Program and Abstracts

Technical Meeting on

Assessment of Core Structural Materials and Surveillance Programme of Research Reactors

(TM-40035)

Organized by

International Atomic Energy Agency,
Vienna, Austria

14 – 18 June 2010
IAEA Headquarters, Vienna, Austria

FOREWORD

Research reactors have played and continue to play a key role in the development of the peaceful uses of nuclear energy and technology, particularly in various domains of research as, fundamental and applied science, industry, human health care and environmental studies, as well as nuclear energy applications and the development of nuclear science and technology related human resources. However, more than 50% of operating research reactors today are over 40 years old and continued operation has to be carefully assessed, especially from the structural materials point of view. In many instances data for the radiation-induced changes of research reactor core materials resulting from exposure to very high neutron fluences are not generally available. Further data is needed in order to evaluate the reliability of research reactor core components. Age-related degradation mechanisms can cause unplanned outages of the research reactors which could in many cases have been predicted by implementation of appropriate surveillance programs.

Typically, neutron-based irradiation programmes are carried out at research reactors for several purposes, with particular attention to structural and moderator materials and fuel samples from conventional nuclear power plants. The aim of such experiments is to determine the neutron fluence effects on mechanical properties of materials. Research and development of new advanced materials is also carried out and many member states with research reactors are involved or interested in such R&D projects. Unfortunately, very little information from analysed structural materials can be used as inputs to evaluating research reactor structural materials because of marked differences in the materials and operating environment between power reactors and research reactors. However, the methods used in such programs could be applied to research reactors, especially in the preparation of a predictive/preventive maintenance program supporting extended RR operation.

With the present international resurgence in interest in nuclear technologies the demand for research reactor irradiation services is expected to increase and older, heavily utilized reactors may be required to extend their operation to provide these services. The uncertainties regarding the reliability of their core structural materials consequently need to be addressed and discussed at a technical level.

CONTENTS

1. OBJECTIVES

The overall objective of this TM on “Assessment of Core Structural Materials and Surveillance Programme of Research Reactors” is to provide a forum to exchange best practice, practical experiences, maintenance procedures and information through scientific presentations and brainstorming discussions. It is anticipated that this meeting will achieve the following specific objectives:

- To exchange and review detailed technical information on the degradation, assessment and ageing management of core structural components and materials,
- To identify structural materials of interest for further investigation,
- To support the RR ageing management programs related to the material issues of critical components.
- Determine the need, viability, terms of reference and organisation for a new IAEA CRP on Assessment of Core Structural Materials, as well as identification of RRs and other institutions prepared to participate in a proposed CRP.
- To strengthen international and regional cooperation and networking between scientists from developing and developed countries with special emphasis on the transfer of knowledge

2. PROGRAM /SESSIONS

Tentative Agenda

Assessment of Core Structural Materials and Surveillance Programme of Research Reactors

(TM-40035)

14 – 18 June 2010

IAEA Headquarters, Vienna, Austria

Meeting room: M0E Press Room

Monday, 14 June 2010

08:30 - 09:00	Registration (Gate 1, IAEA)
09:00 - 09:30	Opening remarks and organizational issues (H.Forsstroem, DIR-NEFW, A.Zeman & E.Bradley, IAEA Scientific Secretaries)
09:30 - 10:00	NPP Surveillance program: NNP best practice – ageing management and surveillance program (E.Liszka, IAEA/NS)
10:00 - 10:30	Aluminum alloy selection for use as structural material in research reactor (R.Haddad, Argentina)
10:30 - 11:00	Beryllium reflector assemblies behavior in a MTR core according to FEM analysis (C. Durione, Argentina)
11:00 - 11:30	ANSTO experience in the assessment of the core structural materials from two research reactors: HIFAR and OPAL (R.P.Harrison, Australia)
11:30 - 12:00	Ageing Management and In-service Inspection of the 3MW Triga MK-IIResearch Reactor of Bangladesh (M.M. Haque, Bangladesh)
12:00 - 13:30	*Lunch Break*
13:30 - 14:00	Beam port leakage problem in the BAEC Triga Mark II research reactor and the corrective measures implemented (M.A.Zulquarnain, Bangladesh)
14:00 – 14:30	BR2 some aspects of structural mechanics (F. Joppen, Belgium)
14:30 - 15:00	Ageing Assessment of the Brazilian Research Reactor IEA-R1 Core Support Structures (S.Marcelino, Brazil)
15:00 – 15:30	Corrosion of aluminum alloys in research reactor cores: Process and assesment (L.V.Ramanathan, Brazil)
15:30 - 16:00	*Coffee break*
16:00 - 16:30	Implementation of a maintenance manual for ageing surveillance activities in Triga Mark II of CGEA / CREN K (V.L.Mwamba, Congo)
16:30 - 17:00	Irradiation surveillance capsules in the reactor of 300Mwe NPP (S Lin, China)
17:00 - 17:30	Day summary (all)

Tuesday, 15 June 2010

08:30 - 09:00	IAEA INIS presentation (N.Rashkova, IAEA/INIS)
09:00 - 09:30	Assessment of Core Structural Materials and Survilliance Programme of Research Reactors in Egypt (S.Mekhemar, Egypt)
09:30 - 10:00	In-reactor Experiments in Fast Breeder Test Reactor for Assessment of Core Structural Materials (S.Murugan, India)
10:00 - 10:30	In Service Inspection to Assure the Safety and Long Term Operation of Batan Reserach Reactors (S.Nitiswati, Indonesia)
10:30 - 11:00	*Coffee break*
11:00 - 11:30	In-service inspection of the HANARO Core Components (H.K.Kim, Korea)
11:30 – 12:00	Investigation of core structural materials for decommissioning of salaspils research reactor (A.Abramenkovs, Latvia)
12:00 - 13:30	*Lunch Break*
13:30 - 14:00	Inspection of Puspati Triga reactor (RTP) core and control rod (Z.Masood, Malaysia)
14:00 – 14:30	Complex degradation and ageing phenomena of research reactor core structural materials – experience at 14 MW Triga reactor from INR Pitesti (M.Ciocanescu, Romania)
14:30 - 15:00	HFR Petten Reactor Vessel Surveillance (B. Van der Schaaf, Netherlands)
15:00 - 15:30	Second NL presentation (N.Luzginova, Netherlands)
15:30 - 16:00	*Coffee break*
16:00 - 16:30	Material aspects of the IR-8 research reactor lifetime substantiation (D.Erak, Russia)
16:30 - 17:00	Theoretical ageing assessment of the SAFARI-1 core structure (A.D'Arcy, South Africa)
17:00 – 17:30	Day summary (all)

Wednesday, 16 June 2010

08:30 - 09:00	The evaluation the reliability of WWR-M Research Reactor Core Components Using Tensile Test Data (V.Revka, Ukraine)
09:00 - 09:30	Relicensing and Utilisation of the Penn State Breazeale Reactor (K.Unlu, USA)
09:30 - 10:00	Challenges and Lessons-learned During the Reactor Pool Repair at the Penn State Breazeale Reactor (K.Unlu, USA)
10:00 - 10:30	Electro-conductivity of aluminum alloys irradiated with neutrons (S.A.Baytekesov, Uzbekistan)
10:30 - 11:00	*Coffee break*
11:00 – 11:30	Assessment of Structural Materials Inside the Reactor Pool of the Dalat Research Reactor (D.Nguyen, Vietnam)
11:30– 12:00	Research Reactors nuclear safety issues (A.Shokr, IAEA/NS)
12:00 - 13:00	*Lunch Break*
13:00 - 15:30	Working group on surveillance and monitoring / CRP inputs (all)
15:30 - 16:00	*Coffee break*
16:00 - 17:30	Drafting of the report and the technical publication (all) (cont'd)

Thursday, 17 June 2010

8:30 - 12:00 Drafting of the report and the technical publication (all) (cont'd)

12:00 - 13:00 *Lunch Break*

13:00 - 16:30 Technical tour at Atom Institute Vienna (M.Villa, AI guide/ P.Salame, IAEA guide)

18:00 *Hospitality*

Friday, 18 June 2010

08:30 - 12:00 Finalization of the report

12:00-13:00 *Lunch Break*

13:00-14:30 Presentation of meeting output and follow-up

14:30-15:00 Meeting closure

3. ABSTRACTS

ALUMINIUM ALLOY SELECTION FOR USE AS STRUCTURAL MATERIAL IN RESEARCH REACTORS.

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Information on a range of aluminium alloys commonly used in the construction of research, experimental or production reactors (series 5XXX and 6XXX)) has been reviewed to assess their capability to sustain a full 40 years life period of use as RPV, reflector tank or other core component material, taking into account their corrosion resistance, fracture properties and irradiation damage.

The corrosion behaviour of the studied alloys is acceptable when used in nuclear grade water; they do not suffer of pitting corrosion, crevice corrosion or stress corrosion cracking (SCC). However, in certain conditions they may be susceptible to intergranular corrosion. Based on the available published information, it could be concluded that the alloy 6061-T6 would be the most suitable material to be employed in research reactor conditions (maximum working temperature 120 ºC, normal radiation level, water flow, etc.) and sustain 40 years of service if a correct water control is assured.

For working temperatures around 60 ºC, 6061/T6 alloy would undergo a moderate irradiation hardening degree and show good ductility retention for over 40 years. In case of short temperature excursions, this material will not suffer of any significant overaging. However, this parameter must in all circumstances be maintained below 150 ºC. From the activation point of view, the chromium content can be an additional advantage.

Due to lack of information, especially threshold propagation data, the mechanical behaviour and loss of some important properties could not be assessed for periods as long as 40 years; among them, resistance to fatigue and in service material toughness, which are RPV life limiting factors. Nevertheless, based on shorter experience, 6061-T6 alloy could be used, provided a proper surveillance programme is carried out, which with this material could be efficiently done, in view of its mechanical characteristics.

BERYLLIUM REFLECTOR ASSEMBLIES BEHAVIOR IN A MTR CORE ACCORDING TO FEM ANALYSIS

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Beryllium (Be) is used in numerous research reactors to moderate neutron energy and reflect neutrons back into the core, thus intensifying the thermal neutron flux. However, beryllium components are susceptible to mechanical property degradation and swelling due to fast neutron interaction, even at low temperatures. Such effects undermine the structural integrity of Be reflectors.

The present study analyses the lifespan of a square-section reflector assembly (RA) located in a nuclear reactor's core. A finite element method analysis (FEM) was applied to study strains and stresses taking place in the RA. A high fast neutron flux of ~1,00x1014 n. sec–1. cm–2 along different periods was assumed. Different constraint conditions were evaluated ("bottom nozzle fixed", "bottom nozzle fixed and top nozzle with zero displacement in x and y direction"). A non-homogeneous neutron flux was considered. The temperature parameter was ignored as its influence on swelling is negligible at operating temperatures (T<150°C).

Simulation results show that the RA will experience different degrees of strain due to the non-homogeneous neutron flux. The highest swelling will be experienced at the central zone of the RA and on the face closer to the fuel assemblies. The RA will bend as a result of such behavior, in a direction according to the constraint conditions. The stresses produced were studied and compared with the tensile strength.

ANSTO EXPERIENCE IN THE ASSESSMENT OF THE CORE STRUCTURAL MATERIALS FROM TWO RESEARCH REACTORS; HIFAR AND OPAL

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ANSTO has operated research reactors for more than 50 years. The first a DIDO class, 10 MW tank-type reactor, was commissioned in January 1958 and ceased operation in 2007. Its replacement, OPAL, is a 20 MW pool-type reactor and was commissioned in 2007 and is now in full operation. HIFAR was heavy water cooled and moderated with the inventory of coolant being contained principally in the reactor aluminium tank (RAT), which acted as a primary defence against a LOCA. OPAL is light water cooled and heavy water moderated, with the moderator being contained in a zirconium alloy reflector vessel that surrounds the core region.

There are significant differences between HIFAR and OPAL in the materials of construction and in the methods of evaluating core material aging. The core materials in HIFAR are high purity aluminium alloys (1080 and 1050 grade) while the OPAL reflector vessel is constructed from mainly Zircaloy-4 with some Zr-2.5Nb in areas where higher strength is required (the cold neutron source vacuum containment in particular) and Zircadyne (an industrial alloy with about 0.9% Hf) in areas where the neutron flux is low. The main cause of mechanical property changes in neutron-irradiated aluminium is the increase in tensile properties caused by thermal neutron transmutation silicon and the increase in dislocation density, both of which act to reduce the fracture toughness of the material. Other effects, such as annealing, also change properties. In the case of zirconium alloys the effects are due mainly to fast neutrons, with hardening occurring, also accompanied by a reduction in fracture properties. Baseline material data is important in the unirradiated condition as this provides a means of comparing property changes. Similarly, acceptance criteria for property changes are important and extensive use of finite element modelling has been used to determine the effect of property changes on the integrity of core components.

There was no surveillance program implemented for HIFAR from commissioning as these were not common at the time. When a remaining life assessment was commenced in the late 1980s it was necessary to infer the condition of the RAT from properties of materials from retired components. Test pieces were sectioned from safety rod liners and flux scan tubes and were tested in hot cells. Tensile and fracture properties were measured and the results used in a finite element model of the reactor to determine the effect of these changes on the integrity of the vessel. The mechanical test results were supplemented by hardness measurements made during the 1995 and 2000 major shut downs. These hardness values were converted to tensile strength using known correlations.

The OPAL reactor has been designed with a surveillance program from the commencement of operation. Test samples (tensile and small punch) are located in a dedicated irradiation position and will be removed at intervals during the life of the reactor. While it is not possible to achieve a fast fluence on the test pieces higher than the component being monitored (core chimney of the reflector vessel) due to the very compact core with no in-core irradiation facilities, the samples are otherwise in a similar environment. The other main component being monitored is the cold neutron source vacuum containment. The flux on the test pieces representing this component is well in advance of the actual component and will hence give information on change in mechanical properties that can be compared with the estimates made in the original design. Corrosion coupons are also included in the surveillance program, and include the common dissimilar metal couples aluminium-to-stainless steel and aluminium-to-zirconium alloy. These will be examined periodically to assess changes to corrosion potential.

The presentation will provide a description of the types of test pieces used for both reactors and will detail the future plans for undertaking mechanical testing of the irradiated materials at ANSTO; new hot cell facilities are currently being designed that will be ready to test the first sample removed from OPAL in about 2012. Also described will be the types of condition assessment tests, such as hardness and thickness measurements, that are essential complements to the mechanical testing of irradiated materials.

AGEING MANAGEMENT AND IN-SERVICE INSPECTION OF THE 3MW TRIGA MK-II RESEARCH REACTOR OF BANGLADESH

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Ageing management and in-service inspection are issues of significant concern, particularly for a facility like the BAEC TRIGA research reactor (RR), which has been in operation for more than twenty-four years. Since its inception the BAEC RR is being operated and maintained by the Reactor Operation & Maintenance Unit (ROMU) of AERE. During the past years, ROMU carried out several refurbishments, replacement and modification activities in the RR facility. These included- replacement of the damaged N-16 decay tank by a new one, replacement of the fouled shell and tube type heat exchanger replaced by a plate type one, installation of a chemical injection system for the secondary cooling system, modification of the piping layout of the primary and secondary cooling systems, modification of the supports for the primary cooling system piping, modification of the shielding arrangements around the N-16 decay tank, modification of the ECCS system, etc. ROMU is also responsible for the planning and implementation of the ageing management and in-service inspection programs for the RR facility. Under these programs vibration levels of different components at vulnerable locations of the systems contain rotating machines are monitored and recorded. Inspection and cleaning of the reactor tank internals are also carried out under the program. The paper focuses on the experience with major maintenance, repair, modification and replacement of the different systems of the RR as well as on the ageing management and in-service inspection aspects of the BAEC research reactor.

BEAM PORT LEAKAGE PROBLEM IN THE BAEC TRIGA MARK-II RESEARCH REACTOR AND THE CORRECTIVE MEASURES IMPLEMENTED

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The 3MW TRIGA Mark-II research reactor (RR) of Bangladesh Atomic Energy Commission (BAEC), which has been in operation since 1986, has four Beam Ports (BPs). One of the BPs, which remained plugged with the beam port plugs (mainly a graphite shield plug) for about 23 years, was found to be leaking when its plugs were removed for installation of a high resolution powder diffractometer in it. The leak, which was developed because of corrosion, was found in the aluminum part of the BP, which is located inside the reactor pool at a depth of about 8 m. Condensate accumulated in the annular space between the graphite plug and inner wall of the aluminum beam port initiated the corrosion. The graphite plug, which got stuck very tightly inside the BP, was removed using locally designed and fabricated hand tools. Leakage of water came to the notice of the reactor operation personnel a couple of days after the beam port had been cleared. Water leakage was stopped by installing a split type encirclement clam around the damaged part of the BP by some remote handling mechanisms, designed and fabricated locally. The paper presents in detail the description of the leakage problem and the remedial measures implemented so as to make the reactor operational again after about 16 months.

BR2: SOME ASPECTS OF STRUCTURAL MECHANICS

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This article discusses some of the important aspects of structural mechanics of BR2, namely: the follow-up of the beryllium matrix and of the reactor vessel and the seismic qualification

According the licence, a follow up program for the beryllium matrix is mandatory. This inspection is necessary because of the swelling of beryllium during irradiation. Due to this swelling, the individual beryllium blocks make contact between each other. This results in mechanical stresses and, because beryllium is a brittle material, cracks. At regular intervals inspection are made to evaluate the evolution of the swelling and the cracks. The maximum allowed neutron fluence is 6.4 10^{22} fast neutrons (energy more than 1 MeV) per cm². After this time the matrix has to be replaced. This has been done already twice.

During the replacement an inspection of the reactor pressure vessel must be made. Last inspection was performed in 1996, using ultrasonic and eddy current inspections. On this occasion a fracture mechanics calculation was made and the minimum allowed fracture toughness of material was determined. Since very little information on irradiated aluminium 5052-O is available, a number of samples were cut out of a second wall around the vessel. This aluminium had received nearly the fluence. Out of the samples test pieces (tensile and charpy) were made. A number of them were tested immediately, while the other was loaded in the reactor for accelerated irradiation. In this way a material follow up program was started. This program still continues.

During the period safety reassessment the authorities requested a seismic qualification. It was decided to make a full dynamic calculation, with input a 0.1g zero period peak ground acceleration and a regulatory guide 1.60 spectrum. The installation can withstand this earthquake, considered as a safe shutdown earthquake. A few structural reinforcements were necessary. The main ones were the primary piping outside the containment building, for which horizontal support were added, and a ventilation pipe bridge between the containment building and the ventilation building. This pipe bridge could have cause secondary damage while collapsing;

AGEING ASSESSMENT OF THE BRAZILIAN RESEARCH REACTOR IEA-R1 CORE SUPPORT STRUCTURES

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IEA-R1 is a research reactor developed by B&W (1) and operating in Ipen-Cnen/SP since 1957.

The core of the reactor is located 7 meters below the swimming pool water level and mounted over an eighty holes supporting plate. Over theses holes are located fuel and control elements, guides, and other structures, displaced in a way to optimize experimental arrangements. The main plate is supported by a frame that is connected to an overhead crane through aluminum profiles.

This work evaluates the support structure of the core and estimates its service life, taking into account the deformation of the aluminum alloy 6061 – T6 due to a critical integrated neutron flux of 0.5xE22 neutrons/cm^2.

It also estimates a change in the reactor power from 2 MW to 5 MW.

Considering the reactor neutron flux as the main life criteria to the aluminum profiles that support the core structure, we evaluate the remaining working hours of the frame.

Future works should include a visual inspection and an evaluation of the frame materials.

CORROSION OF ALUMINIUM ALLOYS IN RESEARCH REACTOR CORES – PROCESSES AND ASSESSMENT

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Around 250 research reactors (RR) are presently operating and these consist of several types, use fuels of different designs and have varying power levels and core configurations. The commonly used RR core materials include aluminium alloys, stainless steels and zirconium alloys. Aluminium alloys are used for cladding fuels and absorbers, targets, thermal columns and other irradiation facilities. Regardless of reactor type or application, most RR cores are surrounded by water that functions as a coolant, moderator and biological shielding. In this metal/environment system, aluminium alloys are prone to different types of corrosion, namely uniform, pitting, crevice and galvanic corrosion. Parameters that affect these forms of corrosion are water chemistry, temperature, solids in suspension, flow rate, bimetallic contacts and crevices. Much information is available about Al corrosion and is used in the design stage of core components. Nevertheless, many factors cause these ‘well designed components’ to corrode and these include transients in specific water parameters, synergism in the effects of certain water parameters, planned but inappropriate design changes, lack of or inappropriate surveillance practices and other site-specific constraints.

The corrosion resistance of Al alloys can be seriously impaired in the presence of very small quantities of chloride ions in the primary coolant water and by contact with other materials. Oxide growth on Al alloy surfaces depends on surface state, temperature and water parameters such as pH, conductivity, dissolved species, flow rate and heat flux. Aluminium alloys, like other metals that rely on surface oxide films for protection are particularly susceptible to pitting and crevice corrosion. Galvanic corrosion is driven by differences in electrochemical potential, ratio of surface areas of the metals in contact and the distance between the metals in contact.

This paper will present: (a) a brief overview of the different forms of corrosion of Al alloys and the effect of specific chemical and physical parameters on corrosion; (b) details of the on-going and previous corrosion surveillance programs in the IEA-R1 reactor at IPEN; (c) guidelines to plan and execute a corrosion surveillance programme to monitor and assess corrosion degradation of Al alloy core components in RR; (d) use of on-line and off-line measurements of specific parameters as well as visual inspection techniques to monitor the status of core components; (e) two case studies to highlight interpretation of data from on-line measurements, video imaging, off-line measurements (SEM/EDS/XRD) and a corrosion surveillance programme to explain fuel cladding degradation in the IEA-R1 research reactor at IPEN in Brazil.

IRRADIATION SURVEILLANCE CAPSULES IN THE REACTOR OF 300MWE NPP

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This paper introduces the irradiation surveillance capsules installed in the reactor of 300MWe NPP to monitoring the material degradation caused by irradiation. The contents of this paper include the function of the irradiation surveillance capsules, layout of the capsules in the reactor, specimens and samples inside the capsules, design verification tests and analyses and withdrawing schedule of the capsules.

IMPLEMENT OF A MAINTENANCE MANUAL FOR AGEING SURVEILLANCE ACTIVITIES IN TRIGA MK II OF CGEA/ CREN K.

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Pascal Kankunku-Katubadi, Mechanical Engineer and Reactor Supervisor,
Dieudonné Kombele Gelembo, Electronic Engineer and Reactor Manager.
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Since November 2004: Triga II is in extended shutdown condition.

For permitting to recover the safety and to keep the reactor alive while awaiting for the final decision on its future, the CGEA/CRENK developed this above programme.
The mains activities are:
(i)Visual inspection, (ii) Performance and operational test , (iii)Calibration, (iv) Leak test, (v)Mechanical maintenance (pump or/and motor lubrication, ...).
In addition, other ageing surveillance activities are implemented (according to action plan developed to ensure the safety and security of CREN-K RR under IAEA TC project ZAI9009), and two following types of tests are used : monthly tests and general (annually) tests.

ASSESSMENT OF CORE STRUCTURAL MATERIALS AND SURVEILLANCE PROGRAMME OF RESEARCH REACTORS IN EGYPT

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The main structural materials to be used in the reactor core, support structures are stainless steel, aluminum and zirconium alloys (zircadyne). Other materials are also used, for example such as polymers in seals and protective coating, and hafnium (HF) as absorber materials in the control rod plates. Stainless steel is used for the reactor pool. The mechanical properties of stainless steel alloys change when they are subjected to irradiation. The main phenomena observed are swelling and irradiation – induced creep. The swelling phenomenon depends on the operating temperature and neutron fluence. For the reactor facility, components will operate at temperature below 70 °C and are expected to see a lifetime fluence of approximately 1×10^{23} $n.cm^{-2}$ .these conditions are well below the conditions where swelling becomes significant. Stainless steels have strong resistance to corrosion over a wide range of environments and temperature. The reactor pool and primary circuit water is demineralized water with controlled low conductivity of less than 100 $\mu.sm^{-1}$. no failure mechanism is known under such process conditions.

Aluminum alloys will be used for the constructions of some reactor internals which working in radiation environment as their properties are well understood and show predictable behavior under such conditions. Aluminum is extensively used in water – cooled research reactors because of its low cross-section for the capture of thermal neutrons, excellent corrosion resistance and thermal conductivity.

Irradiation damage of polymers strongly depends on the fluence received by the materials. Irradiation effects of polymers also depend on their compositions and molecular structure. if the content of natural rubber is high, irradiation induces an increase in the tensile strength. Where the content of polypropylene is high, irradiation reduces the strength.

A materials surveillance plan has been developed and will be implemented from the commencement of reactor operation. The materials to be used to construct the components of the reactor facility are widely used in other nuclear reactors. Some of them, such as zircaloy-4, were developed for more demanding conditions (high temperature , pressure and doses) than those conditions to be present in the reactor.

Despite this, the surveillance plan will be implemented in order to integrity of the reactor facility components. The plan will be based on the guidelines provided in the ASTM –E 185 standard.

SURVEILLANCE PROGRAM AND AGING MANAGEMENT AT ETRR-II

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Surveillance program, periodic testing, and in-service inspection are tools to manage the aging of the research reactors and keep the operation of the research reactors within its design, safety analysis, and operational limits and conditions (OLC's) for the lifetime of the research reactor. This paper describes the surveillance program and aging management of the most important systems, structures, and components (SSC's) of the second research reactor of Egypt (ETRR-II). We believe that the exchange of information and discussion between the members states, about this topic, can lead to the best practice for the research reactor surveillance and aging management. It is important for the members states to try to find benchmarking measures for this topic to make the aging management and surveillance program measurable and achievable.

IN-REACTOR EXPERIMENTS IN FAST BREEDER TEST REACTOR FOR ASSESSMENT OF CORE STRUCTURAL MATERIALS

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Fast Breeder Test Reactor (FBTR) at Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam, India is a sodium cooled reactor with neutron flux level of the order of 10^{15} n/cm^2/s and temperature of coolant in the range of 650-790K (380-520°C). This reactor is being used as a test bed for the development of fuel and structural materials required for Indian Fast Reactor Programme. FBTR is also used as a test facility to carry out accelerated irradiation tests on thermal reactor structural materials.

In-reactor experiments on core structural materials are being carried out by subjecting prefabricated specimens to desired conditions of temperature and neutron fluence levels in FBTR. Non-instrumented irradiation capsules that can be loaded at any location of FBTR core are used for the experiments. Pressurised capsules of zirconium alloys have been developed and subjected to irradiation in FBTR to determine the irradiation creep rate of indigenously developed zirconium alloys (Zircaloy-2 and Zr-2.5%Nb alloy) for life assessment of pressure tubes of Indian Pressurised Heavy Water Reactors (PHWRs). Technology development of pressurised capsules was carried out at IGCAR. These pressurised capsules were filled with argon and a small fraction of helium at a high pressure (5.0-6.5 MPa at room temperature) in such a way that the target stresses were attained in the walls of the pressurised capsules at the desired temperature of irradiation in the reactor. FBTR was operated at a low power of 8 MWt during this irradiation campaign to have an inlet temperature of about 579 K (306°C) which was close to the temperature of pressure tubes at full power in PHWR. Irradiation of thirty pressurised capsules was carried out in FBTR using six irradiation capsules for different durations (upto 79 days). The fluence levels attained by the pressurised capsules were up to 1.1 x 1021 n/cm2 (E> 1 MeV) at temperatures of 579 to 592 K. Post-irradiation increase in diameter of the pressurised capsules was measured in the hot cells to determine the steady state irradiation creep rate that is used to estimate the permissible working life of pressure tubes in PHWR due to irradiation creep.

The grid plate of a fast reactor supports the core subassemblies and is made of stainless steel. It is a permanent core structure subjected to a low dose neutron irradiation over the life time of the reactor. To assess the change in mechanical properties of grid plate due to prolonged low dose exposure, an accelerated irradiation test with dose levels upto 2.6 dpa (at 350°C) has been carried out in FBTR on small size stainless steel tensile test specimens with composition similar to that of the grid plate material. The post irradiation examination results indicate that the material has enough residual ductility.

The mechanical properties of 20% cold worked Type 316 stainless steel irradiated to displacement damages upto 80 dpa at irradiation temperatures of 400 – 500°C have also been determined as a part of performance assessment and life extension of clad and wrapper materials of FBTR. The results indicate that cladding undergoes *significant decrease in both strength and ductility at displacement damages greater than 60 dpa, while tensile properties of* the wrapper evaluated using small specimen tests indicate significant hardening and a decrease in the uniform elongation.

This paper will discuss the salient features of irradiation facilities available at FBTR, the design and implementation of the above mentioned irradiation experiments in FBTR and the important results obtained during post irradiation examination.

IN SERVICE INSPECTION TO ASSURE THE SAFETY AND LONG TERM OPERATION OF BATAN RESEARCH REACTORS

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The safety of research reactor operation requires that system, structure and components (SSCs) of reactor should be maintained in a high level of reliability. Consequently, the SSCs should be regularly inspected, tested and maintained during operation. This paper describes implementation of surveillance program to SSCs through in service inspection (ISI) especially to critical components of BATAN research reactors such as reactor tank liner, heat exchanger, and cooling piping system regularly as part of ageing management program. The objectives of ISI are to observe and evaluate the condition of BATAN research reactors includes detecting any possible welding defect and determining whether any degradation/deterioration of reactor components that have significant impact to the safety occurred. Root cause analysis of component degradation/deterioration and remedial action are also discussed. The safety status of BATAN research reactors have been obtained through inspection utilizing NDT techniques to ensure that the reactor will remain safe for long term operation throughout its service life.

Key words: BATAN, surveillance program, ageing management program, in service inspection

AGEING MANAGEMENT FOR RESEARCH REACTORS IN INDONESIA

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Ageing management in research reactors activities are carried out by doing some evaluations and assessment of the documents related to research reactors ageing. Some activities related to ageing have been performed by operating organizations as categorization of System Structure and Components (SSCs) important to safety related to ageing as well as in-service inspection for some components.The measurement of reactor tank thickness as well as visual examination using under water camera has been done. Water chemistry, such as pH were keep to avoid tank corrosion. Some equipments related to visual inspection were used with the help of expert program during IAEA Mission Statement C3-INS/9/022-05-01. Some examples in RSG-GAS of such activities related to ageing of components are presented in some tables. BAPETEN as the Regulatory Body have been established the Regulation for research reactors related to ageing management by decree of the BAPETEN Chairman No. 8 year 2008 on Provision for the Safety of Ageing Management. The Regulation carry out regulatory reviews, inspections and assessments to determine the effectiveness of ageing management programmes as well as oversee that corrective actions are taken by the licensee if unsafe or potentially unsafe conditions are detected. Some implementations have to be taken due to the reactor ageing.

IN-SERVICE INSPECTION OF THE HANARO CORE COMPONENTS

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The inner shell of the reflector vessel surrounding the core is the most critical part from the viewpoint of a neutron irradiation. The neutron irradiation embrittles the reflector vessel and induces its dimensional change. The periodic measurement of the dimensional changes in the vertical straightness of the inner shell should be considered as one of the in-service inspections.

According to an analysis, it was estimated that the inner shell would deform by a maximum of 0.84 mm toward the core center due to creep and swelling during a reactor operation period of 20 years. Even though the inner shell was properly designed and fabricated adequately to meet the design limits, this deformation should be checked by 10 year period in-service inspection to confirm not only the structural integrity but also the physical clearance with the neighboring fuel channels and control and shutoff rods. The periodical dimensional measurement of the inner shell, control rods and its related guide tubes are the main in-service inspections of the core components to confirm the structural integrity and the operation clearances. The thorough measurement of the straightness of the inner shell in August 2004, which was the first measurement after 9 years of reactor operation, showed a deformation of a maximum of 0.26 mm toward the core center which is much smaller than the prediction of the original design analysis [1].

The control absorber or shutoff rod moving up and down is guided by two guide tubes, i. e. a shroud tube at the top and a flow tube at the bottom of their pass. The potential change of the clearance between an absorber rod and its guide tubes is also one of the important developments to consider for ensuring the safe insertion of the absorber rod into the core, as the diameters of the absorber rod and its related guide tubes could change due to neutron bombardment. Therefore their physical dimensions should be monitored periodically to confirm a certain clearances as necessary for the safe insertion. The periodic in-service inspection has been performed to confirm the clearances. The diameters of the absorber rods, flow tubes and shrouds were measured in 2006 and showed properly maintained clearances between the neighbor components.

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INVESTIGATION OF CORE STRUCTURAL MATERIALS FOR DECOMMISSIONING OF SALASPILS RESEARCH REACTOR.

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In May 1995, the Latvian government decided to shut down the Research Reactor Salaspils (SRR) and to dispense with nuclear energy in future. The reactor is out of operation since July 1998. A conceptual study for the decommissioning of SRR has been carried out by Noell-KRC-Energie- und Umwelttechnik GmbH at 1998-1999 years. The Latvian government decided in October 26 1999 to start the direct dismantling to “green field" in 2001 year. The characterization of reactor’s systems for decommissioning purposes was performed using computational methods and analysis of samples. The experience of utilization of irradiated reactor core construction materials is discussed in paper. The main radionuclides and its distribution in core elements was determined and discussed in the paper.

The change of reactor’s pool cladding during reconstruction of research reactor IRT-5000 in 1974 was caused by corrosion of aluminum tank. The replacement of aluminum core internals with stainless steel components significantly increases the content of radionuclides, especially ^{60}Co amount, in reactor’s pool components and creates additional problems for decommissioning of the research reactor.

The main problems for decommissioning activities of Salaspils research reactor are connected with the rather complicated design of the bioshield. It consists from different ferroconcrete parts with density up to the 6.5 tons/m^3. The steel roads constructions around the neutron beams significantly hinder investigation of bioshield composition and radionuclides content in the bioshield.

The additional efforts were devoted to collecting all information and to characterization of special materials –beryllium reflectors, graphite from thermal column and lead shield for gamma irradiator in the reactor’s pool. Together with the experts from IAEA the final solution for the safe management of the special materials were elaborated. The corresponding activities plans were prepared and approved by the regulatory body.

INSPECTION OF PUSPATI TRIGA REACTOR (RTP) CORE AND CONTROL ROD

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The 1 MW PUSPATI TRIGA Reactor (RTP), located at Malaysian Nuclear Agency has been operated since its first criticality on 28 June 1982. The RTP uses uranium zirconium hydride fuel enriched to about 20% of U-235. The RTP has four control rods made up of boron carbide where three are fuel-followers and one is an air-follower. The aluminium cylindrical core can accommodate up to 127 fuel elements while the reflector surrounding it is made from high purity graphite. Since, the reactor power is relatively small, natural convection is used for cooling. Light water is used both as a coolant and as well as a moderator. Visual inspection of the core, fuel and control rods are carried out routinely to ascertain their integrity. An underwater camera and boroscope was used to visually inspect the top grid plate of the core as well as the control rods. No visible defect was detected at the top grid plate however, two of the fuel-follower control rods had blemishes on its surface. This paper will describe the findings of the visual inspection as well as corrective actions taken.

Keywords: TRIGA reactor, visual inspection, core, control rod

COMPLEX DEGRADATION AND AGEING PHENOMENA OF RESEARCH REACTOR CORE STRUCTURAL MATERIALS - EXPERIENCE AT 14 MW TRIGA REACTOR FROM INR PITESTI

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The 14 MW TRIGA Research Reactor designed in the early '70s is a relative new research reactor with an operational experience of 30 years.

The specific design of reactor core objectives, were to manufacture, build and operate a flexible structure which incorporate previous experience of pool type research reactors. Aluminum alloy 6061 and stainless steel are only materials used for core structural components, which are all easily remotely removable and replaceable by simple hand tools. Properties of those categories of materials were well characterized / known for many other reactors predecessors, and no special criteria or preliminary tests were performed. The mechanical core structure is presented in the paper and designed procedure for periodic testing and inspection is also described. In spite of well known materials properties, the behavior uncertainties of those materials in each reactor case may have special aspects related to design of components, manufacturing technologies, surface finishing and processing, quality control methods, price of specific components, complex conditions in core and vicinity, history of operation, inspection and verification of components, radioactive waste characterization at the end of life of components.

Limited assessment of materials properties and suitability for certain application without considering the each individual component load, exposure and life time, may produce limited information on material itself, in fact the issue is the selection criteria for a standard material suitable for a certain application and consequent failure of components. The degradation and ageing are specific to components starting from design, manufacturing technology and expected life when the component should be replaced.

The paper presents the practical experience on maintenance requirements specific to TRIGA core components and some techniques of material investigations available at Institute for Nuclear Research Pitesti Post Irradiation Laboratory as well as in the Materials Development and Research Department. Some consideration concerning correlation between the reactor safety and materials or component conditions are also presented.

MATERIALS ASPECTS OF THE IR-8 RESEARCH REACTOR LIFETIME SUBSTANTIATION

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Commissioning of IR-8 research reactor at the Russian Research Center "Kurchatov Institute" (RRC "KI") was carried out in 1981 with thermal capacity ~8 MW. IR-8 is pool-type reactor and is used for investigations in the field of nuclear physics, solid state physics, radiation chemistry, material science and in biology (production of isotopes).

Beryllium is used in IR-8 reactor as a reflector. The reflector consists of two parts: internal one is formed by removable beryllium blocks and external one is formed by prismatic beryllium blocks with holes for beam tubes and vertical experimental channels. The reflector is fixed in the vessel. The reactor itself is located on the support platform. The reactor vessel and support platform are made from aluminium alloy SAV-1.

The IR-8 reactor is located in the pool with water (its depth is 11 meters). The pool walls are faced by stainless steel. There are 12 beam horizontal and 30 vertical channels at IR-8 reactor. Each of horizontal channels has the zirconium alloy (Zr-2,5%Nb) bottom which adjoins to the reactor core.

When IR-8 research reactor was put in operation the designed operation lifetime was determined as 30 years. At the present time the work on substantiation of IR-8 research reactor lifetime to prolong its operation for next 10-15 years is carrying out.

The most responsible for IR-8 safety operation structural components are the support platform and channel bottom adjoining to reactor core. There is no special surveillance program for structural core materials at IR-8. At the same time RRC «KI» has substantiation testing results to confirm the possibility of program. Main objectives of this program are defined as follows:

– assessment of properties degradation and ageing of the core structual components and materials;

– experimental-calculation investigations to confirm the substantial reliability for the most responsible structural component of the IR-8 research reactor using available in RRC «KI» data on mechanical properties and corrosion resistance of the aluminium alloy (SAV-1) and zirconium alloy (Zr-Nb 2,5%);

– assessment of properties changing for used structural materials to substantiate IR-8 lifetime extension.

THEORETICAL AGEING ASSESSMENT OF THE SAFARI-1 CORE STRUCTURE

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The SAFARI-1 Research Reactor is a 20 MW high-flux MTR and has been continuously operational for more than 45 years. The reactor core is supported in an Aluminium 5052-O vessel that forms part of the pressure boundary of the reactor primary cooling system. An ageing assessment of the reactor vessel and core support structure is currently underway to determine the condition of the reactor vessel, to establish a date for the projected end-of-life thereof and to examine options for life extension. This paper reports on some of the results of an assessment of the neutron fluence in the core structure, with particular attention to the north core face that forms part of the primary pressure boundary. The assessment includes an evaluation of the material composition changes due to neutron capture and a comparison with the results of destructive tests conducted on the replaced reactor vessel of a similar reactor. The paper also examines the strength properties of the core structure with reference to published results of high neutron fluence tests conducted at ORNL in the USA (on the same material), to determine the effects of neutron damage to the microstructure and tensile properties of the material. The paper further discusses work in progress on analysing the fatigue strength in terms of the history of the core structure, its frequency response characteristics, and crack propagation at a hypothetical "flaw" at the most highly stressed weld location.

HFR PETTEN REACTOR VESSEL SURVEILLANCE

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The HFR Petten was designed and built in the fifties of the 20th century as a research and materials test reactor. It started operation in 1961 with 20 MW thermal power which was gradually increased to 45 MW in 1970. The licensing authorities requested an accelerated material irradiation to assess the reactor vessel material degradation. The displacement damage from fast neutrons and the effects of the silicon formed from transmutation of aluminum atoms by thermal neutron interaction reduce the tensile ductility and increase the strength. The subsequent destructive tensile test results, used to determine the state of embrittlement of the vessel core box, are discussed together with the optical and electron microscopy providing further information on the radiation damage in the vessel material. The end of life of the first HFR reactor vessel was reached in the early eighties. The relevance of the fracture toughness property test results on the hot spots of the first vessel for the surveillance program for the second vessel is highlighted.

The first HFR reactor vessel replacement was accomplished in 1984. The surveillance program for the second vessel aimed for the measurement of the fracture toughness right from the start of its operation, based on the experience built up earlier. The surveillance program has been set up to measure the property degradation of samples in advance of the hot spot vessel areas. The importance of the allowable defect size in the vessel and the minimum fracture mechanics values are the key parameters for the surveillance program of the second vessel, as will be explained. The selected in-service inspection procedure of the vessel allows detection of defects well below the critical size for fast fracture. The in-service inspection also assesses the development of indications developed during manufacturing.

The single lead parameter for the degradation in the HFR vessel is presently the silicon content that increases most rapidly in the pool-side of the core-box. The decreasing importance of the ratio of thermal to fast neutron flux ratio as a factor for the damage influence will be discussed. The next phase of the program resulting from this consideration will be presented. The consequences of the concentration on silicon content for the surveillance program of the successor to the HFR, PALLAS, will be indicated.

THE EVALUATION THE RELIABILITY OF WWR-M RESEARCH REACTOR CORE COMPONENTS USING TENSILE TEST DATA

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WWR-M research reactor which is situated on the territory of Institute for Nuclear Research (Kiev, Ukraine) was brought into operation in 1960. At the time of reactor commissioning the surveillance program was not provided for the estimation of changes in mechanical properties of the structural material due to irradiation. To evaluate the reliability of WWR-M research reactor core components and extend a reactor service life the static tension test was performed in 2001. Main concern was a significant radiation-induced decrease of plasticity for an aluminum alloy CAB-1 which the reactor tank and internals are made of.

In this paper the tension test data for an aluminum alloy CAB-1 is analyzed. For the tension tests the ring specimens cut from the reactor control system component (a channel of the automatic regulator) were used. The tested material was irradiated in the reactor core with a maximum fast (E > 0.8 MeV) neutron flux of $0.7\cdot10^{18}$ $n/m^2\cdot sec$ and a 50°C temperature from the criticality date (1961) until 2000. The max fluence was $2.27\cdot10^{26}$ n/m^2. The specimens were tested at the temperature of 20°C, 60°C and 110°C.

The radiation hardening of the aluminum alloy CAB-1 was revealed. A tensile strength of irradiated alloy is about 260 MPa at the room test temperature. The long term irradiation results in a significant decrease of plasticity for the material tested. A total elongation was ~ 2 % only for all test temperatures. The low plasticity could affect the reliability of WWR-M reactor structural material. However, the fracture toughness analysis has shown that integrity of the reactor core component (in particular a support grid) is ensured and the license could be renewed.

RELICENSING AND UTILIZATIONS OF THE PENN STATE BREAZEALE REACTOR

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The Pennsylvania State University has operated a research reactor at the Radiation Science and Engineering Center (RSEC) on its University Park campus since 1955. The Penn Sate Breazeale Reactor (PSBR) is now the second longest continuously operated research reactor in the world and the first to be licensed by the Atomic Energy Commission, the forerunner of the US Nuclear Regulatory Commission (NRC). In 2009, the NRC relicensed the Penn State reactor for an additional twenty years of operation, the only U.S. reactor to undergo a second license extension. The PSU license renewal application package included the updating of the licensing basis documents and a review of the safety analysis for the reactor. The Penn State reactor uses a special type of TRIGA fuel that differs from the standard loadings. This required the RSEC staff to research many diverse resources, many from the 1960s and 1970s, to provide the basis for the safety analysis.

The mandate of the RSEC is to pursue research and development activities, educate students and provide a service to the region, the nation and the world. The degree to which a facility can remain relevant in the twenty-first century is inextricably tied to these goals. The RSEC excels in many research areas with the existing facilities and capabilities and utilized by internal and external entities over the years. An expansion of the RSEC facilities is planned. The planned expansion of the current neutron beam experimental area and the upgrade of the neutron beam ports at the RSEC will accommodate the state-of-the-art experimental equipment required for cutting-edge research and for graduate education utilizing nuclear techniques in nuclear, material and biological sciences. The planned expansion will make Penn State the leader in the revitalization of nuclear science and engineering in the U.S. by establishing state-of-the-art neutron beam facilities that take full advantage of the existing PSBR and RSEC capabilities. The RSEC has been able to maintain safety and reliability during the last 55 years while expanding its capabilities for research, education and service.

CHALLENGES AND LESSONS-LEARNED DURING THE REACTOR POOL REPAIR AT THE PENN STATE BREAZEALE REACTOR

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On October 10, 2007, operators of the Penn State Breazeale Reactor (PSBR) observed the reactor pool level had decreased more than expected over the weekend. Upon further investigation, the staff confirmed that a small leak of 10 gallons had developed in the 52 year old unlined pool. The staff immediately informed the appropriate regulatory authorities and set about finding a solution. Over the next six weeks, the reactor staff worked with University personnel, contractors and regulators to fix the leak and return the facility to normal operation.

The Penn State reactor was the first university research reactor licensed by the Atomic Energy Commission. The facility has had several minor pool leaks since its construction in 1955, with the latest occurring in 1976. Each time, the leak was located and repaired with concrete and an epoxy coating. The 2007 leak repair was more extensive involving three-step process that required hydro-lazing the pool wall, removing old sealant, and covering the areas with an epoxy concrete. After these processes the entire pool wall surface was sealed with a polyurea coating.

The response from the State of Pennsylvania and US-NRC regulatory authorities was much more involved than earlier leak events. Although public risk was never an issue, the US-NRC immediately dispatched inspectors to the facility so that senior officials could be knowledgeable and responsive to the public's information needs. Additionally, the University set up a team to provide the public and news organizations with ongoing status of the investigation and repair activities. This team allowed the reactor staff to remain focused on the technical aspects of the repair and interface with the regulators. The PSBR pool leak detection and repair will be discussed. Also, PSBR administration and staff, other PSBR functions, coordination with US-NRC and the state of Pennsylvania officials will be presented and experiences gained from this event will be shared with the other university reactor community.

ELECTROCONDUCTIVITY OF ALUMINIUM ALLOYS IRRADIATED WITH NEUTRONS

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Influence of ionizing radiation, temperature and product of water radiolysis on fuel elements materials at the nuclear reactor during operation are considered in the article. Examinations of reactor irradiation influence on electrical conductivity and change of linear dimensions of SAV-1 and AMG-2 aluminum alloys are carried out.
In view of small δ (tensile strain), it is possible to draw a conclusion on possibility of application of investigated aluminum alloys to fluences of 10^{20} cm^{-2} without appreciable stability losses.

ASSESSMENT OF STRUCTURAL MATERIALS INSIDE THE REACTOR POOL OF THE DALAT RESEARCH REACTOR

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Originally the Dalat Nuclear Research Reactor (DNRR) was a 250-kW TRIGA MARK II reactor, started building from early 1960s and achieved the first criticality on February 26, 1963. During the 1982-1984 period, the reactor was reconstructed and upgraded to 500kW, and restarted operation on March 20, 1984.

From the original TRIGA reactor, only the pool liner, beam ports, thermal columns, and graphite reflector have been remained. The structural materials of pool liner and other components of TRIGA were made of aluminum alloy 6061 and aluminum cladding fuel assemblies. Some other parts, such as reactor core, irradiation rotary rack around the core, vertical irradiation facilities, etc. were replaced by the former Soviet Union's design with structural materials of aluminum alloy CAV-1. WWR-M2 fuel assemblies of U-Al alloy 36% and 19.75% ^{235}U enrichment and aluminum cladding have been used.

In its original version, the reactor was setting upon an all-welded aluminum frame supported by four legs attached to the bottom of the pool. After the modification made, the new core is now suspended from the top of the pool liner by means of three aluminum concentric cylindrical shells. The upper one has a diameter of 1.9m, a length of 3.5m and a thickness of 10mm. This shell prevents from any visual access to the upper part of the pool liner, but is provided with some holes to facilitate water circulation in the 4cm gap between itself and the reactor pool liner. The lower cylindrical shells act as an extracting well for water circulation.

As reactor has been operated at low power of 500 kW, it was no any problem with degradation of core structural materials due to neutron irradiation and thermal heat, but there are some ageing issues with aluminum liner and other structures (for example, corrosion of tightening-up steel bolt in the fourth beam port and flood of neutron detector housing) inside the reactor pool.

In this report, the authors give an overview and assessment of structural materials inside the reactor pool and present some recent results of the visual corrosion observation of the pool liner and components inside the pool of DNRR by using a high-resolution color camera and direct examination of aluminum samples used for monitoring corrosion.

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