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Project Experiences in Research Reactor Ageing Management, Modernization and Refurbishment

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PROJECT EXPERIENCES IN RESEARCH REACTOR AGEING MANAGEMENT, MODERNIZATION AND REFURBISHMENT

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IAEA-TECDOC-1748

PROJECT EXPERIENCES IN RESEARCH REACTOR AGEING MANAGEMENT, MODERNIZATION AND REFURBISHMENT

REPORT OF A TECHNICAL MEETING ON RESEARCH REACTOR AGEING MANAGEMENT, MODERNIZATION AND REFURBISHMENT

INTERNATIONAL ATOMIC ENERGY AGENCY
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FOREWORD

Research reactors have played an important role in several scientific fields for around 60 years: in the development of nuclear science and technology; in the valuable generation of radioisotopes for various applications; and in the development of human resources and skills. Moreover, research reactors have been effectively utilized to support sustainable development in more than 60 countries worldwide.

More than half of all operating research reactors are now over 40 years old, with many exceeding their originally conceived design life. The majority of operating research reactors face challenges due to the negative impacts of component and system ageing, which manifest in a number of forms. This situation was highlighted by a serious medical isotope supply crisis which peaked in mid-2010, when several major producing reactors underwent prolonged shutdowns due to extensive necessary overhauls of various systems.

Several facilities have established a proactive systematic approach to managing ageing or mitigating its impact on safety and availability of isotopes. Others have tried to prevent or remedy the drawbacks of ageing on a case by case basis. Overall, a large body of knowledge related to ageing issues exists in many Member States. Collecting and sharing this information within the research reactor community can provide a solid foundation to develop a more systematic approach — that is, an ageing management programme to prevent negative consequences of ageing on the safety, and the operability and lifetime of operating, or even future, reactors. It may also help organizations to manage research reactors that have been in an extended shutdown state by ensuring that any required systems are operated and maintained in a safe manner prior to final decommissioning and disposal of fuel to safe storage facilities.

Sharing experiences from projects undertaken to refurbish or replace equipment and systems, satisfy safety and regulatory requirements, improve performance, or provide new products and services to existing or potential users and customers, as well as share information on the technical details of the work involved, will allow other organizations contemplating similar work to consider and better understand their own challenges. Such understanding can help to optimize future planning in terms of budget, schedule and resource expectations, and to enable informed decision making.

The IAEA is working to systematically collect existing knowledge on research reactor ageing management, modernization and refurbishment to share within the community of reactor owners, operators and oversight authorities. To this end, experts from the research reactor community gathered at IAEA headquarters from 10–14 October 2011 for the Technical Meeting on Research Reactor Ageing, Modernization and Refurbishment. The outcomes of the meeting and the submissions of the participants are documented in this publication.

The IAEA is grateful to all contributors to the drafting and review of the publication and wishes to express particular acknowledgement of the contributions made by E. Lau (United States of America). The IAEA officers responsible for this publication were E. Bradley and N.D. Peld of the Division of Nuclear Fuel Cycle and Waste Management and A.M. Shokr of the Division of Nuclear Installation Safety.

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SUMMARY

With the majority of research reactors having more than 40 years of operating experience, maintenance, modernization and refurbishment is increasingly more important for safe and viable operation. On the other hand, the consequences of operating conditions upon reactor structures, systems and components (SSCs) have been observed and studied, resulting in numerous techniques and regimes for controlling and mitigating the negative effects. These efforts can be subsumed under the premise of ageing management, an engineering, operation, and maintenance strategy and corresponding actions to control within acceptable limits the ageing degradation of SSCs [1]. Although ageing management is vital to safe operation at all stages of a research reactor's lifetime, operating organizations have formulated various methods and tools to implement programmes in line with a graded approach dependent on operational conditions, equipment specifications and regulatory guidelines. The design and best practices in the management of SSC ageing provide insight into establishing an effective programme to improve research reactor reliability and safety.

Research reactor operating organizations now possess immense information and experience on the design and implementation of ageing management programmes. For the large number of relatively aged reactors, ageing management has risen from routine inspection, corrective maintenance and system overhaul, resulting in thorough analyses of SSC degradation under natural wear and reactor level conditions of temperature, pressure and radiation. Through the provision of technical guidance as well as networking and cooperation among organizations within the international research reactor community, strategies for managing similar systems and components and tools for monitoring degradation and auditing programmes can be made available. Such information has spurred a recent trend of incorporating ageing management programmes into the design and construction of new research reactors, as reported for newly operating and planned reactors in Argentina, China, India, Morocco, Ukraine and Vietnam.

Research reactors of various ages, power ratings and utilization schemes were represented at the IAEA Technical Meeting on Research Reactor Ageing, Modernization and Refurbishment. Rather than repeat the conceptual basis and management recommendations from IAEA Safety Standards Series SSG-10 Ageing Management of Research Reactors [2], some common themes among the presentations and papers are presented.

1. LIFETIME MANAGEMENT OF SSCs

The SSCs most commonly described as particularly vulnerable in terms of ageing management are cooling towers, primary system pumps and welds, ventilation systems, control rods, and instrumentation and control (I&C) electronics. As the SSCs critical to research reactor operation vary, however, a number of approaches have been formulated to address their wear and maintenance. Methods such as routine walkthroughs, in-service inspection (ISI), wire continuity checks and pump vibration monitoring are utilized to track corrosion and degradation of SSCs, resulting in numerous instances of detection of leakage and component breakdown. Periodic inspection has been instrument to ageing management following long overhaul periods, such as those caused by vessel wall leakage at the NRU reactor in Canada and radial beam port (RBP) leakage at the TRIGA Mark II reactor in Bangladesh. ISI capabilities have recently been advanced through the utilization of digital cameras. Although radiation resistance varies substantially among models, the greater visual accessibility and ability to record surface inspections of vital components such as the reactor pool liner, even with an inexpensive monochromatic camera, has enhanced the analysis of degradation. Also, the IAEA will soon begin a programme of expert missions for the purpose of performing and training local staff in non-destructive examination and ISI of primary components.

The modernization of I&C, the replacement of analog equipment with digital, has improved the reliability of automatic protective functions and data management, making the latter synonymous with modern operation. In one example, the higher resistance to noise and thus lower frequency of spurious signals with new digital equipment at the TRIGA II Vienna research reactor in Austria was necessitated by a failure analysis that demonstrated half of their registered incidents over 50 years of operation stemmed from I&C problems, including many erroneous reactor protection system actions. Replacement often includes large scale re-design of spaces and operation procedures to accommodate new control rooms, including the addition of remote secondary or emergency control panels. Consequently, operation procedures must be amended, occasionally to a great extent. In addition to the accounts detailed in the submitted papers, more specific information on the experience, and benefits gained, of various research reactor organizations in the conversion to digital I&C will be contained in the forthcoming IAEA TECDOC Digital Instrumentation and Control Systems for New and Existing Facilities [3].

2. AGEING MANAGEMENT PROGRAMME DESIGN

Various designs of ageing management employ a systematic approach, beginning with an overall priority, from enhancing safety performance to minimizing unscheduled shutdowns to, in the case of the HFIR reactor, USA, an optimized operational predictability. The strategy under a comprehensive programme can then attempt to be proactive, predictive, preventive and corrective. The identification of SSCs to be covered under an ageing management programme can be based on safety values, manufacturer specifications or failure analyses to determine the SSCs most at risk, which may include experiences from other research reactors. Systems are typically subdivided according to common degradation mechanisms. In the case of the NRU research reactor in Canada, the three classifications prior to enlargement following the calandria leak in 2009 were concrete structures, the reactor and channels, and instrumentation; while for the HFIR reactor, I&C, mechanical and electrical systems are grouped.

The actions to be taken under an ageing management programme typically include inspections and safety reviews with a well defined periodicity, but could also encompass ISIs and peer reviews as well as staff retention and continuous training programmes. Seismic surveys are also increasingly important in the wake of the Fukushima Daiichi Nuclear Power Plant (NPP) Accident and subsequent stress tests. For the SAFARI-1 research reactor of South Africa, actions have been grouped in categories of safety, sustainability, management, resource based and life extension, as the expected lifetime of the reactor has now been prolonged beyond the original 50 year limit to 2030. Additionally, dialogue and review by regulatory authorities is highly recommended, as is translating experiences and research outcomes into the design of programmes for new research reactors, such that ageing management can be implemented at all stages of a reactor’s lifetime.

Finally, one common problem observed by many research reactor staff with regard to ageing components is the difficulty of obtaining and stockpiling new and spare parts. Obsolescence and the transitory nature of many businesses, in addition to limited facility monetary resources, jointly produce a climate of significant unreliability in refurbishment and corrective maintenance. Following a noticeably increasing trend in the difficulty of obtaining spare parts, the proactive segment of the ageing management programme for the NIRR-1 research reactor in Nigeria was modified to include the timely procurement and storage of critical spare parts in a dedicated warehouse.

3. ANALYSIS OF DEGRADATION AND COMPONENT FAILURE

With a series of SSCs and degradation mechanisms, a matrix may be generated for a simplified overview of ageing management. This matrix is populated with events recorded by staff, typically in a sort of logbook that may be converted into an electronic database, or failure data library, for easier data manipulation. This database may also receive automatic tracking data of operating conditions from event monitors featured in many new control consoles. Many operating organizations have developed various behaviour models that take into account SSC materials and corrosion or degradation mechanisms for the purpose of predicting failure. Many operating organization have encountered positive results as well. With cautious and thorough monitoring, reporting and long range planning, Oak Ridge National Laboratory, USA, boasts 99% confidence in their predictability for critical SSCs.

4. ADMINISTRATIVE ISSUES

Many research reactor representatives claimed to be implementing ageing management programmes as part of an integrated management system according to IAEA publications [2, 4]. This may assist in informing each branch of the operating organization of ageing management concerns, activities and results and facilitate the performance of reviews and decision making. In terms of allocation of resources, experiences in Argentina have yielded a central ageing management administration for the low power research reactors, while the larger reactors, having more resources at hand, have each been able to devote dedicated staff and budget. Nevertheless, the high power research reactors continue to share information on SSC performance and degradation issues. Finally, the appropriate upgrading of Safety Analysis Reports (SARs) to reflect modernization, refurbishment and alterations in ageing management programmes is incumbent upon all operating organizations.

5. OTHER ISSUES

Additional issues that may significantly affect ageing management relate to reactor fuel conversion and waste storage and management. Many research reactors are undergoing conversion from highly enriched uranium (HEU) to low enriched uranium (LEU), particularly as deadlines for US and Russian Federation take back programmes approach. Aside from the impact of new core designs, many operating organizations utilize the long overhaul period to install systems appropriate to the new operating conditions as well as new experimental capabilities or modernize existing SSCs. As in the example of the Maria reactor in Poland, the primary cooling system and associated electrical equipment will have effectively been replaced over the course of eight years, notably including the installation of four main circulation pumps and three auxiliary pumps to overcome the additional hydraulic losses from the new core and adequately remove decay heat after shutdown, respectively. Additionally, broad upgrades to its equipment for production of radioisotopes have been completed such that the reactor's primary utilization will be ^{99}Mo generation. Ageing management for the research reactor consequently must be updated substantially to reflect these vast changes.

As many research reactors approach the end of their service, radioactive waste management is becoming more prominent in the practice of ageing management, particularly in Member States with small nuclear programmes and little incentive to establish waste processing centres and permanent repositories. In the absence of processing and disposal sites, many institutes consequently have been required to design their own liquid and solid radioactive waste storage systems. Notably, the Institute of Energy and Nuclear Research (IPEN) in Brazil devised a pneumatic transport system, La Reina Nuclear Centre in Chile constructed an overflow tank and system for the reactor pool and a special storage rack was developed for the replacement of the beryllium reflector elements of the SAFARI-1 reactor in South Africa. With additional necessary periodic inspection and maintenance, these systems will support continued operation during the latter stages of each reactor's lifetime and serve as a temporary solution until final processing and disposal.

Radioactive waste and spent fuel also constitute an important aspect in the contexts of fuel conversion and decommissioning. In the absence of national depositories, which were widely originally intended, on-site spent fuel storage pools and associated equipment for integrity checks today commonly serve research reactor operating organizations, not least in the preparation for fuel return during reactor conversion. As noted in the experience of the Pamir-630D reactor in Belarus, the receiving Member State assumes responsibility for the disposal of spent fuel and attendant waste generated during the process. This practice is a facet of both American and Russian programmes. Meanwhile, as demonstrated in the decommissioning plan for the IR-100 reactor in Ukraine, the spent fuel pool, though small in size, merits considerable attention regardless of the decommissioning strategy chosen. The long process of decommissioning, or in the case of Venezuela, the conversion of the reactor into a gamma irradiation facility, can be rendered more efficiently with adequate radiation monitoring, water chemistry control and fuel inspection while the reactor is still operating.

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LIFE MANAGEMENT PROGRAMME FOR LONG TERM OPERATION OF REACTORS AND NUCLEAR FACILITIES

Ageing management of research reactors in Argentina

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1. RA-0 RESEARCH REACTOR

1.1. General description

The reactor RA-0 is a critical facility for the performance of exercises for research, education and training. It is located in the city of Cordoba, in the building of the Faculty of Physical Sciences. Its rated power is 1 W, which minimizes the shielding requirements for civil work and dispenses with a core cooling system.

The core consists of two concentric and removable tanks of anodized aluminium with an active volume of 70 l. Moderator is demineralized light water, which enters through the bottom of the external tank with an overflow at the top of the inner tank that is removable.

The fuel elements of 20.00% enriched UO_2 are housed vertically in a grid with 232 holes of diameter greater than the fuel cladding. Other holes of smaller diameter allow circulation of the moderator, and four holes accommodate detector tubes or tubes for mounting experience.

The control system consists of four control rods built with a cadmium sheet wrapped in a stainless steel cladding. These are inserted vertically and tangentially to the outer tank. Each bar is magnetically coupled to an electromagnet secured to the rise and fall mechanism, which also allows them to function as control rods. In case of emergency the electromagnet stops, and the bar falls under the influence of gravity.

Obsolescence was detected in some elements of instrumentation, notably in former relays, starting systems and scram functions, but checks for updates in the data acquisition system have also been undertaken proactively to keep updated to the reactor. No ageing in mechanical systems has been verified.

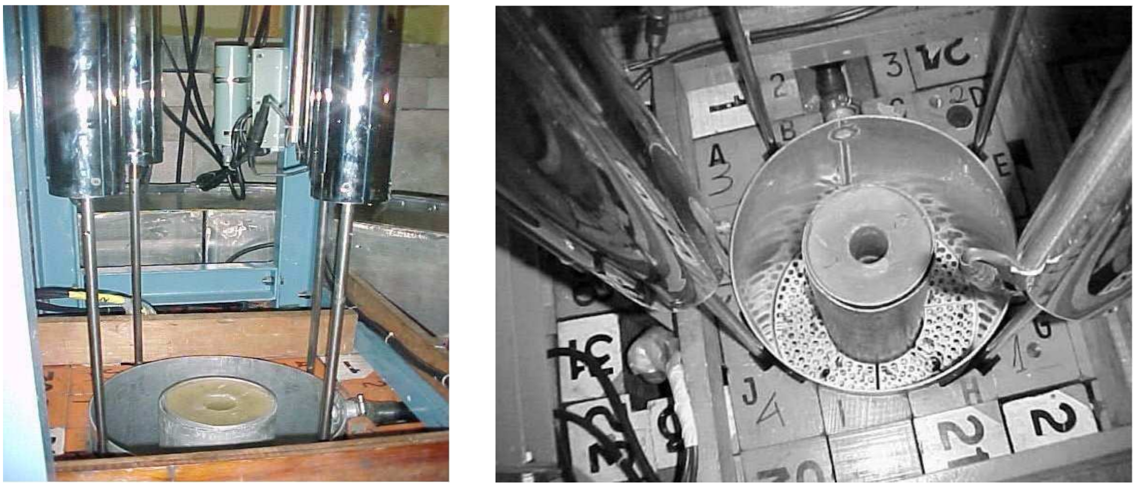

FIG. 1. Reactor tank views.

1.2. Ageing status

1.2.1. Instrumentation and control

Signaling system ionization chambers containing BF_3 and the scram system are handled through relay logic. Due to technological obsolescence it is difficult to obtain spare parts for replacement before failure. On the other hand, they are located in a precarious cabinet in the reactor room without any protection against ingress of dirt and dust, and most items are out of the plastic to prevent deposition of dirt on the contacts, so that its operation is at a great risk of malfunction. The lack of certain features of the current scram logic has also been realized. Area monitors and the implementation of a 2 of 3 logic for startup and operating channels, among others, will be added in a new technology implementation schedule. The wiring presents good quality visual inspection. Although at the time of inspection the data acquisition system was not running due to a specific problem, we were informed of its functionality, the collection of all analog and digital signals, level, rate, flow and bar positioning signals. This data is accessible by internet, CD storage and two servers in tandem, and will provide the operator with access to management system information stored in a friendly and dynamic way.

FIG. 2. Control console view.

1.2.2. Mechanical systems

The anodized aluminium tank containing the core was changed recently, and it is in good condition, as no corrosion was found in general and especially in the welded area. The radial biological shield presents no water leaks or signs of significant degradation. The whole moderator system is in a good general condition. While the original stainless steel pipes were replaced by plastic, they do not show signs of degradation or loss. The visible section of the fast drain pipe for the moderator, which is the only stainless steel component, has no apparent signs of deterioration. The storage tank is not leaking or corroded. The pump works fine with normal vibration levels. Valves operate properly without losses; in particular, the safety valve is in good general condition. Resin filters are functional and in good condition. The mechanisms of control rods are in good condition both functionally and in relation to ageing.

1.3. Conclusions

The RA-0 reactor has subsystems and components that are in a state of technological obsolescence, both starting and operating systems, and in some cases in a state of precariousness, such as the drive relays in the reactor room. It has current technology systems such as the data acquisition system, and a plan to change equipment has been created to lessen the risk of ageing. A stock of spare parts for electronic equipment maintenance or I&C does not exist, and obtaining replacements for certain equipment is not easy. The pressure vessels are apparently in good condition, as well as pipes, valves, motors and control systems. It will be necessary to evaluate the possibility of a technology upgrade that would initially replace operating channels, relays and control rod drive mechanisms with electronic systems [1].

2. RA-1 RESEARCH REACTOR

2.1. General description

The reactor is an open tank type with an authorized thermal power of 40 kW and a core of uranium enriched to 20% ^{235}U reflected by graphite. Its moderator and coolant is demineralised light water. The fuel elements are cylindrical bars of UO_2 with a graphite active length of 540 mm and a graphite reflector at each end, making a total length of 660 mm, encapsulated in aluminium 1 mm thick for a 10 mm outer diameter. The core consists of 228 fuel elements distributed in five concentric circles, forming an annular geometry of 153 mm in internal diameter and 330 mm outer diameter. Control is performed by four cadmium rods encapsulated in stainless steel located between the core and the external reflector at 90 degrees to each other. They are attached by magnets to the drive mechanisms located in a metal structure on the tank. The mechanisms are driven by electric motors of stepper type commanded from the control room and a series of sensors that constantly give the exact position of each bar. The primary coolant circuit includes demineralised light water circulation forced upward by one or two centrifugal pumps, according to the power requirement, built entirely of stainless steel. Two towers of demineralization with anionic and cationic resins keep the water within the range of values of purity required. The secondary cooling circuit, by means of a modular plate heat exchanger, transfers heat from the primary to secondary circuit.

2.2. Ageing status

2.2.1. Instrumentation and control

Control console components are suffering from obsolescence due to missing spare parts. Control console components are being gradually replaced by our own designs based on a complex programmable logic device.

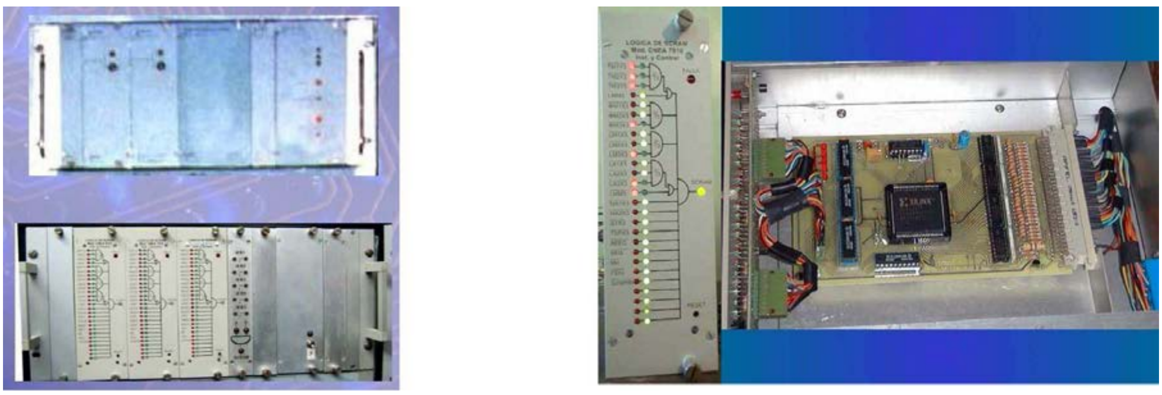

FIG. 3. Old and new scram logic inside view.

2.2.2. Mechanical systems

The primary cooling system has two centrifugal pumps. During normal preventive maintenance, abnormal vibrations were detected. In 2011 the two primary cooling pumps were replaced. The new pumps will be equipped with online vibration monitoring [2].

FIG. 4. Control console evolution.

3. RA-4 RESEARCH REACTOR

3.1. General description

The reactor RA-4 is a thermal homogeneous reactor with a solid core built for purposes of instruction, training and research. Maximum power is 1 Watt. Given its very low reactivity and negative temperature coefficient of reactivity, it has an intrinsically safe power to avoid any increases leading to dangerous levels. The core consists of overlapping 24 cm circular plates consisting of a homogeneous mixture of U_3O_8 pressed as fuel and polyethylene as moderator. Uranium is enriched to 20%, and the critical mass is 674.8 grams. The core is surrounded by a graphite reflector 20 cm thick. A 10 cm layer of lead and a borated water layer 60 cm thick in the external tank provide gamma radiation shielding. Reactivity control is accomplished by the introduction of cadmium vertical plates sheathed in aluminium in the graphite reflector. The vertical drive mechanisms are under the container. They have mechanisms moved by an AC motor that separate the plates of the reactor core to each other, which reduces the critical mass and performs the scram function. Obsolescence has been detected in instrumentation systems in the signalling and channels in place. Borated water leakage from the reactor tank and auxiliary tank due to degradation of the internal protection has been detected.

3.2. Ageing status

3.2.1. Instrumentation and control

Signalling and power systems of the ionization chamber and BF_3 chambers are handled through relay logic that has not presented any inconvenience to date, but given technological obsolescence, it is difficult to obtain spare parts for replacement before failure. In a similar situation are the channels that have launched hybrid valve gaseous technology for replacement of field effect transistors that were current at the time. The internal wiring for its functional logic are in constant movement and bending when the reactor is in operation, and since they are exposed to radiation, replacement would be convenient, although they are not in a bad state according to visual inspection.

FIG. 5. Rod motion mechanism.

There are currently some low voltage relays in need of eventual replacement parts, but since there are 42 parts in the system, our reserve may only last a couple of events of failure. On hand are no parts for high voltage relays, valves or components related to gaseous systems.

FIG. 6. Control console view, front and back.

3.2.2. Mechanical systems

The reactor staff reported that the reactor tank was leaking borated water at the bottom due to possible corrosion. The tank was repaired by welding in the affected area on the plate thickness. The protective coating may be degraded in some areas, leaving the metal exposed to borated water, especially at the bottom due to deposition of potentially corrosive substances. If this is so, the plate could present internal pitting in various magnitudes, compromising their integrity and tightness, as indeed already there has been a leak.

FIG. 7. Leak affected zone.

3.3. Conclusions

The RA-4 reactor is in an advanced state of technological obsolescence in its relay logic systems and motion systems with hybrid gas valves. The reactor vessel has apparent interior corrosion, which causes leakage of borated water. Evaluation of a technology upgrade with replacement of up channels is contingent on obtaining gas valves [3].

4. RA-6 RESEARCH REACTOR

4.1. General description

The core of the reactor RA-6 comprises a plate type fuel arrangement and is contained in a stainless steel tank 2.40 m in diameter and 10.40 m deep resting on a grid to 6.60 m below the water level. The RA-6 reactor tank is open, using rectangular plate type fuel elements enriched to 19.70% uranium by weight, and each section has 19 plates. The fuel element plates consist of an aluminium–uranium alloy called "meat", which is covered by aluminium cladding 6061. The aluminium matrix and the cladding of the fuel are the first barrier against the release of fission products. The core of the reactor RA-6 supports variable configurations of fuel elements. It also includes five pairs of neutron absorbing plates located in fixed positions that are called control rods, and it uses light water as a reflector in the axial direction and graphite as a reflector in the radial direction. The core is moderated by light water that also serves to shield the top. The five control rods are moved vertically by bars mounted on the bridge of the mechanisms located above the mouth of the reactor pool. Therefore, the five pairs of control plates are inserted into the core from above. Four of the control rods are called safety bars and are involved in the regulation of reactivity and fall by gravity into the core when the protection system generates a signal of scram. The fifth slider bar is called regulation and participates in the regulation of small variations of reactivity but does not participate in a scram. The reactor power has 3 MW_{th} in power. This power is transmitted to the primary circuit coolant circulating through the core in a downward direction. Through a heat exchanger, the primary circuit transmits this power to the secondary circuit, which eventually dissipates power into the atmosphere through three cooling towers. Water from the primary cooling circuit is of high purity and has the necessary systems to ensure that quality. Control is done by absorbing the 5 bars mentioned above: 4 are part of the extinguishing system, and the fifth is intended for fine control of power. Core cooling in normal operation is forced downward, while emergency cooling takes place under natural convection.

4.2. Heat transport system modifications

Primary and secondary circuits modifications include:

- The primary and secondary cooling systems were modified in order to dissipate 3 MW_{th};
- In the primary cooling system, the water flow rate was increased from 150 m^3/h to 340 m^3/h;
- This implies changes in the pumps and heat exchanger and modification of the syphon breaker and flapper valve;
- A diffuser for the decay tank was designed to homogenize the inlet flow and reduce the impact on the buffers;
- New inertia flywheel was installed to drive the pump for seven seconds to accomplish residual heat removal in case of power supply failure;
- A new cooling tower system was connected to the secondary circuit [4].

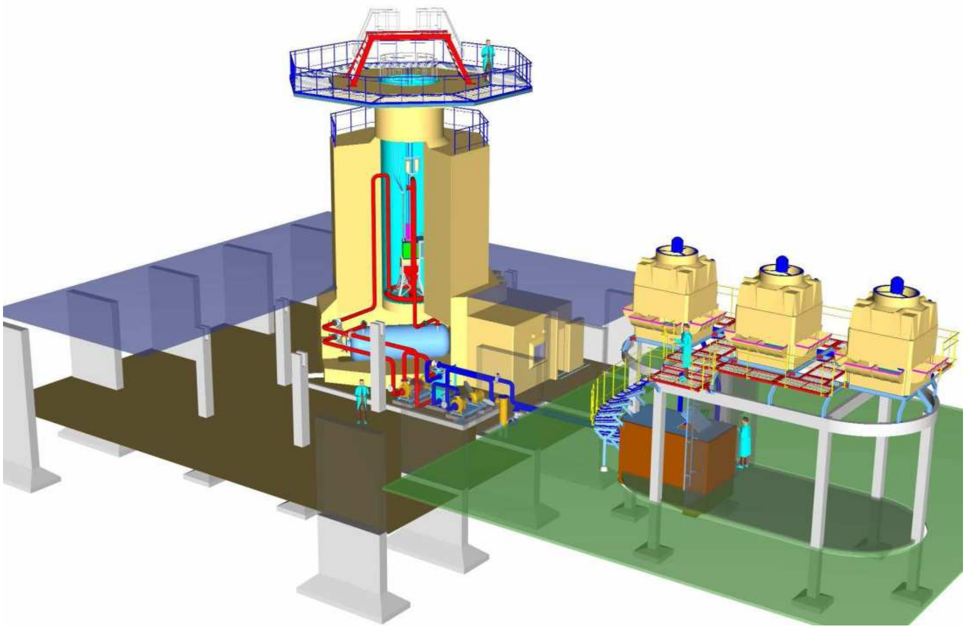

FIG. 8. New primary pump and inertia flywheel with online vibration monitoring system.

FIG. 9. New secondary pump.

FIG. 10. New plate heat exchanger.

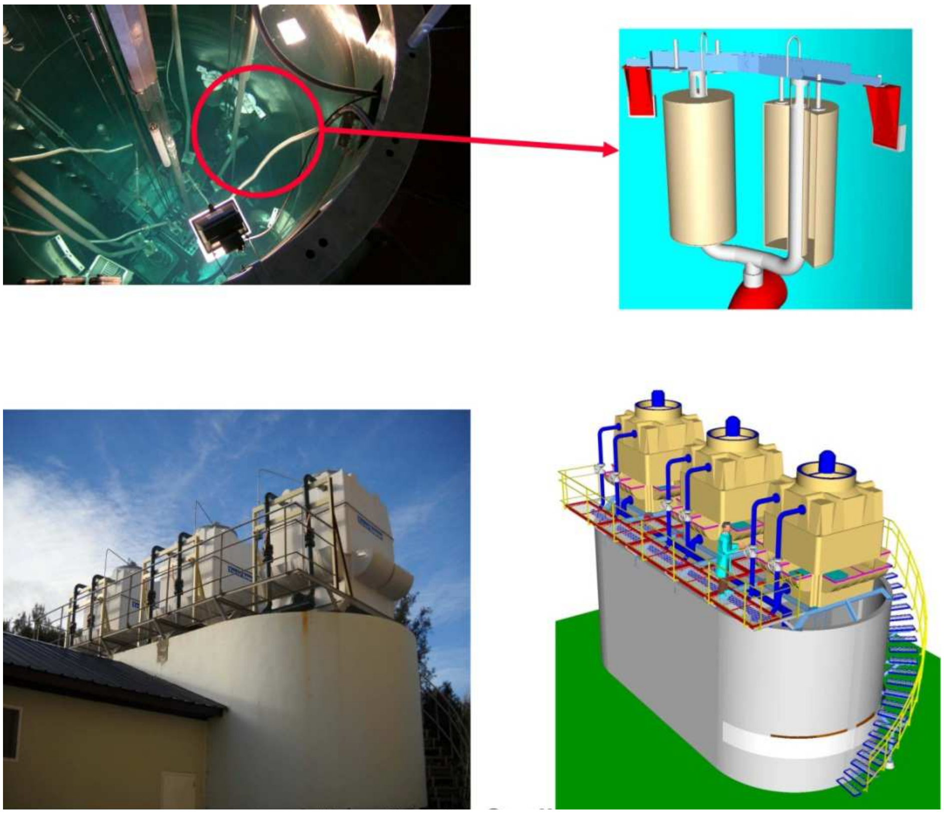

FIG. 11. New redundant syphon breaker.

5. LIFE MANAGEMENT PROGRAMME FOR LONG TERM OPERATION OF REACTORS AND NUCLEAR FACILITIES

Argentina has six research reactors. Five of them are operational, and one is temporarily shutdown pending a new location. Each reactor has its own maintenance programme and maintenance staff. They perform mostly corrective maintenance but don't have experience on developing and performing an ageing management programme. For this reason, we have created an ageing management group on research reactors to assist each reactor staff with their own ageing issues. We perform SSC inspections on each facility and make suggestions based on our experience. Our experience comes from working at NPPs since the 1970s, and we translate all this experience into research reactors. A new research reactor project is being carried out by our country. This is a great opportunity to develop an ageing management programme from the very beginning [5].

Aspects to evaluate are:

- Analog technologies without technical support and no spare parts available;
- Suppliers that no longer exist;
- Equipment without the required functionalities;
- Unresolved ageing threats to security and income;
- Higher maintenance and operation costs;
- Staff retirement and new professionals unaware of old technologies;
- Could have regulatory constraints that require upgrades;
- Increased cost and loss of safety margin concept due to ageing;
- Plan life management: a set of coordinated and systematically designed actions to achieve a safe and profitable plant for as long as possible with an acceptable tolerance.

Plant life management is concerned with the evaluation, early detection and resolution of the effects of degradation mechanisms related to ageing, encompassing the maintenance and replacement of SSCs. In our developed scheme, prioritization and selection of SSCs, plan design, and plan implementation and continuous improvement are the three components of systematic process management. This attempts to anticipate and monitor various aspects of obsolescence and modernization of SSCs.

In terms of obsolescence, examples of monitored parameters are:

- Obsolescence state of each I&C subsystem;
- Failure rate impact and availability;
- Maintenance and operation cost;
- Regulatory requirements.

Measures accounted for in modernization, particularly of digital technology, are:

- Selection of system or subsystem to upgrade;
- Best technology to implement;
- Method of implementation, i.e. incremental or during a single outage;
- External contracts or long term technology support;
- Workforce needed, both full and part time staff;
- Functional requirements;
- Time and monetary budget;
- Interface between old and new technology;
- Profit versus cost evaluation;
- Degree of automation required in testing, storage, control, etc.;
- Staff capacity;
- Licensing issues.

Finally, several challenges in the process of modernization, necessarily involving the coordination of the utility, vendor and regulator, are analysed:

- Functionality: the incorporation of new features not available with the old system;
- Obsolescence: the shorter average life of new technologies and planning of upgrades;
- Versatility: capacity for expansion and ease of upgrading;
- Human factors: operator interfaces and automation by computer systems;
- Security: specifications of redundancy, diversity and independence;
- Economics: costs, payment forms, cash flow and return on investment.

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CORROSION INDUCED LEAKAGE PROBLEM OF THE RADIAL BEAM PORT 1 OF BAEC TRIGA MARK-II RESEARCH REACTOR

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1. INTRODUCTION

The Bangladesh Atomic Energy Commission (BAEC) TRIGA Mark II research reactor is a light water cooled, graphite reflected reactor designed for steady state and square wave operation up to a power level of 3 MW_{th} and for pulsing operation with a maximum pulse power of 852 MW. The reactor achieved its first criticality on 14 September 1986.

The BAEC TRIGA Mk II research reactor has four BPs. These are: (1) tangential BP, (2) radial piercing BP, (3) RBP-1 and (4) RBP-2. When not in use, the BPs are filled up with removable inner and outer BP plugs mainly consisting of: (a) graphite plug (6 inch diameter, 40 inch length:), (b) lead plug (8 inch diameter, 5 inch length) and (c) high density polyethylene plug (8 inch diameter, 3 inch length).

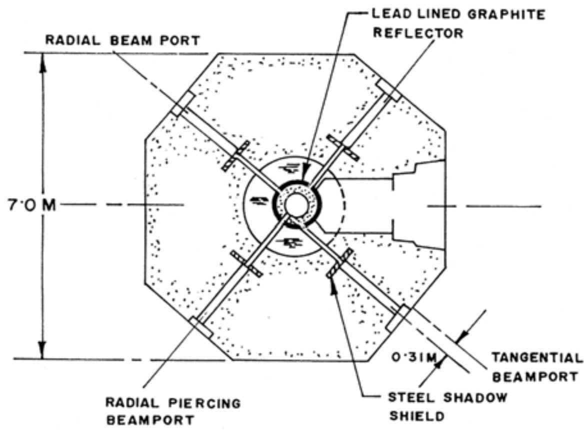

FIG.1. Location of BPs in the reactor shield structure.

It became necessary to remove all BP plugs from RBP-1 in order to install a high resolution powder diffractometer (HRPD) collimator in front of the port. This collimator is installed inside RBP-1.

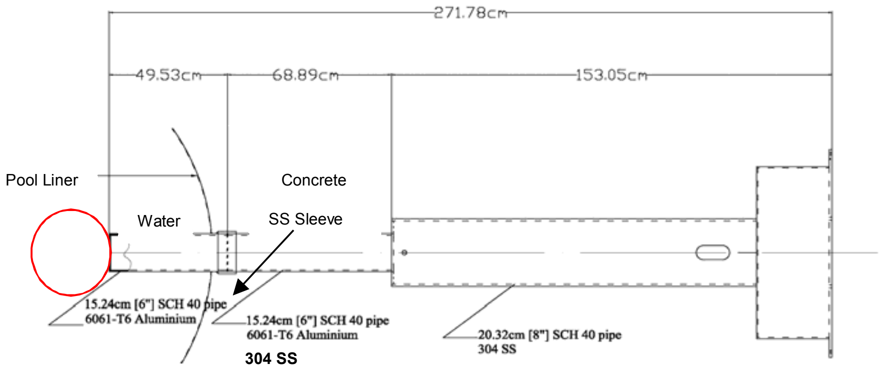

FIG. 2. Details of RBP-1.

The high density polyethylene plug is a lightweight device and can be removed easily by hand. For removing the graphite and lead plug, however, equipment called cask and carriage was used. The graphite part of the inner BP plug was broken, as shown in Figure 3 while being removed on 30 March 2009 with the help of the cask and carriage. The graphite plug was broken again while it was being removed on 10 April 2009 with a scoop like device (Figure 4).

FIG. 3. Broken graphite plug (1st break).

FIG. 4. Scoop used for BP plug removal.

The inner part of the graphite plug (33.02 cm in length, 14.94 cm in diameter) remained stuck inside the aluminium section of the BP tube (Figure 5). The part of the graphite plug, which is about 33 cm in length still remains inside the aluminium section of the BP tube in its stuck out condition.

FIG. 5. Broken graphite plug (2nd break).

TABLE 1: DIMENSION OF INNER AND OUTER SHIELD PLUG ASSEMBLY OF BP-1

Name of shield plug	Material	Dimension
Outer plug assembly	HD polyethylene	Ø 7.880×430
Inner plug assembly	Stainless steel	Ø 7.880×2.250 #2
	Lead	Ø 7.880×50
	Graphite	Ø 5.880×400

2. DOCUMENTATION FOR CONDUCTING THE WORK

The following section was prepared in order to report the removal of the broken and stuck graphite plug from RBP-1 of the TRIGA Mark II research reactor as part of the regulatory requirements.

2.1. Work plan

The objective of the work was to clear RBP-1 by removing the 13 inch broken part of the graphite plug while ensuring minimum radiological risk to the working personnel and also to the reactor and the surrounding environment, so as to facilitate the installation of the collimator of the HRPD inside RBP-1. It is to be noted that the collimator helps provide collimated neutron beams to the HRPD.

2.2. Quality assurance programme (QAP)

QAP was developed, and checkpoints were identified regarding the resource requirements needed for removing the broken and stuck graphite plug from inside RBP-1 of the BAEC 3 MW TRIGA Mark II research reactor. These ensured safety of the working personnel, the reactor and the environment through the implementation of good engineering practices and good radiation practices in all phases of the work.

2.3. Radiation protection programme

All radiation protection measures were implemented under the direct supervision of the Radiation Control Officer. Adequate numbers of trained personnel were engaged with a mindset to accomplish the radiation protection tasks in a safe and competent manner to ensure protection of the personnel and the environment against any external radiation hazard during all the phases of the work planned for removing the broken graphite plug from inside RBP-1.

2.4. Emergency response plan

An emergency response plan was developed for the proper management of a probable emergency situation that may arise during the removal operation of the broken graphite plug from RBP-1 of the TRIGA Mark II research reactor.

3. THE ACTIVITIES CARRIED OUT FOR REMOVING BROKEN GRAPHITE PLUG

Total work activities have been categorized in three groups: (1) safety implication and risk analysis, (2) removal of an adequate number of fuel elements from the reactor core, and (3) removal of the broken part of the graphite plug from the beam port RBP-1.

3.1. Safety implication and risk analysis

3.1.1. Beam tube rupture incident

Considering the matter of accidental BP rupture, it should be mentioned that the wall thickness of the aluminium BP tube is about 0.28 inches. The material of the tube is Type-6061T6 aluminium alloy, which has a tensile strength of 52 000 psig (for mild steel, the value is about 40 000 to 60 000 psig). The chance of any rupture of the aluminium BP tube while cutting the graphite using a manual auger is thus very negligible. However, the Reactor Operation and Maintenance Unit (ROMU) has designed and fabricated a high density polyethylene plug that was kept ready before undertaking the graphite plug removal operation. The plug will be inserted into the BP tube so as to prevent any loss of reactor pool water in the event of an unwanted BP tube rupture incident. An adequate amount of makeup water, reactor grade demineralized water, was also kept ready to fill the reactor pool if needed.

3.1.2. Fuel element storage rack dry up incident

In case of a loss of coolant accident (LOCA) through the ruptured beam tube, the in-pool fuel storage racks may go without cooling water. After careful analysis, however, the chance of such a situation is ruled out under the present work plan. However, if such a type of LOCA occurs, the fuel storage racks will be lowered from their present position such that they remain immersed under pool water. Nylon rope or galvanized iron wire will be used for lowering the fuel storage racks. It is to be mentioned that during the ^{16}N decay tank leakage incident of 1997, the fuel storage containing one used fuel element was lowered similarly so as to reduce the radiation dose rate at the reactor top area. The pool water makeup system and online purification system shall be kept in ready during the implementation of the activities of the work plan.

3.1.3. Design and development of a high density polyethylene plug

The purpose of the high density plug is to close the opening of RBP-1 and prevent any loss of reactor pool water in the event of a BP tube rupture or puncture incident. As described above, the chance of any rupture or puncture of the aluminium BP tube while cutting the graphite using a manual auger could simply be negligible. However, to be very conservative, the matter of fabricating the high density plug has been taken into consideration with its design approved by the Director of ROMU. Once the high density plug has been fabricated, a mockup test will be performed so as to demonstrate that the high density plug fits into the 6 inch part of the BP tube appropriately. The plug will then be removed, decontaminated and stored in a suitable location such that it can be deployed immediately in case of need.

3.2. Removal of fuel elements from the reactor core

The gamma dose rate at the open end of RBP-1 was close to 1000 μSv/h, which is about 100 times the permissible dose rate for a radiation worker. It is to be noted that the main source of this gamma radiation is the fission products contained in the irradiated fuel elements present in the reactor core. To reduce the gamma radiation dose at the work place located in front of RBP-1, an adequate number of fuel elements were removed from the part of the reactor core that falls within the line of sight when viewed through RBP-1. This helped to reduce the radiation dose rate, which was very high at that moment. The removed fuel elements were kept in the fuel storage racks located inside the reactor tank. At the end of the graphite plug removal operations, the fuel elements were put back to their respective locations in the reactor core. The fuel element history file was updated accordingly.

3.3. Removal of the broken graphite plug

The removal operation for the 33.02 cm (13 in) long broken part of the graphite plug was undertaken after getting formal approval of the regulatory authority. Before that ROMU was required to submit to the regulatory authority the detailed work plan, emergency response preparedness plan and QAP to be implemented in connection with removal work. While performing the removal operation, extreme care was taken such that the corroded aluminium BP pipe would not be damaged, and also the amount of graphite dust produced could be kept at a minimum. The stainless steel auger, designed and fabricated mainly by ROMU personnel, was found to be very effective in this regard. Details of the auger and associated fixtures used are shown in Figures 6a and b.

FIG. 6a. Details of hollow saw auger and associated fixtures.

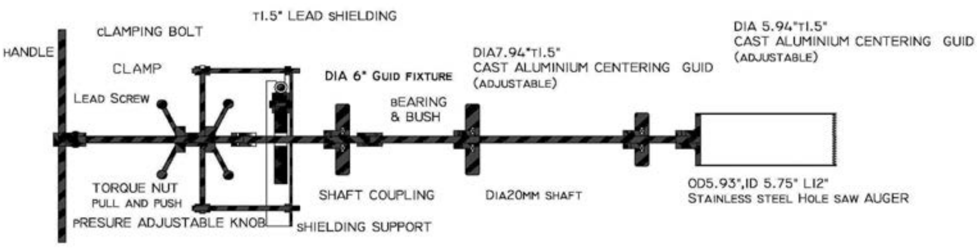

FIG. 6b. Details of hollow saw auger and associated fixtures.

It is to be mentioned that the auger was operated manually so as to be sure that it did not cut the aluminium BP tube. The cutting operation, which is shown in Figure 7, took 12 days. A picture of the last segment of the graphite plug is given in Figure 8. In this picture clear corrosion marks are seen on the surface of the graphite plug that was not trimmed by the auger. It was confirmed that condensate accumulation in the annular space between the graphite plug and inner wall of the aluminium BP initiated the corrosion.

FIG. 7. Cutting operation by using auger.

FIG. 8. The last 33 cm segment of the broken graphite plug.

4. STUDY OF THE CORROSION LEAKAGE PROBLEM

After a couple of days, the stuck graphite plug was removed, and reactor pool water was found to have been leaking through the BP as shown in Figure 9. Leakage of water was stopped by putting a rubber strap around the leaking part of RBP-1. Efforts were then made to carry out visual inspection of the BP using a digital camera and at the same time develop a device that could be used to solve the water leakage problem.

FIG. 9. Water leaking through BP.

4.1. Location of the damage

As shown in Figure 2, the aluminium part of RBP-1, which has a length of 48.26 cm (19 in), pierces the reactor pool liner and terminates near the outer surface of the aluminium liner of the reactor core. The end of the BP tube is closed with a 6.125 cm (0.25 in) thick aluminium disk. The graphite plug occupied about 34.29 cm (13.5 in) of the aluminium pipe. Therefore about 15.24 cm (6 in) of the aluminium pipe of the BP from its dead end remained void. The reported leaks that developed from corrosion damage mainly took place on the inner bottom surface of the aluminium pipe located at distances from 15.24 cm (6 in) to 35.56 cm (14 in) from the dead end of RBP-1. Corrosion damage also occurred to some extent on the stainless steel pipe and at the aluminium–steel interface.

4.2. Visual inspection

The graphite plug removed from RBP-1 was wrapped with a polyethylene sheet and stored in a wooden box. Upon visual inspection, water condensate was visible inside the polyethylene sheet. This clearly indicated that the corroded part of the graphite plug absorbed moisture from the condensate that had accumulated in the annular space between the plug and the BP tube. Deposits or scales were found on the outer surface of the lead plug, and these deposits consist of species resulting from the corrosion. Photographs obtained from a digital camera inserted into RBP-1 clearly showed the corrosion in the form of metal removal and pits on the inner bottom surface of the aluminium pipe. It also revealed that damage in the form of pits had been initiated at the interface between the stainless steel and aluminium. Brown stain marks were observed on the stainless steel surface, but no pits or metal removal was found on the stainless steel surface (Figure 10).

FIG. 10. Inner surface BP aluminium–steel interface.

4.3. Scanning electron microscopy and energy dispersive X ray (EDX) analysis

Samples of corrosion products, debris, deposits, etc., were collected from the inner surface of the aluminium part of RBP-1. The specific radioactivity of these samples was found to be quite high, and as such, these could not be examined. However, scanning electron microscopy and EDX analysis of the deposits collected from the surface of the lead plug were performed in the Department of Materials and Metallurgical Engineering (MME), Bangladesh University of Engineering and Technology (BUET), Dhaka. The results showed the presence of oxygen, carbon, lead, silicon and aluminium. The friable and porous nature of some of the debris indicated the presence of hydroxides. It is to be mentioned that BUET was involved to study the problem and make recommendations to BAEC for remedial measures to be undertaken.

The BUET report pointed out that the aluminium pipe of RBP-1 suffered extensive damage as a result of corrosion due to the presence of air and moisture inside the beam tube. It also attributed the damage in terms of corrosion to the aluminium-stainless steel interface, where a stainless steel sleeve had been used to cover the circumferential gap between the SS and Al pipes. It is to be noted that the interface sleeve was not wrapped with sealant during installation. As a result water vapour from the surrounding concrete condensed and trickled into the gap between the graphite plug and aluminium pipe. Later this moisture initiated the corrosion process on the aluminium pipe. The continual presence of moisture transformed the protective aluminium oxide layer into aluminium hydroxide, which is porous, cannot prevent air and moisture seepage and attacks the fresh aluminium underneath. The cracks and pits that resulted from corrosion went through the aluminium pipe inter-granularly and allowed water from the reactor tank to drip through and out of the BP. The rubber sleeve used underneath the aluminium pipe temporarily sealed the mass flow of water from the reactor tank and stopped water seepage through the BP.

5. ANALYSIS OF RESULTS

Aluminium has good corrosion resistance in oxidizing environments due to rapidly formed adherent aluminium oxide on exposed surfaces. In a humid atmosphere, this aluminium oxide transforms into a porous, friable aluminium hydroxide. Air and moisture penetrates through this porous layer and causes further corrosion of fresh aluminium underneath in a continual process. Aluminium may also exhibit pitting, a localized corrosion phenomenon, where there is a breakdown of the passive oxide film. Chlorides and sulphate accelerate localized corrosion.

6. RECTIFICATION MEASURES IMPLEMENTED

The above mentioned BUET report recommended several actions, including installation of an outside aluminium sleeve to protect the outer surface of the aluminium pipe and seal the flow of water from the reactor tank into the BP. In line with the recommendation, a split type encirclement clamp (STEC) with a silicone rubber lining as shown in Figure 12 was designed and fabricated locally using Type 6061 aluminium alloy. The STEC was then installed around the segment of RBP-1 is located inside the reactor pool at a depth of about 8 m by using remotely operated nut tightening tools that were also designed and fabricated locally. The STEC was designed and fabricated with a provision such that it can be dismantled for replacement of the silicone lining and reinstalled again. About 48 hours after installation of the STEC, the inside of RBP-1 was inspected with a camera, and no trace of water was found.

FIG. 12. Split type encirclement clamp.

It is to be noted that approval of the regulatory authority was granted before implementing the above mentioned rectification measures, and also that several fuel elements had been removed from the part of the reactor core that was in the line of sight with RBP-1 such that radiation streaming from the core could be minimized. The job was implemented under constant supervision of the Radiation Control Officer responsible for the research reactor and the Reactor Supervisor/Manager.

7. CONCLUSIONS

The BAEC reactor has so far been operated as per the technical specifications and procedures laid down in the SAR of the research reactor. The BP leakage problem of the BAEC research reactor was an issue that could lead to a situation close to a LOCA. Therefore, the matter was handled carefully, taking all measures so that such an incident could be prevented. Assistance of agencies outside BAEC was taken for solving the problem. It is understood that the silicone rubber lining of the encirclement clamp may become damaged by neutron irradiation.

Therefore, while designing the clamp, provisions were kept such that it can be dismantled and reinstalled again following lining replacement. As a moderately aged facility, the ageing management BAEC TRIGA research reactor deserves significant attention. BAEC, together with its strategic partners, are doing what is needed in this regard.

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THE EXPERIENCE OF STORAGE AND SHIPMENT FOR REPROCESSING OF HEU NUCLEAR FUEL IRRADIATED IN THE IRT-M RESEARCH REACTOR AND PAMIR-630 MOBILE REACTOR

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1. INTRODUCTION

At the end of 2010 under the Global Threat Reduction Initiative (GTRI), the Joint Institute for Power and Nuclear Research–“Sosny” (JIPNR–Sosny) of the National Academy of Sciences of the Republic of Belarus repatriated HEU spent nuclear fuel to the Russian Federation. The spent nuclear fuel was from the decommissioned Pamir-630D mobile reactor and IRT-M research reactor.

The paper discusses the Pamir-630D spent nuclear fuel; experience and problems of spent nuclear fuel storage; and various aspects of the shipment including legal framework, preparation activities and shipment logistics.

The conceptual project of a new research reactor for Belarus is also presented.

2. SPENT NUCLEAR FUEL OF NPP PAMIR-630 D

In the 1970s and1980s, JIPNR–Sosny, earlier the Institute of Nuclear Power, National Academy of Sciences of Belarus, developed the mobile nuclear power plant Pamir-630D. Appointment of NPP Pamir was maintenance with the electric power of the mobile and stationary objects located in remote regions.

The mobile NPP Pamir-630D could produce 630 kW of electrical output with a thermal power of 5000 kW during a reactor campaign of 10000 hours [1]. The plant consisted of five basic modules: the reactor; turbine generator set; two modules of control and protection systems; and auxiliary module. All the modules were installed on semi-trailers that could be transported by trucks.

A picture of the reactor module on a semi-trailer is provided below (See Figure 1).

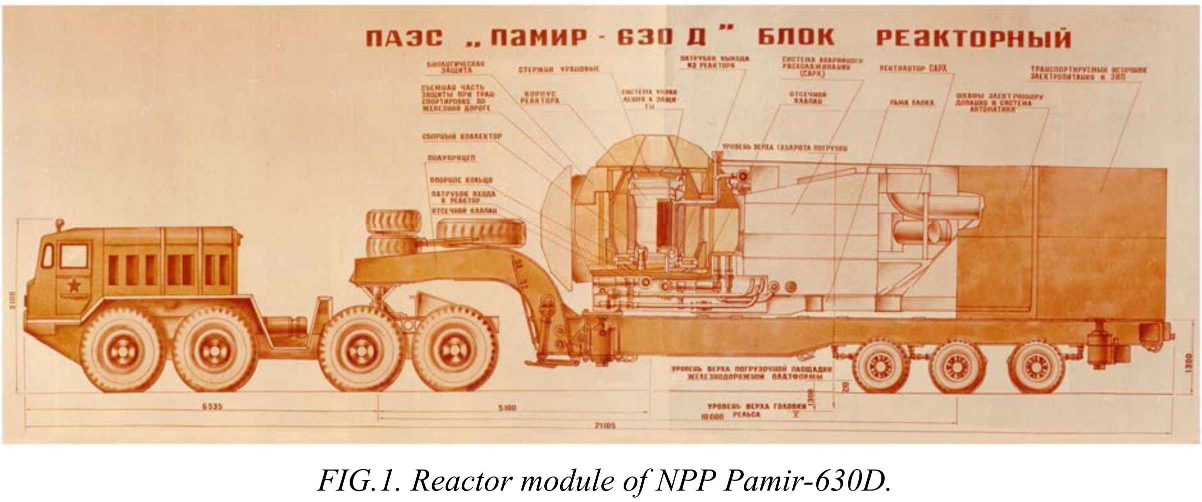

FIG.1. Reactor module of NPP Pamir-630D.

Two pilot reactors of NPP Pamir-630D were produced for testing. The first reactor was put into operation in 1985, and testing was halted in 1986. During the test the reactor worked at various levels of power. The common energy was estimated to be 6.9×10^6 kWh. Average burnup of fuel was estimated at 0.78% of uranium atoms.

The reactor used zirconium hydride $ZrH_{1.9}$ as a moderator and reflector. The coolant of the reactor active zone was dinitrogen tetroxide, N_2O_4.

The active core consisted of three types of fuel assemblies fixed within a hexagonal lattice 45 mm. 12 control rods were used for reactor regulation. Europium dioxide, Eu_2O_3, was the neutron absorber material.

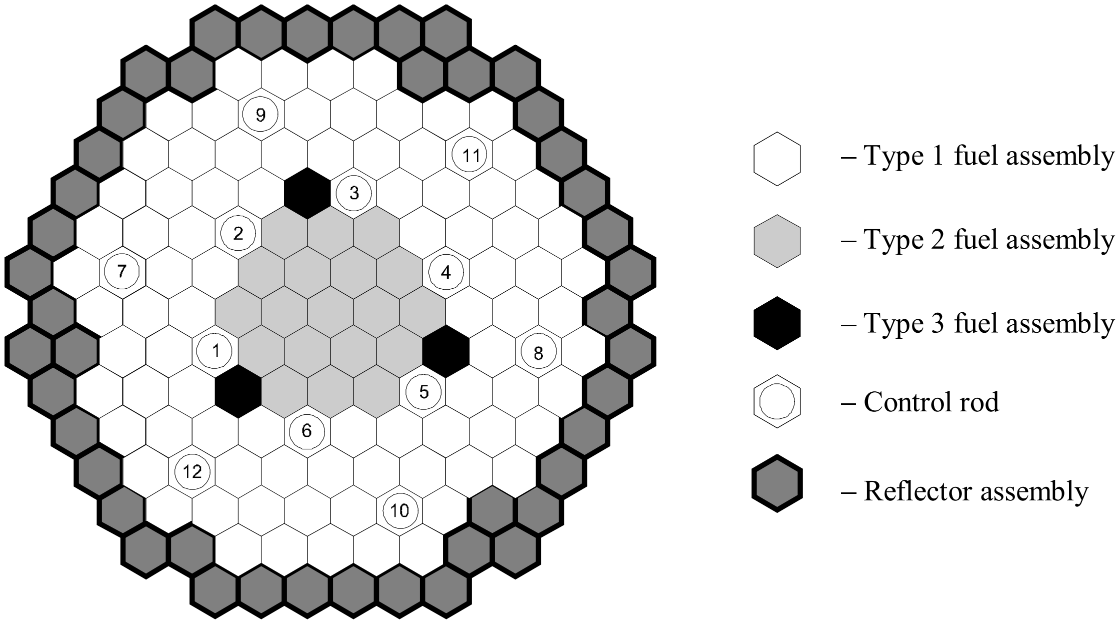

FIG. 2. Cartogram of Pamir-630D reactor active core.

The reactor core consisted of 106 fuel assemblies, 12 control rods and 45 reflector assemblies (See Figure 2). Three types of fuel assemblies with different burnup absorbers were used. Each spend fuel assembly had seven rod-shaped fuel elements made of UO_2 particles enriched to 45% ^{235}U in a nickel and chromium matrix (UO_2–Ni–Cr) and cladding. The diameter of the fuel rods was 6.2 mm and length 650 mm.

The fuel was discharged in 1991 from the active core of the Pamir-630D reactor. All 106 spent fuel assemblies were loaded into the spent fuel pool of a storage facility. One of the Pamir assemblies failed in the test period and was placed in a sealed IK canister.

Besides fuel from mobile NPP Pamir-630D, spent nuclear fuel irradiated in the experimental channels of the IRT-M research reactor has been disposed of also in the storage facility pool in five sealed IK canisters (See Figure 3).

FIG. 3. Loaded IK canisters viewed from an underwater camera.

3. EXPERIENCE AND PROBLEMS OF SPENT FUEL STORAGE

The spent nuclear fuel facility in JIPNR–Sosny is a wet storehouse that includes two pools with the following dimensions: length 4500 mm, width 800 mm, height 4200 mm. The thickness of biological protection from water is 3100 mm (the full height of the water pole is 3900 mm), above 1800 mm of concrete. The admissible water temperature is 3–40°C.

Design capacity of the pool is 207 spent fuel assemblies of type Pamir-630D and 10 hermetic canisters of type IK. Distilled water floods the pool with a pH value equal 6.0 at 20°C).

The periodicity of water control in the pool is once per week. Pool water is checked for specific activity of isotopes, specifically, the fission products Cs-134, Cs-137 and Co-60. Mechanical and ion exchange filters support demanded quality of water.

During the storage period the replacement of the following ageing equipment and installation of new systems have been made:

- The circuital pump;
- The drive of special ventilation system;
- The electronic systems of physical protection of facility;
- The protective rotary covers under pools.

The spent nuclear fuel facility was not intended originally for long storage of nuclear materials in the form of fuel rods and assemblies. Its purpose was for maintenance of nuclear materials of short-term endurance following unloading from the Pamir-630D reactor active zone. In the period of 20 years of storage, the facility developed the following problems: corrosion of metal wares, ageing of engineering and technological systems of the facility, and insufficiency of means for carrying out deactivation and recycling of fuel assemblies in case of mass leaking. Special anxiety was caused by the IK canisters and Teflon interleaf sealant between canisters because unhermetically sealed fuel assemblies and fuel rods have been disposed in the IK canisters. The interleaf between canisters was in a zone of high radiation from fission products. In cases of loss of tightness, radioactivity from unhermetically sealed assemblies or fuel rods will reach the pool with negative consequences.

All complicated factors of spent fuel storage have demanded project attention during preparation activities and spent fuel shipment for reprocessing from Belarus.

4. SNF SHIPMENT

In the framework of the GTRI and Russian Research Reactor Fuel Return programme, JIPNR–Sosny with cooperation from organizations of State Corporation ROSATOM of the Russian Federation and financial supporting from the US Department of Energy started preparation activities in September 2009 for spent nuclear fuel shipment.

The legal framework of spent fuel shipment was:

- "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" from 27 May 2004;
- "Agreement between the Government of the United States of America and the Government of the Republic of Belarus Regarding Assurances Concerning the Provision of Technical Assistance that May Be Provided by the Government of the United States of America, through the United States Department of Energy and Its Contractors, to Support the Transfer of Spent Research Reactor Nuclear Fuel from the Joint Institute for Power and Nuclear Research "Sosny" of the National Academy of Sciences of Belarus to the Russian Federation" from 1 October 2010;
- "Agreement between the Government of the Republic of Belarus and the Government of the Russian Federation on Cooperation for Import to the Russian Federation of Irradiated and Fresh High-Enriched Nuclear Fuel of the Research Reactor and Delivery to the Republic of Belarus of Fresh Low-Enriched Nuclear Fuel" from 8 October 2010.

Initial requirements for spent fuel shipment from Belarus to the Russian Federation were:

- Shipment for spent fuel reprocessing should be carried out in radiochemical factory FGUP NPO Mayak of the Russian Federation with requirements of temporary technological storage of radioactive waste and further waste arrangement in the Russian Federation;
- Spent fuel shipment should be carried out according to the international and national requirements of Belarus and the Russian Federation concerning nuclear and radioactive safety;
- Spent fuel shipment should be carried out in Škoda VPVR/M containers, which should be located in ICO containers accordingly;
- Spent Fuel shipment should be carried out by automobile and railway transport on Belarusian territory and railway transport on territory of the Russian Federation. Cargo liability and physical protection of spent nuclear fuel should be transferred from Belarus to the Russian Federation at a designated point on the boundary of Belarus with the Russian Federation;
- All standard requirements on common industrial safety, and also other special requirements operating in Belarus, should be executed for realization of activities on overloading and spent fuel shipment by automobile and railway transport;
- Physical protection of nuclear materials should be provided at all stages of activities during preparation and spent fuel shipment;
- Spent nuclear fuel should be checked for tightness of shells around fuel elements, except a radioactivity exit, at any stage of activities during preparation and shipment.

All these initial requirements of spent fuel shipment were successfully executed in Belarus. On 19 October 2010, the Republic of Belarus removed all its HEU spent fuel (See Figure 4). It also demonstrated how Belarus continues to support global non-proliferation programmes.

FIG. 4. Train departure with HEU spent nuclear fuel.

5. DEVELOPMENT OF THE CONCEPTUAL PROJECT OF A NEW RESEARCH REACTOR FOR BELARUS

The necessity of creation in Belarus of a new research reactor and the basic technical requirements for this reactor are being developed in the Belarus state programme on development of nuclear power for the period until 2020. In addition Belarus is interested in technical cooperation with the IAEA concerning the development of a conceptual project of a new research reactor for Belarus in 2012–2014.

The main purposes of a new research reactor for Belarus are:

- Research of fuel rods and fuel assemblies;
- Research of new construction materials;
- Research of new radiation proof materials providing minimal change of form;
- Test of models of new equipment such as devices, diagnostic means, steam and gas generators, etc.;
- Research of structure of substances by neutron activation analysis;
- Applied research in manufacture of isotope production of various purposes, use of ionizing radiation for medical purposes and for reception of modified materials;
- Fundamental scientific research.

The basic requirements for a new research reactor for Belarus are stated below.

- Thermal power of 10–20 MW;
- Pool type reactor with water moderator and coolant;
- The maximum density in neutron beams 5×10^{14} $cm^{-2}s^{-1}$;
- Uranium fuel with enrichment less than 20% uranium-235;
- The reactor should be multi-purpose, allowing for the performance of research on problems of safety in nuclear installations, nuclear fuel and the fuel cycle, construction materials, new coolants, basic topics and also applied research and work;
- Reactor and reactor facility should satisfy completely the safety requirements of modern normative documentation;
- Reactor lifetime not less than 50 years;

- The reactor should allow for the use of operating experience of pool research reactors with water coolant, and also the results executed early in the design of similar reactors;
- The research reactor should be constructed on an existing platform in the organization, to further the operating experience of research reactors, and the necessary infrastructure for reactor maintenance and decommissioning research.

6. CONCLUSION

Due to excellent examples of international cooperation between the various organizations of Belarus, the Russian Federation, the United States and IAEA, the shipment of spent HEU nuclear fuel from the Joint Institute for Power and Nuclear Research–Sosny marked the end of a successful project that removed all of the spent HEU from Belarus. Now the Republic of Belarus does not have spent nuclear fuel storage problems. A conceptual project of a new research reactor for Belarus has been developed.

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MODERNIZATION OF SAFETY AND CONTROL INSTRUMENTATION OF THE IEA-R1 RESEARCH REACTOR

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Abstract

The research reactor IEA-R1 located in the Institute of Energy and Nuclear Research (IPEN), São Paulo, Brazil, obtained its first criticality on 16 September 1957 and since then has served the scientific and medical community in the performance of experiments in applied nuclear physics, as well as the provision of radioisotopes for production of radiopharmaceuticals. The reactor produces radioisotopes ^{82}Br and ^{41}Ar for special processes in industrial inspection and ^{192}Ir and ^{198}Au as sources of radiation used in brachytherapy, ^{153}Sm for pain relief in patients with bone metastasis, and calibrated sources of ^{133}Ba, ^{137}Cs, ^{57}Co, ^{60}Co, ^{241}Am and ^{152}Eu used in medical clinics and hospitals practicing nuclear medicine and research laboratories. Services are offered in regular non-destructive testing by neutron radiography, neutron irradiation of silicon for phosphorous doping and other various irradiations with neutrons. The reactor is responsible for producing approximately 70% of radiopharmaceutical ^{131}I used in Brazil, which saves about US$ 800 000 annually for the country. After more than 50 years of use, most of its equipment and systems have been modernized, and recently the reactor power was increased to 5 MW in order to enhance radioisotope production capability. However, the control room and nuclear instrumentation system used for reactor safety have operated more than 30 years and require constant maintenance. Many equipment and electronic components are obsolete, and replacements are not available in the market. The modernization of the nuclear safety and control instrumentation systems of IEA-R1 is being carried out with consideration for the internationally recognized criteria for safety and reliable reactor operations and the latest developments in nuclear electronic technology. The project for the new reactor instrumentation system specifies three wide range neutron monitoring channels, one linear channel using a compensated ionization chamber for reactor control, one linear safety channel using a non-compensated ionization chamber (1 to 110% of reactor power), and one N16 power monitoring channel, in order to guarantee diversity in nuclear power monitoring channels. The interlock and safety logic based on electromagnetic relays will be replaced by electronic modules based on dynamic logic. All information on reactor operations, parameters and variables will be recorded in the reactor computer and displayed for the operators in the new digital control room.

1. INTRODUCTION

Modern I&C is based on the use of digital technology, distributed control systems and the integration of information in data networks, otherwise known as distributed control and instrumentation systems. This has a repercussion on control rooms, where the operations and monitoring interfaces correspond to these systems.

These technologies are also used in modernizing I&C systems in currently operational nuclear reactors. The new interfaces provide additional capabilities for operation and supervision, as well as a high degree of flexibility, versatility and reliability. An example of this is the implementation of solutions such as compact stations, high level supervision screens, overview displays, computerized procedures, new operational support systems or intelligent alarms processing systems in a modernized man–machine interface (MMI). These changes in the MMI are accompanied by newly added switch controls and new solutions in automation.

Incorporation of these systems in control rooms substantially modifies the interface between the operator and plant systems as compared to the classical configuration of the majority of current control rooms. In many cases the conceptual model of operations is also altered.

For adequate licensing, the design process of new control rooms requires an in-depth and detailed analysis of human factors engineering (HFE) in order to achieve a design of the interface between the operator and the plant systems that ensures and optimizes safe operation and supervision by the operating personnel under any plant operating condition, be it during normal operation, incidents, an event of plant or I&C equipment failure, or emergency operation.

In general, and unlike in the case of new facility design, processes for modernizing control rooms are projects that usually move forward slowly, with the operator interface and control being modified to different extents. As the individual impact of each modification in the interface is not usually important, and modifications tend to be developed over years, it is more likely that in this case the analyses of impact on the interface are limited. Also they are typically performed individually and lack an overall long term vision.

An adequate design of the interface between the operator and the systems will facilitate safe operation, contribute to the prompt identification of problems, and help in the distribution of tasks and communications among the different members of the operating shift.

2. MODERNIZATION OF IEA-R1 SYSTEMS

The IEA-R1 reactor is used for research in the areas of nuclear and neutron physics, nuclear metrology and nuclear analytical techniques. The reactor also produces some radioisotopes with applications in industry and nuclear medicine and has been used for training as well. It is a swimming pool type reactor, light water moderated and with graphite reflectors, designed and built by Babcock & Wilcox Co. The first criticality was obtained on 16 September 1957. Although designed to operate at 5 MW_{th}, in the first three years the maximum power operation was 1 MW_{th} and later 2 MW_{th} in order to accomplish radioisotope production.

However, the Brazilian demand for radioisotopes has grown rapidly, and it has become no longer possible to meet the national demand with local production. Until 1980, all ^{99m}Tc generators used in the country were imported. To meet the ever increasing demand, IPEN started producing its own ^{99m}Tc generator kits from fission ^{99}Mo purchased from Canada.

To reduce the heavy importation costs of the primary radioisotope ^{99}Mo, as well as to minimize increasing reliance on the few global suppliers of this product, IPEN concluded a decision making process by starting the production of ^{99}Mo inside its own nuclear reactor using the (n,γ) reaction. Aiming at the local production of ^{99}Mo, the precursor of the radioisotope ^{99m}Tc used in nuclear medicine, IPEN decided during the period 1996–2005 to upgrade the reactor power from 2 MW_{th} to 5 MW_{th} and operational cycle from 8 h/day for 5 days per week to 120 h continuous per week [1]. For this purpose, several modifications in the reactor systems and components had to be implemented. At the same time, an ageing management, inspection and modernization program was developed [2]. Although the basic structures are almost the same as the original project, several improvements and changes in components, systems and structures had been made in the reactor's lifetime.

To achieve the power upgrade and five day continuous operation goals to allow for commercial radioisotope production, the project was divided into three main groups of actions:

- Improvement of fuel elements;
- Adequacy of systems, structures and components, and
- Adequacy for radioisotope production.

2.1. Fuel elements and reactor core improvements

The core configuration was changed from a mixed HEU and LEU 5×6 array to LEU 5×5. The new core configuration provides a thermal flux of magnitude 5×10^{13} $cm^{-2}s^{-1}$. Also a beryllium irradiator was bought from CERCA, France, and is used to produce ^{99}Mo in the center of the nuclear core [3].

2.2. Emergency core cooling system (ECCS)

It is a fully passive system employing two redundant legs with four automatic valves designed to work in fail safe concept that sprays water directly from reservoirs into an uncovered reactor core to avoid fuel element melting. ECCS was designed for continuous operation cycles of 26 hours after reactor shutdown [4].

2.3. Cooling system pumps (primary and secondary loops)

The IEA-R1 reactor also had components changed due to ageing, and their operation is monitored by a new vibration monitoring system. The objective of this implementation is to establish a strategy to monitor and diagnose vibrations of the motor pumps used in the reactor cooling system primary and secondary loops to verify the possibility of using this system in a continuous manner.

2.4. Pool water treatment and purification system

After 50 years of operation without any huge alteration in design and working schedule, the water purification systems became obsolete because the contamination levels of the primary cooling system increased continuously, causing piping corrosion problems to occur at unacceptable rates. Therefore a new water treatment system has been implemented, as shown in Figure 1.

FIG. 1. Water treatment system.

2.5. Reactor control and safety elements

The older control and safety elements of the reactor began to show signs of ageing, and after study and inspection the Reactor Division Head decided for replacement with identical elements of fork type Ag–In–Cd. These new devices were fabricated at IPEN.

2.6. Radiation monitoring system

New portal monitoring equipment was implemented together with replacement of scintillators detectors for air duct monitoring.

2.7. Pneumatic transfer systems

Pneumatic transfer systems are classified as mechanical equipment widely operated for transport of large sorts of objects, samples and materials located at nearby terminals or even at separated ones. System applicability is often recognized in many fields such as medicine (hospital settings, clinical analysis labs), industry (steel, automobiles, mining, chemical, food, construction), trading (gas stations, movies, supermarkets, banks, e-commerce) and federal agencies (post services, federal courts, public enterprises). In the nuclear setting, these systems support also a vast array of applications, as a part of radioisotope production as well as short lived radiopharmaceuticals, including ^{67}Ga, ^{201}Tl, ^{18}F and ultra-pure ^{123}I. Furthermore, pneumatic transfer systems are also used at radioactive waste management plants and research institutes that apply neutron activation analysis. The old reactor transfer system dated from 1957 was refurbished with new electrical control circuitry in 1979, but no mechanical part or air injection system any modernization. After a study and inspection of the old system and a technical report, or SAR, about the risk of LOCAs, the reactor operational division decided to design a new one. This new project took into account a new air injection system, or blower, pipelines with improved materials, a replacement electrical panel control and an advanced shape and material for the irradiation capsules. The new system is shown in Figure 2.

FIG. 2. Pneumatic irradiation system, user station (left) and control panel (right).

2.8. New heat exchanger

Studies regarding ageing management were conducted according to IAEA procedures described in the Technical Report 338 Methodology for the Management of Ageing of NPP Components Important to Safety (2001) and Technical Document 792 Management of Research Reactor Ageing (1995). As a result of these studies, many critical components within the ageing management programme were identified. Also recommendations on the implementation of a testing schedule and a procedure to organize data and documents were made. The main result was a recognized need to carry out the inspection of both heat exchangers, the two primary pumps and the data acquisition system. During the monitoring process, difficulties were observed in the operation of the reactor at the 5 MW power level mainly due to the ageing of heat exchanger A from Babcock & Wilcox and vibration abnormalities observed at high flow rates in heat exchanger B from the CBC Mitsubishi Group.

By mid-2005, it was decided to work at the 3.5 MW power level and to provide a new heat exchanger with 5 MW capacity manufactured by IESA, Ltd., that could substitute the old heat exchanger A. This action demonstrated that the IEA-R1 reactor can be safely and continuously operated at a power level of 5 MW with the new heat exchanger. Both heat exchangers are shown in Figure 3.

FIG. 3. Heat exchangers A and B located in the reactor basement.

2.9. Data acquisition and storage

Due to the need of several other centres to access information about reactor nuclear, thermohydraulic and radiation protection parameters, a data bank for the IPEN scientific community was designed and implemented in which all IEA-R1 operational data are stored. Nuclear, thermohydraulic and radiation protection parameters, as well as digital data from a domestic meteorological tower (2010) are available to scientific staff. Figure 4 shows the new data bank system.

FIG. 4. Data acquisition system in the control room.

Thus, during the past years, a concentrated effort has been made in order to upgrade the reactor power to 5 MW through ageing, refurbishment and modernization programmes. One of the reasons for this decision was ^{99}Mo production. The reactor cycle will be gradually increased to 120 hours per week of continuous operation.

The next step of this programme is the modernization of the I&C room systems. Figure 5 shows the actual control room.

FIG. 5. Reactor control room with control console for the data acquisition system and closed circuit television surveillance.

3. REQUIREMENTS FOR DESIGN OF DIGITAL CONTROL ROOMS FOR NUCLEAR REACTORS

The design of new digital control rooms for power or research reactors constitutes a project of enormous technical and organizational complexity. On the technical level, all plant systems and their components must be analysed in great detail, taking into account their operation and supervision requirements under all possible plant conditions. As regards organization, it is important that all support divisions involved in operation participate in the design. Additionally, all those responsible for plant operation should be connected such that their work is adequately coordinated. All this must be incorporated in a complex overall plan of control design and construction, with the availability of the reference documentation constituting one of the most critical aspects.

The design of new control rooms must necessarily combine two fundamental aspects:

1) Capacities provided by the technology used in I&C systems. These systems, based on the use of digital technology, offer interesting possibilities for the development of the new MMI of these control rooms, which often are completely software based.
2) The application of a HFE programme to ensure a correctly designed and implemented MMI, increase human reliability and guarantee safe plant operation. In this respect, development of methodologies ensuring the adequate application of HFE is of key importance in order to ensure the optimum design of the MMI.

3.1. HFE in the design of digital control rooms

Design of digital control rooms begins with the definition of the objectives to be met by the control room. This is followed by the initial establishment of the operating principles that describe the practices and characteristics of the MMI necessary to fulfil these objectives.

The use of methodologies ensuring an adequate application of HFE allows for an optimum MMI design. These methodologies are based on international standards such as IEC 60964, NUREG-0800, NUREG-0711, NUREG-6393, NUREG-6400, IEC-6177, ISO-9241 and

NUREG-700. These methodologies are used during the MMI specification, design, testing and implementation phases.

3.2. Methodologies for the implementation of HFE in the design of new control rooms

Methodologies are developed to incorporate the different elements of a HFE programme in the design of new control rooms, taking into account the considerations below.

3.2.1. HFE programme management

The methodology is used to develop an integral plan for the incorporation of HFE principles in the design of the new MMI. The activities are defined, and a HFE team with the necessary qualifications is established. The human factors team must be part of the design team and work in close collaboration with the engineering organizations, while at the same time maintaining independence in its actions and freedom to assess designs and propose changes improving human performance.

3.2.2 Operating experience reviews

The objective is to ensure that HFE problems and issues of previous designs have been identified and analysed, with negative characteristics eliminated and positive characteristics maintained.

3.2.3 Analysis of functional requirements and assignment of functions

The next step consists in analysing the functions of each of the plant systems. The basic functions of the system are to be elaborated. The conclusion of this analysis must lead to the achievement of a comprehensive group of state parameters to be monitored and a list of equipment requiring control and status indication in the control room, of which all are intended to accomplish the systems' functions. In the analysis scope, also the assignment of the supervision and control of the functions to man, machine or both is to be taken into account, considering human strengths and weaknesses. The responsible party for surveillance and control must be identified.

3.2.4 Task analysis

The next step describes how the system is managed and which elements are required for optimizing this management for each of the plant systems. This analysis is used to identify the actions to be taken by personnel to fulfil the functions assigned and the information and control required to achieve this, which may encompass control and supervision devices, alarms, communications systems, procedures, specific training requirements, etc.

3.2.5. Personnel organization

The methodology analyses the personnel required in the control room and the organization of these personnel, taking into consideration the results of the assignment of functions and task analysis. As a result, the initial organization established in the early stages of operation may change.

3.2.6. Human reliability analysis

Within the framework of the analysis, consideration must be given to the mechanisms of human error in order to minimize their impact on the design of the control room MMI. An important aspect is their integration in probabilistic safety assessment.

3.2.7. MMI design

The results and information obtained during the previous phases materialize in the design and development of the content and form of the MMI. In new control rooms, the MMI will mainly consist of software. There will also be certain hardware components that serve as a backup in the event of MMI software failure. What is now to be defined is the hierarchical organization of the operations and supervision screens and the incorporation of the functionalities of the backup systems: computerized procedures, operations support systems, classification of operator support systems or intelligent alarms processing systems. New automation requirements may arise during this design phase.

3.2.8. Procedure development

In parallel with that stated above, the methodology develops the content and form of the plant operating procedures. This is based on the results of the task analysis and on HFE principles.

The objective is that the resulting procedures should be technically accurate, understandable, explicit, user friendly and validated.

3.2.9. Development of training

In addition to designing the interface and procedures, the process identifies training needs to provide the information required for the establishment and development of the learning tools, the evaluation parameters associated with these objectives and the mechanisms for review deriving from the on-the-job performance of the personnel.

3.2.10. Verification and validation

Prior to the beginning of the implementation and installation work, the design faces an independent critical analysis in order to check its conformity to HFE principles. Task support and design verification processes are carried out. The first process is intended to evaluate whether the designed MMI and its associated components allow for the complete performance of the tasks described in the task analysis. In addition, a check is conducted so that no component included in the MMI lacks an associated task. On the other hand, the design verification process assesses compliance with applicable HFE guidelines. An integrated validation of the system is also carried out. This assesses the MMI by means of tests based on the actuation of operating equipment to verify the acceptability of the design and the operability of the MMI. This is performed on a simulator through monitoring of operator actions in response to selected scenarios related to the overall operation of the plant

3.2.11. Design implementation

It is necessary to ensure that the design implemented in the control room is the same as that which has been verified and validated. The aim is to confirm that all HFE issues or discrepancies generated during the design, construction and verification and validation process have been resolved.

3.2.12. Surveillance of human interventions

The methodology provides a human intervention surveillance process that ensures that the conclusions of the verification and validation process are maintained over time without the need for the process to be periodically repeated.

Surveillance guarantees the following:

- The design can be used;
- The changes made to the MMI, the procedures and training required have no negative effect on human activity;
- The human actions may be performed within the time limits defined;
- The level of actuation established during verification and validation is maintained.

3.2.13. Freezing of control room design

The results obtained from the application of the methodology are used to definitively establish the concept and the operating principles that will initially lead to the design of the control room overall. Some important considerations are the spatial distribution and design of the supports housing all the operation and supervision devices (screens, alarms, controls, etc.) in relation to the anthropometric data of the population, the ergonomic calculation of console and panel profiles, considerations regarding the frequency of use of the controls and indications, response times in operation and requirements related to the distribution of the equipment in the plant.

The next step consists in harmonizing the design of the displays or screens that constitute the basis for normal plant operation with the hardware instruments that are installed on the panels and consoles for redundancy, backup or safety reasons.

4. MODERNIZATION OF THE IEA-R1 CONTROL ROOM

Most operating nuclear research reactors are undertaking projects for the modernization of their I&C systems that have an important impact in the control room, either because of the obsolescence of the current equipment, the incorporation of equipment facilitating operation and supervision or the fact that the plant is facing lifetime extension. These projects have already begun to alter the appearance of original control rooms, and it is reasonable to imagine that control rooms at the end of their service lifetime will have undergone a highly significant evolution and that they will surely be closer to the designs of today's new plants than their initial designs.

Given that the modernization process extends over multiple years, it is important to have an overall long term vision of the evolution of the control room beyond the specific projects carried out to incorporate new items of equipment that are often provided by different suppliers, who apply specific and not always uniform criteria.

The modifications made to I&C systems and the MMI in the control room have an impact on human activity in the control room, the required know-how and the training requirements. In order to guarantee safe and effective operation it is essential to specify, design, implement, operate and maintain the changes made to the control room MMI and to provide training on them. The application of an adequate HFE programme guarantees achievement of the above.

As has been pointed out above, the experience acquired in the development and application of HFE methodologies in the design of new control rooms is of great use for the HFE programmes applied in the modernization of the control rooms in the existing plants.

However, there are certain additional factors to be considered that are related to specific aspects of modernization, such as the following:

- Conceptual model of the new control room MMI;
- HFE guidelines;
- Migration strategy.

The processes of modernization of control room I&C systems and MMI may be divided into three types, depending on their complexity and on the degree of modification:

1) **Component by component change:** the old components are replaced with new digital units having identical functions. The changes to the MMI are minimal.
2) **Hybrid:** the instruments in the control room are replaced with new units. Some of these will be similar and others may be integrated in displays, giving rise to the so called software MMI. Furthermore, new displays are added to the control room MMI in order to improve operational efficiency and safety. The changes to the MMI may be significant.
3) **Fully computerized control room:** this is the most comprehensive modernization process. The old instruments are replaced with software instruments integrated in displays that form part of the compact work stations assigned to each operator. The MMI is completely different.

The aspects analysed below focus on hybrid modernized control rooms and apply to the definition of what the desired control room is like and how it evolves from its current status to that desired. These two aspects are included in the conceptual model of the control room and in the migration strategy, respectively. Figure 6 shows how the actual IEA-R1 control room evolves to the new design.

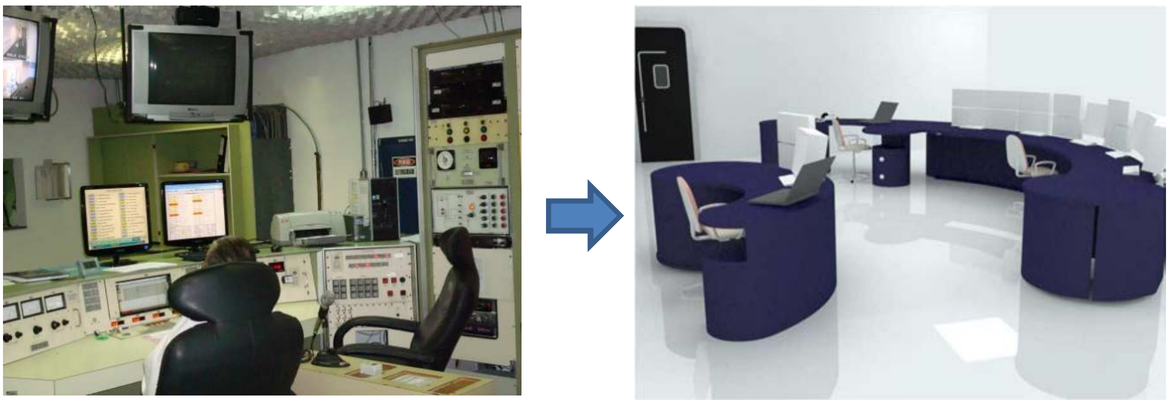

FIG. 6. Modernization of IEA-R1 control room.

4.1. Control room MMI modernization plan

In modernization processes, changes in the I&C systems and MMI carried out step by step, in which each change is incorporated in the control room individually without considering the characteristics of the MMI, usually pose integration problems.

Furthermore, if they are performed individually, without taking into consideration the overall modernization plan, they are often arranged and located in the control room attending only to space constraints, which possibly penalizes the capacity of the operator. In addition, as the modernization process continues, the changes implemented in this way may even degrade the effectiveness of the control room, which might imply the need for re-design with associated additional costs.

The modernization plan will allow for the following:

- An overall view prior to making changes of what the MMI will be like at the end of the process and in each of the phases of modernization;
- Guaranteed consistency and integration of the changes to the MMI over time;
- Assurance that whatever HFE problems might exist have been solved;
- Improvement of the existing MMI, making appropriate use of the technology associated with new I&C systems;
- Assurance that the MMI meets the regulatory requirements, both at the end of modernization and in the intermediate configurations.

The aspects considered in the plan are as follows:

- The conceptual model of the modernized control room;
- The migration strategy;
- The HFE guidelines.

The most important requirements to guarantee a satisfactory plan are as follows:

1) **Involvement of the operating personnel:** in order for the design of the MMI to be effective, it must take into account the capabilities and experience of the operator known as the "human centred" concept. Each plant has its own concept of operation based on the practices developed as a result of the modifications performed over the lifetime of the plant and on the experience of the operators.

2) **Capacities of new technologies available for the MMI:** In order to define the concept of the definitive MMI, it is necessary to understand how a control room with a modern MMI based on digital technology responds and the differences in the way it is operated compared to a traditional interface. Aspects such as operating stations based on a software MMI, the use of overview displays, the integration of 1E and non-1E software controls, the advanced treatment of alarms or the management of I&C and MMI failures must be considered.

3) **Application of HFE:** In order to ensure the adequate application of HFE in the development and subsequent implementation of the plan, a combination of factors is required, such as the following:
    - Commitment of the plant managers;
    - Personnel with knowledge and experience of HFE;
    - A methodology that allows for the performance of an adequate process for modernization.

4.2. Conceptual model of the modernized control room MMI

The objective of this model is to reflect the MMI that is to be obtained at the end of the modernization process on the basis of the concepts of operation and modernized MMI.

The operating concept of the modernized control room must show how the operating team will supervise and control the reactor with the modernized MMI. This includes aspects such as the following:

- Composition of the operating crew;
- Distribution of functions among the operating team and the I&C systems as a result of changes in the levels of automation;
- Assignment of supervision and control tasks and stations to the team members under different operating conditions;
- Coordination and supervision of the operating team;
- Operation in the event of failure or degradation of the I&C system or the MMI.

It is also important to describe how the status of the plant systems and components are monitored, how an alert is issued in the event of conditions requiring operator intervention, how incidents are diagnosed, how activities to solve incidents are planned, how the components and systems are controlled, how the different members of the operating team are coordinated and how operations are supervised. This concept may vary depending on the functionalities and degree of automation of the new I&C systems.

The concept of the modernized control room MMI underlines those aspects that provide a view of the modernized MMI coherent in light of the operating concept. The experience gleaned in the design of solutions for the MMI in new control rooms is of great use.

The following are examples of aspects to be analysed:

- Changes in the type and structure of the displays that supply information to the operating team and undertake its distribution in the control room;
- Installation of an overview display as a matter of necessity and what type of information it displays;
- Integration of traditional indicators and registers in information displays;
- Implementation of software controls, As to what extent they replace the traditional controls;
- Hardware controls that are maintained and others that are implemented as a backup in the event of software MMI failure;
- Changes in the way the alarms are displayed and filtering and prioritization criteria, if applicable;
- Distribution of panels, consoles and operations stations in the control room.

4.3. Migration strategy

The migration strategy defines the steps to be carried out in order to modernize the MMI as the modernization of I&C systems progresses; the aim being to develop the definitive MMI.

The strategy should contemplate aspects such as the following:

- Planning and evaluation of MMI transitions;
- Change in the operating concept of the MMI in each phase of modernization;
- Training and changes to the operating procedures;
- Incorporation of changes on the simulator;
- Integration of MMI failures in the operating procedures;
- Regulatory requirements;
- Evaluation of hybrid configurations.

An important aspect to be underlined in the strategy is the evaluation of provisional hybrid configurations. The performance of modernization by stages gives rise to provisional hybrid MMIs that must be evaluated in order to guarantee that there are no problems in MMI operation, coordination among the members of the team or monitoring of the procedures.

The following are examples of situations that might arise:

- Differences in the design and operation of different systems or different sections of the MMI when these are to be used together or alternately by the operator;
- Control tasks requiring the use of digital and analog controls;
- Differences in the level of automation of digital and analog controls.

4.4. Development of HFE guidelines

Two types of guidelines must be developed for the design of the MMI and for its implementation.

The MMI design guidelines are based on international standards, for example NUREG-0700, and supply the HFE criteria to be considered during the design of the different elements of the interface. These guidelines are used throughout the different phases of modernization.

The following are examples of their use:

1) **Specification phase:** the specifications of the MMI components requirements are drawn in diagrams and displays for operating consoles, panels, etc..
2) **Design phase:** guidelines serve as a basis for the development of the design specifications and style guidelines of the new diagrams and displays of systems in which the user interface changes from physical elements to computer displays (e.g., soft controls, diagrams, graphic items, information displays), as well as for the layout and design of the operations stations.
3) **Testing phase:** guidelines provide criteria for the verification and validation of the new MMI.

The guidelines for the process of MMI implementation describe the HFE programme to be applied while modernizing the MMI. They allow for the adaptation of a generic HFE programme based on international standards, for example NUREG-0711, to the specific modifications made in the plant. These guidelines are based on international standards, for example NUREG-0800, US Nuclear Regulatory Guide 1.174, NUREG-1764. They are used to classify the modernization of the MMI of each system from the point of view of HFE and depending on the level of risk. By means of this classification, a level of application of the HFE programme is associated with each modernization, in keeping with the dimension of the modification.

It is important that the scope of HFE activities is optimized and planned efficiently in order to prevent the cost of such activities from becoming burdensome in comparison to the cost of modernization. Furthermore, the specific methodologies for the modernization of control rooms should be oriented in this respect.

5. CONCLUSIONS

Given the enormous capacities of new digital I&C systems, the detailed analysis of HFE takes on a special relevance, the objective being to determine the ideal interface for the operator, taking into account the tasks to be addressed under any plant operating condition. This interface, jointly defined on the basis of the operating concept, defines the design of the new control room or modernization in the case of an existing control room.

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AGEING MANAGEMENT PROGRAMME FOR THE IEA-R1 REACTOR IN SÃO PAULO, BRAZIL

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1. AGEING MANAGEMENT PROGRAMME FOR THE IEA-R1 REACTOR

IEA-R1 is a swimming pool type reactor. It is moderated and cooled by light water and uses graphite and beryllium as reflector elements. First criticality was achieved on 16 September 1957, and the reactor is currently operating at 4.0 MW on a 64 h per week cycle. In 1996, a reactor ageing study was established to determine general deterioration of systems and components such as cooling towers, secondary cooling system, piping, pumps, specimen irradiation devices, radiation monitoring system, fuel elements, rod drive mechanisms, nuclear and process instrumentation, and safety system. The basic structure of the reactor from the original design has been maintained, but several improvements and modifications have been made over the years to various components, systems and structures.

During the period 1996–2005 the reactor power was increased from 2 MW to 5 MW and the operational cycle from 8 h per day for 5 days a week to 120 h continuous per week, mainly to increase production of ^{99}Mo. Prior to increasing reactor power, several modifications were made to the reactor system and its components. Simultaneously, a vigorous ageing management, inspection and modernization programme was put in place.

The following changes and modifications have been made during the last 15 years:

- The core configuration was changed from a mixed HEU and LEU 5×6 array to LEU 5×5. This latter provides a thermal flux of $\sim 5\times10^{13}$ cm^{-2}s^{-1};
- A Be irradiator was installed in the reactor core to produce ^{99}Mo;
- The emergency cooling system was made fully passive. Two systems with four automatic valves designed to work in fail safe mode were installed to spray water directly into the core to prevent meltdown;
- A new vibration monitoring system was installed to detect and monitor vibrations caused by the pump motors used in the cooling system's primary and secondary loops;
- The water purification system was replaced to reduce contamination induced by corrosion products;
- The old control and safety elements of the reactor were replaced with control elements manufactured by IPEN;
- New portal monitoring equipment was installed alongside scintillation detectors to monitor air ducts;
- Pneumatic transfer systems were installed;
- Due to excessive vibration and insufficient heat exchange the two heat exchangers were replaced;
- A new data acquisition and storage system was installed to meet increasing demand for information about nuclear, thermohydraulic and radiation protection parameters of the reactor;
- The beam holes were refurbished, consisting of general cleaning, substitution of the shielding and inclusion of a water reservoir to improve neutron shielding;
- The control room was modernized.

MODERNIZATION AND REFURBISHMENT OF THE RECH-1 REACTOR

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Abstract

The Chilean Nuclear Energy Commission (Comisión Chilena de Energía Nuclear, or CCHEN) has operated the RECH-1 research reactor since 1974. This reactor is located at La Reina Nuclear Centre in Santiago, Chile. It is a pool type reactor using LEU MTR fuel assemblies, light water as moderator and coolant, and beryllium as reflector. The reactor has been operated at the nominal power of 5 MW in a continuous shift of 20 hours per week, 48 weeks per year. The main utilizations of the RECH-1 reactor are radioisotope production and neutron activation analysis. Among the most relevant refurbishment and modernization campaigns undertaken at the reactor are: full core conversion to the use of LEU fuel, replacement of the cooling tower, improvement of the containment building by changing the doors and gates and by a better sealant for the penetrations, introduction of an additional source of water by connecting the raw water supply system to the emergency cooling system, improvement of the emergency ventilation system, introduction of a fire detection and alarm system for detection and mitigation to protect the I&C racks, introduction of a radioactive liquid release for those generated at the reactor, introduction of a delay tank degasification system and renewal of the environmental monitoring system. At present we are assessing the possibility of replacing the old analog electronics of control for new digital systems. Detailed descriptions of these diverse activities are presented in the paper.

1. RELEVANT FACILITY BACKGROUND

RECH-1 is a 5 MW open pool type research reactor cooled and moderated by light water, using beryllium reflectors and MTR type fuel. The reactor is located at La Reina Nuclear Centre in Santiago, and its first criticality was achieved on 13 October 1974 using HEU (80% ^{235}U) plate type fuel assemblies fabricated by the United Kingdom Atomic Energy Authority (UKEA) at Dounreay, Scotland, with uranium enriched in the USA. A second load of fuel assemblies was also fabricated by UKEA using British HEU with 45% ^{235}U. From 1985–1998 the reactor operated with a mixed core configured with HEU fuel assemblies of these two different enrichment levels.

After CCHEN developed the capability to produce U_3Si_2Al, a total of 47 LEU fuel assemblies were finally manufactured and delivered to the RECH-1 reactor. From 1998 to 2006 the RECH-1 reactor operated with a mixed core; at this time configured with HEU (45% ^{235}U) and LEU fuel assemblies. The full core conversion to LEU of the RECH-1 reactor was completed on 11 May 2006.

A total of 58 spent fuel assemblies from the RECH-1 reactor were returned to the Savannah River Site, USA, in two shipments; both within the frame of the US Foreign Research Reactor Spent Nuclear Fuel Acceptance Program. The first shipment, containing 28 spent fuel assemblies, was carried out in August 1996, and 30 additional spent fuel assemblies were shipped in December 2000. A last shipment under the same programme was done in February 2011, including 40 fuel assemblies with 45% ^{235}U and 22 90% ^{235}U fuel assemblies belonging to the Lo Aguirre research reactor.

In recent years the reactor has been operated at the nominal power of 5 MW in a continuous shift of 20 hours per week, 48 weeks per year.

The emphasis in the utilization of the RECH-1 reactor is placed on radioisotope production, neutron activation analysis and geological samples irradiation.

2. MODERNIZATION AND REFURBISHMENT SCOPE

The main motivation of the modernization and refurbishment carried out in the reactor was aimed to guarantee continuous and safe operation within its operational limits and conditions. These activities were the replacement of components and systems that had reached their lifetime. New systems have been installed, and some of the current systems have been improved. All these improvements have been made in order to increase safety and security and to comply with new environmental regulations.

The main modernization and refurbishment activities are described in the following sections.

2.1. Reduced enrichment

As said above, a complete programme for reduce fuel enrichment was carried out, reducing enrichment from 80% to 19.75%. The LEU fuel assemblies were fabricated at the Chilean fuel fabrication plant and based on a dispersed fuel containing LEU as U_3Si_2. Irradiation for qualification of the fuel elements manufactured in Chile was performed in the High Flux Reactor, Petten, Netherlands, in May 2003 to 2004. Post irradiation tests were successfully completed in October 2005.

2.2. Replacement of the cooling tower

The original reactor cooling tower consisted of a structure 4.6 m high and 6.7 m wide constructed from European redwood, but use and exposure to sunshine for more than 35 years had damaged the tower, reducing its performance. A new battery of modular towers, three modules with four towers each, and two individual towers have replaced the original to recover cooling capacity. A wireless control system to operate the towers from the control room has been installed.

2.3. Improvement of the containment

Reactor doors and gates are equipped with seals and are normally kept closed. During normal operation, the building is maintained at a negative pressure of 25 mm water to ensure leakage only from outside to inside. These reactor doors and gates were replaced by new reinforced ones, while the main reactor gate has a special kind of seal using compressed air in rubber conducts against the wall to keep the necessary negative pressure.

2.4. Improvement of the emergency cooling system

The reactor tank water level is constantly monitored by six water level probes set at various heights, which initiate audible and visual alarms when the tank water rises or falls. If the reactor tank level decreases too much, water from the de-mineralized storage tank (10 m^3) flows into the reactor tank. In case of an emergency situation, if the storage tank is empty, raw water could flow to the reactor tank. This raw water comes from a large storage tank (120 m^3) outside the reactor building; if this tank is empty also, the system allows connecting the emergency flow water directly with the raw water supply pipes.

2.5. Modification of the ventilation system

As stated previously the reactor hall is a sealed building, and the ventilation system is designed to maintain a negative pressure. The original ventilation system was based on 4.3 volume changes per hour, in which one third is fresh air and two thirds are re-circulated. In the case of a nuclear incident or high radiation alarm the air flow is forced to pass through a charcoal filter before being discharged through the stack. At present, in emergency conditions, all the air re-circulates through the charcoal filter until its activity decreases.

2.6. New fire detection and mitigation system

A new system for fire detection in the control room, ancillary building, ventilation plant and electric power supply station was installed. An automatic system under electronic control to extinguish the fire at the reactor control racks was added. Its fire suppressant agent is FM-200 gas.

2.7. Degasification of the delay tank

The vertical disposition of the delay tank produces the formation of a gas bubble, that is, dissolved air in water, at the top of the tank that increases continuously with reactor operation time. Thus the reactor cannot operate for more than 72 continuous hours. To solve this problem we installed a hydraulic ejector with Venturi effect at the top of the tank, and the vacuum formed by the pass of compressed air through the ejector allows the evacuation of the gas.

2.8. Liquid radioactive discharge system

The purpose of this new system is to drain any radioactive primary coolant water that has spilled, leaked or overflowed from the pool, delay tank well, beam tubes, water treatment plant or also from the walkway drain. The water is piped to two low level effluent tanks below the station in the main tunnel. After a suitable decay period, the water is then emitted into an external drainage system for disposal.

2.9. Radiation monitoring system

The original system was formed by four gamma monitor units, with each monitor having an ionization chamber connected. This was provided for personnel protection in the reactor hall and active filter room areas. Each gamma monitor initiates an urgent audible and visual alarm when high gamma activity is detected. To improve protection three new monitors were added, one of them to detect releases of noble gases from the reactor pool.

2.10. Increasing security

Important activities have been carried out in the La Reina Facility in order to hinder as reasonably as possible access to the nuclear material that exists there. These improvements in reactor security have been done to fulfil a contract with shared financing between the Sandia Corporation, USA, and the Chilean Nuclear Energy Commission that was signed in December 2005. In the frame of this project we can mention: fabrication and installation of a reinforced steel door at the hardened part of the control room, installation of closed circuit television in many reactor areas, installation of reinforced steel window grates over windows to the reactor from the control room, addition of infrared and microwave sensors and cameras over the reactor pool, installation of a balanced magnetic switch on all doors to the vault, procurement and installation of a personnel access door for first floor access to the reactor ancillary building and procurement of radio communication system.

2.11 Data acquisition system

The installation of a system to monitor core reactivity and other operational reactor parameters has been very helpful for an improvement in safe reactor operation.

3. MAIN AGEING MANAGEMENT PLAN

3.1. Reactor electronic replacement

The electronics of the reactor are based on 1970s technology, and therefore its maintenance and replacement of parts are difficult due to the lack of spare parts and discontinuation of products. As a result, electronics repair and maintenance has spawned a rare and hybrid mixture of old and new components. This fact has been evaluated by electronics personnel, who have recommended its modernization. A first assessment was done by traditional electric engineers not specialized in the nuclear field, while a second step is for us to participate in the IAEA Technical Meeting on "Research Reactor Ageing Management, Modernization and Refurbishment, Including New Research Reactor Projects" to learn about other reactors and what they have done.

AGEING MANAGEMENT OF CARR

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1. INTRODUCTION

China Advanced Research Reactor (CARR), which is a beam type research reactor, has been under construction in CIAE. Its power is 60 MW, and it contains an inverse neutron trap compact core. CARR uses H_2O as a coolant, and the core is surrounded by a D_2O reflector. The unperturbed thermal flux is 8×10^{14} $cm^{-2}s^{-1}$. There are nine tangential horizontal beam channels available for experiments and 25 vertical irradiation channels for radioisotope production. Other aspects of the reactor are given in Table 1, in addition to these:

- Ground was broken in the year 2002;
- After eight years of construction, the reactor was completed, and first criticality was in 2010;
- Full power operation in 2012.

TABLE 1: MAIN PARAMETERS OF CARR

Fuel meat material	U_3Si_2-Al dispersion
Cladding material	6061 Al alloy
Fuel U^{235} enrichment (wt%)	19.75+0.20
Initial loading of uranium (kg)	55.534
Mean burnup of spent fuel (%)	32.15
Core height (mm)	850
Equilibrium diameter of core (mm)	399.2
Water pool depth (m)	15.0
Inner diameter (mm)	ϕ5500
Heavy water reflector inner diameter (mm)	479
Outer diameter (mm)	2200

We have just started the operation and commissioning of CARR. It is the best time to plan a long period of preventive maintenance and a periodic examination programme. This includes planning the missions of ageing management and the management of different components and systems.

In the paper safety management in the initial criticality tests of CARR is introduced. In order to illustrate the implementation of research reactors, some examples are given.

2. THE AGEING MANAGEMENT OF REACTOR STRUCTURES

2.1. Radiation effects in reactor structures

The CARR structures include the guiding tank, heavy water tank, decay tank and core vessel.

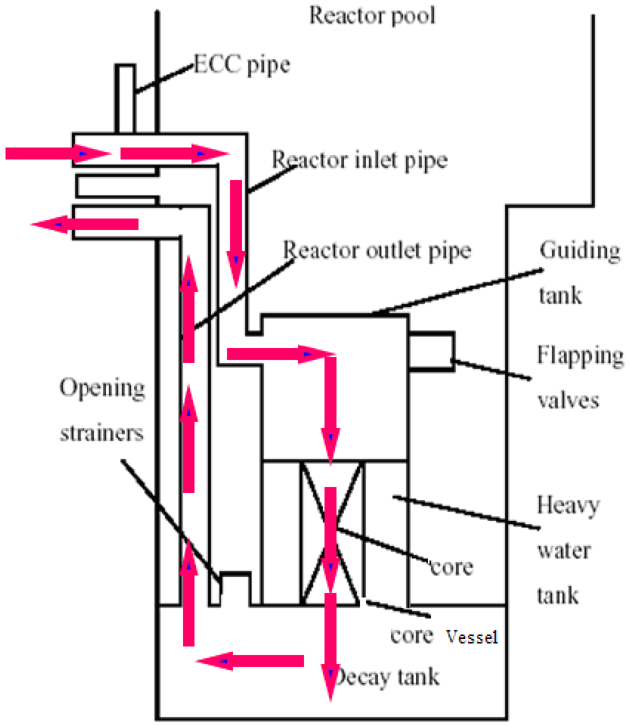

FIG. 1. CARR structures.

Based on previous operational accidents, the method of system safety analysis is applied to analyse the safety of reactor structures under an irradiation environment. Radiation effects in metallic material include the changing of material properties and size variation of the substrate.

The material of the guiding tank, heavy water tank and decay tank is 0Cr18Ni10Ti, and the core vessel uses 6061-T6 Al alloy. Ageing management must take all factors into consideration. The design life of 0Cr18Ni10Ti is 30 years, but the design life of 6061-T6 Al alloy is 10 years. So in order to get correct information, attention should be paid to the following points:

- Structural wall thickness due to spot corrosion during years of service;
- Integrated neutron flux, thermal and fast, of the core vessel should be acceptable for the aluminium alloy;
- Total thermal cycles imposed on the core vessel will be considered.

2.2. In-service inspection

The main methods for the ISI of structures are as follows:

- Visual detection method (underwater camera, remotely operated vehicles, etc.);
- Material irradiation monitor device for the CARR core vessel.

TABLE 2: MATERIAL IRRADIATION MONITORING FOR CORE VESSEL

Number	Runtime (y)	Thermal flux ($cm^{-2}s^{-1}$)	
		Core vessel	Examination
1	3	2.0×10^{22}	2.3×10^{22}
2	7	4.7×10^{22}	5.4×10^{22}
3	10	6.7×10^{22}	7.8×10^{22}
4	16	1.08×10^{23}	1.24×10^{23}
5	22	1.48×10^{23}	1.71×10^{23}
6	28	1.89×10^{23}	2.18×10^{23}

3. THE AGEING MANAGEMENT OF THE SEALING STRUCTURE

There are legible durable identifying codes on all welds of CARR. They should be checked regularly by sight to make sure that there is no external corrosion or other problems.

We have applied for exemption from examination on the guiding tank, heavy water tank and decay tank because they are limited by structure, examination technology and feasibility.

The most important sealing structure is located on the interface between the heavy water and light water. We have 21 vertical channels and 9 horizontal beam tubes, and all of them penetrate the interface. Double means for sealing and a leakage inspection system have been used.

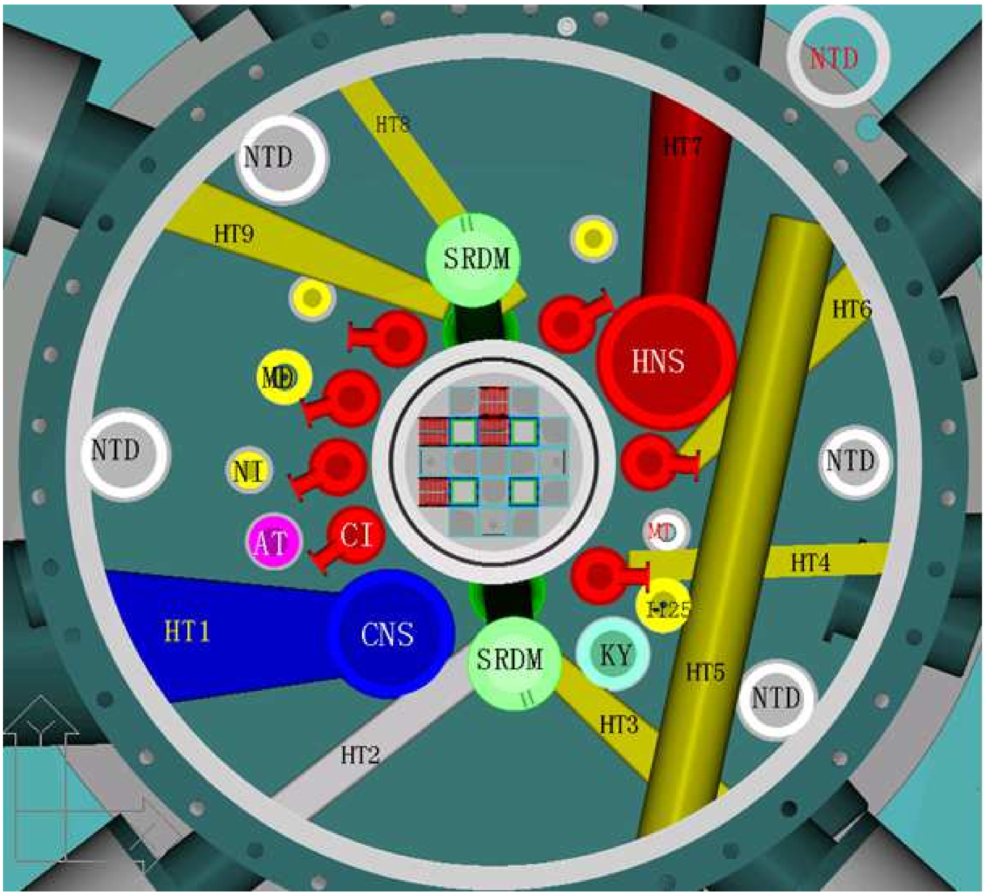

FIG. 2. Horizontal cut-view of CARR.

4. THE AGEING PROBLEM OF THE SECONDARY COOLING SYSTEM

The function of the secondary cooling system is to supply cooling water for the main heat exchanger and heavy water heat exchanger. This system uses water from a deep well, and the cooling tower is exposed to a natural ventilation environment.

Poor quality of injected water will cause the systems to corrode, and scale will form, both of which will damage the pipe and equipment and especially the heat exchanger, thus leading to a great reduction in heat transmission.

Because we have neglected the work of rust proofing in the earlier stage, the surfaces of all the valves were seriously corroded and some of them, as well as several machines, could not work.

To solve the problem, there are two ZJ-11500 type automatically fed facilities have now been used in the secondary cooling system. Also, we are building a filtration system to remove the suspended matter in water and using two chlorine dioxide water purifier facilities to remove the algae and microorganisms. Rust proofing can be accomplished during this stage of reactor lifetime.

We use an Alfa Laval plate heat exchanger from Sweden as the thermal exchanger. We analyse the quality and scale of the thermal exchanger once a year and make a decision about descaling. We do major repairs every six years, which includes replacing the seal gasket, removing rust and checking the condition produced by corrosion.

5. IN-SERVICE INSPECTION

The inspection includes type 2 and 3 safety systems and the assembly unit. The method mentioned in the preceding section, that is, sighting monitoring, irradiation monitoring, continuous leakage monitoring, can also be used in ISI. In addition, surface examination and dimension inspection can be used as well.

The project of ISI must be finished in a limited period. The length of the period should be decided by a conservative assumption.

TABLE 3: ISI PROGRESS SCHEDULE

Inspection period (y)	Served years (y)	Completion (%)	Main inspection restriction
10	0-5 5-10	34 100	66 100
20	10-15 15-20	34 100	66 100
30	20-25 25-30	34 100	66 100

6. PERIODIC TESTING AND EXAMINATION

Besides ISI, we also test and examine regularly to assess the capacity of ageing management, for example:

- The blocking situation of fuel assembly channel before operation;
- Exact positions of control rods once per 6~12 months;
- Lifting time of the safety rod before operation or once per 6 months;
- Time location of control rods before operation or once per 6 months.

In view of operation experience, checking the lifting time and time location of the control rods is extremely important since it can be an indication of serious problems.

7. CONCLUSION

The paper introduced the ageing management of CARR, including the ageing management system, instances of ageing components, difficulties we met and follow up plan, and put forward some suggestions on strengthening and promoting ageing management. We hope to enhance international exchange and collaboration. It is a great challenge to do research on how to manage ageing effectively in the beginning of life, but it will enhance the safety of reactor operation, extend the life of the reactor and improve the quality of operation, making this work very meaningful.

AGEING MANAGEMENT OF THE KINSHASA TRICO II RESEARCH REACTOR COMPONENTS AND STRUCTURES

A case study of the 5 tonne overhead travelling crane and the ventilation system inlet and outlet

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Abstract

The cable isolation sheath of the overhead traveling crane became fragile and brittle on a length of more than two meters. This degradation of the sheath has been caused by the ageing of the cable and the effects of heat, with changeable ambient temperatures in the reactor hall combined with the Joule effect, in relation to the cyclic use of the crane. This ageing effect was discovered when a failure occurred at the end of a nearly completed routine operation of displacing slightly radioactive spent ion exchange resins into waste storage. During the return of the hoist to its rest position, a severe short circuit happened between the cable and the mass of the hoist motor support, followed by a strong detonation that produced sparks and immobilized the overall system. A visual examination of the cable showed a change of its physical properties as mentioned above. A further investigation showed that two master contacts of the hoist were also burned. The bridge crane was inspected and certified in January 2010 by a competent authority eleven months before the event. On the other hand, the reactor ventilation system started presenting its limits in the late 1990s after more than 17 years of operation. Many failures occurred in the extraction motors and the air conditioning system, causing both a temperature increase and a lack of negative pressure in the reactor hall. The crane problem was solved after replacing the damaged cable and the two burned contacts. A review of the overall status of the bridge crane by the licensing authority is scheduled before the end of 2011. The paper also describes steps related to the renewal of the air inlet system and the restoration of negative pressure in the reactor hall.

1. INTRODUCTION

The Democratic Republic of the Congo (DRC) is a country in Central Africa with more than 60 000 000 people and a member state of International Atomic Energy Agency. It joined the IAEA in 1961 and subscribed to peaceful nuclear activities.

DRC has two research reactors; the first is a 50 KW swimming pool type TRIGA Mark I reactor built by General Atomics in 1959. It operated for 11 years and was definitely shut down in June 1970 but is not decommissioned yet. The second reactor called TRIGA Mark II is also a swimming pool type. It was built under General Atomics specifications and supervision with the assistance of experts from SCK·CEN, Mol, Belgium, in the years 1970–1971. TRIGA Mark II is a 1 MW steady state reactor with pulsing capability. It reached its first criticality in March 1972, such that that its facilities are now around 40 years old, and its ageing problems are electronic, mechanical, and structural.

Due to rapid development in the field, we experience real problems in obtaining spare parts for the overall plant. Each component is affected by the negative impacts of fatigue, degrees of wear, deformation and sometimes by some faults in fabrication.

1.1. TRIGA Mark I

As stated earlier, the TRIGA Mark I reactor operated from 6 June 1959 to 29 June 1970 at 50 kW of steady state power. Although it was definitely shut down in 1970, its swimming pool is still full of demineralized water and contains all of its fuel spent elements. A programme for reactor decommissioning is being written.

An ageing management programme has been implemented and is operational. It mainly concerns:

- Measurement of water features to prevent fuel element corrosion;
- Improvement of pool water quality and maintenance of its conductivity below 1 µsiemens per cm with pH around 5.9.

During an inspection of this reactor's internals in December 2008 we performed verification of the status of all its fuel elements and found that all were in good physical condition.

This practice permitted us to develop an engineering design in the form of a scissors-like pick up tool, such as can be seen in Figure 1.

FIG. 1. Scissors for picking up objects.

In this facility the next operations are routinely practiced:

- Cleaning of the swimming pool;
- Continuous cleaning of the demineralization of pool water;
- Checks of the physical state of spent fuel (twice per year);
- Managing of the water level in swimming pool.

1.2. TRIGA Mark II

As said previously, the TRIGA Mark II research reactor became critical and operated at 50 KW until its power level was upgraded to 1 MW steady state with a pulsing capability of 1600 MW) in November 1974. TRIGA Mark II has been in extended shutdown state since Nov 2004, after 32 years of operation.

In the present meeting, the study concerning ageing is based on two cases. The first concerns a short circuit that happened on the power cable of the overhead crane due to ageing, while the second concerns the renewal of the air inlet system and the restoration of the under pressure in the reactor hall including the overall system.

2. SHORT CIRCUIT ON THE POWER CABLE OF THE OVERHEAD CRANE

2.1. Description of the equipment

TRICO II utilizes a 5 tonne overhead travelling crane assembly of type ZLX50-16.5 manufactured by R. Stahl AG, a German manufacturer, and composed of one bridge crane, a hoisting gear coupled with a trolley and electrical equipment. Control is supplied through one isolating switch and one master contactor. Movement can be entirely disconnected from the control station by the master contactor, which works as a fuse cutout. Travelling cranes equipped with pole changing travelling motors for two travelling speeds are supplied inclusive to the reversing contactors and time lag relays for delayed action of the brake.

Current is supplied along the crane bridge via a 13.5 m flexible power cable insulated by neoprene and separate cable carriages and track. Working voltage is 380 V and frequency 0.02 Hz. Detailed crane technical characteristics are presented in Table 1.

TABLE 1: TECHNICAL CHARACTERISTICS OF THE TRICO II 5 TONNE BRIDGE CRANE

Designation	Stahl overhead travelling crane, type ZLX50-16,5, with crane bridge, hoisting gear with trolley and electrical equipment
Capacity	5000 kg
Span	16.53 m
Drive	Gear motors with built-in brake
Motor output	2×0.5 kW, 40 DC%
Travelling speed	25 m/min
Paint coating	Anti-corrosion Ancora "grey"
Hoisting gear lifting height	18 m
Hoisting speed	8 m/min
Hoist motor output	8.8 kW, 40 DC%
Length of control cable	13.5 m
Working voltage and frequency	380 V, 0.02 Hz
Control switch type	SW4

FIG. 2. Different views of the crane components: bridge, trolley, power cable, etc.

2.2. Incident and corrective action undertaken

Figure 3 below shows different views of a piece of the damaged power cable with many cracks, physical breakup and detachment of the insulating outer sheath from the conductor for a length more than two meters. This may be considered as the result of multiple cyclic flexions exerted on the wiring insulation that became brittle with age. The part electric cable plays to preserve wiring integrity is well understood including several critical functions such as electrical isolation of conductors as well as addition of some measure of mechanical stability. Obviously, the first line of defence to assure appropriate wiring function was lost in the case under study.

The degradation of physical properties mentioned above can be seen as the result of the combined action of four factors: ageing, thermo-electrical effects, severe environmental conditions and last but not least the heightening effect of the non-application of a proper ageing surveillance programme concerning sheath testing. This variety of testing should be coupled with visual inspection to monitor overall status and insulating capacity over time.

Thus the sheath fault described above was in large part the triggering cause of the cable breakdown that resulted in acute short circuits between the cable conductors and the mass of the hoist motor support with production of sparks, destruction of contactors C1 and C2 of the master contactor system and the immobilization of the entire system.

The explanation of these negative effects relies on the fact that sheath damage, if not associated with direct damage to the cable insulation, will seldom lead to fast breakdown and failure of the cable installation. From the moment of damage until the appearance of the breakdown, many months or even years can pass, as was the case at CREN-K.

In order to solve the problem, the damaged cable was partially replaced over a length of approximately ten meters, and two new contactors were installed. The reactor team took advantage of the repair period to perform preventive maintenance of crane components. Drive motors, hoisting gear, traversing gear, hoist cable, etc., were inspected and lubricated as necessary. A review of the overall status of the bridge crane by the licensing authority is scheduled before the end of 2011.

FIG. 3. Different views of the defected cable piece.

3. THE RENEWAL OF THE AIR INLET SYSTEM AND THE RESTORATION OF NEGATIVE PRESSURE

3.1. Facility description

The TRIGA Mark II reactor hall is a concrete room of 19.34 m length, 17.36 m width and 13.60 m height, representing a total volume of 4560 m^3.

In the inlet part of the ventilation system, admission and recycling are accomplished by two independent cooling machines through Cambridge high efficiency particulate air (HEPA) filters. Emergency ventilation adds uncooled air into the reactor hall. In the outlet, air is sucked from the reactor hall and pumped into a monitored chimney through Sofiltra pre-filters and HEPA filters by two separate extraction motor groups A and B, which have two ventilators each.

A separate security circuit containing Sofiltra pre-filters, a charcoal filter and HEPA filters collects air from the four neutron beam tubes, thermal column and reactor pool water surface, all of which is pumped into the chimney.

The reactor hall operates in a negative pressure condition of about 10 mm Hg in ordinary situations. In an emergency case all the exhaust machines, including emergency ventilation mentioned above, turn automatically to high speed, and the negative pressure becomes higher.

3.2. Ageing observations

3.2.1. Ventilation system machines

After several years of operation, many problems began to appear both in the inlet air condition system and on the motors of the outlet air system, such as the power cable isolation becoming fragile and brittle, and overheating and burning of the motor. The ultimate solution to these problems was to renew the overall system.

3.2.2. Inlet and outlet filters

In general all the filters are about 30 years old, such that they are physically very worn, while others have significantly deformed. Our efforts to replace them were in vain because of a lack of supplier.

3.2.3. Lack of negative pressure

When the new inlet system first operated, there was an over pressure in the reactor hall, which is an unacceptable operating condition from a nuclear point of view.

3.3. Provided solutions

3.3.1. Ventilation system

In general, choosing a ventilation system involves knowing first how much heat capacity is necessary to cool the hall.

In our case, the initial ventilation system provided references to old machines features, so no sophisticated calculations were necessary. As far as activities in reactor hall are concerned, no changes have taken place.

To resolve this problem, we contacted a local specialist in ventilation who could provide equipment with the same features. This specialist acknowledged provision of corresponding equipment. The work time was approximately six weeks, the cost was US $38 564, and the work was expected to be done mainly by a specialist team with the support of the reactor team. The renewed inlet and outlet systems are operational, and the average temperature is 24°C.

3.3.2. *Depression condition*

This problem was solved by establishment of the recycling of most of the air in the reactor, which was deemed sufficient. Actually the pressure drop between the reactor hall inside and outside is 10 mm Hg. From the nuclear point of view, this is a safe condition.

3.3.3. *Filters*

All filters are affected by age, and the only solution is replacement with new ones. Before we could find a pre-filter supplier, we reused the old set after an air treatment.

4. CONCLUSION

The ageing problem is a reality after a certain number of years of operating, and CREN-K wishes to share with experts from various countries the exchange of experiences on general topics concerning ageing, component acquisition and replacement projects, specifically, problems due principally to not having a ready supplier. On the specific topic of ageing of electric cables for the bridge of 5 tonnes used in the TRIGA Mark II hall, we shared how to resolve the associated problems.

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AGEING MANAGEMENT BASED MAINTENANCE AT ETRR-II

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1. INTRODUCTION

The second research reactor of Egypt, ETRR-II, is an open pool type multi-purpose reactor of 22 MW that has been in operation since 1998. The main purposes of ETRR-II are the production of radioisotopes, personnel training, research and development. Ageing management of ETRR-II is based on the maintenance programme. This programme was planned to have a proactive maintenance strategy that mainly depends on adequate preventive and predictive maintenance. Screening of SSCs is based on safety classes, needs and quality levels. The operation condition and maintenance history are the basis to assess the level of degradation. The ultimate aim of the maintenance team is to perform adequate maintenance work only when it is really necessary.

The objectives of the ageing management based maintenance are to maintain and improve equipment availability, confirm compliance with operational limits and conditions, and detect and correct any abnormal condition that might affect reactor safety. The programme also helps to detect trends in ageing so that a plan for mitigating ageing effects can be prepared and implemented.

Reactor service conditions are monitored through many parameters and conditions such as conductivity and pH of coolant water, water radiochemical analysis, thermal and hydraulic parameters, a leak detection system, corrosion rate of piping, vibration level of rotating equipment and control rod drop time.

2. AGEING MANAGEMENT PRINCIPLES

Ageing management based maintenance principles in ETRR-2 are:

- Reactor operation conditions and maintenance history are the basis to assess the level of degradation;
- Complying with the requirement of the regulatory bodies helps to achieve long life of SSCs;
- Effective management requires a good and balanced maintenance strategy;
- Screening of SSCs is according to their importance to safety;
- Degradation mechanisms could be corrosion, embrittlement, fatigue, wear and fabrication defects;
- Ageing management is a continuous process;
- The ultimate aim is to perform maintenance work only when it is really necessary;
- The maintenance team carries out the aging management programme.

3. SERVICE CONDITION MONITORING

Service condition monitoring is one ageing management strategy used at ETRR-2. Examples of applied service condition monitoring are:

- Core pressure difference;
- Heat exchanger pressure difference;
- Cooling pumps pressure;
- Coolant flow rate and temperature;
- Vibration level of primary cooling pumps;
- Conductivity of water coolant;
- pH value of water coolant;
- Radiation dose over the reactor pool;
- Water level of the reactor pool;
- Water radiochemical analysis;
- Non-dissolved elements level in coolant water;
- Control rod drop time.

4. PLAN–DO–CHECK–ACT (PDCA) PROCESS CYCLE

ETRR-II applies a process technique called the PDCA process for ageing management based maintenance. This technique is implemented for reactor SSCs as a continuous process. It is worth mentioning that this technique greatly helps to keep the reactor within its operating limits and conditions. The PDCA process technique includes: 1) understanding the ageing mechanisms of SSCs; 2) definition of system ageing management; 3) SSC operation or use; 4) SSC inspection, monitoring, and assessment; and 5) mitigation and repair. Figure 1 shows the cycle of the PDCA process.

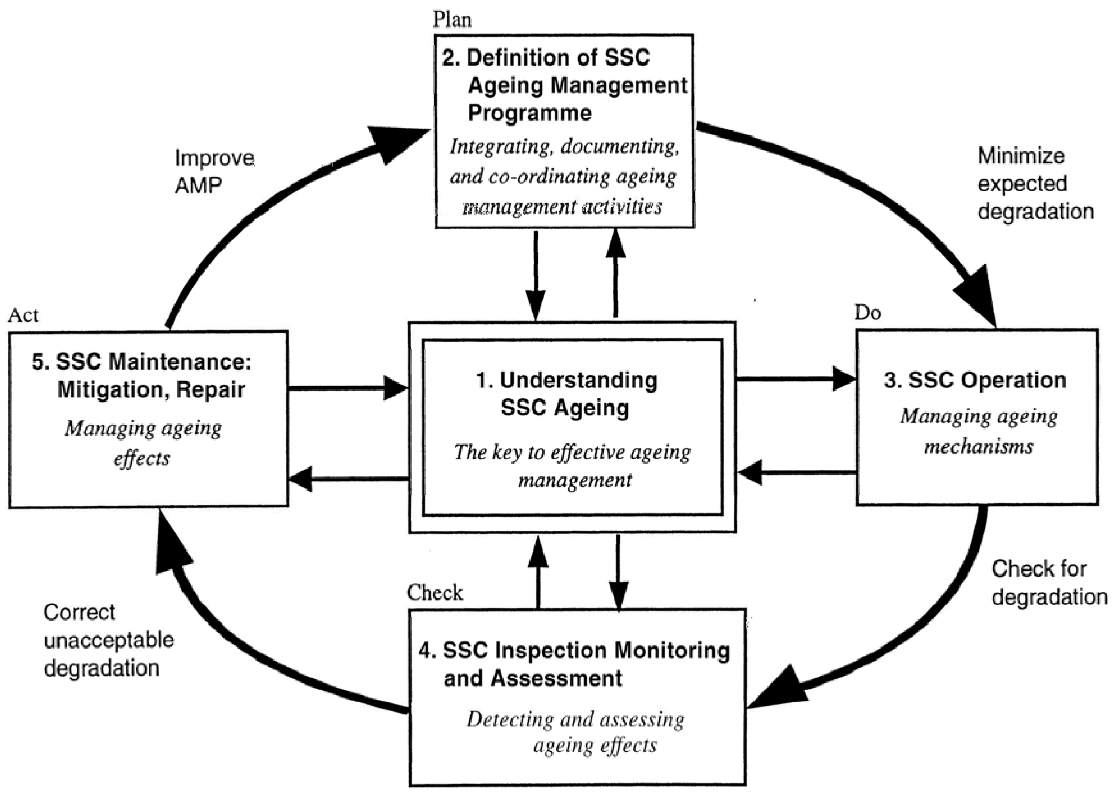

FIG. 1. PDCA process.

Applying the PDCA technique to ageing management shows that the program is dynamic and works effectively, and consequently no damage has occurred to reactor components.

5. APPLICATION OF THE PDCA PROCESS ON THE SECONDARY WATER COOLANT SYSTEM

Ageing management of the secondary water coolant system is an example for applying the PDCA process at ETRR-2. This system is responsible for the final dissipation of the energy produced in the reactor. This energy is transferred as heat from the primary to the secondary circuit through heat exchangers. The energy is ejected to the environment through a cooling tower made of galvanized steel and consisting of six separate cells; each cell is equipped with its own blower and its own water supply. The piping material of the system is carbon steel with different sizes ranging between 15.2 and 61.0 cm (6 and 24 inches). A simplified flow diagram of this system is shown in Figure 2. Untreated water can cause harm to the equipment and waterlines, as it contains impurities like dissolved and suspended solids and microbiological organisms. These impurities can cause three main problems, scale, corrosion and microbiological growth, which are known as the industrial water triangle visualized in Figure 3. These problems can occur jointly, causing reductions in the thermal efficiency of the circuit due to loss of heat transfer capacity, puncturing of waterlines, pitting, irregular flow of water, system shutdowns and finally shorter lifespans of the equipment. These problems should thus be overcome by the proper application of mechanical and chemical means. The chemical means include treating the circulating cooling water with biocides and algaecides as well as scale and corrosion inhibitors. Due to new restrictive laws concerning the environment, all these compounds must be non-toxic and biodegradable.

As the cooling system is a semi-open circuit, particles such as dust and sand may get inside. A cooling tower is an air washer; it scrubs all of the incoming air with water. It collects insects, fly ash, plant and seed fibres, animal hair, clothing fibres, paper, leaves, grease and more. Besides these, a cooling tower basin contains all types of deposits formed due to deposit producing organisms. The accumulation of dirt and debris creates a natural habitat for growing microorganisms and can be dangerous, as a dirty tower basin robs the tower system of treatment chemicals and acts as a breeding ground for microorganisms. Keeping the cooling tower clean and free of mud and sediment is thus the best solution for preventing the growth of microorganisms and for the prevention of corrosion. Solid separators and sand filters are mechanical methods for the removal of suspended matter entrained in cooling water systems on a regular basis. Constantly recirculating the cooling tower water through a solid separator will remove approximately 98% of the solids in the tower. In addition, a biological programme must be applied.

Applying the five steps of the PDCA process to the secondary water coolant system yields the following:

1) Understanding the ageing mechanism

The expected ageing mechanisms are scale formation, corrosion and growth of micro-organisms.

2) Definition of ageing management

Ageing management is defined to be a chemical treatment programme, subsequent analysis of the added chemicals, adjustments of water quality of the system for proper performance of the added chemicals, continuous blowdown and cooling tower cleaning.

3) Managing ageing mechanisms (Water treatment system operation)

Good water quality should be kept at all times within specified limits to minimize ageing effects. This is to be achieved through the addition of:

- Corrosion and scale inhibitors. The chemicals applied in semi-open systems are mostly based on low zinc programmes for environmentally friendliness. The discharge of zinc to the environment is restricted to 1–2 ppm (mg/l). Beside zinc usually phosphates, either organic or inorganic phosphates or mixtures of both types, are implemented;
- Oxidizing (e.g. chlorine) and non-oxidizing biocides (e.g. based on isothiazolinone mixtures) to control the growth of microorganisms;
- Sulphuric acid to set the pH value of the coolant water within specified limits because each water treatment programme requires a specific range of pH, alkalinity and hardness to be most effective;
- Blowdowns, practically carried out at about 5 m^3/hr to control the concentration of salts and suspended solids as well as help decrease pH, alkalinity and hardness to the required limits. Each chemical treatment programme requires a specific maximum of a concentration number, which is the total dissolved solids value of coolant water divided by the value of feed water. This number implies the concentration of salts in water due to evaporation;
- Mechanical filters, especially self-cleaning filters, to remove particles of size greater than 150 µm to maintain a clean tower basin. When suspended matter rises to unacceptable limits, the tower is emptied, cleaned and refilled with fresh water with the corresponding amounts of the required chemicals.

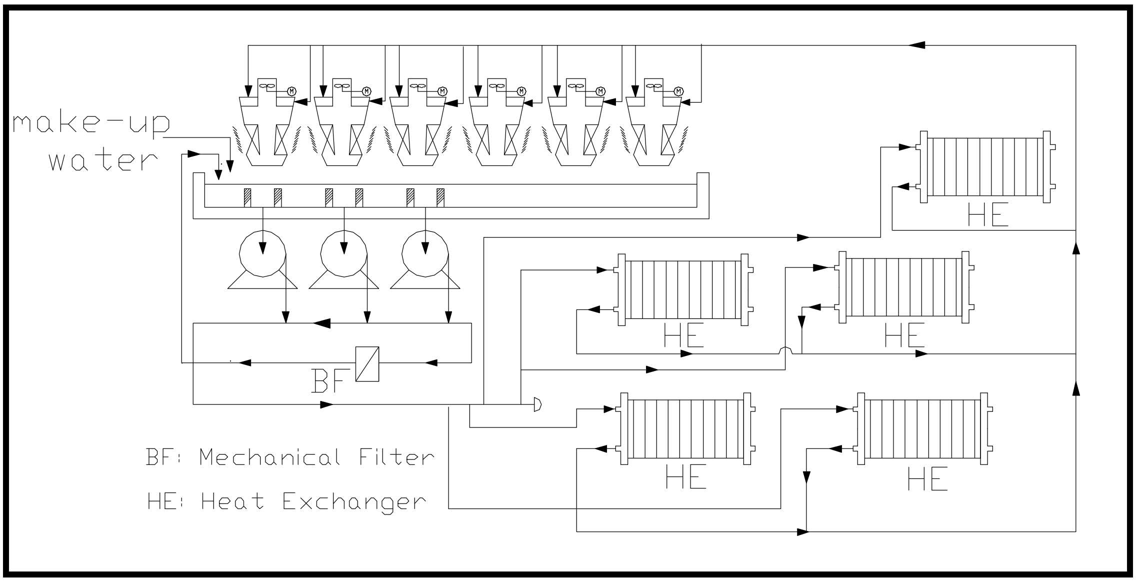

FIG. 2. Simplified flow diagram of the secondary cooling system of ETRR-2.

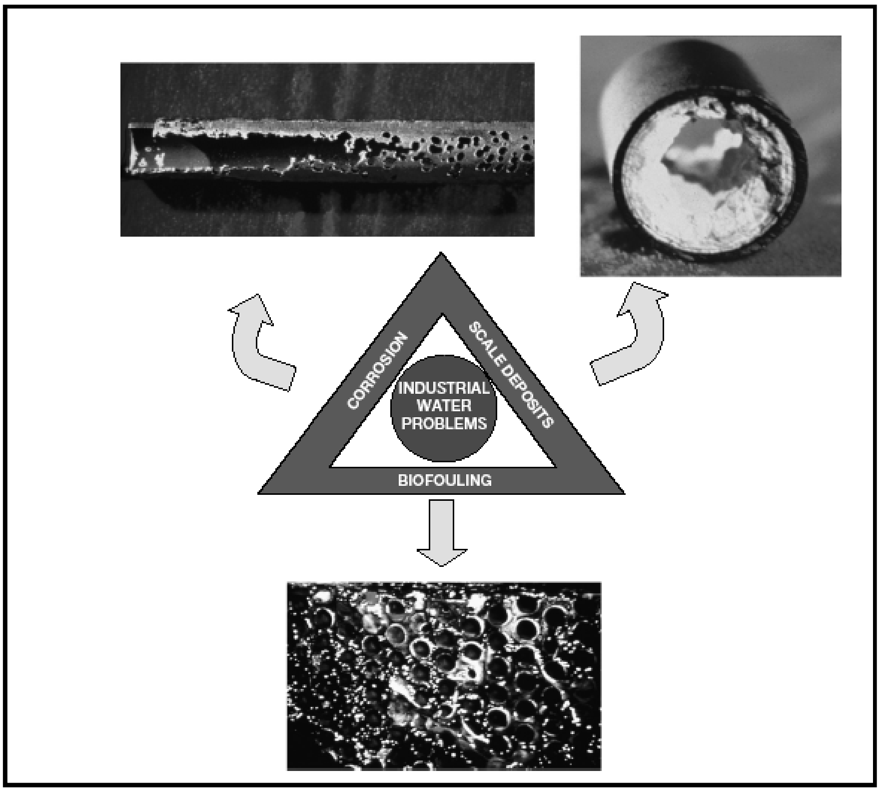

FIG. 3. The three major problems arising from uncontrolled process waters.

4) Inspection, monitoring and assessment

Inspection of pipeline and monitoring of water quality to determine water chemistry and piping condition include:

- Measuring the pH value, alkalinity total and calcium hardness to conclude if the water quality is satisfactory for the chemicals to work properly;
- Measuring chloride and sulphate concentrations, as upper limits exist for their concentrations in the cooling water. It is to be taken into consideration that chloride specifically promotes pitting of carbon and stainless steel;
- Measuring the electric conductivity of both the coolant and feed water to calculate the concentration number. The calcium hardness of both the coolant and feed water are measured. Their ratio, the calcium hardness number, when compared with the concentration number, indicates the presence or absence of scale formation;
- Measuring the residuals of the added treatment chemicals, usually zinc and phosphate, to conclude if the corrosion and scale inhibitors exist in the coolant water in sufficient concentrations;
- Measuring the iron content, as it indicates the extent of corrosion in the system;
- Inserting carbon steel coupons (Figures 4 and 5) in the corrosion test racks in the return lines to the cooling tower to measure the corrosion rate (Figure 6);
- Measuring thickness of pipes by ultrasonic testing.

For assessment the following parameters should be used as references:

- pH value between 7.90 and 8.40;
- Calcium hardness number≥concentration number≤3;
- Corrosion rate≤2 mils per year (1 mil=10^{-3} inch=0.0254 mm);

- Chloride content<100 ppm;
- Sulphate content<900 ppm;

5) Mitigation and repair

The following results are obtained from applying the ageing management programme:

- No damage or repair within the SSCs of the secondary cooling circuit;
- The program is dynamic and works effectively;
- Change in the programme according to the result of parameter monitoring and assessment;
- Consultation with a special company in water treatment is useful to identify, follow and use a suitable chemical treatment programme to get the maximum possible efficiency.

FIG. 4. Left) Coupons of different constructive materials; right) corrosion test rack.

FIG. 5. Left) Carbon steel coupon before insertion; right) after insertion for a one-month period in the corrosion test rack in the cooling tower.

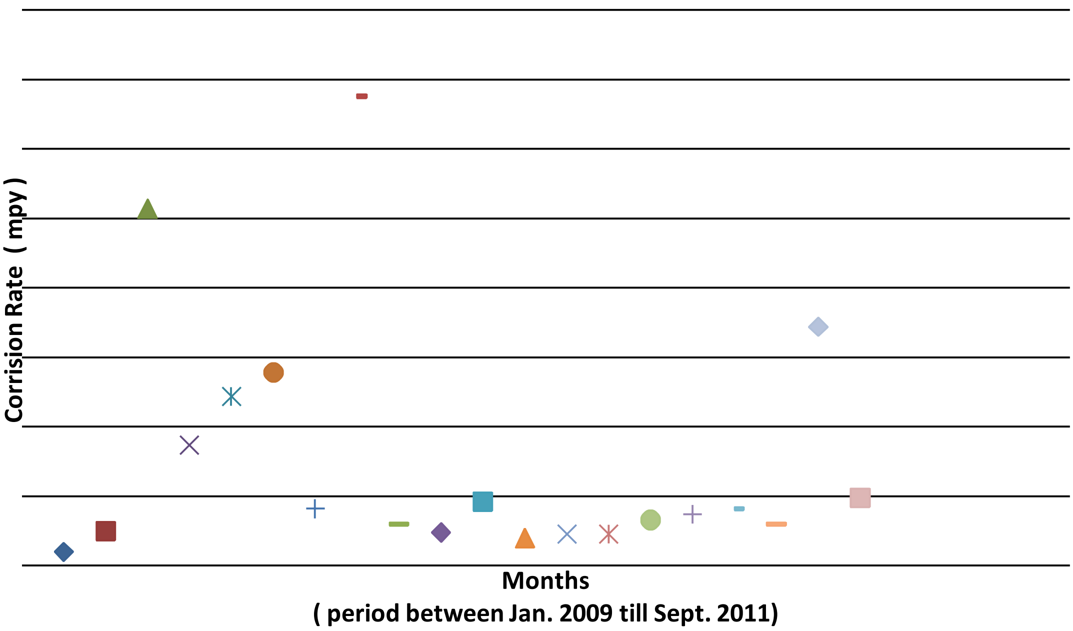

FIG. 6. Corrosion rate estimation of carbon steel in the secondary water cooling system.

5. CONCLUSION

The PDCA process cycle is used to effectively monitor and anticipate the status of SSCs and correct any abnormal condition at an early stage. It is a good practice that provides very useful information for the long term of both maintenance planning and ageing management.

AN OVERVIEW OF AGEING MANAGEMENT AND REFURBISHMENT OF RESEARCH REACTORS AT TROMBAY

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Abstract

Three nuclear research reactors have been in operation at Bhabha Atomic Research Centre, Mumbai, India. India has a rich experience of about 120 research reactor operating years including ageing management. A well structured programme is in force for plant life management, refurbishment and upgrading reactors in operation. Apsara, commissioned in August 1956, was the first research reactor. Apsara is a 1 MW_{th} swimming pool type of reactor with a movable core loaded with enriched uranium fuel and immersed in demineralized light water pool, which serves as coolant, moderator and reflector besides providing radiation shielding. Apsara was shut down during May 2009 for partial decommissioning and upgrading to a 2 MW reactor with several safety upgrades, e.g. a LEU based reactor core with higher neutron flux, a new reactor building meeting seismic qualification criteria and two independent shutdown devices. Cirus, a 40 MW_{th} tank type reactor utilizing heavy water as moderator, graphite as reflector, demineralized light water as primary coolant and natural uranium metal as fuel; has been in operation since 1960. After about three decade of operation, the availability factor started declining mainly due to outage of equipment exhibiting signs of ageing. After ageing studies and performance review, refurbishment requirements were identified. A programme for refurbishment was drawn that included safety upgrades like civil repairs to the emergency storage reservoir to meet seismic qualification criteria and a new iodine removal system for better efficiency. The reactor was shut down during 1997 for execution of this refurbishment programme. After completion of refurbishment, the reactor was brought back into operation during 2003. It has completed about seven years of safe operation after refurbishment with a significant increase in availability factor from 70% to about 90%. The reactor was permanently shut down during December 2010. The reactor core was unloaded, and preparatory work for decommissioning is at hand. Dhruva, a 100 MW_{th} tank type research reactor using heavy water as primary coolant, moderator and reflector has been in operation since 1985. The reactor is being utilized for production of radioisotopes and neutron beam research applications. Modernization, safety upgrades and replacement of some important equipment in Dhruva have been planned to achieve smooth operation with better utilization and enhanced safety. The paper highlights the experience gained in refurbishment, re-commissioning and full power operation of Cirus, the plan for upgrading Apsara and ageing management of Dhruva.

1. INTRODUCTION

In view of increasing global concern for the safety of nuclear reactors and the protection of the environment, plant life management assumes special importance in the quest for consistent improvement in safety and availability of research reactors. A well structured life management programme must address the issues arising out of various degradation mechanisms and their effects, upgradation and refurbishment requirements; and it should encompass thorough surveillance, operation within design parameters and operating limits and conditions with better margins, periodic performance review, ageing studies, regular inspection and testing, operating experience feedback, etc. A dynamic correction programme of predictive and preventive maintenance should complement. The reactor availability and its safe operating life can be significantly enhanced through implementation of timely refurbishment, replacement or upgradation actions based on assessments. The ageing management programme for research reactors in India is based upon operating experience of over 50 years, with safety having been accorded high importance and utmost priority. Out of the main three operating reactors, Cirus was refurbished, upgraded, operated with enhanced safety and availability and permanently shut down in December 2010; Apsara has been shut down for refurbishment and upgrading; and Dhruva continues in operation with an action plan in force for refurbishment and upgrades.

2. APSARA REACTOR

Apsara, the first Indian research reactor, attained its first criticality on 4 August 1956. It is a pool type reactor fuelled by plate type HEU fuel. The reactor core is suspended from a movable trolley in a pool of light demineralized water, which, in addition to shielding, acts as coolant, moderator and reflector. The core was mounted on a square grid of aluminium with

49 holes on a 7×7 square lattice (77 mm lattice pitch) containing standard fuel elements, control fuel elements, reflector elements, a neutron source, irradiation tubes, fission counters for reactor startup, etc. Figure 1 shows a typical Apsara operational core configuration. The reactor core can be positioned at three different pre-designated locations to facilitate a wide range of experiments at beam tubes, in the thermal column and in the shielding corner in addition to the in-core irradiation requirements. The three reactor core positions (A, B and C), beam tubes locations, thermal column and shielding corner are shown in Figure 2.

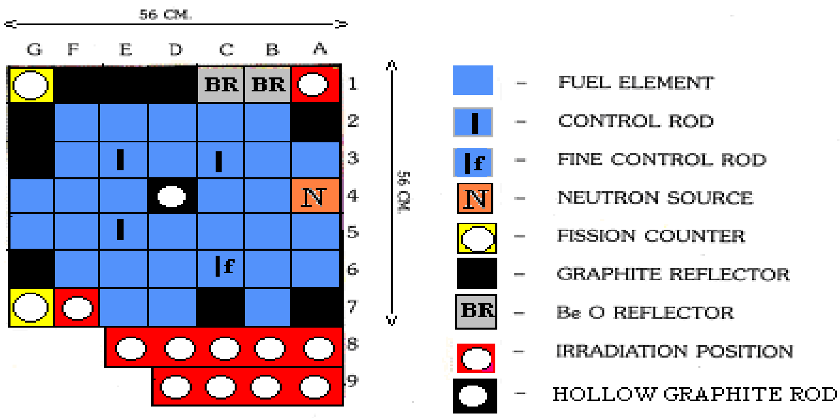

FIG. 1. Typical Apsara core configuration.

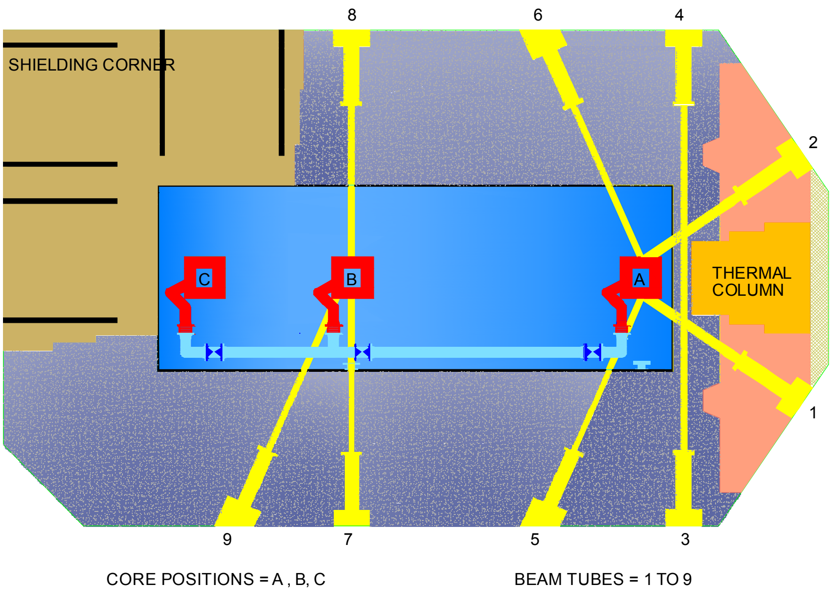

FIG. 2. Pool block structure.

During initial trial runs at the rated power level of 1 MW, the radiation field at the pool surface was observed to be high, about 140 mR/hr, mainly due to ^{16}N activity. Since flow reversal and introduction of delay tanks did not help, reactor power had to be restricted to a maximum of 400 kW. Shortly after commissioning the reactor, significant loss of inventory from the reactor pool was observed mainly due to seepage through the concrete structure. The attempts to arrest seepage by applying a dry mix of shotcrete, plaster, grout and epoxy paint resulted in limited success. Hence, the entire pool was lined with stainless steel.

The reactor was normally operated up to 400 kW in a cyclic mode of operation, i.e. 7 hours operation each day for 5 days per week. At the irradiation positions, the maximum unperturbed thermal neutron flux was about 3–4×10^{12} cm^{-2}s^{-1}. These facilities were used extensively to carry out research in a number of areas in the basic sciences, production of radioisotopes, neutron radiography, detector testing, shielding experiments, material characterization, etc. The Apsara reactor has contributed enormously towards training Indian engineers and scientists for experience and familiarity with the technology involved in the design, construction, operation and maintenance of nuclear reactors.

2.1. Ageing management

The instrumentation for reactor regulation, protection, control console and operator information systems was based upon vacuum tube technology. After about two decades of operation, in the early 1980s difficulties were faced in procurement of necessary spares, as these components were no longer manufactured (i.e. obsolescence) by the original suppliers and could not be developed locally. The replacement of old vacuum tube based systems with semiconductor technology based systems was the first major job undertaken towards ageing management of the reactor. Design, development, fabrication and testing of the new system had been done in-house as a part of modernization project for the research reactors Zerlina, Apsara and Cirus. Safety and reliability analyses were performed, and regulatory clearance was obtained. Installation and commissioning was performed during a planned shutdown during the late 1980s. The system had performed well for about three decades until the reactor was partially decommissioned for modernization and upgrading.

To augment safety against fire hazards, a new fire alarm system was procured, installed and commissioned during the late 1990s. The work was executed with support from local suppliers.

During the early 1950s, a safety manual similar to SAR covering safety analyses and assessment was prepared and issued. Based upon experience, a document named standing instructions similar to technical specifications was prepared and issued. During the early 1990s regulatory requirements necessitated revision of these documents. A SAR and technical specifications were prepared in accordance with current guidelines, reviewed and issued after approval by the competent authority.

2.2. Modernization and upgrading programme

The need for permanently shutting down the Cirus reactor by December 2010 and the ageing of the Apasra reactor over 53 years of operation necessitated upgrading the Apsara reactor to supplement the services provided by the Dhruva reactor in the context of catering to the growing demand for radioisotope production and neutron beam tube research in the country. The programme mainly included extensive refurbishment of the reactor to extend its useful life and enhancement of safety margins by strengthening the SSCs to meet current safety standards. The major design objectives of the upgraded reactor are:

- To enhance the neutron flux level, power and irradiation volume for better utilization of irradiation and beam tube facilities;
- Use of LEU fuel in line with current international norms;
- Adoption of a sub-optimal $H/^{235}U$ atom ratio in the design to maintain all reactivity coefficients negative under different operating conditions;
- Providing sufficient core excess reactivity to cater to various experimental, irradiation, operational and fuel burnup loads;
- Incorporation of two shutoff rods in addition to the two control shutoff rods such that two control shutoff rods or two shutoff rods or any combination of one shut off rod

and one control rod can shut down the reactor with at least a 10 δk sub-criticality margin.

Reactor power was increased to 2 MW to enhance the maximum thermal neutron flux to 6.1×10^{13} $cm^{-2}s^{-1}$ and fast neutron flux to 1.3×10^{13} $cm^{-2}s^{-1}$. The core of the upgraded Apsara reactor will be mounted on a grid plate having 64 lattice positions arranged in 8×8 square arrays with a lattice pitch of 79.7 mm (refer to Figure 3). The central 4×4 array supports 11 standard fuel assemblies, 2 shutoff rod fuel assemblies, 2 control fuel assemblies and 1 hollow beryllium oxide plug. The BeO reflector elements are provided to achieve adequate core excess reactivity and to maximize the irradiation volume for isotope production and material testing. The reflectors will house 7 irradiation positions, 2 fission counter assemblies and 1 fine control rod assembly. The reactor core is movable and can be operated at three different positions in the reactor pool. The reactor will be loaded with LEU silicide dispersion fuel and will use light water as coolant and moderator. The fuel has an enrichment of 17% ^{235}U, and the fuel meat has a loading density of 4.3 g/cm^3. The burnup of the discharged fuel assemblies should be about 25% of initial fissile material content. Two shutoff rods are provided to shut down the reactor. Reactor power is controlled by means of two control shutoff rods and one fine control rod. The control shutoff rods also function as shutdown devices. The control and shutoff rods are located inside the control fuel assemblies. The fine control rod is located in the BeO reflector region adjacent to the core. Each control and shutoff rod contains two fork type hafnium blades, and the fine control rod contains a single hafnium blade. Two control or two shutoff rods or any combination of one control and one shutoff rod are capable of shutting down the reactor with a subcritical margin of at least 10 δk.

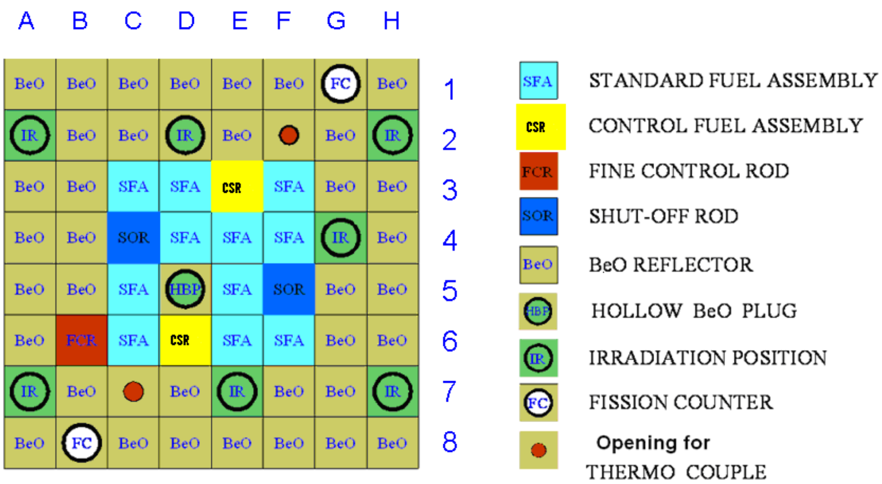

FIG. 3. *Core layout including reflector element.*

2.2.1. Ageing assessment of the reactor building

The reactor building is an approximately 29 m long, 16 m wide and 18.5 m high reinforced cement concrete framed structure. The footings and columns, below and above the plinth level, are constructed using M20 grade concrete with mild steel reinforcement. As per the current regulatory requirements, the reactor building was evaluated for seismic qualification as per the requirements of IAEA-TECDOC-1347 Consideration of External Events in the Design of Nuclear Facilities other than Nuclear Power Plants, with Emphasis on Earthquakes. Analysis using a finite element based software program indicated that it is necessary to improve the capacity of the footings, columns and beams of the building in order to meet the seismic qualification criteria. Non-destructive tests were carried out to assess the integrity and strength of the reactor building and the reactor pool block. An ultrasonic pulse velocity test, Schmidt rebound hammer test, carbonation test, half-cell potential test, chemical test, perfometer test and core sample test were carried out also. Based upon the results of tests and analyses, it was concluded that many of the structural members of the reactor building could be classified as doubtworthy with an internal structure of inhomogeneous concrete mass containing pores and possibly microcracks. Hence the decision was taken to build a new reactor pool block and new reactor building.

2.2.2. New reactor building and pool block structure

The new reactor building will be a reinforced cement concrete structure measuring 28.80 m in length, 20.90 m in width and 20 m in height above the plinth level. The pool block reinforced cement concrete structure will be 17 m long, 9.3 m wide and 9.45 m deep. To prevent seepage of pool water through the pool structure, it will be internally lined with 3 mm thick stainless steel plates on all sides and with 6 mm thick stainless steel plates on the floor. The design of the pool block and reactor building conforms to IAEA-TECDOC-1347.

2.2.3. Primary coolant system

The primary coolant system transfers the heat generated in the reactor core and the irradiated fuel assemblies stored in the reactor pool to the secondary coolant system through a heat exchanger. The secondary coolant rejects the heat to the atmosphere through a cooling tower. In order to maintain the radiation field at the pool surface below acceptable limits, primary coolant flow through the core is still directed from top to bottom (refer to Figure 4), the height of the water column above the core has been increased by one meter and a hot water layer $5^{o}C$ higher than the pool water temperature has been provided on the top of the pool water. The core outlet coolant passes through a delay tank housed in a shielded room to provide a 150 s delay so as to permit decay of ^{16}N activity sufficiently to minimize the radiation field in the process equipment room. The process piping from and to the reactor pool passing through the reactor building is provided with adequate shielding to ensure that the radiation field inside the reactor building is well within the permissible limits.

In the event of main coolant pump outage, natural convection cooling flow through the core is established by automatic opening of two natural circulation swing check valves installed on the core outlet plenum. Provision of flywheels in the primary coolant system pumps ensures adequate coast down flow to cool the core during the transient period between the pump trip and opening of natural circulation valves. In the event of a LOCA, the piping layout with a siphon break arrangement ensures adequate submergence of the core, thus providing core cooling and shielding in the shutdown state of the reactor. As an extended safety feature, emergency spray cooling of the core is provided in the unlikely event of a beam tube rupture, which may result in pool draining and partial core exposure.

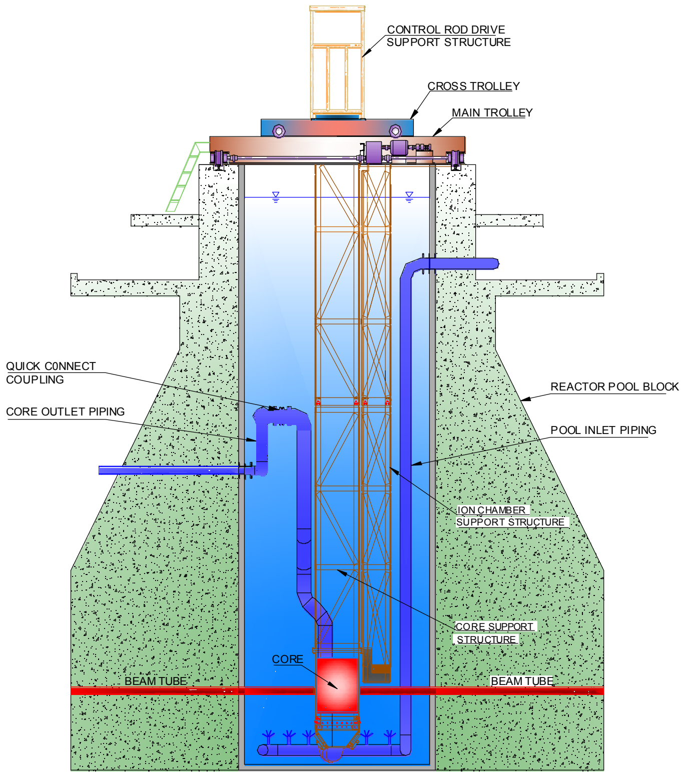

FIG. 4. Diagram of modernized Apsara reactor.

2.2.4. Control and instrumentation

For reliable neutronic power measurement and protection from the source range to power range, neutron detectors and associated electronics that work on diverse principles and have sufficient redundancy like triplicated instrumentation are used for regulation and the protection system. Reactor power is controlled automatically by vertical movement of hafnium absorber elements provided in two control shutoff rod assemblies and in one fine control rod. The existing reactor regulating system based on on–off controllers is being replaced by a proportional control system. The feedback signals now include both linear power and log rate signals, and the combined error signal is used to position the fine control rod. In order to improve positioning accuracy, a stepper motor is used for the drive mechanism of the fine control rod. The new system will make use of the standard microcomputer family of boards, of which a full scale prototype system has been set up for the purpose of development and testing of regulating system software. The software is being developed as per software quality assurance plan standard guidelines.

The reactor was shut down in June 2009 and partly decommissioned to facilitate up-gradation. Several safety upgrades have been incorporated based on experience and feedback. The design of SSCs has been completed, and review by regulator is in progress. Procurement and contract awards for major jobs have been initiated. Ageing management has been considered in the design stage, and it will be integral part of construction, fabrication, installation, testing, commissioning and operation.

3. REFURBISHMENT OF CIRUS

CIRUS is a 40 MW_{th} vertical tank type reactor using natural uranium fuel, heavy water moderator and light water coolant. The reactor vessel is made of aluminium and surrounded by two annular rings of graphite for reflection, a cast iron thermal shield and a heavy concrete biological shield. On the top and bottom of the reactor vessel, shielding by water cooled aluminium and steel is provided. Concrete biological shields are placed at the top. Fuel assemblies are located inside lattice tubes and cooled by forced circulation of demineralised light water in a closed loop with coolant flow from top to bottom. The heat from recirculating primary coolant water, heavy water moderator and thermal shield cooling water is transferred to seawater in tubular heat exchangers with seawater flowing in a once-through mode. The reactor performed well for more than three decades with an average availability factor of about 70%.

3.1. Ageing studies and refurbishment

After 30 years of safe operation, signs of ageing started surfacing, and the availability of the reactor started declining due to an increase in the frequency of equipment outages that necessitated considerable efforts and time for bringing the equipment back into service. During the early 1990s, detailed ageing studies on reactor SSCs were undertaken to examine in detail the technical viability of extending the life of the reactor by 10 to 15 years by appropriate corrective measures to compensate for ageing. For the comprehensive review, the SSCs were classified into four groups, namely in-core components, safety systems, out-of-core components and civil structures. After completion of detailed evaluation of the results of the study, refurbishing requirements of critical plant components were identified, and a comprehensive refurbishment plan was drawn. The refurbishment outage was seen as an opportunity for safety upgrades, and accordingly the action plan encompassed several safety upgrades to meet current safety standards. Regulatory consent was obtained for implementation of the plan. The procurement of necessary equipment was initiated well in time. The reactor was shut down during October 1997 to carry out major repairs and various activities as per the refurbishment plan. Detailed inspections that were not possible earlier due to the fuel present in the core were also performed, additional refurbishing requirements were identified and the refurbishment plan was accordingly modified.

3.1.1. Reactor vessel

The reactor vessel, made of 1S aluminium alloy, is a vertical cylindrical tank with 199 vertical lattice tubes. Between the shell and the top tube sheet an expansion bellow is provided to accommodate thermal expansion and contraction. Several ageing studies were carried out for the vessel.

Samples from the flow tube of an isotope irradiation assembly that is made of the same material as the vessel and had seen comparable neutron fluence were examined for fast neutron irradiation induced embrittlement and also for thermal neutron induced transmutation, which can cause precipitation hardening of aluminium. Test results showed a tendency of saturation in tensile properties for the fluence experienced by the material. It was concluded that the reactor vessel can reasonably sustain many more years of operation.

Visual inspection, eddy current tests for wall thickness monitoring and volumetric examination for flaw detection of all vessel tubes were carried out. The condition of the tubes was found to be good, and no unacceptable flaws were detected.

A finite element analysis was performed to assess the fatigue life of the reactor vessel expansion bellow, which experiences a fluctuating thermal load. The fluctuating stresses in the bellow were assessed to be well below the endurance limit of the material.

From these studies it was concluded that replacing the reactor vessel was unnecessary.

3.1.2. Graphite reflector

The graphite reflector around the reactor vessel is in two annular layers and is cooled by air flowing in between them. During full power operation, the inner layer undergoes concurrent annealing. Since the reactor had been operated at 20 MW for a long period, the reflector had accumulated Wigner energy at lower temperatures that was likely to be released during operation at powers higher than 20 MW. It could have posed a safety concern under anticipated operational transients like station blackout. Hence detailed studies were conducted to estimate the rate of release of energy with a rise in temperature and assess thermal behaviour of the graphite reflector following a reduction in cooling air flow.

Sample blocks were removed from the reflector, and the rate of release of Wigner energy per unit rise temperature was measured by differential scanning calorimetry. A small segment in the inner portion of the outer graphite layer showed negative specific heat (see Figure 5). In view of uncertainties in thermal conductivity and specific heat values of irradiated graphite, experiments were conducted to generate temperature data under steady state operation at 4–20 MW of power and anticipated operational transients, e.g. reactor trip with reduced cooling flows. The data was utilized for developing a computational model that was used for predicting steady state and transient temperatures following a reactor trip from 30 MW and 40 MW power levels with normal as well as reduced cooling flow. The observed temperatures closely matched with predicted temperatures. Thereafter, the reactor was shut down for refurbishment. After completion of refurbishment, reactor power was raised to the full 40 MW in small steps during 2004 after incorporating as an additional safety feature a reactor trip on a pre-set value of graphite temperature about 10°C more that than the estimated temperature. No sudden rise in graphite temperature was observed at any stage. After reactor operation at full power for a sufficiently long time, graphite samples were again drawn and subjected to differential scanning calorimetry, which indicated a disappearance of the negative specific heat region and an overall reduction in stored energy.

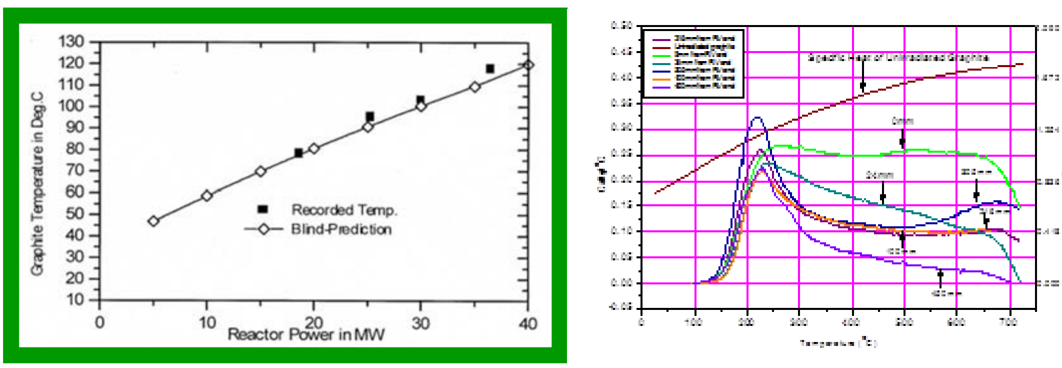

FIG. 5. Graph of the outer graphite layer by differential scanning calorimetry.

3.1.3. Piping

A large part of the primary coolant system piping, made of carbon steel, is laid at a depth of about 4 m below grade. The pipe sections are joined by Dresser couplings using elastomer sealing rings. Towards assessing the condition of the inside surface of the primary coolant loop piping, a 1 m long piece of the pipe from the hot leg of the coolant loop was removed and subjected to metallurgical examination. No evidence of any preferential attack appeared. The actual corrosion rate was estimated to be about 0.035 mpy, as compared to the design corrosion rate of 0.1 mpy.

Pressure testing revealed small leakage in one of the pipelines. The Pearson survey and acoustic emission techniques were tried for the purpose of locating the leak, but these were not successful. Due to the large soil cover over the piping, use of the isotope tracer technique or a pipe inspection gauge were also found infeasible. Sample pits were dug at earmarked locations, and fluorescent chemical dye was injected from one end in the suspected pipe. The pits were dewatered and checked for the presence of dye. This method was successful in identifying the leaky pipe section, and the pipes were exposed between the two pits by excavation. As expected, leakage was found from a coupling on one of the main coolant recirculation pipes. As weathering protection on the external surface of pipelines was found to have suffered ageing degradation, it was decided to expose the entire length of subsoil piping for reconditioning. The condition of the large diameter piping was found to be good, but small diameter pipes were seen to have surface pitting, especially near the welds. All small diameter pipes were replaced. Elastomer sealing rings on all couplings were replaced, and the pipes were pressure tested. Old protective coatings made of tar felt and bitumen were removed, the pipe surfaces were cleaned by high pressure water jets and a new protective coating made of a rubber modified bituminous compound with polyvinyl chloride backing was cold applied in two layers. Couplings were separately coated with epoxy compounds. Figure 6 shows a picture of the piping section after application of the new protective coating.

FIG. 6. Repaired subsoil primary coolant piping.

3.1.4. Helium system piping

There are eight flange joints having elastomer gaskets between the aluminium pipes that extend from the top of the reactor vessel and the stainless steel helium cover gas pipelines located above the upper steel thermal shield. The leakage from the helium system had gradually increased and had reached a maximum value just prior to the refurbishing outage. After detailed checks it was confirmed that the major leak was from the reactor vessel region. Furthermore, by extensive helium and tritium sniffing, it was established that the flange joints between the aluminium and stainless steel pipes were leaking. This was suspected to have been caused by deterioration of the elastomer gasket material due to irradiation and ageing. These flanged joints are located in a 200 mm vertical gap between the top steel thermal shield and the top concrete biological shield at a distance of about 4 m from the top of the reactor (refer to Figure 7).

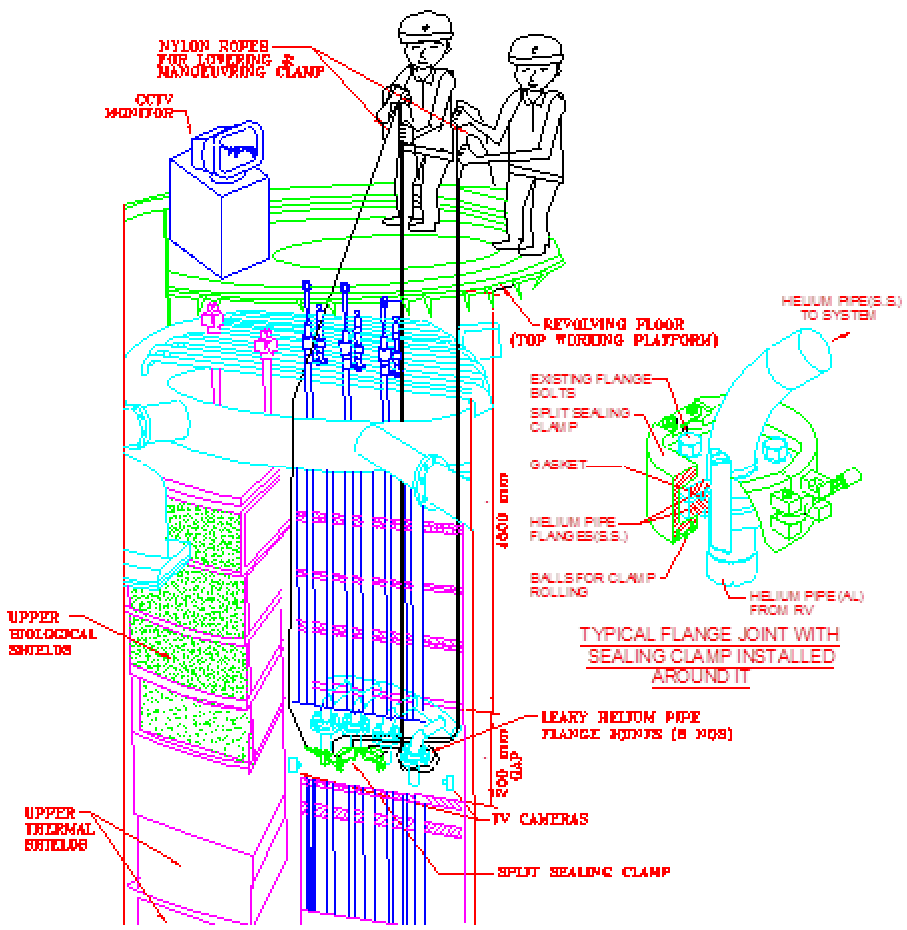

FIG. 7. Flanges and joints above the reactor vessel.

To rectify the above leaks by remote means without dismantling the massive reactor components, special split sealing clamps with tapered inner faces were developed. These were remotely installed around all eight flange joints and tightened to compress the old elastomer gaskets. A full scale mock-up station was established to qualify the tools, procedures and personnel and to ascertain the effectiveness of sealing by using similar old elastomer gaskets. This was a challenging task that was accomplished by a well coordinated technological effort by plant personnel. All the activities of lowering, maneuvering and enveloping the flanges with the clamps were done from the top of the reactor by using nylon ropes for maneuvering and by remote viewing through cameras installed near the flange joints.

Pressure testing of thermal shields above and below the reactor vessel indicated a minor leak in the top aluminium thermal shield. Remote visual inspection established that the leak was from the weld joint between one of the coolant inlet pipes and the top aluminium thermal shield. The location of the leak was confirmed by monitoring the stabilised water level in the vertical pipe under stagnant conditions after temporary sealing of the leak with an inflatable seal (refer to Figure 8).

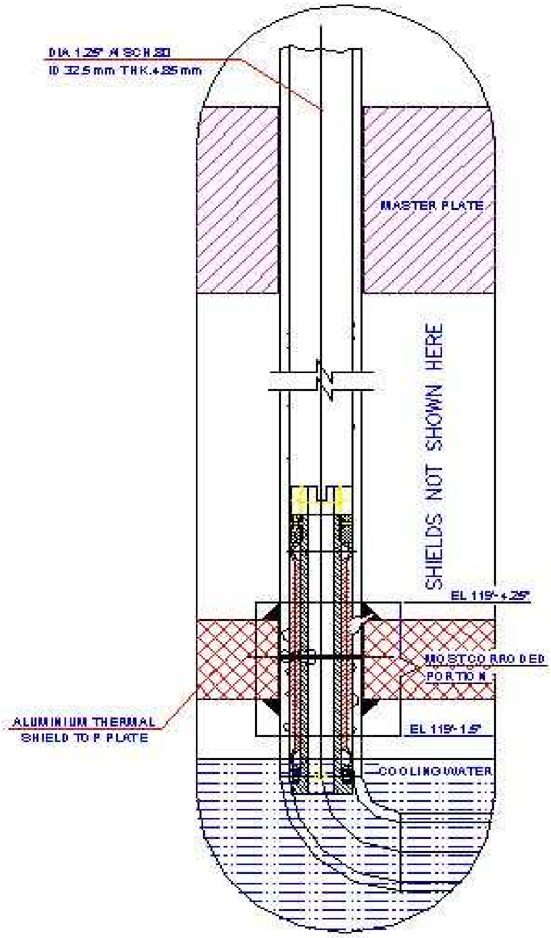

FIG. 8. Hollow plug installed in an aluminium thermal shield.

The aluminium thermal shield coolant inlet pipe was inspected with the help of a fiberscope and micro video camera to assess the condition of its inner surface. Eddy current testing was also carried out for volumetric examination. About a 65 mm length of the pipe around the leak location was found to be corroded. As the leak location was near the core region and 5 m below the top of the reactor, a hollow aluminium plug with expandable "C" shaped rings at both ends and a straight portion in the middle to cover the leaky section was developed and installed remotely after extensive mock-up trials. In order to overcome site constraints, a flexible link assembly was engineered for installing the hollow plug remotely. This was another challenging task that was carried out after extensive in-house developmental work and mock-ups.

3.1.5. Primary coolant pumps

The main coolant pumps are horizontal centrifugal pumps coupled to motors through gear elements used to increase speed. All pump assemblies are provided with fly wheels for improving the system flow coast down characteristics following a pump trip. The maintenance efforts on these pumps had increased primarily on account of frequent damage suffered by the gear elements. Vibration levels on the pumps had also increased over the years. Worn impellers and age related degradation of cork pads below floating foundations were identified to be the main reason. New gear elements with an improved design were procured and installed. The worn impellers were replaced. New cork pads were procured, and after satisfactory testing these were installed to improve the isolation efficiency to over 60% from the old value of less than 10%.

3.1.6. Primary coolant heat exchangers

Frequent tube leaks in primary coolant heat exchangers were one of the main reasons for the decline in the availability of the reactor. Visual examination and ultrasonic thickness measurements indicated the shells were in good condition. The tubes within the shell inlet nozzle area were seen to have suffered perforation attacks due to erosion and related corrosion. Metallurgical examination indicated a loss of nickel in the tubes caused by impinging water. Simulated studies indicated that the velocity of the liquid was higher than the threshold velocity for the stabilization of the oxide layer, and a higher amount of dissolved oxygen was responsible for progressive corrosion by erosion. Two new heat exchanger units and three new tube bundles were procured.

3.1.7. Overhead emergency water storage tank

A spherical emergency water storage tank is provided for gravity driven cooling of a shutdown core. The spherical tank bottom is in the shape of a cupola. Through the bottom cupola, a vertical concrete shaft extending over the water surface in the tank is provided for inspection. In the early 1970s, a small leak developed near the bottom of this central shaft. All efforts to rectify the leak while the tank remained full of water, e.g. pressure grouting, were futile. Emptying the tank warranted core unloading. Seismic evaluation of this tank showed that under its own weight, coupled with sloshing loads from the large mass of water stored in the tank, the area around the joint of the shaft for the bottom cupola would experience greater than permissible stress under the postulated ground acceleration. During the refurbishing outage, the tank was emptied, and repairs were carried out from inside the tank with a polymer modified epoxy mortar. For rectification of seepage from the central shaft of the primary coolant emergency reservoir, or ball tank, the tank was emptied soon after core unloading. The concrete wall in the leaky region was repaired by chipping old plaster and cementing with polymer modified mortar followed by pressure grouting at selected locations. Minor leaks still existed. Strengthening of the central shaft and cupola joints was achieved by additional reinforcement with mild steel plates and epoxy grouting to make it a monolithic structure capable of meeting safe shutdown earthquake qualification criteria. The arrangement is shown in Figure 9.

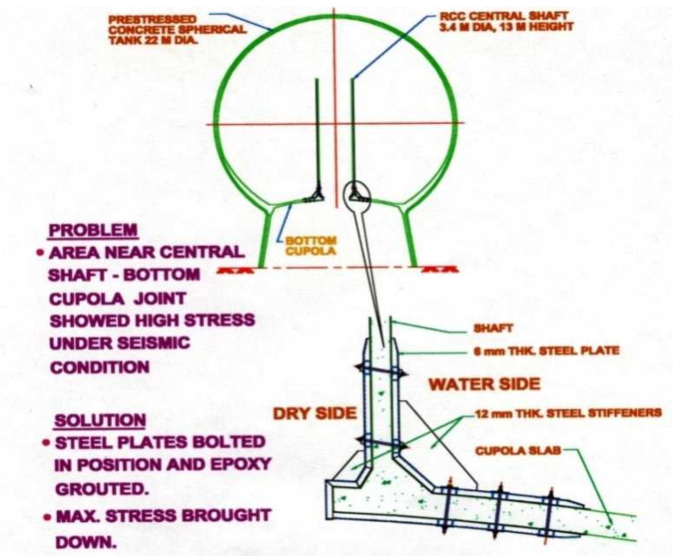

FIG. 9. Diagram of the reinforced ball tank.

3.2. Safety upgrades

The long refurbishing outage was also utilised to carry out certain safety upgrades. A new fire detection system and other fire safety measures, like fire barriers and fire retardant coating on cables, were incorporated.

Physical separation of some of the safety related components was carried out to guard against common cause failures due to fire, flooding, etc.

The emergency ventilation exhaust system of the reactor building was earlier provided with an alkali scrubber and silver coated copper mesh filters for trapping radioiodine under accident conditions. These were replaced by the more efficient and easy to maintain combination filters of activated charcoal and HEPA filters.

The old failed fuel detection system was based on the gaseous fission product stripping method and used to result in substantial radiation exposure of operations and maintenance personnel. A new system based on gamma radiation monitoring was designed and installed.

3.3. Other jobs

All radioactive liquid waste collection sumps were drained, inspected, repaired or reconditioned and tested to establish integrity. This job required considerable effort.

A complete overhaul of the air conditioning equipment of reactor building and control room was performed. Several ventilation air ducts and dampers, where ageing related degradations had been observed, were repaired or replaced. For refurbishment of reactor building ventilation and air conditioning equipment, a temporary filtered air supply system was installed.

Several other pipes and equipment like pumps, heat exchangers, travelling water screens, valves, cables, relays and instruments of the auxiliary systems such as seawater, service water, chilled water and machinery cooling water that had shown ageing related degradation were replaced. In addition, this opportunity was also utilized for a major overhaul of mechanical handling equipment including the 30 t capacity polar crane in the reactor building, fuel handling machine and emergency power supply diesel generators. The refurbishing activities included replacement of over 4 km of piping, servicing or replacement of about 350 valves of various sizes and overhauling of about 65 pieces of major mechanical equipment. In the electrical systems over 200 control relays, 100 electrical motors and about 3 km of power supply cables were replaced or serviced, in addition to overhauling major equipment like transformers, motor alternator sets and switch gear. Refurbishment of control instrumentation included replacement of over 4 km of instrument tubing, 5 km of thermocouple wires and 1 km of control cables. In addition, over 100 regulating valves and control valves, 200 thermocouples and resistance temperature detectors, and 300 assorted instrumentation items were either serviced or replaced.

A desalination unit of 30 t/day capacity based on the low temperature vacuum evaporation process was integrated with the primary coolant system of the reactor in order for a demonstration programme for utilization of waste heat from the nuclear reactor.

3.4. Resumption of operation at full power

After the completion of refurbishment, reactor systems were recommissioned, safety related tests were performed and the reactor was made critical after necessary clearance from safety authorities during October 2002. A reactivity anomaly of about 12 δk was observed. The core reactivity anomaly after detailed investigations was attributed to the wetting of the graphite reflector caused by water spillage from the primary coolant system in the upper service space due to failure of a trunnion valve during fuelling operations. Minor seepage was observed from a few locations on the pour joints of the spherical concrete surface of ball tank. The reactor was kept shutdown for repairs to the seepage from this tank. The ball tank was emptied again, and necessary repairs were carried out by lining the entire internal surface exposed to water with eight layers of epoxy coating impregnated with two layers of fiberglass cloth. The ball tank was commissioned and ensured to be free of leak or seepage. After review of various options, operation of the reactor at high power was found to be the better option for removal of moisture from graphite. The reactor was restarted during October 2003. The reactivity anomaly was reduced to 6 δk.

At the first step of reactor operation at a power of 4 MW, it was observed that the thermal power of the reactor was about 12 MW. This mismatch between neutronic and thermal power was attributed to attenuation of the neutron flux by moisture present in the graphite reflector. The mismatch was offset by an appropriate combination of repositioning of the reactor regulating system ion chambers and adjustment of amplifier gain after careful review and a safety assessment. The reactivity anomaly gradually decreased with progressive operation of the reactor at power. After significant improvement in the neutron flux at the location of the ion chambers and reduction of the reactivity anomaly to about 2 δk, reactor power was raised in small steps to 30 MW. At every step a thorough review of neutronic, thermal and radiological parameters apart from equilibrium moderator height and graphite reflector temperature was executed.

The reactor had been in operation with enhanced safety and an availability factor of about 90% until 31 December 2010, when it was permanently shut down to comply with international commitments. The core was then completely unloaded such that the fuel assemblies, experimental and irradiation assemblies, adjuster assemblies, primary shutdown devices, etc., have been removed. A few dummy assemblies were installed for periodic circulation of primary coolant. Heavy water was transferred to a storage tank disconnected from the system. The system has been dried to recover residual heavy water and is kept under a helium cover. Technical specifications for normal operation still remain in force. Technical specifications for permanent shutdown of the reactor are under review by safety authorities. After reviewing various options for decommissioning, delayed decommissioning was considered the best choice. Accordingly, preparatory jobs are at hand.

4. DHRUVA

Dhruva is a 100 MW_{th} reactor with heavy water as moderator, coolant and reflector. The reactor vessel is made of stainless steel with replaceable guide tubes of made of zircaloy. The fuel assembly consists of seven natural uranium metallic pins clad in aluminium. The secondary coolant is demineralized light water in a closed loop with seawater as the ultimate heat sink. A systematic life management plan has been instituted for enhancing the life and safety of the reactor. The plan is based upon the review of surveillance data, performance of systems and components, ISI results, experience feedback and periodic safety review. Major refurbishment jobs have been planned and executed with an emphasis on minimizing reactor down time without compromising safety.

4.1. Refurbishment of control and instrumentation systems

The control and instrumentation systems of Dhruva were designed in the late 1970s with limited diagnostic features. Obsolescence led to difficulties in the availability of spares and maintenance. Timely action has been initiated to refurbish the major systems with the latest computer based systems.

4.1.1. Coolant channel low flow trip and reactor trip logic system

The old coolant channel low flow trip system was based upon processing of contacts actuated by differential pressure gauges. The trip logic system processed a total of 72 trip contacts consisting of absolute and conditional trips to generate 72 outputs to the alarm annunciation system. It also generated outputs that were multiplied by relays to actuate the shutoff rods and control and dump valves of the moderator system that act as a slow acting secondary shutdown device. The system was based on a 2/3 global coincidence voting logic.

Use of 2/3 global coincidence in the existing reactor trip logic system had at times resulted in unwarranted trips upon activation of two different channels of two unrelated parameters. In addition, the trip logic system uses two optocouplers, one for generating alarm outputs to the alarm annunciation system and one for generating trips for each of the 72 absolute and conditional parameters. The difference in the current transfer ratio characteristics of the two optocouplers together with the difference in response time of the alarm annunciation system (10 ms) and response time of the trip logic system (20 ms) on several occasions has resulted in a situation when the reactor did not trip on annunciation of two trip parameters during transients, thus creating confusion in the mind of the operator. To overcome this problem, a new power PC based reactor trip logic system is being designed in which the trip and alarm generation will be combined, and instead of 2/3 global coincidence logic, 2/3 grouped local coincidence logic will be adopted.

4.1.2. Upgrades to the startup logic system

The existing startup logic system is a hardwired complementary metal oxide semiconduction integrated circuits and relay based logic system to generate outputs for raising and lowering shutoff rods in sequence and also associated trips and alarms upon double device failure, triple device failure, delays in rod start or travel, etc. The existing system as a single channel system leads to unnecessary reactor trips for a single component failure out of a large number of relay contacts.

To overcome this problem, a new triplicated startup logic system that will also directly process limit switch contacts signals has been proposed, thereby eliminating a number of relays as compared to the earlier system.

4.1.3. Upgrades to the alarm annunciation system

The existing alarm annunciation system is comprised of three Intel 8085 based systems. The system has a conventional Windows based display as well as a PC based event sequencing display in the control room.

Since events are generated by three systems, the time-stamping of the events done by the event sequence PCs results in errors in sequence analysis during an event marked by a large number of alarms. Due to reasons of obsolescence and limitations in expanding its capacity, a new dual redundant computer system has been proposed. The new system is a dual redundant computer based system capable of processing 408 parameters with time stamping, thereby eliminating error during sequence analysis of an event consisting of a large number of alarms. The new system also has a dynamic alarm facility that helps to expand the system.

4.2. Replacement of process water and seawater heat exchangers

The heat exchangers have been in service for about 25 years. Periodic inspection revealed that the tubes had gradually suffered erosion in the seawater entry region despite installation of polyvinyl chloride inserts at the mouth. Eddy current testing also indicated substantial wall thinning at the tube sheet region as given below in Table 1.

TABLE 1: RESULTS OF TUBE SHEET INSPECTION

Period of ISI	Number. of tubes tested	Category I tubes >40%	Category II tubes (20-–40%)
PSI 1981	4100	Nil	5
ISI 1993–96	3480	14	1304
ISI 2001–06	2700	128	1608

The replacement of the heat exchangers in a phased manner was initiated. The replacement of the first heat exchanger is in progress.

4.3. Power supply systems

Dhruva electrical systems have been in operation for about 25 years with satisfactory overall performance. However, based on operation and maintenance feedback and upgradation requirements of some systems for improved safety and reliability, several refurbishment jobs were identified. These activities were divided into two groups. The first group consists of jobs that could take place during normal monthly reactor shutdowns, and the items of the second group require longer reactor shutdowns. The refurbishment activities of each group are listed below.

4.3.1. Installation of automatic ground fault detection system

A new system to detect ground faults in the 48 V DC class I system has been designed. The system is programmable logic controller based and works on the principle of sequential scanning of all circuits by isolation from the main class I bus while receiving power from an alternate source. This upgrade will aid in fast detection and rectification of ground faults, leading to enhanced availability of the plant.

4.3.2. Retrofitting of 22 kV and 415 V circuit breakers

Generally, circuit breaker life is indicated in the number of electro mechanical operations under standard conditions of operation. However, due to environmental variations from site to site and other factors, such as the quality and frequency of preventive and corrective maintenance, the actual life of a circuit breaker could vary from case to case.

The panels of 22 kV minimum oil circuit breakers were obsolete, and no spares were available in the market, as the manufacturer had closed his factory many years before. Though the problems associated with switch gear were not so frequent in Dhruva, the onset of ageing related problems like wear and tear of mechanical parts and drift in precision trips and latch-in mechanisms, had to be considered after 25 years of operation. A lack of products for spare part management support was anticipated, and prudently we replaced them before the problem took an acute form.

4.3.3. Refurbishment of transformers

All transformers of Dhruva have been in operation for more than 25 years and their consumables started to show signs of degradation like cracks and minor leakage. Considering these signs, it was decided to refurbish all transformers one by one and replace all the consumable items including sealing gaskets, valves, radiators, marshalling box wiring, oil temperature and level gauges, and transformer oil. Core inspection as well as cleaning and drying to enhance transformer life were also undertaken.

4.3.4. Replacement of 3.3 kV cables and switchgears

All 3.3 kV motors at Dhruva were powered through aluminium conductor flame retardant polyvinyl chloride cables. These cables terminated in copper lugs at the motor as well as circuit breaker ends. Due to thermal cycling during startup and shutdown of motors and differential expansion rates of copper and aluminium metal at the lug joint, coupled with vibration at the motor end, the cable in the lug would slightly loosen over a period of time, resulting in premature failure at the terminals due to overheating aided by high current during motor starting. The frequency of failure was nearly one failure every quarter out of 27 terminals of nine motors in spite of quarterly terminal tightness. The cables were replaced with copper conductor cross linked polyethylene insulated cables, and now the frequency of failure at terminals due to loose connections has been reduced to almost nil.

4.3.5. Replacement of class II motor alternator sets

150 kVA motor alternator sets showed signs of ageing in frequent spurious trips and failure of the control cards due to ageing, hence their refurbishment was required. Considering the additional load arising from modifications in the ECCS, it was necessary to augment the rating of the units to 250 kVA. An additional motor alternator set was also installed as a standby to ensure higher reliability of the class II supply.

4.3.6. Replacement of class I battery banks

In view of upgrades to the capacity of the motor alternator sets from 150 kVA to 250 kVA, the capacity of the battery banks was also enhanced from 240 V, 1000 Ah to 1200 Ah. Since new safety standards require class I battery banks to be mounted on seismic stands, this requirement was also fulfilled during this replacement. Automatic control voltage regulators using thyristor controls that had also seen a service life of 25 years were replaced with new sets using insulated gate bipolar transistor based control.

4.4. Upgradation of other systems/components

In addition to the above systems, other systems such as the ECCS instrumentation logic and fuelling machine control and instrumentation are also being converted to electronic instrumentation. All the pneumatic instrumentation used for control room indication and recording are being replaced by electronic recorders and PC screens, with operator consoles for each system.

5. CONCLUSION

Introduction and timely implementation of an ageing management programme has contributed significantly towards more efficient and safer operation of the research reactors at Trombay. The highly cost effective refurbishment of Cirus was completed with less than 5% of the funds that would be required for building a replacement reactor of similar design. These valuable experiences will be useful for incorporating ageing management plans at the design stage of new projects.

AGEING MANAGEMENT PROGRAMME: AN EXPERIENCE OF IN-SERVICE INSPECTION OF THE KARTINI HEAT EXCHANGER

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Abstract

This paper discusses an experience on ISI of the Kartini reactor heat exchanger, as part of the implementation of an ageing management programme. Kartini reactor is located in Yogyakarta, Indonesia. The heat exchanger was constructed for 250 kW capacity. The type of heat exchanger is shell with tube recirculation. Tube material is stainless steel 304, and the shell and baffle plate materials are carbon steel. The heat exchanger has 72 tubes, its outer and inner diameters respectively are 19 and 16 mm, and tube thickness is 1.5 mm. The aim of ISI was to obtain and evaluate the heat exchanger's condition including the detection of any possible local tube thinning, pitting corrosion or gradual thinning, and determine whether any degradation or deterioration of the heat exchanger could have a significant impact to safety. The heat exchanger was inspected utilizing eddy current equipment in 2003 and 2006. Inspection results in 2003 determined that 12 heat exchanger tubes have a thinning degradation level ranging from 10% up to 60% of the outer diameter due to pitting corrosion. Deterioration of baffle plates has been linked to general corrosion attack. Inspection results in 2006 showed a consistent thinning degradation level with the previous inspection in 2003. So far heat exchanger performance is still satisfactory, as defined by the transfer of primary heat for a 2°C difference between inlet and outlet as required.

1. INTRODUCTION

The National Nuclear Energy Agency (BATAN) is a nuclear operating agency in Indonesia. BATAN operates three research reactors. All of the reactors are open pool type, and one is the TRIGA Mark II Kartini reactor in Yogyakarta. The reactor has been designed for 250 kW_{th}, but the operation license is only for a thermal power rating of 100 kW. The reactor went critical in 1979, and the first license received approved operation for 26 years. The TRIGA Mark II Kartini reactor liner material is 1050 aluminium (99.5% pure Al). Tank dimensions are 2 m in diameter, 6 m deep, and 6 mm in thickness. The pool liner construction consists of four shells that had been assembled in the reactor pool. In order to establish adequate mechanical strength and achieve a watertight construction, at each section 300 mm wide belts of 5 mm thick aluminium were fillet welded over the joints [1]. Reactor utilization includes training, research and education of students from the university in the nuclear field. The TRIGA Mark II Kartini reactor is equipped with two types of heat exchangers, one a shell and tube recirculation type and the other a plate type. An ageing management programme has been implemented through ISI to SSCs, such as the tank liner, heat exchanger and core support structures.

The paper discusses an experience on the ISI of the Kartini heat exchanger as an implementation of ageing management programme.

1.1. Kartini heat exchanger

The TRIGA Mark II Kartini reactor heat exchanger was constructed for 250 kW capacity. The type of heat exchanger is shell and tube recirculation. Tube material is stainless steel 304, and the shell and baffle plate material is carbon steel. The heat exchanger has 72 tubes or 144 tube holes, the outer and inner diameters are 19 and 16 mm, and tube thickness is 1.5 mm. The primary reactor coolant passes through the tubes, and secondary coolant on the outside of the tubes decreases the temperature of primary reactor coolant. The temperature difference (ΔT) between the inlet and outlet water of the heat exchanger is maintained above 2°C [2]. The schema and cross section of Kartini heat exchanger are showed in Figures 1 and 2 respectively.

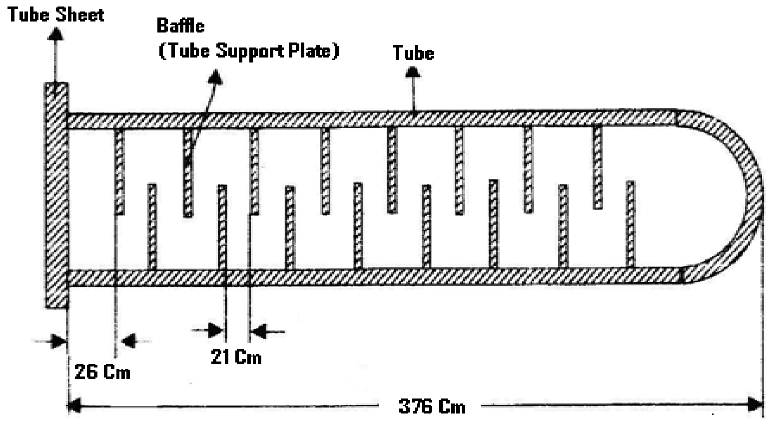

FIG. 1. Schema of Kartini heat exchanger.

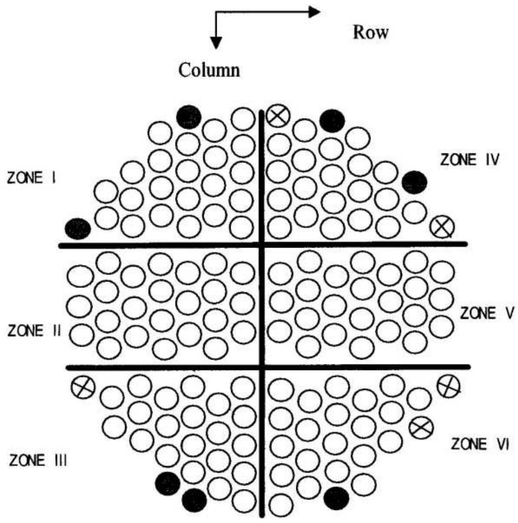

FIG. 2. Cross section of Kartini heat exchanger.

1.2. ISI history

The ageing management programme of the TRIGA Mark II Kartini reactor heat exchanger has been implemented through water chemistry management and conduct of ISI of the heat exchanger. The aim of ISI is to obtain and evaluate the heat exchanger condition, including detecting any possible local tube thinning, pitting corrosion or gradual thinning, and determining whether any degradation or deterioration of the heat exchanger could have a significant impact on safety. ISI of the heat exchanger was first conducted in 2003, after 24 years in operation. The scope of inspection covered the whole heat exchanger tube structure, including both the straight and U-bend tubes and utilizing eddy current equipment with an internal differential probe at a frequency of 88.000 Hz [3].

The second inspection was conducted in 2006, and utilized new eddy current equipment obtained from the IAEA under the Technical Cooperation project INS/9/022. The scope of inspection and condition were the same as in the previous inspection in 2003 [4]. Some photos taken during the described inspection activities are shown in Figures 3–8.

FIG. 3. Handling of heat exchanger [5].

FIG. 4. Shell and tubes bundle of heat exchanger [5].

FIG. 5. Heat exchanger tubes inspection conducted in 2003 [5].

FIG. 6. Deterioration of baffle plates [5].

FIG. 7. Heat exchanger tubes inspection conducted in 2006 [6].

FIG. 8. Discussion on eddy current signal analysis [6].

2. RESULTS AND DISCUSSION

Inspection of the heat exchanger tubes was performed during the major shutdowns in 2003 and 2006 and utilized eddy current equipment. The first inspection in 2003 showed that 12 heat exchanger tubes (24 tube holes) had thinning degradation levels ranging from 10–60% of the outside diameter (secondary side) or in category I to III, due to pitting corrosion. No gradual thinning degradation was seen. The 60 heat exchanger tubes were still in good condition. Deterioration of baffle plates was due to general corrosion attack. These conditions seem natural due to ageing consequences and have no impact on safety [3].

In 2004, a remedial action that replaced the heat exchanger baffle plates with new carbon steel materials was conducted successfully.

Inspection results in 2006 were consistent in thinning degradation levels with the previous inspection in 2003. No gradual thinning degradation due to corrosion was seen. The heat exchanger baffle plates are still in good condition [4].

3. CONCLUSION

In conclusion, although thinning degradation of 12 heat exchanger tubes occurred, ranging from 10–60% of the outer diameter, the 60 heat exchanger tubes (120 tube holes) were still in good condition and safe for continued operation. Plugging of tube degradation was not needed, since plugging criteria for defective research reactor heat exchanger tubes is still uncertain. For example, one primary criterion for action varies from 40–90% degradation in wall thickness. So far, heat exchanger performance is still satisfactory, as defined by its ability to transfer primary heat for a temperature difference of at least 2°C between inlet and outlet as required.

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REFURBISHMENT OF THE PRIMARY COOLING SYSTEM OF THE PUSPATI TRIGA REACTOR

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Abstract

The refurbishment of the 27 year old primary cooling system of the 1 MW PUSPATI TRIGA reactor was completed in April 2010 over an eight month outage. The project was implemented with the dual objective of meeting current user needs as well as a future reactor core power upgrade. Hence the cooling system was partly modernized to cater for a 3 MW_{th} reactor by installing higher capacity heat exchangers and pumps while maintaining the piping and valve sizes. The old 1 MW tube and shell heat exchanger, which had lost 25% of its heat exchange capacity, was replaced with two 1.5 MW plate type heat exchangers. Several manually operated valves were replaced with motorized units to allow remote operation from the control room. The installed cooling system was flushed with distilled water and then subjected to hydrostatic pressure tests. In the cold run test, the system was operated for an hour for every pump and heat exchanger combination while all operating parameters were checked. In the hot run test, the same was done at four levels of increasing reactor power, and dose measurements were also recorded. The paper gives the design, installation, testing and commissioning details of the project.

1. PROJECT BACKGROUND

The PUSPATI TRIGA reactor (RTP) is a 1 MW_{th} research reactor that has been safely operated since June 1982. During 29 years of operation, a few of the RTP systems have been replaced or modified to ensure the safety of the reactor. The RTP primary cooling system is a primary loop that consists of a heat exchanger, centrifugal pump and aluminium piping for transferring heat away from the reactor core. After going through a long operation time and various process of ageing, refurbishment was needed after a finding that the previous shell and tube heat exchanger had lost 25% of its heat exchange capacity. The refurbishment was completed in April 2010 over an eight month outage. The project was implemented with the dual objectives of meeting current user needs as well as a future reactor core power upgrade.

The safety objectives of the project were to enhance the efficiency of RTP primary cooling. In addition, it was also done to optimize natural circulation for sufficient heat removal during operation time as well as residual heat removal after reactor shutdown. The cooling system was also partly modernized to cater for a 3 MW_{th} reactor by installing higher capacity heat exchangers and pumps while maintaining the piping and valve sizes. With the new supervisory control and data acquisition (SCADA) control system the operation of RTP can be more simplified, and now the operator can automatically control the pumps and valves remotely from the control room.

2. PROJECT SCOPE

The existing primary cooling system consists of one shell and tube heat exchanger, three centrifugal pump units, a ^{16}N diffuser and associated aluminium piping and valves together with demineralizer system. In this project the equipment to be replaced or newly installed are as follows:

- Replacement of one shell and tube heat exchanger with capacity of 1 MW with two plate type heat exchangers with a capacity of 1.5 MW for each unit;
- Replacement of the existing three centrifugal pump units with three new units with higher capacity;
- New SCADA control system to control and monitor all primary water parameters such as pH, conductivity, flow and status of all primary cooling system equipment.

The heat exchangers and centrifugal pumps were installed in redundancy, such that only one will operate at a time while the other will be in standby. Each pump and heat exchanger will operate alternately for long lasting use. A different pump is used each day, and they are operated automatically or manually via a SCADA control system in the control room. Each pump can deliver up to 80 m^3/h of water through the heat exchanger to the reactor tank. All parts of the heat exchanger and pump that are in contact with primary water are made of type 316 and type 304 stainless steel. The affected components are shown in Figure 1, while the schematic of the new system is shown in Figure 2. Table 1 below shows the performance of the plate type heat exchanger.

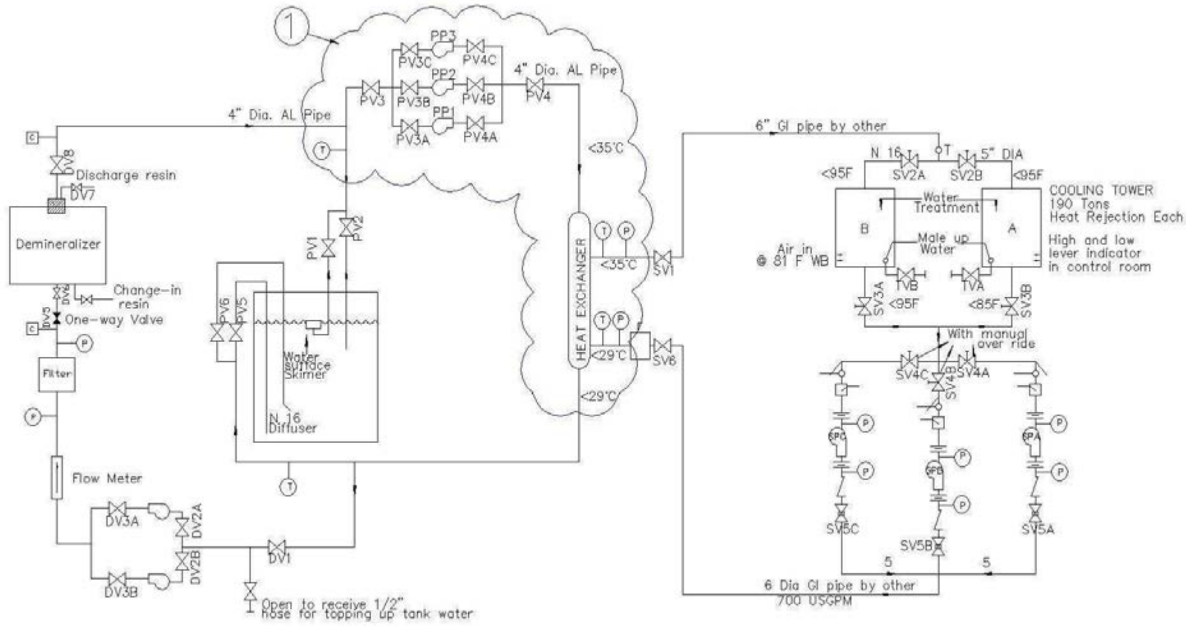

FIG. 1. Reactor cooling and purification schematic.

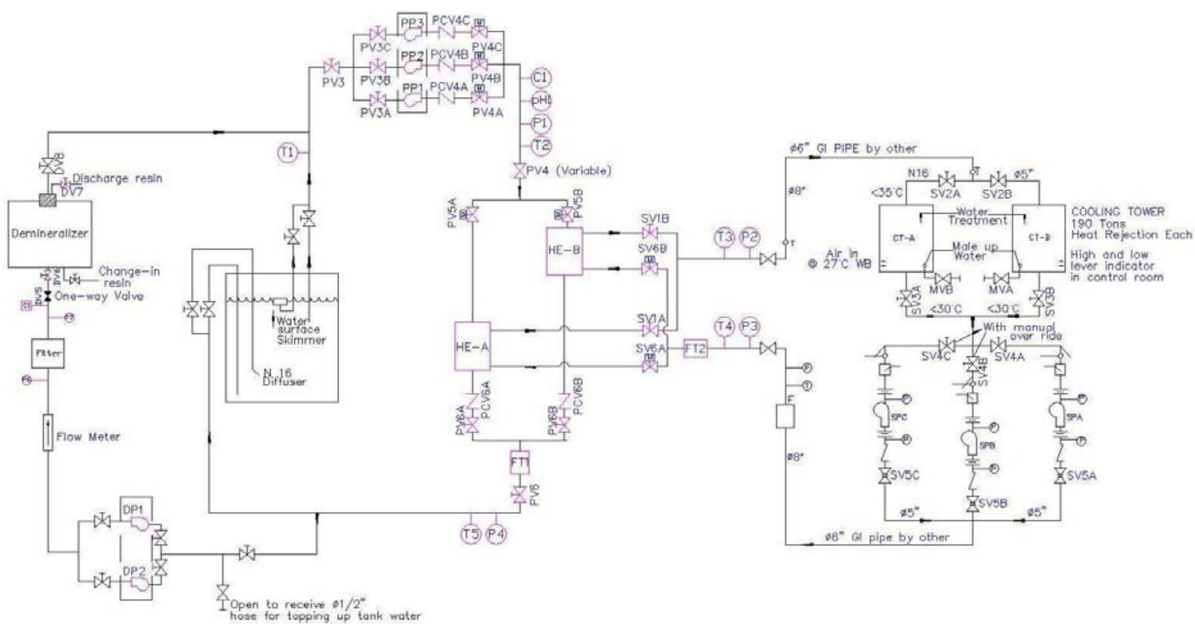

FIG. 2. New RTP primary cooling and purification schematic.

TABLE 1. HEAT EXCHANGER PERFORMANCE

Parameter	Hot side primary water	Cold side secondary water
Flow rate, m^3/h (gpm)	80 (350)	160 (700)
Temperature in, °C (°F)	43 (110)	27 (80)
Temperature out, °C (°F)	32 (90)	32 (90)
Design pressure, bar (psi)	6 (87)	6 (87)
Test pressure, bar (psi)	9 (130)	9 (130)
Design temperature, °C (°F)	100 (212)	100 12)

3. CONSTRUCTION AND INSTALLATION STAGE

Implementation of the project started with the construction stage. The scope of work performed by the contractor was to dismantle and remove the existing system, such as the shell and tube heat exchanger including its support structure and centrifugal pumps including their electrical wiring and switch board. Figure 3 shows the dismantling activities during the construction stage.

FIG. 3. Dismantling shell and tube heat exchanger.

The new plate type heat exchanger and centrifugal pumps will occupy the former space for the dismantled shell and tube heat exchanger.

FIG. 4. New plate heat exchanger and centrifugal pumps.

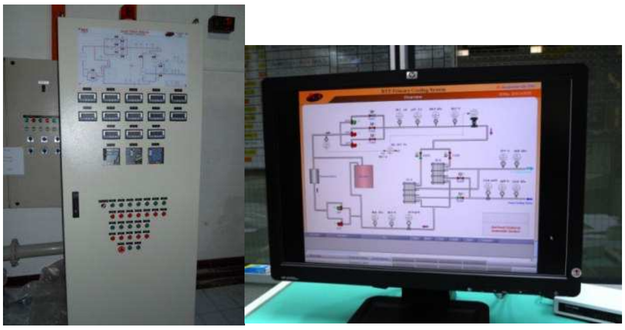

FIG. 5. SCADA control system.

Figure 4 shows the newly installed dual units of the plate heat exchanger with the capacity of 1.5 MW_{th} per unit and the three centrifugal pumps. Several manual valves from the old system that isolate each pump for operation were replaced with motorized valves to allow remote operation from the control room. The SCADA system layout shown in Figure 4 enables the reactor operator to manually or automatically, via programming, select the valves before operating the reactor. Each pump and heat exchanger will be operated alternately to ensure minimum maintenance outage and long lasting use.

4. TESTING AND COMMISSIONING STAGE

A testing and commissioning programme was developed to ensure that all the equipment functions are in accordance with the design intents and requirements as well as in compliance with objectives to ensure the long term safety of RTP. The programme for testing and commissioning is as follows.

4.1. Installation verification

All installation for each equipment is verified, and the verification activities are shown in Table 3 below.

TABLE 3: INSTALLATION VERIFICATION

Plate heat exchanger	a)	The plate heat exchanger is not operated with impermissible flow media, pressures and temperatures.
	b)	All pipe connections are firmly connected to the plate heat exchanger.
	c)	All required components of the plate heat exchanger are completely installed.
	d)	Maintenance access to the heat exchanger and to remove plate are acceptable.
	e)	Pressure surges are avoided. The normal operation of the plate heat exchanger may be jeopardized. Flow media could escape.
	f)	No residues of previous processes (e.g. cleaning agents) are present in the plate heat exchanger.
Primary pumps	a)	Equipment tag and nameplate are permanently affixed.
	b)	All pipe connected to the pump are correctly fastened.
	c)	All pump grouted in place.
	d)	Manufacturer clearance for service access and air circulation around equipment is obtained.
	e)	Pump vibration isolation devices are functional.
	f)	Pump alignment is verified.
	g)	Piping installed is in accordance with design drawings.

	h)	Piping is flushed clean and leak tested.
	i)	Pump is not leaking.
	j)	Lubrication is complete.
Piping valves and flow meters	a)	Pipe fittings are complete and pipes properly supported.
	b)	Piping type and flow direction are properly labelled.
	c)	Maintenance access to valves, flow meters and panel is acceptable.
	d)	Temperature transmitter is installed correctly.
	e)	Flow meter and transmitter is installed correctly.
	f)	Pressure transmitter is installed correctly.
	g)	Piping system is properly flushed.
	h)	Valves areinstalled properly.
	i)	Valves are open and closed.
	j)	Valves are properly tagged.
Electrical cables and wiring	a)	No visible damage that could impair safety appears.
	b)	Cables are protected against mechanical damage.
	c)	Correct phase identification is provided at both ends of the cable.
	d)	All non-armoured cables susceptible to damage are protected with steel conduit or trunking.
	e)	Conductor is sized to suit the rating of the fuse/MCB protecting the circuit.
	f)	No cable joint exists in the final circuit.
	g)	All metal conduits, trunking, switch boxes and exposed metal parts are effectively grounded.
Finalizing the installation	a)	Safe operating ranges have been reviewed and accepted.
	b)	Sequence of operation adequately shows all information.

4.2. Water flushing and air purge for piping

Water flushing and air purging was done in order to clean the assembled piping together with the heat exchanger, and this step is recommended when fine debris is not removed prior to the assembling process for the piping system. Distilled water was used to clean and purge the air inside the assembled primary piping.

The requirements for water flushing and air purge are as follows:

- Distilled water must be free of silt or debris;
- Flush velocity of 1.5–2 m/s for lateral cleaning.

4.3. Pressure test (hydrostatic)

In order to check for any leakage in the piping, pumps or heat exchanger, a pressure test was done to the whole system.

The requirements for pressure testing are as follows:

- RTP primary cooling system shall be capable of withstanding an internal water pressure not less than 1.5 times the operating pressure;
- Operating pressure for the primary cooling system is 3.3 bar, and a test pressure of 5 bar will be applied to the system;
- The pressure will be maintained at 5 bar for 24 hours;
- The piping system is divided into three sections in order to detect leakage during testing period.

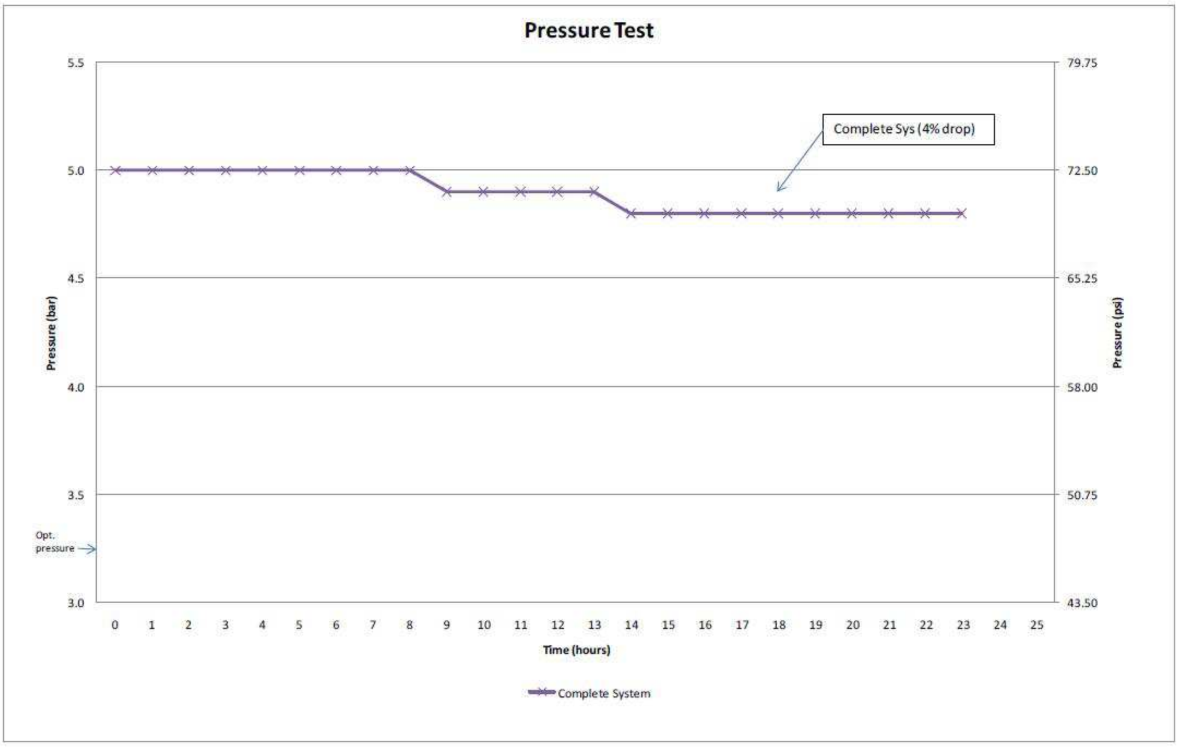

FIG. 6. Test result.

The full system was subjected to the test pressure of 5 bar for 24 hours. The test was considered successful if the pressure did not drop below 5 bars within 30 minutes, and there was no visible leakage throughout the test. The pressure trend for the hydrostatic test showed that the pressure was maintained at 5 bars for about 8 hours, which is above the requirement. No visible leakage was seen.

4.4. Reactor test run

RTP was operated in sequential power levels in order to verify that the new RTP primary cooling system was able to perform its functions and that parameters remained within acceptable limits.

The reactor was operated as follows:

- 0 kW (cold test) for 6 hours
- 100 kW for 2 hours;
- 500 kW for 6 hours;
- 750 kW for 6 hours;
- 1 MW for 14 hours.

The performance of the new system was based on the following criteria:

a. Temperature profile at 1 MW [3]

Inlet temperature, primary side	T1	110°F	43.33°C
Outlet temperature, primary side	T2	90°F	32.22°C
Inlet temperature, secondary side	T3	80°F	26.67°C
Outlet temperature, secondary side	T4	90°F	32.22°C

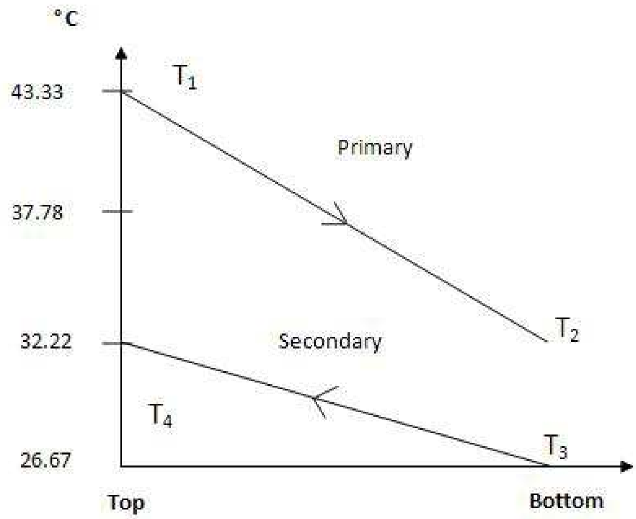

FIG. 7. Primary cooling temperature profile.

b. Temperature difference between inlet and outlet (ΔT) at 1MW [3]

$\Delta T = T_{in} - T_{out} = 43.33 - 32.22 = \mathbf{11.11°C}$

c. Logarithmic mean temperature difference (LMTD) [3]

$$LMTD = \frac{\Delta T_A - \Delta T_B}{\ln \dfrac{\Delta T_A}{\Delta T_B}} = 8.01$$

A cumulative run of 28 hours was done at stated reactor powers. The coolant flow, temperature and pressure were monitored. The temperature profile during hot test operation at 1 MW is shown in Figure 8.

Inlet temperature, primary side	T1	38.20	°C
Outlet temperature, primary side	T2	28.60	°C
Inlet temperature, secondary side	T3	26.60	°C
Outlet temperature, secondary side	T4	30.70	°C

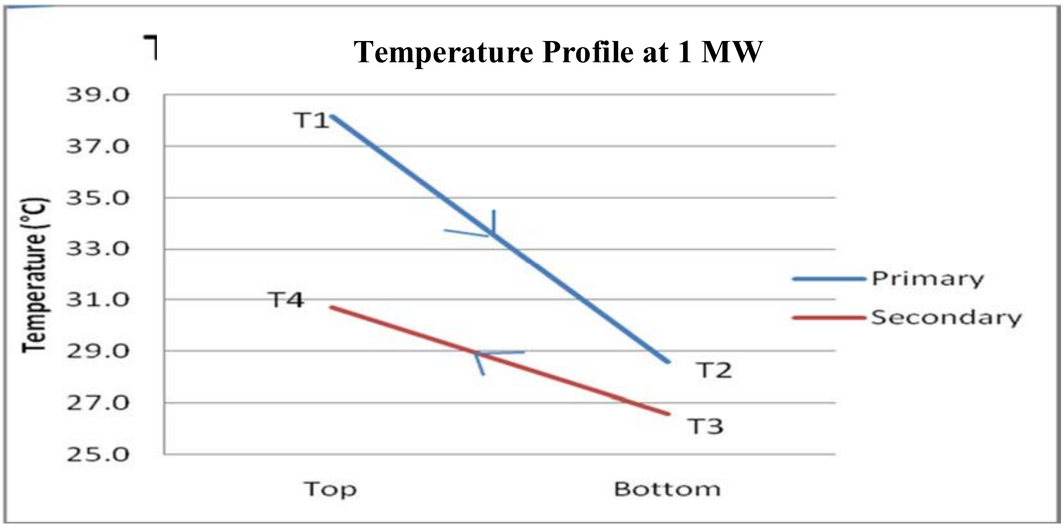

FIG. 8. Temperature profile during hot test operation at 1 MW.

Temperature difference between inlet and outlet (ΔT) at 1 MW during hot test:

$\Delta T = T_{in} – T_{out} = 38.2–28.6 = \mathbf{9.6°C}$

The results compare well the 2008 SAR, in which the difference was 1.51°C, i.e. 13.5% difference. LMTD during the 1 MW hot test:

$\Delta T_A = T1–T2 = 38.2–28.6 = 9.6$;

$\Delta T_B = T4–T3 = 30.7–26.6 = 4.1$;

$$LMTD = \frac{\Delta T_A - \Delta T_B}{\ln \frac{\Delta T_A}{\Delta T_B}} = 6.48$$

The LMTD difference between the 1 MW [3] and hot test is 1.53, an approximately 20% difference.

5. PROJECT MANAGEMENT

The RTP primary cooling system refurbishment project was carried out by a local contractor appointed by Nuclear Malaysia. With a budget allocation around RM 500 000.00, the project was successfully completed within 8 months from September 2009 until April 2010. The project management team was appointed to manage the project with members from the Reactor Operation and Maintenance and Technology Assessment and Engineering Safety Sections. The organization chart for the project management team is shown in Figure 9.

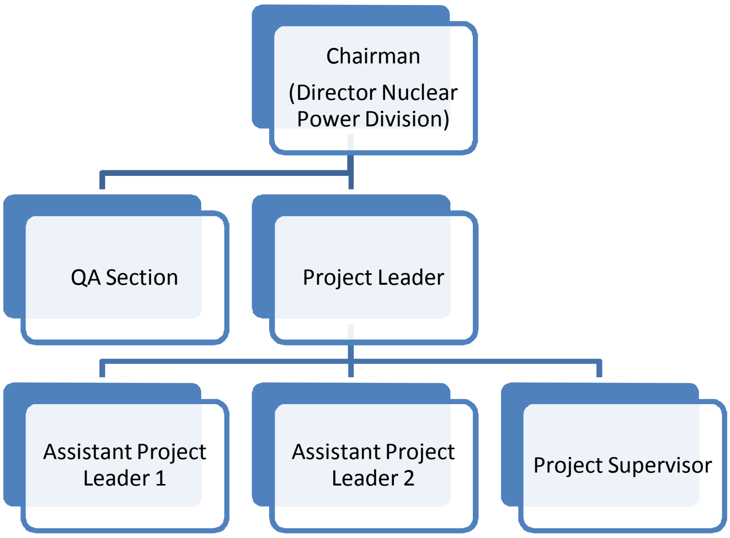

FIG. 9. Project management team.

Initially, the project duration was only four months from September to December 2009. However, due to the delay of some equipment procurements, the project was extended to 8 months. The timeline for the project is shown in Figure 10 below.

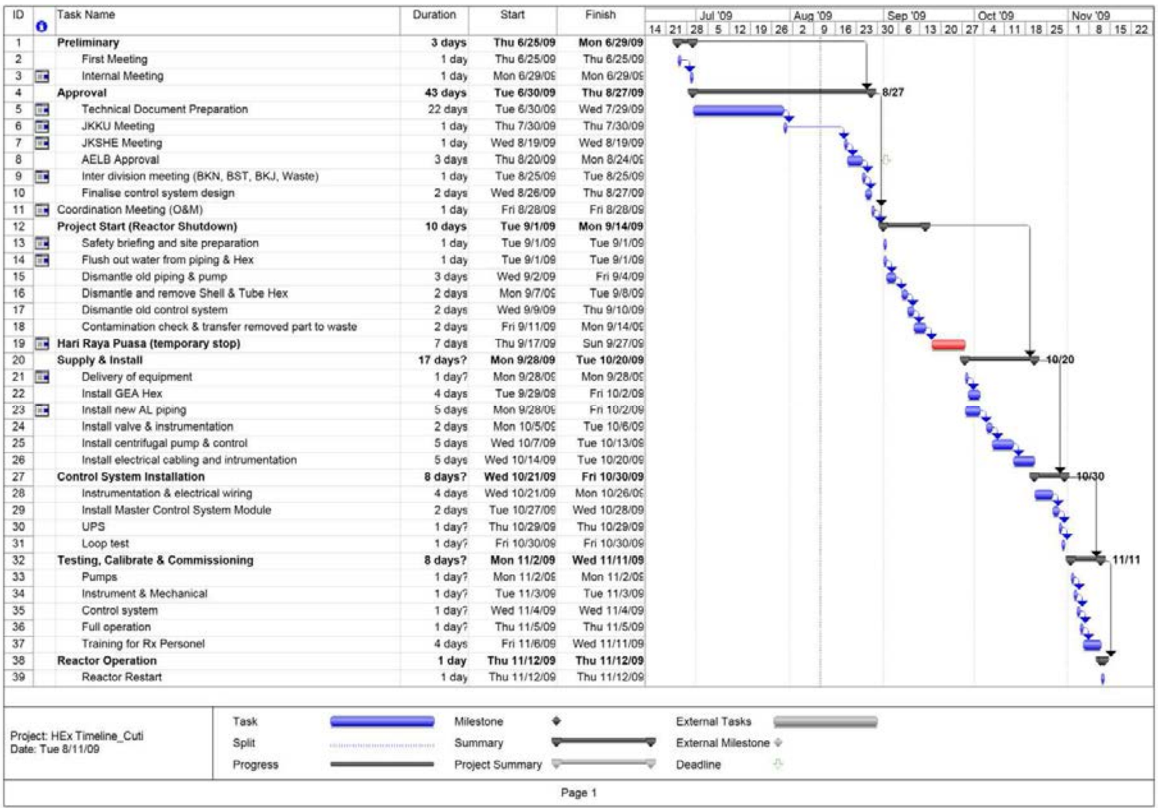

FIG. 10. Project timeline.

6. DESIGN AND SAFETY REVIEW

The design of the new RTP primary cooling system was reviewed by the Major Facilities Safety Sub-committee (JKKU) and endorsed by the Safety Health and Environment Committee (JKSHE). The responsibilities of the JKKU included review of experiments utilizing the reactor facilities; review of all proposed changes to the facility, procedures and operational limits; and determination of whether a proposed change, test or experiment would constitute an unreviewed safety question or a change in the operational limits and conditions. Figure 11 shows the flowchart of design and safety review.

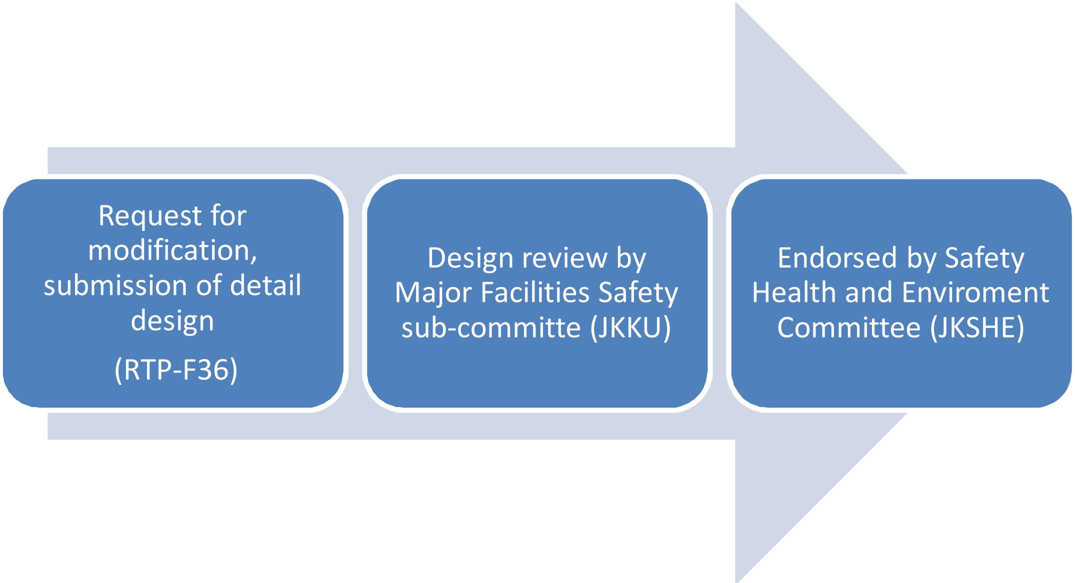

FIG. 11. Flowchart of design and safety review.

7. REGULATORY REQUIREMENT

The final design and calculation of the new RTP primary cooling system was submitted to the Atomic Energy Licensing Board (AELB) for review and to ensure the requirements are incorporated into the operating license, in particular, in the operating limits and conditions. After the completion of all project stages, a commissioning report was submitted to AELB to ensure conformance with the requirements for maintenance; inspection under the operating limits and conditions; and applicable regulations, codes and standards.

8. LESSONS LEARNED

Throughout each project activity, lessons are learned, and opportunities for improvement were discovered and discussed.

8.1. Aluminium piping

Aluminium is known as a soft, durable and lightweight alloy. It is also a difficult alloy to weld, which requires a skilful welder. Hence, Victaulic coupling is used instead of welding in order to assemble the aluminium piping. However it was discovered that welding was actually needed to place the instruments such as temperature probes and pressure, pH and conductivity sensors. To verify the quality of the weld, in-house non-destructing testing was conducted.

8.2. Centrifugal pump

The new centrifugal pump is capable of delivering up to 120 m^3/h, 50% higher than the previous pump. Since the flow for the primary cooling system is only 80 m^3/h, a modulating valve is used to control the flow. The opening of the modulating valve is only 45% in order to reach the required flow and place the valve under high pressure. To overcome this situation an adjustable speed drive or an inverter was installed to reduce the pump speed to obtain the required flow instead of using the modulating valve.

As part of a continuous improvement process, documenting lessons learned helps the project team to discover the root causes of problems that occurred and avoid these problems in later project stages or future projects.

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AGEING MANAGEMENT IN THE CENM TRIGA MARK II RESEARCH REACTOR

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Abstract

Physical ageing is one of the most important factors that may reduce the safety margins calculated in the design of safety system components of a research reactor. In this context, special efforts are necessary for ensuring the safety of research reactors through appropriate ageing management actions. The paper deals with the overall aspects of the ageing management system of the Moroccan TRIGA Mark II research reactor. The management system covers among others, management of structures, critical components inspections, the control command system and nuclear instrumentation verification. The paper presents also how maintenance and periodic testing are organized and managed in the reactor module. Practical examples of ageing management actions of some systems and components during recent years are presented.

1. INTRODUCTION

In order to ensure continued operation of the research reactor and compliance with the operational limits and conditions, an ageing management programme should be established. This ageing management includes, among others, different activities such as repair, refurbishment, and replacement of SSCs important to safety. This paper describes the ageing management system for the CENM TRIGA Mark II research reactor including reactor system categorisation, maintenance and periodic testing management, and record keeping. A practical example of ageing management for control rod connectors is presented.

2. THE CENM TRIGA MARK II RESEARCH REACTOR

The reactor, which is part of the National Centre for Energy, Sciences and Nuclear Techniques (CNESTEN), achieved initial criticality on 2 May 2007. It is a standard design 2 MW natural convection cooled reactor with a graphite reflector. The reactor is located near the bottom of a water filled aluminium tank that is 2.5 m in diameter and approximately 8.2 m deep. The water provides adequate shielding for a person standing at the top of the reactor. The control of the reactor is assured by five independent B_4C control rods mounted at the top of the tank on a bridge structure. The reactor uses solid, homogenous U–ZrH fuel–moderator elements developed by General Atomics technology in which the uranium is enriched to 20%. The reactor contains different irradiation facilities that converge at the reactor core and allow simultaneous performance of multiple experiments. These facilities include four BPs, a pneumatic transfer facility, a graphite thermal column, a rotary rack and a central irradiation facility that extends into a water filled flux trap.

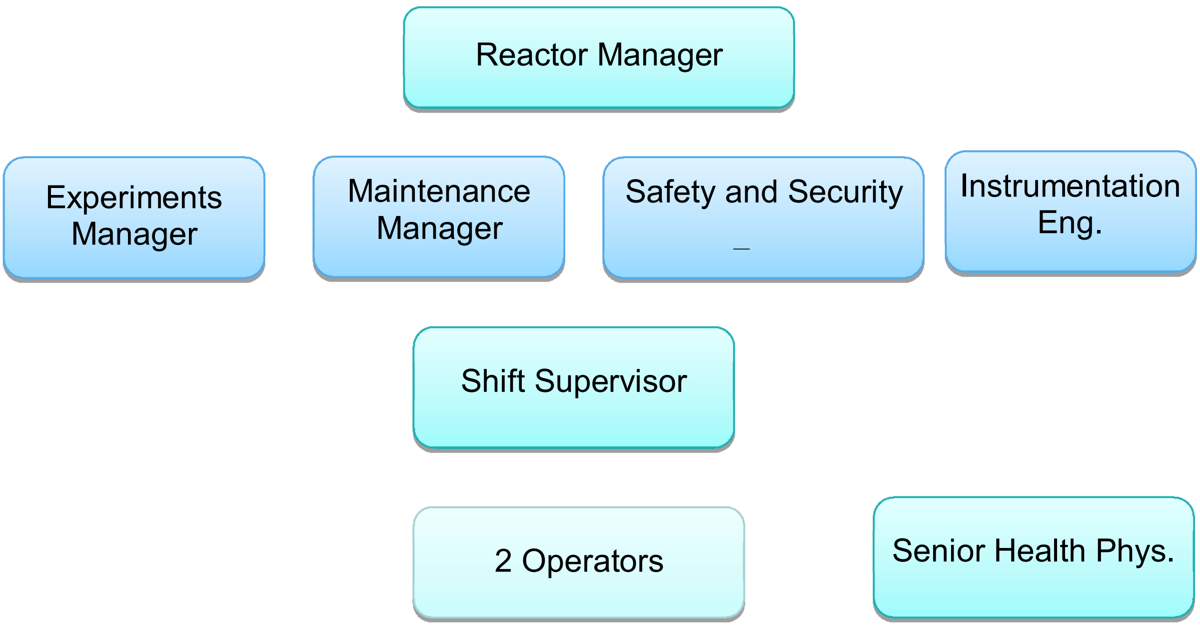

FIG. 1. Reactor organization.

2.1. Maintenance and periodic testing management

One of the key points of ageing management is maintenance and periodic testing. The preventive maintenance, activities such as lubrication, routine checks, monitors calibration and water sampling, is performed by a private company in the reactor module according to a programme submitted previously to the maintenance manager who ensures its execution. The performance of maintenance activities requires the approval of the maintenance or reactor manager. Activities like power calibration and control rod calibration are done by qualified persons from the reactor stuff. All maintenance activities are performed in alignment with the requirements of safety as derived from the operational limits and conditions, regulatory body requirements and design requirement.

2.1.1. Reactor systems categorization

Systems and components are categorised into four main categories: water systems, control command, radioprotection systems and ventilation systems.

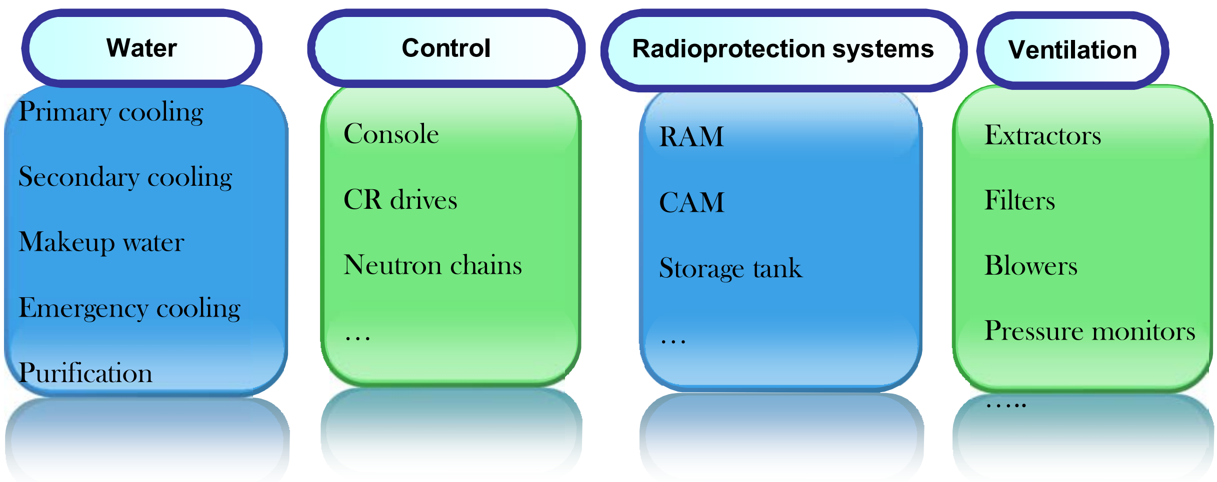

FIG. 2. Categroization of reactor systems.

2.1.2. Record keeping

Maintenance activities of apparatus and equipment are recorded in a maintenance log book to track any change or modification. The records include the equipment, the cause of failure, the date, the nature of activity performed, etc. The person who performed the work is properly identified. If an inspection, test or repair is required, a separate record is used. To facilitate this, each item has a separate identification number or designation. These records are important, as more than one person may have worked on an item and workers may not be aware of its maintenance history. Establishing an electronic database of maintenance records is under development.

2.2. Practical case: control rod connectors

Five fuel–follower control rods are provided to control reactor power during steady-state operation, and to quickly shut down the reactor when desired. They have an upper B_4C filled section that absorbs neutrons and a lower fuel section that enhances reactivity. Inserting the control rods into the core simultaneously removes fuel and adds a neutron absorber, quickly reducing reactivity.

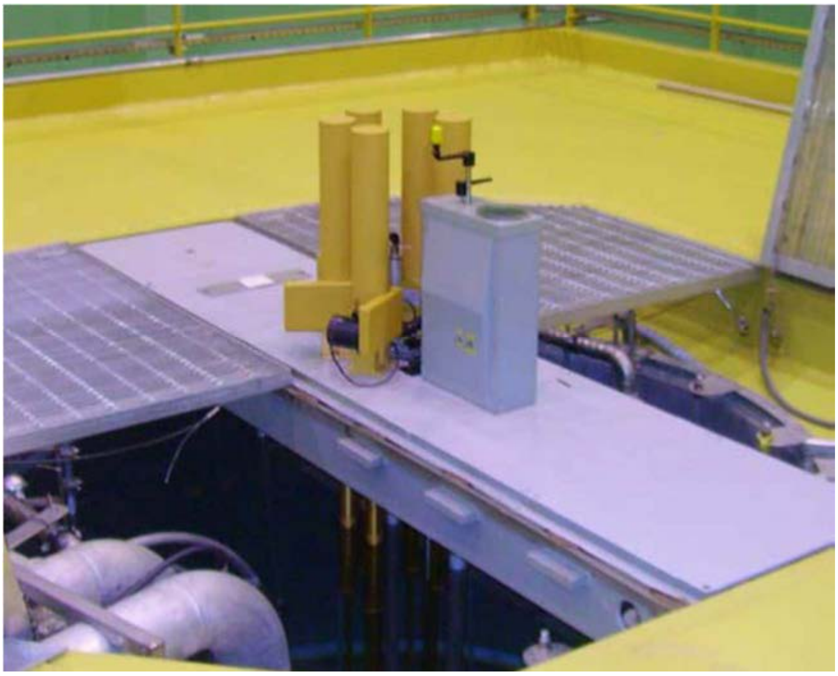

FIG. 3. Control rods atop the MA-R1 reactor.

The control rod drive mechanism is an electric stepping motor actuated linear drive equipped with a magnetic coupler and a positive feedback potentiometer. A series of micro-switches are provided on the drive assembly to control the up and down movements of the control rod and rod drive. Rod motion is controlled from the control console by signals to energize the associated magnets. One of these control rods presented problems in the display of its position on the console screen. The inspection of the electronics associated showed some corrosion trapped inside the connector on the top of the reactor.

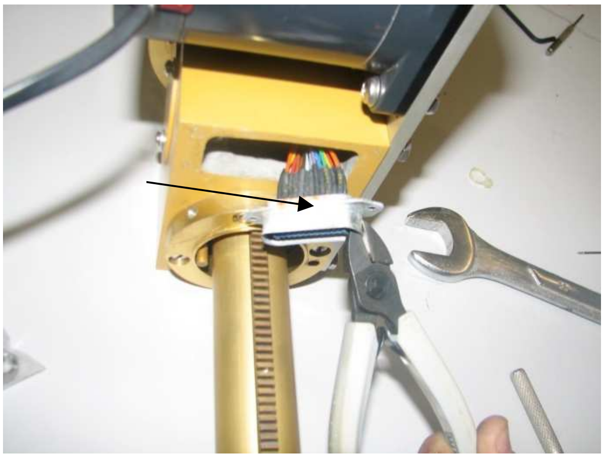

FIG. 4. Control rod drive mechanism.

This corrosion was a result of continuous exposure to significant relative humidity since the connector is installed on the top of the water pool.

3. CONCLUSION

As a corrective action, various plugs have been added at the lower part of the mechanism in order to reduce the humidity effect. The inspection of the control rod connectors has been added to the annual preventive maintenance programme.

PREVENTIVE AND PREDICTIVE MAINTENANCE, WAREHOUSING OF SPARES, PERIODIC TESTING AND IN-SERVICE INSPECTION ACTIVITIES AT THE NIGERIAN RESEARCH REACTOR-1 FACILITY

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Abstract

The Nigerian Research Reactor–1, or NIRR-1, is sited at Centre for Energy Research and Training, Ahmadu Bello University, Zaria, Nigeria. Activities on preventive or routine maintenance have been institutionalized since the commissioning of the reactor in February 2004. This has grossly reduced the rates of corrective maintenance activities and helped the reactor management a great deal in predicting failure rates of reactor components and other auxiliary units. Routine maintenance of systems and components are being carried out on a weekly, quarterly and annual basis based on manufacturer's recommendations, which have been reviewed and improved over the years. The paper presents the implementation of maintenance activities in NIRR-1 from its initial criticality in 2004 till today and the new scheme for periodic testing and in-service-inspection developed after an IAEA Integrated Safety Assessment of Research Reactors mission. The measures put in place are envisaged to reduce the negative impact of ageing on NIRR-1 and its auxiliary systems.

1. INTRODUCTION

NIRR-1 has a tank in pool structural configuration and operates at a nominal thermal power of 31 kW corresponding to a maximum neutron flux of 1×10^{12} $cm^{-2}s^{-1}$. The reactor core is cylindrical with a height and diameter of 230mm and fuelled by $U–Al_4$ enriched to about 90% in aluminium alloy cladding. It has a total number of 347 fuel elements and 3 dummies in its fuel lattice. It uses light water as moderator and coolant, with beryllium as the reflector. It has one control rod serving as the regulating rod in addition to the safety rod. An alternative means of reactor control is provided using external cadmium materials encapsulated in vials for transfer into the inner irradiation sites of the reactor, which ensure reactor subcriticality in case of stuck rod accidents [1, 2, 3].

Measures put in place to ensure safe operation of the facility include standard procedures developed for pre-startup, startup and shutdown; strict adherence to routine maintenance; and prompt response for corrective maintenance. These maintenance activities are carried out by a team of engineers and technicians who are also reactor operators under the authority of the reactor manager. Depending on the complexity and importance to safety, routine maintenance activities scheduled for the reactor and its auxiliary systems are carried out on weekly, quarterly and annual bases.

2. AGEING MANAGEMENT OF NIRR-1

NIRR-1 has no specific ageing management programme. However, maintenance activities implemented in the maintenance programme of the facility are aimed at reducing the negative impact of ageing. On the other hand, periodic testing and ISI activities were carried out indirectly during maintenance exercises until the IAEA Integrated Safety Assessment of Research Reactors (INSARR) mission of December 2009. Since the mission, a scheme has been developed to ensure safety in the operation of NIRR-1. The successful maintenance culture developed for the NIRR-1 facility is hinged around the availability of spare parts in storage for the facility and the management's efforts towards replenishment of the spares bank as they are consumed. Details of these activities are discussed below.

2.1. Maintenance programme of NIRR-1

The maintenance structure adopted for NIRR-1 is proactive and reactive based. The proactive component is based on preventive and predictive maintenance, while the reactive component is executed by corrective maintenance activities. The maintenance programme developed for the reactor is hence divided into two: routine maintenance, during which preventive and predictive maintenance are carried out, and corrective maintenance. Maintenance logbooks for routine and corrective maintenance activities are properly kept to track maintenance records of systems and components.

Because of the importance attached to maintenance, the last day of the working week, Friday, is dedicated for routine maintenance of NIRR-1. This is to ensure that the reactor is always ready for safe operation in the week to come. Routine maintenance activities for the reactor and its auxiliary systems are scheduled and carried out on weekly, quarterly and annual bases depending on their importance to safety and complexity. The preventive maintenance activities are relied upon for predictive maintenance, especially when consumables are required [4].

2.1.1. Weekly maintenance

Weekly maintenance is scheduled for systems that provide limiting conditions for safe operation as enshrined in the Final SAR of the facility [5, 6]. Details of system checks for weekly routine maintenance are tabulated in Table 1. These systems and their bearing to safety in the operation of NIRR-1 are as follows:

(a) Rabbit transfer system

This is a pneumatic control system used for the transfer of vials into the irradiation channels of the reactor for the purpose of utilization. The system comprises an air compressor, air pressure meter, sample positioning equipment (Type A and Type B) and connecting tubes. It is necessary for the system to be functional during reactor operation because it provides the alternative means of bringing the reactor to sub-criticality by sending cadmium rabbits into the reactor in the event of rod failure.

(b) Gas purge system

The gas purge system is operated once a week for one minute. The system injects clean air into the reactor vessel to purge radioactive gasses and hydrogen that might have accumulated during reactor operation within the week. The system is made up of a vacuum pump, mechanical filter, radioactive filter, pressure level indicator, valves and connecting tubes. It is necessary to ensure the functionality of the system before its next operation. This ensures that hydrogen gas generated in the reactor vessel is maintained below a potentially explosive level and radioactive effluents are filtered before being released to the environment.

(c) Ventilation system

The ventilation system is operated continuously during reactor operation. This keeps the reactor hall at a negative pressure in relation to the adjourning rooms. The system is comprised of the stack, ventilation fan and the ventilation ducts.

(d) Reactor water monitoring system

The reactor water monitoring system ensures that the quantity and quality of the water in the reactor vessel is within the acceptable limits. The system comprises reactor vessel water level high and low indicators, water conductivity meters and water purification equipment. Of utmost importance to safety are the reactor vessel water level indicators, which are ensured to be within acceptable limits prior to any operation.

(e) Pool water monitoring system

The pool water monitoring system ensures that the quantity and quality of the water in the reactor pool is within the acceptable limits. The system comprises reactor pool water level high and low indicators, water conductivity meters, water purification equipment and pool water cooling equipment such as a chiller. Of utmost importance to safety are the reactor pool water level indications, which are ensured to be within acceptable limits prior to any operation.

TABLE 1: SYSTEMS SCHEDULED FOR WEEKLY MAINTENANCE

Systems	Details of maintenance checks
Rabbit transfer system	• Air compressor oil level; • Automatic compressed air regulator settings: auto trip and auto restart; • Operation of glove box air exhaust pump; • Operation of samples stripper; • Operations of dividers and transformer; • Draining of condensed water from the collection points in the system.
Gas purge system	• Oil level in gas purge pump; • Noise and vibration level of gas purge pump; • Pressure difference during gas purging.
Ventilation system	• Oil level in ventilation pump check; • Noise and vibration level of ventilation pump; • Air leakage in the ventilation system.
Reactor water monitoring system	• Zero and Full-scale check for reactor water conductivity meter; • Reactor water flow rate meter; • Reactor water purification pump noise and vibration level; • Water leakage checks in the reactor water purification loop.
Pool water monitoring system	• Zero and full scale check of pool water conductivity meter; • Pool water flow rate meter; • Pool water purification pump noise and vibration level; • Water leakage checks in the pool water purification loop.
Provisional pool water cooling	• Chiller water inlet and outlet temperature control and control indicators’ • Chiller pressure and water level.

2.1.2 Quarterly maintenance

Quarterly routine maintenance is mainly conducted on the reactor control console and the associated control instrumentation. Table 2 provides the details of the activities carried out in these maintenance exercises. The importance of the control console and the closed loop computer control system to safety in the operation of the reactor cannot be overemphasized.

TABLE 2: DETAILED SCHEDULE FOR QUARTERLY MAINTENANCE

Item	Maintenance details
Computer closed-loop control system	• Confirm communication of control interface card and control transfer switch; • Operate reactor in automatic and manual mode; • Test for all trip settings, including scram.
Main control console	• Cleaning of dust and contact points for contactors with power supply off; • Visual inspection of solder points; • Confirmation of voltage levels between identified points; • Manual and automatic startup, shutdown and scram functions; • Response of all warning lights; • Responses of reactor overpower and over ΔT lights to test trip conditions; • Confirmation of control of reactor access via rabbit systems; • Functionality test for uninterruptable power supply (alternative power supply)'

2.1.3 Annual maintenance

The annual maintenance activities involves the servicing of the control rod drive mechanism in addition to all the systems and components involved in the weekly and quarterly maintenance schedules. During this exercise also the calibration checks on systems related to safety settings including radiation monitors are carried out. Table 3 provides the details of the activities carried out during the maintenance exercise.

TABLE 3: DETAILED SCHEDULE FOR ANNUAL MAINTENANCE

Item	Maintenance details
Control rod drive mechanism	• Servicing in accordance with a written procedure, approved by reactor safety committee; • Self-lock test, sensitivity test, rod drop test, operation performance test, and position indicator test.
Main control console and computer control system	As in Table 2.
Calibration checks	• Water conductivity meters; • Water level upper and lower sensors; • Temperature difference monitor; • Neutron flux monitor; • Control rod limiting position preset; • Gamma probes settings.
Others	• Cleaning of resins for reactor water and pool water purification systems; • Testing of public address, fire alarm and fire control system.

2.1.4. Corrective maintenance

Corrective maintenance is performed to overcome anomalies observed during routine maintenance and at breakdowns. The breakdown experienced that had a bearing on safety occurred while the reactor was in operation, and was caused by a control rod stuck at a position 185 mm from bottom of the core (height of the core is 230 mm). This accident occurred when the reactor neutron flux exceeded 1.2×10^{12} $cm^{-2}s^{-1}$ (corresponding to 37.2 kW), and the reactor safety system was automatically actuated with the control rod position at 220 mm. The automatic shutdown signal de-energized the electromagnetic clutch of the control rod drive mechanism, which allowed the control rod to fall freely due to gravitational influence. Due to some broken pieces of gear teeth in the gears of the control rod drive mechanism, however, the free fall was intercepted and the rod was stuck at the position 185 mm. The reactor was controlled using the alternate means by sending cadmium rabbits into the inner irradiation sites, which brought it to subcriticality while corrective maintenance was carried out.

2.2. Scheme for periodic testing and ISI

The scheme was established to confirm the settings for over-power and over-temperature signals, which are responsible for the activation of the scram signal for the reactor, and also for the confirmation of the signal levels for operation status such as water levels, warning lights, rod movement, reactor access by the rabbit system and calibration of detectors for utilization. ISI is conducted during quarterly routine maintenance activities, which often leads to minor corrective action even prior to component failure or system breakdown.

2.3. Stocking of spares

The successful maintenance culture developed for the NIRR-1 facility is hinged around the availability of spare parts in storage for the facility and the management's efforts towards replenishment of the spares bank as they are consumed.

3. CONCLUSION AND ACKNOWLEDGEMENT

The measures put in place are envisaged to reduce the negative impact of ageing on NIRR-1 and its auxiliary systems. However, the limitation of availability in the local market for some spares poses a threat to successful implementation of an ageing management programme in the facility. Reliable leads and other necessary assistance from the IAEA for the procurement of some of these spares would help to overcome this threat. The management of NIRR-1 is grateful to the IAEA for its concern and assistance towards the safe and effective utilization of the facility.

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REPLACEMENT OF THE PUMPS FOR FUEL CHANNEL COOLING CIRCUIT OF THE MARIA RESEARCH REACTOR

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Abstract

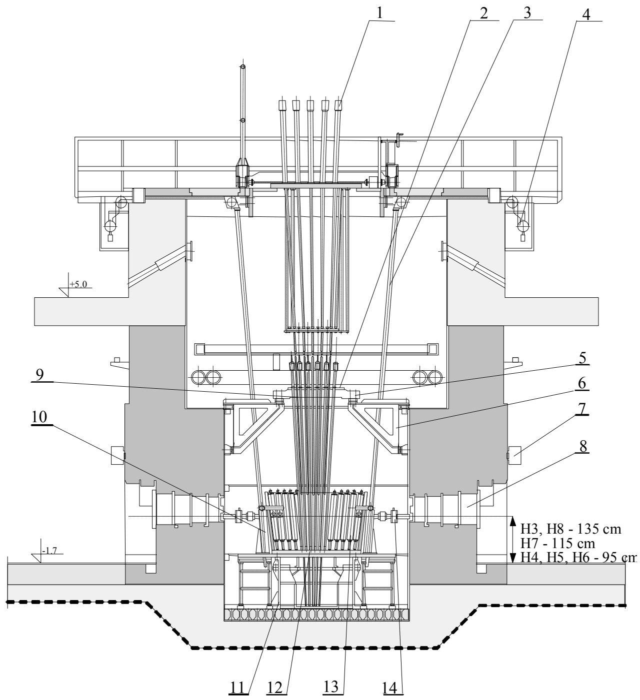
The high flux Maria research reactor is operated by the National Centre for Nuclear Research in Świerk. It is a pool type reactor with pressurized fuel channels located in the beryllium matrix. According to the Global Threat Reduction Initiative programme our goal is to convert the Maria reactor from HEU to LEU fuel. Hydraulic losses in the new LEU fuel produced by CERCA are about 30% higher than the existing HEU fuel of type MR-6. For the MR-6 fuel were installed four two speed pumps. These pumps performed the function of the main circulations pumps during reactor operation with residual pumping power provided by emergency pumps. In the new system four main pumps will be used for circulating coolant while the reactor is operation with three auxiliary pumps for decay heat removal after reactor shutdown, meaning that the conversion of Maria research reactor will be possible after increasing flow in the primary cooling circuit of the fuel channels. The technical design of replacement of the pumps in the primary fuel channel cooling circuit was finished in April 2011 and accepted by the Safety Committee. After delivery of the new pumps we are planning to upgrade the primary fuel channel cooling circuit during October–November 2012.

1. INTRODUCTION

The Institute of Atomic Energy (IAE) decided to utilize LEU silicide fuel qualified under the GTRI programme for conversion of the Maria reactor. IAE contracted with CERCA in France to supply LEU lead test assemblies. These LEU assemblies during irradiation testing required a higher coolant flow rate per fuel assembly and have more hydraulic resistance to coolant flow than do the current HEU fuel assemblies manufactured in the Russian Federation. Maria has four two speed pumps in parallel for maintaining flow of primary coolant through the fuel assemblies. Normal practice is to operate two pumps at full speed and have the other two pumps in standby. The half speed mode is used for shutdown cooling. After a study, IEA proposed replacing the four two speed primary pumps with four single speed main pumps that have a higher capacity than the current pumps for normal operation, and two single speed auxiliary pumps will be used for shutdown cooling. An additional recommendation was that all replacement pumps, valves and piping be resistant to intergranular stress corrosion.

2. MARIA RESEARCH REACTOR DESCRIPTION

The multipurpose high flux research reactor Maria is a water and beryllium moderated pool type reactor with graphite reflector and pressurised channels containing concentric six tube assemblies of fuel elements, as in Figure 1. It was designed to provide a high degree of flexibility.



1. control rod drive mechanism
2. mounting plate
3. ionization chamber channel
4. ionization chamber drive mechanism
5. fuel and loop channels support plate
6. plate support console
7. horizontal beam tube shutter drive mechanism
8. beam tube shutter
9. fuel channel
10. ionization chambers shield
11. core and support structure
12. core and reflector support plate
13. reflector blocks
14. beam tube compensator joint

*FIG. 1. A vertical cross-section of the reactor pool.*The main characteristics and data of Maria reactor are as follows:

- Nominal power: 30 MW_{th};
  - o Thermal neutron flux density: 4.0×10^{14} $cm^{-2}s^{-1}$;

- Moderator: H_2O, beryllium;
- Cooling system: channel type;
- Fuel assemblies:
  - o Material UO_2–Al alloy;
  - o Enrichment 36%;
  - o Cladding aluminium;
  - o Shape six concentric tubes;
  - o Active length 1000 mm;

- Output thermal neutron flux:
  - o At horizontal channels $3–5\times10^{9}$ $cm^{-2}s^{-1}$.

The functions of the fuel channel cooling system are:

- Removing the heat generated in fuel elements during normal reactor operation;
- Removing the decay heat following reactor scram;
- Retaining the fission products in the circuit during the situation when fuel element cladding has lost its integrity.

It is a circulation circuit closed with elevated static pressure provided by the pressuriser. A simplified diagram of the fuel channel circuit is presented in Figure 2.

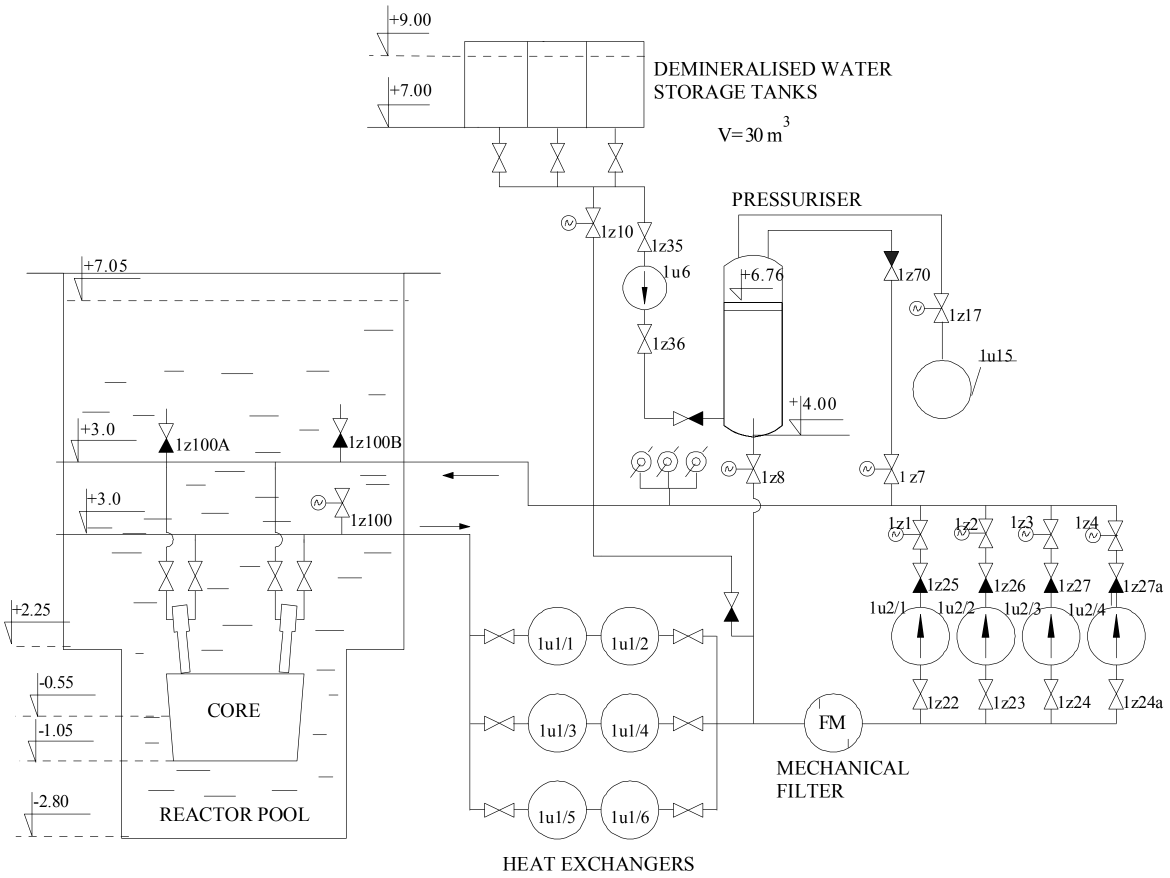


FIG. 2. Simplified diagram of the fuel channel cooling system.

The fuel channel cooling circuit consists of the following main equipment and components:

- Main circulation pumps 1u2/1 and 1u2/4;
- Heat exchangers, three sets of two exchangers in each one in a series system, 1u1/1 and 1u1/6;
- Mechanical netting filter 1u3.

In the Maria reactor each fuel element is placed in a separate fuel channel connected to the outlet and inlet collectors by means of pipes and valves. The collectors and the valves are located in the reactor pool. The main circulation pumps, heat exchangers and mechanical filters are situated in the pumping station of the primary circuits. The main circulation pumps provide coolant circulation during normal operation and in the dormant periods after reactor scram. They are driven by two speed electric motors, each of which provides rotation at basic feeding of ~3000 rpm. If feeding is lacking they are automatically switched to emergency feeding of ~1500 rpm to provide decay heat cooling.

3. INITIAL DATA FOR TECHNICAL DESIGN

Technical design was elaborated, taking into account the recommendations of IEA POLATOM Nuclear Safety, which are as follows:

- Maximum water flow rate through the fuel channel of 30 m^3/h;
- Maximum number of simultaneously operating fuel channels is 25;
- Water flow rate through the degasifier of 30 m^3/h;
- Total flow rate of water through the circulation pumps of 780 m^3/h;

- Hydraulic losses of the circuit for the flow rate 780 m^3/h of 125 m H_2O;
- Quantity of the main pumps installed is 4;
- Number of operating main pumps simultaneously is 2;
- Water flow rate circulating through the fuel channels during cooling down and after collapse of the power supply of 120 m^3/h;
- Water flow rate circulating through the bypass of an auxiliary pump of 15 m^3/h;
- Three auxiliary pumps;
- Two auxiliary pumps operate during normal reactor operation and for post-shutdown cooling.

Parameters of the main pump:

- Flow rate: 400 m^3/h;
- Pressure head: 128 m H_2O;
- Power demand: 179 kW;
- Engine power: 200 kW.

Parameters of an auxiliary pump:

- Flow rate: 70 m^3/h;
- Pressure head: 12 m H_2O;
- Power demand: 3.1 kW;
- Engine power: 4 kW.

4. SCOPE OF MODERNIZATION

The necessity for reactor cooling system modernization is caused by the anticipated change of the fuel channels from MR-6 to MC-5. The new fuel channels have dissimilar hydraulic characteristic, notably a larger cooling water flow rate and greater hydraulic resistance. They therefore require the use of circulation pumps with appropriate characteristics.

The scope of the modernization is described below.

4.1. Technological scope

- Replacement of the four main circulation pumps with two speed motors by single speed pumps with different characteristics;
- Installation of three additional auxiliary pumps to be used for emergency cooling and after reactor shutdown;
- Installation of additional racking and hangers for tubes.

4.2. Electrical scope

- Modernization of the normal and emergency power supply of the pumps and fittings;
- Steering of pumps and fittings.

4.3. Building scope

- Accommodation of main pumps' foundation for installation of the new main pumps;
- Replacement of vibro-insulators under the foundations of the main pumps;
- Construction of foundations for the auxiliary pumps.

5. OPERATION OF THE REACTOR FUEL CHANNEL COOLING CIRCUIT

5.1. Normal operation

When the reactor operates at power, the two main pumps and two auxiliary pumps are on. The remaining main pumps and an auxiliary pump are left in reserve. The capacity of the two main pumps, 800 m^3/h, ensures cooling for the reactor, which contains 25 fuel channels.

The auxiliary pumps during reactor operation circulate coolant through the bypass with a flow rate of 15 m^3/h. After disengaging the main pumps the operating auxiliary pumps circulate coolant through the core with a flow rate of 120 m^3/h.

The startup of the main pumps is performed by the operator, and in this time the two main pumps, electrically isolated, are set into motion. The main pumps are set into motion successively. The operator then starts the auxiliary pumps, which are supplied from an emergency source.

The main pumps are to run automatically with gate valve drives installed behind each main pump. When the main pump is off, the gate valve behind the pump is closed. Setting into motion the pump brings about opening of the gate valves to full discharge. Disengaging the main pump brings about closure of the gate valve in automatic mode. If during the startup one of the pumps was not activated, the operator can start a reserve pump.

Disconnecting the main pumps after reactor shutdown causes a decrease of pressure at the delivery collector in the circuit. When the pressure difference between the delivery and suction collectors of the main pumps reaches the pressure head of the auxiliary pump, water flows to the delivery collector of the reactor cooling system, ensuring cooling of the fuel channels.

The automatic mode of operation of the main circulation pumps should protect against an excessive overload when only one main pump is running. The hydraulic characteristics of the main circulation pump enable increasing the flow capacity from 400 m^3/h up to 620 m^3/h in the case when only one pump is being operated. Such a great change in capacity can cause:

- Large overload of the pump and consequently, burnout of the motor;
- Possible vibration and chatter of the pump and piping;
- Eventual cavitation of pump.

One can avoid burning of the motor by employing one of greater power.

To prevent such consequences in the case when only one main pump is being operated, it is necessary to impose a flow rate limit of 500 m^3/h by means of the partial opening of the gate valve behind the operating pump. This limitation depends on measuring water flow rate in the circuit.

5.2. Emergency situations

The operation of the main and auxiliary pumps in various emergency situations, for example electric supply collapse, breakdown of pumps and fittings drives, has been analysed.

5.2.1. Main circulation pump

Collapse of the electric supply brings about disengagment of the main circulation pumps and a reactor scram. The reactor is scrammed by the pressure drop pulse at the delivery collector. The cooling role after shutdown is taken over by auxiliary pumps supplied from an emergency electric source.

The nature of a breakdown of the main circulation pump, on the other hand, may cause prompt self-acting disengagement of pump in the case of a motor failure or give indications of the abnormality of pump operation, e.g. bearing overheating and leakage. In case of motor failure disengagement of pump entails automatic reactor shutdown by the resulting reactor coolant pressure drop and reactor cooling by operating auxiliary pumps. In case of abnormal operation the reactor operator decides whether to scram the reactor or to prolong reactor operation by starting a reserve pump and removing from operation the pump exhibiting defects.

5.2.2. Auxiliary pumps

Loss of the basic electric supply from the grid does not disempower the operating auxiliary pumps since they are supplied from the emergency supply, a switchboard of warranted voltage. Failure of the operating auxiliary pump during reactor operation causes a self-acting activation of the reserve pump. In this case the reactor operator makes the decision on whether to shut down the reactor or continue reactor operation until termination of the experiment.

6. SCOPE OF ANTICIPATED ALTERATIONS IN THE REACTOR COOLING SYSTEM

The National Centre for Nuclear Research will arrange for component installation. This includes removal of existing components, preparation for mounting foundations, procurement for cables for electricity and I&C system modification as well as the actual installation of new components.

The scope of anticipated alterations is as follows:

- Replacement of the four circulation pumps. Instead of the existing two speed pumps manufactured by Guinard Pumps, Ltd, one speed pumps of 12A32-P7 type fabricated by Grupa Powen-Wafapomp SA will be installed;
- The suction pipeline of the main pumps will remain the same. The removal of the pipeline DN300 between pumps 2 and 3 is foreseen to accomplish an initial pipeline;
- Installation of an additional support for the pipeline DN200 near the pump No 4;
- Replacement of the gate valves DN200 at each suction of the main pumps;
- Installation of new segments of piping between the gate valves and suction nozzles of pumps;
- Replacement of the gate valves DN200 with electric actuators and check valves to be installed upon delivery of pump piping. Fabrication of new reducers DN200/DN150, linking the pump delivery ferrules with fittings, is anticipated;
- Removal of the delivery pipeline DN300 and fabrication of an insert due to necessary spacing for the ferrules of the new main pumps;
- Installation of an additional spring actuated delivery tube hanger DN200 in the region of pump 4;

- Installation of three auxiliary pumps of 8A20-P type produced by Grupa Powen-Wafapomp SA, as agreed upon with the investor;
- Disassembly of the pipeline node connecting the auxiliary pumps with the suction and delivery pipelines DN300 of the reactor cooling system.

7. CONCLUSION

A pump installation plan will be submitted for review and approval by the Polish regulatory authority National Atomic Energy Agency. The endpoint for this task is the necessary approvals for pump installation and commissioning to begin.

After delivery of the new pumps, which is foreseen in August 2012, the Maria reactor will be shut down for two months during October–November 2011 for modernization of the fuel channel cooling circuit.

In the first stage, the assembled installation will be cleaned and degreased. The main step for acceptance will be the pressure test to be performed with a pressure 1.25 times as great as the working pressure. Starting tests will be performed for different combinations of pump operation. During the starting tests vibration measurements will be performed.

TRIGA RESEARCH REACTOR CONVERSION TO LEU AND MODERNIZATION OF SAFETY RELATED SYSTEMS

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1. FULL CONVERSION OF THE TRIGA RESEARCH REACTOR FROM HEU TO LEU

The USA and IAEA proposed an international programme to reduce the enrichment of uranium in research reactors by converting nuclear fuel containing HEU into fuel containing 20% enriched uranium. The Government of Romania joined the programme and actively supported political, scientific, technical and economic actions that led to the conversion of the active area of the 14 MW TRIGA reactor at the Institute for Nuclear Research in Piteşti in May 2006.

This confirmed the continuity of the Romanian Government's non-proliferation policy and their active support of international cooperation. Conversion of the Piteşti research reactor was made possible by completion of milestones in the Research Agreement for Reactor Conversion, a contract signed with the US Department of Energy and Argonne National Laboratory. This agreement provided scientific and technical support and the possibility of delivery of all HEU TRIGA fuel to the United States. Additionally, about 65% of the fresh LEU fuel needed to start the conversion was delivered in the period 1992–1994. Furthermore, conversion was promoted through IAEA Technical Cooperation project ROM/4/024 project funded primarily by the United States that supported technical and scientific efforts and the delivery of the remaining required LEU nuclear fuel to complete the conversion.

Nuclear fuel to complete the conversion was made by the French company CERCA with a tripartite contract among the IAEA, CERCA and Romania. The contract was funded by the US Department of Energy with a voluntary contribution by the Romanian Government. The contract stipulated manufacturing and delivery of LEU fuel by CERCA with compliance measures for quality, delivery schedule and safety requirements set by IAEA standards and Romanian legislation. The project was supported by the ongoing technical cooperation, safeguards, legal and procurement assistance of the IAEA, in particular its Department of Nuclear Safety [1].

For Romanian research, the conversion of the TRIGA reactor core in Piteşti demonstrated the ability of international cooperation and professional competence to implement long term, multiple and unique programmes to ensure elimination of the used fuel containing HEU. At the same time, the conversion was the first of the TRIGA reactors, and it ensured no constraints for using nuclear research necessary to support Romania and international cooperation.

This project through international cooperation has allowed political commitments to reduce the use of HEU fuelled research reactors and achieve a useful experience for numerous conversion projects of the IAEA.

2. MODERNIZATION OF SAFETY RELATED SYSTEMS AT THE TRIGA RESEARCH REACTOR

Research reactors play an important role in creating and maintaining the necessary infrastructure for the progress of energy programmes and research in areas of national development.

In this context and taking into account operating experience over 27 years, as of 2008, of the 14 MW steady stateTRIGA reactor, or TRIGA-SSR, it was necessary to modernize both the operating programmes and associated systems to achieve the following objectives:

- Safe operation of the reactor;
- Maintaining the reactor at a competitive level in accordance with requirements;
- Performance of the current and anticipated requirements for nuclear research.

TRIGA reactor modernization aimed to improve the performance of reactor irradiation in order to satisfy domestic and foreign partners.

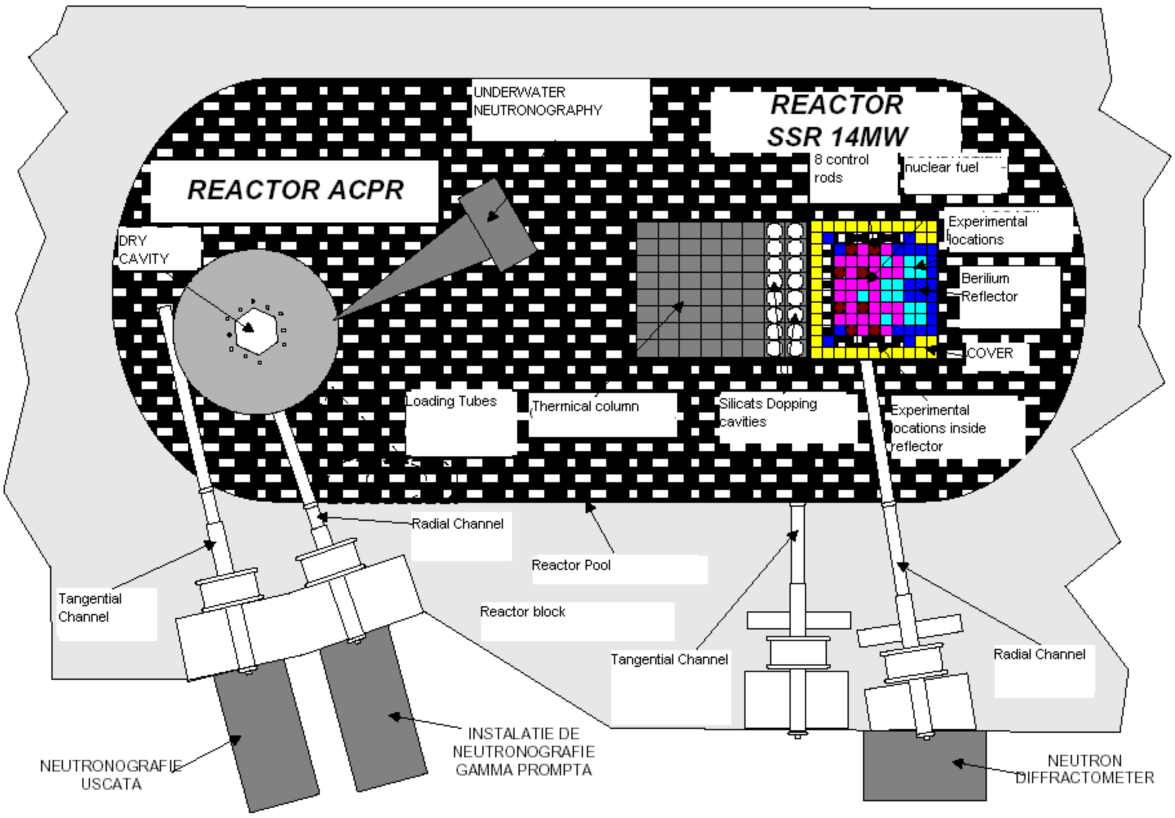

FIG. 1. TRIGA structure.

The main uses of the TRIGA reactor are nuclear fuel testing and materials testing. These types of tests are performed on specific radiation devices. The main types of irradiation tests that can be performed in the TRIGA reactor are:

- Power ramps;
- Ramp superpower;
- Measurements of fission gas pressure;
- Fission gas analysis;
- Residual deformation of the fuel element cladding;

- Tests on structural materials (Zr–2.5%Nb);
- Power ramps on CANDU fuel elements.

The modernization of the control command system of the TRIGA SSR followed the separation of the operating and security systems according to current regulations and recommendations of the National Commission for Nuclear Activities Control (CNCAN) and the IAEA. A console provided by INVAP S.E., Argentina, was built and tested under a comprehensive CNCAN control regime. The commissioning tests for the new control and command system of the reactor, performed in December 2009 under CNCAN surveillance, demonstrated the fulfilment of nuclear safety objectives and pre-defined design requirements.

The Institute for Nuclear Research developed a new concept of control rods of boron carbonate. In order to fulfil nuclear safety requirements, the design of new rods took into account the conservation of neutron absorption capacity as well their hydraulic behaviour. Also, the new control rod design has the advantage of easier maintenance by eliminating the need for a system to create vacuum pressure. Considering the requirements mentioned above, the new mechanical design of the control rod improves the structural stability of the entire control bar. The main drawback of the original bars was corrosion that led to water penetration and swelling of the absorbent section of the rod; thus, this modernization increased the reliability of the reactor control system.

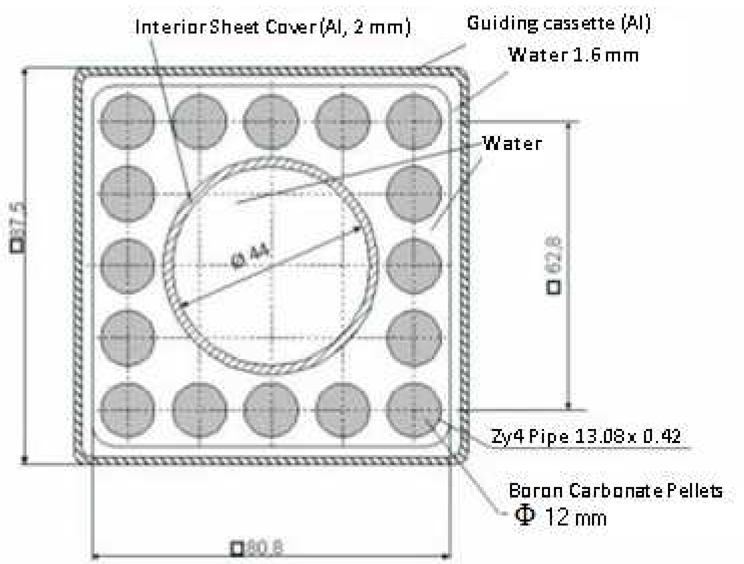

FIG. 2. New control rod design.

Rehabilitation of the reactor primary coolant circuit has as major objectives:

- Controlling, measuring, displaying and storing the operational parameters of the reactor;
- Executing security functions imposed by technical limits and conditions;
- Enabling evaluation in real time of the operational parameters for decision making regarding the operational state of the system;
- Ensuring redundancy.

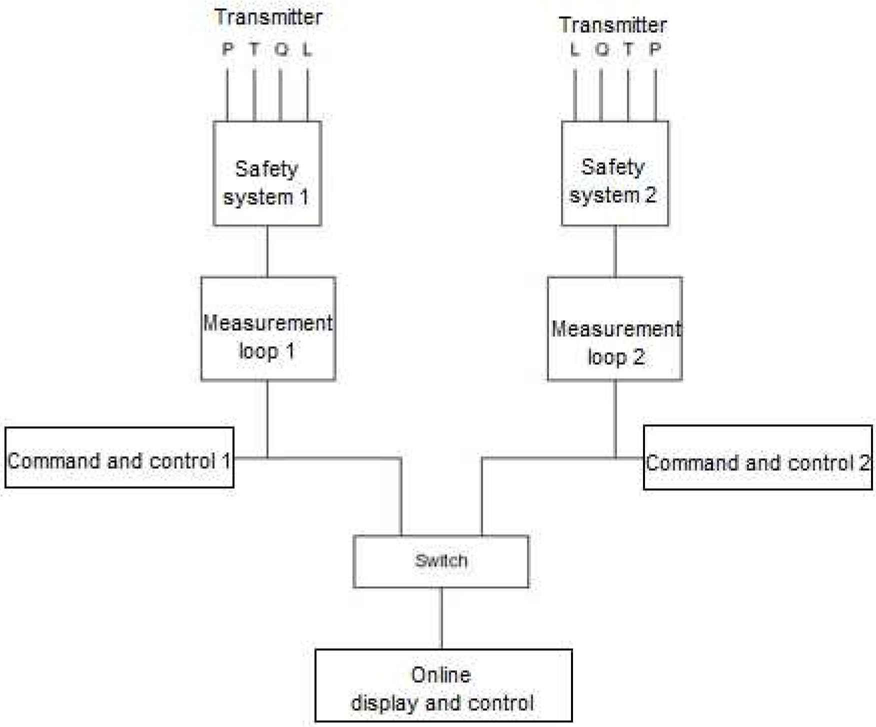

FIG. 3. Primary cooling circuit.

In accordance with the CNCAN fundamental norms on radiological safety [2], among which is the mandatory reduction of the dose limits for occupationally exposed individuals and populations, dosimetry system rehabilitation was determined necessary. The rehabilitation project included new configurations of fixed equipment dosimetry with the possibility of remote data transmission, processing and online evaluation parameters. The new draft contained more measurement points, taking account of the fundamental norms on radiological safety and international radiological safety recommendations.

Rehabilitation of the secondary cooling circuit of the reactor aimed to increase the performance of the heat exchangers servicing the cooling towers in order to ensure heat removal from reactor operation up to 21 MW. Rehabilitation activities on the secondary cooling circuit had no direct consequences on nuclear safety because this circuit is an auxiliary classical circuit. Test results in October 2007 after completion of rehabilitation activities confirmed the ability of removing heat from operation of the reactor at a power of 21 MW.

Rehabilitation and conditioning of the ventilation system involved replacing equipment and components with new elements that better accommodate the operating characteristics of original system. With the completion of activities of modernization and rehabilitation of these systems an increase of the power rating of the TRIGA-SSR reactor from 14 MW to 21 MW is possible, ensuring competitiveness in relation to current and future requirements.

Rehabilitation of the irradiation devices took into account national regulations and IAEA recommendations. This activity also monitored compliance with the irradiation specifications of the internal and external beneficiaries and with nuclear safety operating conditions. The main irradiation devices of the TRIGA reactor with implications for its use and availability are the C2 capsule and loop A.

In over 30 years of operation, the TRIGA reactor had no events with major consequences on nuclear safety, personnel or population. This is because reactor operation complied with:

- A set of technical specifications and administrative guidelines;
- A quality assurance manual;
- IAEA recommendations;
- Internal working instructions and procedures;
- A programme of training and staff appraisal.

The existence and the use of these tools led to minimized risks and increased nuclear safety. In fact, the IAEA INSARR missions developed in 2000 and 2006 confirmed by the reports prepared after the missions an appropriate level of nuclear safety and proper operation of the nuclear facility.

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REGULATORY OVERSIGHT FOR AGEING MANAGEMENT OF NUCLEAR RESEARCH INSTALLATIONS IN THE RUSSIAN FEDERATION

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Abstract

Rostechnadzor reviews reports of the operating organizations on notifiable events and their annual reports on the status of nuclear and radiation safety at nuclear research installations, which contain data on the conditions of the SSCs important to safety. The evaluation of outcomes is being used by Rostechnadzor for methodical oversight of nuclear research installation safety, including control of efficiency of ageing management programmes in operating organizations. The results of the evaluation of notifiable events in 2010 regarding ageing are given in the paper. The events were classified in terms of type of ageing and item of concern.

1. INTRODUCTION

At present Rostechnadzor, the authority within the regulatory framework for safety of the use of nuclear energy in Russian Federation, conducts state safety regulation and supervision at 70 civil nuclear research installations (NRIs) that are managed by 19 operating organizations governed by different authorities. NRIs are classified as structures and complexes with nuclear research reactors, critical assemblies or subcritical nuclear assembles that have been designed for utilization of neutrons and ionizing radiation for research purposes. In operation are 60 NRIs, among them 23 research reactors, 26 critical assemblies and 11 subcritical assemblies), while 7 NRIs (5 research reactors, 2 critical assemblies) are under decommissioning and 3 NRIs (2 research reactors, 1 subcritical assembly) are under construction. The utilization factor of the powerful research reactors ranges from 0.5–0.9. Data on the state of NRI safety are published in the annual Rostechnadzor report [1].

In accordance with Federal Norms and Rules NP-027-10 [2] operating organizations shall gather, process, analyse, record, and store information related to events and incidents at NRIs during their life cycle. The proper report resulting from the investigation of the events most significant to safety shall be submitted to Rostechnadzor.

Moreover, the operating organizations shall submit to Rostechnadzor annual reports entitled Contents of the Operating Organization's Annual Report on Assessment of the Status of Nuclear and Radiation Safety of Research Nuclear Installations (RB-025-03) on the safety status of each managed NRI in line with the safety regulations. Included in this report are the data on the state of SSCs important to safety, as well as results of identified deficiencies that are not obligatory for notification but were investigated in accordance with internal procedures established by the operating organization.

The operating organization shall classify all operational occurrences in NRIs as indicators of ageing for use of the outcomes to evaluate ageing management programmes. The objective of the paper is to illustrate good practices of regulatory supervision to establish, implement and improve NRI ageing management programmes for enhancement of their efficiency. As an example, an assessment of efficiency of the ageing management programmes of operating organizations is given on the basis of evaluation regarding ageing phenomena of notified events at NRIs of the Russian Federation in 2010.

2. REGULATION OF NRI AGEING MANAGEMENT PROGRAMMES IN THE RUSSIAN FEDERATION

The Basic Codes and Standards for development of ageing management programmes at NRIs to detect, monitor and control ageing include the following documents:

- Requirements of QAPs for NRIs, NP-042-02;
- Rules for Design and Safe Operation of Equipment and Pipelines of Nuclear Power Installations, PNAE G-7-008-89;
- Rules on Design and Safe Operation of Control Rod Actuators, PNAE G-7-013-89;
- Rules for Safe Storage and Transportation of Nuclear Fuel at Nuclear Facilities, NP-061-05;
- Rules to Examine for Compliance of Equipment, Components, Materials, and Half-Finished Products Delivered for Nuclear Facilities, NP-071-06;
- Requirements for Justification of Possible Extension of Assigned Operational Life of Objects Using Nuclear Energy, NP-024-2000;
- Requirements of Content of SARs for Nuclear NRIs, NP-049-03.

There is no special regulatory document in the set of the Federal Norms and Rules in the field of the use of nuclear energy in the Russian Federation that would formulate requirements for NRI ageing management programmes. However, NP-024-2000 establishes the requirement that an operating organization shall develop and implement a management programme concerning the service life of SSCs.

When a nuclear facility's established service life expires, or after a 30 year term, the operator should assess the possibility of a nuclear facility service life extension and perform the following activities:

- A comprehensive survey of the nuclear facility;
- Definition and justification the residual resources of SSCs;
- Preparation of a nuclear facility for operation during the additional service time including:
  - Additional surveys to define residual service life of SSCs;
  - Replacement of overage equipment and, if necessary, modernization and reconstruction of the nuclear facility;
  - Testing of systems or elements of a nuclear facility to prove that they meet their design requirements;
  - Correction of documents justifying safety;

When the facility operating life has not reached the operating time established by design or a period less than 30 years, the oversight for ageing issues is performed within the licensing process. In Rosechnadzor's practice, the license for this kind of activity is issued for a period of 5 years. When the license time expires, the operating organization shall obtain a new license upon providing proper expertise on all issues related to safety, taking into account the complexity and potential risks of the facility. This expertise is needed in case the operator applies to change the license conditions. As part of obtaining a new license, the operating organization shall submit safety justification confirming that systems and elements related to safety meet the criteria of durability, are able to withstand external effects and are functional. In line with requirements of NP-049-03 the safety assessment reports of NRIs shall include information on the frequency and conditions of testing, ISIs and maintenance of the systems and elements important to safety, as well as procedures to examine the state of the NRI vessel, internals and primary circuit as a whole.

In addition, the operating organization shall submit a yearly report to Rostechnadzor assessing the status of nuclear and radiation safety of an NRI. In line with the safety guide Contents of the Operating Organization's Annual Report on Assessment of the Status of Nuclear and Radiation Safety of Research Nuclear Installations, RB-025-03, this report contains data on the state of vital safety systems and elements such as:

- Data on the technical state and lifetime indicators of the actuators and instrumentation of the protection and control systems, special cranes and hoists;
- Information on performed periodic ISIs, scheduled examinations and repairs of the SSCs important to safety;
- Information on replacement of the equipment with expired service life;
- Data on fast neutron fluency exposure on the main elements of a research reactor;
- Results of assessment or ISI of the state of metal and main welded joints of a research reactor.

The outcomes of the reviews of annual reports are used by Rostechnadzor for verification of fulfilment of licence conditions and corrective actions taken by operating organizations in line with ageing programmes.

3. OUTCOMES OF ANALYSIS OF REPORTABLE EVENTS AT NRIs OF THE RUSSIAN FEDERATION IN 2010

In 2010 10 notified events and incidents at NRIs were submitted to Rostechnadzor for review:

- Six incidents due to deficiencies in control and protection system of an NRI;
- Four incidents due to deficiencies in external power supply systems of an NRI.

The outcomes of incident analysis for 2010 performed in line with coding recommended by the IAEA [3] are represented in Table 1.

TABLE 1. DISTRIBUTION OF REPORTABLE EVENTS IN 2010 RELATED TO AGEING MECHANISM AT NRIs

No.	NRI	Thermal output (MW)	Beginning of operation	Code	System [4.1]	System [5.1]	Total number of events at NRIs
1	MIR.M1	100	1966	RU-0013	K+K	M	3
2	BOR-60	60	1969	RU-0027	D	-	1
3	IRT-T	6	1967	RU-0014	-	K	1
4	WWR-M	18	1959	RU-0008	-	M	1
5	WWR-TZ	15	1964	RU-0019	K+ K	-	2
6	IVV-2M	15	1966	RU-0010	D	K	2
Total number of events					6	4	10

Ageing mechanism codes:

- D: mechanical displacement, fatigue or wear from thermal cycling, flow induced vibration, or other cycling loads;
- K: obsolescence or technology change;
- M: other (should somehow have a time dependent attribute).

System codes:

- [4.1]: reactor protection, including secondary shutdown;
- [5.1]: main power supply.

The types of ageing phenomena of SSCs important to safety have been identified as a result of analysis of reports about incident investigations [4, 5]:

- Four deficiencies due to physical degradation of equipment, types of ageing mechanisms D and M, which demand improvement of maintenance and replacement of worn-out components;
- Six deficiencies, type of ageing mechanism K, due to lack of safety design, manufacture and mounting of equipment, i.e. non-physical ageing effects.

4. BRIEF DESCRIPITON OF REPORTED EVENTS IN 2010

4.1. Incidents in protection systems

4.1.1. IVV-2M reactor

IVV-2M is a pool type research reactor with light water moderation and cooling, beryllium as the reflector and HEU fuel. In the protection system the absorber rods through extension rods are electromagnetically coupled to the drive mechanism. The logic unit provides galvanic isolation and transformation of defence signals to drive the mechanism of the control rods and annunciators. The design and mounting of the logic unit as well as cyclic startups and shutdowns of the reactor have an effect on the service life of the relay contacts included in the electrical circuit of the electromagnetic couplings of the control rod drive mechanisms. Disturbance of the electrical contacts of a relay results in de-energization of the electromagnetic couplings and, as a result, a reactor scram.

On 3 February 2010 a reactor scram occurred due to the drop of a safety rod. The event investigation discovered degradation of an electrical contact in the relay of the logic unit.

Corrective measures were more attention to timely maintenance and replacement of elements close to being worn out and the development of a design for protection system modernization.

4.1.2. BOR-60

BOR-60 is a sodium cooled fast reactor. Its protection system provides the following automatic regimes: fast safety protection (BAZ) and slow safety protection (MAZ). During MAZ the control rods are inserted into the reactor core with a determined speed. MAZ is initiated as a result of de-energization of the appropriate relay in the protection system.

On 5 February 2010 the MAZ regime was initiated. During the investigation of the incident it was determined that the spurious control signal was initiated due to failure of the relay in the control circuit. The short circuit in the relay induction coil happened as a result of deterioration of wire insulation.

Corrective measures were replacement of the broken relay and an unscheduled examination of performance attributes of all relays in the circuits BAZ and MAZ.

4.1.3. WWR-TZ

WWR-TZ is a tank type light water reactor with HEU fuel. For redundancy considerations three safety channels provide control and protection during the reactor period. The performance of all units of each channel excluding the detector is automatically checked. In case of failure of any unit of the electronic equipment a proper safety signal is initiated by the channel. Coincidence of two of three signals initiates a reactor scram.

On 24 March 2010 a reactor scram occurred due to the coincidence of two of three signals related to electronic equipment failure in the safety channels during reactor operation. The event investigation discovered the failure of two high voltage power supply units of the detectors in different channels in addition to a broken fuse box in one unit and a broken indicator tube in the other unit.

Corrective measures taken were the examination of the high voltage power supply units of the detectors in all safety channels during preventative maintenance.

As background for a second incident, one of the safety signals that initiates a reactor scram is a signal of excessively increasing coolant temperature at the inlet of the reactor core. Failure of the unit of the temperature control leads to reactor scram.

On 29 April 2010 a reactor scram occurred due to failure of the device of temperature control. The event investigation discovered a failure of the relay contacts set for the device of temperature control due to insufficient tension between the contacts included in the circuit of the safety signal.

Corrective measures after the campaign's completion and reactor shutdown were the unplanned fixing of tension between the contacts in all thermal–technical devices that initiate safety signals in the protection system as well as the systematic replacement of aged I&C devices.

4.1.4. MIR-M1

This reactor is a channel type HEU fuelled research reactor immersed in a light water pool and provided with a beryllium reflector. Each experimental loop, which typically supports material science experiments, includes 1–2 fuel channels. Some parameters important to safety of the experimental loops can initiate signals for reactor scram.

One of the safety signals inducing reactor scram is a signal of excessively increasing coolant temperature at the outlet of the loop channel. On 6 May 2010 reactor scram occurred due to failure of the annunciator for coolant temperature at the outlet of the channel of the water loop. The event investigation discovered failure of the relay coil in the annunciator. The failure of the annunciator for coolant temperature at the outlet of the channel is a repeated event.

Corrective measures were replacement of the defective device and that of the reserve circuit and implementation of a developed design for modernization of the control system of loop parameters, which included implementation of a third control channel and coincident logic for reactor scram concerning critical parameters of the loop.

Likewise a signal of excessively decreasing coolant flow through the channel of the loop can signal a reactor scram. The measuring channel includes impulse lines connected to a differential manometer through screw joints. The integrity of the screw joint depends on the quality of the joint gasket. Ageing of the joint gasket materials may lead to integrity loss, spurious readings and initiation of spurious safety signal for decreasing coolant flow and reactor scram.

On 5 September 2010 a reactor scram was initiated by a safety signal for excessively decreasing coolant flow through the channel of the loop. The event investigation discovered coolant leakage through the joint of the positive impulse line with the differential manometer due to the loss of joint gasket integrity.

The corrective measure was replacement of the joint gasket in the screw joint.

4.2. Incidents in power supply systems

Safety assessments of NRIs during the licensing and target inspection of power supply systems at sites of research centres have indicated that the state of main, reserve and emergency power supply systems is in line with the requirements of effective federal regulations. The loss of external sources of power supply does not lead to violation of operation limits and conditions at NRIs, as reactor cooling is provided by means of natural circulation. However, the loss of external sources of power supply has an impact on the stability of NRI operation and may lead to an outage of its experimental facilities.

The major causes of failures in power supply systems in the frame of the responsibility of an operating organization are ageing of the main electrical equipment and personnel errors.

4.2.1. WWR-M

WWR-M is an open tank HEU fuelled reactor with light water moderation and cooling and a beryllium reflector. On 21 May 2010 a reactor scram occurred from the total loss of its external power supply. The failure of an exterior electrical system in the neighbouring city resulted in a short circuit in the electricity supply network.

Corrective measures were taken by the organization that serves the municipal electricity supply network.

4.2.2. IRT-T

IRT-T is a pool type HEU fuelled reactor that is also provided with light water moderation and cooling and a beryllium reflector. On 25 May 2010 an undervoltage condition in the 10 kV line at the operating organization's electric power substation 35:10 kV transformer resulted in a reactor scram.

Corrective measures were the replacement of switchgear at the electric power substation and plans for modernization of the power supply system.

4.2.3. MIR-M1

On 27 June 2010 a reactor scram occurred due to an undervoltage condition in the 6 kV line of the main power supply system of the research centre site that disabled the 110 kV power transmission line in the external power supply system.

The cause of failure was damage to the single phase 110 kV voltage step-down transformer due to extremely high outdoor air temperature (58°C) and a hidden fault in the isolation of the phase winding high voltage circuit. By an automatic shielding function the 110 kV line deactivated.

The corrective measure was to modernize the power supply system at the research centre site.

4.2.4. IVV-2M

On 22 September 2010 an undervoltage condition in the external power supply system caused a reactor scram. The corrective measure again was to plan the modernization of the power supply system at research centre site.

All the events mentioned above are repeat events that show the necessity of more careful review of modernization plans of the power supply systems of research sites that include the provision of an uninterruptible power supply.

Partial modernization of power supply systems at research centres is being resolved in the framework of the Programme of Long Term Activity of the State Corporation Rosatom in accordance with the federal target programme Providing Nuclear and Radiation Safety from 2008 until 2015.

5. SUMMARY

- In 2010, ten events and incidents requiring notification happened at nuclear research reactors managed by 5 operating organizations, but no events were reported by critical and subcritical stands.
- The proportion of repeated events was still high in 2010 and made up 50% (2009: 69%, 2008: 80%). Most of the repeated events can be characterized by undervoltage conditions in lines or the loss of the external power supply. To enhance the stability of operation of NRIs and experimental facilities, the modernization of site power supply is desirable, including the introduction of an uninterruptible power supply to all systems important to safety.
- Event analysis since 1999 declared a tendency of a decrease in the total number of events at Russian NRIs, but increasingly events have resulted from the ageing of I&C equipment.

6. RECOMMENDATIONS

With regard to ageing mechanisms the following specific data should be systemized to benefit regulatory decision making in the case of license extensions for NRIs:

- Clarification of the mechanism of corrosion damage to the steel reinforcement of concrete items and the impact of concrete wear on seismic stability of buildings and structures;
- Information from the assessment of the impact of cyclic loadings typical for research reactors, e.g. reactor shutdown for core refuelling and cyclic variation of power during experiments, on the ageing rate of structural materials;
- Information from the assessment of the impact of long term shutdown periods on the ageing processes of SSCs.

Information on the ageing mechanisms of idle equipment would be beneficial to assess the lifetime extension of critical assemblies that may remain preserved for a few years.

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AN AGEING MANAGEMENT PROGRAMME FOR THE SAFARI-1 RESEARCH REACTOR

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Abstract

The SAFARI-1 Research Reactor is a 20 MW high flux material test reactor and has been continuously operational for nearly 47 years. In this period, ageing of the facility has been addressed by means of various, mainly reactive, maintenance and upgrade initiatives to replace components that have become unmaintainable for various reasons such as wear, corrosion and obsolescence. With the facility now approaching 50 years of continuous operation, a programme has been implemented to assess, address and implement ageing management in a more formal and proactive manner. The programme conforms to the recently published IAEA Safety Guide SSG-10 Ageing Management for Research Reactors and makes extensive use of the guidelines set therein as well as other tools and methods developed in various international meetings and workshops at the IAEA for identifying ageing issues at the facility. The paper presents an outline of the methodology for implementing the ageing management programme at the SAFARI-1 research reactor. Identification of SSCs important to the safety and sustainability of the facility that are susceptible to ageing, the ageing mechanisms affecting them and the remedial actions identified to mitigate or remove the effects of ageing are discussed. A methodology for determining priorities is also elaborated. Remedial actions are divided into four groups: safety critical, mission critical, lifetime extension and organizational, and an implementation strategy is described.

1. INTRODUCTION

The SAFARI-1 research reactor is a high flux material test reactor and has been continuously operational since March 1965, i.e. for nearly 47 years. It is licensed for a power of 20 MW. Between 1976 and 1992 it operated at a maximum of 5 MW for only 5 days per week. Currently, however, it operates for, on average, 303 full power days per year at 20 MW. Throughout its operational history, ageing of the facility has been addressed by means of various, mainly reactive, maintenance and upgrade initiatives to replace components that have become unmaintainable for various reasons such as wear, corrosion and obsolescence. With the facility now approaching 50 years of continuous operation, a programme has been implemented to assess, address and carry out ageing management in a more formal and proactive manner.

The ageing management programme is implemented at a high level through the SAFARI-1 integrated management system, which is subject to periodic review and audit. It conforms to IAEA Safety Guide SSG-10 [1] and makes extensive use of the guidelines set therein and other tools and methods developed in various international meetings and workshops at the IAEA for identifying ageing issues at the facility. A series of workshops were held at the facility in which expertise and experience from the Operations, Maintenance, Engineering, Utilisation and Quality Sections at the facility, as well as from the NECSA Licensing Department and other support groups, were combined to perform a detailed ageing assessment and to formulate remedial actions to address ageing issues. A remedial action is formulated as an action, a small project, a major project or multiple projects addressing one or more ageing related issues. The paper does not separate the remedial actions into their component projects.

The design basis for the facility contains no information relating to a pre-designated or intended design lifetime for the facility, but an assessment of, for example, the effect of neutron fluence on the fixed core structure based on current operation confirms that the soundness of this structure is easily predictable until about 2020 based on actual experience and reported data of other facilities. This analysis then forms the basis for an operating threshold for the reactor. The result of a remedial action addressing this issue may change the prediction significantly, e.g. by adding a decade or so to the projection, but for the purpose of this ageing management programme, the above date presents a convenient threshold extrapolated to the rest of the facility separating the concept of a current lifetime from that of lifetime extension. Operation of the reactor beyond the currently projected end of life may follow one of two paths:

- Extension of the lifetime of the facility by a few years to a decade or so;
- Complete rejuvenation of the facility during the 2020s to support operation for another 40–60 years.

The remedial actions identified in the present assessment to address lifetime extension focus mainly on the former, as the latter possibility is expected to entail substantial redesign of the facility and, in all likelihood, an extended shutdown of two or more years to implement and commission the modernised facility. Many of the remedial actions, especially those involving instrumentation and information technology, involve technologies with a maximum life span of 10–15 years; hence the targeted lifetime extension of the current ageing management programme is 2030. Operation beyond that time will need to be reviewed at a later date.

2. EFFECT OF AGEING DUE TO SERVICE CONDITIONS

One of the basic causes of ageing degradation of a SSC are the service conditions that support the activation of particular ageing mechanisms leading, unless properly managed, to loss of SSC functionality[1]. These service conditions can be categorized as normal operation, abnormal operation and environmental conditions and include such effects as wear and tear, corrosion, erosion, chemical changes and physical damage due to operational incidents.

The effects of ageing, i.e. the probable consequence or failure, for each condition under these three categories with the associated ageing mechanism are summarised in Tables 1–3.

A further ageing mechanism identified in many SSCs at SAFARI-1 is technological obsolescence, in which either whole technologies, e.g. vacuum tubes, become obsolete or suppliers and vendors of equipment simply no longer exist or have discontinued certain products to the point that even their support is discontinued and spares are not available. The latter affects electronics and instrumentation most readily, but it has also, in the past, afflicted such unlikely equipment as primary pumps and cooling towers.

TABLE 1. EFFECT OF AGEING FOR SEVERAL NORMAL SERVICE CONDITIONS

Condition	Ageing mechanism	Consequence/failure
Radiation	Change of properties	Chemical decomposition; Strength change; Ductility change; Swelling; Resistivity change; Burnup

Condition	Ageing mechanism	Consequence/failure
Cycling of temperature, flow or load; Flow induced vibrations	Motion	Displacement; Change of position or setpoint; Loose connections
	Fatigue	Break or collapse; Deformation
	Wear	Deterioration of surface; Change of dimensions

Temperature	Change of properties	Strength change; Resistivity change; Ductility change	Flow	Erosion	Strength change
			Fluids chemistry	Corrosion/ galvanic cells	Release of radioactivity; Strength reduction; Deposition of particles; Short circuits; Leakage
Stress (pressure)	Creep	Changes of geometry; Stress corrosion cracking	Technological	Obsolescence	Ineffective maintenance

TABLE 2. EFFECT OF AGEING FOR SOME ABNORMAL OPERATIONAL CONDITIONS

Condition	Ageing mechanism	Consequence or failure
Power excursion	Thermal and mechanical damage	Deterioration of systems; Accelerated ageing
Flooding	Deposition and chemical contamination	Corrosion; Blockages; Reduction of strength
Fire	Heat, smoke, reactive gases	Reduction of strength; Loss of insulation; Corrosion

TABLE 3.EFFECT OF AGEING FOR SEVERAL ENVIRONMENTAL CONDITIONS

Condition	Ageing mechanism	Consequence or failure
Humidity, salinity	Corrosion/galvanic cells	Leakage; Release of radioactive material; Strength reduction; Deposition of particles; Short circuits
Chemical agents	Chemical reactions	Undesirable chemical product; Deterioration of structures
Wind, dust, sand	Erosion and deposition	Strength change; Deterioration of surface; Malfunction of components

3. SAFARI-1 AGEING ASSESSMENT

The ageing assessment of the SAFARI-1 facility was facilitated by a generic spreadsheet based matrix developed during the same IAEA workshops that resulted in the publication of SSG-10, with minor adjustments to fit the specifics of the reactor and its components. The matrix gives a comprehensive set of SSCs in its vertical axis and an equally comprehensive set of ageing mechanisms in its horizontal axis, which will be seen to correspond with those in the IAEA Research Reactor Ageing Database [2]. In order to ensure a consistent approach to all potential ageing issues by a large and divergent group of specialists over the extended period of time taken to conduct the ageing assessment workshops, the ageing mechanisms must be clearly defined. The clarifications used are listed in Table 4.

The completed assessment matrix for SAFARI-1 is shown in Table 5, in which the alphanumeric indicators in non-blank cells indicate identified real or potential ageing issues requiring some form of remedial action. In many cases the remedial action is to perform an assessment to determine whether there is an incipient problem requiring a proactive solution, while in other cases the remedial action addresses a known or existing problem. The complete list of remedial actions, tagged with the same alphanumeric indicator as appears in Table 5, is given in Table 8.

The remedial actions formulated to address the identified real or potential ageing issues are divided into four distinct groups of ageing management namely:

(1) Safety critical: remedial actions without which the reactor will probably not be able to operate safely until the currently projected end of life, not considering the purpose of operating the reactor;

(2) Mission critical: remedial actions without which the reactor will be safely operated until end of life, but reliability or availability may be compromised;

(3) Lifetime extension: Remedial actions required for lifetime extension of the facility;

(4) Organizational: safety, health, environment, human resources and quality issues requiring attention to maintain and improve the safety management and culture of the facility.

TABLE 4. CLARIFICATION OF AGEING MECHANISMS

Mechanism		Description
A	Radiation: change of properties	Neutron and other radiation damage. Generally well known and predictable phenomena, for which studies and data are fairly widely available.
B	Temperature: change of properties	Affects many synthetic materials, electronic circuits and sensors, cables and wiring, electric motors, transformers etc. and also concrete subjected to heat deposition. Also consider effects of historical fire events in the facility (also see J).
C	Creep due to stress and pressure	Typical examples are core components subject to the effects of A, e.g. Be reflector, Graphite components, even fuel elements that are loaded or stored for very long periods in some reactors.
D	Mechanical displacement: fatigue or wear from vibration, cyclic loads	Routine loosening and fastening of bolts, periodic repair of breakages (e.g. re-tapping of threads, re-welding etc), changes due to operating modes, general wear and tear.
E	Material deposition (e.g. crud)	Particularly in inaccessible places such as regions below the reactor core, the decay tank, BP front (core-side) chambers, experiment penetrations and cavities in the pool structure (e.g. in the space below pool gates).
F	Flow induced erosion	Shouldn't affect the normal flow paths of most RRs, but look e.g. for erosion of flow measuring orifices (dulling of edges) that can affect accuracy. Erosion of concrete can occur in the biological shield and other concrete structures due to pool leaks.
G	Corrosion	This is by far the biggest contributor to the record of ageing in RRs, and is not limited to old facilities. Look particularly for corrosion on the concrete side of embedded pipes, components, re-enforcing, etc. – especially if the pool has been leaking. Note that stainless steel is not immune to corrosion! There are many instances of (e.g.) incorrect welding procedures or inadequate protection against galvanic action that have promoted rapid corrosion of SS components. Look also at electronic and instrumentation components, where corrosion can lead to imperfect connections.
H	Damage due to power excursions, operational events	A single abnormal event or accident may cause permanent damage. Look also for historical handling errors and accidents causing mechanical damage. Ever dropped something heavy into the pool? Or into the reactor core? Or onto an elevated concrete floor?

Mechanism		Description
I	Flooding: deposition; chemical contamination	Both internal and external flooding. The latter can cause erosion around foundations etc. (see also F). Chemical contamination can occur in demineraliser plant, pool liners (e.g. Hg/Al reactions) and may lead to corrosion (see also G)
J	Fire: effects of heat, smoke, reactive gases	Both internal and external fires (induction of smoke and gases by the ventilation systems).
K	Obsolescence and technology change	This affects practically all aspects, especially of old facilities (design, mechanical, electrical, instrumentation, documentation, staff etc.). However, even new facilities have reported rapid obsolescence and loss of support from vendors due to discontinuation of products etc. Also look at as-built status of drawings.
L	Changes in requirements or acceptable standards	This is typically applicable to regulatory requirements. Codes and standards (including IAEA Safety Series docs for RRs) also evolve with time and facilities' documentation and safety cases gradually become contextually out-dated. Furthermore, the operational focus of many RRs today is far removed from their original design intent.
M	Other, e.g. time dependent phenomena	Typical aspects considered under this mechanism are incorrect or defective control over design or over installation – both during the original construction and during modifications or upgrades. This may include SSCs not to specification or at an unacceptable quality, which may result in a much shorter lifetime

TABLE 5. SAFARI-1 RESEARCH REACTOR AGEING MANGEMENT ASSESSMENT

SSCs Relevant to Safety and Sustainability.		Ageing Mechanisms											
		A	B	C	D	E	F	G	H	I	J	K	L
1. Reactor block, fuel and internals													
1.1	Fuel assemblies including control rod followers										1a,b	1a,b	
1.2	Fuel storage												
1.3	Removable core components other than reflector and fuel	1c		1c			1c	1c					
1.4	Reflector	1d	1d	1d				1d					
1.5	Reactor tank and vessel including fixed core structure and components	1e		1e									
1.6	Pool liner, pool gates and jambs, etc.												
1.7	Pool structure (Concrete structure excluding biological shield)					1f	1f		1f				1f
1.8	Beam tubes	1e											
1.9	Control rods, shutdown rods, drive mechanisms												
1.10	Biological shield					1f	1f		1f				1f
2. Cooling systems													
2.1	Reactor primary cooling system			2g	2c		2f,c				2a		
2.2	Pool primary cooling system				2c		2f				2b		
2.3	Emergency cooling system						2f						
2.6	Purification												
2.8	Secondary and Tertiary cooling			2d	2g		2d,e				2e		

SSCs Relevant to Safety and Sustainability.		Ageing Mechanisms											
		A	B	C	D	E	F	G	H	I	J	K	L
3. Confinement and containment													
3.1	Structure			3a			3b		3b				
3.3	Ventilation			3c			3c				3c,d		
3.4	Emergency ventilation			3c			3c				3c,d		
3.6	Penetrations			3e			3e						3e
3.8	Exhaust and stack			3a			3b		3b				
4. Instrumentation and controls													
4.1	Reactor protection including second shutdown	4a		4a		4h	4a,h				4a,h	4c	
4.3	Operational supervision and control	4b		4b,h		4h	4b,h				4b,h		
4.4	Radiation monitoring										4f,g	4f,g	
4.5	Control console			4e			4e				4e	4e	
4.6	Annunciators			4i							4i		
4.7	Data acquisition										4d	4d	
4.9	Signal cabling and routing	4a,b					4h		4h		4c,h	4c,h	
4.10	Remote shutdown and monitoring											4j	
5. Power supply													
5.1	Main power supply			5a			5a	5a	5a	5a	5a	5a	
5.2	Emergency power supply			5b			5b		5b		5b	5c	
5.3	Power distribution, cabling and routing			5a,b			5a,b		5a,b		5a,b	5a,b,4c	
6. Auxiliaries													
6.1	Fire protection										6b	6b	
6.2	Lightning protection and grounding						6a				6a	6a	
6.4	Communications and alarms										6c		6c
6.5	Compressed air			6h	6h		6h				6h		
6.6	Crane			6d,e							6d,e	6d,e	
6.7	Handling and storage facilities			6l	6l		6l	6l				6l	
6.8	Hot cells						6f				6f,g	6f,g	
6.12	Clean waste handling, storage and disposal											6m	
6.13	Radioactive waste handling, storage and disposal				6i,j,k		6i,j,k				6i,j,k	6i,j,k	
7. Experimental facilities													
7.1	Beam tube lines (pool), shutters						7a						

SSCs Relevant to Safety and Sustainability.		Ageing Mechanisms											
		A	B	C	D	E	F	G	H	I	J	K	L
7.3	Rabbit and conveyer			7b								7b	
7.10	Dry irradiation rooms						7c				7c	7c	
8. Documentation and configuration management													
8.1	Licensing and commitment tracking											8a	8a
8.2	SAR										8b	8b	8b
8.3	Operating limits and conditions											8c	
8.4	Design including as-built drawings and descriptions												8c
8.5	Management system											8d	8d
8.6	Operation and maintenance manuals and procedures											8e	8e
8.8	Reviews and assessments											8f	
9. Other (non-SSC)													
9.1	Staff and staff training												8a, 9a
9.2	Training facilities												9b
9.3	Industrial safety											9c,d	9c

4. PRIORITIZING

A simple mathematical model was developed to aid in the objective prioritisation of remedial actions and for application throughout the process. The methodology followed allocates a score to each remedial action for each impact factor given in Table 6. The sum of these scores is then adjusted by multiplication with a supplementary weighting factor on a scale of 1–10 based on the answers to the questions raised in Table 7. The weighting factor for each remedial action is a blend of the considerations given in Table 7, and the final scores determine the priorities of the remedial actions in relation to one another.

TABLE 6. IMPACT FACTORS AND SCORE

An ageing issue leads to one or more of the outcomes listed below:	High priority 10	Medium 5	Low 0
Non-availability of: • Reactor • Backup system • Production facility • Site services (power, water, air)	 >1 week >2 weeks >1 week >2 weeks	 <1 week <2 weeks <1 week <2 weeks	 <1 day <1 day <1 day <1 day
Reportable nuclear event	High category	Intermediate category	Low category
Radiological exposure	Greater than allowable annual dose or any dose to public and environment	Greater than allowable daily dose or any dose to site personnel	Dose to staff not as low as reasonably achievable
Injury, radiological or conventional (anyone)	Disabling injury >30 days, permanent disablement, fatality	Injury <30 days	Minor or none

An ageing issue leads to one or more of the outcomes listed below:	High priority 10	Medium 5	Low 0
Environmental releases	Uncontrolled or uncontained (>investigation or intervention levels and >FAADQ)[1]	Uncontrolled but contained (>investigation or intervention levels but <FAADQ)	Controlled (>investigation levels but <intervention levels and <FAADQ)
Licence complication	Regulatory intervention	Regulatory tightening	*Not applicable*
Lifetime limitations	Premature permanent shutdown of the facility prior to intended lifetime	Obstacle to meeting intended lifetime. Obstacle to lifetime extension beyond intended lifetime	*Not applicable*
Public non-acceptance	Strong public demand to shut down	Negative media coverage; questions or doubts by a local community or institution	Management questions or doubts
Stakeholder non-acceptance (Operating organization, government, regulator, customers, nuclear industry)	Loss of support or strong stakeholder action to stop reactor operation	Stakeholder intervention (e.g. by imposition of strong financial or operational constraints)	Stakeholder and or customer complaint

TABLE 7. WEIGHTING FACTOR

Questions for each remedial action	If "Yes", the weighting factor should be
Is the issue easily monitored, controlled, evaluated?	Low (<5)
Is the remedial action easily implemented?	Med–high (5–8)
Is it viable, given the age of the plant as a whole?	Med–high (5–8)
Is the expected cost of implementation compatible with the anticipated return?	Med–high (5–8)
Is it urgent for some reason other than considered in the impact table?	High (8–10)

To follow a particular example, consider remedial action 1d "replace beryllium reflector elements", identified in the ageing assessment matrix against SSC 1.4 "reflector" with identified real or potential issues for ageing mechanisms A, C, D and H (see Table 4). The priority assessment identifies the following impacts:

(1)	Non-availability	10
(2)	Reportable nuclear event	0
(3)	Radiological exposure?	0
(4)	Injury to anyone?	0
(5)	Environmental releases?	0
(6)	Licence complication?	5
(7)	Lifetime limitations?	10
(8)	Public non-acceptance?	0
(9)	Stakeholder non-acceptance?	5
	Score:	30

The weighting factor gives a value of 8, based on the fairly easy implementation of Be replacement, albeit at a relatively high cost.

[1] *Facility Annual Authorised Discharge Quantity*

Thus, the total priority score for a remedial action to replace the beryllium reflector is 30×8=240. This value can be seen in the right hand column of Table 8.

Applying the same process to all the remedial actions produces a range of scores from 40 at the low end to 400 at maximum. The resulting priorities need a further assessment at management and stakeholder level to determine the relative urgency of the remedial actions and to cast them into an overall action plan to fit available resources.

5. LISTING OF REMEDIAL ACTIONS

The list contained in Table 8 is sorted by group and priority. The first column contains the alphanumeric label used in Table 5. The remedial actions appear as short, concise statements without providing details of the scope, cost and other resource requirements. These aspects are not addressed in the paper. The assessment identified 57 remedial actions that are expected to devolve into some 80 separate projects, requiring more human resources than has nominally been available to the facility and additional capital funding. A remedial action, 9a — see under organizational ageing management in Table 8, is identified to specifically address the human resource requirements for the programme, while funding is reasonably assured by the high profile of the programme and the high levels of commitment by the main stakeholders.

TABLE 8. REMEDIAL ACTION LIST

RA No	Remedial action description	Priority
	Safety critical ageing management	
4a	Upgrade (modernize) safety critical neutron and gamma detectors and instrumentation	400
4c	Safety critical instrumentation segregation and separation of routing	280
4j	Implement remote shutdown of reactor and selected plant; also investigate dedicated emergency control room	250
3e	Evaluate, and where necessary, redesign and re-implement all confinement penetrations such as cables, pipes and access doors	200
8f	Overall review (10 yearly), assessment by IAEA INSARR	200
	Mission critical ageing management	
6e	Refurbish and replace reactor hall crane	400
4f	Upgrade (modernize) radiation and contamination monitoring including fission product monitor	400
4g	Upgrade (modernize) ventilation stack monitors and data transmission	400
2f	Robotic eddy current ISI of built-in reactor and pool primary pipes and emergency spray nozzle	400
1a	Convert to LEU fuel	350
1b	Convert to LEU targets	350
3d	Implement standard charcoal ventilation filter efficiency and effectiveness measurement capability	350
6g	Investigate the adequacy of the hot cell ventilation filters for an expanded commercial programme and upgrade	350
8c	Re-evaluate the basic design package and safety of the reactor and all experimental facilities	350
5a	Refurbish electrical supply transformers, switchgear and reticulation	320
6a	Refurbish grounding and lightning protection; investigate power stabilization	300
8b	Upgrade SAR	300

RA No	Remedial action description	Priority
6k	Re-conceptualise, redesign active liquid waste storage and disposal	300
2g	Manufacture spare primary heat exchanger, then refurbish the others in rotation	250
6d	Install auxiliary crane in reactor hall	250
1d	Replace beryllium reflector elements	240
4b	Upgrade control instrumentation, e.g. nuclear and process controllers, rod drop monitor	230
2b	Replace pool primary switchgear	160
7b	Upgrade rabbit systems; refurbish worn equipment and control systems and re-establish capacity	150
7c	Inspect dry gamma facility; re-establish capability	150
2a	Replace reactor primary pumps, motors and switchgear	120
Lifetime extension ageing management		
1e	Assess reactor vessel lifetime and make recommendation for enhanced surveillance and replacement	400
5b	Refurbish electrical equipment, generators, power supplies, batteries, switchgear and reticulation	400
3c	Modernize and rationalize ventilation logic, machinery and control and switchgear	400
3a	Assess reactor building structure and stack integrity; determine actions needed for lifetime extension of facility.	350
4h	Replace, upgrade and modernize process I&C systems including isolation of the control room	350
4e	Refurbish control room	350
6b	Investigate effectiveness of fire protection system	300
4i	Replace and upgrade annunciator electronics and logic units	300
6f	Evaluate condition of hot cell liners and internal equipment	300
6h	Redesign and implement compressed air supply system including standby and emergency compressor systems	300
6i	Re-conceptualize, redesign compressible waste handling areas and equipment	300
7a	Rehabilitate beam tubes as required	300
2c	Inspect, clean and flush reactor, pool decay tanks and twin storage tanks	250
5c	Redesign and rationalise uninterruptable power supply consumer loadings	250
3b	Inspect, clean, seal and coat all surfaces of reactor building and stack exposed to weather or flooding.	200
4d	Implement industrial network and supervisory control and data administration upgrade	200
6j	Re-conceptualize, redesign active solid waste interim storage for in-pool and dry storage	200
1c	Replace core support grid and control rod bearings	150
1f	Assess condition of pools and civil structures and implement stabilisation if necessary	140
2d	Replace secondary pumps, pipes and in-line components	100
2e	Refurbish or build new cooling towers	100
6c	Modernize communication system with particular attention to cabling from control room to outside	100
9d	Asbestos removal programme	50
Organizational ageing management		
8e	Review, revise and update procedures	300
6l	Organise storage areas and control measures for critical spares, handling tools, maintenance equipment, etc.	300
6m	Organise storage areas and control measures for non-radioactive solid waste and materials, e.g. $ZnBr_2$	300

RA No	Remedial action description	Priority
9a	Develop expertise base by filling vacancies and training suitably qualified and experienced personnel and criteria and procedures to implement	240
9c	Implement industrial safety monitoring and tracking systems, e.g. NOSA, SHEQ, BBS, safety culture	200
8d	Implement an integrated management system compliant with ISO 9001/14001, OSHAS 18001, RD-0034, NQA-1, NS-R-4, SHEQ-INS, etc.	150
8a	Establish human resources commensurate with requirements and proper succession planning	50
9b	Update, translate and modernize training material; expand to cover relevant SAFARI-1 disciplines	40

6. IMPLEMENTATION

There are numerous approaches to the coordinated implementation of an ageing management programme of the scope elaborated in this paper. Many of the remedial actions listed in Table 8 are relatively straight forward and can be implemented under the normal operational and maintenance programmes at the facility. Others are more complex and capital intensive and need to be implemented in a coordinated manner under the ageing management programme. The division of responsibility for implementation of the remedial actions at SAFARI-1 was, however, based mainly on a priority score threshold:

(1) All remedial actions in the safety critical group, irrespective of their priority score, and all remedial actions in the other groups with a priority score equal to or greater than 350 are implemented under a formal ageing management programme [3].

(2) All projects implementing remedial actions with a priority score less than 350, other than in the safety critical group, are implemented by the various internal groups within SAFARI-1 in the form of management, maintenance or refurbishment projects.

7. CONCLUSION

The application of the guidelines provided by the IAEA for the implementation of an ageing management programme at the SAFARI-1 research reactor has produced a comprehensive and highly satisfactory result. Plans are underway at SAFARI-1 to obtain the resources, processes and funds to address the identified ageing issues, and while a number of the remedial actions have already been completed or are close to completion, a great deal of work is still needed in order to fully realize the programme.

It should also be noted that while the IAEA guidelines are sufficient to identify practically all the real or potential issues related to ageing safety, mission, lifetime extension and organisational aspects of a research reactor facility, there needs to be awareness that some unrelated but equally important issues could be missed. A particular omission experienced by SAFARI-1 after completion of the protracted ageing assessment workshops described in Section 3 was the very old building elevator, which provides personnel access to the four operating levels in the reactor hall. A project to replace the elevator was started by the maintenance group independently of the ageing management programme and now needs to be incorporated into the ageing management programme. The reason established for the omission is that the SSC list in Table 5 did not trigger an evaluation of the elevator during the assessment workshops.

To remedy, it is proposed to add to the list of SSCs under “6 — auxiliaries” an SSC labelled “support buildings and services”, which would trigger discussion on such things as the elevator and numerous other aspects such as office accommodation, sanitation, drinking water supplies, air conditioning, interior decoration, lighting and electrical distribution in non-plant areas of the facility.

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REPLACEMENT OF THE CORE BERYLLIUM REFLECTOR IN THE SAFARI-1 RESEARCH REACTOR

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Abstract

The SAFARI-1 Research Reactor is a 20 MW high flux MTR and has been continuously operational for more than 46 years. In this period, the core beryllium reflector had never been replaced. An ageing management action to replace the reflector received priority due to the risks involved with failure or deformation of elements. This paper elaborates on the actions taken to replace the old and manage the new reflector. To this extent a reflector replacement procedure, backed up by core neutronic calculations and a test plan, was developed for the safe replacement of the reflector. A reflector management programme will ensure that records of reflector elements are kept and used to optimally manage usage of every element. Due to the historic nature of reflector utilisation in the SAFARI-1 core, deformation of the elements was unavoidable. These deformations will be monitored in the management programme for the new reflector. Deformation measurement of the old reflector is planned and could yield interesting comparisons with analytical results. The action plan for final disposal of the old reflector, although still in development, is also mentioned in this paper.

1. INTRODUCTION

The SAFARI-1 reactor is located at Pelindaba approximately 30 km west of Pretoria. The tank in pool reactor, similar in design to the Oak Ridge Reactor, is light water cooled and moderated with an 8×9 core lattice that currently contains 26 fuel assemblies (active height 600 mm) and 6 control assemblies. The remaining lattice positions are either aluminium or beryllium reflector elements. A representation of the core is depicted in Figure 1.

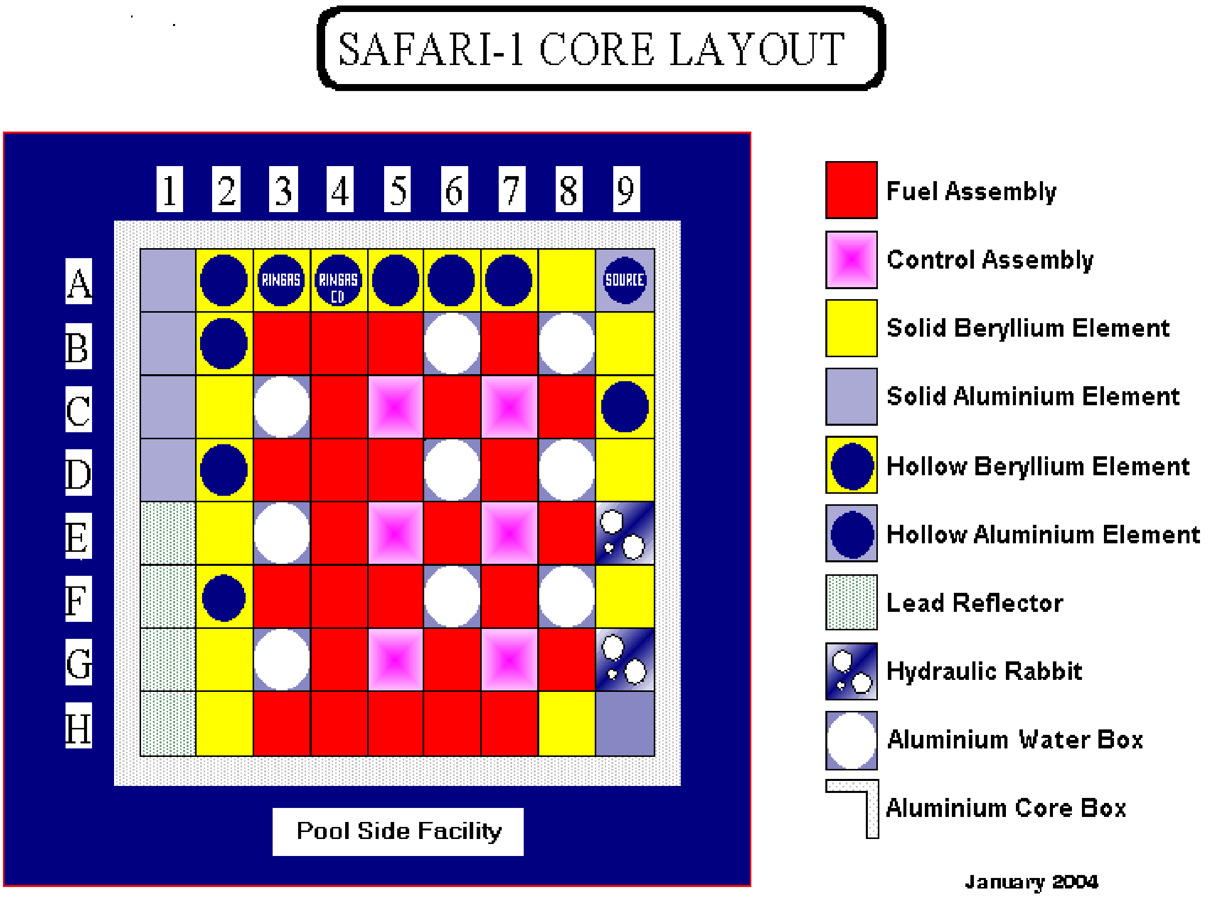

FIG. 1. A schematic representation of the SAFARI-1 core layout.

Beryllium reflectors are used in SAFARI-1 to enhance the neutron flux density. The beryllium reflectors consist of hollow and solid elements and plugs. The SAFARI-1 reflectors are in the process of being replaced as part of an effort to systematically and methodologically counter the effects of ageing degradation of the plant and as part of SAFARI-1's ageing management programme.

Mechanisms that contribute to ageing of SAFARI-1's beryllium reflectors are [1]:

- Changes of the material properties due to neutron irradiation. The good structural characteristics of beryllium are negatively affected by the formation of gaseous atoms of helium and tritium and the displacement of atoms, which form voids in the material that can combine into hairline cracks [2]. This also negatively impacts the ductility of beryllium;
- Creep or stresses due to pressure buildup of gaseous atoms. This can contribute to bowing or swelling of material;
- Damage due to operational events.

Actively managing ageing reflector elements will ensure safe and effective utilisation of the reflector through:

- Effective and appropriate actions and practices for managing ageing that provide for timely detection and mitigation of ageing effects in the reflectors;
- Indicators of the effectiveness of the strategy.

Replacement of beryllium reflectors in the SAFARI-1 core received a medium priority in the SAFARI-1 ageing management programme prioritisation assessment [3]. Notwithstanding this, the project received a much higher priority after real appreciation of the geometrical condition of the SAFARI-1 reflector emerged later when problems were experienced with core loading in late 2010. Fast fluences higher than 6.4×10^{22} cm^{-2} bring about accelerated swelling behaviour [4]. Visual observation depicted bowing of certain elements that occurred in the SAFARI-1 core where a fast fluence greater than 2×10^{22} cm^{-2} was reached. Beryllium irradiated to a fluence of 10^{21} cm^{-2} and tested at temperature below 100°C exhibits increased yield strength and nil ductility [5].

Figure 2 below is a snapshot from visual inspection in the SAFARI-1 core. In Figure 2 it is noticeable that the two elements fully visible have a varying gap between them that could indicate some bowing may have taken place.

FIG. 2. Snapshot of possible bowing of reflectors taken November 2010.

2. THEORETICAL ASSESSMENT

A neutronic evaluation was done to address the expected accumulated fast fluence in order to evaluate the possibility of swelling and bowing of reflector elements.

Accumulated fast fluences were calculated for the SAFARI-1 core, based on mostly a representative core indicating that fluences of approximately 0.6–3×10^{22} cm^{-2} could be expected at various localized sections in the reflectors. It is reported that the High Flux Reactor, Petten, Netherlands replaced their reflectors after a fast fluence of approximately 5×10^{22} cm^{-2} [6]. It must be mentioned that their decision to replace was based on operational experiences such as handling problems.

The representative core was defined as the core that has the highest fast flux in the beryllium elements. During the course of this evaluation, it came to attention that the (γ,n) reaction in the beryllium reflectors may lead to an approximately 5% higher fast flux then normally predicted by software codes [7]. Therefore, the fast flux was adjusted accordingly in order to more correctly predict the swell behaviour and poison buildup in the beryllium.

The swelling behaviour of beryllium for irradiations at temperatures below 75°C as a function of fast fluence (E_n>1 MeV) can be determined by the following equation [4]:

$$\Delta L/L=0.00185(\Phi_t) \qquad (1)$$

in which Φ_t is the fast fluence in units of 10^{22} cm^{-2}, and L is the dimension of the reflector, as indicated in Table 2 below.

This formula can be used to predict the swelling for fluences less than 6.4×10^{22} cm^{-2}. At fluences higher than this value, accelerated swelling behaviour occurs, resulting in swelling values higher than predicted by Equation 1.

Figure 3 below presents the axial dimensional swell per beryllium reflector. Each reflector was divided into 10 mm axial segments, and the swell was calculated in each segment to give an indication of the mean swell axial distribution across the width of the reflector per element.

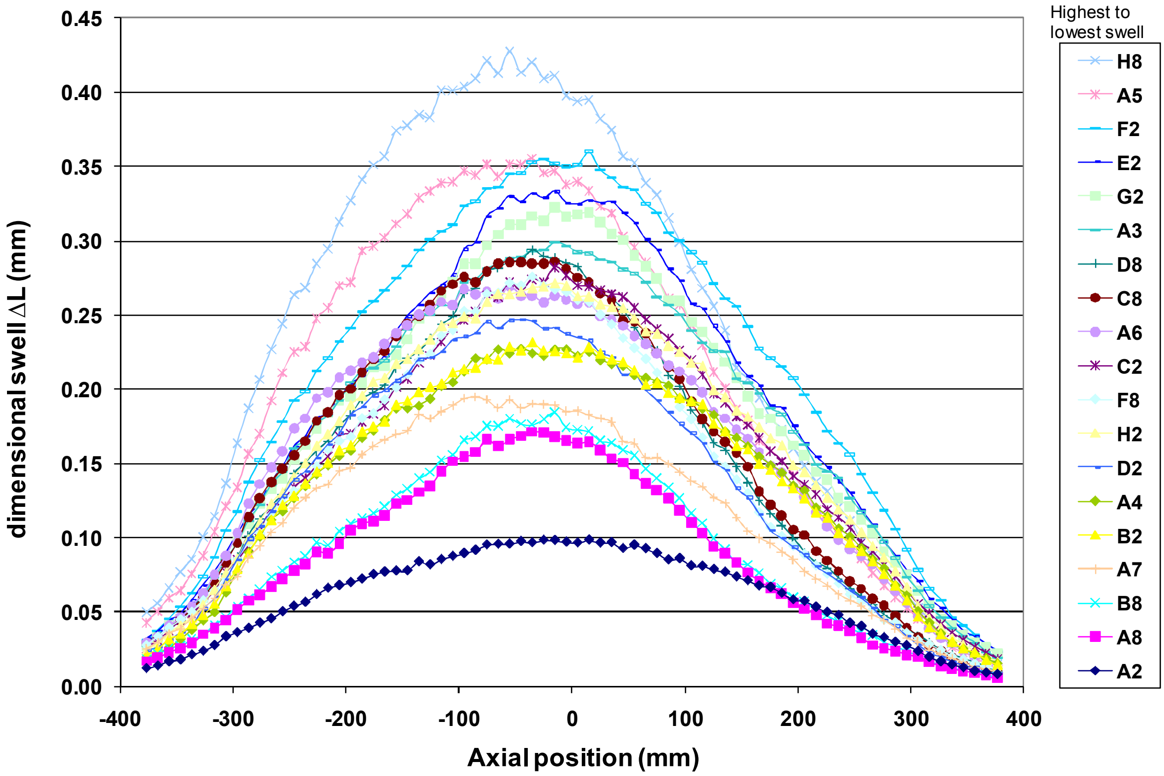

FIG. 3. Dimensional swell per element.

It is evident from Figure 3 that none of the reflectors are theoretically close to the swell limit. Of greater concern is the fact that although the theoretical assessment above may not provide adequate proof for replacement, the practical reality in terms of buckling, mechanical damage due to handling and wear and tear may prove otherwise. Moreover, this swell assessment assumes the gas produced follows the same axial and radial profile as the neutron fluence; however, in practice these gases experience diffusion mechanisms and may form local stresses in other locations that affect the actual swell or bow characteristics and could result in higher dimensional changes and swell characteristics than indicated above.

The impact of fast neutron fluence on the ductility is in general a function of the material fabrication process [5]. Thus quantitative evaluation of the ductility of the present reflectors is problematic due to the fact that the fabrication process for the reflectors is an unknown. At best it can be assumed that mechanical failure due to embrittlement can be expected at the predicted accumulated fluences.

Measuring old reflector elements for bowing and swelling could aid in calculation validation.

3. REFLECTOR REPLACEMENT PROCEDURE

It was decided to approach replacement of the reflector elements in a few separate stages. The specification and purchase of a new reflector were initiated immediately due to long lead times from suppliers. Once new elements are ready for loading into the core, baseline straightness measurements of each element will ensure a reference for future geometrical comparisons as part of a monitoring programme for each element. To ensure safe replacement of the reflector a procedure was derived to ensure that replacement of old elements in the core is controlled and corresponds with calculated core flux mapping predictions. The last stage of the replacement process, the storing of old reflector elements, will be executed by temporary storage inside the pool and then relocation to dry storage and final disposal.

Once baseline straightness measurements are completed and the core reflector replacement can commence, the new beryllium elements will be placed in wet storage in a special basket rack designed for storing these elements inside the reactor pool. Also stored in the reactor pool in a separate rack will be the old elements removed from the core. Although calculations show that there will be no drastic influence on the core due to differences in the neutronic properties between the old and fresh beryllium elements, it is still appropriate to confirm this in practice. Hence a requirement to obtain a critical bank before and after beryllium exchange on the same core loading will be implemented. In addition, foils will be irradiated in selected core positions to provide valuable data on the change in neutron flux due to beryllium replacement.

The startup process to confirm theoretical predictions is expected to be as follows:

- The reactor will be prepared for restart on low power only after 48 hours following shutdown for replacement. This is obviously necessary to allow for ^{135}Xe decay in the core after operation. It is required that the end of cycle core will still be loaded, including the old reflector elements.
- The reactor will then be started, and a stable critical bank will be obtained for at least 15 minutes. The control rod bank and primary inlet temperature will be recorded.
- Reactor power will be increased to a level per the foil irradiation request (~2 MW) and remain there for the required period of time.
- After this the reactor will be shut down.
- These steps will be repeated with only two new reflector elements loaded in the core at the highest reactive reflector positions but still with the end of cycle fuel elements loaded. The critical bank obtained will be recorded.
- Again the steps will be repeated but this time with all reflectors replaced.

Measured values will be compared with theoretical predictions, and if this is found to be within an acceptable or explicable difference, the reflector replacement will be considered as safe for the loading of the new core for operation [8].

4. REFLECTOR MANAGEMENT PROGRAMME

Due to neutron damage to beryllium material, induced swelling and hence bowing as well as embrittlement take place, and the reflectors need to be replaced at periodic intervals. For this reason, a reflector management programme is established at SAFARI-1 to ensure:

- Availability of a complete irradiation history of the beryllium elements at any point in time;
- The establishment of a knowledge base to assist in the understanding of the behaviour of beryllium reflectors in an irradiated environment;
- The extension of the lifetime of the reflector elements through ageing management.

To achieve this, the following actions and practices will be put into place at SAFARI-1 [9]:

- Monitoring of tritium levels (kBq/l) in the primary water will assist in the detection of crack formation;
- Periodic visual inspection of the reflector elements;
- Periodic measurement of the straightness of the elements to track geometric deformation including swelling, bowing and cracking. Deformation will be referenced to reference measurements of new elements before they are inserted in the reactor core;
- Records of accumulated fast fluence for all reflector elements in the core;
- The position and orientation of the beryllium elements in the SAFARI-1 core will be indicated on the core map for each core cycle and tracked for the life of the element. All elements will be periodically relocated in the core and rotated by 180° to ensure a more even accumulation of fluence over time;
- A database to log all recorded data will be developed. This database will be incorporated into SAFARI-1's fuel management software program;
- Establishment of end of life criteria for beryllium reflector elements.

5. REFLECTOR GEOMETRICAL MEASUREMENT

A device designed to measure each reflector element's geometrical straightness will be used to capture reference readings for each reflector element. Future readings will be used to monitor element deformation due to irradiation.

The jig design is still in progress, and the design is based on requirements that include among others:

- A requirement that zero damage will be caused to the reflectors during the measurement process;
- Simplicity of design and operation of the measurement jig;
- The jig should be radiation tolerant;
- The jig should be submerged under water, where measurements shall be done;
- The jig should measure all four sides of the reflector element for geometrical straightness.

With simplicity in mind, one of the concepts currently most favoured makes use of dial gauges evenly spaced along the length of the reflector element. Placing the element vertically in a socket and then laying the element down horizontally on top of the gauges through a simple action places the element in the correct position for measurement. Ledges that support the element at both ends in the horizontal position ensure that each element can be measured with good repeatability.

Figure 4 below shows a drawing of the proposed jig with a reflector element in the measurement position.

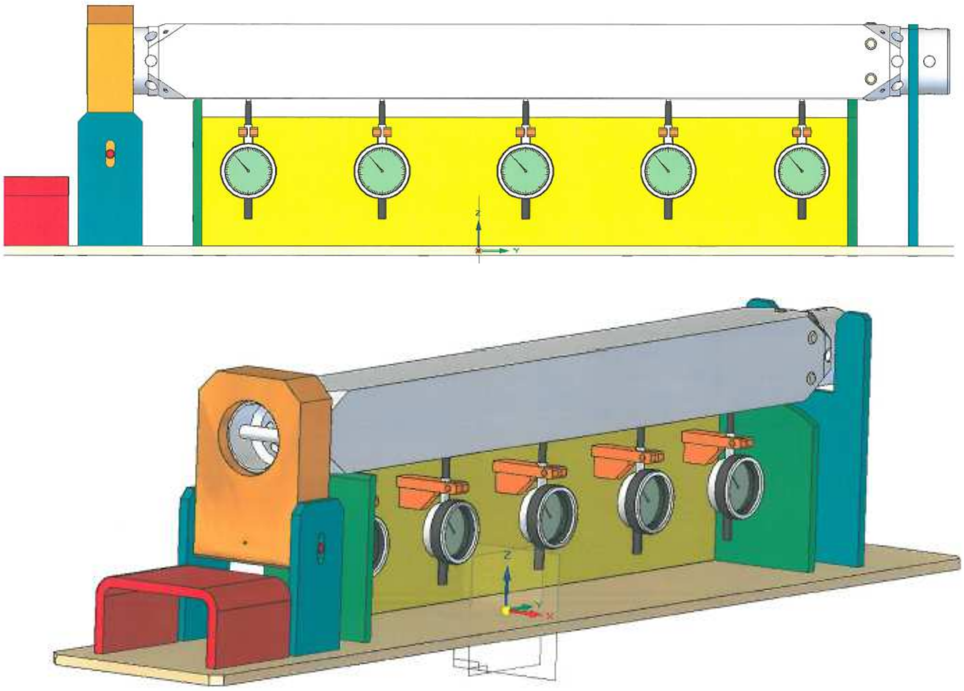

FIG. 4. Reflector element geometrical measurement jig.

6. DISPOSAL OF SPENT REFLECTOR

The intermediate and final storage of the removed spent reflector will be handled in two separate phases. Phase 1 encompasses intermediate storage of the spent reflector. Intermediate storage of the reflector will be through storage of reflector elements in a vertical position in a specially manufactured rack designed for housing these reflector elements. The rack filled with spent reflectors will initially be housed in the reactor pool.

Phase 2 entails the final disposal of spent reflectors. IAEA Technical Report Series No. 441 Management of Problematic Waste and Material Generated during the Decommissioning of Nuclear Facilities, together with information published by the International Energy Agency Irradiated Beryllium Disposal Workshop held at Idaho Falls, USA, during May 2002 will be used to guide the effort in reaching a solution for final disposal. A container for dry storage of the reflector is currently being investigated.

7. PROJECT INFORMATION

This paragraph is intended to provide some basic project information that could be of benefit to readers who plan similar activities and require some basic project indications.

A decision was made early in the project to split execution of the project into three phases. Phase 1 of the project deals with the specification and purchase of the new reflector. Phase 2 deals with the reflector replacement, reflector management programme and interim storage of spent reflectors. Phase 3 deals with removal and final disposal of spent reflectors.

Although time schedules heavily depend on urgency and project priority, supplier lead times could push beryllium reflector replacement projects like the one at SAFARI-1 to more than a year for phases 1 and 2.

Other project information that should be kept in mind includes:

- This project is within the range €500k–1000k excluding internal labour;
- Total project internal resource time is estimated to amount to 2100 man-hours;
- Regulatory involvement and oversight as prescribed by regulations could affect schedules if neither well understood nor included in planning.

Although the safety significance of the reflector in the core is low, the project and quality of the product is rated high. This has an implication on project cost and schedule.

8. CONCLUSIONS

Historical operational data for the SAFARI-1 reflector were not adequately collected and recorded during the first 46 years of operation of the reactor. This led to many uncertainties regarding the condition of the reflector and also introduced many uncertainties in theoretical predictions of the core and reflector. Information about fluence exposure and orientation of different reflector elements are not available.

SAFARI-1's decision to replace the core reflector is justified by primarily the ductility calculation and operational difficulties that indicated possible bowing or swelling of the reflector beyond allowable specification.

SAFARI-1 decided to implement a reflector management programme with the new reflector to address shortcomings in the historical management of the reflector.

POSTSCRIPT

The results and information presented in this paper are part of ongoing work to finalise the replacement of the SAFARI-1 beryllium reflector. Results obtained from measuring spent reflector elements, possible material analysis and relocation to dry storage as well as final disposal could lead to further publications.

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APPROACH FOR ESTABLISHMENT OF AN AGEING MANAGEMENT PROGRAMME FOR THAI RESEARCH REACTOR-1/MODIFICATION 1 (TRR-1/M1)

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Abstract

Thai Research Reactor-1/Mod. 1 is a TRIGA Mark III reactor and has been in operation for more than 30 years. As the reactor has become older, ageing issues have been more prominent. Therefore, the ageing management programme of TRR-1/M1 is being systematically formulated to manage ageing issues. The purpose of this paper is to discuss the approach being taken for the establishment of the TRR-1/M1 programme. Essentially, the approach is based on the new IAEA Safety Guide SSG-10 Ageing Management for Research Reactors, which was published in 2010. The key to success for the approach is to develop the ageing management programme and then integrate it into the current quality assurance programme. The formulation of the ageing management programme takes into account both major ageing categories, i.e. physical ageing and non-physical ageing. The physical ageing management begins by screening the SSCs for ageing management review. The SSCs in the SR-A class (SSCs performing safety functions) and the SR-B class (SSCs not performing safety functions but safety relevant) are evaluated whether to be included in the ageing review routine. The ageing mechanisms of these SSCs are then thoroughly studied to better understand the ageing degradation processes. The examples of ageing mechanisms of these SSCs are fatigue, corrosion and erosion, stress corrosion cracking and irradiation effects. Due to the wide variety of disciplines involved in the evaluation, external experts in each specific field are sought for consultation. The results from the study are to be reviewed for improvement of practices used in operation, maintenance, inspection and testing. The programme will also identify the measures to be taken for detection, monitoring and analysis of ageing degradation trends. The measures will be formulated and included in the routine inspection, maintenance and testing programme. The current conditions of the SSCs are to be factored into the programme. In addition, mitigating actions such as periodic replacement and major refurbishment of SSCs will be scheduled in advance. For non-physical ageing, procedures to address and monitor obsolescence from technology, regulation and standard changes and to maintain up to date information will be included. To assure that all elements of the programme are implemented properly, the current quality assurance programme will be reviewed to identify gaps among current practices and elements specified in the ageing management programme. New procedures will be developed and incorporated in the quality assurance programme to fulfil the ageing management programme.

1. INTRODUCTION

The Thailand Institute of Nuclear Technology (TINT) is working on the establishment of a systematic ageing management programme for Thai Research Reactor-1/Modification 1 (TRR-1/M1). TRR-1/M1 is a swimming pool type TRIGA Mark III reactor designed and manufactured by General Atomics, USA. The reactor core is cooled by natural convection, and heat is removed to the environment in an external cooling circuit. The reactor uses two types of TRIGA fuel elements, i.e. 8.5 wt% and 20 wt% ^{235}U. Currently, there are nine in-core irradiation facilities and several out of core facilities. The utilization of the reactor includes isotope production, neutron activation analysis, gemstone coloration, education and training. Historically, TRR-1/M1 was converted from an MTR type research reactor (TRR-1) that was built in 1962. The conversion took place in 1975 after Thailand signed the Non-Proliferation Treaty. The MTR reactor used HEU as fuel, which was replaced by TRIGA fuel that uses LEU in order to comply with the NPT. The conversion was completed in 1977, and the reactor essentially became a TRIGA reactor. In the conversion, a number of SSCs were replaced including the reactor core, I&C system and radiation monitoring system. Some SSCs were modified or refurbished; e.g. the reactor pool was repaired and re-painted, and the cooling system was modified to include a new heat exchanger to accommodate a higher reactor power. On the other hand, some other SSCs are original components, e.g. reactor building, cooling tower unit, piping system, electrical system and ventilation system. The reactor conversion was a major milestone for Thai nuclear technology, and the reactor has been used for more than 30 years. The current core loading of TRR-1/M1, as of 2011, is core loading number 19.

As the reactor has become older, ageing issues have become more prominent. Some of the SSCs have more issues than the others since they have been used for a longer period of time, and some SSCs are facing obsolescence especially with electronic components. Although there has never been a safety concerned accident with ageing issues for TRR-1/M1, the management of reactor operation is more difficult. There is sometimes an urgency to acquire unplanned budget resources to manage ageing issues in order to provide continuous support for the utilization of the research reactor. Budget planning is thus not well managed or optimized. The management difficulty and the safety concerns of the reactor ageing issues trigger attention to establish a more systematic ageing management programme. With the country's demands on research reactor's utilization, TRR-1/M1 is still expected to continue operations for a certain period of time until it is replaced by a new research reactor. Therefore, the ageing management programme will be integrated as an important element in the operation of the research reactor. The main objective of the paper is to discuss the approach being taken at TINT to establish the ageing management programme and to provide a practical example in the establishment of the ageing management program.

2. AGEING MANAGEMENT METHODOLOGY

To establish the ageing management programme, the IAEA Specific Safety Guide SSG-10, Ageing Management for Research Reactor [1], has been consulted for guidelines. TRR-1/M1 is nominally operated at 1200 kW for 46 hours per week and 46 weeks per year. The establishment of the ageing management programme takes into consideration the operation schedule, which is reflected in the frequency of inspections, detection methodology, preventive measures, etc. The SSCs of TRR-1/M1 are subjected to essentially different environments during reactor operation; hence, they are also diverse in ageing issues. Reactor ageing issues can be generally classified into two broad categories: physical ageing and non-physical ageing. Each of the categories is studied in more detail for the establishment of the ageing management programme.

2.1. Physical ageing

Physical ageing refers to the ageing of SSCs in their physical properties. Ageing can lead to degradation of the materials of the SSCs and may result in loss of ability of the SSCs to function properly, especially for the SSCs important to safety. The establishment of an ageing management program for TRR-1/M1 physical ageing begins by screening the SSCs for ageing management review. The basic concept is to evaluate the importance of each SSC to the safety of the reactor. A short list of SSCs that are safety relevant is selected as a priority for the ageing management programme. The list of the selected SSCs of TRR-1/M1 is shown in Table 1 below. The safety class designation for each SSC in Table 1 identifies the safety relevance classification. The SR-A class describes a SSC that performs a safety function, while the SR-B class contains SSCs that do not perform a safety function directly but have functions that must be accomplished to achieve proper reactor operating conditions, prevent accidents or mitigate accident consequences. Clearly, the SSCs in the SR-A class are priorities for the ageing management programme.

TABLE 1. SCREENING OF THE SSCs OF TRR-1/M1 FOR AGEING MANAGEMENT REVIEW

SSCs	Safety class
Reactor core and structures	
Core structures	SR-A
Reactor pool structures	SR-A

Reactor bridge and platforms	SR-B
Fuel elements	SR-A
Beam tubes	SR-A
Thermal column	SR-A
Reactivity control system	
Control rods	SR-A
Control rod drives	SR-A
Control rod guide tubes	SR-A
I&C system	
Reactor protection system	SR-A
Reactor control system	SR-B
Reactor coolant system	
Primary coolant loop	SR-B
Pool water purification loop	SR-B
ECCS	
Solenoid valve	SR-A
Uninterruptable power supply for solenoid valve	SR-A
ECCS pump	SR-A
Pressurized tank	SR-A
Water supply tank	SR-A
Radiation monitoring system	
Continuous air monitors	SR-B
Remote area monitors	SR-B
Ventilation monitors	SR-B
Personnel monitors	SR-B
Electrical power supply system	
Emergency power supply	SR-A
Experimental Facilities	
In-cores	SR-B
Reactor building and structures	
Reactor hall	SR-B
Reactor hall crane	SR-B
Ventilation stack	SR-B
VAC system	
Normal ventilation system	SR-B
Reactor confinement system	SR-A
Fuel handling and storage system	
In-pool fuel storage facility	SR-B
Fuel handling equipment	SR-B
Auxiliary systems	

Fire detection and protection system	SR-B
Emergency communication	SR-B
Evacuation alarm	SR-B
Radioactive waste system	SR-B

After screening the SSCs for ageing management review, each SSC is thoroughly studied for specific ageing mechanisms to better understand the ageing degradation processes. The study considers various factors that could affect the degradation process. These factors include materials of the SSCs, environments that the SSCs are subjected to while in operation, the operating schedule of the reactor and frequency of surveillance and maintenance. The study of ageing mechanisms is conducted by researching degradation processes of the SSC in industrial applications that have similar environments, reviewing feedback from maintenance records and consulting with specialists of each SSC. The examples of ageing mechanisms of these SSCs are fatigue, corrosion and erosion, stress corrosion cracking and irradiation effects. The results of the studies are documented for subsequent analysis to formulate the ageing management measures.

The formulation of ageing management measures follows the study of ageing mechanisms. The SSCs with similar ageing mechanisms are grouped together for the purpose of developing the measures. The ageing management measures include surveillance activities on specific schedule with specific inspections and periodic replacement and repair on a specific schedule. The results of surveillance and maintenance are to be documented as a database for further review. The database shall provide information about the conditions of SSCs as a function of time. The information is to be used for ageing trend analysis. The implementation of these ageing management measures should minimize the ageing degradation and should provide early indications of ageing issues so that preventive activities can be performed in time.

2.2. Non-physical ageing

Non-physical ageing occurs when SSCs become out of date in comparison with current technology, knowledge, standards or regulations. Non-physical ageing also includes documentation that is out of date and does not reflect the actual conditions of the facility. For the TRR-1/M1 ageing management programme, the non-physical ageing also considers the knowledge loss from retired or resigned personnel as well. The process of screening SSCs for ageing management review also includes the consideration of non-physical ageing. Some SSCs are more susceptible for obsolescence, especially electronics components in the I&C and radiation monitoring systems. The ageing management measures against obsolescence for these SSCs include keeping adequate spare parts for the expected lifetime of the reactor and studying alternative replacements or technology. Since the original SSCs of the TRR-1/M1 were manufactured abroad, replacements for SSCs that can be found domestically are studied. However, for some SSCs, replacement may not be possible, and they should be replaced by new technology. The list of alternative replacements and technology from the study is documented, including potential vendors. When obsolescent SSCs are out of service, the replacements can be procured in time.

In addition to the non-physical ageing of SSCs, the ageing management programme of TRR-1/M1 includes ageing measures for safety relevant documentation. The documents to be covered by the ageing measures are the SAR with operating limits and conditions, emergency plan, engineering drawings and system manuals. The SAR, emergency plan and system manuals may particularly face obsolescence issues, while engineering drawings that were previously not well managed face additional issues; some of the drawings have been lost, worn or damaged. The measures against non-physical ageing for safety documentation include periodic review and update of the documents, implementation of a document control system and updating following modification or changes in conditions. The ageing measures are integrated as part of the QAP.

Another issue for non-physical ageing of TRR-1/M1 is knowledge loss from retired or resigned personnel. The personnel structure of TRR-1/M1 is experiencing a generation gap. This issue may lead to lack of information needed for safe operation of the reactor. The measures to encounter this ageing issue include the implementation of the knowledge management programme and enhanced training for the younger generation. Key information on reactor operation, maintenance, analysis, emergency planning and system design is the focus for knowledge retention. The implementation of knowledge retention and training is planned to be integrated as part of the QAP.

3. QUALITY ASSURANCE PROGRAMME

When formulated, the ageing management measures will be integrated into the current QAP of TRR-1/M1 [2]. This will enable the ageing management measures to be part of the normal operation activities. A gap analysis is to be performed in order to identify the gap between current quality assurance practices and the elements specified in the ageing management programme. The ageing management programme will integrate the main elements of the QAP, i.e. management responsibility, resource management, process implementation and measurement and assessment.

For management responsibility, the integrated quality assurance system shall include clear responsibilities of the management for ageing management and also the framework for ageing management planning. This shall include the planning for monitoring, detection, prevention and mitigation of ageing effects. The budget planning for ageing management shall be part of the routine management review as well. The integration of ageing management in the QAP will make clear the continuous support of top management for ageing management of the reactor. Moreover, the resource management of the QAP shall address the provision of resources including infrastructure, equipment, qualified personnel and training for ageing management activities as needed. It shall also provide the framework for the preparation and issue of specifications and working procedures.

For the process implementation, the QAP shall have the requirements and instructions for the operation of the ageing management activities. It shall provide the framework for performing specific ageing management activities such as ageing surveillance, conduct of preventive maintenance and mitigation of ageing effects. The framework shall also identify clear responsibilities of personnel at each level for implementation of ageing management. Lastly, the measurement and assessment element of the QAP shall cover the framework for checking and assessing the effectiveness of ageing implementation activities. It shall include procedures for review of relevant documents such as the quality manual, work procedures and ageing surveillance and maintenance records, and shall also assess the timeliness of ageing management practices.

4. IMPLEMENTATION STATUS

The establishment of an ageing management programme is a long process and requires various types of expertise. Moreover, the integration of ageing management in the QAP is another issue that is dynamic in nature. The establishment of the ageing management programme of TRR-1/M1 is currently an ongoing work. The screening of SSCs for ageing management review has already been completed, and the study of ageing mechanisms for these SSCs is in progress. Due to the wide variety of SSCs involved, specialists in diverse disciplines, e.g. material science, chemistry, mechanical engineering, electrical engineering and control engineering are required. At TINT, external experts are sought from among industrial specialists and faculty members from universities. When the study of ageing mechanisms for each SSC is complete, ageing management measures will be formulated and planned for implementation in the QAP. It is expected that the formulation of ageing management measures shall improve budget planning and operational management of the reactor.

The establishment of the ageing management programme has experienced some difficulties including incomplete information in relation to SSCs, obsolete or missing documents (particularly for engineering drawings) and inability to find alternative technologies or replacements. To lessen the difficulties, engineering drawings are scanned and retained in electronic copies. This measure should preserve the drawings and retain them for further use. In addition, the drawings are being redrawn from the originals, and some of the drawings are updated based on the actual survey. Ageing management measures to implement the periodic update of safety documents are to be included in the QAP.

The last step on the approach to establish an effective ageing management programme of TRR-1/M1 is to integrate the ageing management measures in the QAP. This process will be performed on a continuous basis. As new ageing management measures are formulated, the gap analysis between the QAP and the new measures will be performed. The QAP will be edited to integrate the new measures.

5. CONCLUSION

The approach to establish the ageing management programme of TRR-1/M1 has been discussed in the paper. Essentially, the ageing management programme of TRR-1/M1 is developed for two major ageing categories: physical and non-physical ageing. For physical ageing, the screening of SSCs for ageing management review has been performed, and the ageing mechanisms of SSCs are being studied in order to formulate the ageing management measures. For non-physical ageing, SSCs, safety documents and knowledge loss were identified as subject to ageing issues. Ageing measures against non-physical ageing are being formulated as well. Some difficulties were found because there is incomplete information about SSCs, and some documents are out of date and sometimes worn. The key success for the approach is to develop the ageing management programme of the reactor and then integrate the ageing management programme into the current QAP for continuous realization.

REFERENCES

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AGEING MANAGEMENT AND MODERNIZATION OF TR-2 RESEARCH REACTOR

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Abstract

TR-2 is a 5 MW pool type research reactor. It is in the same building and in the same pool of the former TR-1 research reactor. TR-1 was active between 1961 and 1977, and during the construction period of the TR-2 reactor (1977–1980), TR-1 was partially refurbished. TR-2 started operation in 1981 and continued to operate until 1995. In 1995 operation was stopped because of concern for the seismic safety of the old reactor building. From 1995 to 2009 the reactor was operated at a limited power of 300 kW. In 2009, HEU fuel was shipped to the country of origin, and LEU fuel was provided. Presently, the reactor is in a shutdown state for the seismic reinforcement project of the building. The building was built in the 1950s, and this and the pool are the only structures inherited from TR-1. Other parts of the TR-2 reactor were first installed in the 1980s. During the 15 years of operation certain components of the reactor aged. Considering the future operating programme, modernization of some systems are in planning. In the paper, ageing management and modernization of the TR-2 research reactor are given.

1. INTRODUCTION

Ageing of a facility is a common problem for those of a certain age. This phenomenon must be considered both before commencing and during operation. For a research reactor, many components can be modernized and refurbished to counter the effects of ageing. This can be done by repair or replacement. Mostly, repair is the first option, but when it is not the solution of replacement is required. Repeated repair extends the lifetime of the component, and this delays refurbishment of the system. Considering replacement, new components may be unsuitable for the old system. This is a common example of ineffective management of ageing, which can cause inadequacy in the facility.

On the other hand, different types of management strategies can be used to minimize the effects of ageing. In the TR-2 research reactor, various components have been changed or repaired. Generally, ageing is managed through components, but this can cause an inadequacy in the system. Likewise, intermittent operation of the reactor makes ageing management more difficult in terms of detection of ageing and faster ageing of components.

2. HISTORY

The first reactor of Turkey, TR-1, was constructed from 1959–1962. TR-1 started operation on 27 May 1962 and continued to operate at 1 MW for 15 years until 13 July 1977. An increase of power to 3 MW was considered during the first half of the 1970s, and for this reason, various changes were made. Although, instead of this project, a new project was commissioned, and the 5 MW powered TR-2 research reactor was constructed in the large part of the TR-1 pool.

TR-2 started to operate in 1981, and continued to operate until 1995. In 1995, operation was stopped because of concern for the seismic safety of the old building. From 1995 to 2009, the reactor was operated at a limited power of 300 kW. In 2009, HEU fuel was shipped to the country of origin, and LEU fuel was provided. Presently, the reactor is in a shutdown state for the seismic reinforcement project of the building.

3. TECHNICAL SPECIFICATIONS

TR-2 reactor characteristics are contained in Table 1.

TABLE 1: SPECIFICATIONS OF THE TR-2 REACTOR

Reactor type	Open pool
Pool	Heavy concrete, 450 m^3, stainless steel surface
Reactor power	5 MW
Moderator	H_2O
Coolant	H_2O
Reflector	H_2O, Be
Fuel	19% enriched U_3Si_2
Control rods	4 rods with 80% Ag, 15% In, 5% Cd
Mean thermal flux	3×10^{13} $cm^{-2}s^{-1}$
First loop pumps	2
First loop flow rate	600 m^3/h
Second loop pumps	1
Second loop flow rate	370 m^3/h
Heat exchanger	Plate type

4. MODIFICATIONS DONE WHEN INSTALLING TR-2

The project for TR-2 was planned to benefit from the existing installation of TR-1. Still, some modifications in installation were needed for establishing TR-2.

4.1. Pool

The TR-1 reactor pool had ceramic lining. The old lining was causing water leakage from the pool. The pool surface was coated with stainless steel.

4.2. Primary coolant loop

Aluminium pipes in the primary cooling circuit of the TR-1 reactor were embedded in concrete under the pool. Pipes were investigated for leakage after disassembly of the TR-1. Several tests were conducted. After these tests, usage of the old aluminium pipes was cancelled. Stainless steel pipes were installed for the TR-1 and TR-2 reactors.

4.3. Demineralization system

Pipes of the TR-1 demineralization system were embedded in pool walls and the bottom part of the pool. The old demineralization system was intended for use by TR-2. However, investigations after disassembly of TR-1 showed that usage of the old system for TR-2 was impossible. A new system was implemented with stainless steel pipes.

4.4. Reactor hall

Airlock doors were added in order to decrease air leakage from the reactor building. The TR-1 reactor facility, however, was not constructed as an airtight volume. There were numerous factors that affected the sealing of building. Many systems were also changed or modified for these reasons.

5. AGEING MANAGEMENT PROGRAMME

Ageing management in TR-2 is performed in three stages. The first is ageing prevention. It consists of actions taken in order to minimize corrosion. pH and conductivity values of pool water are investigated on a daily basis for this reason. Regeneration of ion exchangers is performed due to these water values.

Second in the ageing management programme is ageing investigations. Routine investigations in systems are used for inspecting ageing, as sometimes ageing can be seen during unsatisfied operations. In these cases, aged SSCs are found as the reason for unsatisfactory operation.

The last step in ageing management is ageing handling. Repair is the first option in ageing handling. It has a cost advantage but can cause obsolescence in SSCs. If repair is not possible for aged SSCs, modernization is undertaken.

6. AGEING MANAGEMENT BASED ON SSCs

Certain components in various systems of TR-2 have aged during its lifetime. Some of them were replaced, and some continue in service with routine maintenance.

6.1. The building

The reactor was shut down because of concern for the seismic safety of the building. A refurbishment project for the building was started and is planned to finish at the end of 2012.

6.2. Secondary coolant loop

After the shutdown of 1995, the reactor was restarted to operate with limited power in 1998. At the beginning of the first operation, secondary coolant flow could not reach its nominal value when a secondary coolant pump was activated. The reason for this unsatisfactory operation was found at the pump filter. A huge amount of rust was seen at the filter. Three years of shutdown caused an air inlet to the pipes. The existence of air in secondary coolant pipes increased the corrosion rate by oxidation. These pipes were used until the previous shutdown in 2009. Pipes will be replaced in 2012.

There are six cooling towers in the secondary coolant loop. These cooling towers will be replaced with newer versions. Two new cooling towers were installed but have not yet been connected to the system; they will be connected to the new pipes.

6.3. Fuel elements

A relatively high amount of ^{137}Cs was detected in the pool water. A sipping test was performed. Relative radioactivity was compared along all irradiated fuels. Standard fuel elements S102 and S105 and control fuel elements CO13 and CO14 were then classified as suspect for leakage of considerable ^{137}Cs activity in the water. Examination of these fuel elements showed corrosion at their aluminium cover. The S102 fuel element was one of the most aged fuel elements in the reactor core; its burnup was 44% at that time. While ageing seemed the most possible reason for leakage from S102, leakage from other elements were considered as a result of production defects. These fuel elements were replaced with newer ones.

6.4. Control rods

The nickel coating of the control rods contains small amounts of Co impurity. This causes ^{60}Co contamination of the pool water. Since the reactor was not in continuous operation after 1995, the contamination level was not so high. Replacement of control rods is being considered for future operations.

Drive mechanisms were unloaded for malfunction inspection. Drive motors consequently must be changed.

Electromagnets were investigated. Huge amounts of rust were seen on the stainless steel surface of the electromagnets. This rust flowed through the pipes that connect the electromagnets to the drive mechanisms.

6.5. Air conditioning system

The air conditioning system was changed.

6.6. Demineralization system

The recirculation pump and some of the pipes were changed. Also, there is corrosion at the welds of stainless steel water softeners.

6.7. Heat exchanger

Stainless steel plates of the heat exchanger were damaged during routine maintenance and then replaced with titanium plates.

6.8. Leak through the pneumatic tubes of TR-1 reactor

There are two pneumatic tubes in the TR-1 reactor. Some parts of these tubes were cut exterior to the pool to block leakage.

6.9. Electric system

Some modifications were done in the electric system. Connectors at the TK3 electric panel were changed, but the whole system needs refurbishment.

6.10. I&C system

There have been malfunctions in some relays in the safety logic of the I&C system, and the fission chamber need to be changed. Mechanical problems in recorders have also arisen in recent years, necessitating replacement by digital recorders.

7. CONCLUSION

Ageing management in TR-2 is a straightforward programme. Since TR-2 is already an aged facility, this ageing management is not designed to be proactive. As a consequence, the programme is rather weak as a preventative scheme. Ageing management in TR-2 is generally focused on ageing inspection and handling.

Additionally, the shutdown after 2009 has given the staff enough time for maintenance, but has caused a delay in the detection of malfunctioning. Likewise, a long shutdown period can cause corrosion in some components. Refurbishment of the building and secondary coolant loop components and partial modernization in the I&C system are planned to finish at the end of 2012. Any extension of this period may lead to permanent shutdown. There is, however, routine maintenance for some systems, and repairs to certain components are generally delayed because of the indefinite future.

Ageing of staff is also a problem at the TR-2 reactor, like other reactors of the same age. There is currently only one shift supervisor, who is 63 years old.

All these problems indicate the necessity of a short refurbishment period in order to operate the TR-2 reactor in the future.

RESEARCH REACTORS OF UKRAINE

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Abstract

Ukraine today operates two nuclear research reactors: WWR-M (total capacity of 10 MW), which is located on the site of the Kyiv Nuclear Research Institute of the National Academy of Sciences of Ukraine, and IR-100 (total capacity of 200 kW), which is located on the site of Sevastopol National University of Nuclear Energy and Industry. Both of them have been in operation since the 1960s. The operation project period of WWR-M for which it is licensed is limited to 31 December 2013. In order to improve safety at WWR-M several modernization projects, development of the reactor vessel and the first loop equipment ageing management programme were conducted. According to the license for operation of IR-100 the operation period of the reactor depends upon results from assessments of critical safety elements such as the tank, control and protection system, cable lines and electrical switchgear. Currently the operation period of this equipment has been justified until 2013.

1. INTRODUCTION

Nuclear power is an important component of the fuel and energy complex of Ukraine. Ukraine operates currently (see Figure 1):

- 4 NPPs (15 nuclear power units);
- 2 research reactors:
  - WWR-M (total capacity of 10 MW);
  - IR-100 (total capacity of 200 kW);

- Physical test bench and subcritical uranium–water assembly.

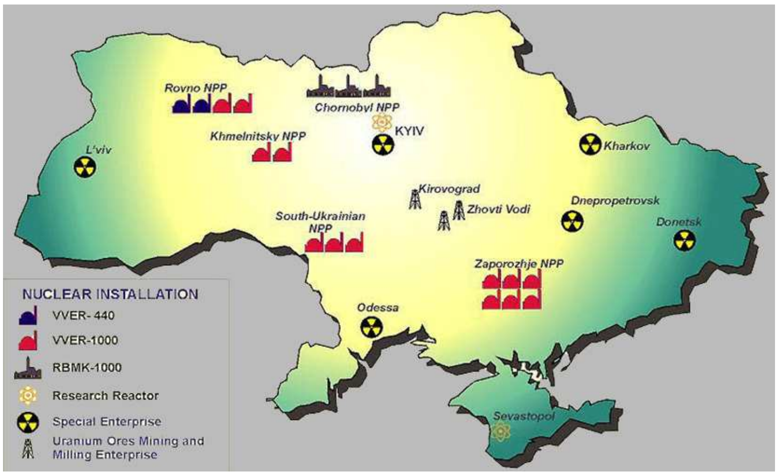

FIG. 1. Nuclear power engineering in Ukraine

2. RESEARCH REACTORS OPERATED IN UKRAINE

2.1. WWR-M research reactor

The site of the WWR-M research reactor is located in the southeastern part of Kiev, the capital of Ukraine, on the site of the Kiev Nuclear Research Institute of the National Academy of Sciences of Ukraine (NASU) [1]. It was one of the first research reactors constructed and commissioned in the former USSR.

The reactor was created more than 50 years ago to implement a programme for providing regional nuclear centres with research reactors on the initiative of academician I. Kurchatov.

From its first years, the scientific and technical basis for research using the reactor has constituted nuclear and radiation physics, nuclear power, solid state physics, production of radioisotopes and radiation biology.

The WWR-M reactor is a heterogeneous water moderated pool type research reactor operating with thermal neutrons at a power of 10 MW_{th} to give a maximum neutron flux of 1.5×10^{14} $cm^{-2}s^{-1}$ at the core center [2]. The reactor has 9 horizontal experimental channels (HECs), a thermal column and 13 vertical isotope channels in the beryllium reflector. It is possible to install 10–12 vertical channels in the core. An overall view of reactor is shown in Figure 2.

FIG. 2. Common view of WWR-M reactor.

The main characteristics of the reactor are presented in Table 1, and the reactor's vertical section is shown in Figure 3.

TABLE 1. MAIN CHARACTERISTICS OF WWR-M

Reactor power	10 MW_{th}
Number of fuel assemblies (WWR-M2 type)	262 maximum, 156 minimum
Fuel enrichment	Currently 36%, after conversion, 19.8%
Core volume	82 l
Maximal density of heat flow	490 kW/m^2
Water flow through primary circuit	1200 m^3/y
Water flow rate in the core	2.6 m^3/s
Water pressure at the core input	1.35×10^5 Pa
Pressure difference in the core	1.5×10^5 Pa
Maximal water temperature at the core outlet	50°C
Maximal temperature of the fuel assemblies	95°C
Maximal density of the thermal neutron flux:	— In the core: 10^{14} $cm^{-2}s^{-1}$ — Near the reflector (isotope channels): 6×10^{13} $cm^{-2}s^{-1}$
Maximal density of the fast neutron flux (E>0.8 MeV)	— At the bottom of hot cell: 4.8×10^{14} $cm^{-2}s^{-1}$ — On the supporting grate: 5.2×10^{12} $cm^{-2}s^{-1}$

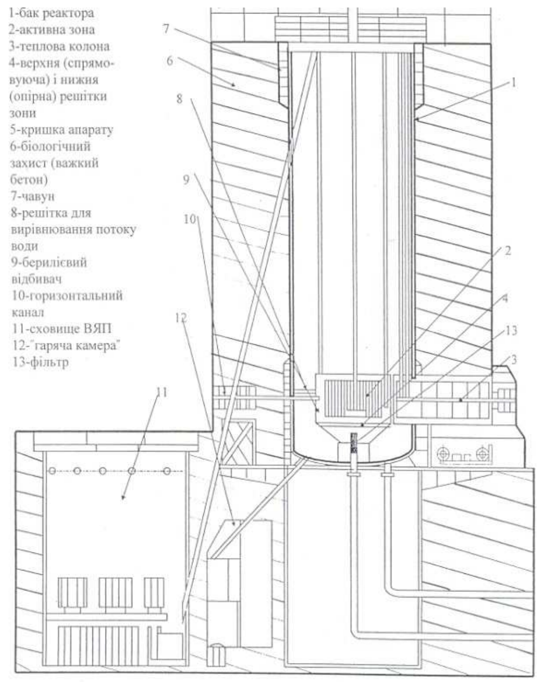

FIG. 3. Reactor's vertical section: 1) reactor vessel; 2) core; 3) thermal column; 4) core grids; 5) reactor cover; 6) biological shielding (heavy concrete); 7) iron; 8) grid for water flow; 9) Be reflector; 10) HEC; 11) SF storage; 12) hot cell;13) filter.

The WWR-M reactor was commissioned on 12 February 1960.

The reactor was shut down in 1993, and the core was entirely unloaded to the spent nuclear fuel wet storage facility. The reactor did not operate again until May 1998. From May 1998 until the end of 2001, the reactor was operated according to the interim permit issued by the regulatory body. Since May 2001, the Nuclear Research Institute has had a permanent license for reactor operations.

From 1994–97, the Nuclear Research Institute undertook numerous measures to improve the nuclear and radiation safety of the reactor, in particular [1]:

- Modernization of physical protection system was commissioned;
- Computer system for nuclear materials accountability was commissioned;
- New automated fire alarm system was commissioned;
- Two diesel power plants of 100 kW power each were installed and connected; this is the source of the emergency power supply system;
- System equipment lifetime for reactor control and protection was extended;
- Reactor tank, piping and primary circuit equipment lifetime were extended;
- Operation of the liquid radioactive waste processing facility was renewed;
- Lifetime of cables and exchanging units of safety related systems were extended;

The reactor remained shutdown until 1998. After verification that the new equipment complied with new safety standards, operation was authorized until the end of 2000, and the core was reloaded.

In accordance with the requirements of Article 8 of the Law of the Ukraine "On Permissive Activity in the Sphere of Nuclear Energy Utilization", which has been in place since 2000, the basis for activity, work, and operations related to the facility lifetime is the license issued by the state nuclear and radiation safety regulatory body. On 15 March 2002, the Board of the State Nuclear Regulatory Committee of Ukraine (SNRCU) issued a permanent license to the Nuclear Research Institute to operate the WWR-M reactor. This was the first operating license issued by SNRCU. The license states the term of operation; in this case, until the end of the reactor's lifetime. The term of operation was continued until the end of 2008. In 2009 a renewed operating license was issued until the end of 2013.

The reactor operations schedule is determined by the requirements of the experimental programmes. The reactor usually operates in weekly cycles (24 hours per day from Monday to Friday). If necessary (in accordance with the conditions of experimental works), the reactor can operate continuously for two to three weeks.

2.2. IR-100 research reactor

The research reactor IR-100 with a thermal power of 200 kW is assigned to conduct science research and training activities in the fields of nuclear and molecular physics, radiation chemistry, radioactive isotope production, material properties, irradiation of devices and equipment in neutron and gamma fields, as well as for training of specialists for nuclear reactor operation.

The reactor site is located in the northwestern part of Sevastopol on the site of Sevastopol National University of Nuclear Energy and Industry. The campus of the IR-100 reactor is surrounded by a fence equipped with a physical protection system.

The IR-100 reactor is a heterogeneous thermal neutron pool type research reactor using pure ordinary water as coolant and moderator [3]. An overall view of reactor is shown in Figure 4.

FIG. 4. Ground view of IR-100 reactor.

The reactor has the below experimental devices [3]:

- Rolling box (800×800 mm);
- Thermal column (1200×1200 mm);
- Tube transfer system (30 mm diameter);
- HEC having diameter of 100 mm in 3 pieces;
- Vertical experimental channels (VEC) having diameter of 76 mm in 2 pieces, and having diameter of 48 mm in 6 pieces;
- Central experimental channel (CEC) having a diameter of 36 mm;

- Hot chamber to allow for conducting experiments with quantities of radioactive substances up to 9 g equivalent Ra of ^{60}Co;
- Experimental unit sigma–sigma.

The main characteristics of the reactor are presented in Table 2.

TABLE 2. MAIN CHARACTERISTICS OF IR-100

Reactor power	200 kW_{th}
The working core loading is the following	47 fuel assemblies, 35 graphite displacers and 1 beryllium block
Fuel enrichment	10%
Core volume	84.15 l
Maximal water temperature at the core outlet	55°C
Maximal density of the thermal neutron flux:	— In core: 5.4×10^{12} $cm^{-2}s^{-1}$ — In horizontal channels: HEC-1: 1.04×10^{12} $cm^{-2}s^{-1}$; HEC-2; 7.4×10^{10} $cm^{-2}s^{-1}$; HEC-3: 1.2×10^{12} $cm^{-2}s^{-1}$; — In the vertical channels: CEC: 5.4×10^{12} $cm^{-2}s^{-1}$; VEC-1: 2.24×10^{12} $cm^{-2}s^{-1}$; VEC-2: 2.22×10^{12} $cm^{-2}s^{-1}$; VEC-3: 0.68×10^{12} $cm^{-2}s^{-1}$; VEC-4: 0.48×10^{12} $cm^{-2}s^{-1}$; VEC-5: 1.98×10^{12} $cm^{-2}s^{-1}$; VEC-6: 1.94×10^{12} $cm^{-2}s^{-1}$; VEC-7: 0.6×10^{12} $cm^{-2}s^{-1}$; VEC-8: 0.44×10^{12} $cm^{-2}s^{-1}$; — In the rolling box: 0.24×10^{12} $cm^{-2}s^{-1}$; — In the thermal column: 2.0×10^{8} $cm^{-2}s^{-1}$.

The reactor design and construction enterprise was Power Engineering Science Research and Design Institute in Moscow. The reactor design power is 100 kW.

The construction of the reactor complex was conducted in 1967. Physical reactor startup was performed on 18 April 1967.

The research reactor was operated at the parameters defined by the original project until 29 August 1972:

- Thermal power of 100 kW_{th};
- Maximal flux in the CEC of 2.5×10^{12} $cm^{-2}s^{-1}$;
- Working core loading of 44 fuel assemblies and 35 graphite displacers.

In 1973 the reactor personnel together with the design institute representatives conducted a series of experiments with increasing reactor power, and subsequently the reactor nominal power was increased to 200 kW (4.8×10^{12} $cm^{-2}s^{-1}$).

The modernization of the reactor control and protection system was made in the period from 12–26 December 1977.

In September 1975 the reactor was used to study the effectiveness of neutron traps located in the central region of the reactor core. In this study, the active zone was reconstructed with a re-organization of the neutron trap, and 56 fuel assemblies were loaded. Upon completion of the work, the load of the reactor core was restored to a working load.

The nuclear facilities of the Sevastopol National University of Nuclear Energy and Industry, research reactor IR-100, physical test bench and subcritical uranium–water assembly, are operated in accordance with the conditions of the license for operation. Design documentation has not established the operation period of IR-100. In this case, according to the license the operation period of the reactor depends on results from assessments of critical safety elements such as the tank, control and protection system, cable lines and electrical switchgear. Currently the operation period of this equipment has been justified until the end of 2012.

3. FUTURE PLANS

3.1. Multi-Purpose Nuclear Reactor [4]

Currently in Ukraine there are only two research reactors in operation, and only one of them is considered a high power reactor. Both were commissioned more than 40 years ago, and the period of their operation is coming to an end.

Ukraine maintains its own nuclear industry for the construction of new NPPs and expansion of existing plants. Providing safe and secure NPP operation along with further development of nuclear science and technologies definitely requires a new research reactor and scientific centre.

The Institute for Nuclear Research at NASU has prepared the Concept of the New Multi-Purpose Nuclear Reactor. This document was approved by the Cabinet of Ministers of Ukraine in Decree No. 1299-r from 8 October 2008.

The concept determines the conceptual basis and the main requirements for the design, construction and operation of the new Ukrainian reactor, providing accepted protection levels for both humans and the environment along with reducing risks of radiological accidents. The concept highlights methods for selection of reactor type, reactor characteristics, possible site locations, proposed reactor utilization and the necessary infrastructure for reactor operation.

The following prerequisites underlie the concept:

- The new reactor is intended as a replacement for the existing WWR-M reactor;
- The new reactor will be designed and constructed as a power source of neutrons with the aim of satisfying current and future needs of the state;
- It will have multi-purpose use for both fundamental and applied investigations;
- The new research reactor will be an integral part of the Ukrainian nuclear power industry;
- The design will follow modern international trends for increased reactor utilization while maintaining all nuclear and radiation safety requirements;
- The construction will benefit the domestic industry and economy;
- The new research reactor should be the core installation of a new national nuclear centre.

Unfortunately, the Concept has not been implemented because of financial difficulties.

3.2. Neutron source facility based on a subcritical assembly controlled by a linear accelerator of electrons

As agreed between Ukraine and the USA at the Washington Nuclear Safety Summit in April 2010, the Kharkov Institute of Physics and Technology (KIPT) is constructing a neutron source based on an accelerator driven subcritical assembly.

The project of this neutron source has been developed by KIPT in collaboration with Argonne National Laboratory of the USA [5]. The facility will have an accelerator driven subcritical assembly with LEU fuel from the Kiev research reactor WWR-M.

The existing KIPT electron accelerator will be used for generating neutrons through photonuclear reactions with a high mass number material such as tungsten or natural uranium to drive the subcritical assembly. The facility will be utilized for producing medical isotopes, supporting the Ukrainian nuclear industry and training young specialists.

The subcritical assembly facility uses proven techniques and practices for its design, operation and maintenance to enhance its utilization. The main facility components are the target assembly, the subcritical assembly, the biological shield and the auxiliary supporting systems. Figure 5 shows an overview of the facility and an example of possible experiments using the radial neutron beams.

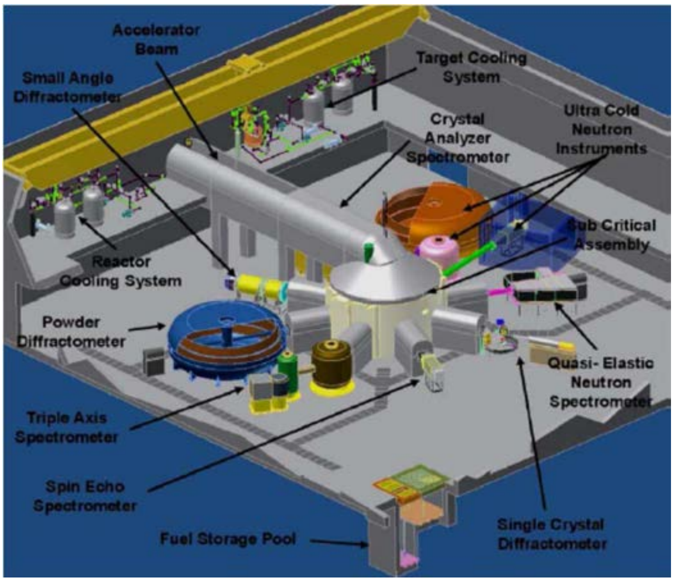

FIG. 5. Overview of the facility and an example of possible experiments using the radial neutron beams.

Ukraine legislation contains only two regulatory documents with requirements for subcritical assemblies:

- Rules on nuclear safety for subcritical stands, PBYa-01-75 (1975);
- General provisions on safety of research reactors in design, construction and operation, OPB IR (1988).

These are documents of the former Soviet Union that were adopted in Ukraine in the early 1990s.

New regulations on safety of the nuclear subcritical installation are under preparation:

- A first draft was prepared on 31 March 2011. It is posted on the SNRIU website, www.snrcu.gov.ua, for public comments and has been sent to interested ministries and organizations for professional discussion;
- At present a second draft is under preparation that takes into account obtained comments and recommendations.

4. LEGISLATION

Operation of research reactors in Ukraine and their supervision is based on:

- Law of Ukraine on Nuclear Energy Use and Radiation Safety;
- Law of Ukraine on Permissive Activity in Sphere of Nuclear Energy Use;
- Regulatory (Normative) Documents.

The main regulatory documents are:

- General Provisions on the Research Reactors Safety Assurance during Design, Construction and Operation (OPB IR 1988) is the main document in research reactor safety regulation;
- Rules on Nuclear Safety of Research Reactors (PBYA-03-75, 1975), which specifies requirements provided by general regulations in research reactor nuclear safety provisions;
- Safety Rules for Nuclear Fuel Storage and Transportation at Nuclear Facilities (PNAE G-14-029-91) defines approaches for nuclear safety assurance in the process of nuclear fuel management at NPPs and research reactors;
- Requirements to Nuclear Installation Modifications and Order of Assessment of Their Safety (NP 306.2.106-2005);
- General safety provisions for decommissioning of NPPs and research reactors;
- Requirements for structures and content of SARs for decommissioning of NPPs and research reactors.

The first three documents are documents of the former Soviet Union developed during 1970–1990. They do not correspond to IAEA recommendations, modern science achievements and technological progress, national experience or international best practices.

According to the Plan of Normative Regulation of the State Nuclear Regulatory Inspectorate of Ukraine the simultaneous review of the above mentioned documents is planned, which shall allow SNRIU to gain assurance for the full package of regulatory documents on nuclear safety of the research reactors as well as nuclear fuel management, including research reactors, in accordance with the existing approaches of the IAEA.

Ukrainian policy is aimed at harmonizing national and EU legislation. The Law of Ukraine On Nuclear Energy Use and Radiation Safety states that nuclear and radiation safety standards of international organizations must be considered while developing relative national regulations and standards. Therefore the above mentioned documents are revised under Task 2 Assistance to SNRCU in Making Regulatory Decisions during Design, Construction and Operation of Research Reactors, Critical and Subcritical Assemblies of Project INSC U3.01/07 (UK/RA/07) Institutional and Technical Cooperation with SNRCU to Develop their Capabilities on the Basis of Transferred European Safety Principles and Practices (Strengthening of and Cooperating with the Nuclear Regulators. As a result, two new documents are planned:

- The basic document for research reactor safety. Its provisions will be based on reviewing the general provisions, PBYa and PNAE-G-029-14-91;
- A document establishing the storage and transportation requirements for nuclear fuel of NPPs and other nuclear installations, except for research reactors.

5. AGEING MANAGEMENT OF RESEARCH REACTORS

During 2008–2009 the Kiev research reactor WWR-M was under a lifetime extension process. The meeting of the Board of SNRCU concerning the changes in the operation license of WWR-M was carried out on 21 April 2009. The decision of this meeting contained a new requirement for the Kiev Nuclear Research Institute to develop an ageing management programme for equipment and pipelines of WWR-M.

Ukrainian normative documents have no requirements on lifetime extension and ageing management of research reactors. In this situation SNRIU recommended the Kiev Nuclear Research Institute to use during the lifetime extension process the approaches, taking into account the specificity of research reactors, that are described in:

- General Requirements to NPP Life Extension above Project Period as a Result of Periodic Safety Review Conducting;
- Requirements to the Structure and the Context of Periodic Safety Review Report of Existing NPP Units.

An ageing management programme must provide for, according to page 3.5. of General Requirements to NPP Life Extension above Project Period as a Result of Periodic Safety Review Conducting:

- Development of a list of systems and elements that are part of the programme;
- Maintenance of a database of the technical condition of elements and found defects based on the aggregation of information on production, operation, maintenance, repairs, tests qualifications, etc.;
- Assessment of the current technical condition of elements and forecast of their changes due to aging;
- Reassignment of element resources, replacement or refurbishment that are not planned until the next periodic safety review;
- Plans for technical and organizational measures to refurbish and replace elements;
- Development of technical and organizational measures for moderation of degradation due to aging;
- Optimization of software maintenance, repair and inspections;
- Introduction of additional controls and diagnostics of the current technical condition of elements and systems, where necessary;
- Study of mechanisms of degradation.

“Program of Ageing Management of the Tank, Pipelines and Equipment of the Primary Circuit of WWR-M Research Reactor of the Nuclear Research Institute of NAS of Ukraine was developed by the Nuclear Research Institute and is being finalized according to state expertise in nuclear and radiation safety.

6. CONCLUSION

Both of the Ukrainian research reactors are in operation, in compliance with existent regulatory documents that were revised in line with IAEA recommendations, modern science achievements and technological progress, the national experience and international best practices.

In accordance with international practice and requirements of SNRCIU operation organizations of research reactors provide ageing management programmes of tanks, pipelines and equipment important for safety.

Both research reactors are more than 40 years old and have exhausted their resources. In this case, Ukraine, a country in which nuclear power is an important component of the fuel and energy complex, will construct two new nuclear installations, a Multi-Purpose Research Reactor and a Neutron Source Facility.

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PROBLEMS OF DECOMMISSIONING RESEARCH REACTOR IR-100

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1. REACTOR IR-100

The research reactor IR-100 with a thermal power of 200 kW is assigned to conduct science research and training activities in the fields of nuclear and molecular physics, radiation chemistry, radioactive isotope production, material, irradiation in neutron and gamma fields of devices and equipment, as well as for training of specialists for nuclear reactor operation.

1.1. Nuclear facility IR-100

The research reactor IR-100, a physical stand containing a critical assembly located in the biological shield of reactor and a subcritical uranium–water assembly located in the laboratory of the department of nuclear steam producing facility in the main educational building of the university comprise the nuclear facility. The nuclear facilities of the university as an educational research complex were put on the list of facilities of national heritage of the state by the act of the Cabinet of Ministers No. 1709 of 19 December 2001.

1.2. The main technical characteristics of the reactor IR-100

The IR-100 reactor is a heterogeneous thermal neutron pool type research reactor using pure ordinary water as coolant and moderator. Its maximum thermal power is 200 kW, the height of the core is 500 mm and its circumscribed diameter is 460 mm. The nuclear fuel is UO_2 enriched to 10% ^{235}U, and the total ^{235}U weight in the core is 2.68 kg. The maximum thermal neutron flux density at the core center is 5.4×10^{12} $cm^{-2}s^{-1}$. The number of fuel assemblies is 45–50 with 7 rods in each.

1.3. Experimental facilities of reactor IR-100

The experimental facilities include:

- Rolling box (800×800 mm);
- Thermal column (1200×1200 mm);
- Tube transfer system (30 mm diameter);
- HEC having a diameter of 100 mm in 3 pieces;
- VEC having a diameter of 76 mm in 2 pieces, and a diameter of 48 mm in 6 pieces;
- CEC having a diameter of 36 mm;
- Hot chamber;
- Experimental unit sigma–sigma.

1.4. The main directions of activity

IR-100 is currently used for:

- Radiation doping of high resistance single crystal silicon;
- Tests of material resistance against radiation
- Long term testing of specimens useful for reactor vessels
- Investigations of absorbed gamma radiation doses in substances.

For training Ukrainian NPP specialists with the help of IR-100 reactor a series of training activities is conducted in the following areas:

- Nuclear physics;
- Nuclear reactors physics;
- Radiochemistry and sample preparation;
- Physical protection;
- Radiometry;
- Ecology

2. THE PROBLEMS OF DECOMMISSIONING REACTOR IR-100

2.1. General terms

Before removing the reactor from service, the stages of decommissioning are described.

The main purpose of the first stage is to clear the reactor facilities of nuclear fuel. It is possible to store nuclear fuel on the reactor's campus only in a special reservoir for nuclear fuel waste that is intended for long term safe storage of such substances.

For permission for removal of the reactor from service the responsible organization should provide to the state regulatory authority the following documentation:

- The programme for removal of the installation from service;
- The SAR;
- The changes in installation technological regulation.

In accordance with General Terms about Providing Safety during Removal of Atomic Power Stations and Research Reactors from Service it is necessary to follow several steps during removal of an installation from service: final closing, conservation, delay and dismantling.

The necessity of each of the mentioned steps is defined and based on the programme for removing the research reactor IR-100 from service.

Before removing the reactor from service, the project and the programme for performing the work are developed and also subjected to a full inspection with representative persons from the state regulatory authority of nuclear safety supervision.

In accordance with the results of the mentioned inspection, the regulatory authority makes a decision about removing the object from service using the prescribed procedure.

A full inspection should take place in accordance with special programmes that are approved by the state regulatory authority of nuclear safety supervision.

2.2. Service termination of reactor IR-100

After a decision to remove reactor IR-100 from service, plans will be made for the final stage of its operation. Subsequently the stage of reactor service termination will start. The main purpose of the termination stage is to clear the reactor facility of nuclear fuel. It is possible to store the nuclear fuel on the reactor campus only in special reservoirs for nuclear fuel waste intended for long term safe storage of such substances.

This stage includes the realization of the following measures:

- Cleaning of the nuclear fuel;
- Cleaning of the solid and liquid radioactive waste products that accumulates during the period of operation;
- Final shutdown of systems and elements;
- Deactivation of systems and elements;
- Organized measures for preparing the removal from service.

The process of cleaning nuclear fuel includes unloading the nuclear fuel for temporary storage and developing the measures for its removal.

2.2.1. Transfer of nuclear fuel to the storage silo

The technology of nuclear fuel transfer from the core into silo storage was perfected over a period of many years. According to the transportation instructions for the nuclear fuel located in the core of the reactor, it must be kept in silo storage at least three years before transportation to any other place.

2.2.2. Organization following nuclear fuel placement

Various measures for fuel maintenance during this period can be realized with appropriate decisions by the Ukrainian Cabinet of Ministers or Ministry of Energy and Coal Industry. One of these measures is the process to clean liquid and solid radioactive waste products after the storage term.

2.2.3. The route development

After determining the place of continuous storage or processing of nuclear fuel waste, the route of transportation will be developed.

2.2.4. The final stop of the systems and elements

During this period all systems and elements will be shut off except those that provide for the normal work of the staff such as ventilation, radiation and dosimetry control, water cooling in the silo storage and others.

2.2.5. The development of work reports

A radiation inspection will be implemented for radioactive waste products and documentation will be prepared for the license for removal from service, including a program for final closure of the reactor. In the state inspection of nuclear regulators the report on performed work and proof that the installation has been prepared for shutdown will be presented.

3. REMOVAL OF FRESH NUCLEAR FUEL

The receiving organization, most likely the Mayak Production Association of the Russian Federation, for the removed fuel nuclear fuel must be determined.

The development of adoption and approval procedures for the prescribed documentation for transferring and storage of fresh nuclear fuel to the organization with the appropriate license can then proceed. Furthermore, the development of a programme and a technique for nuclear fuel movement, as well as its adoption and approval in the prescribed manner, must be developed. This work will be accompanied by a report analysing the provision of nuclear and radioactive safety during movement and transportation of the fresh nuclear fuel.

The manufacturing and order of the appropriate transport containers to be provided by Mayak will then take place, along with the order of a special transport, train and other measures for physical protection of the nuclear fuel during transportation to the destination point of Belbek Sevastopol International Airport. Transportation by plane must be done by a border agency special subdivision. The performance of the programme and the technique for moving the fresh nuclear fuel from storage to transportation containers, which will be accomplished by IR-100 personnel with involvement from the Kiev Institute of Nuclear Research, will be licensed especially for movement and loading. The performance of fresh nuclear fuel transportation from the customs point in Belbek Airport to the destination point will be achieved by a convoy provided by the Russian Federation Ministry of Internal Affairs.

The performing of works for fresh nuclear fuel storage deactivation, as well as cleaning and clearing procedures for the equipment and the tools used during movement process must be designed. Finally, accounting and documentation for the change of the nuclear fuel quantity in the installation will be developed.

4. THE STRATEGY OF REACTOR IR-100 DECOMMISSIONING

In accordance to point 3.3 of the General Safety Regulations, four should be accomplished. The final closing achieves the transition of the installation to a position that excludes the possibility of use for its directly intended purpose. Conservation then places the installation into a position to provide safe long term storage of the radioactive substances contained within.

A delay period allows for the gradual decrease of the quantity of radioactive substances contained in the conserved installation through natural radioactive nuclide disintegration. Dismantling provides for the removal of sources of ionizing radiation from the installation.

All possible strategies of decommissioning will include all four mentioned stages. Stages 1, 2, and 4 are possible to perform in a short period of time, as little as one year. All decommissioning stages must be performed in the reactor building with total adherence to radioactive and sanitary regulations. Thus the influence on the environment and population will not exceed the influence of normal operation of the reactor.

The variants of decommissioning strategies in general differ in performance of the third stage. The first variant may delay performance of the final stage for 2–3 years. The second variant on the other hand may require 30 years for satisfactory completion. The advantages and disadvantages of both are described in Table 2.

TABLE 2: COMPARISON OF THE TWO DECOMMISSIONING VARIANTS

Variant 1	
Advantages	Disadvantages
— In a shorter period of time the reactors construction and other units and aggregates turns into position that allow to use them for other needs; — It is possible to save the skilled staff for performance of deactivation of rooms and units and dismounting equipment;	— It is necessary to use more staff for deactivation and dismantling; — Financing should be provided in a short period of time; — A larger quantity of radioactive substances will infiltrate the environment through ventilating system and special canalization during performance of the dismantling

— Costs for security and personnel maintenance are reduced.	stage than in the case of variant 2.
Variant 2	
Advantages	Disadvantages
— The time of financing for delay and dismantling is longer; — A lower quantity of radioactive substances will infiltrate the environment through ventilating system and special canalization than in the case of variant 1.	— The physical and technical properties of the reactor units and elements are poorer; — Costs for security and personnel maintenance is larger; — Skilled personnel can leave at any point to the moment of dismantling; — For a significant period of time the construction cannot be used for another purpose.

Analysing the two variants in terms of positive and negative we can conclude that the first variant has an important advantage versus the second variant from the financial as well as mental and ethical point of view.

The work included in all stages of the decommissioning process will be performed in accordance with National Radiation Safety of Ukraine, Basic Sanitary Rules for Ukraine and other normative documents and regulations.

5. THE FINAL CLOSING OF REACTOR IR-100

In this stage the reactor IR-100 will be deactivated as a neutron source. After removing the nuclear fuel from the reactor's active area it will not be able to produce neutrons, and the equipment of the experimental channels will be dismounted. Thus the main purpose of the stage will be reached.

In this stage the following activities have been planned:

- Performance of radiometry and dosimetry inspections of the reactor IR-100 and the development of cartograms;
- Development of a nomenclatural list of radioactive pollution and activated reactor systems and elements;
- Development of radioactive pollution cartograms;
- Dismounting of experimental installation equipment;
- Preparation for removing solid radioactive substances, for instance in the special factory Radon;
- Strengthening of the protection near the concrete case of the reactor, especially at the gate installations and around solid radioactive substances;
- Processing of liquid radioactive substances;
- Preparation of the units and elements for conservation and their deactivation;
- Development of the work report;
- Development of the programme for the conservation stage.

Radiation monitoring will include:

- Gamma radiation dose monitoring; neutron equivalent dose monitoring; and neutron, beta and alpha flow density monitoring at job sites, in adjacent rooms, on the reactor campus, in the sanitary protected zone and in the monitoring zone;
- Monitoring of radioactive gases and spray content in the air in working and other rooms of IR-100;

- Radioactive pollution level of working surfaces and equipment and skin and clothing monitoring;
- Atmospheric monitoring for spikes in radioactive substances;
- Monitoring of radioactive substances in the liquid waste products and waste waters;
- Control of waste liquid and solid radioactive products collection, elimination and neutralization;
- Monitoring of pollution levels for environmental objects outside IR-100;
- Monitoring of radioactive pollution levels for vehicles;
- Monitoring for inner and outer exposure doses as well as radioactive substance content in organisms;

The radiation control system of IR-100 includes the following technical facilities:

- Continuous control based on the data of stationary technical facilities, an installation for radiometric control RMS-2 for strict monitoring, and gamma monitoring in the neutron shield tank;
- Operative control based on mobile technical facilities DKS 96 (gamma and beta) and the neutron radiation meter KND;
- Alpha radiation monitoring if the heat generating elements will be destroyed;
- Laboratory analysis based on data from the stationary laboratory equipment, extraction facilities and preparation probes.

6. THE PROGRAMME FOR PROVIDING QUALITY

The process of quality provision during the decommissioning of a nuclear installation in accordance to the guidelines of the operating organization should be in agreement with these purposes:

- The development of an optimal control system;
- The provision of radiological safety.

The programme of quality provision during the decommissioning of a nuclear installation is based on Regulating Quality for Operating organization SNUNEI, taking into account the IR-100 operation experience. The QAP takes into account the factors that have an influence on nuclear and radiation safety during the decommissioning of a nuclear installation.

In the QAP during the decommissioning of nuclear installation IR-100 the following processes will be performed in accordance with Ukraine Standard DSTU ISO 9001:2009 The System of Quality Control Requirements:

- Management responsibilities;
- Resources management;
- Financial provision for decommissioning;
- Regulation requirements;
- The development of techniques and technology for nuclear installation decommissioning
- The development of techniques and technology for radioactive waste products handling;
- Social aspects;
- Nuclear installation decommissioning.

The management responsibilities include the distribution of charges between official persons of the operating organization analysed from the management point of view, planning, distribution of authority among official persons of the operating organization, as well as information provision.

Resource management entails provision of material resources, selection, alignment and qualification of personnel, and development and maintenance of appropriate infrastructure.

Financial provision includes estimating the cost of nuclear installation decommissioning, estimating the cost for handling and disposal of radioactive waste products and identifying the funding source.

The requirements contained in regulations provide for the development of the conception and the strategy for decommissioning the nuclear installation IR-100. The techniques and technology development for nuclear installation decommissioning include the application of modern technologies, equipment and techniques for handling radioactive waste products. The social aspect includes development of programmes for radioactive safety, natural environment monitoring, physical protection for nuclear installations and nuclear materials during decommissioning, as well as measures in case of a radioactive incident.

Nuclear installation decommissioning means the step by step performance of the stages for completing operation of the nuclear installation, final closing, conservation, delay and equipment dismounting. These stages are performed while taking into account the real state of the nuclear installation, the experiences of decommissioning such installations in Ukraine and IAEA recommendations. Document management during nuclear installation decommissioning will be performed in accordance with Technique of Documents and Records Management in the System of Quality Management of the Operating Organization, Guidance Document 000.7.02/3.001-07.

REPLACEMENT OF THE ELECTRONIC PART OF INSTRUMENTATION AND CONTROL OF THE WWR-SM RESEARCH REACTOR

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Abstract

WWR-SM research reactor was put into operation in 1959. Since that time the reactor has been very effectively operated, but all electronic equipment of I&C is now outdated. On 7 December 2009 a contract among the International Atomic Energy Agency, Joint Stock Company 'SNIIP-SYSTEMATOM' and the Institute of Nuclear Physics of the Academy of Sciences of the Republic of Uzbekistan concerning design, manufacturing, delivery and commissioning of a complex control and protection system equipment for the nuclear research reactor WWR SM was signed. The implementation of this contract started in 2010.

1. INTRODUCTION

The research reactor WWR-S, a light water pool type reactor, reached its first criticality on 10 September 1959 [1]. The reactor operates at a power rating of 2 MW.

In 1978, the reactor active core was reconstructed, and partial replacement of the I&C equipment was accomplished. The reactor name was changed to WWR-SM, and nominal power was increased to 10 MW. Modernization of the physical protection of the WWR-SM reactor was carried out in 1996, with modernization of the radiation control system in 2006. In 2009 the reactor active core was fully converted to LEU fuel.

Positive reactivity insertion by I&C functions must be provided, but not at a speed more than $0.07\beta_{eff}$/s. Insertion of negative reactivity on safety alarm signals is provided at higher speeds. For alarm rate and controlled rate executing bodies with efficiency value of $0.7\beta_{eff}$, insertion of positive reactivity must contain a step weight of no more than $0.3\beta_{eff}$, and an ability to break the power circuit from the control room must be provided. The new I&C system must meet all requirements of the Uzbekistan nuclear safety rules [2].

2. REASONS FOR I&C ELECTRONIC PARTS REPLACEMENT

Installed I&C equipment is now outdated. Most of the parts were produced 40–50 years ago, and details for replacements cannot be found on the market today. This equipment is physically aged. Below in Figures 1–3, samples of installed outdated equipment are presented.

FIG. 1. Ionization chamber power unit made in 1966.

FIG. 2. Test device for prestart adjustment of emergency safety amplifiers made in 1956.

FIG. 3. Automatic recording KIP device made in 1971.

3. ORDER OF EVENTS

To replace old electronic parts of I&C equipment and thus increase reactor safety, the Institute of Nuclear Physics (INP AS RUz) approached the IAEA for consultation and financial support.

As a reply to the INP AS RUz approach, the IAEA sent an INSARR mission to INP AS RU in June 2008. Experts together with INP staff defined equipment that needed to be updated and upgraded during the first stage. A performed inspection led to the development of technical specifications for I&C of WWR-SM in August 2008.

Next year bidding was announced by the IAEA, and Joint Stock Company (JSC) SNIIP-SYSTMEATOM won the bid. In December 2009, a contract between the IAEA, JSC SNIIP-SYSTEMATOM and INP AS RUz concerning design, manufacturing, delivery and commissioning of complex control and protection system equipment for nuclear research reactor WWR-SM was signed.

During 2010 actual work started to prepare for installation the electronic parts of the I&C system equipment:

- March 2010: initiation of a technical task for complex equipment of the control and protection system for WWR-SM reactor;
- August 2010: audit of JSC SNIIP-SYSTMEATOM within the quality providing programme;
- November 2010: preparation of premises and areas for the ASUZ-11R complex.

Replacement of the old I&C system equipment will not only solve the problem of spare parts availability, but also will bring more advantages that were not accessible with the old system.

These new advantages to the ASUZ-11R complex include:

- Reactivity control;
- Rod control operation and falling time determination;
- Automatic diagnostic of protection and control channels operability;
- Detection of seismic activity;
- Mapping and registration of information in digital and graphical modes;
- Archiving and documenting information;
- Protection from unauthorized access;
- Preliminary amplification and galvanic decoupling of signals from neutron flux detectors;
- Utilization of a wide range neutron flux detectors, i.e. eliminating movement of detectors;
- Tightness of neutron flux detector cables;
- Automatic prestart check of alarm signals and emergency protection signals;
- Design and aesthetics.

4. FUTURE WORK

Much preparation work has been done since the beginning of the project. Now the project is at the stage of receiving at the ASUZ-11R complex and its installation. After installation work is finished, the following work must be done:

- Junction of ASUZ-11R complex with the rod control equipment of WWR-SM reactor;
- Autonomous and complex trials, jointly with the supplier;
- Physical and power startup of WWR-SM reactor, jointly with the supplier;
- Renewed operation of the WWR-SM reactor.

5. CONCLUSION

Work on I&C replacement is planned to finish in 2012. Actual installation time and junction of the new ASUZ-11R complex with the rod control equipment of the WWR-SM reactor and autonomous and complex trials are planned to take two months. This will not cause a long delay with radioisotope production.

Installation of the new I&C system will significantly increase the safety of WWR-SM reactor operation.

For the second stage of reactor ageing management, the replacement of the heat exchanger and secondary piping is planned.

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VENEZUELAN EXPERIENCE AND CHALLENGES ON PARTIAL DISMANTLING AND CONVERSION OF THE VENEZUELAN RV-1 NUCLEAR RESEARCH REACTOR INTO AN INDUSTRIAL IRRADIATION PLANT

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Abstract

The RV-1 nuclear research reactor of the Venezuelan Institute for Scientific Research is a pool type reactor of 3 MW of thermal power that was operated from 1961 to 1991 despite having a high level of obsolescence. Since 1991, IVIC has studied different options in spite of its obsolescence to upgrade, decommission or convert the reactor facility into an industrial irradiation plant. The option to convert the RV-1 into an industrial irradiation plant prevailed politically. In December 2003, the reactor conversion activities had been completed at a cost of US $3 million. The RV-1 conversion project consisted mainly of the characterization of the radioactive contamination of structures, systems and equipment in preparation of their dismantling and removal from the reactor building, waste classification and disposal. Structural modifications were made, e.g. new ventilation and water systems for treatment of the reactor pool and the ^{60}Co source pool, concluding finally with the installation and commissioning of a new irradiator with a maximum rack capacity of 1 MCi. Currently, the RV-1 remains permanently shutdown, and the industrial irradiation plant is operating 22.5 hours per day, 365 days a year. All of these activities carried out have represented a challenge for our country.

1. INTRODUCTION

Venezuela acceded to the US Atoms for Peace programme in June of 1955. The construction of the research reactor building started in 1958 with a US $1.5 million total cost including a load of 36 fuel assemblies. The reactor was named RV-1, which was a material test reactor type. The first criticality occurred on 12 July 1960, and it was designed to operate at 3 MW.

The RV-1 was in operation for 31 years, and it was shut down in 1991 upon safety recommendations by the IAEA. The decision to permanently shut down the RV-1 was made in 1997.

Since 1991, the Venezuelan Institute for Scientific Research (IVIC) studied different options due to its obsolescence, among others reasons, to upgrade, decommissioning or convert the RV-1 into an industrial gamma irradiation plant. Between 1991 and 1997, IVIC analysed these three options. The first option related to the reactor upgrade, with potential use of the reactor at an estimated cost of US $10 000 000. The second option related to final reactor decommissioning, recognizing the problems of not having in our country a final repository for radioactive waste resulting from demolition and dismantling as well as an estimated cost of US $20 000 000. For the third option, the IVIC authorities were not very convinced, and for this reason, I made arrangements for an IAEA expert mission.

In 1997, the IVIC received the IAEA expert mission to evaluate the feasibility of conversion of the research reactor into an industrial gamma irradiator. The conclusions of the mission [1] were that the existing building and the equipment would be useful for this purpose. An unusual aspect for this type of installation is that the storeroom must be on the ground level, whereas the processing, as well as the main and auxiliary equipment, is on the subterranean level. Taking into consideration that the storeroom; access for trucks; power; water pipeline; canal system; basic concrete shielding; labyrinth and auxiliary equipment like ventilation, demineralization, compressed air, dosimeter and trained personnel already existed, after a comparison of the two investments for updating and the reconversion, the concept of reconversion appeared rather attractive.

The Venezuelan government approved in 2000 the conversion of the RV-1 research reactor, and the commissioning of this modified facility and the irradiator was achieved in December 2003. MDS Nordion Inc., Canada, was in charge of the irradiator project, while IVIC was in charge of all matters related to changes in the reactor and its safe condition.

In September 2004, IVIC received authorization to operate the panoramic wet source storage gamma irradiator, category IV, with a nominal capacity of the ^{60}Co source plaque in the irradiator of 10^6 curies (37 PBq). The radiation shield meets the specification for the maximum radiation field recommended by the IAEA and the US Nuclear Regulatory Commission for an installed source of 3×10^6 Curies (111 PBq).

2. REACTOR MODIFICATIONS

2.1. Characterization of radioactivity in SSCs

This activity was carried out by IVIC personnel who evaluated the removable radioactivity contamination on different surfaces, production by neutron activation and the spread of radioactive substances across the surfaces of the reactor structure.

In assessing the majority of the external surfaces on the SSCs (See Figure 1), we found no traces of radioactive elements. Surface contamination was evident only in the following:

- Pipes and drain pipes of the primary cooling system, mainly the reactor recirculation water through the deionizer. The assessment of these pipes showed a surface contamination of less than 4 Bq/cm^2 of ^{137}Cs and traces of ^{60}Co. It was established that surface contamination exceeding 4 Bq/cm^2, would be considered free for release;
- Cationic and anionic resins in the demineralization system for the reactor pool water and spent fuel storage. In this case there was contamination above the limit for free release, and therefore they were stored in the IVIC low activity repository;
- No evidence of surface contamination on the exterior of the stainless steel primary cooling system pipes, but when these pipes were opened, we found surface contamination slightly above the limit. These pipes are stored in a controlled area of IVIC;
- The primary pump was not stainless steel, and internal contamination slightly higher than the limit was evident. This radioactivity was determined to be a result of activation of the pump structure;
- Sampling was conducted for all topsoil, excavated material and loose or distributed material, but the presence of radioactive elements determined not to be from reactor activities.

FIG. 1. Assessment of internal contamination of the primary system piping.

2.2. Reactor building

To locate the irradiator within the RV-1 reactor building several civil works were conducted.

2.2.1. Demolition and removal

At the basement level in the heat exchange room, concrete pedestals for support piping were demolished, and the biological shield walls of the deionizer water system and areas of floor slab were dismantled, as in Figure 2. Excavation for the source pool foundation was done.

The existing building foundations were protected from damage or movement resulting from a loss of bearing support during the excavation of the source pool foundation.

FIG. 2. Concrete pedestal demolition with the use of expansive chemicals.

All piping was cut and dismantled in the heat exchanger room, and also the heat exchanger and pump of the primary reactor cooling system were disassembled. See Figures 3 and 4.

FIG. 3. (top) RV-1 primary cooling system; (middle) view of the primary cooling system of the reactor and its removal; (bottom) view of the part of the secondary cooling system and the pipe cutting operation.

FIG. 4. Primary cooling system and removal of the primary pump.

The ventilation system of the reactor was entirely dismantled and replaced by a new ventilation system. The new ventilation system is simple and maintains a negative pressure in the irradiation room and at the top of the reactor building. See Figure 5.

FIG. 5. Different views of the ventilation system in the reactor basement and dismantling.

The main electricity supply already existed at the proper place to suit the needs of the irradiator even after the upgrades.

2.3. Building modification to install the irradiator

The irradiator was installed in an existing radiation shield and built in the heat exchanger room. Figure 6 shows the concrete structures of the reactor basement.

The wall thickness located in the heat exchanger room had at least 1.2 m thick reinforced concrete. In order to place a source of 10^6 Curies, it was necessary to increase the biological shield around the ^{60}Co sources. The thickness of the walls and ceiling were increased to 1.8 m. Additionally, it was necessary to build two additional walls, as seen in Figure 7.

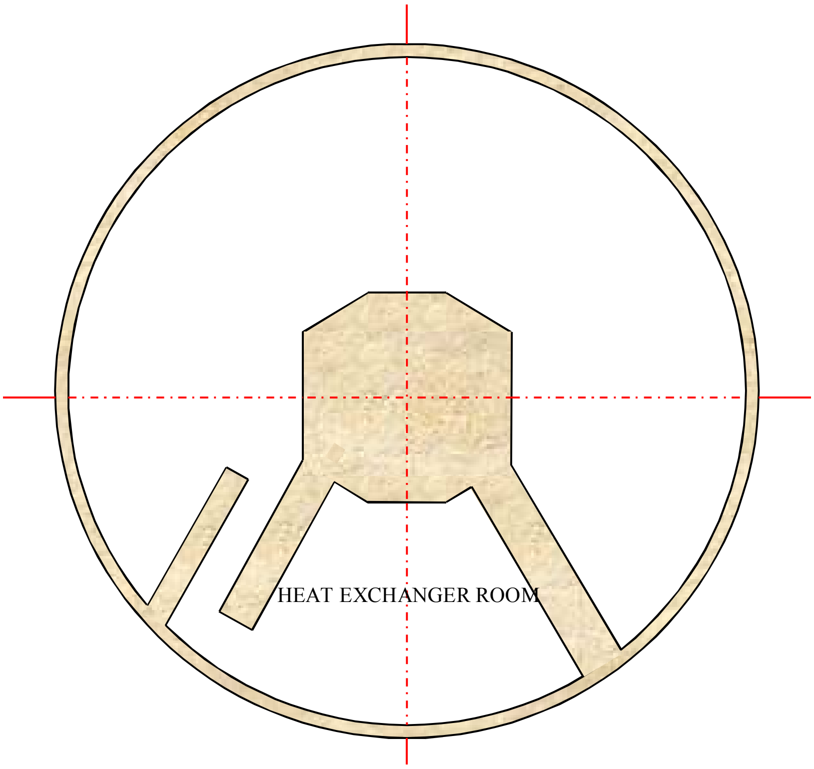

FIG. 6. Floor plan of the subterranean processing level of the reactor building.

2.4. Industrial gamma irradiation type

Considering that the existing reactor structure was adequate to house an industrial gamma irradiator, several considerations in relation to the proposed conversion of the nuclear research reactor into an industrial gamma irradiator were developed entirely by IVIC. A company with great experience in the design, operation and supply of ^{60}Co sources, and preferably one capable of receiving the exhausted sources, was sought.

Although IVIC had experience operating a panoramic mini irradiator that had been updated twice, the organization did not have the experience to undertake a project of this magnitude due to factors of irradiator safety and operation. The research reactor is an unusual place to house an irradiator, and it was necessary to comply with national regulations and international standards.

Taking these into consideration, IVIC authorities decided to hire an international firm with recognized expertise in irradiator design and installation and commissioning of industrial gamma irradiators. The company selected was MDS Nordion Inc. of Canada, and after reviewing several design options, IVIC authorities decided on the acquisition gamma irradiator JS-9500HD.

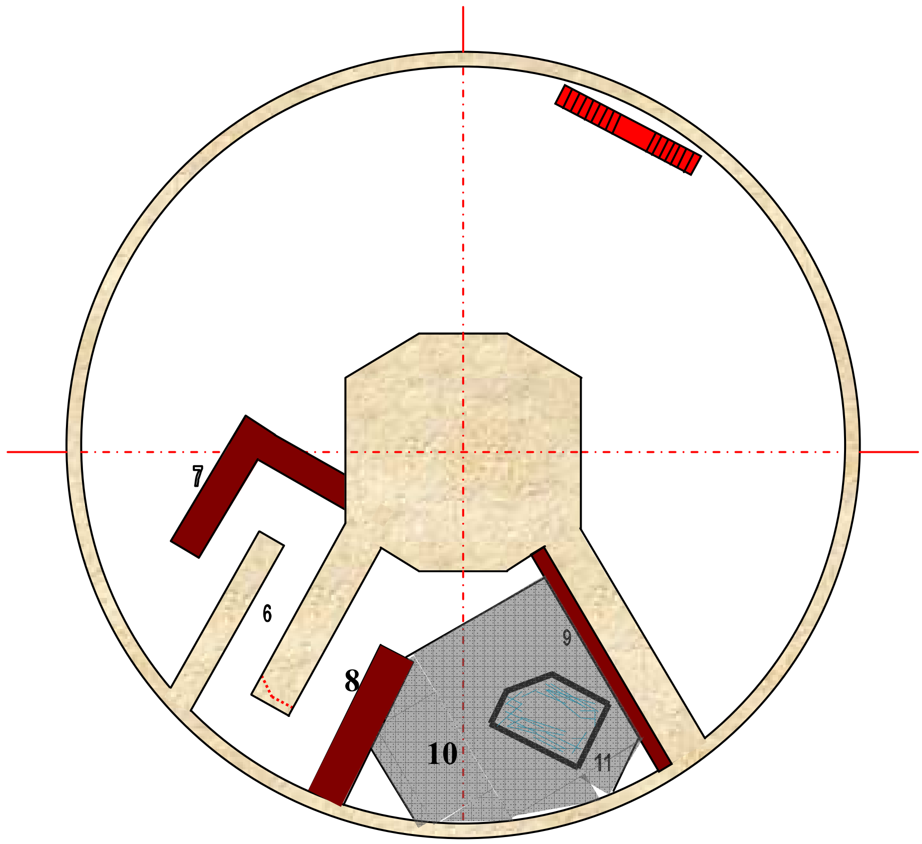

FIG. 7. Reactor basement with increased wall and ceiling and two extra walls. 6: labyrinth; 7–9: additional concrete walls; 10: additional concrete ceiling; 11: excavation of the ^{60}Co sources.

2.5. ^{60}Co irradiator IR-216 IVIC

The ^{60}Co irradiator IR-216 is a panoramic, wet source storage gamma irradiator, category IV.

The ^{60}Co irradiator IR-216 [2] is an industrial gamma irradiator, model JS9500HD, designed by MDS Nordion Inc., that processes pre-packaged material by exposing them to controlled doses of gamma radiation emitted from a large ^{60}Co source rack. Products move along a predetermined number of steps around the source to receive sterilizing doses of gamma radiation.

The irradiation facility, in addition to having a radiation room in the heat exchanger room and labyrinth, includes a control room, an equipment room and a storage area. The control room containing the control console and programmable logic controller cabinet is adjacent to the labyrinth entrance door. The equipment room contains the water treatment plant and water chiller. The ventilation system for ozone exhaust is located in the roof penthouse. A biological shield of thick concrete encloses the entire arrangement. See Figures 8 and 9 showing the general layout of the irradiator.

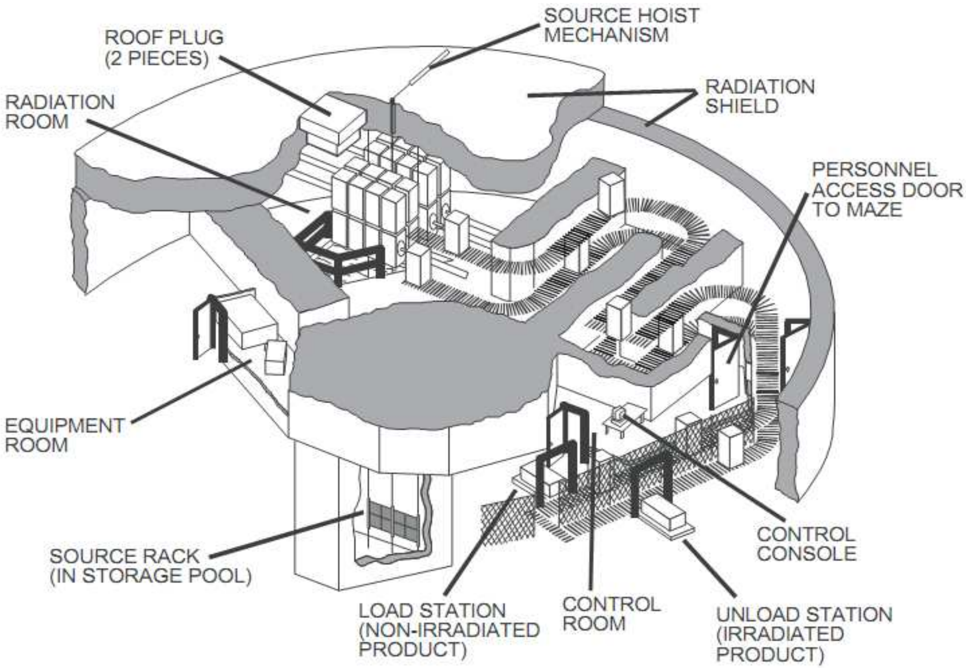

FIG. 8. General layout of the irradiator.

2.5.1. Radiation shield

The radiation shield was designed to reduce the radiation leakage level to the outside from a 111 PBq (3×10^6 Ci) source for an average exposure rate of less than 2.5 µSv/h (0.25 mrem/h). This allows a person to work near the shield for 40 hours per week and limit their exposure to a maximum dose of 0.5 mSv (500 mrem) per year.

Stepped concrete plugs located in the roof are removable to allow lowering of ^{60}Co shipping containers into the source storage pool. See Figure 10.

Personnel and products enter the radiation room through a labyrinth that prevents loss of shield integrity. Personnel enter through an access door. The product is transported by the conveyor system through separate entry and discharge ports, as in Figure 11. These ports are blocked to personnel entry by metal barriers when no product is present.

2.5.2. Source storage pool

The water filled source storage pool in the radiation room contains the source rack. The source rack, loaded with radioactive sources capsules, remains at the bottom of the pool when it is not in use in a safe storage position. The depth of the pool water provides an adequate shield from radiation, and thereby it allows the safe performance of service operations.

To minimize the possibility of corrosion damage to components in the pool, the water circulates through a water treatment plant.

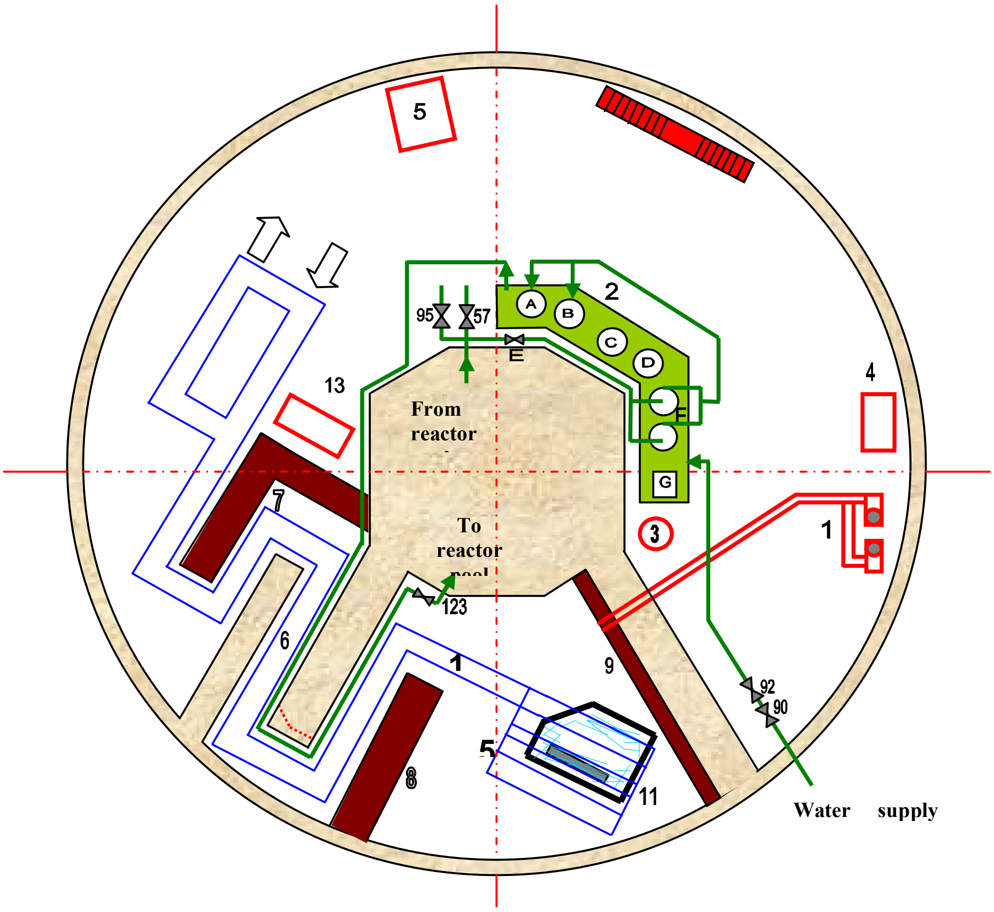

FIG. 9. General gamma irradiator layout in the reactor basement. 1: ventilation system; 2: water reactor deionizer; 3: compressed air supply; 4: main electric supply; 5: irradiation room; 6: labyrinth; 7–9: additional concrete walls; 11: ^{60}Co sources pool; 12: conveyor; 13: control room.

2.5.3. Radiation source mechanism

The hoist mounted on the shield roof raises the source rack when automatic operation starts. The hoist is interlocked with the irradiator control safety system that promptly returns the source to the bottom of the storage pool if any system faults are detected.

2.5.4. Source rack

The radioactive ^{60}Co source material is doubly encapsulated in MDS Nordion type ^{188}C sealed source capsule. These sealed source capsules are inserted into subassemblies called modules after they are assembled in the rigid stainless steel source rack that constitutes the source. The source rack has 2 module rows, each with enough length for 3 modules as in Figure 12. Each module has a capacity of 42 sealed sources, called radioactive pencils.

2.5.5. Source hoist

The source rack is raised out of the safe position in the pool into the irradiation position by a stainless steel cable that runs through the ceiling to a pneumatic hoist.

3. SAFETY ASPECTS OF THE PROJECT

Given the presence of an unusual industrial irradiator within the research reactor structures, a number of considerations relating to the safety of having two application areas in one facility were taken.

The research reactor was permanently shut down. We have designed two systems for processing water in the facility. The first system is used to deionize and maintain the reactor pool and the transfer canal water. The second system was designed to deionize the ^{60}Co source pool water. The reason for this physical separation is due to the fact that the reactor pool and the transfer canal water are contaminated with ^{137}Cs and ^{60}Co, while the irradiator pool of sources is not contaminated with fission and activation products. This helps to avoid transport cask contamination when performing ^{60}Co reloads.

To avoid affecting the reactor foundation, it was necessary to verify the absence of water loss from the irradiator pool. During the excavations in the basement of the reactor, we were careful not to affect the foundations of the reactor. We installed two 60 m^3/min exhaust fans for the removal of ozone that is produced in the irradiation room. External air is drawn into the facility in order to prevent exhaust entrainment and contamination from extracted ozone and other gases.

In the reactor building, a few reactor fuel assemblies remain in wet storage. 74% of all fuel assemblies have already been returned to their country of origin, but still some remain in a portion of the facility external to the irradiator [3]. For security reasons, the fuel assemblies that remain in the facility must be removed.

4. COMMISSIONING AND AUTHORIZATION OF THE FACILITY

The commissioning of the facility started in December 2003. Our regulatory body in charge of the Atomic Energy Direction decided to seek an expert mission by the IAEA in order to evaluate the entire installation, safety and operating permit. This expert mission concluded that there were no technical impediments to operate and deliver the authorization. In September 2004, IVIC received the authorization to operate the facility under a number of requirements. These requirements have been fulfilled for seven years. The authorization has been renewed every two years, and we received regulatory body inspections every six months.

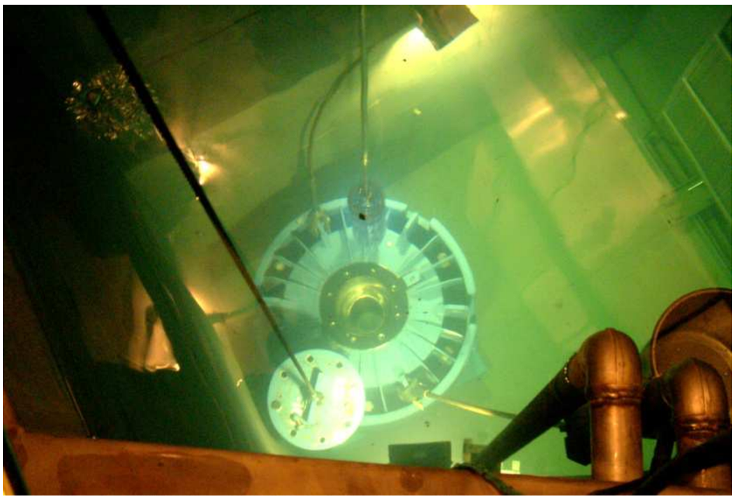

FIG. 10. ^{60}Co shipping container lowered into the source storage pool.

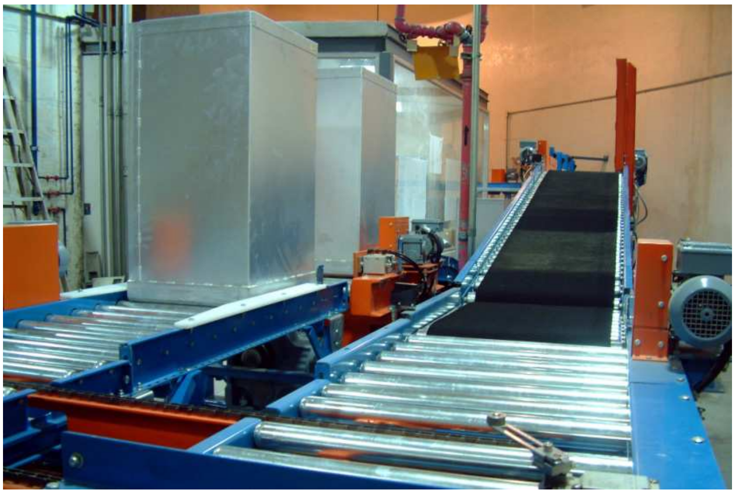

FIG. 11. Conveyor system.

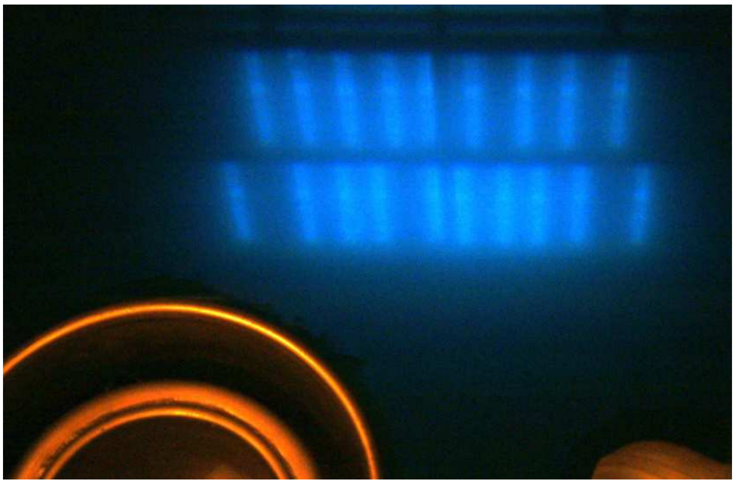

FIG. 12. Source rack with nineteen ^{188}C and ^{60}Co sources.

5. CONCLUSION

After seven years of modifications to our nuclear research reactor RV-1 and having converted it into an industrial gamma irradiator, we state several conclusions.

With a little re-engineering, an obsolete facility can be recovered at modest and for reduced environmental liability.

In order to maintain safety standards, it is important to train personnel in knowledge and expertise about the new technology, irradiation plant operation and nuclear research reactor maintenance.

We believe that the conversion of the reactor into an industrial gamma irradiator is a productive project that generates economic resources to finance reactor maintenance. Also conversion is a viable option when a country does not have resources for reactor decommissioning, a long term radioactive waste repository and reactor maintenance.

Of a total of 617 nuclear research reactors built, 170 have been decommissioned, and 242 are in a shutdown condition [4]; a number that in the future will continue to rise. Our positive experience could be taken into account when considering only two options for a nuclear reactor: to upgrade or decommission.

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ABBREVIATIONS

BAZ	Fast safety protection
BP	Beam port
CEC	Central experimental channel
ECCS	Emergency core cooling system
GTRI	Global Threat Reduction Initiative
HEC	Horizontal experimental channel
HEPA	High efficiency particulate air
HEU	Highly enriched uranium
HFE	Human factors engineering
HRPD	High resolution powder diffractometer
I&C	Instrumentation and control
INSARR	Intergrated Safety Assessment of Research Reactors
ISI	In-service inspection
LEU	Low enriched uranium
LOCA	Loss of coolant accident
MAZ	Slow safety protection
MMI	Man–machine interface
NPP	Nuclear power plant
NRI	Nuclear research installation
PDCA	Plan–do–check–act
QAP	Quality assurance programme
RBP	Radial beam port
SAR	Safety analysis report
SCADA	Supervisory control and data acquisition
SSC	Structures, systems and components
STEC	Split type encirclement clamp
VEC	Vertical experimental channel

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