^{239}Pu	0.072	5.41
^{240}Pu	0.074	5.3
^{241}Am	0.05	38.32
^{242}Cm	0.026	18.99
^{244}Cm	0.012	13.83

6. Conclusion

During 10 years of measurement, the activities of fission products in primary circuit water have been decreasing, in average. From comparison of theoretical results and measured values of volume activities of fission products and ^{239}Np the enrichment of the irradiated uranium have been estimated. The resulted 39 % enrichment has high uncertainty due to complicated chemical and physical processes inside of the reactor. Despite of this uncertainty (estimated of factor 2 to 5) the enrichment is much closer to value of fuel enrichment (36 %) than for natural Uranium (0.72 %). The origin of the uranium can be contamination of fuel cladding with uranium or residuum from the damage of fuel assembly in 1996. The presence of alpha radionuclides is probably also connected with this event.

Acknowledgments

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DEVELOPMENT OF INTEGRATED MANAGEMENT SYSTEM FOR THE RESEARCH REACTOR IN SOFIA

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ABSTRACT

The main purpose in establishment of Integrated Management System (IMS) is to guarantee safety operation of the nuclear facilities as well as to increase their exploitation effectiveness. To ensure the safety operation of the nuclear facilities the Bulgarian Nuclear Regulatory Agency has created requirements and norms to prevent potential nuclear incidents, overdose irradiation or terrorist attacks opportunities. The IMS of the Institute for Nuclear Research and Nuclear Energy has been developed in a way to create an environment which to guarantee the ways and means for: quality management according with ISO 9001:2000, environmental management according with ISO 14001:2004, management of safety requirements of the Bulgarian Nuclear Regulatory Agency, security and physical protection, management of the safe and health working conditions for the employees. The IMS is based on the concepts recommended by the IAEA: the entirety of work can be structured and interpreted as a set of interacting processes that can be planned, performed, measured, assessed and improved, and, those performing assessing work, all contribute in achieving quality and ensuring safety. The IMS has been developed in the way to be continuously refreshable and additive. The IMS will be added with new instructions, procedures and others, which will correspond to new activities arising during the reactor reconstruction. The IMS was developed on the base of state-of-the-art software ARIS in the way to achieve ease in communication, visualization, possibilities for assessment and continues improvement of the IMS.

I. Introduction

The Institute for Nuclear Research and Nuclear Energy (INRNE), Bulgarian Academy of Science, with its Research Reactor IRT is the biggest complex in Republic of Bulgaria for conducting research in the field of nuclear science, nuclear technology and energy and in the field of the monitoring of the radioactive influence on the environment.

The INRNE is the host and an operator of this institutes research reactor complex which is situated in Sofia city, and it is responsible for reactor systems maintenance and controls the permanently shutdown Research Reactor. INRNE is responsible also for the reconstruction activities, which include:

- Planning and preparation for partial dismantling of the Research Reactor equipment;
- Supply of equipment for the IRT Reconstruction;
- Planning of activities and responsibility for reactor modernization;
- Spent fuel control and management;
- IRT radiation monitoring for all implemented activities;
- Radioactive waste (RAW) management and control, for the RAW generated during the reconstruction process.

An Integrated Management System has been elaborated on ISO 9001:2000 requirements for quality management [1], ISO 14001:2004 for environmental management [2], and safety requirements of the Bulgarian Nuclear Regulatory Agency [3], governmental requirements for occupational health and safety and security. The IMS application guarantees the safety of activities as well as reduces of the radiation influence on the environment within the governmental norms. It helps to achieve maximal effectiveness and quality of the performed activities.

II. Processes

The processes, going along with various activities performed on the institute's Research Reactor are described in so called procedures. These procedures are developed in the way to give you an opportunity for simple process control following the Deming's cycle through plan, do, check, action stages. [4] For each activity there is a responsible person, delegated with peculiar obligations and competence.

The job instructions for each activity have been developed and different formularies to inscribe the results are applied. The regulatory body can control safety requirements performance according to these records, and there are documents in case of audit, or to give evidence to civil interest.

Here are shown some of most important and specific processes and procedures for our reactor as: "The IRT nuclear and irradiation safety insurance"; "IRT Research Reactor reconstruction management"; "The radiation monitoring insurance on the Nuclear Scientific and Experimental Centre (NSEC) site"; "Radioactive waste management"; "Preparation of documentation for licence and permissions".

For the process "The IRT nuclear and irradiation safety insurance" corresponded to procedure with instructions attached to, for example "Instruction for distillate water full up in reactor pool and water pool spent nuclear fuel (SNF) storage", "Instruction for the water technology control in water pool spent nuclear storage", "Instruction for activity on duty for the mechanic when IRT special sewage is used" etc. The performance of these activities is documented in the records, which proved the IRT safety assurance.

Other basic process is "The Research Reactor reconstruction management". The realisation of that process is graphically shown on fig. 1 and it is developed in following basic procedures:

- "Management of "Investigations, analysis and design of the Research Reactor with 200 kW power"";
- "The reactor equipment partial dismantling management";
- "The IRT reconstruction work project implementation management";

All processes, that are carried out on the NSEC site are accompanied with permanent radiation control and monitoring. These activities are described in the procedures "Securing the radiation monitoring on NSEC site" and "Providing safety radiation conditions for the personal, working with radioactivity sources, on the NSEC site". The monitoring programs and instructions strict implementation is documented by the records in appropriate formularies.

The „Radioactive waste management" process includes: RAW generation, collecting, sorting, minimization, and storage up to final transportation from the NSEC site. These activities control is realized according to Bulgarian legislation requirements, which are reflected in the procedure "RAW management". The responsible for RAW person makes the records, which give an account about the RAW quantity and the movement.

To make easier the realization of the system process as well as to evaluate it and to apply continuing improvement principle on IMS (in agreement with ISO standards) the ARIS software has been used. [5]

III. Results

From general point of view the good practice is the approach where we plan and conduct training to achieve qualification level. To perform this we can conduct training courses not only concerning nuclear technology and energy knowledge but conducting seminars on IMS also. The necessity, to conduct education courses, is documented in the form – "Training Application". After the end of the education course, the trained person has to write down report.

At daily work we ran into difficulties, connected with establishment and documentation of the suggestions for improvements in separate processes. That's why periodically we are refreshing our courses by the ISO standards requirements.

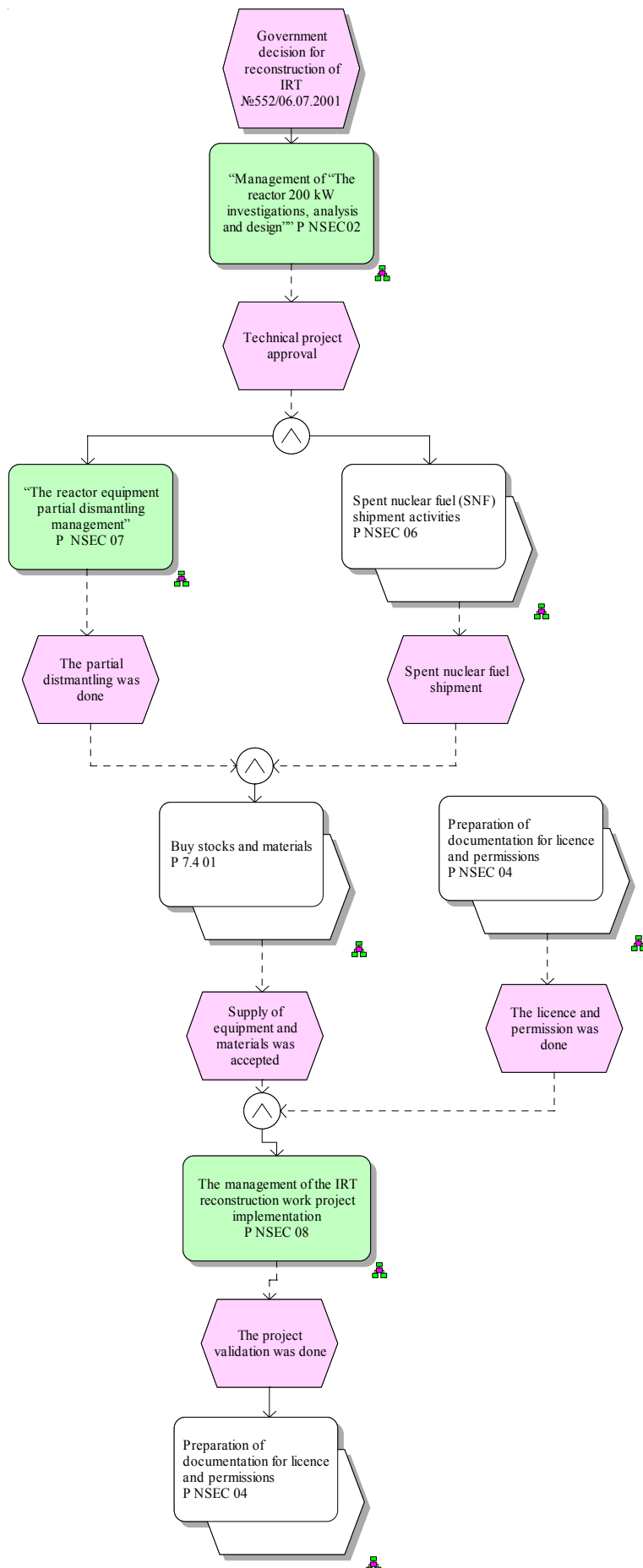


Fig.1. The process chart “The IRT Research Reactor reconstruction management”

The INRNE management policy is directed to guarantee high quality developments implementation, which are in agreement with modern world trends of continuously refreshing knowledge, of long standing experience and cooperation with leading European and International institutions. INRNE have a purpose to satisfy the community needs for development and maintenance of the nuclear science, to create necessary knowledge and skills for development of applied methods and research in the area of nuclear technology medical physics and nuclear industry.

IV. Conclusion

The IMS is based on the concepts recommended by ISO standards and the IAEA prescriptions: the entirety of work can be structured and interpreted as a set of interacting processes that can be planned, performed, measured, assessed and improved, and, those performing assessing work, all contribute in achieving quality and ensuring safety. The IMS has been developed in the way to be continuously refreshable and additive. That's why IMS will be added with new instructions, procedures and others, which will correspond to new activities arising during the reactor reconstruction.

The IMS gives strong level of certainty in the Research Reactor safety assurance, environmental protection, reactor physical protection and secure the normal personal working conditions.

V. References

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HOMOGENEOUS SOLUTION REACTOR ANALYSIS FOR ^{99}Mo PRODUCTION

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ABSTRACT

The neutronic behavior of a homogeneous solution reactor for ^{99}Mo was analyzed. The aim of the research is to study the neutronic characteristics in order to assess important parameters for reactor design and to set the main calculation options. All calculations show very safe neutronic behaviors with large negative reactivity coefficients and good ^{99}Mo production.

1. Introduction

The most widely used radioisotope in medicine is $^{99\text{m}}\text{Tc}$, used on 80% of nuclear medicine procedures in the world. It is used because of its ideal features for diagnosis. $^{99\text{m}}\text{Tc}$ is the ^{99}Mo daughter which can be obtained irradiating ^{98}Mo enriched targets or, as a fission product, irradiating uranium targets or by homogeneous solution reactors [1].

In this work, an uranyl nitrate homogeneous solution reactor for ^{99}Mo production was analyzed, establishing a neutronic processing chain for calculations related to this kind of reactor. The uranyl nitrate solution fuel (20% enriched in ^{235}U) is contained in a cylindrical stainless steel tank, surrounded with light water as neutronic reflector [2]. The reactor has 6 cadmium rods as control system.

Argentina is one of the six ^{99}Mo producer countries and the first using low enrichment targets. ^{99}Mo is produced in RA-3 reactor by uranium target irradiation and the radiochemical processing is performed at Fission ^{99}Mo Production Plant.

The only purpose of this homogeneous solution reactor is radioisotope production at lower costs. As compared to other radioisotope production options, it gives several advantages like a small burnup to produce large quantities of ^{99}Mo operating at low power.

2. Reactor model description

The analyzed reactor model has liquid fuel consisting of uranyl nitrate in a water solution with extra nitric acid. The solution is contained in a cylindrical stainless steel tank reflected by light water [3].

The reactor is refrigerated by a helicoidal heat exchanger immersed into the solution. The heat is removed from the fuel by natural convection, while water circulation inside the exchanger is forced. The secondary loop coolant is light water.

The reactor control system consists of 6 independent cadmium control rods with light water followers. An auxiliary control system is also possible by changing the solution level, but it was not included in the present modeling.

A schematic drawing of reactor core is shown in Figure 1 and main reactor characteristics in Table 1. It can be seen that the tank is not completely filled with fuel solution, and therefore the upper plenum must be filled with a inert gas (nitrogen for example) to avoid hydrogen accumulation due to water radiolysis and hence chemical explosion risk.

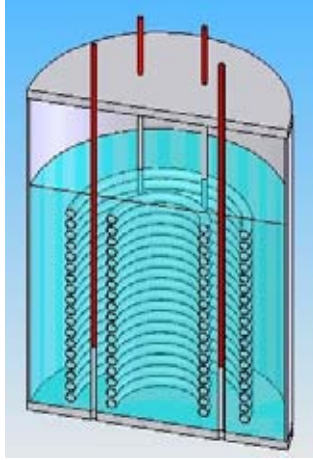


Figure 1. Homogeneous solution reactor nucleus simple model used in neutronic calculations.

Fuel	$(\text{NO}_3)_2\text{-UO}_2$
Volume	90 l
U concentration	200 g/l
U enrichment	20 %
Nitric Acid	1mol/l
Power	50 kW
Control rods	6 Cd

Table 1. Main homogeneous solution reactor features.

The ^{99}Mo production during one-day full power operation with a fresh nucleus is over 500 Ci. The ^{99}Mo activity during full power operation is shown in Figure 2 [4].

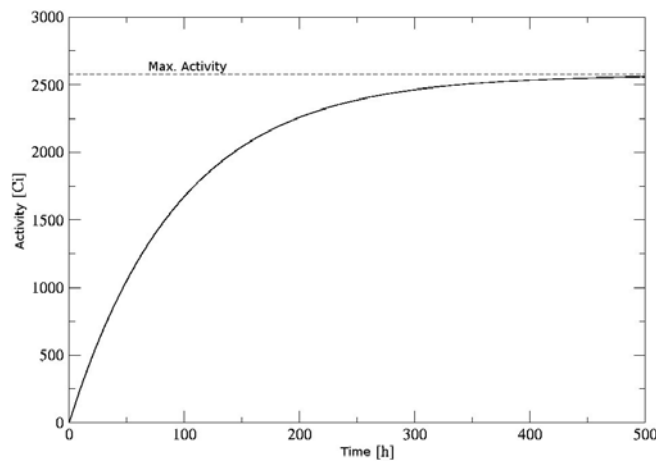


Figure 2. Full power ^{99}Mo production versus time.

3. Reactivity effects in normal operation

The system reactivity change due to several phenomena was studied. One important factor during normal operation is reactor temperature variation, leading to volumetric expansion and spectral effects [4]. Reactivity change due to fuel volumetric expansion is shown in Figure 3. This change could be as large as 600 pcm. In this case, calculation was performed with MCNP code. To study spectral effects in fuel cross sections, MCNP, SCALE/TORT and WIMS codes were used. The results are in Table 2, showing an average change of 1000 pcm for 100°C fuel temperature variation.

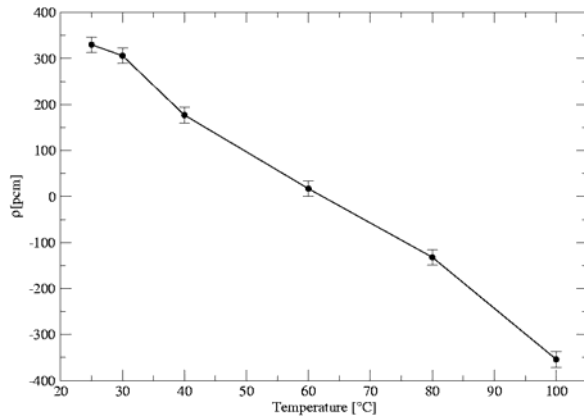


Figure 3. Solution dilatation effect over system reactivity due to increase in fuel temperature

Code	$\Delta\rho$ [pcm]
TORT	-800
MCNP	-1000
WIMS	-1200

Table 2. Reactivity change due to spectral effects in fuel cross sections for 100°C temperature increase.

Another phenomenon inserting changes in system reactivity is uranium concentration variation in fuel (due to burnup and/or chemical processing, for example). Reactivity versus uranium concentration in fuel solution, for a nearly critical system, was calculated with MCNP [5] and results are shown in Figure 4. The slope is about 150 pcm per uranium gram per solution liter. Moreover, the fuel burnup after one year full power operation was calculated with WIMS and analytically (Figure 5), showing a 220 mg/l change in uranium concentration. This leads to a small change in reactivity (about 30 pcm) comparing to temperature effects.

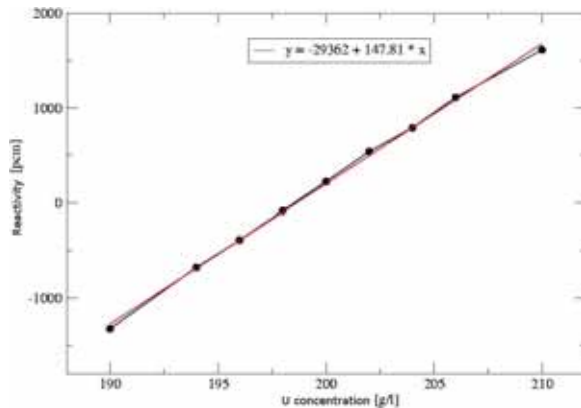


Figure 4. Reactivity versus uranium concentration variation in liquid fuel solution.

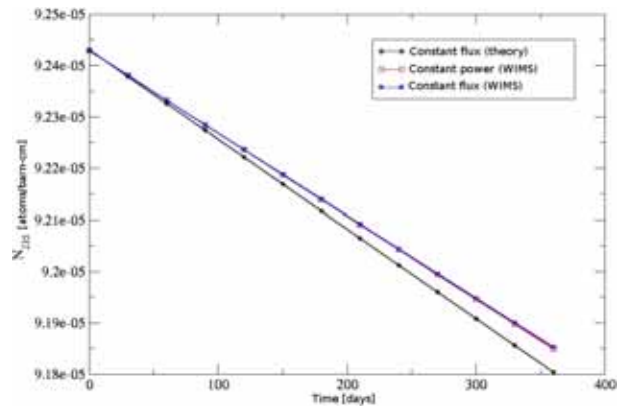


Figure 5. ²³⁵U burnup in one year full power operation.

Another situation in normal operation could be fuel solution height variation reaching the critical volume. Other possible case is exceeding the critical height leading to a supercritical system. For these reasons the reactivity variation with solution height was studied. The results are in Figure 6, where the slope is around 330 pcm per cm increased near criticality.

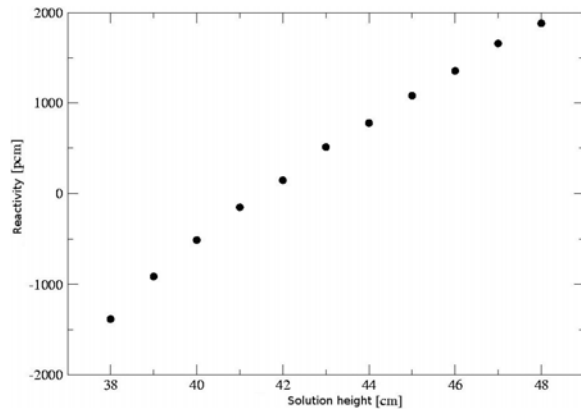


Figure 6. Reactivity variation by solution height.

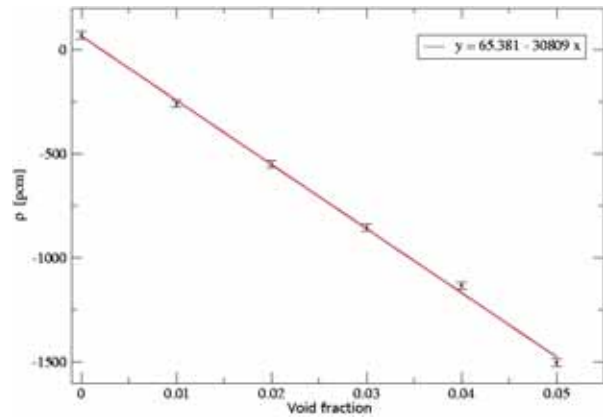


Figure 7. Reactivity function of void fraction in liquid fuel solution.

An important phenomenon in this type of reactors is a considerable void fraction in liquid fuel solution. Boiling water due to temperature and water radiolysis are the main contributors in this case. In Figure 7 can be seen that negative reactivity effect could be important depending on the void fraction value.

4. Reactivity effects in modelling and abnormal operation

As first approximation, the helicoidal heat exchanger was not modeled. It was homogenized with the uranyl nitrate solution as a model simplification. After that, four different heat exchanger models were studied (Figure 8) keeping some specifications of previous thermohydraulic calculations [2], like total tube length and tube diameter.

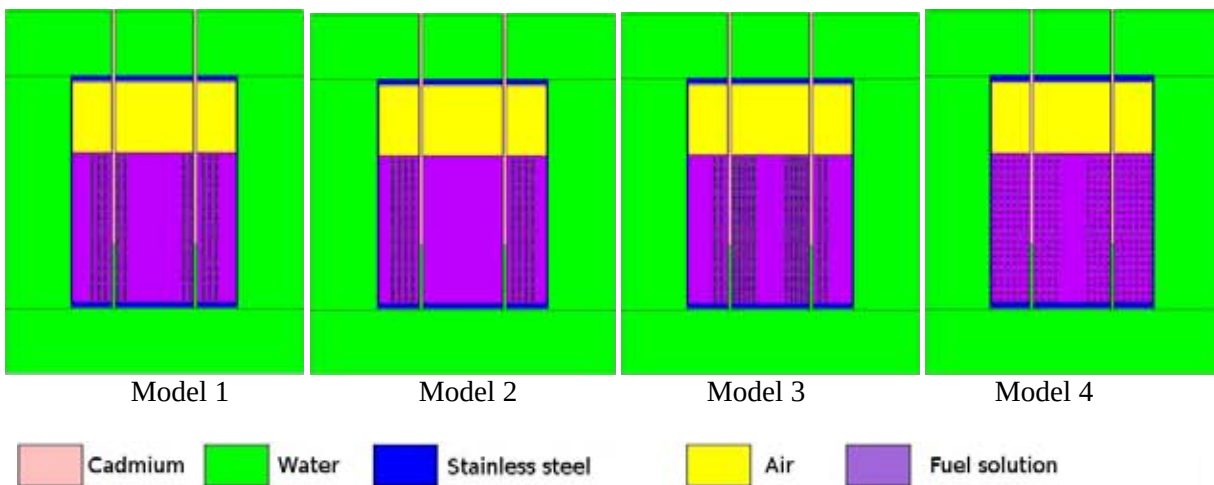


Figure 8. Different helicoidal heat exchanger models used in calculations.

Possible cases of abnormal operation were studied. In the case of loss of coolant in the secondary loop, several heat exchanger models were calculated. In all cases the results shown in Table 3 point out a large negative effect in reactivity system, keeping the reactor behavior from the safety side for this case.

Model	$\Delta\rho$ [pcm]
1	-1000
2	2500
3	-4800
4	1470
Homogeneous	70

Table 3. Reactivity variation due to changes in heat exchanger model.

Model	$\Delta\rho$ [pcm]
1	-1180
2	-850
3	-1000
4	-1240
Homogeneous	-1480

Table 4. Loss of coolant effect on reactivity for different heat exchanger models

As the fuel containment tank is surrounded by water used as reflector, any break could lead to a water insertion in fuel solution. This event produces a more diluted fuel solution and, as is shown in Figure 9, a negative reactivity insertion. On the other hand, a decrease in solution water (due to evaporation for example) leads to a positive reactivity insertion. In the last case, it is supposed that there is not uranium precipitation. Near criticality the slope is about -200 pcm per cm increased, taking into account that 1 cm equals about 2 fuel solution liters.

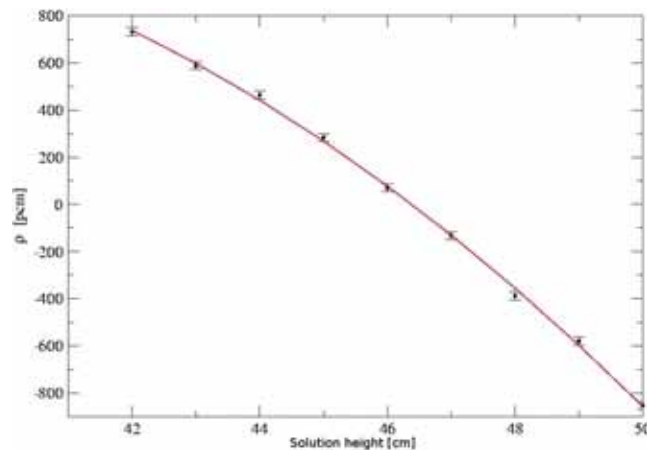


Figure 9. System reactivity versus fuel solution height change due to solution water mass variation.

5. Conclusions

Reactivity variations in normal and abnormal operation due to several phenomena were studied with MCNP, WIMS and TORT codes. A neutronic processing chain for calculations related to homogenous solution reactors was established integrating stochastic and deterministic codes.

All calculations show very safe neutronic behaviors with large negative reactivity coefficients. The fuel burnup and its reactivity effects are low with a considerable molybdenum production leading to a good production efficiency. As reactivity has a strong dependence on the heat exchanger geometry, a detailed thermohydraulic calculation is needed.

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UPGRADING OF JRR-3/JRR-4 NEUTRON BEAM UTILITIES - FOR COLD NEUTRON BEAM AND BNCT -

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ABSTRACT

To response to their requests, we proposed two plans. One is the enhancement of cold neutron beam intensity for Japan Research Reactor No.3 (JRR-3) and the other one is the progress of Boron Neutron Capture Therapy (BNCT) for JRR-4. We are expecting to achieve 10 times the present intensity in our maximum extent so that the good complementary relation with J-PARC in one site can be established with JRR-3. The various medical irradiation technologies in BNCT at JRR-4 should be established in order to promote the medical application of nuclear energy. Although BNCT in JRR-4 has been mainly applied to therapy against brain tumor so far, technical developments to expand its application to therapy against the other cancers such as lung cancer, head and neck cancer are carried out as the needs are increasing these days.

1. Introduction

The JRR-3 of the Japan Atomic Energy Agency (JAEA) was re-constructed in 1990 with a cold neutron source and five neutron guide tubes in order to meet with the neutron beam experiments those that increased rapidly [1]. JSNS (Japanese Spallation Neutron Source) of high intensity pulsed neutron source at the Material and Life Science Facility in the Japan Proton Accelerator Complex (J-PARC) will be constructed by 2008. The time-integrated neutron intensity at 10 m from the coupled moderator of JSNS is estimated to be 4.6×10^8 n/cm²s with a peak wavelength of 3 Å. On the other hand, it is 2.3×10^8 n/cm²s with a characteristic wavelength of 4 Å in a C2-1 beam port at 30 m from the cold moderator of JRR-3. That is to say, it is expected roughly that the steady cold neutron intensity of JRR-3 is half of that of JSNS with a power of 1MW so far. Of course, it is not easily compared with each other. JSNS is desired to exceed the steady beam intensity of the reactor source in the experiments that requires the time-integrated intensity such as spin echo experiment, small angle scattering, neutron radiography, prompt gamma ray analysis. Because of above Japanese neutron supply situation the target of JRR-3 upgrade plan was focused in 10 times of cold neutron intensity for continuous neutron requirement. In this plan, the cold neutron source and the neutron guide tubes should be upgraded without the increase of reactor power because of the budget.

The various medical irradiation technologies in BNCT at JRR-4 should be established in order to promote the medical application of nuclear energy. Although BNCT in JRR-4 has been mainly applied to therapy against brain tumor so far, technical developments to expand its application to therapy against the other cancers such as lung cancer, head and neck cancer are carried out as the needs are increasing these days. In this situation, a collimator and a setting-system to enable BNCT at a seated position were developed and used in practice.

2. Upgrading of JRR-3 Cold Neutron Beams

2.1 Design of Moderator and Neutron Guides

In cold neutron source (CNS) moderator vessel shapes, the spherical, cylindrical annulus or crescent shape with neutron trap such as NIST [2], ORPHEE [3] and CARR [4] are used recently. The JRR-3 liquid hydrogen moderator vessel for CNS is made of stainless steel and is flat shape which is the same type as old ORPHEE vessel [5]. The neutron intensity could be increased by optimizing the shape and material of the vessel. In a condition of our design of a new vessel, we should consider spatial limit and thermal limit to use an existing cooling system and an existing pipe, which the vessel

is inserted in. We also aim to increase the cold neutron beam intensity by replacing the current nickel mirror to Ni/Ti super-mirror in neutron guide tubes

The neutron gain of upgraded JRR-3 CNS was evaluated with the MCNP-X[6]. In addition, the McStats[7] code, which is the ray trace Monte-Carlo code, was used to design the neutron guide. Figure 1 shows the arrangement of cold neutron beam tube. In the MCNP-X simulation model, the core is defined as the homogeneous cylindrical fission source where the vertical fission distribution was determined by experimental results. The new vessel is designed as a boat-bottom shape and made of aluminum. The vessel outline dimensions are outer height 25 cm, outer width 118 cm and vessel wall thickness 2 mm, body radius 6.5 cm in outside / 3.1-4.1 cm inside, body annular arc with a central angle of 146 degrees, outer diameter 2.6 cm / inner 2.2cm in the neck. A 3-D imaging view of the vessel is shown in Figure 2. The horizontal space the body expands to 2.5cm from 2cm in consideration with decreasing by vapor void fraction, and an upper part and a bottom part have ellipsoidal inner face. The new vessel would be cooled by natural circulating saturated hydrogen at near atmospheric pressure. Therefore, the radiation heating of the vessel should be lower than cooling capacity of the existing system. The ortho-para hydrogen ratio with the reactor full power operation also was assumed 65%.

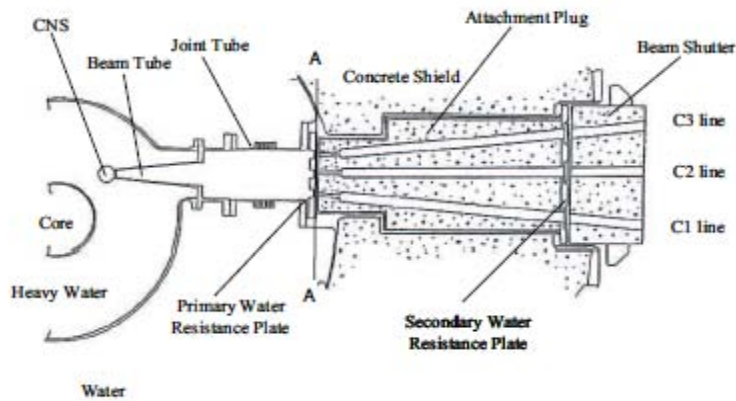


Fig.1 Arrangement of cold neutron beam tube at

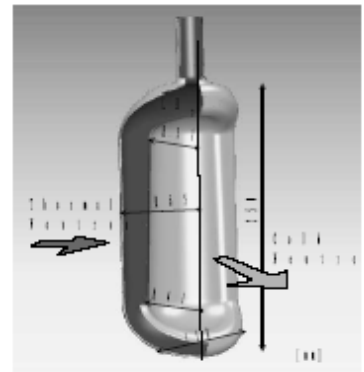


Fig.2 Three dimensions CAD imaging view of new vessel, which is called the boat-bottom shape vessel.

The neutron spectra for existing flat shape stainless steel vessel and boat-bottom shape aluminum vessel were used to connect the MCNP-X results to the McStats simulation. The joint is shown in an A-A line in Figure 1. The idealized mirrors on simulations in the neutron beam guides were arranged with actual positioning data measured during installation of current guides. The mirror optical properties are defined by the experimental neutron refraction data. The inside dimension of a guide tube element is 80cm in length, 12cm in height and 2cm width which is considered not to peep at the core side directly by using the curved guide. The neutron guides are placed after the beam shutter in each line. However, the optimization of neutron guides should be decided on the performance of connected experiment instruments, and should be achieved on the basis of ready-made technology.

2.2 Performance of Cold Neutron Beams

The neutron spectra at the joint of the current vessel and the new vessel are shown in Figure 3. The closed circles show the neutron flux of the new vessel with 0 % void fraction, and open circles show that of the current vessel. The neutron intensity of new vessel was 1.7-2.6 times for current vessel in the energy range from 3.5×10^{-4} eV to 0.1 eV. It has been known as reentry cavity effect in the geometry in hydrogen moderator such as the ORNL, NIST and ILL that the neutron gain is improved. The flux distribution of neutron beam on a plane between the beam tube and the joint tube also was checked in uniformity. It was observed that distribution in cold region (above 0.01 eV) was uniformly flat but that distribution in thermal region (0.01 - 0.1 eV) became low to moderate insufficiently at core side. In consideration of installation condition, this shape was selected from some of other geometries such as a horizontal cross-section into a hollow annulus, a half opened hollow annulus and

a crescent shape. The radiation heating of the new vessel was 211W that is 75% of the current vessel, because gamma heating of aluminum is lower than that of stainless steel by material density. It was shown that the cooling condition also was excellent with advantage of twice neutron gain in relative.

The neutron spectra at C2-1 port were calculated for 1Qc mirror as current mirror, 2Qc and 3Qc performance mirror with industrial specifications. The neutron source was assumed to be uniform spatial distribution and spectra, which was edited from MCNP result. The calculation source became rational approximation because it is difficult to improve statistics accuracy on the joint source distribution. We impose ourselves on a limiting condition that the gain factors for each guide should be controlled in a little impact on the distribution in comparison with current condition. Neutron transport efficiency at 4Å could be improved dramatically 4.4 times if the current 1Qc mirror could be replaced with the 3Qc mirror. Also, the characteristic wavelength would be shifted from 4 Å to 2.4 Å. The divergence of neutron beam increased with the Qc value of mirror in comparison with same wavelength, but the divergence amplification factor decreased with the Qc value. In case of 4 Å required a Low-energy Triple-Axis Spectrometer (LTAS) at C2-1 port, half bandwidth in beam divergence of 1Qc, 2Qc and 3Qc were 0.6, 0.9 and 1.1 degrees in horizontal direction (see Figure 4) respectively, and were 0.8, 1.4 and 1.7 degrees in vertical direction respectively. The gross neutron intensity would be increased 5.2 times in 3Qc case. These results indicated that the cold neutron beam intensity could be improved about 10 times for the current intensity by two modifications.

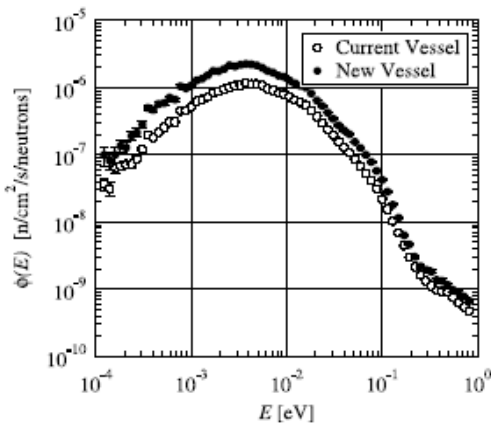


Fig.3 Neutron spectra on the joint point for the flask shape vessel and the boat-bottom shape vessel.

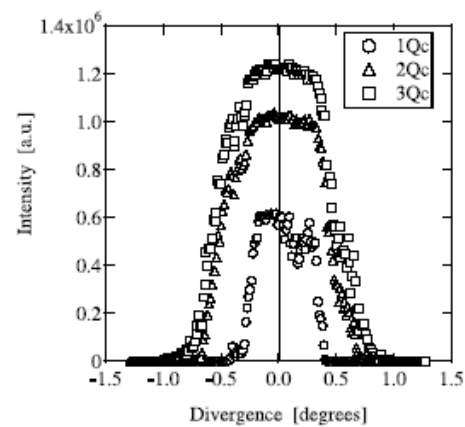


Fig.4 Horizontal neutron beam divergences with 4 Å at the LTAS beam port for 1Qc, 2Qc and 3Qc mirror.

3. Development of Irradiation technology for expanding BNCT applicability

Though BNCT clinical studies for head and neck malignancies were started recently at Kyoto University Research Reactor [8], it is difficult to carry it out in JRR-4. The reason being that positioning to keep an affected part such as the mumps in a irradiation field in case of BNCT for head and neck cancer because the shoulder of patient touches to a wall of irradiation room. So we developed a new collimator, which has a circular opening of 12cm in diameter, for head and neck cancer. The long collimator jutting out from the wall to 15cm was very necessary to apply the BNCT for the head and neck cancer.

The performance of the special collimator were compared with a circular opening collimator of 12cm in diameter, which is usually used for the brain tumor, in the thermal neutron flux

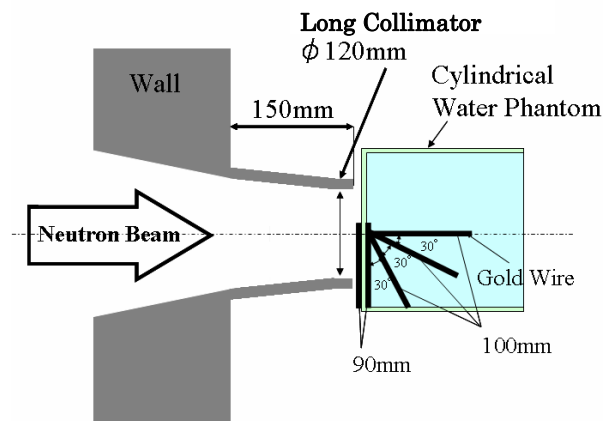


Fig.5. Arrangement of cylindrical water phantom to long Collimator.

distributions in a cylindrical water phantom. The thermal neutron fluxes in the phantom for the long collimator and for the standard collimator were measured with two beam conditions of Thermal Neutron Beam mode 1 (TNB-1) and Epithermal Neutron Beam mode (ENB). The measurement results are shown in Figure 6. Positions of thermal neutron peak, which is generated around the phantom surface, of the long collimator for two beams were a little deeper than that of the standard collimator. However irradiation time doubled because the maximum values for each mode were reduced by half of the standard collimator value. The simulated value with MCNP in thermal neutron flux as shown in Figure 6 shows a good agreement compared with measured values. These results show the propriety of MCNP joining sources that used for dose evaluation calculations for patients.

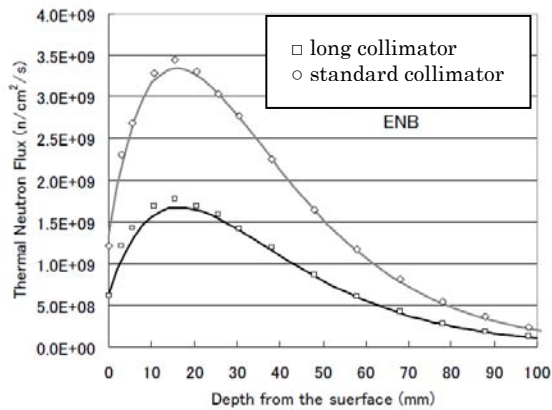


Fig.6-(a) Thermal neutron flux in water phantom for epithermal neutron beam mode

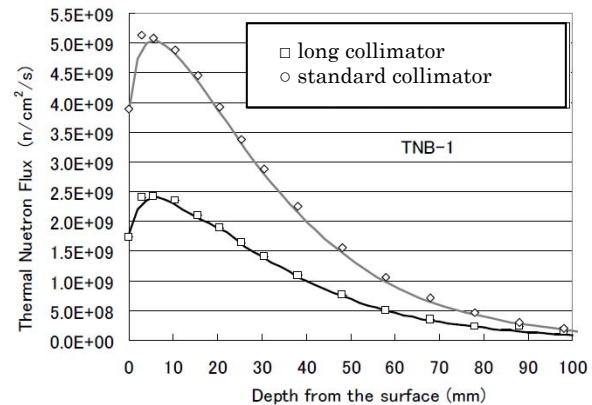


Fig.6-(b) Thermal neutron flux in water phantom for Thermal neutron beam mode 1

We developed a new collimator for head and neck cancer. It is difficult in positioning to keep an affected part such as the mumps on an irradiation field in case of BNCT for head and neck cancer because the shoulder of patient touches to a wall of irradiation room. The long collimator jutting out from the wall to 15cm was very necessary to apply the BNCT for the head and neck cancer.

4. Conclusion

Through the conceptual design for upgrading JRR-3 in cold neutron beam, it has been shown that the intensity of cold neutron beam may enhance 10 times as the current intensity. More specifically optimization of the CNS vessel could increase the intensity twice, and neutron guides replacement with high efficiency 3Qc mirror could increase about 5 times. We developed the long collimator jutting out from the wall to 15cm for head and neck cancer. The long collimator has been already applied effectively as the special collimator for 11 times of clinical trials of head and neck cancer. We will advance the development of irradiation technology for expanding BNCT applicability against the other cancers such as lung cancer.

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U(Al,Si)₃ STABILIZATION BY ZR ADDITION

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ABSTRACT

Four alloys were made within the quaternary system U-Al-Si-Zr in order to assert the minimum Zr content for fixed 0.1 wt. % Si content that could stabilize (U,Zr)(Al,Si)₃ against U(Al,Si)₄ formation. Heat treatments at 600 °C were undertaken and samples analyzed by means of XRD, EMPA and EDS techniques. Evidence was found of both Zr solubility in UAl₃ and presence of UAl₄ phase for a maximum of 6 wt. % Zr content in the alloy. A remarkable conclusion is that Zr was only found in the primary solidified phase (U,Zr)(Al,Si)₃, which does not even partially transform to U(Al,Si)₄ under heat treatment.

1. Introduction

As a means of reducing high porosity formation in the interlayer between uranium alloy solution and aluminum matrix in dispersed fuel, it has been proposed that UAl₃ (cP4, space group 221) compound should be stabilized against UAl₄ (oI20, space group 74) formation. Promising results have already been obtained showing that Si addition to the aluminum matrix is able to inhibit UAl₄ formation [1,2,3]. However, it was also noticed that minor silicium quantities should be required in the presence of fourth element collaboration [4]. An already suggested candidate is Zr, as a theoretical construction based on semi empirical Miedema formation energies predicts [4]. Experimental evidence [5] shows that 14% zirconium third element addition is enough to inhibit UAl₄ formation. No experimental results have been found showing simultaneous behavior of both silicon and zirconium when added to U-Al alloys. We thus decided to determine U(Al,Si)₃ phase stability for constant Si concentration as a function of zirconium content. We performed stabilization experiments in 50U 49.9Al 0.1Si alloys with 0, 1, 3 and 6% Zr addition (weight percentages). Heat treatments at 600°C (100h and 1000h) were undertaken and results were analyzed by x-Ray diffraction, metallographic and composition measurement techniques. UAl₄ was not suppressed with Si and Zr contents evaluated in this work, as a slight evidence of UAl₄ presence was still found in heat treated samples with 6%Zr content. Although, it can be observed that UAl₄ related x-Ray peaks intensities diminish with higher Zr contents. Thus, our efforts served to gain some knowledge about the possibilities of retarding the peritectic reaction that leads to UAl₄ formation.

2. Experimental techniques

Four alloys were made from uranium 99.975 %, aluminum 99,9 %, silicon 99,999 % and zirconium 99,7 % (alloys labeled 0, 1, 3 and 6, Table 1). Alloys compositions were designed taking into account the U-Al binary liquidus, as a base for quaternary solidification path estimation. In this way, we sought for UAl₃ as primary phase avoiding the formation of UAl₂. Figure 1 shows our alloys global composition in a binary U-Al phase diagram. Si content was determined following Boucher work [1] where he established that a content of 0.1 till 0.6 w% Si in a U-Al alloy is not enough to inhibit UAl₄ formation, though time required for transformation increases with Si content. We thus decided to try increasing amounts of Zr addition in order to suppress the reaction between Al and primary (U,Zr)(Al,Si)₃ to give

$U(Al,Si)_4$. Si content was chosen as 0.1 wt % in order to obtain a result evident for short heat treatment times.

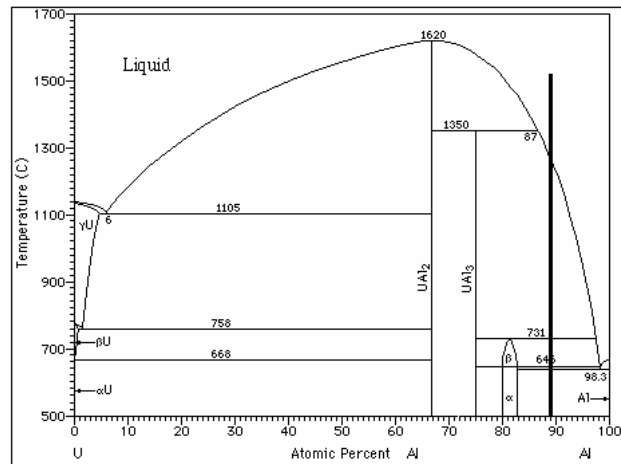


Figure 1. U-Al phase diagram [6]. The vertical line stands for our alloy composition design.

A non-consumable tungsten electrode arc furnace was employed to melt the pure components and produce the alloys under a controlled argon atmosphere using a water refrigerated copper crucible and an oxygen titanium getter. The alloys were melted and remelted at least four times to ensure homogeneity.

Heat treatments at 600 °C were made during 100 hours and 1000 hours. The interest in 100 hours experiments was focused in determining whether Boucher result was reproduced with Zr added at the same temperature and duration he experimented. In 1000 hours experiments we expected equilibria had been reached. Heat-treated specimens were wrapped in tantalum foils and sealed in silica tubes under argon atmosphere to protect them from contamination during heat treatments. Finally, they were water-quenched to room temperature without breaking the containers.

Metallographic techniques and composition measurements were employed to identify the different phases. The alloys were examined by optical microscopy and by scanning electron microscopy (SEM) in a PHILIPS PSEM-500 apparatus. They were also analyzed by Energy Dispersive Spectroscopy (EDS) in the same equipment and electron-probe microanalysis (EPMA) in a Cameca SX50 equipment at 20 KV, fitted with a wavelength-dispersive spectrometer; and by X-ray diffraction (XRD) in a PW3710 BASED PHILIPS equipment employing $Cu-K\alpha_1$ and $K\alpha_2$ radiation at room temperature.

Prior to optical, SEM and EPMA analysis, the samples were grounded on silicon carbide paper, polished with diamond paste (1 μ m) and electrolytically polished at 20 volts during 10s using a solution of phosphoric acid, ethylic alcohol, distilled water and 2-n-butoxiethanol at volume ratio 54: 21.6: 2.8: 21.6. XRD samples were free from electrolytic polishing in order to exhibit a plane surface.

Sample (label)	U (wt.%)	Al (wt.%)	Si (wt.%)	Zr (wt.%)
0	50.00	49.90	0.1	0.0
1	49.47	49.43	0.1	1.0
3	48.48	48.42	0.1	3.0
6	47.00	46.90	0.1	6.0

Tab 1. Alloys compositions.

3. Results and Discussion

a) Morfological evolution with heat treatment at 600 °C

Metallographic examination of 0, 1 and 3 as cast alloys revealed a primary phase surrounded by an eutectic component (Fig 2). The same primary phase is observed in alloy 6, while the eutectic is not quite evident.

After heat treatments, a broadening with time is observed in eutectic precipitates, remaining the distinction between primary phase and eutectic components (Fig 4).

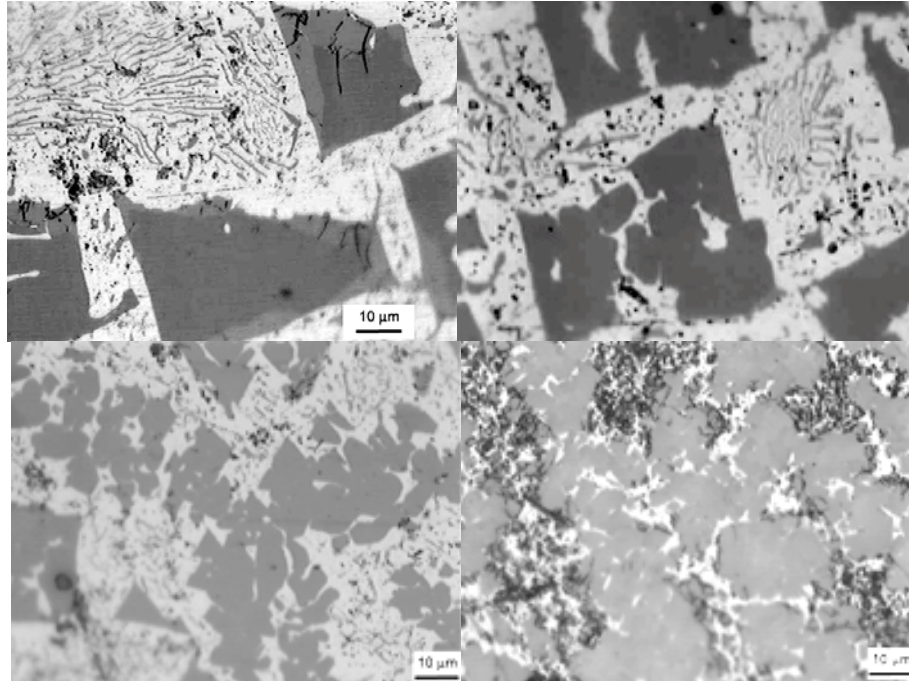


Figure 2. From left to right, as cast metallographies of samples 0, 1, 3 and 6.

b) $U(Al,Si)_4$ formation

Analyses of XRD patterns revealed the presence of UAl_3 , UAl_4 and Al phases in alloys 0, 1 and 3 in as cast samples, and in heat treated samples from all alloys. However, in the as cast sample from alloy 6 only UAl_3 and Al phases were detected. In Table 2 we resume intensities belonging to (013) peak characteristic of UAl_4 phase. Heat treated samples evidence smaller quantities of UAl_4 phase with increasing zirconium content. Though this analysis would not be accurate concerning as cast samples, peaks intensities serve so as to assert UAl_4 presence.

	0	1	3	6
As cast	2.80	7.40	1.40	Not detected
100h 600°C	28.2	53.3	4.80	2.5
1000h 600°C	No pattern	40.2	13.8	7.6

Table 2. Intensities of UAl_4 (013) peak relative to more intensive peak in the sample (in %), denoting UAl_4 presence in samples from all alloys with different heat treatments.

c) Solubility of Zr in ternary phases

A first analysis was made from XRD patterns. In both as cast and heat treated samples, lattice parameter of UAl_3 phase shows a tendency to diminish with increasing zirconium content, while lattice parameters belonging to Al phase are invariant (Table 3). This fact is seen as an evidence of solubility in the UAl_3 phase and not in Al. Variations are also evident in UAl_4 lattice parameters with increasing Zr content. These could arise from the small size of UAl_4 particles, which are expected to render smaller as Zr content is increased, thus leading to

broadener peaks and more uncertainty in the line assessment. Possible microstrains between this phase and surrounding UAl_3 or Al phases could also contribute to uncertainty.

Sample	Lattice parameter ($\text{\AA}$) ($\pm 5 \cdot 10^{-2} \text{\AA}$)		
	Al_3U	Al_4U	Al
As cast			
0	4.27	a=4.40 b=6.25 c=13.73	4.05
1	4.26	a=4.40 b=6.25 c=13.73	4.05
3	4.25	a=4.42 b=6.25 c=13.90	4.05
6	4.22	-	4.05
HT 1000h 600°C			
1	4.25	a=4.39 b=6.22 c=13.67	4.05
3	4.25	a=4.40 b=6.25 c=13.73	4.05
6	4.23	a=4.40 b=6.25 c=13.74	4.05

Table 3. Lattice parameters calculated from experimental diffraction patterns.

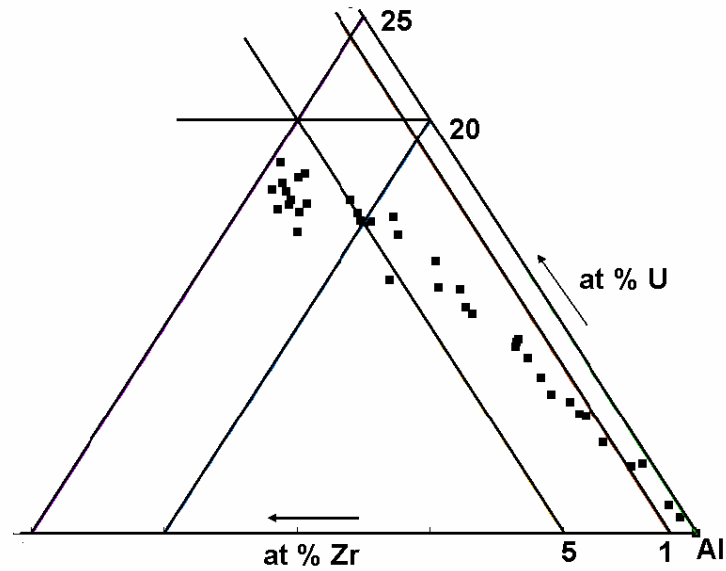


Figure 3. EPMA composition measurements in sample 6 heat treated at 600°C during 1000h.

Composition measurements were undertaken both by EPMA and EDS techniques in samples heat treated during 1000 h at 600°C. Tendencies from EPMA are shown in Figure 3 while Figure 4 shows a typical backscattered electron image obtained by EDS in sample 6. Figure 3 shows composition tendencies in the ternary phase diagram U-Al-Zr. We consider it is representative of the quaternary diagram as all samples have the same Si content and this content is small enough to consider that the ternary diagram is not quite modified. EPMA

data was measured on the primary phase and on the Al matrix. Analysis reveals that the primary phase is $(\text{U,Zr})\text{Al}_3$ and the matrix is Al, measurements in between the two corresponding compositions serve as to depict a segregation path. In agreement with this, EDS measurements showed the presence of Zr throughout the primary phase, but the small precipitates in the matrix (Fig 4) turned out to content 0 % Zr. As XRD demonstrated the presence of UAl_4 phase, and the primary phase is UAl_3 after EPMA, we conclude that the small precipitates in the matrix, forming with it the eutectic component, are UAl_4 particles. And thus, after EDS measurements, we can assert that Zr is not found in solution in UAl_4 but only in UAl_3 .

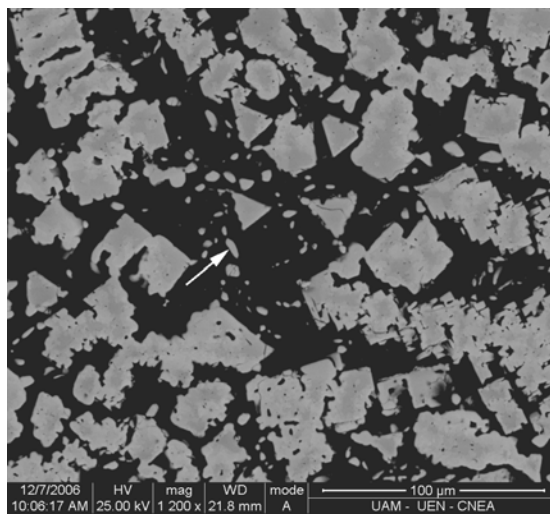


Figure 4. EDS backscattered electron image of sample 6 heat treated at 600°C during 1000 h. The arrow indicates a typical particle that resulted free from Zr.

Conclusions

- The alloys with 0.1wt.%Si studied in this work show a primary phase consisting in $(\text{U,Zr})(\text{Al,Si})_3$ and a eutectic component consisting in an Al matrix with $\text{U}(\text{Al,Si})_4$ particles. $\text{U}(\text{Al,Si})_4$ grew with heat treatments, as it was demonstrated by XRD patterns from sample 6 stabilized for 1000 h at 600°C. Nevertheless, $(\text{U,Zr})(\text{Al,Si})_3$ phase forming the primary phase did not transform to $\text{U}(\text{Al,Si})_4$ as it was demonstrated by composition measurements.
- The addition of Zr in amounts till 6 wt. % to 50U 49.9Al 0.1Si alloys does not inhibit UAl_4 formation. Higher Si contents should be tried. Moreover, attention should be paid to solidification path in order to avoid the eutectic formation.

Aknowledgements

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A STUDY ON POSSIBILITY OF USE OF LEU MR-6 TYPE FUEL FOR ADS DESIGN

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ABSTRACT

Recent studies are carried out at the VINCA Institute for possibility of use of Russian future low-enriched uranium-dioxide fuel (dispersed in an aluminium matrix) of TVS or MR-6 type, assuming that such fuel will be available in Russian Federation soon, due to development of new fuel within RERTR Program. Here are shown initial results for neutron yield and spectrum for interaction of proton beam with idealized cylindrical long target obtained by MCNPX2.4.0 code and results of criticality calculations and neutron spectra obtained by MCNP5 code for a case of the ADSRF with LEU MR-6 type fuel assemblies in lead matrix. The proposed ADSRF, beside its usage as a valuable research machine, may contribute to following and developing contemporary nuclear technologies in the country useful for eventual future nuclear power option.

1. Introduction

The idea of the ADS project is based on the fact that the construction of the TESLA Accelerator Installation in the Vinča Institute will be finished within few years. In addition, a rich experience in design, construction, operation and maintenance of both research reactors (RR) in the Institute, gained in the last 40 years, will be used. Proposal for a conceptual design of an accelerator driven subcritical research reactor was based initially on the stock of available fresh HEU TVR-S type fuel elements (FE) of Russian origin at both Vinča's RR. The basic design request for the ADSRR features is that it should simulate neutron characteristics of high-power ADS with intermediate neutron spectrum. Such ADSRR could respond to worldwide requests for experimental needs for validation of various nuclear parameters of the proposed ADS. Characteristics of such type ADS with HEU (ADSRR-H.) were studied in period 1999 – 2002 [1 – 4].

In compliance with the RRRFR Program, the Vinča Institute has returned fresh HEU FEs back to the Russian Federation [5] in August 2002. As a usage of HEU FEs in research reactors is not further recommended, a new study of ADS facility design was initiated in last few years, based on usage of a modern Monte Carlo particles transport codes and assumed commercial availability of LEU FEs of Russian origin. This conceptual design of the proposed ADSRR with TVR-S type FEs was studied in period 2002 – 2004 and is designated as ADSRR-L. Recent feasibility studies are related to possibility of usage of LEU MR-6 type FAs in ADS and this facility is designated as ADSRF-MR6. Purpose of this paper is to show initial results of neutronic designed parameters of the proposed ADSRF-MR6 and compare them to the basic requirements and spectrum characteristics of the previous ADSRR-H.

2. TESLA Accelerator Installation

The TESLA Accelerator Installation (TAI) in the Vinča Institute is a multi-purpose facility for production, acceleration and use of ions. Construction of the installation began in 1989 and comprises a compact isochronous cyclotron - the VINCY Cyclotron; an electron cyclotron resonance heavy ion source the - mVINIS Ion Source (in operation from 1994); a volume light ion source - the pVINIS Ion Source (in operation from 1994), and several low and high-energy experimental channels. Due to interruptions, from various reasons, in providing funding for the construction, the first beam extraction from the VINCY cyclotron is foreseen at the end of 2008.

The VINCY cyclotron, optimised for extraction of deuteron beam, can deliver either protons with maximum energy of 73 MeV and current of 5 μ A, or deuterons with maximum energy of 67 MeV and current of 50 μ A. These parameters are not favourable for design of an ADS for energy production or transmutation of trans-uranium nuclides. Thus, one of the main tasks in the ADSRF project is examination of interaction of the beam particles with different target materials in order to choose and design an optimal target in respect to escaping neutron spectrum and neutron yield.

3. MR-6 Fuel Design

The MR type of FAs are produced, among other research reactors' FA types, at the Novosibirsk Chemical Concentrate Plant (NCCP), Novosibirsk, Russian Federation produces [6] These FAs may have different number (four to six) concentric fuel element tubes. Fuel meat is UO_2 , dispersed in aluminium matrix. Uranium is enriched to 36 % or 80 %. Numerical neutron multiplication and spectrum studies with LEU (19.7 % ^{235}U) MR-6 fuel assembly (FA) were already carried out for possibly conversion of HEU to LEU fuel at Poland's MARIA reactor [7]. For feasibility study in this paper, MR LEU FAs with six concentric fuel tubes (MR-6) were selected. It is assumed that uranium is enriched to 19.7 % ^{235}U and UO_2 is dispersed in aluminium alloy powder (PA-4) matrix [8]. Mass of 450 g of ^{235}U per FA is assumed [7]. Cross-section of MR-6 type FA is given in Figure 1.

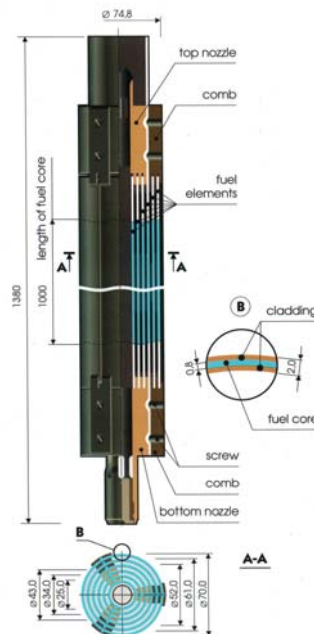


Fig. 1. MR-6 type fuel element (dimension in mm) [6]

The fuel layer of each tube of MR-6 FA is 0.8 mm thick and covered by 0.6 mm thick aluminium cladding [6] that is assumed to be made from Russian aluminium alloy AMSN2 [8]. Inner and outer diameters of fuel tubes are shown in the Figure 1. Gaps between tubes and central axial hole are supposed to be used by moderator (water). Length of the fuel layer is 1000 mm.

4. Conceptual design of ADSRF-MR6

In ADSRF with LEU MR-6 FA, gaps among fuel tubes in the MR-6 are filled by water, except in the central FA, in which axial hole is used for a target position. Total of 21 FAs (supporting external tubes OD 76 mm) are placed in holes (ID 78 mm) drilled in a lead matrix of the subcritical reactor core. The holes form a regular square lattice (5 x 5 matrix) with 110 mm pitch. Water is selected again for the primary moderator inside the FAs in aim to achieve dominant thermal energy neutron fission rate, while lead matrix is used in the subcritical core with the aim to achieve an intermediate energy neutron spectrum. Radial lead reflector is designed with average thickness of about 260 mm. There are also axial lead reflectors, above and below stainless steel grids, mounted at top and bottom of the core,

used to fix FAs at positions in the lattice. Height of the ADSRF-MR6 core is 1380 mm. The total mass of ^{235}U nuclide in the core is 9.45 kg, while total mass of U is five times higher: 47 kg.

The subcritical core is designed in a cylindrical stainless steel (SS) vessel (ID/OD 100/105 cm). The total mass of the whole facility is about 10 tons. Beam of charged particles is introduced to the subcritical core (through the top surface along central axis) by a separate SS tube (ID/OD 12/14 mm) under high vacuum. An idealised target, a solid metal lead cylinder (25 mm high and 10 mm diameter), is placed at the end of the beam tube, near the centre of the central FA in the core. Cross sections of designed ADSRF-MR6 core are shown in Figure 2.

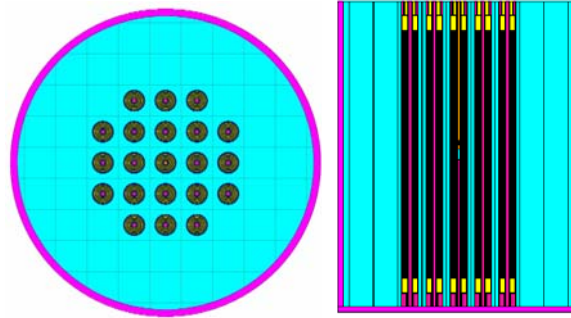


Fig. 2. Horizontal and vertical cross-sections of ADSRF-MR6

In aim to estimate escape neutron spectrum and neutron yield from the target surfaces, a 73 MeV proton parallel beam with circular cross section is assumed to hit the top surface of the lead cylindrical target along the cylinder axis. Neutron spectrum, obtained from calculation of interaction of the beam with the target material in idealized conditions (target surrounded by vacuum) by the MCNPX2.4.0 code [9] with the LA150 [10] continuous energy data library, is used as a neutron source in the subcritical system in this study. Escaping neutron and proton spectra are shown in Figure 3 for beam radius equal to zero and equal to the target radius (0.5 cm). Range of 73 MeV protons in lead is 0.90 cm, and as a consequence, only small percent of incoming protons escape the target. Energy deposited by protons in the target is shown in Figure 4. The total yield of escaping neutrons from all target's surfaces is calculated with the SHIELD [11] and MCNPX codes at 0.143 ± 0.003 neutrons per one 73 MeV proton, regardless the beam diameter. Less than 3.5 % of escaping neutrons have energy higher than 20 MeV. Peak of the escaping neutron spectrum is in energy range from 0.2 MeV to 0.8 MeV. There are no neutrons with energies below 1 keV, i.e., there is no thermalisation of neutrons within the target material. Fraction of protons escaping target surfaces is 14.5% in the case of 1 cm diameter beam and only 0.09 % in the case of infinite thin beam.

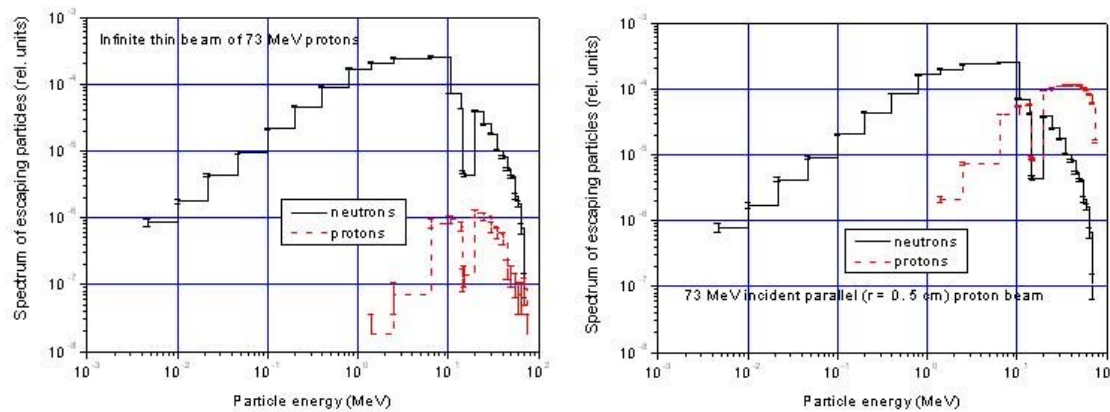


Fig. 3. Escaping neutrons and protons from the lead target

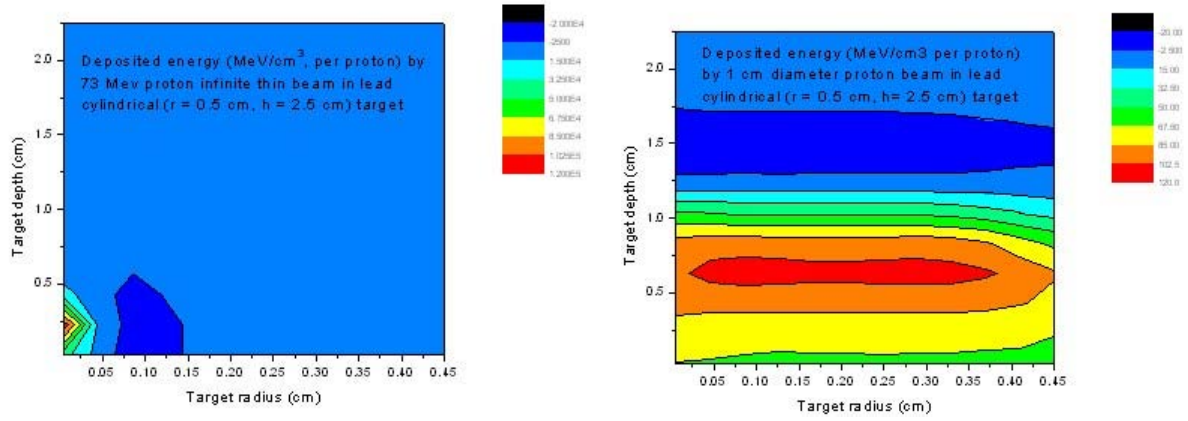


Fig. 4. Energy deposited by proton beam in the lead target

5. Results of neutronic study

The MCNP5 code [12] is used for criticality calculations of the ADSRF-MR6 with combination of the LA150 (neutron $E_{\max} = 150$ MeV) and ENDL92 (neutron $E_{\max} = 30$ MeV) neutron cross-sections data libraries. Only for few nuclides (impurities in the ADSRF materials) ENDF-B/V1.8 (neutron $E_{\max} = 20$ MeV) data are used. Scattering of thermal neutrons at hydrogen atoms connected in light water molecules is encountered by application of the $S(\alpha, \beta)$ scattering law, according to the TMCCS library. Three-dimensional (3D) model of real MR-6 FA and 3D model of ADSRF-MR6 for the MCNPX/MCNP codes are developed and used.

To specify a neutron source (SDEF option) for the MCNP5 code, in the first step of the calculations, neutrons escaping the target volume with group spectra obtained in the MCNP5 code are used. The code is run for 200 000 neutron histories and the five-group spatial distribution of neutron flux and fission rate source in ADSRF-MR6 core are determined with (1σ) error less than 10%. Such spatial distribution of fission neutron source is used for the subsequent MCNP5 calculations of the neutron effective multiplication factor (KCODE option) in the ADSRF-MR6. In this step of calculation by the MCNP5 code, 2000 neutron active cycles are run with 5000 neutrons per cycle, after 100 initial ones. Simultaneously, 56-energy groups neutron spectra in various cells of the ADSRF-MR6 lattice are calculated (Figure 5). Average effective neutron multiplication factor (k_{eff}) of (0.9574 ± 0.0003) and prompt removal neutron lifetime (l_p) of $(65.77 \pm 0.04) \mu\text{s}$ for standard (1σ) statistical uncertainty valid for 0.67 probability are calculated. Average neutron energy producing fission is 33.255 keV.

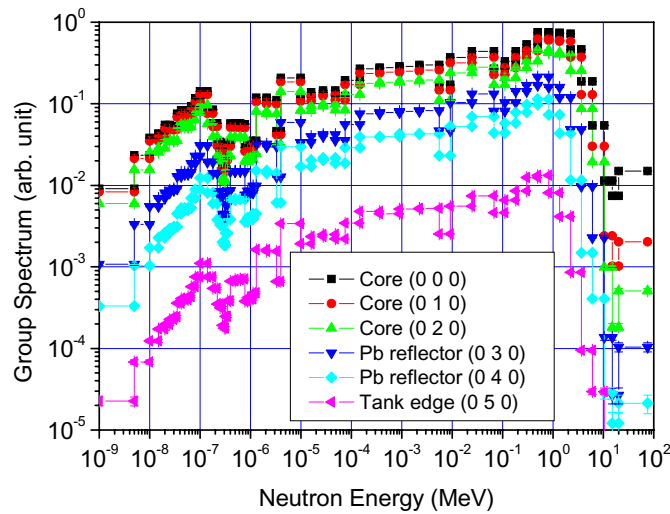


Fig. 5. Neutron spectra in cell zones of the ADSRF-MR6

Figure 5 shows that stationary neutron spectrum in the ADSRF cells is dominantly intermediate. Average neutron flux in lattice cells with LEU FAs is dominantly fast and about from 3 to 7 times higher than the thermal one. Total epithermal and fast energy neutron flux is about one order higher than thermal one at the core-reflector and the reflector-tank edge of the unshielded system. The MCNP5 calculations show that neutrons in the system are generated dominantly (83.78 %) by thermal fission ($E_n < 0.625$ eV) in ^{235}U nuclide. In three group representation, fraction of intermediate fission ($0.625 \text{ eV} < E_n < 100 \text{ keV}$) is only about 14.43 % of total number of fission, and only 1.79 % of all fissions are generated by fast neutrons ($E_n > 100 \text{ keV}$).

6. Conclusion

The basic aim of the ADSRF-MR6 research is to study the physics and development of the technologies necessary to design a small subcritical low neutron flux research facility driven by an accelerator medium energy proton beam. Russian origin research reactor's LEU fuel elements of existing construction are assumed to be available. Initial results of the neutronic study have shown that it is possible to design such ADSRF with dominant intermediate neutron spectrum using fresh low-enriched uranium MR-6 type fuel elements and light water as primary moderator in lead matrix. That system is driven by a neutron source generated in a lead target by interaction of proton or deuteron beam, extracted from the TESLA Accelerator Installation.

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THERMAL CONDUCTIVITY OF HEAVY-ION-BOMBARDED U-MO/AL DISPERSION FUEL

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ABSTRACT

Changes of thermal conductivity during in-pile irradiation are of central importance for the large-scale use of U-Mo/Al dispersion fuel. Recently it was shown [1], that heavy ion bombardment of U-Mo/Al dispersion fuel allows to simulate the effects of radiation damage during in-pile irradiation. Heavy ion bombardment avoids (strong) activation of the specimens. They may therefore be readily examined in simple laboratory experiments. We report the development of a new method to determine changes of the thermal conductivity of U-6wt%Mo/Al and U-10wt%Mo/Al dispersion fuel due to ion bombardment. We derive changes of the DC thermal conductivity from a low frequency heat current as tracked by a digital lock-in technique. A comprehensive set of tests has been carried out that establishes the basic feasibility of our method.

1. Introduction

U-Mo alloys dispersed in Al are most promising candidates for high density fuels for research and materials testing reactors. It is most important for the qualification of this fuel for use in reactors to know the development of its thermal conductivity during burn-up. In this paper a method to measure the change in thermal conductivity of U-Mo/Al dispersion fuel caused by bombardment with heavy ions will be presented. The bombardment with heavy ions simulates part of the damages inside the fuel during in-pile irradiation [1-4]. Values will be presented and discussed.

The area that could be exposed to bombardment with swift heavy ions is approximately $3 \times 4 \text{ mm}^2$ of size. Since swift heavy ions of 80 MeV energy can penetrate only the first $10 \mu\text{m}$ s of the U-Mo/Al sample, the expected change in thermal conductivity of the whole sample is lower than 5%. The size of the samples bombarded with heavy ions was approximately $3 \times 15 \text{ mm}^2 \times 150 \mu\text{m}$. It was a challenging task to resolve such a small change in thermal conductivity inside such a small sample. A new method had to be found since known methods would not provide the needed accuracy.

Common steady-state methods to measure the thermal conductivity of a sample suppose both, bar shaped samples of some 10cm length and big heat currents. However, they can hardly resolve temperature gradients of less than 5K [5].

Known methods that use a heat-wave propagating through the sample to determine the diffusivity κ of the sample suppose partly samples of some 10cm length to give satisfying results [6,7]. Furthermore, the shape of the samples and the position of the temperature measurement on the sample must satisfy strict border conditions [8,9]. The standard laser-flash method to determine the thermal diffusivity of the samples could not be applied since it supposes very homogeneous samples [10]. In our case, the size of the inhomogeneities (the U-Mo/Al particles, diameter $< 150 \mu\text{m}$), and the thickness of the samples ($< 300 \mu\text{m}$) are in the same order of magnitude. Furthermore it is known, that the laser-flash method leads to inaccuracies due to double inflections when applied to thin dispersion samples [11].

Therefore, we developed a quasi-static method that uses a very low-frequency heat wave to determine the diffusivity of the U-Mo/Al samples. Using this method, one can avoid parasitic voltages that may

appear during the temperature measurement. The method will be described in chapter 2. Additional to the measurement of the change of thermal conductivity of the samples due to bombardment with swift heavy ions, the change in electrical resistivity of the samples was measured with a classical four-point-method. The change in the thermal conductivity and the electrical conductivity of the individual samples can be compared using the Wiedemann-Franz law.

2. Experimental set-up

The basic idea of our method is sending a very low-frequency (in the order of some 10mHz) heat wave lengthwise through the sample. The amplitude and the phase of the heat wave are recorded on two different points lengthwise on the sample. The diffusivity κ of the sample can be calculated from the amplitude and the phase respectively. The wavelength of the heat wave is thereby much bigger than the length of the sample. The temperature along the sample is therefore always nearly constant.

The heat wave is produced with a small electrical heater mounted on top of the sample. The bottom of the sample is mounted in a small copper sample-holder that functions as a heat dumb.

We use the well-known equation of heat conduction:

$$P = \lambda \cdot A \frac{\Delta T}{\Delta x} \quad (2.1)$$

where P is the thermal output of the heater, λ is the thermal conductivity of the sample, A is the cross section perpendicular to the surface, ΔT is the difference of the amplitude of the heat wave inside the sample on two different points lengthwise the sample and Δx is the distance of the two measuring points.

Solving equation (2.1) to determine λ leads to an expression that is proportional to the inverse of the difference of the amplitude of the heat wave. This expression has the dimension of a thermal conductivity [$\text{Wm}^{-1}\text{K}^{-1}$] and will be called “AC thermal conductivity” λ_{AC} in the future.

$$\lambda_{AC} = \frac{P \cdot \Delta x}{A \cdot \Delta T} \quad (2.2)$$

We use the model of the propagation of a heat wave in a semi-infinite half space derived by Carslaw and Jaeger [12]. The diffusivity κ of the sample can be calculated from the frequency dependant AC thermal conductivity and from the phase difference $\Delta\Phi$ of the heat wave between the two measuring points.

Using the model of Carslaw and Jaeger [9] the AC thermal conductivity can be best described with

$$\lambda_{AC} = \frac{P_0 \Delta x}{AT_0 (1 - \exp(-\Delta\Phi))} \exp\left(\sqrt{\frac{x_1^2 \pi}{\kappa}} \sqrt{f}\right) \quad (2.3)$$

where f is the frequency of the heat wave and x_1 is the distance from the heater to the first measuring point on the sample.

When plotting the values for the AC thermal conductivity calculated with equation (2.2) over the frequency of the heat wave one can interpolate those with a function of the form $y = P_3 \exp(P_4 \sqrt{x})$.

Comparing this with equation (2.3) leads to an expression for the diffusivity κ of the sample:

$$\kappa = \frac{x_1^2 \pi}{P_4^2} \quad (2.4)$$

For the phase difference of the heat wave on two points x_1 and $x_1 + \Delta x$ on the sample one gets after Carslaw [12] the following expression:

$$\Delta\Phi = \Delta x k + \varepsilon_0 = \sqrt{\frac{\Delta x^2 \pi}{\kappa}} \sqrt{f} + \varepsilon_0 \quad (2.5)$$

$$\Delta\Phi = a\sqrt{f} + b \quad (2.6)$$

Taylor expanding (2.5) to f around 1mHz and aborting after the linear term leads to:

$$\Delta\Phi \approx \left(\frac{1}{2} a \sqrt{1\text{mHz}} + b\right) + \frac{a}{2\sqrt{1\text{mHz}}} f \quad (2.7)$$

The phase difference of the heat wave is plotted against the frequency and interpolated with a function of the form $y = P_1 + P_2 x$. One gets for the diffusivity κ :

$$\kappa = \frac{\Delta x^2 \pi}{4P_2^2 \cdot 1\text{mHz}} \quad (2.8)$$

All measurements were carried out with two digital lock-in amplifiers that controlled the heat wave created by the heater and recorded both, the amplitude and the phase of the heat wave on two different points on the sample at the same time. The heat wave was detected with two type K thermocouples that were glued onto the sample with a two-component, electrical conducting silver epoxy glue. The distance of the two thermocouples was $\Delta x \approx 4\text{mm}$. Figure 1 shows the experimental set-up we used.

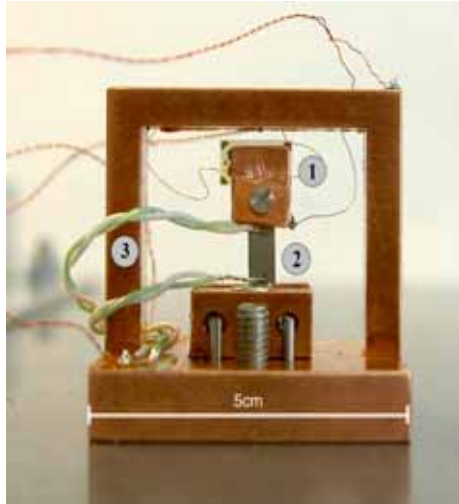


Fig. 1: The experimental set-up we used to determine the amplitude and the phase propagating through the sample (2). A small heater is mounted on top of the sample (1). The heat wave propagates from the heater through the sample into the experimental set-up, which works as a heat sink. The amplitude and the phase of the temperature wave are detected with two thermocouples (3).

Additional to the thermal diffusivity the electrical conductivity of the samples was measured before and after bombardment with heavy ions with a classical four-point-set-up. Furthermore, the specific heat of the U-Mo/Al samples and the silver-epoxy glue was measured with a commercial available device by Quantum Design. Results will be given in chapter 3.

3. Results

There were in total 4 samples irradiated with iodine ^{123}I at 80MeV at the tandem accelerator of the “Maier-Leibnitz Laboratorium” in Garching. All samples were measured before and after bombardment with iodine. The samples were provided by Argonne National Laboratory (ANL). There were samples with U-10wt%Mo (J8, J9) and U-6wt%Mo (R11, R12) examined. The volume loading of the samples is around 55vol%U-Mo in an aluminium matrix. The preparation for examination was done inside a hot laboratory at the Garching campus. Table 1 shows the specifications of the four samples including the fluency of iodine after bombardment.

<i>Name</i>	<i>Alloy</i>	<i>Dimensions [LxWxT]</i>	<i>Distance thermocouples</i>	<i>Fluency [Ions/cm²]</i>
J8	U-10wt%Mo	15,85x3,1x0,3mm ³	4,9-11,45mm	1,5x10 ¹⁷
J9	U-10wt%Mo	14,65x3,35x0,3mm ³	2,85-10,1mm	2,03x10 ¹⁷
R11	U-6wt%Mo	13,1x3,95x0,27mm ³	3,5-9,3mm	2x10 ¹⁷
R12	U-6wt%Mo	13,7x3,35x0,27mm ³	3,9-9,2mm	1,96x10 ¹⁷

Tab. 1: Summary of the U-Mo/Al samples that were examined.

The AC thermal conductivity was determined according to equation (2.2) and plotted over the frequency (figure 2a). The AC thermal conductivity is perfectly fitted with a function of the form $y = P_3 \exp(P_4 \sqrt{x})$, in agreement with equation (2.3). The phase difference of the heat wave is also plotted over the frequency (figure 2b). It is very well reproduced with a function of the form $y = P_1 + P_2 x$, in agreement with equation (2.7).

Table 2 shows the thermal diffusivity calculated from the AC thermal conductivity compared to the thermal diffusivity calculated from the phase difference and the electrical conductivity of the samples. The single absolute values of κ_{AC} , $\kappa_{\Delta\phi}$ and σ match quite well. Values of the electrical conductivity are in the order of magnitude one would expect [11]. However, values of the thermal diffusivity are about a factor 30 smaller than values of U₃Si/Al dispersion fuel of similar volume loading reported by Kim et.al. [11]. Calculating the thermal conductivity of the samples from the thermal diffusivity (the specific heat of U-10Mo was measured to be $\sim 600 \mu\text{J mg}^{-1}\text{K}^{-1}$, the specific heat of U-6Mo was $\sim 400 \mu\text{J mg}^{-1}\text{K}^{-1}$) leads to values also a factor 30 smaller than expected. The most probable explanation for this behaviour is a systematic error due to the model of the semi-infinite half space we used. Relative changes, however, should be unaffected.

Only sample J8 shows an decrease between 2,48% and 6,75% in electrical or thermal conductivity due to irradiation with swift heavy ions. In contrast to sample J8 sample J9 shows an significant increase in thermal and electrical conductivity after bombardment with heavy ions. Sample R11 shows now significant change in thermal or electrical conductivity. Looking at the measured values of sample R12 one notices that the electrical conductivity and $\kappa_{\Delta\phi}$ show an significant increase. However, κ_{AC} shows an decrease. This behaviour is most likely because of inaccuracies during the measurement. In consequence this measurement is no longer considered.

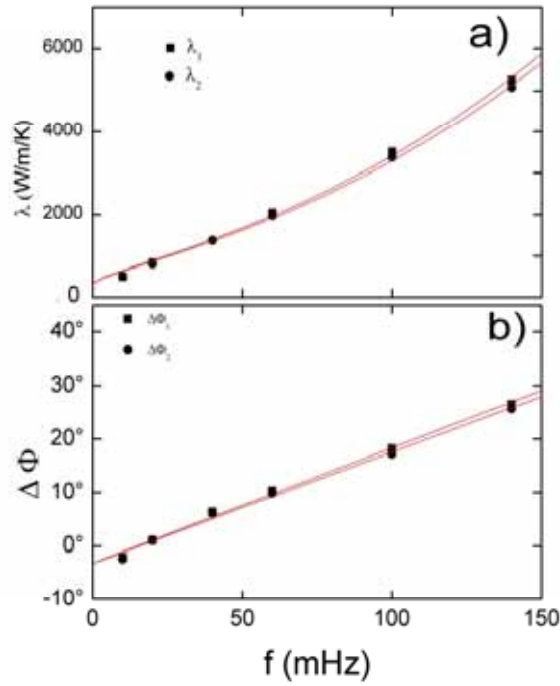


Fig. 2: AC thermal conductivity (a) and phase difference of the heat wave on two points on the sample (b) plotted over the frequency. The amplitude difference (a) is best fitted with a function of the form $y = P_3 \exp(P_4 \sqrt{x})$. The phase difference is best fitted with a function of the form $y = P_1 + P_2 x$ in excellent agreement with literature [12]. Parameters P_2 and P_4 can be used independently from each other to determine the thermal diffusivity κ of the sample.

Sample		$\kappa_{AC}[10^{-3}cm^2s^{-1}]$	$\Delta\kappa_{AC}$	$\kappa_{\Delta\phi}[10^{-3}cm^2s^{-1}]$	$\Delta\kappa_{\Delta\phi}$	$\sigma[10^6\Omega^{-1}m^{-1}]$	$\Delta\sigma$
J8	Before	8,595		12,1		9,99	
	After	8,015	-6,75%	11,8	-2,48%	9,53	-4,6%
J9	Before	5,91		40,6		4,88	
	After	6,335	+7,19%	83,4	+105%	6,01	+23,2%
R11	Before	4,87		17,8		9,11	
	After	4,89	+0,4%	16,95	-4,78%	9,14	+0%
R12	Before	8,65		12,4		9,97	
	After	7,94	-8,2%	16,65	+11,06%	10,4	+4,3%

Tab. 2: Summary of the results of the conductivity measurements. Mean values before and after bombardment with swift heavy ions are displayed. The thermal diffusivity and the electrical conductivity change always in the same direction after bombardment with heavy ions, except sample R12.

4. Discussion

In this chapter a possible explanation for the varying behaviour of the thermal and electrical conductivity of the single samples will be given [13].

It is well known that in principal the conductivity of metals decreases during irradiation due to an increasing number of point defects [14]. In our case we have a complicated mixture of zones of different material combinations. Therefore a combination of different effects on the thermal conductivity has to be considered..

The sample consists of an Al matrix which embeds the U-Mo particles. After irradiation the sample can be divided into two regions: The first region is directly hit by heavy ions which will induce lattice defects in the Al matrix as well as in the U-Mo grains. Furthermore an interdiffusion layer between the aluminium and the U-Mo is created during irradiation [1]. The volume of the irradiated region is small compared to the total volume of the sample. In the first instance the thermal and electrical conductivity of the matrix will decrease due to the increasing concentration of lattice defects. However, if the temperature of the sample during irradiation exceeds $\sim 250^{\circ}\text{C}$ the lattice defects will possibly anneal immediately [14]. No or only a little effect will be measurable in this case. It is generally accepted that the electrical and thermal conductivity of the interdiffusion layer is quite low [18]. The U-Mo particles in the irradiated region are initially mainly in the γ -phase. In [4] it has been shown, that under the irradiation conditions of this experiment the remaining α -U-Mo transforms into γ -U-Mo – see also [2,17,19,20]. Furthermore it was shown by Bleiberg 1956 that γ -U-Mo transforms into a slightly more ordered phase which he called γ' . The electrical conductivity of γ' -U-Mo is about 3% higher than of γ -U-Mo [17].

The second region is the rest of the sample. Since this region is not hit by swift heavy ions only the increased temperature during irradiation can cause effects. No interdiffusion layer is expected to grow in this region [2]. Point defects inside of the Al matrix will anneal due to the increased temperature. A slight increase in electrical and thermal conductivity of the matrix would be the consequence. Most U-Mo particles exist in the γ -phase inside the non-irradiated part of the sample. This phase transforms to the α -phase when exposed to temperatures greater than 400°C [16]. The electrical conductivity of the U-Mo α -Phase is approximately 10% higher than the conductivity of the γ -Phase [17]. In case that the temperature during irradiation was too high it is possible that a phase change from γ - to α -phase inside the U-Mo particles took place. As a consequence there would be a significant increase in electrical conductivity of the U-Mo particles. With it comes an increase in electrical conductivity of the whole non-irradiated region.

During the bombardment with heavy ions competing effects on the conductivity of the samples occur. Thermal annealing and phase changes from γ - to α -U-Mo increase thermal conductivity, whereas irradiation creates point defects which principally decreases thermal conductivity. We conclude that at the here used total fluence and complex sample composition the net thermal conductivity can either increase or decrease.

The here used total fluence roughly simulates 1/10 of a typical burn-up of fuel in a research reactor. Future irradiations have to accumulate larger total fluences while controlling more precisely the temperature within the sample. It is expected to detect under these conditions a clear tendency in the changes of the thermal conductivity.

Acknowledgement

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CHARACTERIZATION OF MONOLITHIC FUEL FOIL PROPERTIES AND BOND STRENGTH*

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ABSTRACT

Understanding fuel foil mechanical properties and fuel / cladding bond quality and strength in monolithic plates is an important area of investigation and quantification. Specifically, what constitutes an acceptable monolithic fuel – cladding bond, how are the properties of the bond measured and determined, and what is the impact of fabrication process or change in parameters on the level of bonding? Currently, non-bond areas are quantified employing ultrasonic determinations that are challenging to interpret and understand in terms of irradiation impact. Thus, determining mechanical properties of the fuel foil and what constitutes fuel / cladding non-bonds is essential to successful qualification of plate-type monolithic fuel. Capabilities and tests related to determination of these properties have been implemented and are discussed, along with preliminary results.

1. Introduction

Monolithic fuel forms are necessary to convert high power nuclear reactors that could not otherwise be converted by low density dispersion fuels. Development of these fuel forms is essential to the success of the Reduced Enrichment for Research and Test Reactors (RERTR) program. Based on the success of initial irradiations of monolithic fuel plates, an aggressive campaign to further fabricate, irradiate and qualify monolithic fuel has been developed [1]. However, challenges associated with the planar interface introduced by a monolithic fuel form still remain, in particular bonding between the fuel and cladding across the interface. Therefore, understanding bond quality and strength in monolithic fuel plates is an important area of investigation and quantification. Specifically, what constitutes an acceptable monolithic fuel – cladding bond, how are the properties of the bond measured and determined, and what is the impact of the fabrication process or change in fabrication parameters on the level of bonding?

1.1 Approach

Currently, potential non-bond areas can be identified by employing ultrasonic determinations. Determination of what constitutes an unbound area and to what degree this constitution is acceptable with high confidence is challenging and somewhat unknown. This challenge creates difficulties in drawing correlations observed in post-irradiation examinations (PIE) with pre-irradiation fabrication observations. A series of tests aimed at addressing the challenges associated with acceptable bonding behaviour determination in monolithic fuel plates is underway at the Idaho National Laboratory. Understanding the bond ‘quality’ in monolithic fuel is essential to the successful qualification of monolithic fuel plates. Two approaches have been identified and are being investigated to determine the level and quality of bonding in the monolithic fuel plates. The first approach is through characterization of the bond layer that is fabrication technique specific, i.e. friction stir welding (FSW), transient liquid phase bonding (TLPB) and/or hot isostatic pressing (HIPing). The second approach is through determining the irradiation performance of the fuel plates, allowing correlations between fabrication processes and post-irradiation examinations to be drawn.

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1.2 Characterization

Ultrasonic testing (UT) has shown some promise in determining the location and degree of non-bond areas. UT is highly desirable in the fact that the technique is non-destructive and provides information on bond quality in an efficient manner. On the other hand, mechanical testing is desirable in the fact that the technique provides quantitative information on bond strength. There are two different types of mechanical testing that can be carried out: a non-destructive technique such as a proof test, or a destructive technique such as an instrumented tensile test and/or shear test. A non-destructive proof test is coupled with UT and consists of applying a known torsional force at the ends of the fuel plate with a defined cycle, analyzing the fuel plate for non-bond areas from UT, and repeating the proof test over until a defined size of non-bond defect appears. This type of testing would demand an extremely high confidence in the UT method. A destructive instrumented tensile or shear test is carried out on both well-bonded areas and suspected non-bond areas determined by UT, and is the subject of the current paper. Changes to fabrication process parameters or conditions, e.g. addition of a secondary interface such as a diffusion barrier, lower HIPing temperatures, etc., and the impact these have on bond strength will be more easily identified and understood prior to irradiation. Microstructural characterization is carried out in a similar manner as that defined for the destructive mechanical tests. Both suspected well-bonded and non-bonded areas are sectioned creating a metallographic specimen that is mounted, prepared and examined with optical microscopy and scanning electron microscopy (SEM). Any reaction layer existing between the monolithic fuel and the cladding is clearly visible and quantified in terms of thickness and composition (employing a semi-qualitative technique such as energy dispersive spectroscopy). The microstructural method is carried out in conjunction with the ultrasonic testing and mechanical testing, creating an ensemble of information relating to bond strength and integrity.

1.3 Irradiation Performance

The general irradiation performance evaluation of bonding in monolithic fuel plates is carried out in two basic areas: modelling and post-irradiation examination (PIE). Although the modelling approach is not discussed at this time, the approach consists of finite element analysis and analytical solutions relating to both thermal and thermo-mechanical aspects of the fuel-clad interface. Specifically, these models investigate the impact of a debond on the fuel meat temperature and stress behaviour, ultimately supporting determination of an acceptable debond size and geometry. The post-irradiation examination (PIE) approach involves examination of plates previously characterized by UT scans, pull tests and/or microstructural analysis after irradiation. Specific results on PIE of the latest monolithic fuel campaign (RERTR-6) may be found in Ref. [2]. Combination of these two approaches allows observations from irradiation to be fed back into fabrication to improve subsequent irradiation experiments, utilizing characterization as an effective means to understand how and what has changed in terms of bond strength and integrity.

2. Experimental Methods

Sample plates were fabricated employing one of three processing methods, hot-isostatic pressing (HIP), transient liquid phase bonding (TLPB) or friction stir welding (FSW). An updated description of each process can be found in Ref. [3]. All of the sample plates contained a DU-10Mo (nominal wt.%) foil approximately 8.26 cm long by 1.91 cm wide with aluminium-6061-T6 used as the cladding. HIP sample plates were subjected to 580°C for ninety minutes at 103 MPa pressure. TLPB sample plates were subjected to 590°C for fifteen minutes at 6.89 MPa pressure. FSW sample plates were welded with an approximate load of 17.8 kN and an unknown temperature, although the processing temperature is believed to be in the range of 400-500 C. Sample plates were subjected to ultrasonic testing to determine whether or not debonds were present. Regions of interest (ROI's) were determined from the UT scans and marked. Test specimens (ROI's) were sectioned from the sample plates using a low-speed saw. Each test specimen was a square approximately 0.876 cm on edge. One test specimen for each fabrication method was bound to aluminium test platens using a high strength epoxy. Bonding of the epoxy involves a heat treatment of 165°C for ninety minutes after application. The low temperature heat treatment does not affect reaction kinetics or growth of an interfacial layer in a significant manner. Pull testing was carried out on the test specimens, similar to that used in determination of bond strength between thermally sprayed coatings and a substrate [4]. An in-house

test rig, shown in Fig. 1 along with a photograph of a mounted sample, was employed to carry out the pull tests. A constant crosshead rate was applied to pull the test while monitoring induced load with a tensile link load cell. A second sample from each fabrication method was cold mounted, polished using SiC paper and examined under a scanning electron microscope (SEM).

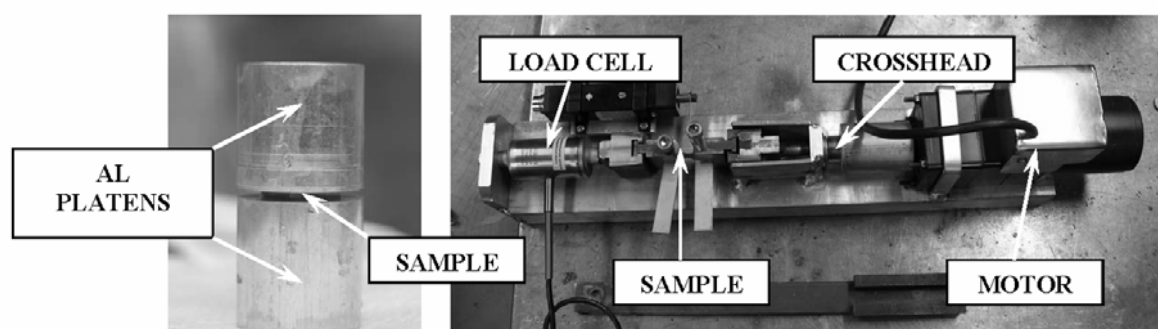


Fig. 1. Photograph of test rig employed to carry out pull tests. Pertinent features of the rig are pointed out

3. Results and Discussion

Ultrasonic testing scans of each sample plate are provided in Fig. 2. Regions of light [white] areas suggest acceptable bonding between the aluminium-aluminium cladding. Regions of light [grey] areas suggest acceptable bonding between the aluminium-fuel interfaces. Regions that appear dark in colour would suggest either a debond or inclusion/impurity in the fuel foil. However, observation of Fig. 2 reveals that this is clearly not the case for each of the sample plates fabricated using HIP, TLPB or FSW, and that each plate, based on this technique, has bonding between the fuel and cladding. Examples of SEM photomicrographs of the fuel-clad interface for each fabrication technique investigated are presented in Fig. 3. Observation of the photomicrograph for a HIP fabricated fuel plate reveals a relatively thin, uniform reaction layer, approximately $6\text{ }\mu\text{m}$ thick. The TLPB fabricated fuel plate contains a thicker ($38\text{ }\mu\text{m}$ thick), non-uniform reaction layer that consists of multiple phases visible on the photomicrograph, i.e. regions A, B, C and D. Finally, the FSW fabricated fuel plate shows relatively no reaction layer at all.

Stress-time plots for each sample pull tested are provided in Fig. 4. The dashed line in the figure indicates the approximate limit (20 MPa) of the test rig, above which the crosshead is turned manually employing a wrench until failure of the interface or epoxy occurs. Observation of these plots show a steady increase in stress until catastrophic failure occurs. Samples are pulled normal to the fuel-clad interface. The HIP specimen profile reveals that the sample has bond strength of 60.3 MPa. However, failure of the epoxy occurred before that of the fuel clad interface, so that the actual bond strength, although unrealized in this plot, is greater than 60.3 MPa. An alternative test method, such as a peel test, will be used to quantify the bond strength of samples with strength greater than that of the epoxy. Currently, 60 MPa is established as acceptable bond strength, since no failures after irradiation have been observed with plates fabricated in this manner, at this time. The TLPB specimen has the second highest strength at 15.4 MPa, while the FSW specimen has the lowest bond strength at 6.42 MPa. Also observed from the stress-time plots is the significant difference in stress rate between the TLPB specimen and the HIP and FSW specimens. Since specimens are subjected to a constant rate up to the approximate limit of the test rig, variations in stress rate can provide some initial insight into the integrity of the as-fabrication reaction layer at the fuel-clad interface. The TLPB specimen has an approximate stress rate of $0.04\text{ MPa}\cdot\text{sec}^{-1}$, while the HIP and FSW specimens have approximate stress

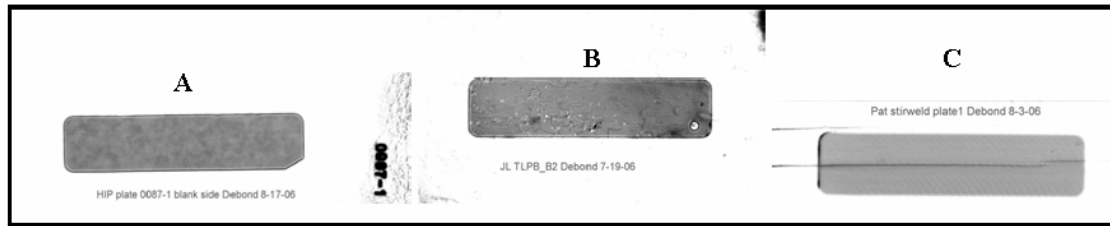


Fig. 2. Ultrasonic testing scans of (A.) HIP fabricated fuel plate, (B.) TLPB fabricated fuel plate and (C.) FSW fabricated fuel plate

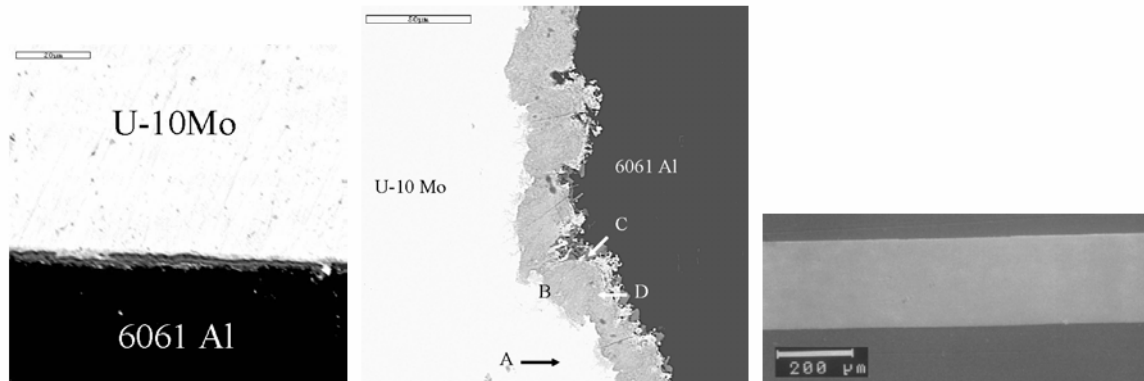


Fig. 3. SEM micrographs of the reaction layer formed from (left) HIP fabrication process, (middle) TLPB fabrication process and (right) FSW fabrication process

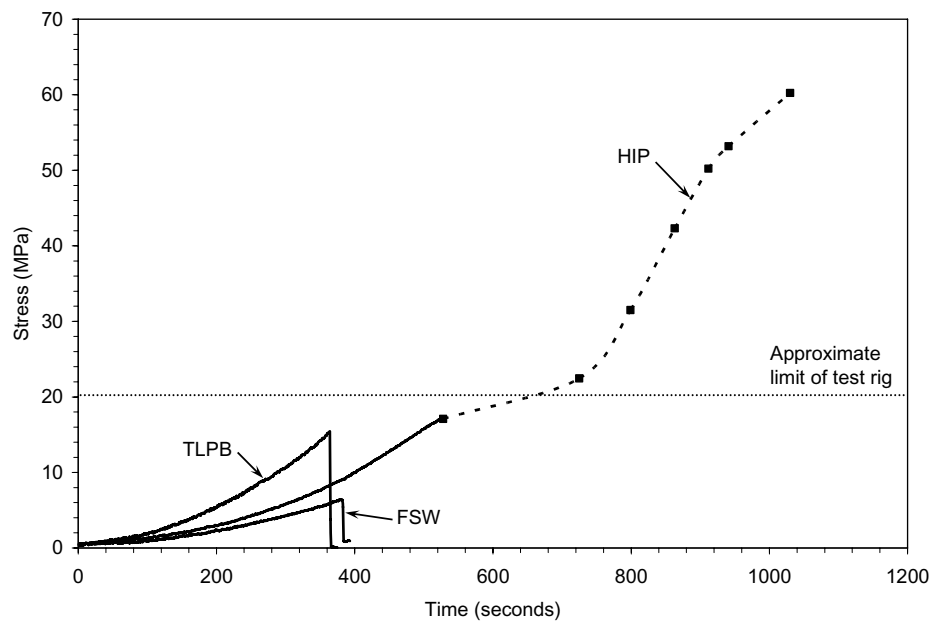


Fig. 4. Stress-time plots for test specimens obtained from pull test fabricated by each [HIP,TLPB,FSW] method

rates of 0.031 and $0.016 \text{ MPa} \cdot \text{sec}^{-1}$, respectively. Hence, a hypothesis may be drawn that the eutectic formation of Al-12Si for the TLPB process behaves in a manner expected of a brittle intermetallic, i.e. high stress rate with sudden failure. In addition, this result appears to suggest that diffusion of aluminium into the foil is significant and results in a thick reaction layer, ultimately lowering the bond strength. Conversely, the FSW specimen has a low stress rate and low bond strength, suggesting that the bond is more mechanical than diffusional. This also seems intuitive since the weld tool has a low thermal conductivity compared to aluminium. Thus, as the FSW process progresses along the plate,

lower loads are applied in order to compensate for the increased temperature, i.e. heat builds up in the plate and is not conducted away from the weld face. Further increases in temperature would ultimately result in increased aluminium plasticity and promote void formation or disturbance of the monolithic fuel foil. The weld surface temperature can additionally be controlled by modifying the weld tool alloy. Increasing the thermal conductivity of the tool face has been shown to significantly increase the bond strength [5]. Finally, the HIP specimen shows the ideal trade-off between fabrication temperature and pressure, promoting diffusion of atoms across the fuel-clad interface resulting in bonding, but not to a degree where the brittle intermetallic nature of the bond dominates the behaviour.

4. Conclusions

The first series of mechanical characterization tests on monolithic fuel plates fabricated by hot isostatic pressing, transient liquid phase bonding and friction stir welding has been carried out. These tests allow a greater understanding of bond strength characteristics and performance prior to irradiation, so that improved correlations between fabrication processes, foil microstructure characteristics and post-irradiation properties can be determined, enhancing the success of the RERTR fuel development campaign. Initial results show that HIPed samples provide the highest bond strength while FSW samples, fabricated in the current manner, provide the lowest bond strength.

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SAFETY ANALYSIS OF A 1-MW POOL-TYPE RESEARCH REACTOR

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ABSTRACT

The following work is performed in view of gathering a better understanding of the behaviour of MTR pool type nuclear research reactors and in particular to present the results of the safety analysis of the 1-MW pool-type research reactor. The process followed passes through the determination of the inherent characteristics of the reactor core and by the analysis of some postulated initiating events considered in the safety analysis report (SAR) of this reactor.

The inherent parameters of interest which are: effective multiplication factor, kinetic parameters, feedback effects, power peaking factors and power defect, are calculated following two major steps. In a first step, cell calculations that consist of detailed models applied to different cell materials identified in the core, are performed by the commonly widely used **WIMS/D4** code. In a second step, core calculations consisting of global core model in two dimensional geometry are performed by **MUDICO-2D** diffusion code.

The accident analysis considered is related to the simulation of some postulated initiating events in two different core configurations of the same reactor. The purpose is to assess the dynamic response of the core and to investigate whether the cladding failure (melting) could be induced. To reach this goal, the in-house coupled thermal-hydraulic-point kinetic **RETRAC-PC** computer code was applied. The results obtained showed that the clad melting threshold is not reached over a wide range of transient situations even when the scram is disabled (unprotected transient). The importance of the inherent safety parameters of a reactor core under different configurations has been emphasized.

The ongoing work is now oriented on the utilisation of **MCNP5** for criticality calculations and validation of the diffusion calculations performed so far and **RELAP5** code for transients and accidents analysis. The validation process of the codes as performed will be also presented in this paper.

1 Introduction :

The degree of consequences for any event postulated in safety analysis of a research reactor is determined by the response characteristics of the reactor system, which is a combination of non nuclear characteristics (like control systems) and inherent or intrinsic characteristics which are strongly nuclear.

In case of severe accidents, the sequence of events could be too rapid for effective control by external means and the course of the response will be firstly fully determined by the inherent characteristics also known as the self-limiting characteristics of the system. Therefore, in case of a power excursion event, the following observable parameters such as the power history shape, the self-limiting behaviour, the peak power, and the released energy should be checked.

In order to perform such a checking, certain intrinsic key parameters namely Prompt neutron lifetime Λ , Delayed neutron fraction β , Reactivity feedback coefficients, and Reactor control

systems, are important to understand the dynamic behaviour of the reactor core during any transient situation [1]. However, most of these key parameters are highly dependant upon the core configuration such as the lattice dimensions, the disposition of the fuel and control assemblies, the control systems, the reflector elements in the grid matrix ...etc; Indeed, all these related configuration's data are used to determine the well known inherent core intrinsic parameters.

The first part of this work concerns the determination of some inherent characteristics of two different configurations of an MTR pool type research reactor: one configuration contains 16-fuel elements (C-16) and the other, 25-fuel elements (C-25) [2] and the second part of the work concern the analysis of some postulated initiating event using the key parameters calculated in the previous step and the importance of such inherent parameters are emphasized by considering the response of the two different core configurations during similar protected postulated initiating events. Furthermore unprotected cases as well as transients without feedback are also considered.

2 Key parameters of the core

2.1. Modelisation of the core:

The determination of the inherent characteristics of any given nuclear system passes throughout the use of some validated computer tools and techniques. For this purpose, a computer code package namely COMPACK-LHW [3, 4] has been developed for MTR research reactor applications. The individual modules of the package and the overall computational strategy along with the techniques of modeling used have been already assessed and qualified in a previous international benchmark problem [5-9].

In the following, the process of determination of the inherent characteristics of a 1 MW MTR (Material Testing reactor) pool type research reactor is outlined.

The computational procedure for the criticality analysis is performed in two main steps and is based mainly on two modules of the computer code system COMPACK-LHW. An adapted cell calculation version of *WIMS-D/4* code [10] provided with an updated 69-group LWR cross sections library and a locally developed diffusion MUDICO-2D computer code [11].

The first step, require a rigorous analysis of the reactor component by an identification of the different types of cells, which may represent physically any region of the core configuration taking into account the fuelled and non-fuelled regions of the core loading, along with the grid plate and the reflector assembly. Thus, in present case, cell calculations are performed using the *WIMS-D/4* code for five (05) different identified and homogenized unit cell types of fuel and non-fuel elements present in the core.

In the second step, global core calculations with the locally two dimensional multigroup diffusion code MUDICO-2D are performed. Effective multiplication factor, reactivity feedback coefficients (Fuel temperature, moderator temperature and density/void coefficients), power peaking factors, power defect are then obtained. Perturbation calculations module of the code is used to evaluate kinetic parameters, like effective delayed neutron fraction and prompt neutron life time. The calculations of macroscopic cross sections for the fuel and non-fuel cells were carried out using 69 energy groups with collapsing procedures version. A 5-energy groups cut-off was adopted (See Table 1) for the following materials identified in the core:

1. Fresh fuel element;
2. H₂O – water reflector;
3. Graphite reflector;
4. Al+H₂O for the external region;
5. Stainless steel + H₂O for the external region in the control blade without absorber.

The broad group cross sections were also generated for different fuel temperatures ranging from 20°C to 150°C and for water temperature ranging from 20°C to 100°C and for change in water density. These cross sections were used to calculate the isothermal feedback reactivity coefficients. In addition, the excess reactivity values for the fresh core considered were also computed at different times of full power operating reactor using the diffusion theory model.

Table 1: Energy boundary for collapsed calculation

Group Energy	Energy range	WIMS - Groups
1	10.0 MeV – 0.821 MeV	1 – 5
2	821 KeV – 5.530 KeV	6 – 15
3	5.530 KeV – 0.625 eV	16 – 45
4	0.625 eV – 0.080 eV	46 – 55
5	0.080 eV – 0 eV	56 – 69

Fuel Element Model:

Fig. 1 shows the dimensions of the standard (SFE) and control fuel elements (CFE). Detailed specifications of the reactor are reported in the following reference [12]. The geometrical fuel model used in the following calculations is represented in Fig.2. Each standard fuel element is made of 19 fuel plates which is represented by two regions :

- The first region of dimensions (6.0 cm × 8.01 cm) represent the active zone of the fuel element;
- The second region with dimensions of (0.95 cm × 8.09 cm) represent two non-fuel (inert) region which consists of an Aluminium side plates and their associated water channels.

Fig. 3 shows the representative unit cell adopted and the associated dimensions used in WIMS-D/4 calculations for cross section generation of the fuel and non-fueled materials present in the reactor. The non-fueled region is represented by an “extra-region” which contains calculated volume fractions of Al and H₂O associated with each fuel plate, together with the water outside the assembly. The thickness of this extra-region was chosen to preserve the water volume fraction in the physical unit cell of each fuel assembly.

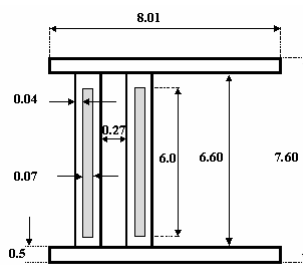


Fig. 1 View of 1 MW MTR Fuel Element (SFE and CFE)
(All dimensions are in cm).

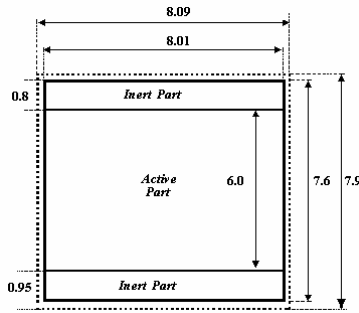


Fig. 2 Model of Standard Fuel Element (SFE).

(All dimensions are in cm).

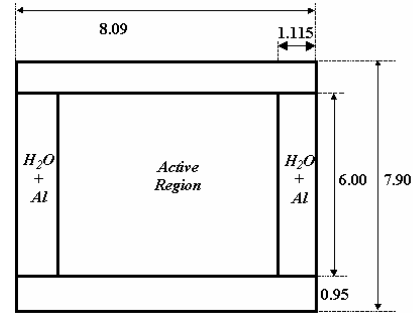


Fig. 3 Model of Control Fuel Element (CFE), without absorber.

(All dimensions are in cm).

CORE model

To evaluate the neutronics and safety parameters of the pool type reactor, and due to the geometrical asymmetry of the core configurations [13], a whole core simulation was performed using the MUDICO-2D two-dimensional multigroup diffusion code. The control rod channels were represented as $H_2O + Al$ zones at both sides of the corresponding fuel elements. A total of 15 and 14 mesh intervals was used in the x and y directions respectively. The axial direction was represented with buckling of $1.64E-3$ corresponding to a chopped cosine axial flux distribution with 8 cm reflector savings ($H_{extr.} = 77.5$ cm). The following parameters were determined for the concerned configuration: Excess reactivity, prompt neutron generation time and effective delayed neutron fraction, fuel and coolant temperature, density feedback coefficient, void coefficient, power defect, and power peaking factors.

2.2 RESULTS

2.2.1. Excess reactivity :

The diffusion calculations for the supercritical core with the buckling of $(1.64 \cdot 10^{-3})$ gave a k_{eff} value of ~ 1.05075 , the experimental value of k_{eff} is 1.04302 [14].

The results should gradually be improved with adequate value of buckling. Thus, some additional and needed sensitivity studies should be performed in the future in order to decrease the difference between calculated and experimental k_{eff} .

2.2.2. Basic kinetic Parameters: (Λ , β)

The perturbation calculation module of the MUDICO-2D [11] code is used to evaluate basic kinetic parameters, namely effective delayed neutron fraction (β_{eff}) and prompt neutron generation time (Λ). The results obtained are as follow:

$$\beta_{eff} = 0.00765$$

$$\Lambda = 46.8 \text{ } \mu \text{ sec}$$

2.2.3. Isothermal reactivity feedback coefficients :

- **Change of fuel temperature only:**

In order to determine the reactivity coefficient for fuel temperature, values of k_{eff} were computed for a fresh fuel at temperature (T_F) of 20°C, 50°C, 75°C, 100°C and 150°C. The deviations of the reactivity to a reference value at 20°C are given by equation (1):

$$\Delta\rho = \rho(T_F) - \rho(20^\circ C) \quad (1)$$

The best polynomial fit of the reactivity variation $\Delta\rho$ (in pcm= 10^{-5}) for the studied case is given in terms of fuel temperature (T_F) by:

$$\Delta\rho \text{ (pcm)} = 45.870 - 2.064 T_F \quad (3)$$

It's interesting to note that for the normal operating range of the reactor, this coefficient remains negative.

- **Change of water temperature only:**

Values of k_{eff} were computed for water temperature (T_w) of 20°C, 30°C, 40°C, 60°C and 80°C. Cross sections libraries at these different temperatures are obtained from WIMS-D4 cell calculations and are used by MUDICO-2D to calculate k_{eff} corresponding to each of these temperatures. The best polynomial fit of the reactivity variation $\Delta\rho$ (in pcm) for the studied case is given by:

$$\Delta\rho \text{ (pcm)} = 159.890 - 7.970 T_w \quad (4)$$

- **Change of Water Density only :**

Values of k_{eff} are computed for water densities of 0.998, 0.996, 0.992, 0.983, 0.969 and 0.958 g/cm³, which correspond to water temperature of 20°C, 30°C, 40°C, 60°C, 80°C and 100°C. The reactivity changes are calculated to reference reactivity at 20°C.

The best polynomial fit representing this variation for the studied core is given by:

$$\Delta\rho \text{ (pcm)} = 125.297 - 5.611 T_w - 0.087 T_w^2 \quad (5)$$

- **Core Void Coefficients:**

To obtain core void coefficients (α_v), the water concentration change was considered in both moderator and extra-region of the unit cell. The reactivity changes are calculated to reference reactivity at 20°C and for void conditions of 5%, 10%, 15% et 20%. This case, which seems to be a realistic one, is represented by the best polynomial fit obtained as:

$$\Delta\rho \text{ (pcm)} = -17.52 - 294.5 \alpha_v - 4.79 \alpha_v^2 \quad (6)$$

The global results obtained for the temperature coefficient of reactivity for the studied critical core configuration over the temperature range of 20 – 80 °C are summarized in table 2. Also is shown in this table, the core void coefficient of reactivity due to change in water density from 0.948 – 0.80 g/cm³ and from 0.998 – 0.948 g/cm³.

- **Power Defect of Reactivity :**

The power defect of reactivity which is an important parameter for reactor operation is defined as the total of all reactivity effects induced by bringing the reactor from cold zero-power conditions to normal operating conditions. Thus, taking into account the different temperature conditions, the loss of reactivity between zero and full power is given by:

$$\Delta\rho_{\text{power}} = (\alpha_{T_w} + \alpha_{D_w}) \overline{\Delta T_w} + \alpha_{T_f} \overline{\Delta T_f} \quad (7)$$

α_{T_w} , α_{D_w} and α_{T_f} are the temperature coefficients of reactivity defined in table 2 and

$\overline{\Delta T_w}$ and $\overline{\Delta T_f}$ are the mean temperature differences in the water and in the fuel from cold zero-power conditions to normal operating conditions.

In the case of interest with an inlet coolant temperature of 40°C and a flow rate of 220 m³/h, the mean temperature difference between zero power and full power is calculated to be around 3.5°C in the water and about 11.1°C in the fuel meat [15]. Table 3 shows the water, fuel and total reactivity

differences between zero and full power computed using isothermal reactivity coefficient calculated above for the temperature range 20-80°C.

Table 2 : Isothermal Reactivity Coefficients

Effect	LAS*
Temperature Range: 20 – 80 °C ($-\Delta\rho/^\circ\text{C} \times 10^{+5}$)	
Fuel Temperature only: α_{Tf}	2.13
Water Temperature only: α_{Tw}	7.93
Water Density only: α_{Dw}	14.52
$\alpha_{Tw} + \alpha_{Dw}$	22.45
Water Density Range: 0.948 – 0.80 g/cm ³ ($-\Delta\rho/\Delta\rho_w$)	
Void Coefficient (%): α_v	41.8
Water Density Range: 0.998 – 0.948 g/cm ³ ($-\Delta\rho/\Delta\rho_w$)	
Void Coefficient (%): α_v	32.8
* LAS: Laboratoire des Analyses de Sûreté, CRNA.	

- **Power Peaking Factors :**

The peaking factor, being the most important safety parameter, was also evaluated. This factor is defined as the product of three factors: radial x local x axial, peaking factors.

The radial power peaking factor is defined as the ratio of the average midplane power in a specified element to the average midplane power in the core.

The local power peaking factor is defined as the ratio of the maximum midplane power to the average midplane power in the specified element.

The axial power peaking factor is defined as the ratio of the maximum axial power to the average axial power in the specified element.

Two-dimensional diffusion calculations were performed by MUDICO-2D for all the fuel elements present in the core. Since the maximum wall or clad temperature is the limiting operational parameter of the core, the peaking factor of greatest importance for steady state operation is the maximum of this product (radial x local). Thus, the results are as follow:

$$F_{rad} = 1.32$$

$$F_{loc} = 1.09$$

The axial power factor was determined, assuming a chopped cosine flux distribution along the channel. Then, we obtain the following value:

$$F_{axi} = 1.32$$

The total peaking factor is the product of these three parameters ($F_{TOT} = F_a \times F_r \times F_l$) :

$$F_{TOT} \text{ (conf.-16)} = 1.90$$

This parameter is very important as well as, for steady state operation and for extreme hypothetical accidents, since it determine the maximum energy released in the hottest fuel element and consequently give us the corresponding peak fuel temperature and the peak cladding temperature which is as outlined before an important limitation safety parameter.

The total peaking factor calculated implies that under extreme conditions, the maximum of heat generation in the hottest fuel element is almost two times higher than that generated in the average fuel element of the core.

Table 3 : Power Defect of Reactivity

$$\overline{\Delta T_w} = 3.6 \text{ }^{\circ}\text{C}$$

$$\overline{\Delta T_f} = 11.1 \text{ }^{\circ}\text{C}$$

<i>Effect</i>	LAS
Fuel Temperature (pcm)	23.6
Water Temperature + Density (pcm)	80.8
$\Delta\rho_{\text{power}}$ (pcm)	104.5
β_{eff} (%)	0.765
$\Delta\rho_{\text{power}}$, ϕ (cents)	13.7

3 Accident analysis

3.1. Description of scenarios

Two different types of initiating events are considered as follows

Case 1: Rapid (kinetic) transient initiated by a positive Reactivity Induced Accident (RIA); this postulated event is initiated by an hypothetic control rod withdrawal at start-up when the core power level was at 1W and the coolant inlet temperature at 40°C;

During RIA events a non-equilibrium between the generated and the removed heat takes place. Consequently, the core component's temperature rise up and their integrity could be damaged in absence of appropriate response of the control systems. The transients under consideration herein are initiated by a super prompt insertion of a positive reactivity of \$1.5 in 0.5 seconds. This transient occurs at reactor startup and the reactor power level was assumed to be 1W; the circulating pumps are assumed to assure full upward cooling flow.

In case of the analysis of protected transients, the safety setting overpower trip point is set to 1.2 MW i.e. the scram system is acted when the power level reach the setting trip point above. This shutdown mechanism introduces a negative reactivity of -\$8 in 0.5 sec with a response delay time of 0.025 s.

Case 2: Relatively slow transient cases related to Loss of core coolant Flow Accident (LOFA) as a consequence of main cooling pump failure when the reactor was operating at its nominal power level and the inlet temperature was 40°C.

Both protected (with scram) and unprotected transients (without scram) are considered.

During a loss of flow accident or LOFA, we can expect a core heat up due to malfunction of the cooling system. To simulate this postulated initiating event, the flow decrease is modeled by an exponential decrease function ($\exp(-t/T)$). The period T is set equal to 1 sec in case of a Fast LOFA (FLOFA) and equal to 25 s in case of a Slow LOFA (SLOFA).

These transients are initiated when the reactor was operating at its nominal power level of 1.2 MW with cooling system allowing full downward cooling flow.

For the protected cases, the reactor scrams is set by a trip point that enable a shutdown reactivity insertion when the flow rate is reduced by 15% of its nominal value; the scram is acted with a delay time of 0.2s.

The basic kinetics parameters and isothermal reactivity feedback coefficients for these two configurations (C25 and C16) of the equilibrium cores summarized in Table 4 [2].

Table 4 : Kinetic and reactivity feedback parameters

<i>Parameter</i>	25 Fuel elements	16 Fuel elements
<i>Effective Delayed neutron fraction β_{eff}</i>	0.00769	0.00765
<i>Generation Time Λ, μs</i>	46.80	46.80
<i>Mean Void reactivity coefficient, \$ / %</i>	0.8128	0.512
<i>Mean Fuel Temperature feedback coeff \$/°C</i>	$2.773 \cdot 10^{-3}$	$2.58 \cdot 10^{-3}$
<i>Mean Coolant Temperature feedback coeff \$/°C</i>	$3.084 \cdot 10^{-2}$	$3.226 \cdot 10^{-3}$

3.2. Reactor Core modeling and computer codes:

In order to simulate the reactor dynamics under the aforementioned initiating events, the reactor is modelled by one unit cell that represents any heated region of the core. Three regions compose this representative cell: fuel-clad and the associated coolant channel.

In this model, a cosine axial power distribution is assumed with a cosine shape set to 1.311; the overall peak/average as calculated above is applied for the hot channel.

3.3. Computer codes used:

- The RETRAC-PC Code [16] is used to perform a numerical simulation of the considered transients. The core power history is derived through the resolution of the point kinetic equations coupled to the thermal hydraulic conservation equations of mass, momentum and energy conservation. In this code feedback contribution due to coolant and fuel temperature changes and void are considered with the possibility to assign separate reactivity coefficients for each component (see Table 4).

- The PARET Code [17] which is similar to RETRAC-PC, is a channel code based on a coupling of kinetic equations, heat conduction and hydrodynamic equations with adjusted feedback. This code has been extensively compared with the SPERT experiments [18]. These comparisons have shown quite good agreement for a wide range of transients including melting of the clad. The PARET code has also been used by the RERTR (Reduced Enrichment for Research and Test Reactors) Program for the safety evaluation of many research reactors candidates for fuel reduced enrichment [19]. According to the good performances shown by PARET, the RETRAC-PC results are compared to PARET ones in order to appreciate the relevant results.

3.4. Results :

3.4.1. RIA Transients:

a. Protected transients:

The power, the clad maximum temperatures and the coolant outlet temperatures histories as obtained by RETRAC-PC and PARET are presented in the fig.s 4 and 5 respectively. The compensated reactivity is also provided in fig. 4. The main transient parameters are summarized in Table 5 for both configurations.

As shown on fig. 4, the power behavior exhibits, as could be expected, an exponential excursion. The trip setting point by power ($P_{trip}=1.2$ MW) is reached few milliseconds after the initiation of the accident. However, the power increase continues during the lag time before effective action of the scram system (due to time response delay of 25ms); this scram system delay response enable the power to reaches almost 50 MW as shown in Table 5. But, in case of unprotected transients, the power reaches very high levels as could be observed in Table 6.

We can notice quite similar responses of both configurations for protected transients especially during the lag time before peak power is reached (see Fig. 4). Indeed, in this kind of transients, the dynamic response of the core is governed essentially by the prompt neutrons and the inherent

feedback mechanisms do not act significantly during this first stage of the transient. This is confirmed by the delay response of the thermal hydraulic mechanisms that induces a beginning of temperatures rises later after the power trip point (see Fig. 4 for example). As a consequence, the feedback effect begins to act slightly just after the trip point and contribute to reduce the power excursion period. Their effect becomes insignificant after the reactor scram. However, we should mention here that some voids production is predicted to occur around the peak power time occurrence. This phenomenon contributed strongly in reducing the power excursion; the void production begins when the clad temperature exceeds 126°C which correspond to the onset of nucleate boiling temperature threshold as calculated by the Bergles-Rhosenow correlation.

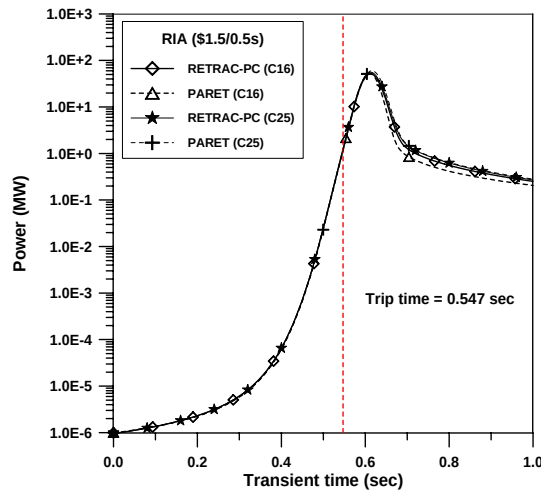


Fig. 4 : Power behavior during RIA transient (C16 and C25 configurations)

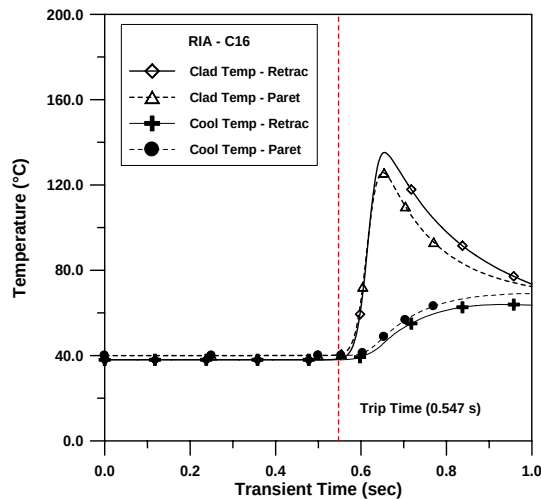


Fig. 5 : Cladding and Coolant Temperature behavior during RIA transient (C16 configuration)

As mentioned above, the PARET code has been also used to simulate the core behavior during protected RIA. As outlined on fig. 4 and Table 5, the two codes showed in general similar results. The observed discrepancies, as well explained in a previous work [20], are essentially due to the feedback model (essentially the void effect) and to the amount of heat directly transferred to coolant without convection.

Table 5. Transient Parameter for protected RIA

Configuration Parameter	C 16		C25	
	RETRAC-PC	PARET	RETRAC-PC	PARET
Trip time (sec)	0.547	0.547	0.546	0.547
Peak Power (MW)	51.676 (0.612)	51.756 (0.611)	57.325 (0.613)	60.087 (0.616)
Energy at Peak power (MJ)	1.41	1.37	1.59	1.67
Peak Fuel Temperature (°C)	157.60 (0.640)	150.60 (0.635)	146.86 (0.644)	142.73 (0.643)
Peak Clad Temperature (°C)	135.22 (0.654)	125.83 (0.651)	131.62 (0.661)	120.66 (0.659)
Coolant Temperature (°C)	63.93 (0.928)	63.47 (0.985)	63.97 (0.999)	70.60 (1.170)

(time of occurrence in seconds)

b. Unprotected transients:

In order to emphasize the strong effect of feedback mechanism and to outline the key parameters that govern the inherent dynamic response of the cores, similar RIA as above but without scram are considered. The C-16 clad temperature and the power excursion under these conditions (without scram) are shown on fig. 6 and 7 with comparison to the similar parameters corresponding to case with scram.

In case of no scram, the power continues its excursion until it is stopped by the inherent feedback's mechanisms of the reactor. For this case, the power reaches a maximum of 150 MW and quenches due to the strong feedback effect even the scram system is disabled during the course of the accident. No damage of the cladding occurs according to the results shown in Fig. 7. The maximum temperature reached (less than 186°C) is far below the melting point of aluminium that is 600 °C or even from the safety margin of 450°C fixed due to swelling of Aluminium.

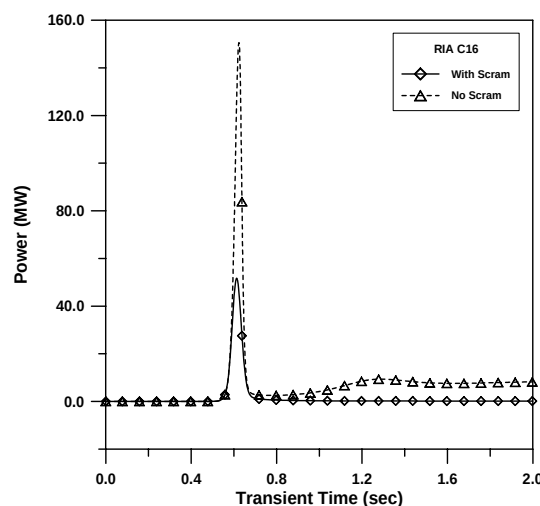


Fig. 6 : Power behavior during RIA transient with and without scram (C16 configuration)

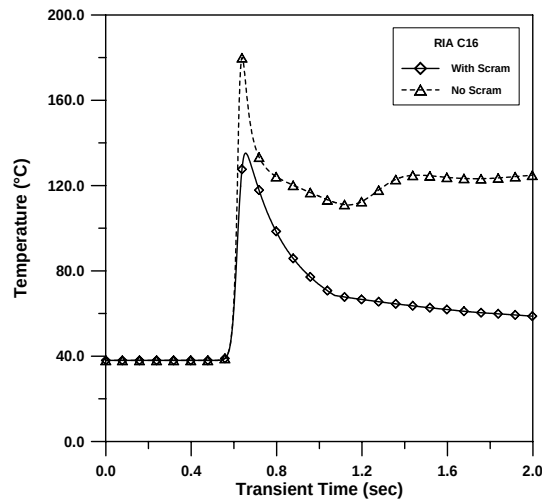


Fig. 7 : Cladding temperature behavior during RIA transient with and without scram (C16 configuration)

Furthermore, the strong influence of the feedback mechanism on the stability of this research reactor are emphasized in the following by investigating the reactor response under protected RIA (with scram) but in absence of any feedback mechanisms. The results obtained in this case are summarized in Table 6 and the power excursion is reproduced on fig. 8 with a comparison to the reference case (in 3.4.1-a). The power excursion for both cases is similar until the trip point (1.2MW) but after this point and under the new assumptions (no feedback), the power continue to increase and reaches 132 MW (instead of 52Mw in previous case) before the control system could stop the excursion. These results emphasize what has been stated in previous sections concerning the strong feedback effects on the excursion runaway before any effective scram. This exercise confirms also the importance of these inherent core parameters for the safety of the considered reactor (self shut-down).

Also, in fig. 8, one can observe a failure of the code after 0.67 seconds because of the limitation of the thermal hydraulic model that can not handle, at this stage of development, the transition from single to two phase convection.

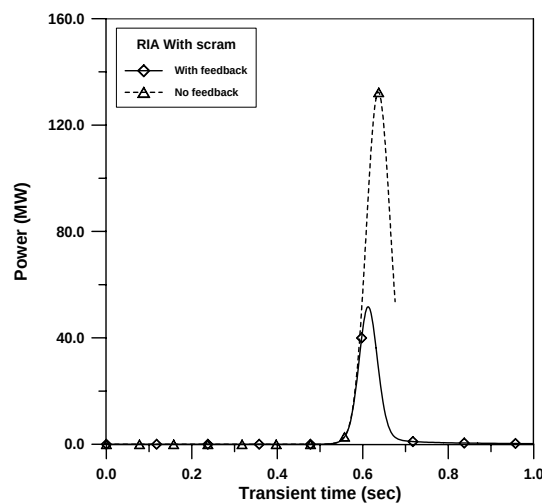


Fig. 8 : C16 response to RIA accident with and without feedback (scram enabled)

Table 6. Transient Parameter for unprotected RIA

Configuration Parameter	C 16		C25	
	No scram	No feedback	No scram	No feedback
Trip time (sec)	0.547	0.547	0.546	0.546
Peak Power (MW)	150.64 (0.623)	132.41 (0.636)	182.18 (0.624)	135.65 (0.636)
Energy at Peak power (MJ)	3.69	4.71	4.55	4.80
Peak Fuel Temperature (°C)	308.04 (0.634)	324.91 (0.653)	294.52 (0.635)	284.36 (0.655)
Peak Clad Temperature (°C)	180.01 (0.638)	185.96 (0.658)	181.01 (0.640)	180.93 (0.660)
Coolant Temperature (°C)	106.16 (0.849)	104.58 (0.672)*	104.64 (0.979)	109.17 (0.700)*

*(t)time of occurrence in seconds * time of end of calculations*

Conclusion :

Through the present study, the overall strategy followed for the analysis of reactivity accidents in an MTR research reactor was outlined. This process passes through two major steps which are the characterization of the reactor core (key parameters) and the simulations of the postulated events.

However, in order to evaluate the uncertainties of the criticality calculations, an additional assessment of this procedure and the computer tools used will be handled, in a near future, by comparison with analytical results from Monte-Carlo calculations.

From the analysis of the postulated events considered here, it appears that the clad melting threshold is not reached over a wide range of transient situations even when the scram is disabled (unprotected transient).

The importance of the inherent safety parameters of a reactor core under different configurations has been emphasized and the investigation has confirmed the strong influence of the inherent parameters (or key parameters) on the core dynamics under transient or accidental situations. However more detailed simulations should be considered in order to confirm the aforementioned conclusions. This could be done by applying best estimate computational tools that are able to take into account the multidimensional kinetics and thermal-hydraulic effects [22].

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OSCAR-3 MCNP INTERFACE (OSMINT5) VERIFICATION AND VALIDATION

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ABSTRACT

OSMINT5 has been developed to set up a SAFARI-1 MCNP model by detecting the whole core configuration and isotopic inventory from OSCAR-3. Its flexibility allows various models, enabling OSCAR-3 comparisons, OSCAR-3 coupling approximation studies, application to OSCAR-4 (reserved), and an exact model, to be used for predicting neutronic parameters and for design purposes.

Using MCNP-Os3 (MCNP – OSMINT – option 3) to compare with OSCAR-3 has shown a difference on average with OSCAR-3 of 1.316% when the bank position is inside an axial node and .997% when the bank position coincides with an axial mesh. The average difference with measurements is -1.765%; however, MCNP-Os5 (MCNP - OSMINT - option 5) comparisons have shown good agreement (0.10813% on average) with measurements in predicting the critical bank position.

Along within the validation process of MCNP models generated with OSMINT, this paper presents a complete analysis of the global core reactivity variation associated with the modelling, and of the modifications accordingly implemented.

1. Introduction

OSCAR-3 is a nodal diffusion code, used to perform SAFARI-1 reactor core follow and safety analysis. As a result of the approximations of nodal diffusion theory, it was necessary to build an automatic tool to run in parallel with OSCAR-3 to provide comparisons, and by using core snap shots to generate MCNP models as input for measurement comparisons and design purposes.

The first stage of the project was to generate MCNP inputs at different time snap shots with isotopics treated explicitly and with approximate core models (MCNP – Os3) [1], with results referred to in the conclusions. The second stage, which is presented in this paper, is to set up detailed modeling, with the rest of isotopes not available in the current MCNP nuclear data libraries (Sm-148), and detailed geometry (MCNP –Os35 and MCNP – Os5) and to analyze the approximate MCNP-Os3 results; these could be taken into account to build the detailed geometry model MCNP-Os5. A complete verification of OSMINT as a tool capable of fulfilling its purposes was necessitated, and future requirements were formulated. The validation of the MCNP model generated and of the modeling options to represent the SAFARI-1 core, is presented in this paper, as well as the validation of MCNP input generated with different modeling options.

The next stage will be to generate fission products as spatial and burnup dependent lumped fission product (LFP) cross sections, resulting in an MCNP model that is more representative of a burned core; in addition, integrated codes will be developed to perform the parallel core follow, fuel management and the optimization of ⁹⁹Mo production.

The ability of OSMINT to set up a valid MCNP model of the SAFARI-1 core, for any burnup case as supplied from OSCAR-3 was verified. Once the generated MCNP-SAFARI-1 core model had been verified, the validation process was performed in order to establish the range of uncertainties for predicting the core neutronic parameters.

However, in order to validate the MCNP model for SAFARI-1, MCNP calculation results are compared with validated results from OSCAR-3 to establish relative uncertainty margins, besides ongoing comparisons with measurements.

2. OSMINT Verification

The verification process of OSMINT mechanics includes the following steps:

1. Isotopic transfer.
2. Check the cell volumes and material masses as calculated with OSMINT and MCNP.
3. Visual check of the core layouts by means of horizontal and vertical cuts.
4. Check that all outputs of OSMINT (1.e., MCNP inputs) are executable without any user modifications, in order to meet the requirements of a future upgrade to a parallel Monte Carlo core follow and fuel management system.

3. Additions to OSMINT.061t

The following capabilities and modifications were made to OSMINT test version [1].061t during the verification process:

3.I) In order to verify the mechanics of OSMINT, some additions were implemented, with the result that the verification process will be part of every OSMINT execution.

3.II) As SAFARI-1 is in the process of converting from HEU to LEU, the ability to handle LEU as well was added to OSMINT.

Note: during each execution of OSMINT, a complete verification of geometry volumes, isotopic IDs, isotopic masses per assembly and in the whole core is performed and reflected in OSMINT output file.

3.III) The ability to handle Mo target plate irradiation rigs with different combinations of fuel and/or dummy aluminium plates. In the OSMINT.i input file, the user enters the number of rigs, and for each rig, the combination of fuel/dummy-Al plates.

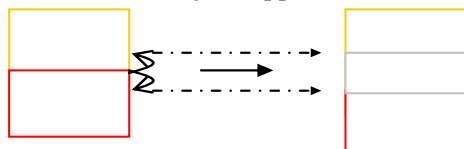
Shown below is part of the OSMINT input file for target plate specification, where the user enters 1 for fuel and 0 for dummy-Al per rig.

```
Molly Control -
Number of Molly devices          : 6
Insert 1/0 fuel/dummy-Al
combination for each Molly
B8 : 1 0 1 0 0 1 1
C3 : 1 1 0 0 0 1 1
D8 : 1 0 0 1 0 1 1
E3 : 1 1 0 0 1 1 1
F8 : 1 0 0 0 0 1 1
G3 : 1 1 0 1 1 1 1
```

The volume fraction is calculated according to the user-specified fuel/dummy combinations.

3.IV) To set up MCNP models representing the following cases:

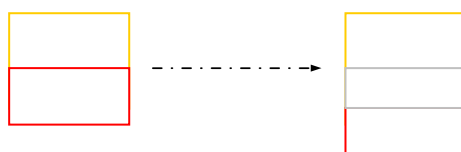
1. ver = 3 (MCNP-Os3), model representing OSCAR-3, with: a. Exact active core length 59.37cm. b. Control follower is not represented beneath the core active length (in reflector). c. Bank positions provided from OSCAR-3 handled. d. Cd section presented above the core in cases where the bank position is more than 0 notches. e. Generic Mo target plate modelling (inside 4 nodes) with the thimble extending above and below the rig up to active core length only.
2. ver = 35 (MCNP-Os35), same as above, but splits the bank positions provided to form the coupling part in order to study its approximation in OSCAR-3, as shown below;



OSCAR-3

MCNP

3. ver = 5 (MCNP-Os5), Exact core model as above, but: a. Control follower extending beneath the core active length. b. Exact models of rig and thimble. c. Bank positions provided are handled as shown below, taking into account that the setting exact models from SAFARI-1 are at the lower tip of the Cd.



OSCAR-3

MCNP

4. Validation

4.1 Case I (MCNP-Os3)

Case I shows the results of a recent core during the core follow calculations with OSCAR-3. After comparing results from OSCAR-3, MCNP and measured values, a comparison also took place with the same core and different control bank settings, in order to study the effect of control rod modelling. The plate combinations in the Mo production rig were obtained from the Mo Irradiation Planning [4] documentation, where the case study is performed for the core at 15.993 days into the cycle.

4.2. Results and Discussion

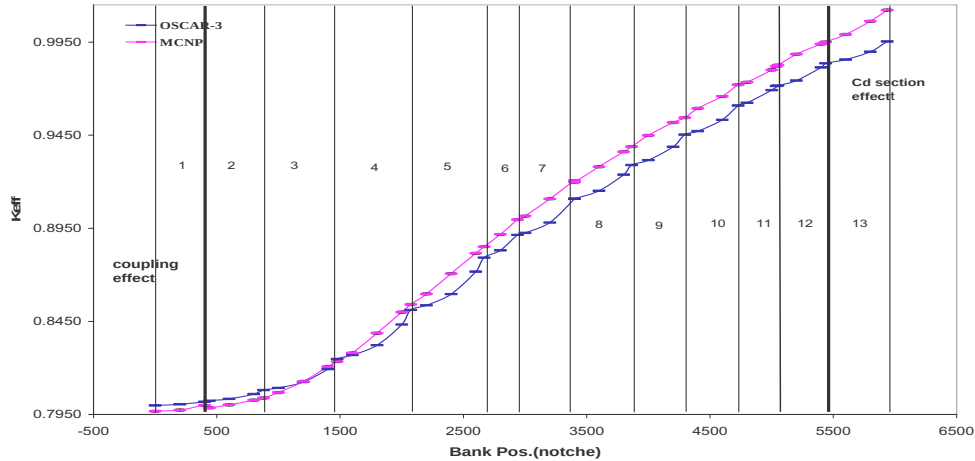


Fig 2. Core reactivity with banks withdrawal position, vertical lines represent the axial nodes tip of nodes numbered from 1 to 13, bottom to top respectively (MCNP-Os3)

Criticality comparisons have been performed, where the difference between OSCAR-3 and MCNP was -1.132%, and the uncertainty in predicting the criticality for cores loaded with Mo target plates is -2.92% for OSCAR-3 and -1.765 for MCNP.

For the core loaded with target plates, criticality calculations have been performed for different bank withdrawal positions, in order to analyze the effect of introducing more or less of the Cd section and fuel follower parts.

Figure 2 shows the global core reactivity with a fine mesh bank withdrawal positions, where positions are taken at the tip of each axial nodes, and few points within each node. OSCAR-3 approximates the cross section presentation of the Cd section and fuel follower inside the node with volume averaged cross sections, resulting in the effect shown by the blue line (cusping effect), whereas the MCNP results show a typical S-curve behavior.

Referring to Figure 3 - the percentage difference between OSCAR-3 and MCNP reactivates - and starting from banks fully inserted, the OSCAR-3 MCNP difference is seen to be 0.392%, the closest representing model of MCNP to OSCAR-3. Withdrawing the banks up to 400 notches (first three points on the curve), the trend is typical. From 400 to the next point, the reactivity jumps down where 0.4 cm Cd is withdrawn, part of Al coupling was replaced by water, and a small amount of fuel is present at the bottom of the core; the global effect to decrease the reactivity by ~100 pcm. Withdrawing the Banks up to 1000 notches, MCNP is still lower and the difference goes from 0.392% (zero withdrawn) to 0.3% (1000 notches); with peaks at the node tips of 0.47% and 0.52%, means 0.1% occurred due to Cd not present above the core. The error increases as the MCNP modelling introduces the first source of deviation from OSCAR-3 model of having Cd section above the core. This decreases the Cd section participating in absorbing neutrons and increases the global core reactivity. This trend will affect the difference with nearly constant value of ~ 1.671% inside the nodes and 1.5564% at the node tips, where 1.671% and 1.5564% are errors ascribed to not modelling the Cd section above the core when the bank position is greater than zero.

Bank withdrawal positions greater than 5552 notches were excluded from the above analysis as it is the second reason of difference between OSCAR-3 and MCNP. The Al-coupling piece is represented in MCNP and approximated with a shift of Cd section and fuel follower to the middle of the coupling

in OSCAR-3; this gives the reason for a jump in core reactivity at the last three points on the MCNP S-curve. Here only fuel follower and coupling are being added and the Cd is no-existent after 5552 notches, as the coupling is 3.85 cm and the MCNP modeling takes the bank position provided as being at the tip of the Cd. Accordingly;

1. MCNP model (MCNP-Os3) representing OSCAR-3 will be modified to introduce Cd above the core when the withdrawal position is greater than fully inserted, and the coupling will be replaced with Cd and fuel follower as in OSCAR-3.
2. A formula will be derived to position the control element with the bank position provided as:
 - a. The zero position (fully inserted) of Cd lower tip will be the active core height, subtracting the total Cd section height, and
 - b. The top (fully withdrawn) will be the active core height plus the coupling height, to reflect OSCAR-4 and physical modeling.

Regarding the graph, vertical lines representing the axial nodes tip, and between the solid bold vertical lines, show that the MCNP S-curve follows a typical trend, while it deviates outside the bold lines according to bank withdrawal position handling (coupling) on the left side, and Cd section not yet modeled above the core on the right.

Concluding the above analysis, it is to be predicted that after the aforementioned OSMINT modifications, the global core reactivity difference, with all cases and all bank withdrawal positions would be ~400 pcm between the OSCAR-3 and MCNP models, with MCNP being lower.

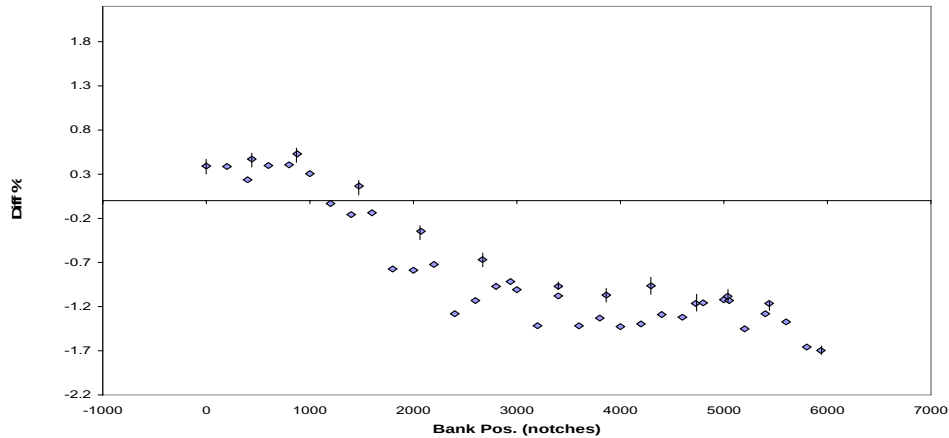


Fig 3. OSCAR-3 vs MCNP reactivity difference with bank withdrawal position
Slashed points are at the node tips, while without slashes are inside the node

4.3 Case I revised (MCNP-Os3r, MCNP-Os5)

Referring to section 3.IV and the analysis of Figure 2, it was clearly necessary to modify the OSMINT MCNP modeling (MCNP-Os3). After modifications to OSMINT following the aforementioned Case I, and referring to section 3.IV of additions to the OSMINT test version, Case I was revised to compare OSCAR-3 with OSMINT ver=3 (MCNP-Os3). The results and discussion are shown below, and figures 4-6 show OSCAR-3 and measurements comparisons with MCNP-Os3, MCNP-Os35 and MCNP-Os5 modeling (see additions to OSMINT, section 3.IV).

4.4 Results and Discussion Revised

Figure 5 shows a difference of 1.316% between OSCAR-3 MCNP-Os3 when the bank position is inside an axial node and 0.997% when the bank position at an axial node tips, on average.

While OSCAR-3, MCNP-Os3r and MCNP-Os5 were compared with measured critical settings, the vertical lines on the graph Figure 4 shows the magnitude of difference between each predicted critical setting and the measured setting as unknown: 5498, 5055 and 5050 for OSCAR-3, MCNP-Os3r and MCNP-Os5 and measurement, respectively.

The exact MCNP-Os5 model showed a good agreement with measured settings, namely 0.10813% average error in predicting the critical pattern (+0.05 cm control rod position).

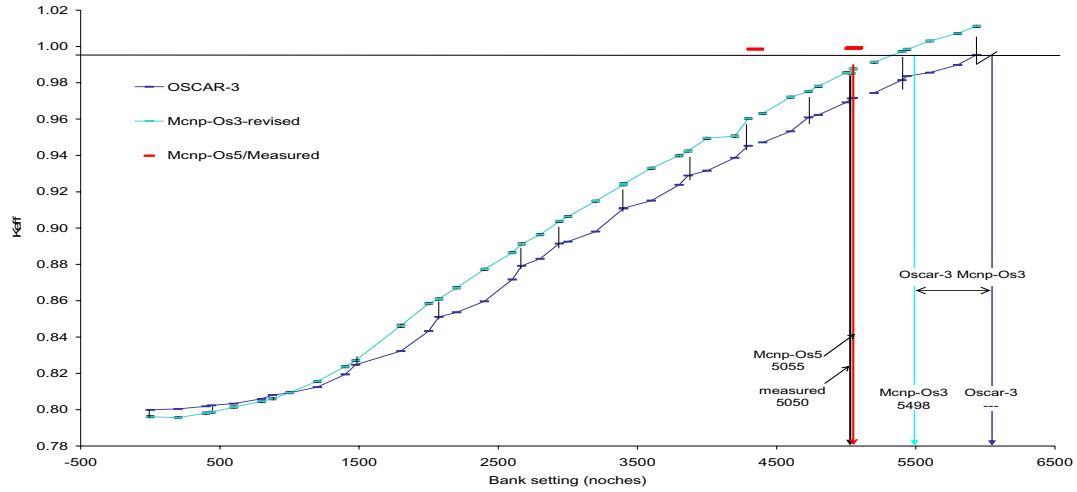


Fig 4. A comparison between OSCAR-3, MCNP-Os3r and MCNP-Os5

In Figure 5, the marked points (in circles) show the effect of coupling at the core bottom and top, and the graph shows a shift of the OSCAR-3 MCNP-Os3r results with some differences in magnitude due to the modifications made in the MCNP-Os3r model during the comparison process. The behavior is about constant after the bank position enters the fifth node from the bottom, the difference being 1.471% on average; the fluctuations of differences inside the nodes is according to the behavior of OSCAR-3 curve, while from first to the fourth node, the behavior is being investigated.

Bank Pos.(noche)	Mcnp-Os5	Std	Error%
4346	0.99854	0.00055	-0.14621
5050	0.99884	0.00056	-0.11613
5058	0.99938	0.00055	-0.06204

Table 1. MCNP-Os5 comparison with measured critical settings

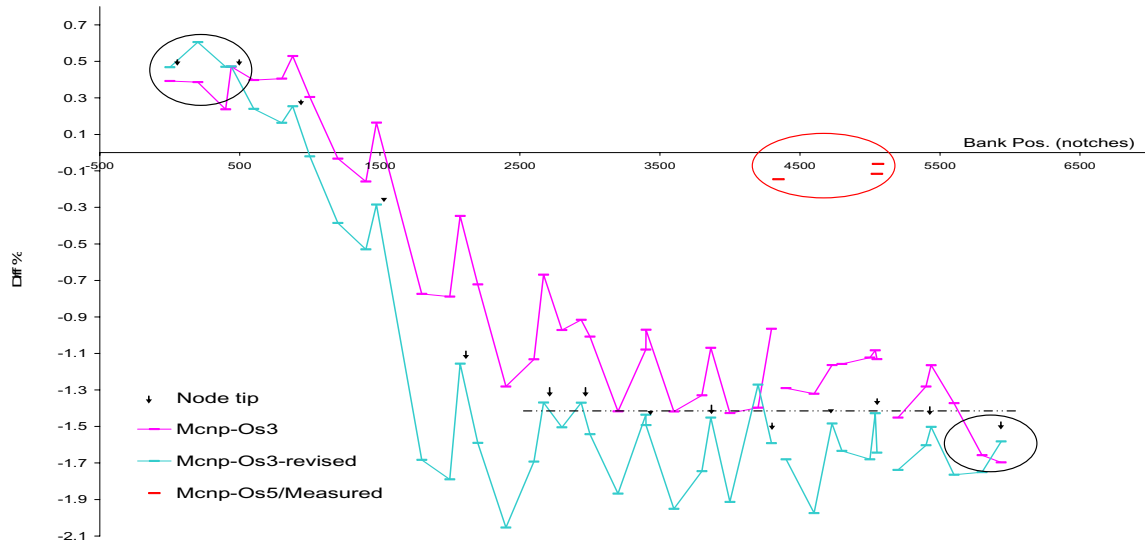


Fig 5. MCNP-Os3 and MCNP-Os3r revised difference relative to OSCAR-3

45. Case III (MCNP-Os5)

Validation against local neutronic parameters has been performed, via foils activation [7], showed a close agreement within the experimental error at the 1/3rd core top and fluctuated overestimations at the bottom 2/3rd, on average, which agree with the abovementioned coupling approximation analysis, and due to isotopic mass distributions prediction, measurements uncertainties, and lack of MCNP LFPs.

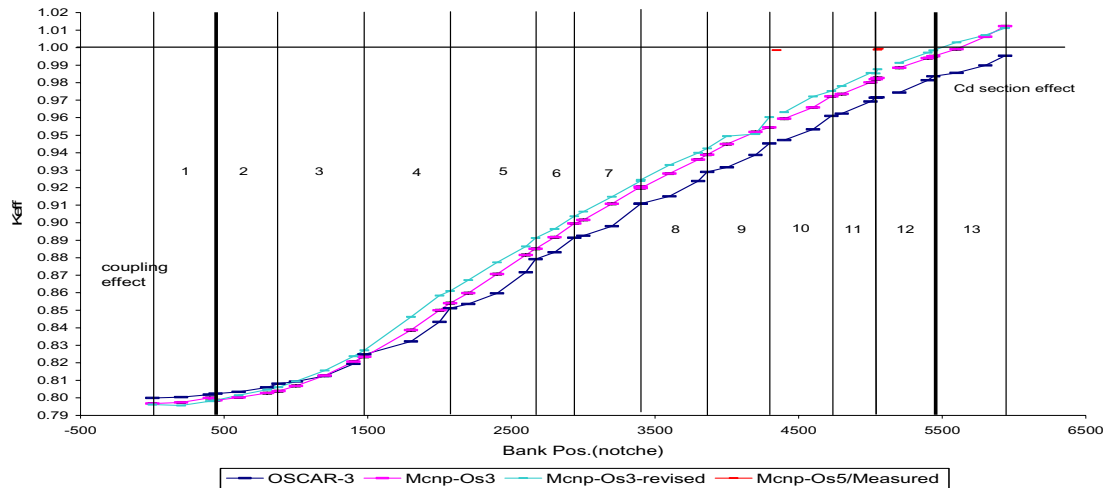


Fig 6. OSCAR-3 MCNP-Os3 and MCNP-Os3r revised comparison. The behavior of Os3 and Os3r are close but for near the bottom and top of the core, and significant differences are being investigated

5. Conclusions

The ability of OSMINT to set up a SAFARI-1 MCNP core model for any case, supplied from OSCAR-3, has been verified, the fuel/control/Mo rigs/peripherals core positions, the isotopic contents per node per plate per assembly and the overall core assembly dimensions. The generated MCNP inputs are ready to run without any modifications, provided the user follows the right process to build the model.

According to the modifications made, the global core reactivity difference between OSCAR-3 and MCNP has been changed, from a wide range of differences, with variation with the control rod withdrawal position and core state, to a very narrow range of max. -1.49% and min. -0.7% regardless of the control rod withdrawal position and/or core state, with no Mo target plates present. The difference for cores loaded with target plates was subjected to farther calculational analysis with fine mesh rod banks movements: the average difference at the node tips is $\sim -0.625\%$, and inside the node is -0.833% .

According to the above considerations and during the validation process, MCNP follows the same trend of uncertainty predicting measured values (K_{eff}), where the uncertainty in criticality calculations with OSCAR-3 is $\sim -8.2\%$ and MCNP-Os3 is $\sim -6.99\%$ for cores without target plates, and -2.92% and -1.765% respectively for cores loaded with target plates. The sources of uncertainty are being studied and a decision will be made whether to accept the MCNP model with its underestimation of the global core reactivity.

The target plate modelling capability added to OSMINT, did not affect the OSCAR MCNP agreement trends; the target plate modelling, core position, isotopic contents, and the fuel/dummy-Al plate combinations have been verified.

6. References

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CONVERSION OF TAJOURA CRITICAL FACILITY FROM HEU TO LEU

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ABSTRACT

The Tajoura Critical Facility has been converted from 80% enriched uranium IRT-2M (HEU) fuel to 19.7% enriched uranium IRT-4M (LEU) fuel. using the Russian manufactured IRT-4M fuel assemblies (FA), the compact core containing ten 3-tube and six 4-tube IRT-2M FA has been replaced by ten 6-tube and six 8-tube IRT-4M FA. The loading of the LEU to the critical facility was completed on January 2006. During the approach to criticality and after criticality was reached many measurements were performed. In this paper calculated parameters are presented. Two codes (WIMS and CITATION) are used for the calculations.

1. INTRODUCTION

The Tajoura Research Center operates both a 10 MW pool type research reactor and a Critical Facility. The fuel of the Critical Facility^{1,2} has been converted from 80% enriched IRT-2M IRT-2M fuel to a less than 20% enriched uranium IRT-4M³ fuel. The IRT-4M 8-tube and 6-tube fuel assemblies were loaded to the Critical Facility in January 2006. It is very important to determine all the neutronic parameters of the facility with the new LEU fuel. Most of these parameters such as the system excess reactivity, the shut down margin, and the reactivity worth of the control rods have been reported in another paper⁷. Cross for core diagram for the critical facility is shown in Figure 1. In this paper a detailed investigation and comparison of the flux distribution for both HEU and LEU cores are presented. Experimental measurements are underway and they will be compared to these calculated in the near future.

2. DATA AND RESULTS

The results presented in this paper are obtained using the diffusion theory (CITATION^{4, 5} code). For the CITATION model the fuel assemblies are homogenized and the cross sections for the homogenized regions are generated using the WIMS³ code. Two types of cells are used for the models in WIMS⁵; first the straight parts of the elements are taken as slabs while the central ring of the 8-tube and the curved corners of the fuel assembly are modeled using the annulus option. Results of the

comparisons between calculated LEU and HEU flux for a 4 watt critical state and for control rods worth are presented below.

The critical facility is modeled with sixteen fuel assemblies (ten 6-tubes and six 8-tubes IRT-4M FA or ten 3-tubes and six 4-tubes IRT-2M FA). The core has eight shim rods (KC rods in figure 1), two safety rods (AZ rods in figure 1), and one regulating rod (RR in figure 1). The core is surrounded by a Be reflector (20 moveable blocks, and a fixed Be reflector) where experiments can be loaded. If no experiments are present, Be plugs are inserted inside the Be blocks.

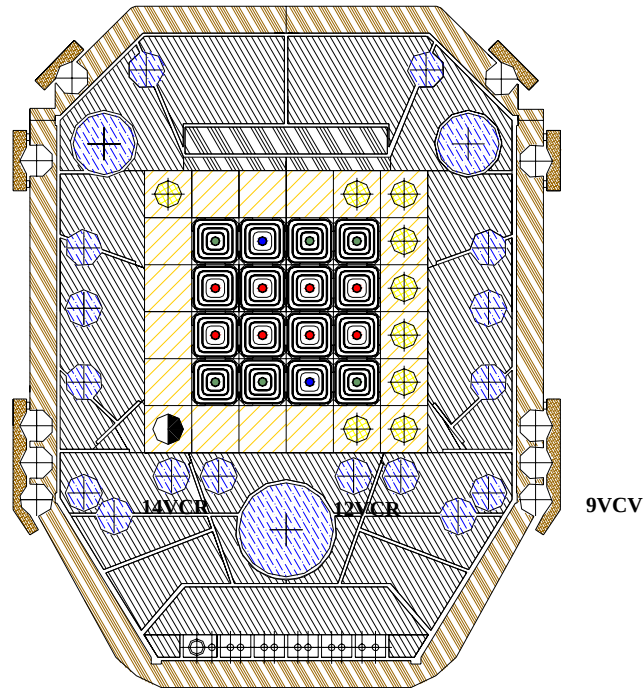


Figure 1 Cross Section of the Critical Facility Core Diagram

The results presented in this paper are obtained using the diffusion theory (CITATION^{4, 5} code). For the CITATION model the fuel assemblies are homogenized and the cross sections for the homogenized regions are generated using the WIMS³ code. Two types of cells are used for the models in WIMS⁵; first the straight parts of the elements are taken as slabs while the central ring of the 8-tube and the curved corners of the fuel assembly are modeled using the annulus option. Results of the comparisons between calculated LEU and HEU flux for a 4 watt critical state and for control rods worth are presented below.

The critical facility is modeled with sixteen fuel assemblies (ten 6-tubes and six 8-tubes IRT-4M FA or ten 3-tubes and six 4-tubes IRT-2M FA). The core has eight shim rods (KC rods in figure 1), two safety rods (AZ rods in figure 1), and one regulating rod (RR in figure 1). The core is surrounded by a Be reflector (20 moveable blocks, and a fixed Be reflector) where experiments can be loaded. If no experiments are present, Be plugs are inserted inside the Be blocks.

2.1 Integral Reactivity Worth of Control Rods

Comparisons between LEU and HEU for the 16 FA core with rods fully withdrawn, all shim rods and regulating rods fully inserted (shutdown state), and all rods (shim, regulating and safety fully inserted) are presented in Table 1. The results in this table show the differences between the two cores.

The reactivity worth of each of the control rods was measured with the 16 FA core configuration with all Be plugs in place. The Be plugs are positioned inside the halls of the Be blocks in the reflector when no experiments are installed. Comparison of the calculated and measured results using CITATION is provided in Table 2 below. The calculated results are essentially within about 10% of the measured results, which are considered to be in a good agreement given that there are some differences in way of calculating and measuring the reactivity worth of the control rods.

Case	Reactivity (\$)	
	CITATION (LEU)	CITATION (HEU)
All Rods Out	21.08	17.73
All Shim and Regulating Rods Inserted	-5.05	-10.81
All Rods Inserted	-11.08	-16.797

Table 1 Reactivity for 16 FA Core with Different Sets of Rods Fully Inserted

Rod	LEU Core Results (β_{eff})		HEU Core Results (β_{eff})	
	Calculated	Experimental Results	Calculated	Experimental Results
Safety Rod AZ1	3.31	3.0	3.16	-
Safety Rod AZ2	2.87	2.5	2.79	-
Regulating R	0.58	0.62	0.446	0.41
Shim Rod KC-1	3.12	3.25	3.02	2.94
Shim Rod KC-2	2.96	3.1	2.87	3.23
Shim Rod KC-3	3.55	3.8	3.78	3.02
Shim Rod KC-4	3.36	3.7	3.61	4.50
Shim Rod KC-5	3.48	3.8	3.73	4.39
Shim Rod KC-6	3.31	3.65	3.57	4.11
Shim Rod KC-7	3.0	3.25	2.94	3.26
Shim Rod KC-8	2.77	2.98	2.74	3.22

Table 2 Control Rods Total Worth for 16 Fuel Assemblies Compact LEU and HEU Cores
(With all Be plugs in place)

Finally, calculations were carried out to determine the fast and thermal fluxes in different positions in the core and reflector regions. Six locations are chosen for the comparison of fast and thermal flux distributions of the LEU and HEU cores. One locations is inside the core namely in the fuel cells (2-2), two locations are in the removable beryllium reflector namely cells (4-1) and (1-2), two locations are in the stationary beryllium reflector namely cells (12VCR) and (14VCR), and one location is in the aluminium vessel in cell 9VCV. Figures 2 to 7 show the axial flux distributions in these chosen locations. Figure 8 shows the thermal flux across cell (2-2) and figure 9 shows the radial flux distributions in the x-direction of the core passing through the center of cell (2-2). The comparison between HEU core and LEU core results is clear.

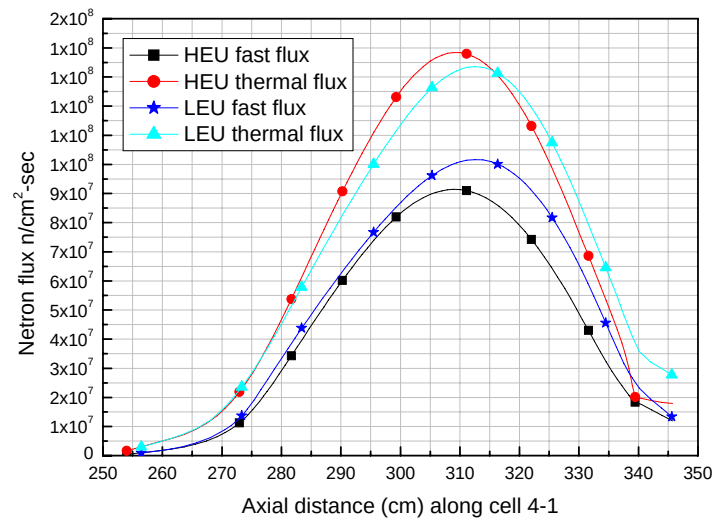


Figure 2 Calculated LEU and HEU fast and thermal flux distributions along cell 4-1

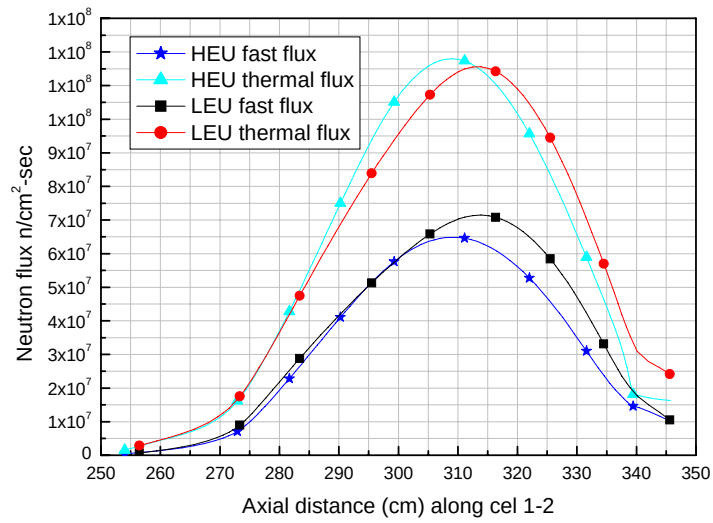


Figure 3 Calculated LEU and HEU fast and thermal flux distributions along cell 1-2

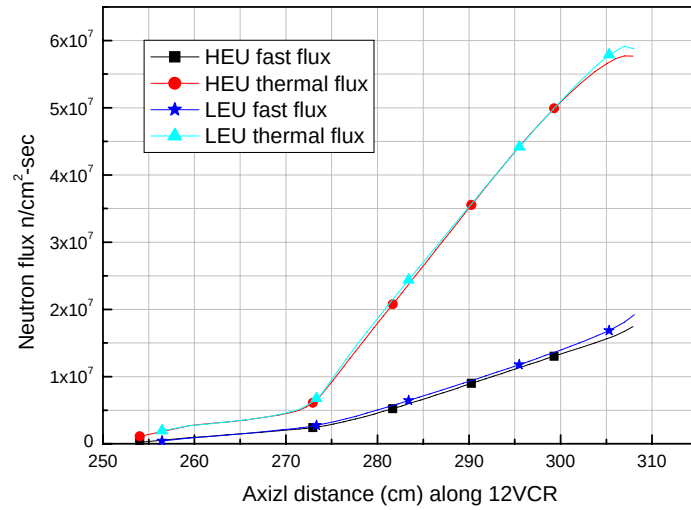


Figure 4 Calculated LEU and HEU fast and thermal flux distributions along 12VCR

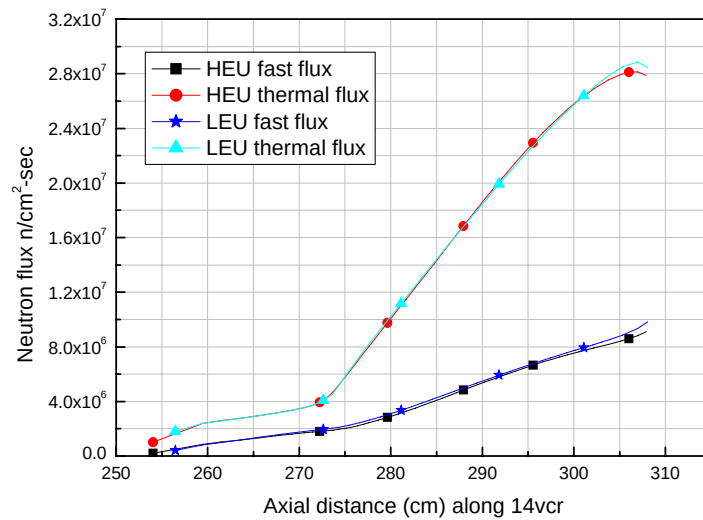


Figure 5 Calculated LEU and HEU fast and thermal flux distributions along 14VCR

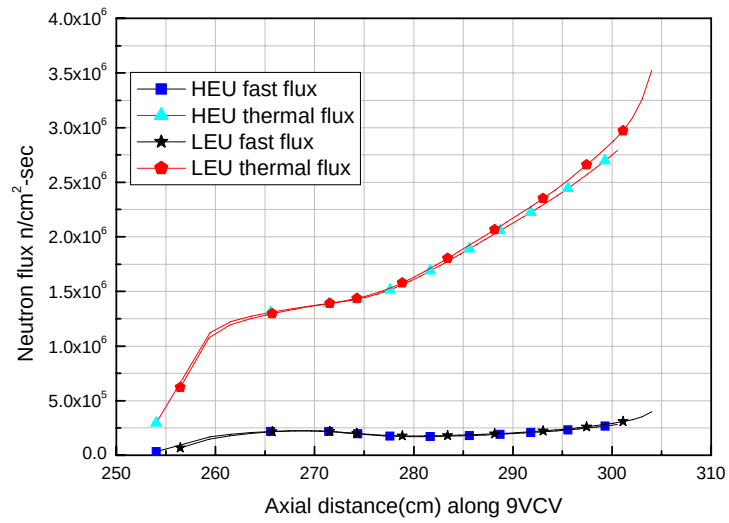


Figure 6 Calculated LEU and HEU fast and thermal flux distributions along 9VCV

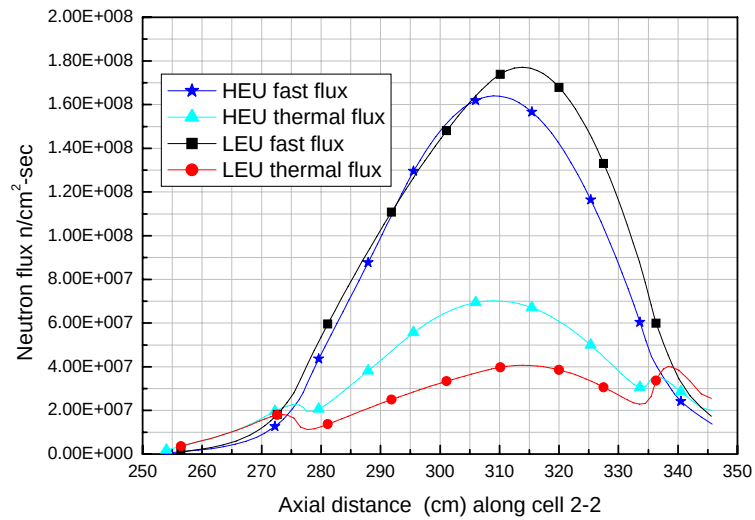


Figure 7 Calculated LEU and HEU fast and thermal flux distributions along cell 2-2

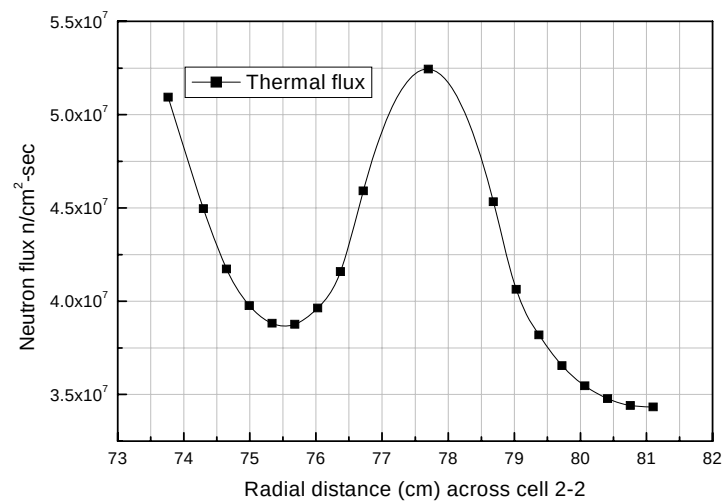


Figure 8 Calculated LEU thermal flux distribution in the x-direction across cell 2-2

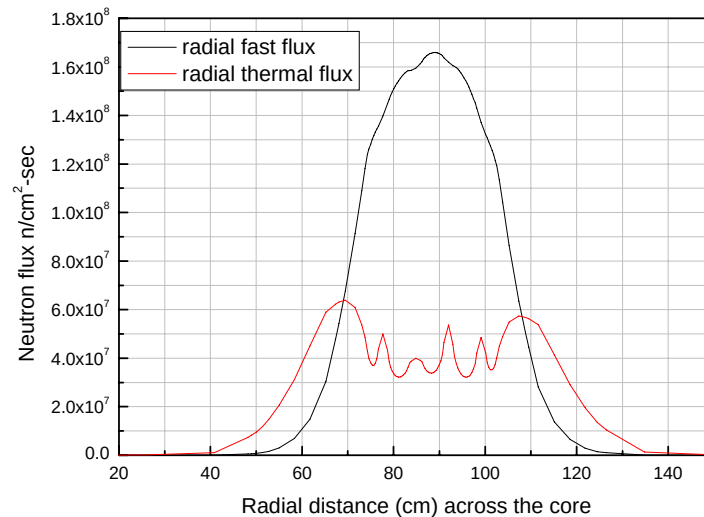


Figure 9 Calculated LEU fast and thermal flux distributions across the core through cell 2-2

3. SUMMARY

The Tajoura critical facility was recently converted from HEU to LEU (Jan/2006) using the Russian-made IRT-4M instead of IRT-2M FA. During the conversion process many calculations and measurements were carried out, and comparisons between measurements and calculated results using diffusion theory (CITATION code) methods were presented. As shown in the figures the fast flux increases and the thermal flux decreases for the LEU core in the fuel cells and in the reflector near to the fuel region. In the reflector away from the fuel and in the vessel both thermal and fast fluxes remain almost the same for both LEU and HEU cores.

4. REFERENCES

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EFFECTS OF Ti IN THE UMo/Al SYSTEM : PRELIMINARY RESULTS

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ABSTRACT

Among the elements which could be added to the UMo fissile compound or to the aluminium alloy in order to prevent or reduce the UMo/Al interaction, Ti is a promising one since its affinity for Al is very important. The first results dealing with the effect of Ti on the UMo/Al reactivity are exposed in this poster. The metallurgical aspects of manufacturing the ternary compounds U-Mo-Ti by arc melting are described. Annealing treatments performed at 450°C on diffusion couple are presented and discussed.

1. Introduction

The UMo dispersion fuel is developed to convert the MTR cores currently working with UAl_x and U_3Si_2 , with a more dense fuel able to reach the exigencies of the non-proliferation nuclear treaty with no or low modification of initial design. This treaty promotes pacific nuclear issues and gives a value of 20% as the upper allowed value for U^{235} enrichment (Low Enriched Uranium) [1]. In operating conditions, the interaction between UMo and its matrix Al results in a degradation of its performances leading sometimes to the failure of the fuel element. The understanding of the UMo/Al interactions is then a key stage for the research and the development of a UMo-based Low Enriched Uranium (LEU) fuel, behaving in a satisfactory way under irradiation [2-3-4-5].

According to the work performed by Park et al. [6], a promising way to reduce the reactivity of the UMo/Al alloys consists in adding element having both strong interactions (e.g. formation of intermetallics) with the matrix and weak ones with the UMo alloys. This assumption indicates that, as calculated by Y.S. Kim et al [7], among the different elements, tetravalent ones like Ti, Si or even Zr could be interesting candidates. However, zirconium reduces the γ phase stability by forming Mo_2Zr . This is the reason why we decided to choose Ti as a stabilizing element.

The binary U-Ti diagram (figure 1) shows the existence of a high temperature extended cubic solid solution between these two elements.

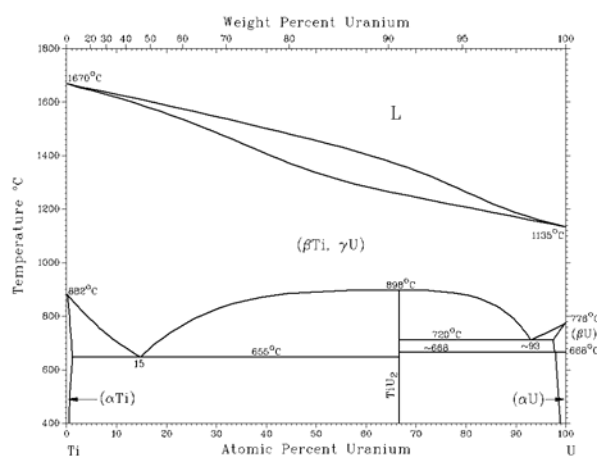


Fig. 1: U-Ti Phase diagram [8]

Compared with the TTT (Time, Temperature, Transformation) curve of the binary UMo8 (wt%) alloy, adding up to 1 %wt of titanium to the alloy does not change the ability of the cubic phase to be retained at room temperature by quenching (figure 2).

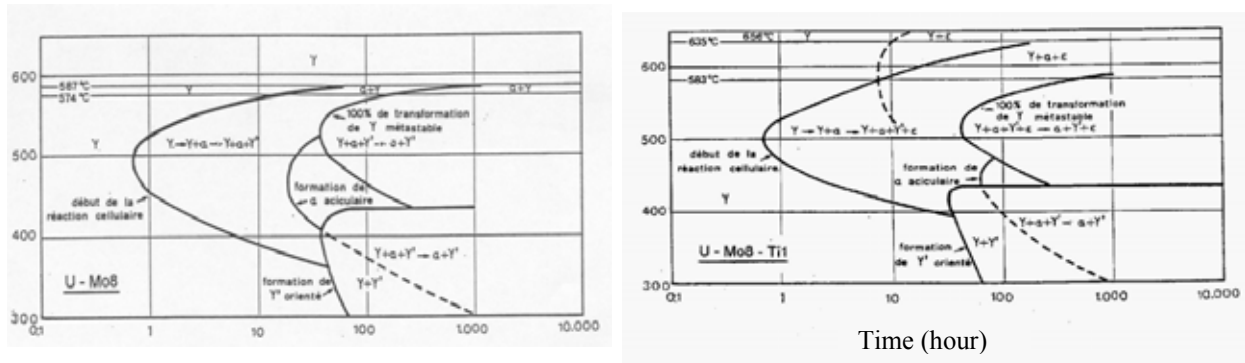


Fig. 2: U-Mo8 and U-Mo8-Ti1 TTT curves

The Ti-Al phase diagram (figure 3) illustrates the great chemical affinity between these two elements. Indeed, several intermetallic compounds can be formed, depending on the Ti/Al ratio. As previously discussed, this affinity could reduce the fuel-matrix interaction process.

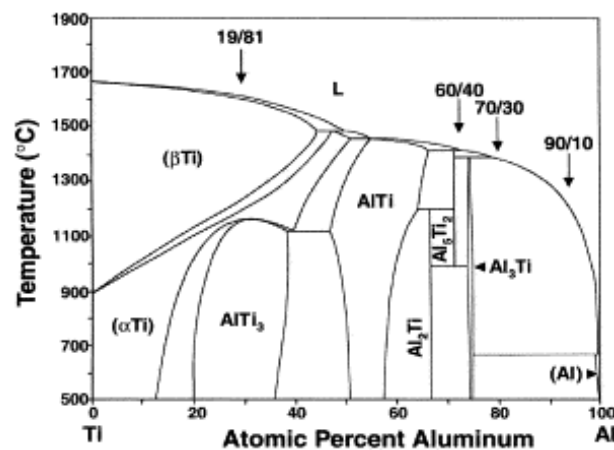


Fig. 3: Al-Ti phase diagram [9]

2. Experimental

2.1. UMoTi alloys

Small arc melted ingots (about 5 mm diameter) of UMo6Ti1 (wt%) were prepared sampling raw UMo5 alloy (11.6 at %) supplied by AREVA-CERCA and wires of pure Ti and Mo.

The metals, weighed in the proportions required for preparing a sample of approximately 1 gram, were molten by arc melting (Mini Arc Melter MAM-1, Bühler Company). To prevent the oxidation during the elaboration, high purity argon is used and Ti oxygen getter is melted before UMoTi alloys. A good homogenisation is expected to be reached carrying out several melting on each ingot, while turning over them each time. No mass loss has been detected during the elaboration process.

Aluminium grade 1050A was used in diffusion experiments : diffusion couples were prepared with samples of approximately $2 \times 5 \times 5 \text{ mm}^3$, cut out from Al foil and a half of a U-Mo-Ti ingot. These two parts were preliminary polished and etched in nitric acid in order to eliminate any trace of oxide. Couples were maintained in a specific device loading samples up to 50 MPa approximately.

2.2. Thermal ageing

Following kinetic considerations given by the TTT curve of the alloys (cf. figure 2, despite a slight different composition of the alloy), thermal annealings were performed on UMo6Ti1/Al diffusion couples under a reducing atmosphere ($\text{Ar} + 5\% \text{H}_2$), at 450°C for 2 hours in order to suppress/limit the influence of an eventual eutectoid transformation.

2.3. Characterization

The raw materials, as well as the aged samples were characterized with a classical XRD device (D8 Advanced Brüker) and by optical and electronic microscopies (FEG-SEM-Philips XL30 / EDAX EDS detector).

3. Results and discussion

3.1. Raw U-Mo-Ti alloys characterization

The XRD pattern (fig. 4) of an UMo6Ti1 ingot after its elaboration (without annealing) shows that the γ -phase of uranium is retained at room temperature and that the sample seems to be homogeneous (no secondary phase detected).

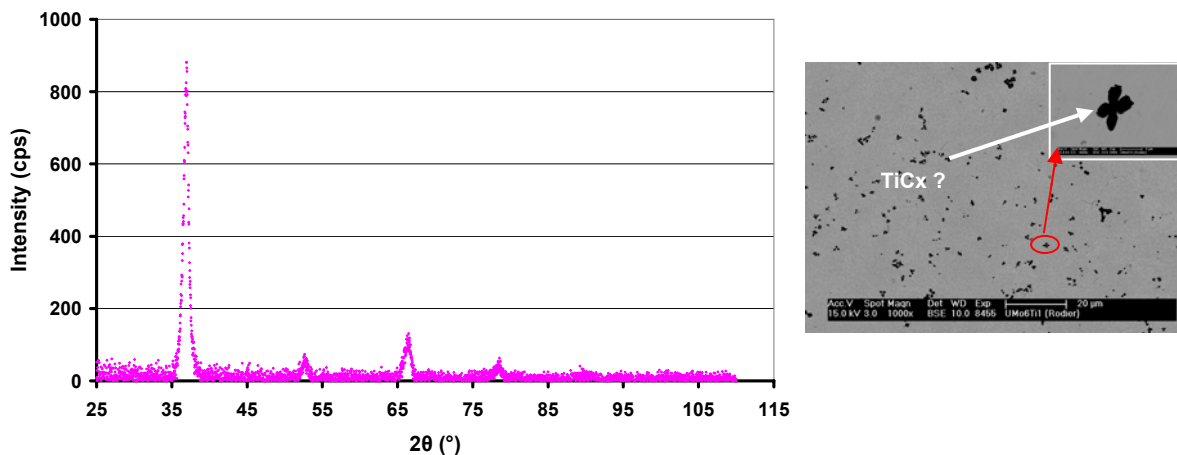


Fig. 4: X-ray diffraction pattern and SEM picture of a raw arc melted UMo6Ti1 ingot

Optical microscopy and SEM examinations showed however that at least a part of the titanium seems to be precipitated, probably as a carbide phase (as starlike carbide in fig. 4). In fact, some UC carbides were initially present in the UMo5 alloy ($[\text{C}]$ content ≈ 1000 ppm). One can assume that this carbon should precipitate with Ti, as titanium carbides are at least as stable as uranium ones.

This partial precipitation of titanium leads probably to a smaller available concentration of this element in solid solution in the UMo alloy. We plan to check if an homogenization annealing will permit a resolution of the titanium containing precipitates. If not, a more pure UMo alloy should be used for the arc melting elaborations.

3.2. UMoTi/Al interaction (annealing treatment at 450°C , 2 hours)

Microstructures of an UMo6Ti/Al couple after a diffusion test at 450°C were examined by SEM. The interaction layer (which is not present along the whole interface of the couple) grows inhomogeneously at this temperature. Some UMo6Ti residual islets are also present in the interaction layer (IL). This tends to indicate that a preferred propagation is involved in the interaction process.

The IL's thickness varies from a few micrometers to $150 \mu\text{m}$, in the same specimen (see figure 5a). This thickness variation could be due either to a residual oxide thin layer at the fuel-Al interface (especially for some parts where no IL is observed) and/or a poor contact between the samples.

Compositional analyses were performed by energy dispersive X-Ray spectroscopy (EDS). Figure 5b shows different zones in the IL with variable compositions of U, Mo and Al, e.g. quoted 1 and 2 on the micrograph. The analyses gave the following results (wt% element):

Wt%	U	Mo	Al
Zone1	68	2	30
Zone2	60	4	36

Table 1

In these two areas, titanium was not detected (its concentration is probably below the detection threshold).

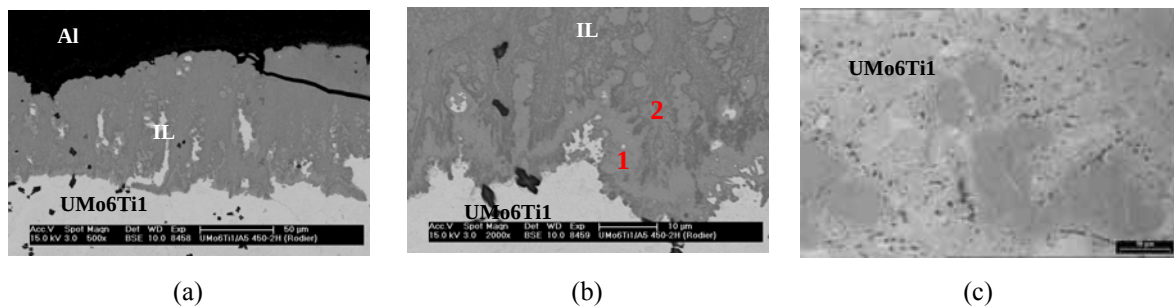


Fig. 5 : scanning electron micrographs of (a) interaction layer, (b) Al side of the IL and (c) two different compositions in IL

It is interesting to note that, in some areas of the UMo6Ti1 alloy (see figure 5c), a lamellar structure, which is typical of an eutectoid structure, is observed. Further XRD characterizations will be useful.

A compositional profile of U, Mo, Al and Ti concentrations was performed on the interaction layer (fig. 6). We are not able to distinguish different zones in the IL by this measurement method but the variations of concentrations in the first half of the IL are more important than in the part near the aluminium matrix. This phenomenon is explained by the residual islets of UMo6Ti which are present in the IL. A light depletion of Mo in the first part of the IL is also slightly visible on this profile. Next to the aluminium matrix, concentrations seems to stabilize around approximately 60%U, 35%Al, 5% of Mo in weight. The titanium concentration seems to stay around 0.5% all along the profile, no variation of its concentration can be detected. It should be due, as previously explained, to its low concentration level in the UMo (1 wt%) in comparison with the detection threshold.

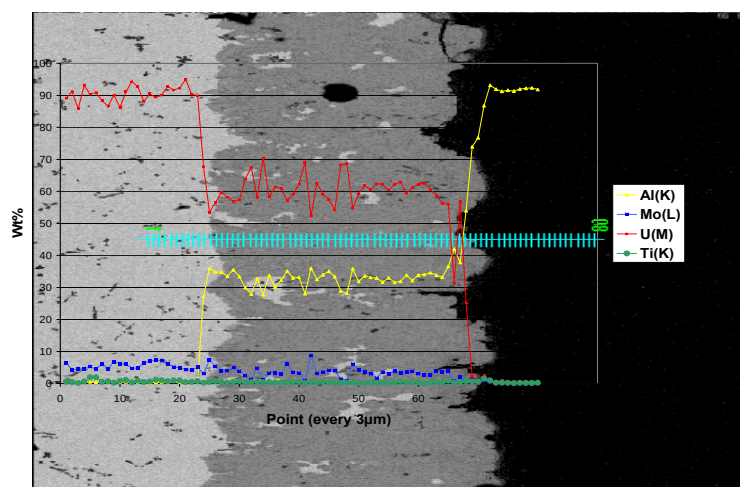


Fig. 6: Compositional profile through the IL in an UMo6Ti/Al diffusion couple annealed at 450°C for 2hours

All along the interface between the IL and Al, there is an accumulation of porosities which could be the result of the Kirkendall effects : to minimize the gradient of concentration between the IL and the Al phase, the aluminium flow generates, in the opposite direction, a flow of vacancies which will agglomerate close to the interface. This accumulation of vacancies forms porosities and then can induce a decohesion between the IL and Al.

4. Conclusions

The addition of Ti in the UMo/Al system is currently studied. We expose the first results, dealing with some metallurgical features of the UMoTi alloys, secondly with the effects of some weight percent of this element in the UMo alloy on the reactivity with aluminium.

Thermodynamic calculations and equilibrium data (i.e. available phase diagrams) show that Ti, acting like silicon, could reduce the interaction between UMo and Al. For this, one may consider its great binding affinity with Al leading to the possible formation of strong bonding intermetallic compounds acting like possible diffusion barriers.

Summarizing, let's point out the first and main results :

- UMo6Ti1 ingots (a few mm diameter, 1 g approx.) have been arc melted elaborated . As cast, the XRD pattern shows only the γ -UMo phase on the ingots. This allows to conclude on the ability of this ternary alloy to be retained in the γ form at room temperature. Nevertheless, Small precipitates of titanium compounds are present in the alloy bulk. Annealing homogenization treatments are planned in our future work.
- The first characterizations of a diffusion couple performed 2 hours at 450°C between 1050A Al alloy and UMo6Ti show:
 - o The existence of a cellular decomposition within the U-Mo-Ti alloy, leading to the formation of a classical eutectoid microstructure.
 - o The existence of a 150 μm max. thick interaction layer, which is neither regular nor homogenous. BSE imaging and EDS measurement shows the existence of a chemical heterogeneities, with a local Mo depletion near the UMoTi alloy, compared to the Mo content in the center of the IL.
 - o The existence of porosities between the Al and the IL, which could be an evidence of the Kirkendall effects, i.e. the consequence of the coalescence of vacancies flowing in the opposite direction to the transport of Al. Due to its mobility and the existence of a chemical potential gradient within the layer, the Al diffusion, possibly more important than that of U or Mo, seems to drive the growth process. Maybe, the diffusion short-circuit of the freshly formed eutectoid microstructure are involved.

This work is still in progress and will focus on :

- The elaboration and characterization of different composition of U-Mo-Ti alloys (with more than 1%wt of Ti), and homogenisation annealing will be performed on every sample to share out Ti.
- New diffusion couples will be tested at an higher temperature and with other aluminium alloys,

Further characterization will be performed.

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FULL CONVERSION OF MATERIALS AND NUCLEAR FUEL – RESEARCH&TEST – TRIGA SSR 14 MW

M. CIOCANESCU, M. PREDA, C. IORGULIS

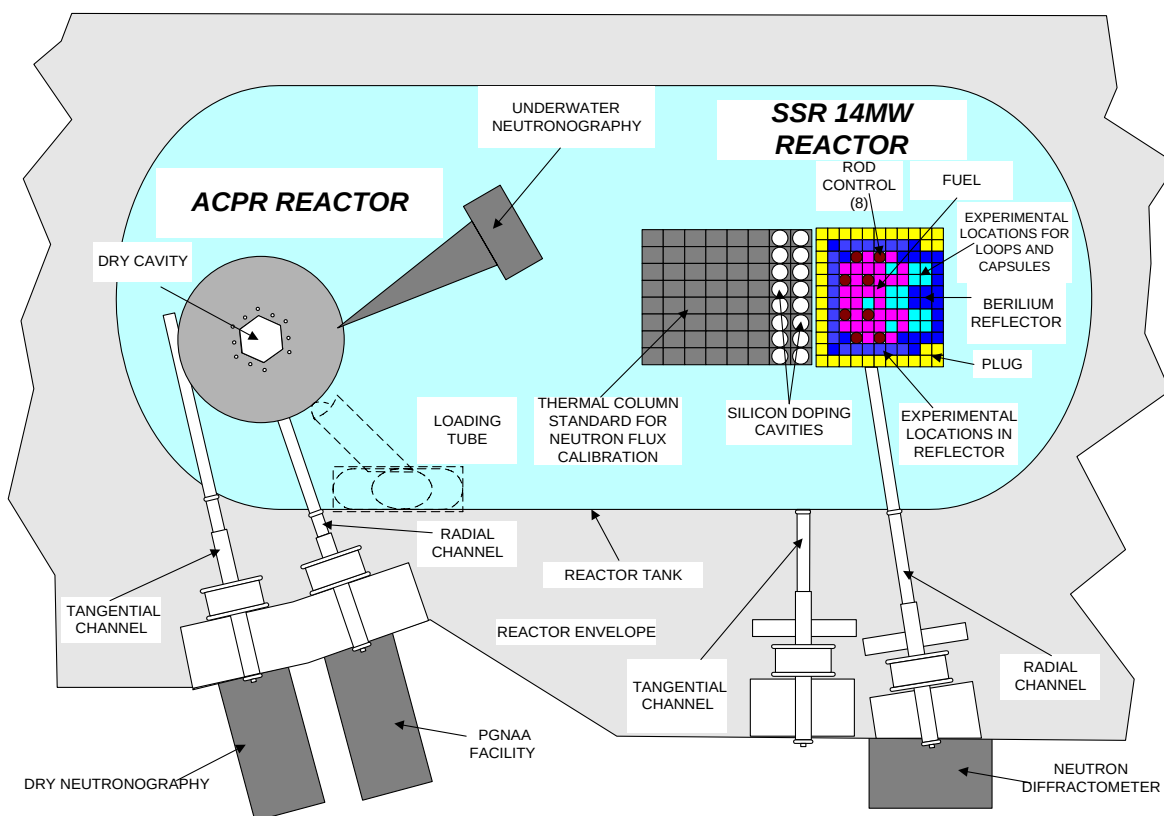
TRIGA REACTOR, INSTITUTE FOR NUCLEAR RESEARCH PITESTI
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1. Introduction

The location of the Institute for Nuclear Research is Pitesti, 100 Km Northwest of Bucharest. Our TRIGA reactor was commissioned in 1980 (first criticality has reached on November 17th, 1979). In fact, as it could be seen in the Figure 1, the TRIGA reactor consists in two reactors:

- ✓ A Steady State Reactor, 14MW, initially loaded with HEU fuel (93% enrichment).
- ✓ An Annular Core Pulsing Reactor of 20.000MW.

Figure1 TRIGA reactor pool arrangement



TRIGA SSR Main Characteristics

LEU fuel core	
Low Enriched Uranium (LEU)	10% weight U235, 23% U235 enrichment
Composition	Er-U-Zr1.6 eutectic alloy fuel moderator, Er 2.8% weight
Clad Material	Incoloy 800
Pellet Diameter	1.27 cm
Clad Diameter	1.372 cm
Fuel Length	56 cm
Total Length	76 cm

Reflector	
Type	Beryllium with and without central hole
Number	20 with central hole & 20 without central hole
Dimension	Same cross section as the fuel bundle
Control Rods	
Number	8
Type	Sinterized Boron
Poison Type	B4C natural

Total Weight	0.438 kg of B ₄ C
Bundles Number	29
Fuel Elements Bundle	25

Actuation Mechanism	Rack and drive pinion
Operational Features	
Max. Power	14MW
Max. Thermal Flux	2,9x10 ¹⁴ n/cm ² s
Max. Central Fuel Temp.	To Be Determined

TRIGA ACPR Main Characteristics

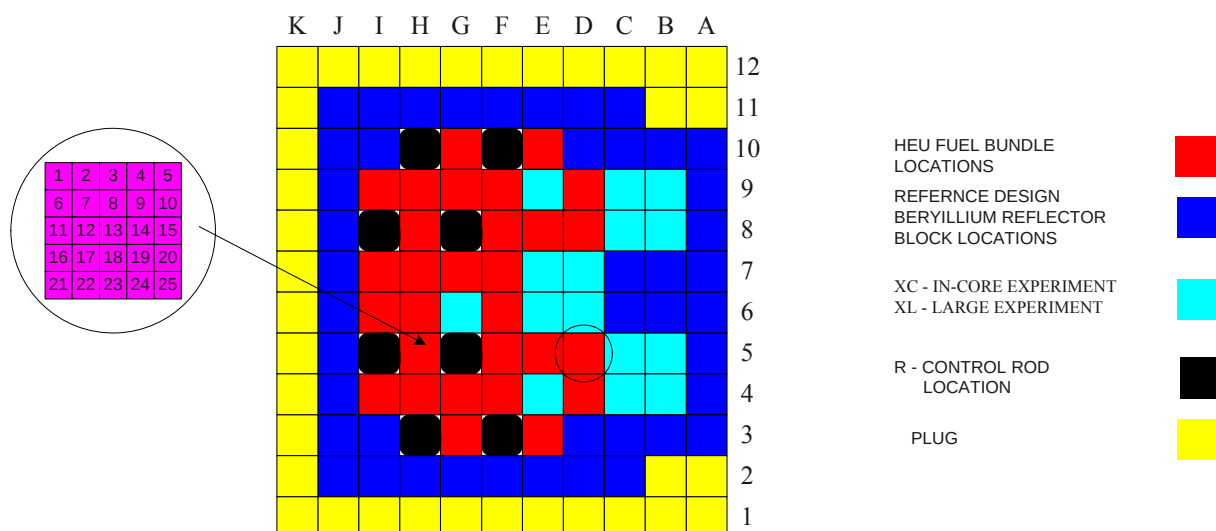
Fuel	
Type and Enrichment	12 wt% U- ZrH fuel, 20 wt% ²³⁵ U
Cladding material	stainless steel with dimples
Diameter	3.56 cm
Cladding diameter	3.76 cm O.D.
Section length	38 cm
Rods number	146+6 fuel followers
Control rods	
Number	6
Type	fuel followered type
Poison material	natural B ₄ C
Rod drive	rack and pinion
Transient rods	
Number	2 fast transient rods and 1 adjustable transient rod
Poison material	92% enriched B ₄ C
Rod drive	fast: pneumatic, adjustable: rack and pinion drive
Reactor performances	
Steady state power	500 KW
Maximum peak power	20,000 MW
Maximum core energy release	106 MW s
Pulse width	4.6 ms 1/2 peak power
Prompt neutron Lifetime	32μs

Initially the standard core contained 29 HEU fuel bundles

- ✓ XC1, XC2, XC3 experimental locations with a similar geometry as one fuel bundle,
- ✓ XL1, XL2, XL3 experimental locations with a similar geometry as four fuel bundles.

That was the configuration used to determine the TRIGA SSR 14MW core performances (see Fig. 2).

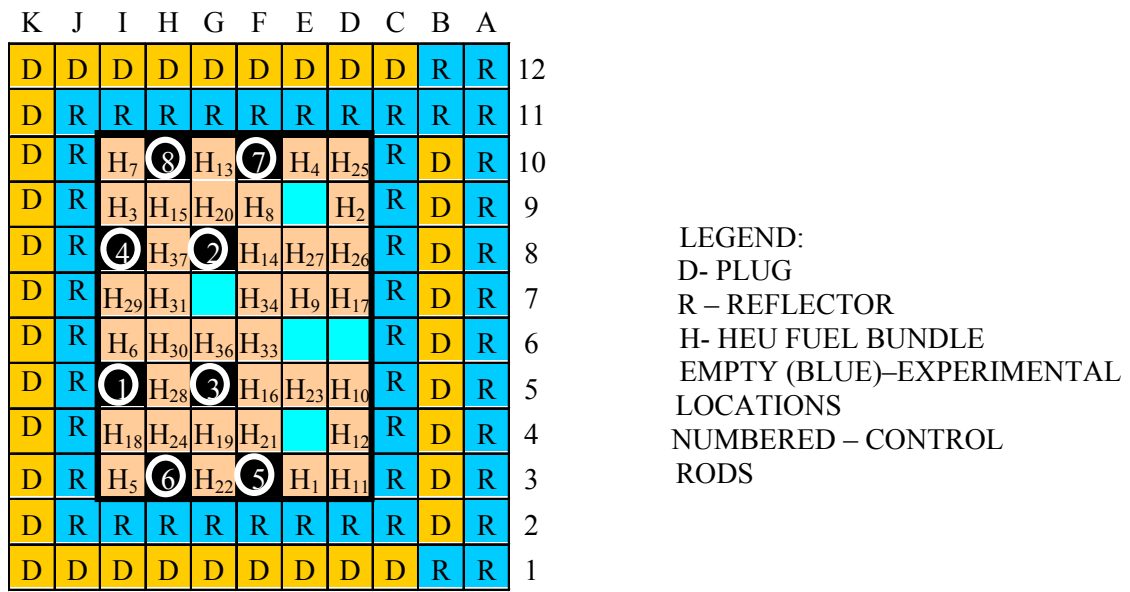
Figure 2 TRIGA SSR Startup Core Configuration, November 17th, 1979



Over time the core configuration was modified due to acquisition/fabrication of the irradiation devices used for CANDU nuclear fuel and materials testing. In any configuration the location XC1 used in connection with Loop A was kept unmodified. The reason is that this irradiation device has a fixed position within reactor core. TRIGA SSR 14 MW use for research and testing program accomplishment imposed HEU refueling in order to maintain an excess of core reactivity.

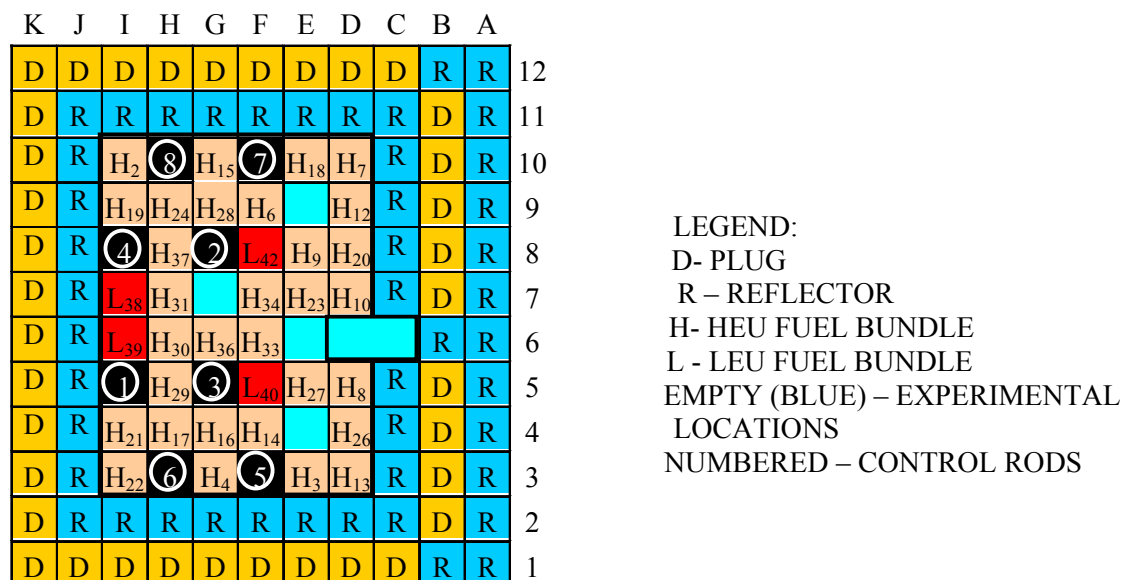
An effective use of nuclear fuel, assessments and measurements and an adequate management allows us to attain an average burn up of 55%, and to reconfigure the core from 29 to 35 HEU fuel bundles. This core configuration was realized in 1991 and is presented in the following Figure 3:

Figure 3 TRIGA SSR core configuration in 1991



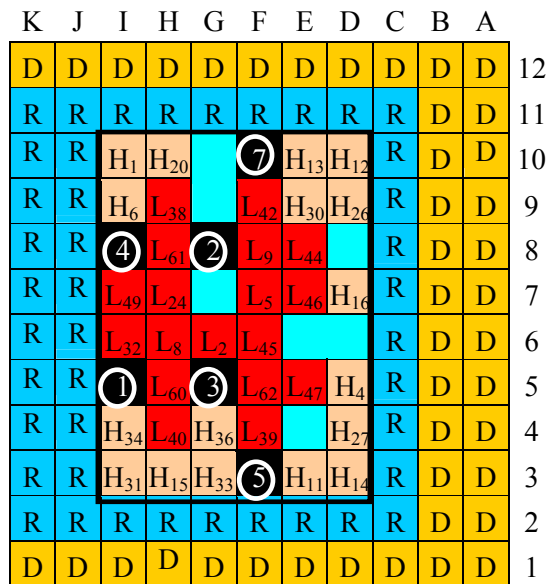
TRIGA reactor conversion has a technical, scientifically, politically and economically significance. The process began in 1992 by core loading with LEU fuel provided by General Atomics, and is related to RERTR program that aims to reduce enrichment in order to decrease the risk of nuclear weapons proliferation. In 1990 it was established an agreement with ANL related to TRIGA SSR core conversion. The main objectives of that agreement were related to neutron and thermal analysis accomplishment in order to determine the feasibility of the conversion project. Once the project was completed at the end of the 2005, in 2006 was performed the conversion from mixed HEU+LEU to full LEU core.

Figure 4 TRIGA SSR core configuration in 1992 after the first step of conversion



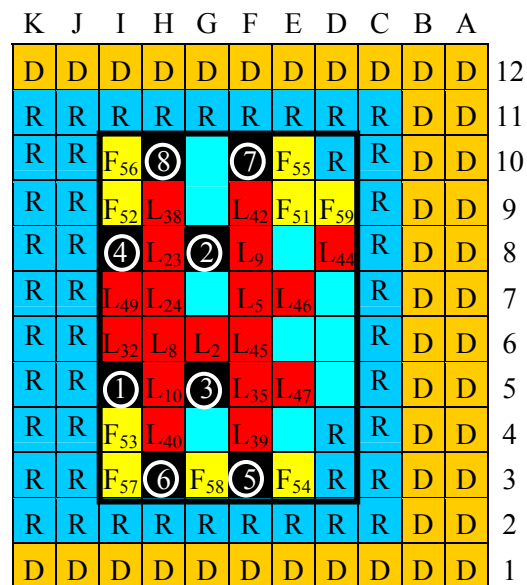
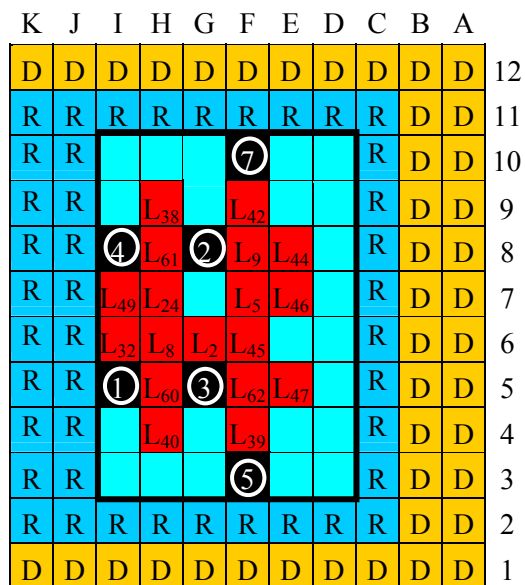
As the Figure 5 emphasize, in March 2004 the mixed reactor core has 18 LEU and 17 HEU fuel bundles by HEU-LEU replacement in successive steps of refueling.

Figure 5 TRIGA SSR core configuration in 2004 before the final step of conversion



Starting from this configuration in 2006 was completed the final HEU-LEU fuel conversion by complete removal of HEU fuel bundles (Figure6) and refueling with fresh LEU fuel manufactured by CERCA, France. As result we have now a standard reactor core with 29 LEU fuel bundles (Figure7).

Figure6 TRIGA SSR core configuration without HEU Figure7 TRIGA SSR final core configuration



LEGEND:

D- PLUG

R – REFLECTOR

F- FRESH LEU FUEL BUNDLE (FRENCH FUEL)

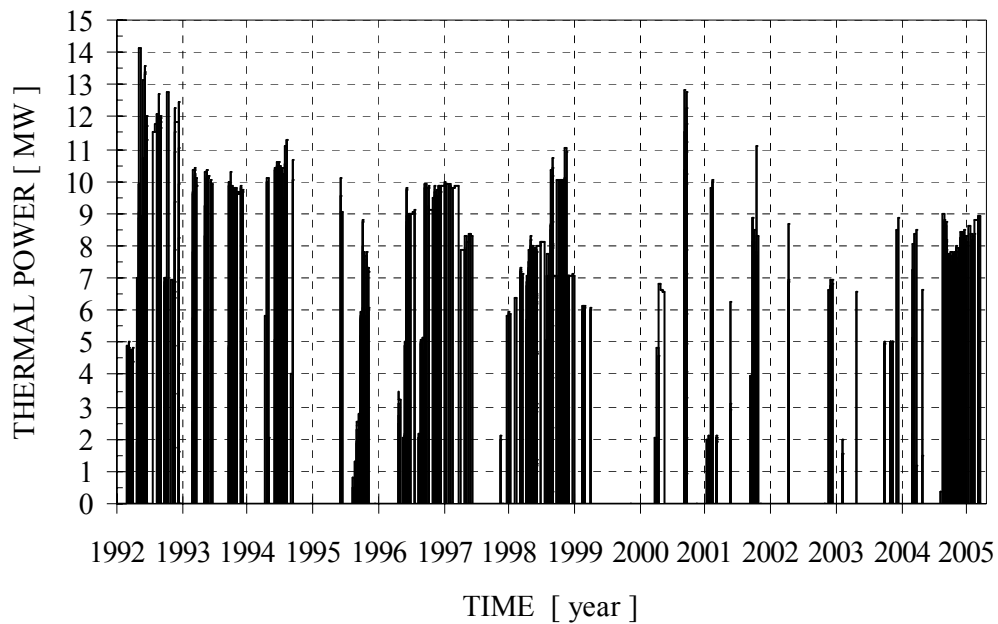
L - LEU FUEL BUNDLE

EMPTY (BLUE) – EXPERIMENTAL LOCATIONS

NUMBERED – CONTROL RODS

From 1992 until 2005 the TRIGA-SSR 14MW reactor with mixed HEU-LEU core was in operation for 1409.93 days. During this time the released energy was 11504.94 MWd. The following graphic illustrates the reactor function history:

Figure7 TRIGA SSR-14MW REACTOR OPERATION IN 1992 - 2005



As a result of our assessments the following table emphasizes the differences between HEU and LEU core:

Table1 TRIGA SSR parameters comparison between 29 HEU and 29 LEU bundles core:

Parameter	HEU	LEU
Fuel bundles	29	29
Reactor Power	14 MW	14 MW
Inlet Temperature	37°C	37°C
Flow rate with one pump in primary cooling system	400 Kg/s	400 Kg/s
Flow rate with two pumps in primary cooling system	700 Kg/s	700 Kg/s
Safety limit for nuclear fuel	1150 °C	1150 °C
Φ_{max} (XC1 location)	$2.9 \times 10^{14} \text{ (cm}^2\text{s)}^{-1}$	$2.82 \times 10^{14} \text{ (cm}^2\text{s)}^{-1}$
PPFmax	2.5	2.14
APFmax	1.30	1.313
CPFmax	3.25	3.01
a	$6.0 \times 10^{-5} \text{ (}^\circ\text{K)}^{-1}$	$5.5 \times 10^{-5} \text{ (}^\circ\text{K)}^{-1}$
Keff	0.00725	0.00730
Tcmax	705 °C	617 °C
Core life time	8000 MWD (29 fresh HEU fuel)	4500 MWD (11 fresh and 18 burned LEU fuel)

Shim bundle removal

Pmax	550 MW	500 MW
T _C max	810 °C	773 °C *

The assessment of LEU nuclear fuel behavior shows differences between maximum temperature values and maximum flux values for LEU and HEU fuel. These differences appear due to loading diagram with fresh LEU fuel: because of the relative large amount of fuel bundles (11 out of 29) that have been introduced in a single step, these bundles were placed on reactor core edge.

RESULTS OF POST-IRRADIATION EXAMINATION OF THE (U-Mo)–ALUMINIUM MATRIX INTERACTION RATE

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ABSTRACT

For improving the irradiation stability of the U-Mo/Al fuel composition it is necessary to reduce the rate of the interaction between the fuel particles and the Al matrix. Using the Al matrix alloyed by Si and creating the barrier coatings on the fuel particles have been considered as the possible ways.

The first results of post-irradiation examination of the fuel particles interaction with the matrix in U-Mo/Al and U-Mo/Al-12%Si compositions and in the same compositions with the Nb, Zr-1Nb and UO₂ barrier layers around the fuel particles are presented in this paper. In the course of irradiation 60% ²³⁵U burn-up has been achieved at 19.7% fuel enrichment.

1 Introduction

It has been reported [1] that for reducing the interaction rate of U-Mo fuel particles with Al matrix the mini-elements, containing the different types of barrier coatings around the fuel particles and two types of matrix material, have been fabricated and exposed to irradiation in the IVV-2M reactor.

This paper presents the first results of post-irradiation examination of the fuel particles - matrix interaction in the U-9%Mo/Al and U-9%Mo/Al-12%Si compositions and in the same compositions with the Nb, Zr-1Nb and UO₂ barrier layers around the fuel particles irradiated to 60% ²³⁵U burn-up at 19.7% fuel enrichment.

2. Fuel composition

The next types of the fuel compositions have been chosen for irradiation:

- U-9%Mo/Al;
- U-9%Mo/**Nb**/Al;
- U-9%Mo/**Zr-1Nb**/Al;
- U-9%Mo/**UO₂**/Al;
- U-9%Mo/Al-12%Si;
- U-9%Mo/ **UO₂**/Al-12%Si.

The basic matrix is the PA-4N (Al-0.3%Si) powder. The fuel particles have been coated by Nb and Zr-1Nb alloy using the plasma method. The thickness of the barrier layer was equal to 1-2.5 and 1-2 μm, correspondingly. The UO₂ layer has been obtained on the fuel particles by oxidizing in air, the layer thickness is ~0.3 μm.

3. Mini-elements

Mini-elements have been fabricated by the following manner. The fuel composition was being put into the aluminium cylinder. Its diameter was equal to 10.9 mm, the bottom thickness is equal to 1.5 mm.



Fig. 1. Typical mini-element cross-section

The cover with thickness of 3 mm was being inserted into the cylinder. Then the hot pressing have been carried out. The diameter of mini-elements is equal to 10.88-11.0 mm, the height is equal to 5.07-5.28 mm. The common shape of the mini-element cross-section is shown in Fig. 1.

4. Irradiation conditions

The scheme of mini-element disposition in the plates of the irradiation assembly in the IVV-2M reactor with the indication of their number and of materials of the matrix and barrier layer is shown in Fig.2. The distance between mini-elements along the plate was equal 14 mm.

The plates were placed vertically in the special irradiation assembly (Fig. 3), and their both sides were being washed by water. The center of the active zone was placed at the level of the third floor mini-element middle.

№52 Al-Si -	№58 PA-4N Nb	№50 Al-Si -	№60 PA-4N Nb
№53 PA-4N -	№48 Al-Si UO ₂	№65 PA-4N Zr-1Nb	№63 Al-Si UO ₂
№40 PA-4N UO ₂	№54 PA-4N -	№61 PA-4N Nb	№66 PA-4N Zr-1Nb
№67 PA-4N Zr-1Nb	№47 PA-4N UO ₂	№57 PA-4N -	№43 PA-4N UO ₂
№62 PA-4N Nb	№68 PA-4N Zr-1Nb	№45 PA-4N UO ₂	№70 PA-4N -

Fig. 2. Mini-element disposition in the plates

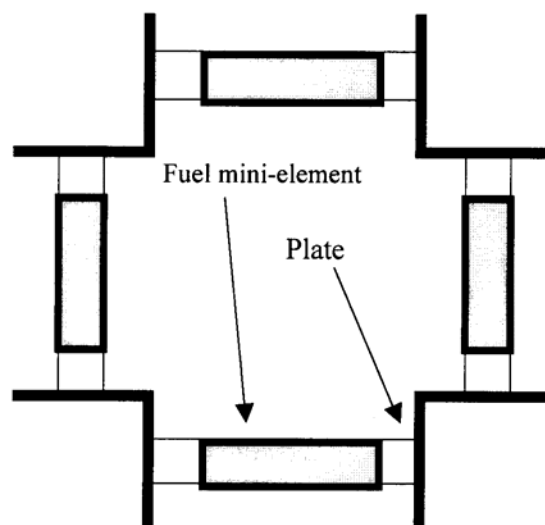


Fig. 3. Cross-section of the irradiation assembly

The irradiation was conducted in the interior of the fuel assembly placed in the 5-5 cell of the IVV-2M reactor active zone for 118 days at the nominal reactor power. The mean heat flux density was equal to 60 W/cm². The mean fission rate was equal to 4.0×10^{14} fis./(cm³c). The mean fission density of 4.1×10^{21} fis./cm³ and the equivalent ²³⁵U burn up of 60% have been achieved to the irradiation end.

The thermocouples were placed in the claddings of two mini-elements. The cladding and water temperatures were controlled at the input and output and in the active zone center in the course of the reactor testing.

The calculation of the temperature distribution along the mini-element cross-section is shown in Fig.4.

5. Post-irradiation test results

The mini-element visible failures and the bulges on the claddings were not observed.

The results of the mini-element thickness measurements, attributed to the fuel layer, are shown in Fig.5. The basic U-Mo/Al composition exhibits the highest swelling (~17%). Coating of the U-Mo particles by Nb and Zr-1%Nb alloy reduces the swelling of the basic U-Mo/Al composition by 20-27%, and oxidizing – ~45%.

Using the Al-12%Si alloy as the matrix and the particles without the coating leads to the swelling reducing by comparison with the basic composition by 20%, and using the UO₂ coatings – more than two times.

The mini-element cross-section metallographic test results are shown in Fig. 6.

Cracking of the composition was not detected in all cases. As a rule, the barrier layer around the particles, the interaction zone «matrix-fuel particles» and the region of the fission fragments output to the matrix have been observed. Moreover, the formation of the interaction zone on the Al matrix side is typical for particles with the barrier coatings. This is indirect evidence of the preferential diffusion of U atoms to region of the reaction with the components of the Al matrix.

The results of measuring the barrier coatings thickness, the interaction zone width and the layer width of the matrix, damaged by the fission fragments for different types of the fuel composition are presented in Table.

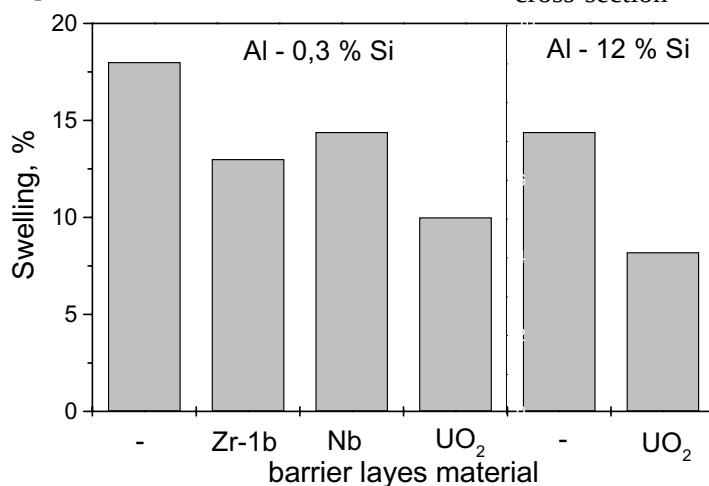


Fig. 5. Swelling of fuel compositions: U-Mo/Al; U-Mo/Nb/Al; U-Mo/Zr-1%Nb/Al; U-Mo/UO₂/Al; U-Mo/Al-12%Si; U-Mo/ UO₂/Al-12%Si.

oxidized fuel particles by comparison with the technically clean Al matrix.

The presence of the metal coatings on the particles surface leads both to reducing the (U,Mo)Al_x layer thickness and to reducing the width of the damaged by the fission fragments Al matrix layer around the particles by 15-20%. This may be an evidence of the less fission fragments (including gaseous fragments) content both in the matrix and in the layer around the fuel particles and, therefore, may positively influence on the behaviour of such compositions under irradiation including the case of the more high burn-up.

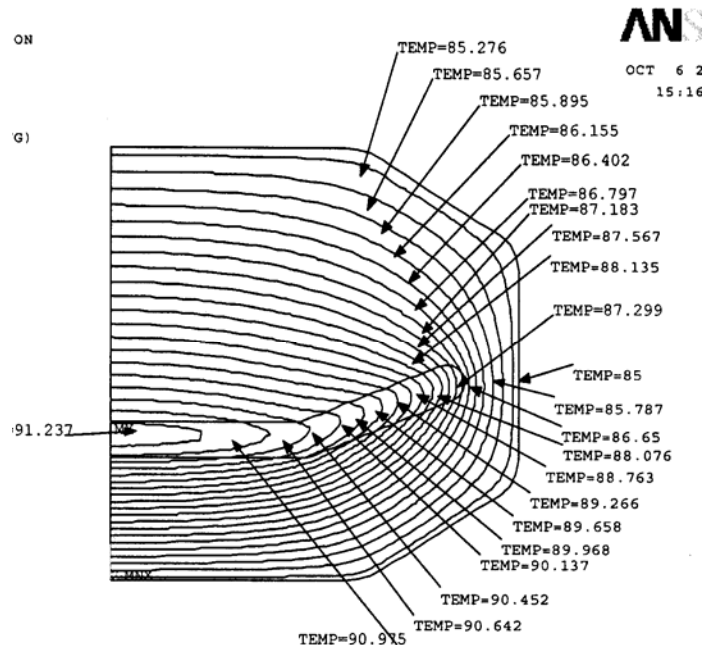


Fig. 4. Temperature distribution along mini-element cross-section

The maximum interaction layer thickness was observed for the basic U-Mo/Al composition and was equal to ~5 μm. The oxidic coatings (thickness is ~0.3 μm) on the surface of the particles insignificantly reduce the (U,Mo)Al_x layer thickness. The (U,Mo)Al_x layer thickness for the UMo/Nb/Al composition is less by ~30% by comparison with the basic composition and is nearly 6 times less if the coatings from the Zr-1%Nb alloy are used.

The addition of 12% Si to the Al matrix leads to reducing the (U,Mo)Al_x layer thickness by ~15% for the particles without the coatings and - by ~30% for the

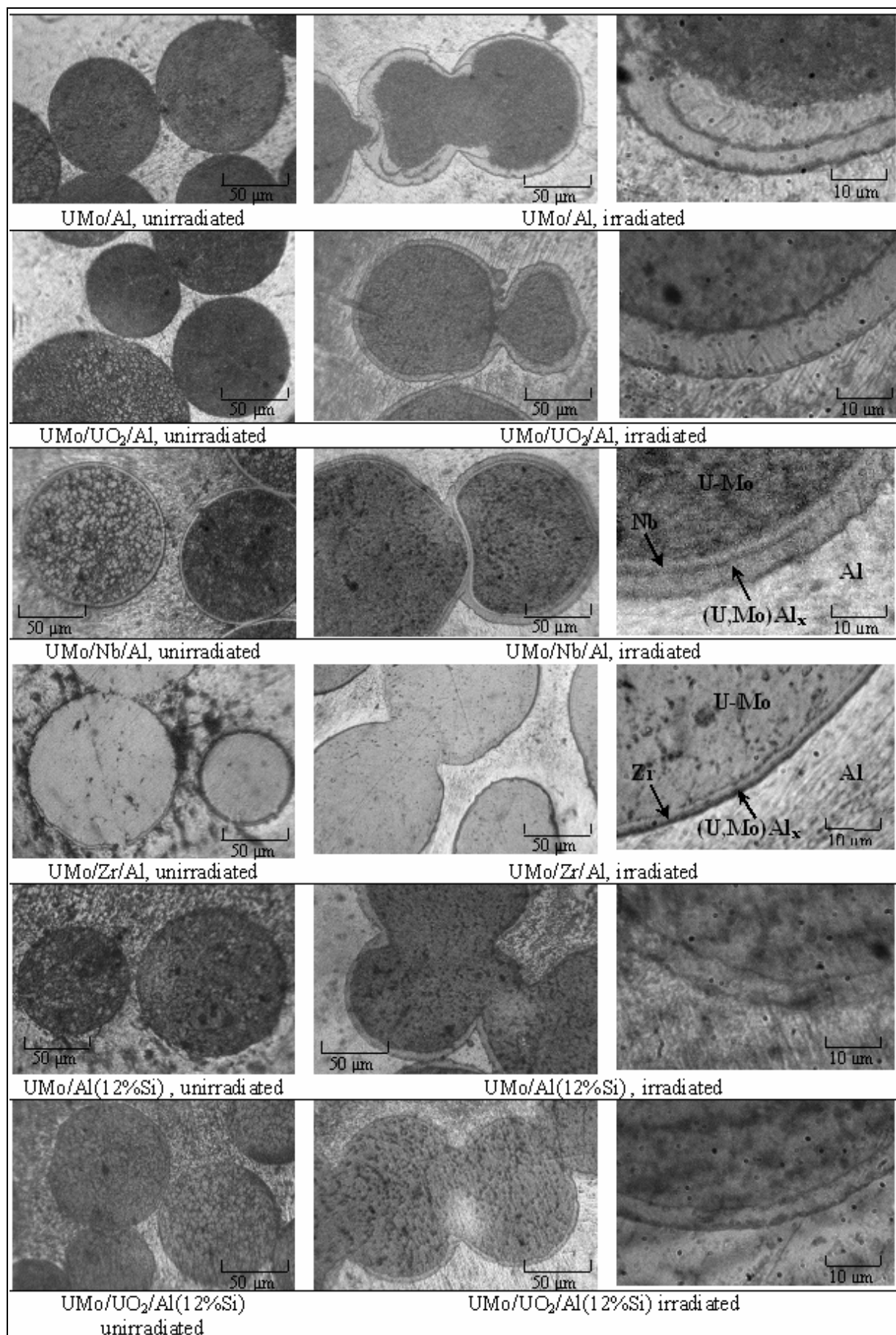


Fig. 6. Structure of the fuel compositions before and after irradiation

Table – Thickness of coatings, (U,Mo)Al_x layer and layer of the matrix damaged by the fuel particles

Composition	Coating thickness, μm		U,Mo)Al _x layer thickness, μm	Layer of the matrix damaged, μm	Thickness of all layers, μm
	Before irradiation	After irradiation			
UMo/Al	-	-	4.9	7.7	12.6
UMo/VO ₂ /Al	~0.3	~0	4.6	8.1	12.7
UMo/Nb/Al	1.1-2.3	1.0	3.3	6.3	10.6
UMo/Zr/Al	0.9-2.1	1.0	0.8	6.8	8.6
UMo/Al-12%Si	-	-	4.2	9.5	13.7
UMo/VO ₂ /Al-12%Si	~0.3	~0	3.3	7.2	10.5

The metal coatings thickness is nearly two times less than the initial one (see Table). However the coatings have preserved their integrity and were strongly coupled with the fuel particles and the interaction layer. The oxidic coatings on the irradiated samples have not been observed. This is probably conditioned by either their small thickness or their interaction with the aluminium matrix.

The analysis of the data on the mini-element swelling and the (U,Mo)Al_x layer thickness gives evidence of the absence of correlation between them. This may be presumably because of both the heterogeneous distribution of the fuel particles along the fuel core and the differences of the chemical composition of the very (U,Mo)Al_x layer.

6. Conclusions

The reactor test of the mini-elements containing the U-9%Mo alloy fuel particles without the coatings and with Nb, Zr-1%Nb alloy and VO₂ barrier coatings dispersed in the technically clean Al (0,3% Si) and Al-12%Si alloy matrix have been successfully carried out.

To the test end the next mean irradiation parameters values have been achieved: the fuel flux density – 60 W/cm², the fission rate 4.0×10^{14} fis./(cm³c), the fission density – 4.1×10^{21} fis./cm³, the fuel temperature ~ 92°C, the equivalent burn up - 60%.

The positive influence of the Nb, Zr-1%Nb alloy and VO₂ barrier coatings and the additions of Si to the Al matrix on the mini-element swelling reduction has been observed. Besides, the oxidic coatings and the Si addition to the Al matrix act synergistically leading to the least swelling among the investigated compositions.

The presence of the oxidic and metal coatings and the Si additions in the Al matrix reduces the formation rate of the interaction layer of the (U,Mo)Al_x type.

The least (U,Mo)Al_x layer thickness has been observed in the composition with the Zr-1%Nb alloy barrier coating.

The metal barrier coatings have preserved their integrity, but their thickness has been reduced nearly two times. The presence of the oxidic coatings on the surface of the fuel particles has not been observed after irradiation because of their small initial thickness (<0.3 μm).

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PLACA/DPLACA SIMULATION OF MONOLITHIC/DISPERSE UMo PLATES

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ABSTRACT

The codes PLACA/DPLACA simulate the irradiation behavior of plate-type fuels under normal operation conditions. They correspond to monolithic and dispersed fuels, respectively. The codes have a modular structure and contain about thirty interconnected and mutually dependent models. They make possible a detailed simulation of the evolution of the more relevant physical parameters of a plate-type fuel element during its permanence within a reactor. In the particular case of the U-Mo/Al system, a model is included to give account of the interaction layer that develops between the alloy and Al. The model assumes that interdiffusion of Al and U through the layer is the rate determining step. The associated Stefan problems are numerically solved. This presentation includes the simulation of irradiation experiments performed with U-Mo/Al fuels: monolithic miniplates fabricated in CNEA and irradiated at the ATR and dispersed-fuel plates belonging to the IRIS-2 experiment. Comparison of the calculation results with the experimental data evidence in these and in previously analyzed cases, the correct performance of the models involved and the good coupling of the ensemble.

1. Introduction

PLACA and DPLACA, which can be considered as two versions of the same code, employ the finite elements method in Cartesian coordinates with linear rectangular elements to solve the differential equations corresponding to the thermal and elastic-plastic problems. Two perpendicular views (xy and yz) of the plate are considered. They make possible a global analysis of the plate since, when combined yield a quasi-three dimensional description of the system. Transverse sections (xy) can be analyzed at any location of the plate, particularly at those considered more critical, allowing a detailed analysis of all the variables involved.

Special attention is paid to the candidate fuel constituted by U-Mo particles dispersed in Al. This system exhibits a singular behavior due to the interaction layer that grows around the fuel particles and can provoke uncontrolled swelling. A model has been developed to give account of this phenomenon [1,2]. It assumes that the kinetics of both interfaces is determined by diffusion of U and Al through the layer. From the computational point of view this is performed by numerically solving the diffusion equations of two species along with the Stefan problems associated to the two moving layer boundaries. This gives to the code a realistic tool since the consumption of fuel particles and matrix is

strongly dependent on the longitudinal position of each finite element, i.e., on the local power history and the consequent temperature distribution.

2. Modelling and calculation procedure

To perform the finite element calculations, the domain is subdivided into linear rectangular elements. The two sections of the plate considered in the codes are shown in Figure 1.

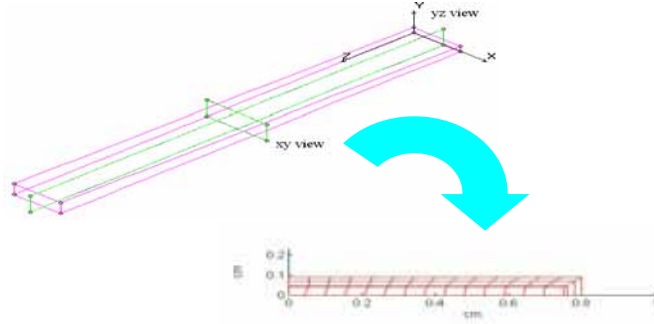


Figure 1: Sections xy and yz of the fuel plate considered in PLACA/DPLACA

The temperature distribution in the yz section is not symmetrical along z, but symmetry exists in the xy section along both axes. Then, the calculation domain used in the xy view is one-fourth of the section while that used for the yz view contains one half of the respective section. Domain discretization is performed by a mesh generator included in the code. The input parameters are the fuel geometry, the constituent materials and the amount of ^{235}U within the plate, the particle shape and its size distribution if the fuel is of the dispersed type, the coolant velocity and mass flow, and the neutron flux or linear power history of the fuel plate. The history is divided into stationary periods and power ramps; these are divided into a number of stationary steps, at the programmer's choice.

The first calculation step is the temperature distribution which serves as input for the stress-strain distribution problem for which plane strain is assumed. A predictor-corrector algorithm is used to solve the non-linear equations associated to the thermal-elastic-plastic problem. Within each finite element the temperature and power history are assumed uniform.

In the particular case of the U-Mo particles dispersed in Al, a representative volume constituted by a U-Mo particle, the interaction layer, a pores shell and the surrounding Al matrix is chosen in each finite element, all of them according to the proportions in which they are present in the whole material. The interdiffusion equations in spherical coordinates are then solved to determine the intermetallic growth and the corresponding decrease of the amount of fuel and matrix. If the particles are not spherical, a correction factor is applied. The decrease of the initial porosity during fuel burnup (densification) is considered. The results obtained in each representative volume are extended to the volume of the corresponding finite element and then to the whole fuel plate volume. In this procedure the particle size distribution is taken into consideration.

Swelling and growth of the intermetallic layer [3,4] take place during fuel life time and modify the volume fraction of the diverse fuel plate constituents. These processes also deteriorate the heat conduction and, in consequence, temperature within the plate increases; in consequence, layers growth and swelling accelerate even in a constant power regime. For this reason, the code actualizes the temperature distribution after a given burnup interval. Moreover, some of the physical parameters of the system change due to irradiation effects and need also to be actualized.

The code considers aluminum or Zircaloy as possible cladding materials. If it is made of Al, the superficial oxide layer on the external cladding surface is taken into account since its low thermal conductivity is responsible for an extra temperature increase in the domain. Its growth rate depends on

the working temperature and on the water composition. The code contains four models to give account of this growth [5]. If the cladding material is Zircaloy the oxide layer is not considered since its growth is very slow at the normal operation temperatures of research reactors.

To analyze the stress and strain distributions, the total strain vector $\{e\}$ is written as $\{e\} = \{\varepsilon\} + \{\varepsilon^{th}\} + \{\varepsilon^{sd}\}$ where $\{\varepsilon\}$, $\{\varepsilon^{th}\}$ and $\{\varepsilon^{sd}\}$ represent the strains due to the applied loadings, to thermal expansion and to swelling-densification, respectively.

After determining the temperature and particle radius in each finite element, the code solves at every time step the interdiffusion problem, evaluates the particle and matrix consumption, the growth of the interaction layer and of the porosity annulus and actualizes the volume fraction of each component. These values are then averaged for the whole domain. For non-spherical particles six shape factors are included to give account of different area/volume relations.

3. Test Cases

Both codes were applied to simulate several irradiation histories. In this work the predictions of PLACA are compared with destructive and non-destructive analysis of two miniplates of monolithic LEU, U-7Mo clad in Zircaloy-4, elaborated in CNEA and irradiated in the ATR during the RERTR-7A experiment [6]. Also, some fuel plates of dispersed UMo particles from the IRIS-2 experiments [7] are simulated with DPLACA.

3.1 Monolithic UMo: MZ25 and MZ50

The monolithic miniplates identified as MZ25 and MZ50 were fabricated by hot co-lamination of U-7Mo plates sandwiched with two Zry-4 plates and designed to have a total thickness of 1 mm and a meat thickness of 0.25 and 0.50 mm, respectively. The dimensional and physical characteristics of the miniplates are summarized in Table 1, along with the available PIE results [8] which include Gamma scanning, dimensional measurements and sectioning for metallographic inspection and determination of burn-up by chemical methods. The Table also includes the results of simulations with PLACA.

Table 1. Dimensional and physical characteristics of miniplates MZ25 and MZ50. Comparison between numerical and available experimental results.

Miniplate	MZ25		MZ50	
Dimensions (mm)	73×18.8×0.25		71×18.6×0.50	
Density (gU/cm ³)	16.5		16.3	
	Exper.	Num.	Exper.	Num.
Burn-up (%)	38	34	33	30
Fission density × 10 ²¹ (f/cm ³)	2.7	2.8	2.3	2.5
Max. heat flux (W/cm ²)	135	127	217	235
Max. internal temp. (°C)		100		163
Max. external temp. (°C)		60		78
Total swelling (%)	3.6	5		4.8
Oxide layer (μm)	2.6	0.0		0.0

3.2 Disperse UMo: IRIS-2

Plates fabricated with more than about 50 vol% of atomized U-7wt%Mo powder with a density of 8.3gU/cm³ and a porosity of 1.5% were used in the IRIS-2 experiment. Figure 2 shows the evolution of temperature (in a xy view calculation) at the meat centreline, at the oxide layer-cladding metal

interface, at the external face of the oxide layer and of the coolant at the maximum power plane. The power history used was obtained from [9].

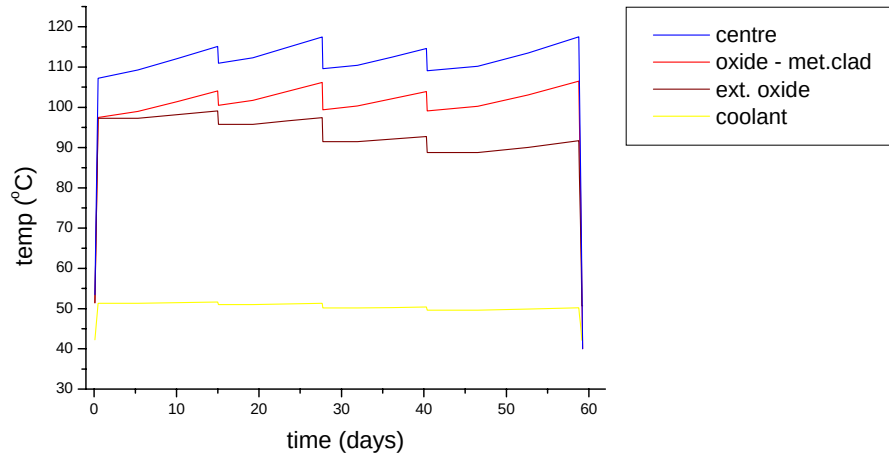


Figure 2: Temperatures of the plate at the maximum flux plane.

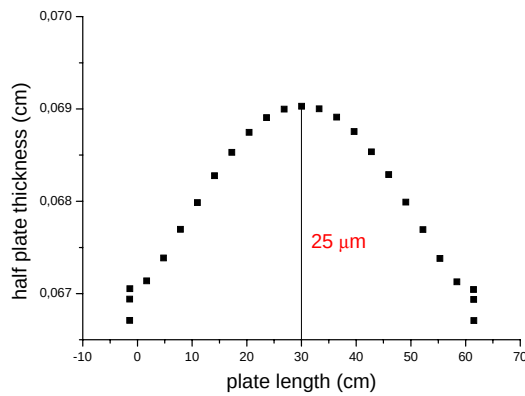


Figure 3. Thickness of one-half of the plate at EOL predicted by DPLACA.

Figure 3 shows the thickness of one-half of the plate along its length at EOL. A yz view calculation is used in this case. Burnup, volume percent of UMo, Al and intermetallic variation along the plate length are shown in Figure 4.

4. Discussion

Previous results obtained with PLACA and DPLACA [10] and those presented in this case show a good agreement with the experimental results and with those obtained with other codes. They make possible a detailed simulation of the evolution of the more relevant physical parameters of a fuel plate during its permanence within a reactor. The results evidence the correct performance of the models involved and a good coupling of the ensemble. The modular structure of both codes allows testing and replacement of the different particular models included. Moreover, the xy view allows a detailed analysis at any section of the plate, particularly at those considered more critical. Superposition of views xy and yz makes possible a global analysis of the plate.

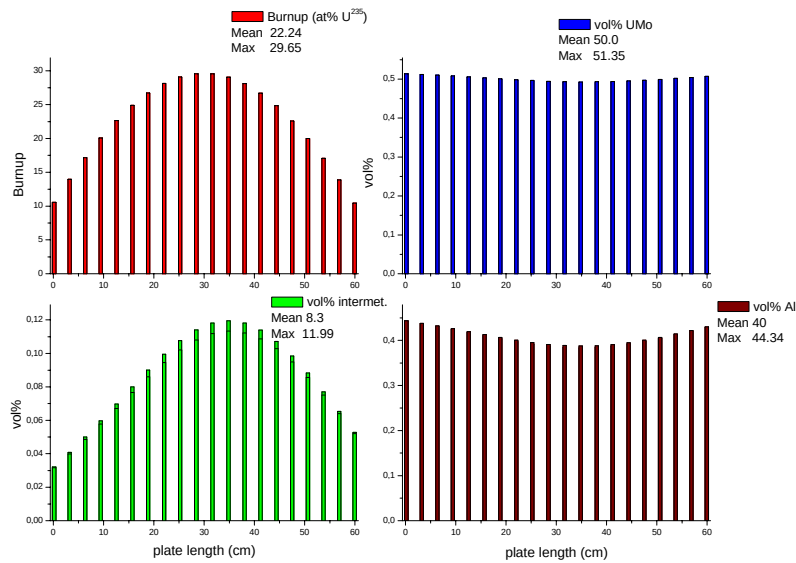


Figure 4 Burnup, Volume % of UMo, Intermetallic and Al vs. plate length at EOL.

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STRUCTURE STUDIES OF DISPERSED U-Mo FUEL AFTER IRRADIATION AND ISOCHRONOUS ANNEALING WITHIN THE TEMPERATURE RANGE OF 150 - 580 °C BY THE NEUTRON DIFFRACTION METHOD

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ABSTRACT

The data about the phase and structural state of (U, Mo)Al_x layer have been scarce so far, thereby precluding from development of the measures for reducing the rate of layer growing.

This paper presents the results of the structure study on the dispersed uranium-molybdenum fuel after irradiation to different burn up levels from 33 to 97% and one-hour annealings within temperatures of 150 to 580 °C by the neutron diffraction method.

1. Introduction

The experiments with U-Mo dispersion fuel which were performed in the frame of the RERTR national programs have revealed formation of an interaction layer such as (U,Mo)Al_x. For the present time neither the phase composition, nor the structural parameters of (U,Mo)Al_x layer have been determined. There are only some scattered experimental data on chemical composition of this layer where the content of Al varies as corresponds to compounds from (U,Mo)Al₃ to (U,Mo)Al₈ [1, 2]. It is assumed that under irradiation temperature below 200 °C (U,Mo)Al_x layer would be amorphous. No experimental data to confirm this assumption are available yet. The only study [3] has demonstrated that the UAl₃ crystal phase formed in the fuel rods at 20 % burnup and irradiation temperature of fuel meat exceeding 200 °C. The lack of data on structure and phase composition of (U,Mo)Al_x layer precludes from developing the measures aimed at reducing or complete inhibition of the layer growth.

The present paper gives the results of studying the structure of U-Mo dispersion fuel following its irradiation in IVV-2M reactor (Zarechny) up to the different values of burnup (from 33 to 97 %) and after annealing during 1 hour in the temperature range from 150 to 580 °C performed by the neutron diffraction method.

2. Materials and the experiment technique

The specimens of 40x8x1.3 mm size cut at different axial points from the fuel elements of the combined FAs of KM003 and KM004 types tested in IVV-2M reactor. The specimens are characterized in Table 1. Two specimens G13 and G100 were taken from KM003 FA, which corresponded to the axial positions of the specimens G96 and G97 from KM004 FA (Fig.1). It should be pointed out that G98 specimen was cut from the fuel element portion, which was in the direct contact with the areas of cladding pillowing. The fuel meat of this specimen contained separated gas pores and their agglomerations as well as minor cracks up to 1.5-2 mm in length.

Each specimen contained two fuel claddings of 0.45 mm thick and one layer of the fuel meat of 0.45 mm thick. Three such specimens were combined into a single package for making the experiments on neutron diffraction.

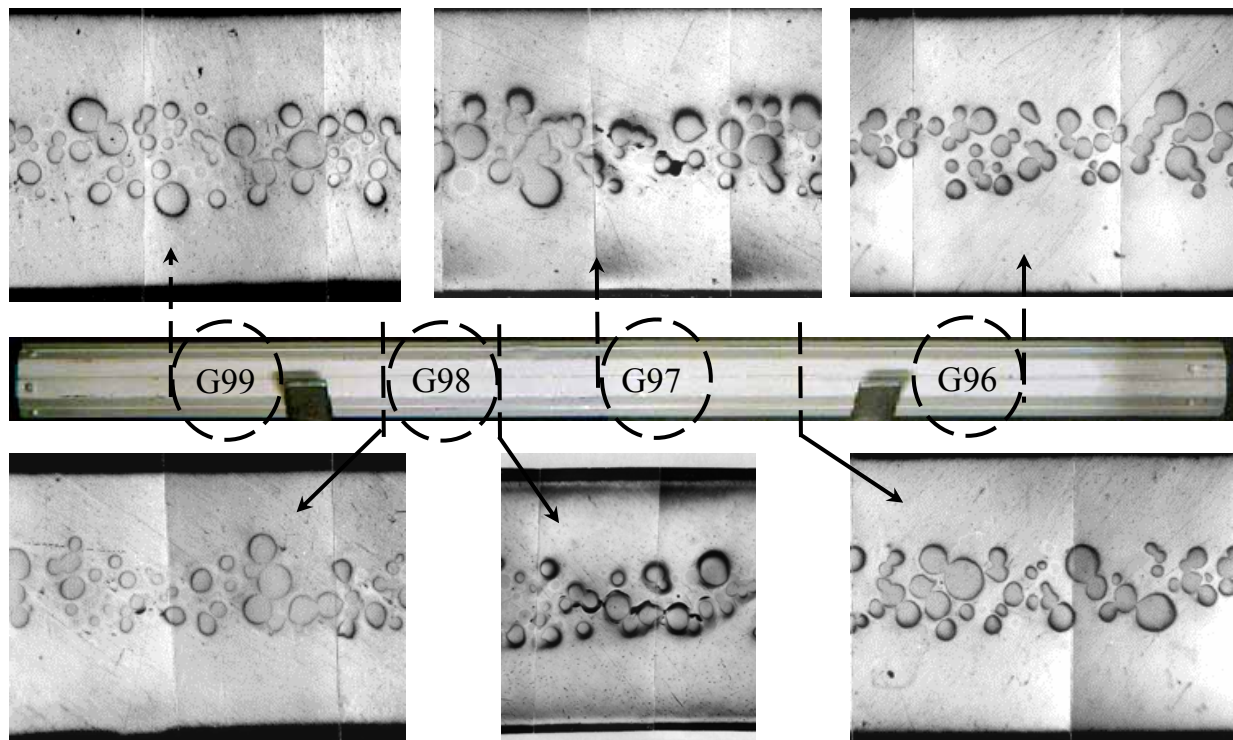


Fig. 1. Appearance of the fuel element from KM004 FA and macrostructure of the fuel meat at different axial cross-sections of the fuel element

Specimen No.	FA No.	Bu ¹⁾ , %	ϕ , 10^{14} fiss/(cm ³ s)	Ψ , 10^{21} fiss/cm ³	T _{BOL} , °C	q _{BOL} , W/cm ²	IL ²⁾ , μ m	Volume ratio of phases in a specimen, %			
								Al	U-Mo	IL	Al _{FP} ³⁾
G13	KM003	32.9	2.6	2.1	43.5	48	1.2	87.0	12.1	0.9	13.4
G100	KM003	51.1	4.5	3.8	61.4	80	3.6	85.6	11.6	2.8	14.2
G96	KM004	55.1	3.1	4.1	48.6	66	3.5	85.7	11.6	2.7	14.2
G97	KM004	96.3	5.5	7.2	77.2	115	11.3	80.7	10.1	9.2	16.9
G98	KM004	96.9	5.7	7.3	85.3	118	11.2	80.8	10.1	9.1	16.8
G99	KM004	78.4	4.6	5.9	86.4	95	9.0	82.3	10.5	7.2	16.0

1) equivalent burnup;
2) thickness of (U,Mo)Al_x layer;
3) aluminum matrix damaged by fission products

Tab. 1. Characterization of specimens under testing

The neutron diffraction experiments have been performed at IVV-2M reactor (in Zarechny). The experimental data were obtained for the angular range from 5° to 105° by 2 θ with the step of 0.1°. The spectra of all specimens were obtained at the first stage. After that three specimens (G96, G98 and G99) have been subjected to step-by-step, 50 °C each, annealing in the temperature range from 150 to 580 °C, annealing time was 1 h. Neutronograms were taken after each annealing.

3. Experimental results and their discussion

Fig. 2 shows the neutron spectra obtained for the specimens irradiated to the different burnups and after isochronal annealing for G96, G98 and G99 specimens. For the purposes of better demonstration, the spectra of each specimen are shifted along the intensity scale.

The specimens under testing contain the strong coherent reflexes from the following crystal phases (Fig. 2a): the alloy on the basis of γ -U-9% Mo, Al as the base of the aluminum matrix and fuel claddings made of SAV-1 alloy. Several other visible minor reflexes have been interpreted as those induced by Mo-based alloy (Mo₉₀U₁₀ phase) and UAl₃. Besides, some neutronograms at angles of 51-52 and 60 degrees show the traces of reflexes induced by γ -FeNiCr alloy caused by the stainless steel

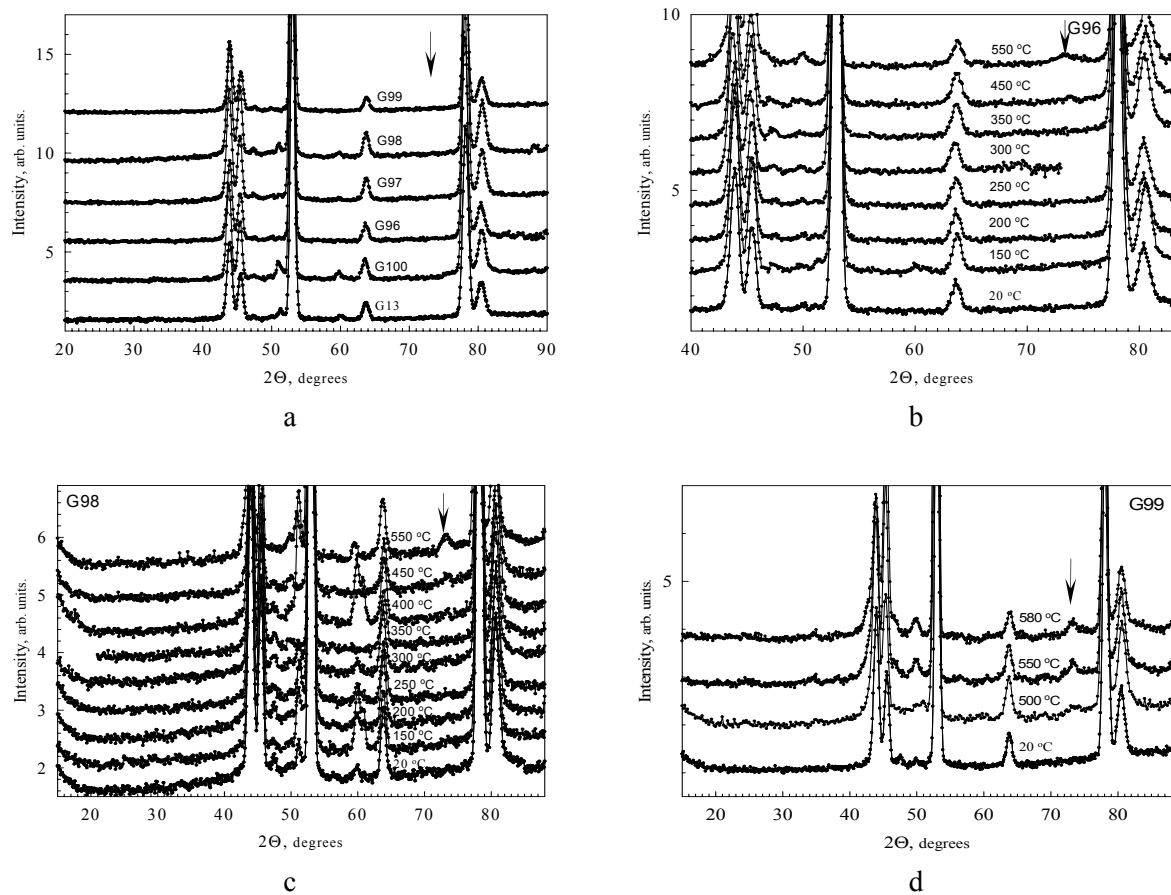


Fig 2. Neutron diffraction spectra for the specimens with different burnups (a) and after isochronal annealing in the temperature range of 150-580°C (b-d). The position of UAl_3 phase line (220) is marked with arrow.

position limiter placed inside the vanadium container. The spectra of specimens under testing did not include any reflexes corresponding to such intermetallide phases as UAl_2 , UAl_4 or to the more complex structures based on U-Mo-Al system.

The thorough investigation of the diffraction spectra given in Fig. 2a made it possible to notice that reflexes corresponding to Al resulted in an increase of diffusion background caused by scattering of the amorphous Al-based phase at short-range order. This phenomenon can be clearly seen in Fig. 3 that illustrates the neutronograms for the specimens with minimum (G13) and maximum (G98) burnups, and minimum and maximum contents of $(\text{U},\text{Mo})\text{Al}_x$, i.e. 0.9 and 9.1 %, respectively.

After processing of the experimental data given in Fig. 2a it is possible to observe that the contents of $\text{Mo}_{90}\text{U}_{10}$ and UAl_3 phases does not practically depend on burnup and makes up ~ 1.5 and ~ 0.5 %, respectively, Fig. 4. It is necessary to specify that the content of UAl_3 phase was found at the level of error for the determination of phases with content less than 1 %. At the same time the amount of Al and γ -UMo phases decreases as burnup rises, that is caused by the reaction between these phases:

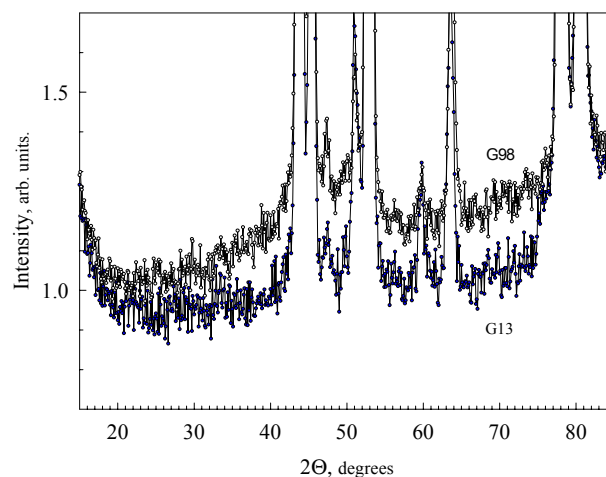


Fig. 3. Neutronograms of G13 and G98 specimens enlarged along the intensity

volume ratio of Al is reduced by ~8 % as burnup rises from 33 to 97 %, whereas the content of γ -UMo phase decreases only by ~0.6 %. As a result of interaction between fuel particles under irradiation, the amorphous phase such as $(U,Mo)Al_x$ is formed. The volume fraction of this phase could be estimated by the results of optical metallography. The fraction of $(U,Mo)Al_x$ phase increases almost by an order of magnitude (from 0.9 to 9.1 %) as burnup rises from 33 to 97 %.

Figs. 2b-d show the neutronograms for G96, G98 and G99 specimens obtained at room temperature and after annealing. These neutronograms have the same specific features as those given in Fig.2a. Besides, it should be noted that at annealing temperature above 350 °C the reflexes increase at angles 49.6 and 72.8, which correspond to the reflections (200) and (220) for UAl_3 .

Processing of the spectra shown in Figs. 2b-d allows to obtain the relationships between the content of the amorphous phase $(U,Mo)Al_x$ and the crystal phase UAl_3 which are illustrated in Fig.5. The percentage of any of these phases does not change until temperature is below 300°C. However, at annealing temperature above 350°C the specimens G96 and G98 demonstrate the noticeable amount of UAl_3 phase and the simultaneous decrease of the percentage of the $(U,Mo)Al_x$ amorphous phase. It is necessary to point out that the curves for $(U,Mo)Al_x$ have been plotted assuming that UAl_3 would be formed inside the former phase. At the same time it is possible that the UAl_3 crystal phase may be formed as a result of interaction between fuel particles and the Al matrix rather than as recrystallization of the $(U,Mo)Al_x$ amorphous phase.

Fig. 6 illustrates changes of UAl_3 content in the specimen under testing as a function of temperature. The figure also shows the calculated variations of the UAl_3 content which were obtained based on the equation derived from the data on heat test on interaction between unirradiated U-9%Mo alloy and commercial pure Al [4]:

$$\delta = 237.8 \cdot \tau^{0.40} \cdot \exp(-6310/RT);$$

where δ – thickness of $(U,Mo)Al_x$ layer, μm ; τ – time, h; T – absolute temperature, K.

In accordance with data given in Fig. 6, for the specimens G98 and G99 with relatively thick $(U,Mo)Al_x$ layers (11.2 and 9.0 μm , respectively) the noticeable percentage of UAl_3 phase, formed only due to thermal interaction between fuel particles and Al matrix, could be expected at temperature exceeding 450 °C. Thus, experimentally found values of UAl_3 percentage in G98 specimen at the annealing temperature in the range of 350-450 °C could be higher by almost an order of magnitude than the

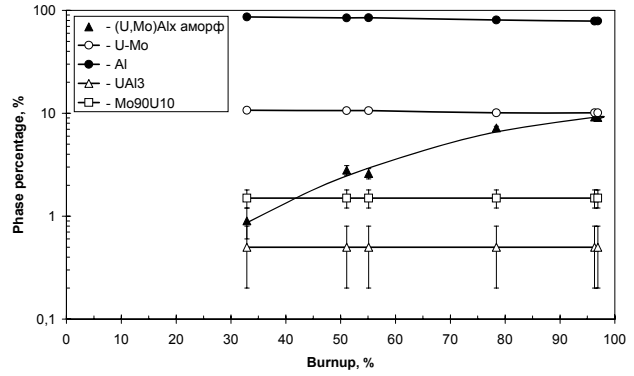


Fig.4. The effect of burnup on changing the content of Al, γ -UMo, UAl_3 and of the amorphous phase such as $(U,Mo)Al_x$

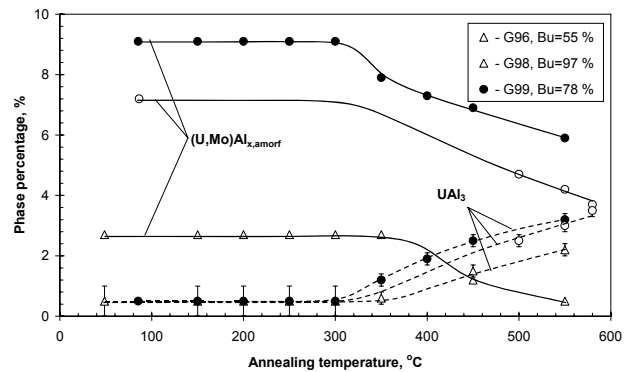


Fig. 5. The effect of annealing temperature on changing the content of the $(U,Mo)Al_{x, \text{amorph}}$ amorphous phase and UAl_3

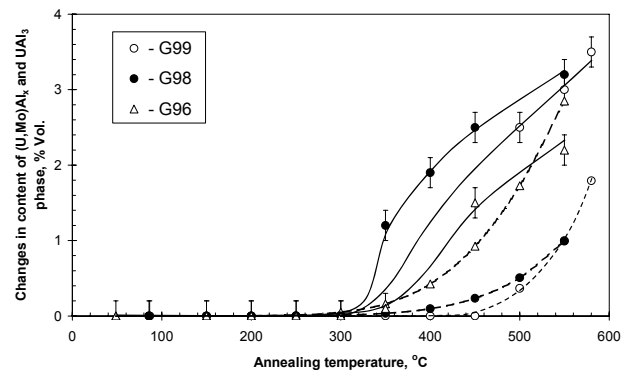


Fig. 6. The effect of annealing temperature on changes in percentage of UAl_3 phase: solid lines - experimental data, dashed lines – calculated values

expected values of UAl_3 percentage, if this phase would be the result of heating effect. Consequently, the occurrence of UAl_3 phase in G98 specimen can be explained only by the process of recrystallization in the $(U,Mo)Al_x$ amorphous phase. As is clear from the data in Fig.6, recrystallization of the $(U,Mo)Al_x$ amorphous phase begins in the temperature range between 300 and 350 °C. It is probably true also for the specimen G99 which was annealed only at 500-580 °C. However, the specimen G99 with less thick (by ~20 %) $(U,Mo)Al_x$ layer as compared with the specimen G98 has demonstrated almost equal percentage of UAl_3 phase at annealing temperature of 500 and 550 °C.

Unlike the specimens G98 and G99, the occurrence of noticeable amount of the heating UAl_3 phase could be found as early as at 350°C for the specimen G96 where thickness of $(U,Mo)Al_x$ layer was less (3.6 μm). However, percentage of this phase is smaller compared to percentage of UAl_3 phase obtained during the experiment in the temperature range from 350 to 450 °C. If temperature is above 500 °C, percentage of UAl_3 phase of heating origin and obtained experimentally is almost the same.

4. Conclusions

It has been revealed that the specimens of fuel elements in KM003 and KM004 FAs with U-9%Mo dispersion fuel are characterized by the following phase composition: γ -UMo, Al, UAl_3 , disordered Mo-based alloy $Mo_{90}U_{10}$ and amorphous phase with short-range order such as Al.

It has been demonstrated with accuracy of 1 % of the volume that tested specimens of FEs had no other crystal phases such as UAl_2 , UAl_4 and more complex structures based on U-Mo-Al system.

It has been experimentally proven that tested specimens contained either amorphous or finely crystalline phase with the size of coherently dispersed particles less than 30 Å.

The following relationships of the said crystal structures as function of burnup and annealing temperature have been established:

- Percentage of γ -U-9%Mo phase depends only slightly on burnup;
- Percentage of UAl_3 crystal phase does not exceed 0.5 % and is almost the same in the range of burnups from 33 to 97 %;
- Percentage of $Mo_{90}U_{10}$ phase is constant and equal to ~1.5 % and hardly depends on burnup;
- Percentage of amorphous phase such as $(U,Mo)Al_x$ depends on irradiation parameters and increases as fuel burnup rises;
- As annealing temperature increases, percentages of the main phases such as γ -UMo, $Mo_{90}U_{10}$ and Al practically do not change;
- In the range of annealing temperature from 150 to 300-350 °C percentages of the amorphous phase and UAl_3 practically do not change;
- Annealing at temperature above 300-350 °C results in crystallization of the amorphous phase with formation of the UAl_3 intermetallide phase. Percentage of the latter increases, as annealing temperature rises, being as high as ~30-50 % of the $(U,Mo)Al_x$ amorphous phase at annealing temperature of 550-580 °C.

Under the specified annealing temperatures ($T=150-580$ °C, $t=1$ h) the entire percentage of the $(U,Mo)Al_x$ amorphous phase could not be transformed to the crystal state.

It has been determined that temperature of 300 °C is the boundary value in terms of accelerated formation of the UAl_3 intermetallide phase.

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REMOVAL OF SPENT NUCLEAR FUEL FROM KURCHATOV INSTITUTE RESEARCH REACTORS FOR REPROCESSING: PROBLEMS AND PLANS

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ABSTRACT

The paper presents problems and main results of activities on removal of spent nuclear fuel (SNF) from research reactors of the Kurchatov Institute for reprocessing. Up to 1990, standard spent fuel assemblies (SFAs) from the Institute reactors were on a regular basis removed to Mayak enterprise for reprocessing. The scheme of removing the standard SFAs was based on the use of casks of TUK-19 type. Between 1990 and 2003, no work on SNF removal from the Institute was conducted, but in 2004-2005, three shipments of SFAs were completed. In 2006 the work on SNF removal was continued. A special feature of this shipment lay in the fact that new generation casks of TUK-128 type were used for transportation of the SFAs. Main design features of the TUK-19 and TUK-128 casks are described. The Institute spent fuel includes almost all kinds of nuclear fuel used in research reactors of Russian design. Classification of the Institute SNF according to Mayak requirements for acceptance of spent fuel for reprocessing is presented. In order to perform further activities on removal of SNF, a “Programme of Management of Research Reactor SNF Accumulated at the Kurchatov Institute Site” has been developed. The main activities on preparation of SFAs for removal, the procedure and schedule of SFA removal for reprocessing are described. It is expected that implementation of the Programme will establish a well-documented basis for organization of further activities on decommissioning of shutdown reactors at the Institute.

1. Introduction

Kurchatov Institute, established in 1943, is currently one of the largest scientific nuclear centres of Russia. Complex solution of the problem of safe and environmentally clean energy generation on the base of nuclear fission reactions, which required a considerable experimental base, was one of the Institute’s main activity areas.

The Institute’s experimental complex has operated the total of 12 research nuclear reactors of different types (the first commissioned in 1946), as well as about 20 critical and sub-critical experimental facilities and 3 “hot” material research laboratories for works with irradiated nuclear fuel.

Currently the Institute has 6 research reactors in operation. Another six reactors have been shut down in various years, and some of them have been dismantled, and some others now expect decommissioning. Research reactors, which form the Institute’s experimental reactor base, are listed in Table 1, together with their basic parameters.

For many years of its experimental reactors’ operation, the Institute has accumulated considerable amounts of spent nuclear fuel (SNF) on its site.

Currently over 5 tons of SNF (including over 900 spent fuel assemblies (SFAs) and their fragments) with total activity exceeding 10^{16} Bq (over 3×10^5 Ci) is stored in the Institute’s territory. This

circumstance determines the urgency of SNF removal from the Institute, the site of which is situated within the borders of Moscow.

Reactor	Reactor type	Dates of start-up/ reconstruction	Capacity/after reconstruction, MW	Status
F-1	Uranium-graphite reactor	1946	0.024	In operation
RFT	Channel-type uranium-graphite reactor	1952/1957	10.0/20.0	Partially dismantled (1962)
VVR-2	Tank-type water-water reactor	1954/1960	0.3/3.0	Dismantled (1983)
IRT	Pool-type reactor	1957	2.0	Dismantled (1979)
OR	Tank-type water-water reactor	1960/1983-86	0.3/0.3	In operation
MR	Channel-type in-pool reactor	1963/1967	20.0/50.0	Shut down (1993)
Romashka	High-temperature reactor with thermoelectric converter	1964	0.04	Dismantled (1967)
Gidra	Homogeneous solution pulse reactor	1972	0.01-30.0 MJ (in pulse)	In operation
Topaz-2	High-temperature reactor with thermionic converter	1973	0.1	Dismantled (1986)
IR-8	Pool-type uranium-graphite reactor	1981	5.0 (design - 8.0)	In operation
Argus	Homogeneous solution reactor	1981	0.02	In operation
Gamma	Tank-type water-water reactor	1982	0.125	In operation

Tab 1: Basic parameters of research reactors of the Kurchatov Institute

2. Storage conditions and classification of the Institute's SNF

SNF accumulated on the Institute's site is represented by spent fuel assemblies (SFAs) and fuel elements from: already dismantled RFT, IRT, Romashka, Topaz-2 and VVR-2 reactors; reconstructed OR reactor; and MR reactor, which was shut down for decommissioning.

SNF was removed to on-site temporary storage facilities created in the Institute's territory in order to meet the research reactors' operation requirements, and to assure safe management of spent fuel.

SNF from RFT, IRT and MR reactors is stored in the central "dry" storage facility situated on the main site of the Institute.

SNF from Romashka and Topaz-2 reactors is stored in the "dry" storage facility of the "R" complex, also situated on the main site of the Institute.

SNF from VVR-2 and OR reactors is stored in "aqueous" storage facility of the "Gas Plant" complex situated on the supplementary site of the Institute.

The technical condition of all temporary SNF storage facilities existing in the Institute's territory conforms to the contemporary requirements related to physical protection and nuclear/radiation safety assurance.

It should be said that the SNF accumulated in the Institute comprises almost all types of nuclear fuel ever used in research reactors of Russian design. This SNF includes not only the working (regular) SFAs of research reactors, but also experimental fuel assemblies and elements used in reactor research and tests of various structures, fuel compositions and construction materials. For this reason, the Institute's SNF is characterized by diverse structural features, fuel composition types, U-235 fuel enrichments, fuel burnup depths, cooling periods, and status of fuel assembly components and construction materials used. SFAs also include leaking fuel assemblies, as well as assemblies containing defective fuel.

Taking into account the requirements the spent fuel should meet to be accepted for reprocessing by PA Mayak, the Institute's SNF could be divided into the three following groups:

- reprocessable fuel, i.e., fuel meeting PA Mayak's acceptance requirements;
- conditionally reprocessable fuel, i.e., fuel not quite fit for reprocessing at PA Mayak, requiring repackaging into leak-tight canisters and baskets, as well as additional coordination of its acceptance for further reprocessing;
- irreprocessable fuel, including defective fuel, which requires long-term storage, since currently there is no reprocessing technology for it.

More details on the Institute's SNF are given in Table 2.

№	Reactor	SFA type	SFA					Fuel elements
			Reprocessable SNF			Irreprocessable SNF		
			completely	conditionally	defective	leak-tight	defective	
1	MR	Pilot	-	-	226	-	-	970
2	MR	Fuel element fragments	-	-	-	-	-	16
3	RFT	Working	-	-	-	108	90	-
4	IR-8	Working	8	-	-	-	-	-
5	Topaz-2	Working-1	-	151		-	-	-
6	Topaz-2	Working-2	-	26	18			
7	Ромашка	Working	-	-	-	11	-	-
8	VVR-2, OR	Working-1	155	-	39	-	-	-
9	VVR-2, OR	Working-2	-	68	17	-	-	-
10	Total spent fuel assemblies or elements		408		300	119	90	986
			708			209		
11	TOTAL		917					

Tab 2: Types and numbers of SFAs to be removed from the Institute's site (status of 2007)

3. Current status of works on SNF removal from the Institute's territory

Before 1990, the working SFAs of the Institute's research reactors have been regularly shipped for radiochemical reprocessing to the specialized enterprise – PA Mayak. Then the technological scheme of SFA transportation has been based on using old-generation TUK-19 shipping casks and TK-5 railway carriages.

Between 1990 and 2003, SNF removal works in the Institute have been frozen because of insufficient financing. In 2004 the Institute resumed SNF removal operations. In 2004-2005, three batches of the Institute's working SFAs were prepared and delivered to the destination on the basis of the restored shipping scheme using TUK-19 transportation casks.

In November 2004 and in March 2005, two special trains delivered 128 working SFAs from MR reactor (64 SFAs in each train) to PA Mayak.

In April 2005, the transportation of the third batch of 64 SFAs was prepared and carried out, again using TUK-19 transportation casks. SFAs sent by this train included spent fuel assemblies from the physical model of MR reactor.

The regular SNF loading and transportation system operating inside MR reactor premises was used for the transportation of MR physical model. In the preparatory process before MR physical model's SFA removal, the technical state and parameters of 70 assemblies were analysed, and 64 assemblies were

selected for transportation. A special certificate was then compiled for each assembly, indicating initial fuel parameters, fuel irradiation regimes and other data required by PA Mayak.

In 2006, the Institute continued its SNF removal activities, and shipped another 40 working SFAs from IR-8 reactor to PA Mayak.

This batch had a pilot status, because it used a new-generation TUK-128 shipping cask to contain and transport the fuel. That's why one of the principal goals of this SNF shipment was to confirm the efficiency of this new-generation shipping cask.

4. TUK-19: structural features and SNF loading/transportation scheme

TUK-19 shipping cask is intended for transporting research reactor SFAs, and includes the container No 19 and the basket. TUK-19 has the following basic technical parameters:

Transportation index	20
Mass, kg	4 750
Maximum U-235 fuel enrichment, %	below 90
Maximum residual power, W	below 112
Outside surface gamma dose equivalent rate, mrem/h	below 200
Surface temperature, °C	
- maximum	60
- minimum	-50

After SFAs were loaded, the container is filled with air or air/inert gas mixture. Structural layout of TUK-19 shipping cask is shown in Fig. 1.

TUK-19 shipping casks are transported on special TK-5 railway carriages, which provide vertical positioning, reliable fastening and transportation of eight TUK-19 shipping casks, and meet the technological requirements of SNF transportation.

TK-5 carriage is a four-wheel transporter with a body coated with stainless steel from the inside and partitioned by walls into one cargo section and two auxiliary rooms. The cargo section contains a frame with 8 cells, in which TUK-19 shipping casks are installed and fixed with special grips. The cargo section deck is made in form of two flaps opened using hydraulic or manual drive. The cargo section also contains decontamination solution and radioactive waste collecting tanks, the control panel and mechanisms opening the deck flaps, as well as spare parts and accessories.

The special train transporting the SNF in TUK-19 casks includes: a locomotive; a cover carriage; a carriage for PA Mayak escort team; an armed guard team carriage; two TK-5 carriages; and another cover carriage.

PA Mayak is an official owner of TK-5 carriages and TUK-19 casks.

All three shipments used the following technology of loading SFAs into TUK-19 and TUK-19 – into TK-5 carriages.

The upper lid was removed from TUK-19, and the cask was installed (using a crane) on a special rail-guided cart, together with the overload transfer cask. After that, TUK-19 and the transfer cask were delivered on the cart to SNF storage premises.

Then the protective plug was removed using the frame crane from the storage cell containing a metallic canister with SFAs to be shipped. Metallic canisters used for SFA storage in the storage cells are equipped with special grips. After removing the protective plug, the first operator, under the shady shielding, hooked the frame crane to the canister and then left the storage premises. The second operator, using video monitoring and frame crane remote control systems, removed the canister containing SFAs from the storage cell and loaded it into the transfer cask. After that, the first operator re-entered the storage area, unhooked the frame crane from the canister, hanged the SFA grip on the frame crane hook, hooked the SFA to it, and left the area again.

Then the SFA was removed – using the frame crane controlled remotely – from the canister placed into the transfer cask, and loaded into a fixed TUK-19 cask cell. Then three other SFAs were loaded into TUK-19 in accordance with the same order. After the last (fourth) SFA was loaded into TUK-19, the upper protective lid was installed on the cask using the frame crane, and loaded TUK-19 was removed from the storage premises on the rail-guided cart.

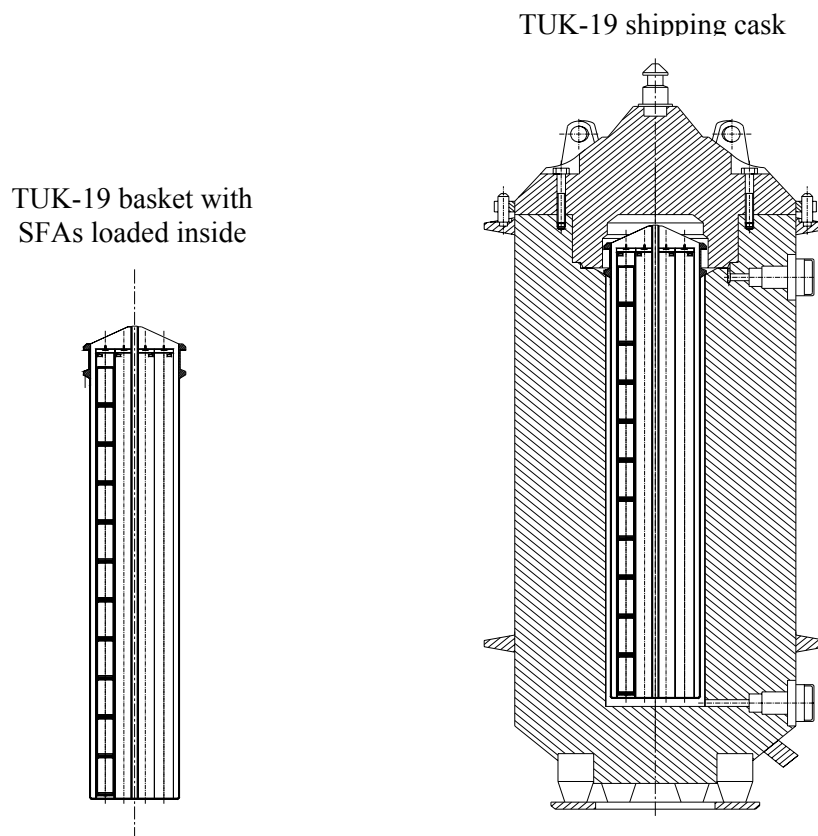


Fig 1. TUK-19 shipping cask layout

After all TUK-19 casks were loaded in accordance with the above technology, the loaded casks were installed by a crane on a special car, and successively delivered to the transfer site organized at the Institute's railway spur, where they were loaded into TK-5 carriage using the truck crane.

Use of the above technology of loading and transportation of the Institute's SNF in TUK-19 and TK-5 carriage had the following benefits:

- over 30 years of SNF transportation experience using TUK-19 and TK-5;
- high reliability of this scheme of SNF transportation to PA Mayak (over time, almost all SNF shipments using TUK-19 and TK-5 were performed in routine conditions; only rarely – in normal conditions; and never – in accident conditions);
- since PA Mayak is the SNF conveyor and TUK-19 and TK-5 owner, the specialists from this enterprise participate in SNF acceptance for reprocessing from its earliest stage;
- PA Mayak discharges TK-5 and TUK-19 on a regular basis, so no development or manufacturing of additional equipment is required;
- no additional reloading of TK-5 with loaded TUK-19 casks is required on the way.

5. TUK-128: structural features and SNF loading/transportation scheme

Unlike TUK-19 shipping cask used earlier, TUK-128 is a dual-purpose shipping cask: its structural features allow using it for both transportation and storage of SFAs (up to 50 years long). Besides, TUK-128 has a larger capacity, which makes it possible to load up to 20 SFAs into it (compare with 8 SFAs loaded into TUK-19).

TUK-128 shipping cask has the following basic technical parameters:

Outside diameter of container body, without dampers, mm	1120
Inner diameter of container body, mm	520
Container diameter including journals, mm	1310
Container cavity height, mm	960

Container height including lids, mm	1520
Container height, including top and bottom dampers, mm	2250
Container body mass, with journals and facing, kg	7857
Outside lid mass, kg	445
Internal lid mass, kg	410
Unloaded basket mass, kg	256
Loaded container mass, without dampers, kg	9227
Loaded container mass, with dampers, kg	11534
Container body material	high-strength, nodular graphite cast iron

TUK-128 shipping cask layout is shown on Fig. 2.

TUK-128 casks are transported in ISO containers, which undergo a special modification beforehand. The ISO container modification includes the fabrication of improved elements fastening these containers to railway flatcar, as well as TUK-128 retention into the ISO container.

It should be noted that, since the required reloading equipment was not yet ready, the standard transfer cask of IR-8 reactor and the standard loading technology previously used for TUK-19 were refined and successfully used in the pilot SFA batch shipped in TUK-128 casks. For that reason, in the framework of preparing the pilot SFA transportation in TUK-128 casks, as well as other transportation, technological and other auxiliary equipment, it became necessary to develop and use in the working process the following organizing and technical documents:

- On-site transportation process flow diagrams of TUK-128 shipping cask, ISO container, standard IR-8 transfer cask and technological equipment management;
- On-site process regulations of IR-8 research reactor SFAs' loading into TUK-128 shipping cask;
- On-site programme of "cold" tests of TUK-128 cask and of the device intended for SFA loading into TUK-128.

In course of the pilot transportation, the SFAs from IR-8 reactor were loaded into TUK-128, and TUK-128 – into ISO container, in accordance with the following technological sequence.

Two TUK-128 in two ISO containers were delivered on the Institute's site on trucks and installed in the reloading area organized at the Institute's railway spur.

With the help of the crane, TUK-128 casks were removed from ISO containers. After their dumpers were removed, the truck loader successively delivered TUK-128 casks to MR reactor building.

On the site near the MR reactor building, TUK-128 casks were installed on a special cart and successively delivered to the MR reactor hall, where their outside and internal lids were removed.

After that, a conductor funnel was installed on the fixed cell of TUK-128, in which a standard SFA from IR-8 reactor should be loaded.

Standard SFAs from IR-8 reactor were delivered to MR reactor hall in a standard IR-8 under-load transfer cask. To do that, the standard transfer cask was first delivered to IR-8 reactor hall and installed on the transfer hatch of this reactor's cooling pond. Then the operator, using a special manual grip, moved the standard SFA of IR-8 to be loaded into TUK-128 from its cooling pond cell and installed it into this pond's loading socket. After that, the grip of the standard transfer cask was put in its downward position and fixed to the assembly; then the grip with the SFA fixed to it was put in upward position and drawn into the standard transfer cask. After that, the protective gate of the standard transfer cask was closed, and the truck loader delivered the transfer cask with SFA inside to the site near the MR reactor building and then to MR reactor hall.

In the reactor hall, the standard transfer cask was installed above the fuelling channel of the shield cell in the reactor hall. After that, the protective gate of the standard transfer cask was opened, and the standard SFA of IR-8 reactor was installed on the bottom of the fuelling channel using the transfer cask grip. Then the grip of the transfer cask containing the SFA was unhooked, and the transfer cask was removed from the fuelling channel. After that, the first operator entered the MR reactor hall, manually fixed the special grip (hanged in advance on the SFA crane hook), and left the reactor hall. Then the second operator, using the crane, removed the SFA from the fuelling channel, moved the SFA to TUK-128, and installed it into a cell of this cask's basket through the conductor funnel.

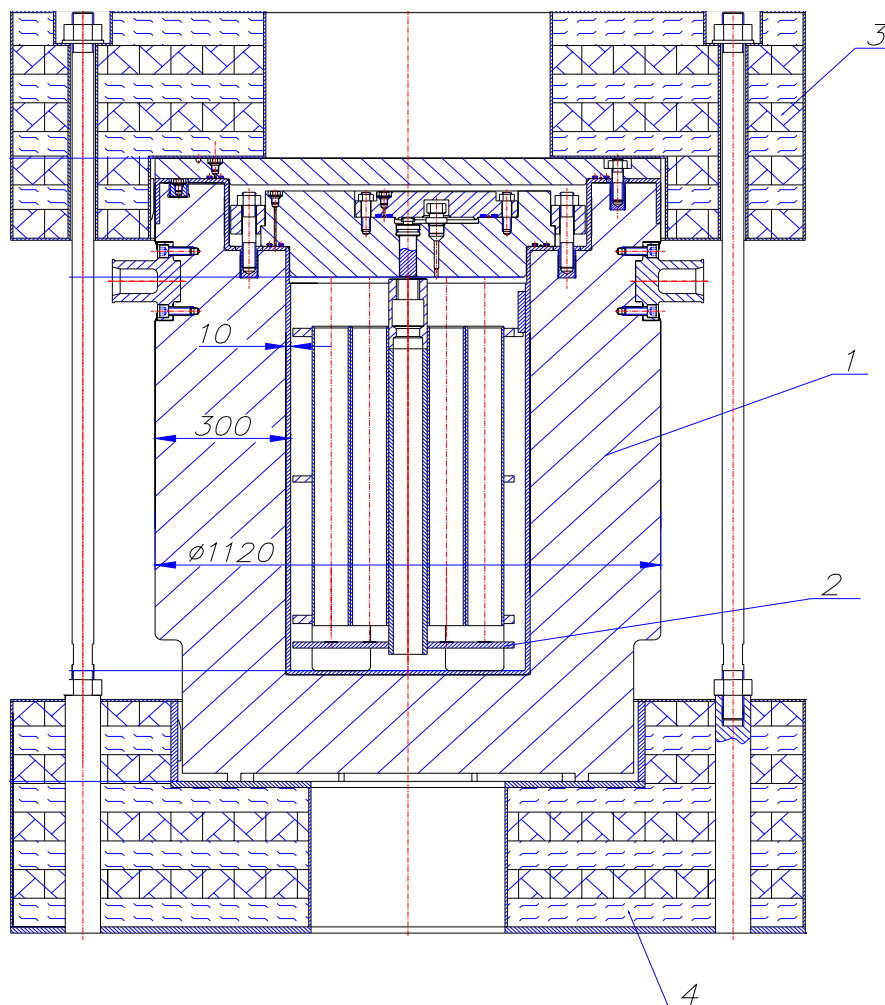


Fig 2. TUK-128 shipping cask layout:

1 – shipping cask; 2 – basket with SFAs; 3 – upper damper; 4 – lower damper

The grip was unhooked from the SFA automatically, after the SFA reached the bottom of the basket cell – thus, any further SFA movement was impossible. The conductor funnel was reinstalled on the next TUK-128 basket cell, and the standard transfer cask returned to the IR-8 reactor hall to receive a next SFA from this reactor.

After the twentieth (last) SFA was loaded into the cask basket, the internal and the outside protective lids were successively installed on TUK-128, and the cask's leak-tightness was checked. Then a special car delivered TUK-128 with SFAs inside to the transfer site at the Institute's railway spur, where the truck crane loaded TUK-128 into ISO container (already installed on a railway flatcar). Inside the container, TUK-128 was unfastened in accordance with the standard scheme.

6. Plans of further activities to remove SNF from the Institute's site

In order to organize and perform further activities on preparing and removing the SNF from the Institute, the "Programme of Management of Research Reactor SNF Accumulated at the Kurchatov

Institute Site” has been developed. The Programme determines the basic activity areas and their related timetables.

Besides the removal of the accumulated SNF, this Programme also provides for the liquidation of old SNF storage facilities, which have exhausted their design lifetime, at the Institute’s site.

The total complex of works included in this Programme would take 5 years to complete.

Financial estimation of activities to be performed in view of preparing and removing all SNF accumulated on the Institute’s site totals to about 30 million USD.

Basic areas and indicators of the activities to be performed under the Programme are summarized in Table 3.

SNF type	Number of SNF trains	Number of shipping casks			Number of SFAs		
		Cask type	In each train	Total	In each cask	In each train	Total
Pilot SFAs and fuel elements of MR reactor	4	TUK-19	16 casks	64 casks	3 SFAs or 4 canisters with fuel elements	3 trains with 48 SFAs in each, 1 train with 30 SFAs and 24 canisters with fuel elements	174 SFAs, and 491 fuel elements
Fuel elements, standard SFAs of VVR-2 and OR reactors; SFAs of IR-8 reactor	2	TUK-128	4 casks in 2 ISO-container	8 casks	7 casks with 20 canisters in each, 1 cask with 12 canisters and 8 SFAs	1 train with 80 canisters with fuel elements, and 1 train with 72 canisters with fuel elements	152 canisters with fuel elements, and 8 SFAs
SFAs from Topaz-2 reactor	4	TUK-19	3 trains with 16 casks in 2 TK-5 carriages, and 1 train with 8 casks in TK-5 carriage	56 casks	3 SFAs	3 trains with 48 SFAs in each, and 1 train with 24 SFAs	168 SFAs
SFAs from Romashka reactor	1	TUK-128	2 casks in ISO-container	2 casks	1 cask with 6 SFAs, and 1 cask with 5 SFAs	11 SFAs	11 SFAs
SFAs from RFT reactor	2	Dual-purpose reinforced concrete casks	3 casks	6 casks	5 casks with 36 SFAs, and 1 cask with 18 SFAs	1 train with 108 SFAs, and 1 train with 90 SFAs	198 SFAs

Table 3. Basic areas and indicators of the activities under the SNF Removal Programme

7. Conclusion

It is expected that implementation of the SNF Removal Programme described above would establish a well-documented basis needed to deploy further activities on the decommissioning of retired research reactors in the Kurchatov Institute.

In future, the Institute's experience in organizing and performing the research reactor SNF management activities could be used to solve similar tasks of removing the SNF from other research reactors, including those built abroad.



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