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Session VI

Research Reactor Operation & Maintenance

KEEPING AGING RESEARCH REACTORS IN GOOD SHAPE

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ABSTRACT

During the 47 years of reactor operation of the TRIGA Mark II reactor Vienna many highly specialized maintenance and inspection methods have been developed which in several cases have been acquired by other research reactors. Further in some cases the TRIGA Vienna team plus equipment was rented for up to two weeks to carry out i.e. reactor tank, cleaning or visual inspections and documentation of tank internals or fuel elements. In addition complete systems such as pool water cleaning systems were tailored according to local needs. These services were either carried out by direct bilateral contracts between the counterpart institution and the Atominstitut or upon request of the IAEA which supported the inspection financially for low-income countries. This paper presents the experience of inspections and maintenance of several research reactors and gives some recommendations about the optimal maintenance frequency of research reactor tank internals in order to keep low power research reactors in good condition.

1. Introduction

The TRIGA Mark-II reactor started up initially on March 7, 1962, with a steady state power of 250 kW and with pulsing capability up to 250 MW. Within the past 47 years no major incidents occurred, however, a number of reconstructions and modifications of reactor systems were carried out. Since the implementation of the Atomic Law in the mid 1970 a detailed re-inspection plan had to be prepared covering all reactor related components and systems to be re-inspected regularly. The completed reinspection forms are controlled by a government appointed expert and are the basis for the continuation of the operating license [1].

One major issue was the renewal of the TRIGA reactor instrumentation in 1992 when the old transistor type instrumentation was replaced by a computer controlled up to date instrumentation. Since this time experience has accumulated with this digital instrumentation which will be presented in this paper.

Another important task is the periodic optical inspection of the reactor tank internals and the regular cleaning of the primary water system. Optical inspection is carried out with a rigid underwater endoscope. This is a modular optical device which allows optical inspection in any place of the reactor tank, including the fuel elements in the core. With integrated lights and various objectives, 0° foreward, 45° foreward and 90° sidewards practically all areas in the reactor tank can be inspected. This endoscope can also be used to inspect a spent fuel element in a special lead container placed in the reactor hall. The spent fuel is transferred from the reactor tank into this container and through several holes the endoscope can be inserted to view directly the fuel surface without being exposed to radiation.

Regular cleaning of tank internals in three months intervals is also very important. A high pressure water jet is used to stir up all deposits from tank surfaces and a special pump with integrated filters is used to collect the deposits.

The endoscope together with the water jet and the tank cleaning pump has been applied succesfully in several other research reactors through bilateral cooperation and assistance. In fact in one case the visual inspection and maintenance saved the operator a tedious repair work of several monthes or a possible permanent shut-down.

The fuel elements are measured with an underwater device for elongation and bowing every two years. Since 1962 out of 104 fuel elements only 8 had to be removed only one due to a cladding defect, the others due to excess elongation. A dry spent fuel storage has been developed to accommodate the removed fuel elements in a controlled atmosphere.

2. Special inspection and maintenance equipment

2.1 Underwater endoscope

The most important in-service inspection equipment is a modular underwater endoscope. It consists of seven 1 m long rigid endoscope modules which are watertight and can, therefore, be inserted directly into the reactor tank water. As the diameter of the system is only 18 mm it can practically be inserted into empty fuel element positions and allows therefore inspection inside the core volume. In many cases it can even be lowered through the lower core support plate (grid) and allows to view the volume below the core. The front end of the endoscope is equipped with an integrated lamp and several viewing angles are possible like 0º, 45º foreward, 90º, 45º backward. To the ocular standard video- or photo equipment can be connected [7-9].

2.2 High-pressure water jet

To clean the tank and the core structures from debris a compressor is used producing a water jet which can be regulated up to 100 bar pressure. The water is taken directly from the tank, compressed and ejected through various types of nozzles (flat, rotary, point direction) to the surfaces to be cleaned. Using a special small nozzle the water jet can be introduced directly into the core volume between top and bottom grid.

2.3 Tank cleaning pump

While the high pressure jet is used, an additional tank cleaning pump is operating with several stages of gross and fine filters. The pump inlet tube is directed to the area of highest debris. All materials are collected in the filters and the cleaned water is returned into the tank. The overall cleaning of a typical TRIGA tank with a typical amount of stirred-up debris takes about 24 hours. During a recent tank inspection a number of washers, screws and metal pieces were removed, some of them with a dose rate up to 0.1 Sv/h (10 rem/h).

2.4 Pick-up tool

To pick up flat or fine objects from the tank bottom (such as coins, washers, buttons) a special pick-up tool was developed at the Atominstitut which acts on a string-and-pull system. The tool is so small that it can be transported in a shoe box. It can pick up items from as far as 10 m below the water surface.

The above mentioned equipment (endoscope, water jet, tank cleaning pump) can easily be transported to any reactor station in Europe with costs of approximately € 1000.-- round trip on road. If the more bulky high pressure water jet pump and the tank cleaning pump are omitted, the endoscope itself can even be shipped by ordinary mail or transported in a passenger car.

All components and systems are re-inspected following an elaborate re-inspection program [2]. This consumes about 4 man-days per month. Once a year all the reactor systems are inspected in presence of an expert nominated by the regulatory body and his expertise is the basis for the annual renewal of the operation license valid again for the coming year. This annual inspection requires approximately 1 man-month (four persons for two weeks). Some of the inspection methods have been successfully applied in other TRIGA reactors [3-6].

3. Recent applications of the endoscope in European research reactors

Pavia, Italy: Upon request of the 250 kW TRIGA reactor in Pavia, the equipment was used to, identify damaged core installations such as the regulating rod guide tube fitting to the lower grid plate and a deformed central irradiation tube (details see below).

Munich, Germany: The endoscope was used to identify a leak in the primary coolant pipe of the 4 MW MTR type reactor and helped to supervise and control its repair.

Imperial College, UK: The endoscope was used during a general inspection and clean-up of the 100 kW CONSORT reactor.

Rome, Italy: Upon request from ENEA a contract was signed between ENEA/Rome, Italy, and the Atominstitut/Vienna, Austria, to carry out the following tasks

- Visual inspection of the TRIGA RC-1 tank
- Maintenance and cleaning of the reactor using special tools
- Removal of objects found during inspection
- Preparation of a final report

Kinshasa, Democratic Republic of the Congo: Visual inspection and verification of the spent fuel of the TRIGA-I facility, CREN-K upon request from the IAEA.

Before the start of visual inspection, the operations staff from the TRIGA-I facility was able to start and operate the mobile purification system supplied by the IAEA one year ago. It was then possible to clean the pool water of the TRIGA-1 facility. After one hour of purification system there was a visible improvement in the water clarity due to a decrease in turbidity. The operation staff of the counterpart then began to clean up small objects and debris from the bottom of the reactor tank and the three fuel storage racks. It should be noted that the reactor ‘tank’ of the TRIGA-1 consisted of a poured concrete circular wall and base. The visible condition of the concrete walls and base appeared to be excellent.

As the inspection equipment with the underwater endoscope had not been delivered from the customs office by the beginning of the mission, a different approach to inspect and identify the fuel elements stored in the pool of the TRIGA-1 facility was tried. Together with the operation staff from TRIGA-I facility each fuel element was individually lifted about 1-2 m with the fuel element handling tool and transferred into an aluminum bucket. The fuel element and bucket assembly was then raised to an elevation where the top of the fuel element was just below the water surface of the TRIGA-1 pool. The fuel element and aluminum bucket assembly was then rotated until a handheld digital camera picture of the serial number could be taken, see figure 1.

As all the elements, except one, had the number written up side, the identification could only be done afterwards on the screen of the digital camera by rotating the picture. After this was done the surface of the corresponding fuel element was visibly viewed, by lowering the aluminum bucket beneath the fuel element. A picture of the fuel element was then taken, see figure 2.

Fig 1. Picture of the fuel element 539 E

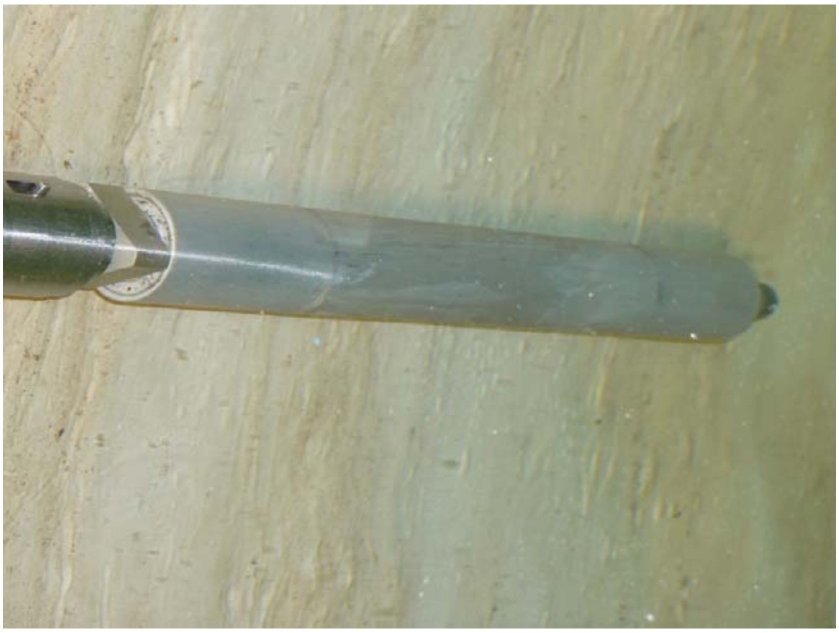

Fig 2. Surface of the corresponding fuel element

Gamma radiation levels were then measured at a fixed distance, about 50 cm, from every fuel element. These measurements then provided a crude measure of relative burn-up of the elements. No previous information on element burn up was available. Each fuel element was then transferred back into a storage rack position at the bottom of the pool. The elements were replaced into their original rack storage locations unless reinsertion proved difficult.

All of the R2 fuel elements were in good condition, with no visible corrosion spots or mechanical damage. Some elements had a visible black or white discoloration at the axial interface locations between the fuel and the top and bottom axial graphite slugs. The discolorations are normal and are attributed to the heat flux variations of the aluminum cladding at these interfaces. No corrosion or mechanical defects of the cladding were observed. Every fuel element was verified to be in excellent condition with no evidence of any mechanical deformation or defect.

After the inspection was finished, several screws, wires and electrical insulation material, from the former control rod drives, were removed from the bottom of the pool with the pick-up tool.

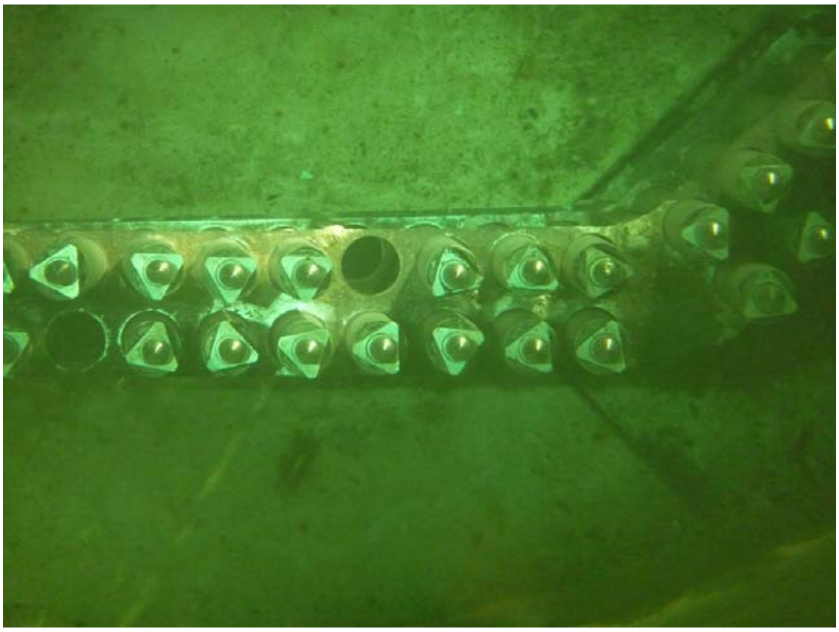

Fig 3. Storage racks located at the bottom of the pool

4. Conclusions

It is obvious that careful maintenance and periodic in-service inspections of the research reactor components have a positive influence on the technical state of the reactor and may extend its lifetime considerably. Reactor facility life extension is best accomplished by establishing and completing a maintenance program at an early stage in the facility's operation. However, high quality routine maintenance of all reactor safety systems and operation within the established technical specifications is also essential to ensure the safety of the reactor and the public. During the past 47 years maintenance and inspection methods have constantly improved and new methods with digital systems have been developed. Together with an elaborated in-service inspection program the TRIGA reactor Vienna and all the other low power reactors around the world could be kept in excellent technical state without any major ageing effects. It is hoped that this facility will "still be going strong" for many more years.

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A STRATEGY FOR MANAGING AGEING COMPONENTS OF A SLOWPOKE REACTOR

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ABSTRACT

The Montreal SLOWPOKE reactor has been operating for 33 years and it is planned to continue operations for at least another 20 years. After the first 21 years, the HEU fuel was replaced with a new long-life LEU core. A strategy is presented for managing the ageing components of the reactor core, the control system, the pool and other auxiliary systems. It deals with problems such as materials degradation, corrosion and the replacement of obsolete components. The strategy includes close surveillance of the fuel, with weekly maintenance tests, yearly inspections of the pool and its contents, and the acquisition of a large stock of spare parts. The strategy also deals with ageing documentation and ageing operating personnel.

1. Introduction

1.1 The SLOWPOKE reactor

The eight SLOWPOKE-2 reactors constructed in the 1970's and 1980's were designed for a lifetime of 20 years [1], with a maximum operating power of 20 kW and a mean power of about 1.4 kW. The five in Edmonton, Saskatoon, Kingston, Montreal and Halifax, Canada and the one in Kingston, Jamaica are still in operation and four of them will likely continue operating for more than 50 years. To achieve inherent safety, the maximum installed excess reactivity is limited to 4 mk. As the reactor is used, the excess reactivity decreases, due to the production of fission product poisons and the burn-up of U-235, and it is periodically brought back up to 4 mk, not by the addition of new fuel, but by adding beryllium shim plates to the upper reflector, see Fig. 1.

The Montreal reactor was one of the more heavily used, and by 1997, after 21 years of use, the beryllium shim tray was full and it was necessary to replace the fuel. The high enriched uranium (HEU) core was replaced [2] with a low enriched uranium (LEU) core. In the original HEU core, 7 mk of beryllium plates were used just to attain criticality, leaving only 13 mk for future reactivity consumption with reactor use. The long-life LEU core is made critical with no beryllium in the upper reflector, leaving space for about 20 mk of beryllium for future use. With 50% more available reactivity, the LEU core is expected to operate even more than 50% longer than the HEU core at the same rate of use, because fission product poisons reach saturation after the first few years of operation. It is now predicted that the Montreal LEU core will last until 2030 at the current rate of use. The life of the reactor can be further extended about another six years by the addition of another beryllium annulus above the main annulus shown in Fig. 1.

Some of the SLOWPOKE facilities have made major upgrades over the years. The second beryllium annulus was added to the Toronto (decommissioned in 2000) and Halifax reactors around 1990. The Halifax reactor will soon decommission also and the two beryllium annuli will be available for re-use at other facilities. The facility at Royal Military College, Kingston, Ontario, Canada replaced its electro-mechanical reactor control system with a digital system designed in-house [3]. The facility at Royal Military College and the one at Saskatchewan Research Council, Saskatoon, Saskatchewan, Canada have both recently replaced their pool water purification systems with a modern treatment system [4,5]. The Montreal facility still has its original systems: the electro-mechanical reactor control system, incorporating a self-powered neutron detector, a voltage comparator and a motor-driven control rod, has proven to be extremely reliable and it is not planned to replace it.

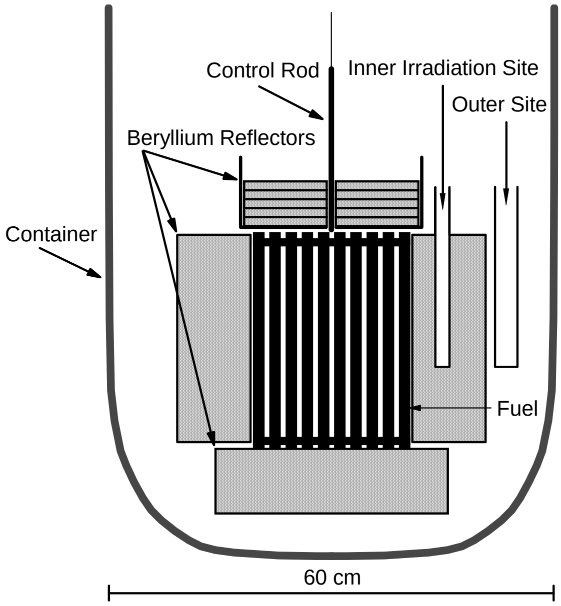

Fig 1. The SLOWPOKE reactor core

1.2 Ageing

Ageing is defined as a general process in which characteristics of components, systems and structures gradually change with time or use [6]. This process eventually leads to degradation of materials subjected to normal service conditions. These include normal operation and transient conditions under which the component, system or structure is required to operate. The effects of such degradation may result in the reduction or the loss of the ability of components, systems and structures to function within accepted criteria. Safety and utilization of the facility may be affected unless preventive measures have been taken, and corrective measures have been established.

At a SLOWPOKE facility, ageing management is necessary to keep the equipment running safely and effectively for the foreseen lifetime of the reactor. It is also a requirement imposed by the regulator, the Canadian Nuclear Safety Commission.

1.3 Ageing experience at SLOWPOKE facilities

We will describe four occurrences, related to ageing, which have occurred at the Montreal facility, and possibly other SLOWPOKE facilities, and mitigating actions which were taken:

1. Early in the history of the Montreal reactor, a high level of humidity was observed above the pool under the concrete cover. There was noticeable condensation on several components. A ventilation system was installed to reduce the humidity.

2. With the aluminium-clad HEU fuel, it was observed that fission product releases from the fuel to the water increased over the years. The visual inspection of the HEU fuel of the Montreal reactor in 1991, with an underwater camera, revealed swelling on several fuel pins. The increased releases of noble gas fission products may have been through minute fissures in these swellings. It is speculated that the temperature cycling associated with daily reactor operation may be the cause of this degradation of the fuel. To reduce the releases of fission products, the Montreal reactor ceased operating at full power, 20 kW, and continued at 10 kW or less from 1991 to 1997, the last six years of operation with the HEU fuel.

3. At the Montreal facility, corrosion formed on one of the contacts of the power selector switch, which is part of the reactor control system, and in 1991 it was the cause of a power excursion up to 40 kW. Although such a small power excursion poses no immediate risk, it is undesirable because it may accelerate ageing of the fuel. Now, all SLOWPOKE facilities have a maintenance program for this switch which eliminates the accumulation of corrosion.

4. Upon inspection, it was found that some of the joints between the aluminium irradiation tubes coming out of the reactor container and the polyethylene tubes, which transport irradiated capsules about the laboratory, were not airtight. In the event of a rise in the level of the pool water, a leak at these joints would allow water to enter the irradiation sites in the reactor. The risk of a rise in the pool water level may be increasing as the ageing pool cooling system may be developing corrosion and approaching failure. The joints have now been sealed and are the subject of periodical inspections.

2. The strategy for managing the ageing components

The overall safety objective is to protect individuals, society and the environment by establishing and maintaining an effective defence against radiological hazards. The SLOWPOKE reactor is inherently safe [1] and the total inventory of radioactive fission products in the core is relatively small for a research reactor, of the order of 10 TBq, and no mechanism is known by which significant quantities of radioactivity could be released. Thus, the achievement of the safety objective is relatively easy with this reactor. As a consequence, the strategy for ageing management is not motivated so much by safety concerns, but rather by the desire to keep the reactor operating usefully as long as possible. Of course the strategy must always maintain the inherent safety, mainly by preserving the multiple barriers, the fuel matrix, the fuel cladding, the container and the pool, which prevent the release of fission products from the fuel to the environment.

2.1 Basic Principles

The management of ageing includes two types of measures:

- Preventive maintenance
- Mitigation

Preventive maintenance is essentially maintaining the optimum environment and operating conditions of the equipment to delay ageing. It includes inspections designed to detect changes in the environment and operating conditions of the equipment which might accelerate ageing. When these changes are detected, they are remedied by preventive maintenance. Preventive maintenance may also include refurbishment and the replacement of equipment even before it shows signs of ageing.

Mitigation includes measures undertaken to counter the effects of ageing once they have been detected. It includes inspections designed to detect ageing as early as possible. It also includes corrective maintenance on the ageing equipment or replacement of components.

2.2 The strategy at the Montreal SLOWPOKE facility

The following are the seven facets of our strategy for the management of ageing:

Along with components and materials, the reactor documentation may also age, in the sense that it may become out of date. We have recently developed a formal Quality Assurance Programme that ensures that all reactor documentation is kept up to date. According to the QA programme, the Operating Manual is continually revised and the procedures in it, especially the maintenance procedures, are continually updated. The QA programme also ensures that the operating personnel are knowledgeable of modified systems and the associated documentation. The operating personnel also follow a documented Continuing Training Programme.

The ageing of the operating personnel should also be considered. At the Montreal facility, a training programme for new operators was developed and two new reactor operator candidates have now almost completed their training. It was found that an overlap of a retiring operator and his replacement of at least one year is necessary.

The most important goal of the ageing management strategy is to preserve the integrity if the reactor fuel, because if any fuel degradation occurs it cannot be repaired. When the reactor was commissioned with LEU fuel in 1997, it was necessary to carry out power excursions to demonstrate the self-limiting behaviour of the reactor, but these were limited to powers less than 80 kW to minimise their possible effect on the fuel. The mechanism which caused the degradation of the aluminium-clad HEU fuel is not expected to occur with the zirconium-clad LEU fuel. No sign of degradation has been observed after 24 years of operation of the Kingston, Canada reactor with identical LEU fuel. The most important aspect of the preventive maintenance is to maintain the purity of the reactor water. It is verified weekly and the resistivity is maintained above 2 Mohm-cm. The radioactivity of the reactor water is also measured each week. An increase in the concentration of gaseous or solid fission products in the reactor water will be the first indication of fuel degradation.

The reactor control system, the pneumatic irradiation systems, the radiation monitoring systems, the pool cooling system and the purification systems for the reactor water and the pool water have many components that are more than 30 years old. It is generally accepted [6] that the risk of failure of such components increases with age, and it may be wise to replace them before they fail to reduce the risk. We do not totally agree with this logic, for two reasons. First, we know our reactor well enough to be confident that the failure of a component of any of these systems will not lead to an unacceptable situation from the point of view of safety. Second, we have found these systems to be acceptably reliable up to now and we are not convinced that new components will improve reliability. Therefore, our strategy is to continue operating these systems until failures occur. Then the failed components will be replaced. The reactor may need to remain shut-down until the replacement is completed. In order to minimise down-time, our strategy is to ensure that the eventual repairs can be completed as quickly as possible.

Many of the original components are now obsolete and it may take considerable time to find or have manufactured suitable replacements and then to install and verify them. For critical components, such as the temperature recorder and the self-powered neutron detector of the control system, replacements were purchased several years ago and are available. A network, with the participation of the five operating SLOWPOKE reactors in Canada, has been set up. Each is drawing up a list of spare parts on hand and has agreed to make them available to the other facilities as needed. As many spare parts as possible have been salvaged from two SLOWPOKE reactors that have already been decommissioned and more will be obtained from the next reactor to decommission. Most of the parts will be stored at the Montreal facility.

The pool and the support structure for the reactor container are large structures which would be difficult to repair in the case of a major failure. Each week, the exact amount of makeup water added to the pool to compensate for evaporation is noted. This would be the first indication of a minor leak, which, hopefully, could be repaired before it became a major leak. In 2008, a programme of annual inspections of the pool was begun. The concrete pool cover is withdrawn and the following components are inspected: the pool wall, the outside of the reactor container, the irradiation tubes, the tubes leading to the pool cooling coil, the support structure. The onset of corrosion was discovered on one of the I-beams of the support structure, likely caused by condensation dripping from the tubes of the cooling system. Remedial measures are being taken to stall the corrosion.

This strategy was developed with the experience acquired over thirty years of reactor operation. It could be improved with the additional information on operational experience available at the other SLOWPOKE facilities. In 2009, each facility will be asked to compile a list of their complete operating experience, including all operational occurrences or other experience that may be related to ageing. The compounded list will benefit the ageing management strategies of all the facilities.

This strategy has not yet been fully documented in accordance with to our QA programme. Once this is done and the strategy is fully implemented, we are confident that our facility will eventually be shut down, not because of the effects of ageing, but because the reactor is no longer useful or has exhausted the available reactivity.

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Irradiation Growth of Graphite in Reflector Elements of JRR-4

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ABSTRACT

A crack was found on the weld of aluminum cladding of a reflector element of graphite in Japan Research Reactor No.4 (JRR-4). A survey on the reflector element confirmed that the crack was caused by a growth of the graphite reflector. The growth was observed in other reflector elements by radiographic testing. We measured the dimensions of irradiated graphite reflectors after dismantling the reflector elements. We found that the sizes had increased with fast neutron fluence under the JRR-4 operation condition which was estimated lower than 200 ˚C.

1. Introduction

Japan Research Reactor No.4 are used for medical irradiation (Boron Neutron Capture Therapy), education and training for engineers, activation analysis and researches in various fields. The JRR-4 is a 3.5 MW pool-type reactor which is cooled and moderated by light water. The JRR-4 core consists of fuel elements, reflector elements, control rods and irradiation tubes. The top view of the JRR-4 core is shown in Fig.1. The reflector element is composed of reflector part, handle part, joint part and plug part as shown in Fig.2, all of which are made of aluminum alloy except for the graphite reflector set in the aluminum cladding. The graphite material is a fine-grained and isotropic one (IG-110) manufactured by Toyo Tanso Co.. The reflector is a rectangular shape of 691x72x72 mm.

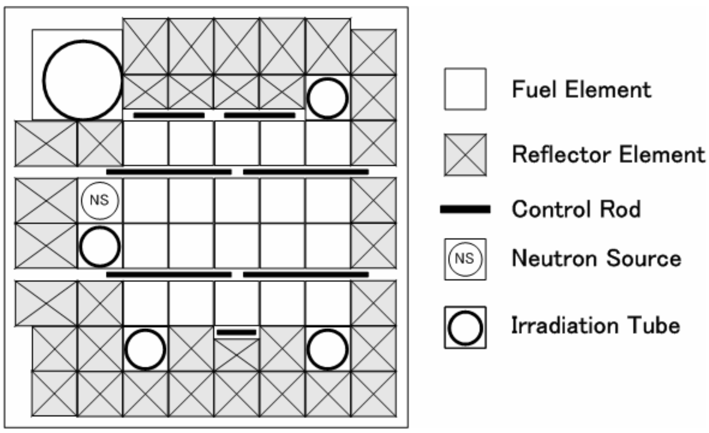

Fig.1. Top view of JRR-4 reactor core.

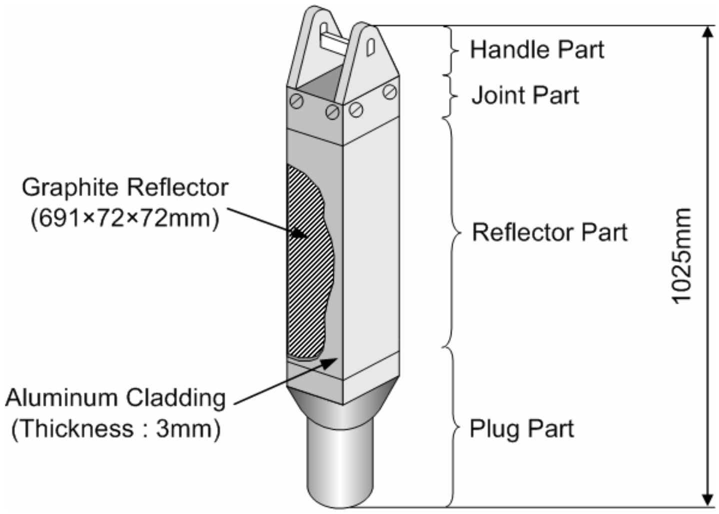

Fig.2. Reflector element of JRR-4.

The crack was found on the weld of the aluminum cladding on December 28th, 2007. A photograph of the cracked reflector element is shown in Fig.3. We investigated the reflector elements by visual examination, dimensional examination, fractography examination, etc. As the result, the main cause of the crack was concluded by growth of graphite reflector due to fast neutron irradiation. The growth was about 7 mm (dimensional change: about 1 %) in longitudinal direction. An excessive stress broke the weld of aluminum cladding, since the dimensional change exceeded the gap of 4 mm between top of the graphite reflector and the joint part. In the next phase, we carried out a radiographic testing of verifying other reflector element. This result showed that the most of the graphite reflector were grown in the aluminum cladding.

When the cracked reflector element was designed, the gap size of graphite reflector was designed based on the irradiation data of nuclear grade graphite material of which irradiation temperature is above 350 ˚C. These irradiation data show that irradiation growth of about 0.05 % happens only on the early stage of irradiation, while irradiation shrinkage proceeds with fast neutron irradiation by 10^{25} n/m^2 after the early stage. The irradiation data of the IG-110 graphite material indicates that the irradiation shrinkage is caused due to the fast neutron irradiation at high irradiation temperatures above 600 ˚C [1]. Irradiation behaviour of anisotropic graphite was reported that the growth was observed in perpendicular to the direction of extrusion at 225 ˚C and 250 ˚C [2].

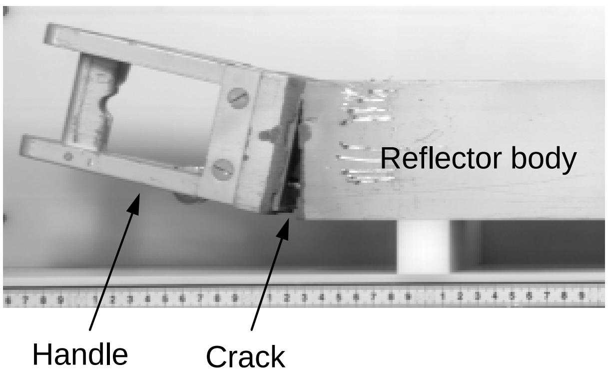

Fig.3. Crack of the reflector element.

2. Measurement of the irradiation growth

In order to investigate the low temperature irradiation behaviour of the IG-110 graphite, we measured the dimensions of the graphite reflectors after dismantling the aluminum cladding. We picked up 13 graphite reflectors for this investigation. The dimensional changes were obtained from the difference between measurement at manufactured period and measurement at present. The fast neutron fluence was determined by multiplying the fast neutron flux by the irradiation time under an equivalent power of 3.5 MW.

2.1 Measurement of the dimensional change

Dimensions of graphite reflectors were measured at the four sides on the cross-section. There might be a difference among dimensional changes, since the neutron irradiation fluence on the fuel side is stronger than that on the other side due to the fact that neutrons are scattered and absorbed in the graphite during passing the graphite reflector. We, then, took the average of the four data. This measurement was carried out at 11 points aligned in a longitudinal line at regular intervals for each graphite reflector. There must be broad peak in the distribution of fast neutron irradiation fluence along the longitudinal direction, the middle of the graphite has been irradiated more than the close to top and down area.

2.2 Estimation of the fast neutron fluence

The flux of fast neutrons (E>0.18 MeV) at each measurement point was estimated by using the SRAC code and the Monte Carlo calculation (MCNP5) [3, 4]. The SRAC code was developed by JAEA (Japan Atomic Energy Agency). The fast neutron fluence at every measuring point was obtained by multiplying the fast neutron flux by integrated irradiation time on an equivalent power of 3.5 MW.

3. Results

We measured the dimensions of all 13 graphite reflectors. The cross-section average between the dimensional change and fast neutron fluence as a function of the distance from top of the graphite reflector are compared in Fig.4 (a) and Fig.4 (b). In the case that the dimensional change was within the gap size (1 mm or 1.5 mm) between graphite and the aluminum cladding, the results of all measurement locations showed a good coincidence between the dimensional change and the fast neutron fluence. While, in the other case that the graphite obviously touched the aluminum cladding, the dimensional change was deformed around its peak and showed a big difference from the fast neutron fluence. This result indicates that the inflation of the graphite reflector was retarded and deformed by aluminium cladding wall.

The dimensional changes of the graphite reflectors are plotted in Fig.5. As the result of the evaluation under the JRR-4 operation condition, the relations between the dimensional change of the graphite and fast neutron fluence were the following. The maximum dimensional change was 1.9 % at the fast neutron fluence of 5.4×10^{24} n/m^2. The maximum irradiation growth per fast neutron fluence (irradiation growth ratio) was 7.13×10^{-25} %m^2/n, the minimum 4.21×10^{-25} %m^2/n and the average 5.71×10^{-25} %m^2/n in the region of fast neutron fluence below 2.5×10^{24} n/m^2. To get these results, a threshold of the fast neutron fluence was set, since the effect on the collapse of the graphite was supposed in the region of fast neutron fluence above 2.5×10^{24} n/m^2.

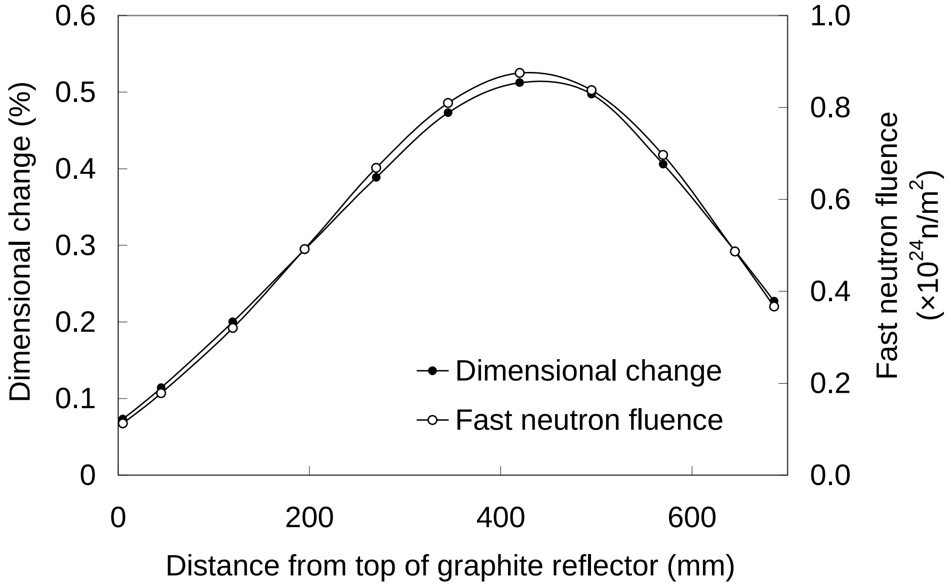

Fig.4 (a). Comparison of the cross-section average between the dimensional change and fast neutron fluence as a function of the distance from top of the graphite reflector (In the case that the irradiation growth was within the gap size).

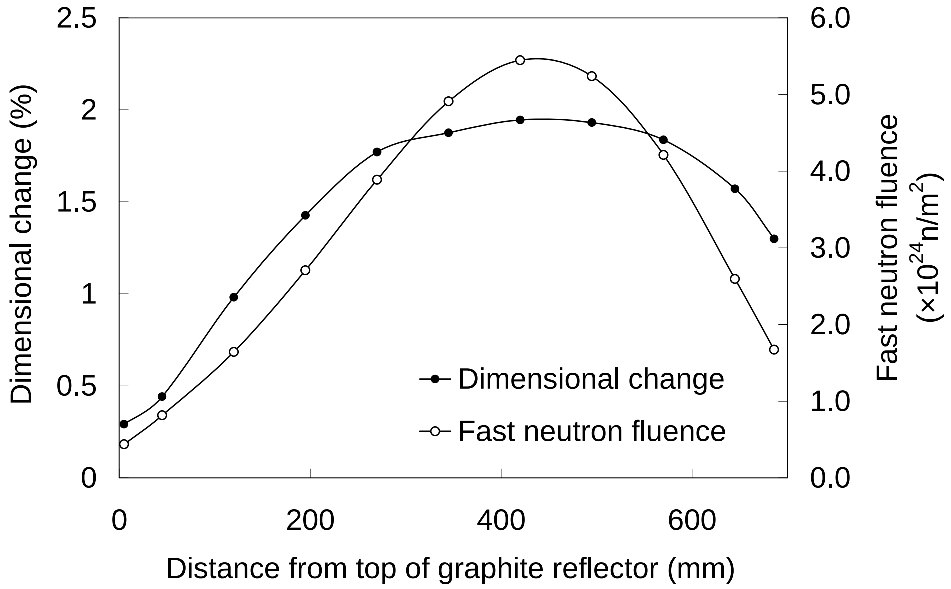

Fig.4 (b). Comparison of the cross-section average between the dimensional change and fast neutron fluence as a function of the distance from top of the graphite reflector (In the case that a part of the graphite reflector obviously pushed the aluminum cladding).

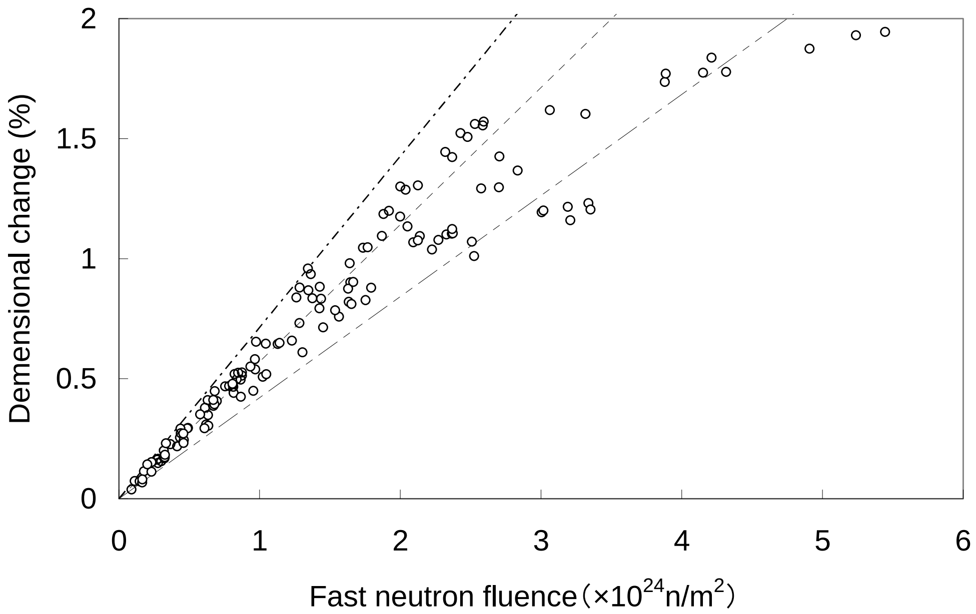

Fig.5. Dimensional change of the graphite under the JRR-4 operation condition as a function of the fast neutron fluence.

4. Conclusion

We found that a large irradiation growth happened to the graphite reflectors under JRR-4 operation condition, of which temperature was lower than 200 ˚C. The survey on the reflector elements confirmed that the crack was caused by irradiation growth of the graphite material by fast neutrons. The irradiation growth ratio was 7.13×10^{-25} %m^2/n in the range of fast neutron fluence below 2.5×10^{24} n/m^2. We will take this result into account in the future reflector design.

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SAFETY CONSIDERATIONS FOR CORE MANAGEMENT AND FUEL HANDLING FOR RESEARCH REACTORS

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ABSTRACT

This paper addresses the safety considerations for the core management and fuel handling for research reactors. These considerations are associated with core calculations, including neutronic and thermal-hydraulic computational methods and analysis, safety requirements related to core configurations, core operation and monitoring, and refuelling process. The safety aspects of handling of research reactors fresh and irradiated fuel are also discussed. All these issues are presented and discussed on the basis of the relevant IAEA Safety Standards.

1. Introduction

During the lifetime of a research reactor, the core configuration is regularly changed for various reasons. These changes in the core configurations are mainly to compensate for fuel burn-up and to meet the utilization needs. A reactor core configuration represents a given layout of fuel elements, reactivity control devices and neutron absorbers, moderator, reflector elements, fixed experimental devices, and in-core instrumentation. Any change in the specifications, position, or number of any of these items, as applicable, is considered as a change in the core configuration. This change must be accommodated in a core management process to ensure compliance with the design intent and assumptions and with the Operational Limits and Conditions (OLCs) as derived from the reactor safety analysis. The activities of research reactor core management include core calculations, core operation and monitoring, and the refuelling process [1].

Fuel handling activities include loading, unloading, transfer, storage, packing and transport of fresh and irradiated fuel. Due to their safety implications, these activities must be subjected to operational constraints and administrative conditions that are usually established by limiting conditions for safe operation as a part of the OLCs [2].

The activities of core management and fuel handling should be performed in such away that ensures safe management of the operational cores throughout the research reactor lifetime and avoids mishandling of fuel and other core components which may lead to inadvertent criticality, overheating, mechanical damages or other kind of failures. The following sections present and discuss, on the basis of the IAEA Safety Standards [1, 2, 3, 4, 5], the safety considerations associated with these activities.

2. Core calculations

The objective of the core calculations is to determine the physical parameters of a proposed core configuration to ensure their compliance with the OLCs. These calculations should be verified by measurements before power operation with the proposed core configuration. The calculations should be performed for the steady-state operations and for anticipated operational occurrences and should include neutronic and thermal-hydraulic calculations using validated computational methods and codes. The uncertainties in the core calculations should be taken into consideration as these uncertainties may have significant influence on the final results. The uncertainties in the modelling process and those resulting from other sources (e.g. fuel fabrication tolerances and deviations in the operational conditions) should be treated using a conservative approach.

The loading pattern, including the locations of fuel and reflector elements, and burn-up value for every fuel element (to ensure that a fuel element is unloaded before its specified maximum burn-up has been reached) are basic parameters in the calculations of a proposed core configuration. The criticality parameters should be also determined. The calculations should evaluate the excess reactivity of the proposed core configuration and the expected changes in its value (due to fuel burn-up, build-up of fission products, and temperature reactivity feedback effect) during the planned operating cycle and should predict the associated movement of the control rods. This evaluation is essential to ensure that there is a sufficient margin for control at all times for shutting down the reactor safely and for keeping it in a safe shutdown conditions.

The safe operation of a research reactor also depends on the effectiveness of its control rods which varies significantly from one core configuration to another. For every proposed core configuration, the reactivity parameters related to control rods must be determined by calculations and verified by measurements. These parameters are the maximum reactivity insertion rate and reactivity worth of each control rod and the reactivity shutdown margin, including its value with the failure of the control rod of the highest reactivity worth, for all operational states. For research reactors that have a second shutdown system (e.g. drainage of the moderator or injection of a neutron absorber), the core management calculations should cover the shutdown capability of that system. The core calculations should cover an evaluation of the effect of the control rods positions (including their positions relevant to each other) on the neutron flux spatial distribution and neutron detectors. These calculations should also evaluate the degradation of the absorbing material of the control rods along operation lifetime of the reactor.

The power peak factor should be determined for any new core configuration. The calculation of this parameter should consider the effects of control rod and reflector positions, and existence of flux-traps. In some complex configurations, it may be necessary to perform measurements to verify that the value of this parameter remains below the limits specified in the OLCs. The power peak factor is used as an input to the thermal-hydraulic core calculations. These calculations should demonstrate adequate safety margins against the thermal-hydraulic critical phenomena for the reactor normal operation and anticipated operational occurrences.

The values of the reactivity feedback coefficients, and reactivity worth and effect of the moderator and reflector elements should be determined by calculations and verified by measurements for the first core. Re-calculation and re-measurements of these parameters may not be necessary for every core configuration change. The reactivity worth of the in-core and in-reflector experimental devices should be determined by calculations and verified by measurements before further reactor power operation. Any proposed change in the location and specification, including materials to be irradiated, of any of these devices should also be analysed from the safety point of view. The effect of the experimental devices on the neutron flux spatial distribution and neutron detectors should be evaluated.

Reactor core fuel conversion from highly enriched uranium (HEU) to low enriched uranium (LEU) is also a core configuration change. However, due to its major safety significance additional safety considerations to those discussed above should be taken. Detailed discussions of the safety aspects for research reactor core conversion from HEU to LEU are provided in Reference [6].

3. Core operation and monitoring

The core management activities include operation of the reactor in accordance with the design intents and conditions as specified in the OLCs. The core operation should be performed in accordance with approved procedures, which include precautions that are necessary for maintaining safe core operation. The operating procedures related to safe core operation include:

- Core configuration change;
- Reactor start-up, operation, power level changes, and shut down;
- Control rod calibration;
- Determination of the excess reactivity, reactivity shutdown margin, and reactivity worth of experimental devices and materials to be irradiated;
- Handling of fuel elements (including failed ones) and core components;
- Determination of the reactor thermal power;
- Determination and adjustment of the safety system settings;
- Performance of routine checks of reactor operation and status of the systems and components.

Monitoring of the reactor core parameters and conditions provides for verifying that the reactor operation is conducted in accordance with the OLCs. The parameters to be monitored or verified include:

- Reactor thermal power;
- Reactivity as a function of the control rod positions;
- Control rods drop time, moderator or reflector dump time;
- Reactor water level;
- Pressure difference across the reactor core, coolant flow rate, coolant temperature at the core inlet and outlet;
- Margins to the thermalhydraulic critical phenomena (derived);
- Fuel temperature (it may be derived from other measured parameters);
- Radioactivity contents in the primary coolant water;
- Physical and chemical parameters of the coolant and moderator.

Integrity of the fuel elements is one of the important parameters to be monitored along reactor operation. Some research reactors use delayed neutron detectors located in the primary coolant flow for on-line monitoring of fuel cladding. Other research reactors may use methods based on detection of the fission products in the coolant or in off-gas from the coolant. In addition, activities such as checking, testing and inspection within an in-service inspection programme should be established for early detection of any deterioration (e.g. bowing, dimensional change, etc.) of the fuel. Failure contingency procedures of the fuel should be established to ensure identification and removal of failed fuel from service, determination of the root cause of the failure, and implementation of the necessary corrective actions that prevent re-occurrences of such events.

4. Core refuelling

The details of the core configurations throughout the reactor lifetime and a schedule for movement of the fuel elements and core components are defined by the refuelling programme. This programme should be developed in the design stage and be subjected to review for further improvement based on the experience acquired from reactor operation and on the changes in the utilization programme. It involves shuffling of the fuel through the core in a predetermined pattern to provide sufficient reactivity to compensate for fuel burn-up and build-up of fission products. The refuelling strategy should be established to achieve a uniform burn-up of the fuel within the bounds of burn-up limitations, which also enhances the fuel economics. For Safe and effective core management this strategy should provide for maximum flexibility for reactor utilization and an optimum use of the fuel without jeopardizing safety.

The main operations related to the refuelling process are loading and unloading of fuel elements and other core components using dedicated operation tools and equipment (which should be checked prior to their use in refuelling operation), storage of unloaded fuel elements and core components, and verifications (by checks, testing, or measurements) that the core has been correctly configured. These operations should be performed in accordance with approved procedures.

During the execution of the refuelling operations, all instrumentation that are required to monitor the evolution of the neutron flux, reactivity changes, and integrity of the fuel elements should be operational. The reactor protection system and the shutdown system(s) should be also operational. Movement of bridge crane over the reactor core should be minimized. The fuel shuffling operations should be designed so that the intermediate core configurations are less reactive than the most reactive one of the OLCs. The identification of the fuel element should be checked each time it is moved to a new location. Considerations should be taken to ensure that fuel elements, including their orientation, are correctly positioned in the core.

In addition to the measurements mentioned in section 2 above and before any further power operations with the newly assembled core configuration, the safety system settings should be adjusted and the control rod withdrawal speed and drop time should be measured. Comparison of the measured and calculated parameters should be made and the results should be assessed from safety point of view for further improvements to the calculation models and tools.

5. Handling of fresh and irradiated fuel

Fuel handling activities include the loading, unloading, transfer, storage, packing and transport of fresh and irradiated fuel. These activities are of major safety significance and must be performed with an increased attention following approved operating procedures and in compliance with the OLCs.

Mishandling of fresh fuel elements may lead to inadvertent criticality, scratches or other physical damages to the cladding that could affect the behaviour of the fuel in the core (e.g. reduction of the fuel coolant channel) or result in a release of radioactive material into the reactor coolant or contamination by material that could degrade the integrity of the cladding. Mishandling of irradiated fuel may also lead to inadvertent criticality, overheating, degradation of cladding material that may lead to release of fission products into the storage media, or radiation exposure.

Fresh fuel elements should be inspected before their loading into the reactor core. Dimensional checks of the fuel, including coolant channels, should be performed. Visual inspection should be performed to ensure the quality of the workmanship, verify accuracy of final machining, and to identify possible defects. A smear test could be performed for contamination control. Fuel elements which have been stored for a long time period should be re-inspected prior to their loading into the reactor core.

Fresh and irradiated fuel elements should be stored in subcritical configurations by applying physical measures (e.g. use of fixed neutron absorbers) and administrative procedures. Surveillance programme should be put in place to ensure retention of the effectiveness of these measures and procedures. The fresh fuel storage areas should be protected against fire and flooding. These areas should also be protected against physical and chemical damage by maintaining them under appropriate environmental conditions (humidity, temperature, etc.). Effective security measures should be implemented to prevent unauthorized access to fresh fuel elements storage areas.

The safety aspects of unloading of irradiated fuel from the reactor core and their movement to the storage facilities inside the reactor building were discussed in the frame of the implementation of the refuelling process above. Adequate storage places should be available to ensure that complete core unloading could be performed at any time during the reactor lifetime. Adequate cooling systems for removal of the decay heat from the irradiated fuel elements should be installed. Conservative values of decay heat should be assumed in the design of such cooling systems. The environmental conditions of the storage media should be kept within specifications to ensure the integrity of the stored irradiated fuel. The other aspects related to monitoring of the fuel integrity during the reactor operation, including handling of failed fuel are also applicable to the case of irradiated fuel.

Handling of irradiated fuel, including unloading from the reactor, moving to and storing at storage areas should be covered by the operational radiation protection programme. The radiation protection measures that should be applied to the irradiated fuel storage areas include: installation of adequate radiation shielding and dedicated radiation detectors providing an effective monitoring of radiation levels; control of the operating personnel access to the irradiated fuel storage areas; and control and minimization of releases to the environment, taking into account the ventilation system and the associated filtration system.

6. Conclusion

The safety considerations of the core management and fuel handling activities for research reactors were discussed on the basis of the IAEA Safety Standards. The needs for a flexible core operation and for an effective use of the fuel for optimum utilization of the reactor should not compromise its safety. Specific precautions need to be taken in performing fuel handling activities to ensure that the fuel integrity and subcriticality are maintained as well as to protect the individuals and environment from radiation hazards. Because of their importance to safety, the core management and fuel handling activities should be performed according to approved operating procedures and in compliance with the OLCs, and be subjected to review and assessment by the regulatory body.

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INVESTIGATION OF POSSIBILITY FOR FUEL DEFECT DETECTION BY ANALYSIS OF RADIONUCLIDE IN PRIMARY COOLANT OF HANARO

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ABSTRACT

The radionuclides in the primary coolant of HANARO were analyzed, and the possibility of the fuel defect detection by using the analysis was investigated. The concentrations of the radionuclides in the primary coolant were determined by using the gamma-ray spectroscopy. Several activation and fission products were detected in the coolant. The source of the fission products was the uranium contamination on the fuel surface. The variation of the concentration of each nuclide was compared with that of the delayed neutron measurements in the primary cooling system. The proper nuclides and gamma-ray peaks for the fuel defect detection were determined, and they were 166 keV peak from Ba-139, 250 keV peak from Xe-135, 307 keV from Tc-101 and 1436 keV from Cs-138. During the real operation period of HANARO, a very small fuel defect which resulted in a much smaller signal than that required to trigger the fuel failure detection system could be confirmed. Therefore, it is confirmed that this method is very useful as one of the auxiliary measures for the fuel defect detection in research reactors.

1. Introduction

In HANARO, a 30 MW research reactor, the fuel failure has been monitored by using the detection of the delayed neutrons from the primary coolant. The delayed neutron precursors are released from the inside of a failed or defected fuel rod to the coolant water [1,2]. Three fuel failure detection systems (FFDSs) were installed near the pipes of the primary cooling system (PCS).

In addition to the delayed neutron detection, continuous monitoring of the gamma-ray emitting nuclides from the inside of the fuel in the primary coolant loop can also be used in real time. Since the energies of the gamma-rays from the fission products are mostly smaller than 2 MeV, this method utilizes the total gamma-ray count rate below 2 MeV. However, it may not provide the sensitive result to detect the fuel failure due to the high Compton background of the high energy gamma-rays from the N-16 or others [2].

Quantitative analysis of the radionuclides in the reactor coolant water using HPGe detection system can also be used for the detection of the fuel failure. This method has weaknesses of the slow response time and the complicated procedure. So, it is difficult to directly apply this method to the reactor protection system. However, an exact and sensitive detection of small defect of the fuel is possible by this method, and other abnormal situations like a leakage of irradiation samples can be also detected. In order to apply this method to the research reactor operation, the species and concentrations of the gamma-ray emitting nuclides in the coolant should be analysed, and the suitable nuclide for the detection of the fuel defect should be determined.

Therefore, in this work, the concentrations of the radionuclides in the coolant water were measured by the gamma-ray spectroscopy, and the origins of the radionuclides were analysed. And then, the possibility for the fuel defect detection by the analysis was investigated.

2. Experimental method

During the normal operation of HANARO, the species and concentrations of the radionuclides in the primary coolant except for the short-lived nuclides were determined by using the gamma-ray spectroscopy. The primary coolant was collected at the entrance to the primary coolant purification system at a reactor power of 30 MW, the normal power of HANARO. The volume of the collected water was 100 cm^3. The gamma-rays from the coolant water were detected by using the spectroscopy system of a coaxial HPGe detector with a relative efficiency of 15% [3,4]. The cooling time was 5 min., and the specific concentration of each nuclide in the coolant at the reactor core was obtained by considering the cooling time. The full-energy peak efficiency for the volume source was calibrated as a function of the photon energy for the HPGe detector. The efficiency calibration was carried out with a cylindrical bottle-type standard source which had a homogeneous radionuclide mixture in the matrix with the same shape as the sample bottle. The density of the matrix was 0.97 g/cm^3.
Fig. 1 shows the typical gamma-ray spectrum of the HANARO coolant water with a cooling time of 5 min. and a gamma-ray collection time of 30 min. During the normal operation of a water-cooled reactor, the major gamma-ray source in the coolant is N-16 generated by $O^{16}(n,p)N^{16}$ reaction. It was confirmed that the effect of the N-16 gamma-rays on the spectrum was negligible through the gamma-ray spectrum obtained with the minimum amplification factor of the spectroscopy system since the cooling time was long enough for the N-16 nuclide to decay out. As shown in the figure, the gamma-ray peaks from Na-24, Mg-27 and Al-28 are much higher than those of other nuclides.

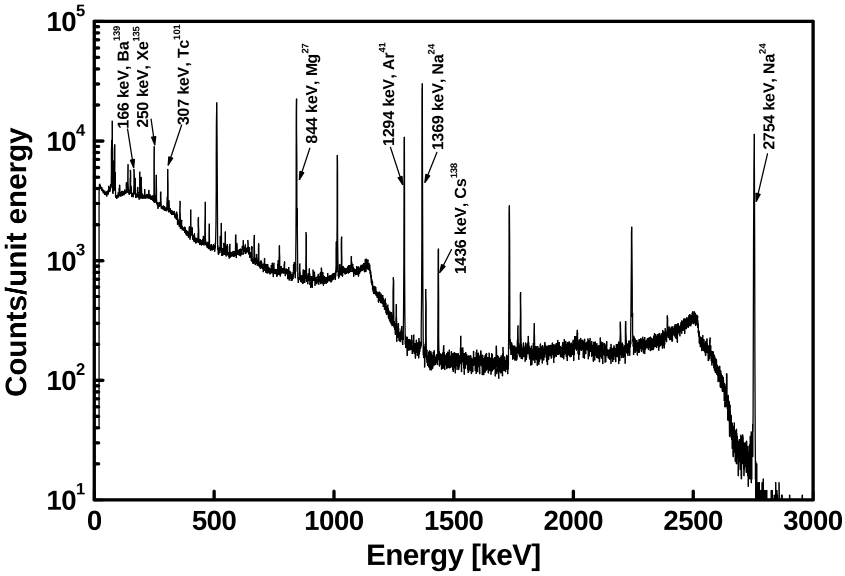

Fig 1. Typical gamma-ray spectrum of HANARO coolant water with a cooling time of 5 min. and a collection time of 30 min.

The species and concentrations of the radionuclides except for the short-lived nuclides in the primary coolant during the normal operation of HANARO were determined by using the full-energy peaks on the spectrum. Table 1 represents the determined concentrations of the confirmed radionuclides in the coolant water. Although very small gamma-ray peaks from several nuclides such as Ce-141, I-131 and Zr-95 were sometimes confirmed, the determination of their concentrations in the coolant was meaningless since their uncertainties were so large. When the sample was cooled for a long time, and the gamma-rays were collected for a long time, the Cs-137 and Co-60 peaks could be confirmed. The uncertainty in the determination for the activity concentration was below 1% for Na-24, Ma-27, Al-28 and Ar-41. The uncertainty for the fission product was relatively high, and it ranged from 5 to 25%. From the table, it is confirmed that the concentrations of Na-24, Mg-27, Al-28 and Ar-41 are much higher than those of other nuclides in the coolant, and they are 3×10^5~5×10^6 Bq/liter. In addition, many fission products such as iodine and xenon nuclides were detected in the coolant. Among them, the concentrations of Rb-89, Xe-138 and Xe-133 were relatively higher than those of others, and they were ~3×10^4 Bq/liter. Xe-133 is a very important gaseous radionuclide source in the research reactor. Its concentration at the beginning of a reactor period is small and increases with the reactor operation time since its half-life is relatively long. The source of the fission products in the coolant during the normal operation was the surface contamination of the nuclear fuel by uranium in the fabrication procedure. However, in the abnormal condition like a fuel defect, the concentrations of the fission products will be increased abruptly.

Tab 1: Determined concentrations of the radionuclides in the HANARO coolant water.

Nuclide	Half-life (min.)	Main gamma-ray energy (keV)	Concentration (Bq/liter)	Nuclide	Half-life (min.)	Main gamma-ray energy (keV)	Concentration (Bq/liter)
Al-28	2.31	1778.8	1.75E+06	Np-239	3391.68	103.7	6.13E+03
Ar-41	109.8	1293.6	3.15E+05	Rb-88	17.8	1836	1.76E+04
Ba-139	83.06	165.8	1.38E+04	Rb-89	15.4	1031.88	3.09E+04
Ba-141	18.267	190.22	1.58E+04	Sr-91	580.14	555.57	-
Ce-141	46632.96	145.45	-	Sr-92	162.6	1383.9	7.48E+03
Cs-138	32.2	1435.86	2.56E+04	Sr-93	7.3	590.9	-
I-131	11579.04	364.48	-	Tc-101	14.2	306.86	1.95E+04
I-132	142.8	667.69	7.17E+03	Tc-104	18.2	357.99	8.35E+03
I-133	1218	529.5	4.01E+03	Tc-99m	361.14	140.51	4.73E+03
I-134	52.6	1072.53	-	Te-131	25	149.72	4.50E+03
I-135	396.66	1260.41	-	Te-131m	1800	852.21	-
Kr-85m	268.86	151	1.52E+03	Te-132	4675.02	228.16	1.70E+03
Kr-87	76.4	402.7	-	Te-133	12.45	312.1	5.05E+03
Kr-88	170.34	2392.11	-	Te-134	41.8	565.99	-
La-142	92.517	641.17	-	W-187	1434	685.74	3.20E+04
Mg-27	9.462	843.73	5.25E+06	Xe-133	7619.04	81	9.32E+03
Mn-56	154.56	1811.2	1.31E+04	Xe-135	544.98	249.79	6.90E+03
Mo-101	14.62	590.82	-	Xe-135m	15.6	526.81	1.19E+03
Na-24	897.54	1368.55	1.54E+06	Xe-138	14.13	258.31	2.45E+04
Nb-95	50364	765.82	-	Zr-95	92733.12	724.18	-

3. Origin of radionuclide in HANARO coolant water

Activation products of the coolant water such as N-16, N-17 and O-19 were not detected in this measurement. The main gamma-ray sources in the coolant water after 5 min. cooling are Na-24, Mg-27 and Al-28. Their origins are radiative reactions of aluminum used as the structural material in the reactor core and the irradiation rigs and the cladding of the nuclear fuel. Ar-41 is generated from the activation of dissolved air. Mn-56 is an activation product of iron in stainless steel used as structural material. W-187 is also an activation product of tungsten used as welding rod. Zr-related nuclide such as Nb-95 can be generated as a fission product or from the activation of zirconium used in the flow tube and the fuel bundle.
Various fission products such as iodine, xenon, etc. are detected in the coolant in spite of the normal operation. These nuclides are generated from the fission process of the uranium contaminated on the fuel surface. The maximum allowable surface contamination of HANARO fuel is 3.25 $\mu g\text{-}U^{235}/ft^2$. From the above result, it is confirmed that even a small amount of uranium contaminated on the fuel surface can give rise to large gamma-ray peaks of the fission products enough to determine their concentrations in the coolant.

4. Detection of fuel defect

The concentrations of the radionuclides in the primary coolant were measured continually once a week during the reactor operation. The variation of the concentration of each nuclide was compared with that of the delayed neutron measurements in the primary cooling circuit. The delayed neutron measurement system is used as the radiation monitor for the primary cooling system (PCS).

Fig. 2 shows the variations of the delayed neutron measurements of the PCS radiation monitor and the activity concentrations of Na-24 and Ar-41 in the coolant for the last two years. As shown in the figure, Ar-41 activity concentration has hardly changed during 2 years. It means that the quantity of the dissolved air in the coolant is almost constant. The variation of the Na-24 concentration for one period of the reactor operation shows no significant change. But, the change between each period is relatively large since the total reaction rate of the aluminium inside the reactor core is changed for each period. At any rate, it is confirmed that the concentrations of Na-24 and Ar-41 are independent of the delayed neutron measurements of the PCS radiation monitor.

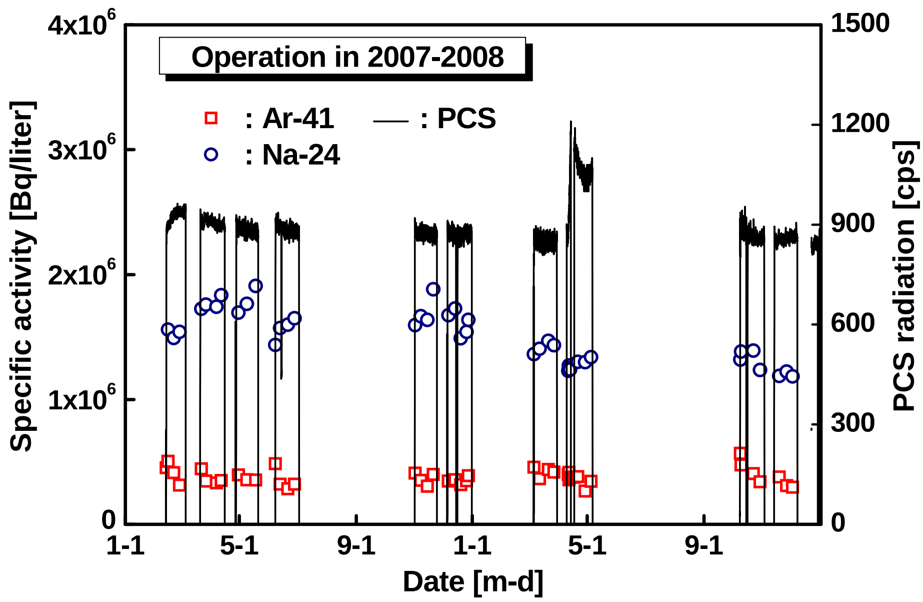

Fig 2. Variations of the delayed neutron measurements of the PCS radiation monitor and the activity concentrations of Na-24 and Ar-41 in the coolant for last two years.

As shown in Table 1, there are a lot of fission products in the coolant water. Among them, in order to choose the proper nuclide for detecting the fuel defect, the half-life, decay scheme, peak area and interference of adjacent peaks were considered. And then, the variation of the concentration of each nuclide was compared with that of the delayed neutron measurements in the primary cooling circuit. From the comparison, the proper nuclides and their gamma-ray peaks for the fuel defect detection were determined, and they were 166 keV peak from Ba-139, 250 keV peak from Xe-135, 307 keV from Tc-101 and 1436 keV from Cs-138. The half-lives of the selected nuclides are from 14 min. to 9 hr. Even though these peaks were located on a large Compton continuum of the gamma-ray spectrum of the coolant, the peak areas were determined with relatively small uncertainties.

Fig. 3 shows the variations of the delayed neutron measurements of the PCS radiation monitor and the activity concentrations of the selected nuclides. In April, 2008, a fuel rod with a very small defect was found during the reactor operation. Due to the defective fuel, the delayed neutron measurements of the primary cooling system (PCS) were increased slightly as shown in Fig. 2 and 3. But, this value was much smaller than that to trigger the fuel failure detection system (FFDS). As shown in Fig. 3, the concentration of the selected nuclide shows very drastic change when the delayed neutron value is increased slightly. The concentrations are increased by over 5-10 times in comparison with the case of no defect. Thus, it is confirmed that the very sensitive and exact detection of the fuel defect would be possible by this method even if the defect is very small.

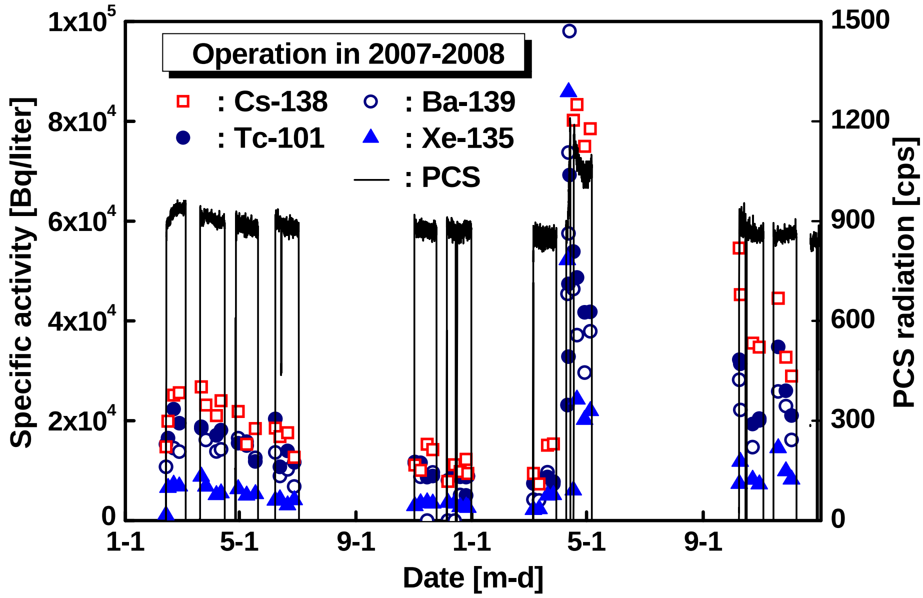

Fig 3. Variations of the delayed neutron measurements of the PCS radiation monitor and the activity concentrations of selected nuclides.

In the real operation period of HANARO, a very small fuel defect which resulted in a much smaller signal than that required to trigger the fuel failure detection system (FFDS) could be detected by this method. So far, the gamma-ray spectroscopy for the reactor coolant takes lots of time, and the procedure is fairly complicated. Thus, it is very hard to apply this method to the reactor protection system. However, this method would be very useful as an auxiliary measure for the analysis of a small defect on the fuel.

5. Conclusion

The radionuclides in the primary coolant of HANARO were investigated by gamma-ray spectroscopy, and this analysis was applied to the detection of the fuel defect. The proper nuclides and gamma-ray peaks for the object were determined. During the real operation period of HANARO, a very small fuel defect which resulted in a much smaller signal than that required to trigger the fuel failure detection system could be confirmed. Therefore, it is confirmed that this method is very useful as one of the auxiliary measures for the fuel defect detection in research reactors.

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AD-HOC AND PREVENTATIVE MAINTENANCE SYSTEM AS PART OF THE SAFARI-1 MANAGEMENT SYSTEM

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ABSTRACT

SAFARI-1, the Nuclear Research Reactor located at Pelindaba, has to meet a multitude of requirements relating to corporate policies, strategic planning, preventative maintenance , in-service inspection, plant ageing, quality, conventional and radiological safety, regulatory specifications, security, commercial and financial goals etc., which need to be incorporated into an overall Management System. The control of these various disciplines within the nuclear facility become quite complex if the procedures are to be coordinated, maintained and at the same time managed so as to achieve suitable levels of staff motivation, which will ultimately ensure appropriate implementation. Maintenance is one aspect that was a challenge over the last 12 years to identify, implement and optimize the various disciplines of maintenance. The objective was to ensure fewer plant stoppages as a result of an effective maintenance programme implementation.

Although SAFARI-1 has achieved ISO 9001 and ISO 14001 Certification, it's present commercial and operational schedule takes a very well coordinated operational and maintenance management system to achieve all stakeholder requirements and still function within the design and safety requirements. In order for SAFARI-1 to maintain an exceptionally high operational schedule of 305 out of 365 days it takes good planning and coordination to fit routine maintenance, upgrades and In-Service Inspection activities. This will ultimately ensure the success of the facility to maintain operation requirements.

The management system procedures incorporate an interdisciplinary approach, which can integrate all the above organizational requirements, with the emphasis on safety and at the same time encompass a more customer driven focus - this includes customers both internal and external to the SAFARI-1 organization. The integration of a management system requires effective implementation at all levels such as operational schedules and procedures, maintenance procedures, project management, managing design changes, training, QHSE manuals, legislative and licensing requirements, work instructions, etc. All these activities have been satisfactorily achieved within SAFARI-1management system

1. INTRODUCTION

The nuclear installations at SAFARI-1 have already been designed, manufactured, erected, commissioned and licensed and have been operational since 1965. Ongoing programmes of development, improvement, upgrading and ageing management have characterized this past operation and are expected to continue for the remaining life cycle of the facility.

It is SAFARI-1 Management's policy to conform to all applicable codes, standards, practices, guidelines and regulatory requirements in the continued operation of the facility and in such ongoing development, improving and upgrading thereof. Safety is the most important issue considered by management and employees during the operation and utilization of the reactor. The requirements for conformity are set out in an integrated Quality Safety Health Environmental Management System (QHSE MS) and therefore are binding on all personnel and all levels of management within SAFARI-1. Amongst others, effective maintenance is one of the most important specific activities to achieve safety and the intent of the design objectives.

2. QUALITY HEALTH SAFETY MANAGEMENT SYSTEM

The SAFARI-1 QHSE MS and Necsa organizational structure defines the organization in terms of management units, covering overall management, quality, health, safety and environmental management, project management, reactor operations, reactor engineering, training and authorization as well as the reactor utilization of SAFARI-1 Research Reactor. The SAFARI-1 management has the primary responsibility for the safety of the facility, covering operation, utilization, maintenance and modification. To discharge this responsibility, management has implemented an integrated QHSE MS prescribing activities that shall:

- Ensure that safety matters are given the highest priority;
- Ensure that established safety policies of Necsa are adhered to;
- Provide a clear definition of responsibilities and accountabilities with corresponding lines of authority, accountability and communication;
- Ensure that sufficient staff at all levels are properly and unambiguously authorized to carry out safety-important work according to appropriate education, training and competence requirements;
- Develop sound work procedures, based on good safety practices, and ensure that these are strictly adhered to and
- Review, monitor and audit all safety matters on a regular basis and implement appropriate corrections where required.

2.1 INTERNATIONAL CODES, STANDARDS, PRACTICES AND GUIDELINES

ASME NQA-1: Quality Assurance Requirements for Nuclear Facilities, ASME, New York.
DS412: Draft Safety Guide “Ageing Management of Research Reactors”, IAEA, August 2008.
NS-R-4: Safety of Research Reactors – Safety Requirements, Vienna, 2005.
NS-G-4.2: Maintenance, Periodic Testing and Inspections of Research Reactors, IAEA Draft Safety Guide, Vienna, ca. 1996.
SABS ISO-9001:2000: Code of Practice: Quality Management Systems – Requirements, SABS, Pretoria, RSA.
SABS ISO-14001:2004: Code of Practice: Environmental Management Systems – Specification with Guidance for Use, SABS, Pretoria, RSA.
SRS 1: Examples of Safety Culture Practices, IAEA Safety Report Series, Vienna, 1997.
IAEA SS No. 50-C/SG-Q: Quality Assurance for Safety in Nuclear Power Plants and Nuclear Installations.
SS 35 G1; G2; G6 Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report, IAEA Safety Guide, Vienna, 1994.
SS 35 P5: Operating Procedures for Research Reactors, IAEA Safety Practice, Vienna, 1994.
SS 50 C/SG-Q: Quality Assurance for Safety in Nuclear Power Plants and Other Nuclear Installations, IAEA Safety Standard and Guides, Vienna, 1996.
SS 75 INSAG-4: Safety Culture – A Report by the International Nuclear Safety Advisory Group, IAEA Safety Report, Vienna, 1991.
TECDOC-TCM (Dec-98): Safety of Core Management and Fuel Handling for Research Reactors, IAEA draft TECDOC, Vienna, December 1998.
TECDOC-967: Guidance for Considerations and Implementation of INFCIRC/ 225/Rev.3, The Physical Protection of Nuclear Material, Vienna, September 1997.
TECDOC-1263: Application of Non-destructive testing and In-service Inspection to Research Reactors, Vienna, December 2001.

3. SAFARI-1OPERATIONAL HISTORY

The operating history of SAFARI-1 is summarized below. By March 2008 the reactor had been operational for 43 years and had produced a total of 261 GWh of thermal energy during that time. The main features are briefly discussed below:

- 1965: First criticality in March 1965 - after which the reactor operated at 6.67 MW thermal power using HEU fuel of US origin with an enrichment of 93%.
- 1968: The reactor was shut down for approximately 9 months to upgrade the heat removal train for 20 MW operation.
- 1977: Due to the negative political climate at this time the US fuel supply to SAFARI-1 was stopped, prompting a reduction in the nominal operation of the reactor to 5 MW during weekdays only.
- 1981: Locally manufactured Medium Enriched Uranium (45% Enriched) fuel was supplied for the first time. The reactor continued to be operated at 5 MW on a weekday schedule for the next 12 years.
- 1988: The reactor was shut down for an extended period of ±6 months to effect repairs to the pool liners.
- 1993: The start of a commercially oriented utilization programme led, for the first time in 16 years, progressively to reactor operation at powers higher than 5 MW, initially at 10 MW but after 1995 towards 2000 increasingly higher powers, 18 MW average, and ultimately to 20 MW continuously until today.
- 1994: Enrichment of locally produced fuel elements increased from MEU to HEU, with a ^{235}U content of 200g.
- 1995: 1 Million MWh (1000GWh) total energy production since first start up.
- 2000: The 235U content in locally manufactured HEU fuel elements was increased from 200g to 300g.
- 2003: 2 Million MWh (2000GWh) total energy production since first start up.
- 2007: The first two LEU fuel elements (340g of 19.75% enriched ^{235}U) of French origin were successfully irradiated in the core.
- 2008: Core conversion to LEU started, currently 19 LEU fuel and 2 LEU Control rods loaded (core contain 26 Fuel Elements and 6 Control Rods)

Table 1: SAFARI-1 AVAILABILITY

Year	1999	2000	2001	2002	2003	2004	2005	2006	2007	2008
a) Available time [hrs]	8760.0	8784.0	8760.0	8760.0	8760.0	8784.0	8760.0	8760.0	8760.0	8760.0
b) Operational time [hrs] (a-c-d)	7393.3	7518.0	7607.7	7445.9	7540.8	7392.0	7501.0	6982.5	7167.1	7314.3
c) Scheduled downtime [hrs]	1306.8	1183.4	1089.5	1278.2	1176.0	1279.1	1214.4	1328.7	1321.5	1331.0
d) Unscheduled downtime [hrs] (e+f)	59.83	82.53	62.64	35.80	43.08	112.79	44.53	448.69	271.33	114.70
e) Plant unreliability [hrs]	44.06	16.61	33.46	10.83	29.47	25.42	2.43	349.5*	239.3*	36.40**
f) Beyond plant control [hrs]	15.77	65.92	29.18	24.97	13.62	87.3***	42.10	99.13	32.00	78.30**
g) Load factor [%] (b/a)	84.38	85.59	86.86	85.00	86.09	84.14	85.63	79.73	81.82	83.50
h) Loss in load factor due to plant unreliability [%] (e/a)	0.50	0.02	0.38	0.13	0.33	0.29	0.03	3.98	2.73	0.42

***Note:** Failure of deminerilizer anion column bottom sieve and resin released into primary system.

**** Note:** Two unreliable channels of the six safety channels caused sporadically the reactor to scram.

*****Note:** Failure of Conical Strainer

From **Table 1**, since 1996 the reactor power levels were progressively increased from an average of 16MW to 20 MW continuously and operate for the last nine years ~305 days a year at 20MW. Operating cycles are ~30 days with a 5 day shutdown and one extended 12 days shutdown per annum.

4. HISTORY OF UPGRADES AND MODIFICATIONS

A continuous programme of improvement, over the life of the facility, has resulted in many upgrades and modifications to the reactor and auxiliary equipment. Many of the upgrades and modifications were aimed at improving safety or replacing obsolete safety related equipment with state-of-the-art equipment (such as neutron safety instrumentation, cooling system pumps etc). Other improvements focused on expanding the versatility and utilization of the reactor and, since ~1994, on providing greater operational flexibility for commercial irradiation programmes.

A list of the more important upgrades and modifications undertaken during the life of the reactor is presented in **Table 2 below**. From Table 2 the following could be concluded:

- More than 60% of refurbishment and upgrades were done during the last 12 years of the reactor life cycle of 43 years and increased steadily as the plant aged.
- More upgrades were performed since 1994 to maintain the increasing operational schedule and to ensure reliability w.r.t. safety.
- 1995 the 1 Million MWh (1000GWh) total energy production since first start up was reached and the 2 Million MWh (2000GWh) total energy production since first start up was reached in 2003.
- During the period 1994 to 2000 the plant had a experience unscheduled stoppages due to minor failures such as old pumps, mechanical seals/bearings, electrical motors, ventilation fan bearings, water leaks in system cooling tower and heat exchanger efficiency, instrumentation and electrical systems. Since the implementation of a good QHSE MS and preventative maintenance program, ISI and competent personnel the plant systematically became more reliable towards 2008. A large amount of the above-mentioned maintenance activities were performed over the last 12 years to replaced or modernized plant systems. All of this was done to ensure reliability and safety of the plant, personnel and environment.

5. MAINTENANCE SYSTEM

The SAFARI-1 research reactor maintenance programs are adopted (with more emphasis on maintenance and personnel competency since 1996 than ever) to ensure that systems, structures and components (SSC) continue to operate as desired with capability to meet the design and safety objectives. The maintenance procedure forms an integrated part of the QHSE MS and prescribes the principles and controls established for periodic inspections and maintenance within the SAFARI-1 facility. This maintenance procedure ensures that plant equipment and related nuclear safety and safety critical equipment or systems, are correctly inspected and maintained in accordance with ISO 9001 and IAEA safety guidelines (SS 35 G7, SS 50C/SG - Q13 and NS-G-4.2) and regulatory requirements.

5.1 MAINTENANCE PROGRAMS

The SAFARI-1 Research Reactor, has since initial operation in 1965, applied a management system which was primarily focused on the technical design and safe operation of the plant. The informal system was never developed to comply with a specific code of practice. Since the early 1990's the challenge was to return to the international arena and to change the employees from a Monday to Friday research culture into a more complex commercial culture, operating the plant 24 hours a day at a fixed operating schedule so as to meet customer and stakeholder requirements. The finding of a government evaluation in 1997 was clear: "be commercially viable, at least 67% self-reliant or closedown". Under the above-mentioned circumstances the decision was made, as part of a systematic strategic plan, to implement a formal quality management system in accordance with the ISO 9001. During this period the decision was made to implement all disciplines of the management system according to IAEA and national standards to ensure development of competency and support in all areas of the organization structure. One of the areas was to prepare new maintenance programmes requiring maintenance at regular intervals in order to reduce the probability of failure. New SSC and optimization of methods were included in the maintenance program systematically and still today programs are revised to change or add maintenance activities or revise frequencies. During 1997 and 1998 the Quality Management System was prepared and formal maintenance programs for instrumentation, electrical, mechanical and reactor specific activities were prepared and include:

- Routine maintenance (Preventative Maintenance).
- Periodic inspections (Preventative Maintenance).
- Ad-hoc maintenance – Request for Maintenance database (Corrective Maintenance).
- Functional inspection (Preventative Maintenance).
- Performance and functional tests (Predictive maintenance).
- Operational checks and maintenance (Reactor pools and hot cells).
- ISI procedure and ISI plan.
- Training and authorization of maintenance personnel.
- Control of equipment and spares.
- Project management and design control.
- Procurement control and release of items.
- Work permits (Radiological and conventional safety and Authorization of personnel).
- Record keeping of suppliers, SSC data and project file containing all information.

The maintenance program lists responsibilities, frequencies, schedules (Daily Weekly, Monthly Yearly and 2;3;4;5;etc yearly schedules), criteria and controls for the maintenance operations, identify systems that can be isolated while the reactor is in operation, applicable restrictions for on line maintenance and records to be kept. It also prescribes or reference additional maintenance instructions that exist for the equipment (e.g. from manufacturer's manuals), as well as any specific requirements for the training of the maintenance staff.

5.2 INSPECTION AND MAINTENANCE OF PLANTS

All maintenance procedures/activities are consistent with the Operating Technical Specification (OTS) and Safety Analysis Report requirements. Operating requirements of the OTS are vital and will not be compromised as a result of maintenance activities.

5.3 MAINTENANCE SCHEDULES AND SHUTDOWNS

The Reactor Manager is primarily responsible for maintenance planning of all routine and special maintenance. Approval of any project (Reactor Safety Committee or Regulatory) for installation is finally accepted by the Reactor Manager. In planning maintenance activities, due cognizance is taken of the reactor operational program, maintenance programs, ISI plan as well as the nuclear safety aspects pertaining to the maintenance work. A Maintenance Shutdown Plan is issued detailing all maintenance schedules and ad-hoc inspections or testing to be performed during a shutdown. After maintenance has been completed, post-maintenance tests and/or inspection, as applicable, are be performed. On completion of maintenance activities, the Shift Supervisor checked that the work is completed and perform re-commission the equipment or system and on completion of final checks and or adjustments, signs off the works permit.
The completed operational check lists or maintenance check lists are used as the maintenance report.
The Reactor Manager, Engineering Manager and relevant support managers/personnel, as applicable, jointly review all reports (including ISI reports) after each planned maintenance shutdown to verify that the plant is being correctly inspected and maintained in accordance with the OLC, OTS and maintenance schedules.
On a regular basis the regulatory body reviews and assesses documents/reports relating to maintenance and periodic inspections/testing to ensure adequate control of maintenance activities. The regulatory body also conducts inspections to ensure conformance with the requirements for maintenance schedules.

6. CONCLUSION

An integrated QHSE management system involving all aspects (various corporate, regulatory and international requirements) required to manage a nuclear facility safely and successfully must be developed in conjunction with other management systems and should include all disciplines from human resources to decommissioning and decontamination. Management systems do not originate by themselves but must be carefully planned and implemented to ensure that they function satisfactorily. Such a management system supports maintenance and operational activities to ensure one of the most important objectives of a nuclear plant that is, “Plant Safety” but also ensures training of personnel, quality of behavior, quality of thinking and quality of decision making to enhance company QHSE culture.

Table 2: SAFARI-1 Major Upgrade Projects Since 1994

DATE	DESCRIPTION	DATE	DESCRIPTION
1994	Developed, designed and installed high-density storage racks for spent fuel in storage pool.	1999	Upgraded heat exchanger temperature and pressure measurements with own signal transmitters. Added signals to data log system.
1994	Upgraded mechanicals (motors, drive shaft, gearboxes and fans) in cooling towers.	1999/2000	Instrumented cooling towers in preparation for PLC control.
1994	Installed power distribution mimic in control room.	2000	Upgraded video surveillance system of reactor building.
1994	Upgraded and re-cabled power distribution in process wing. Split the three safety instrument channels. Removed lead sheathed VIR power cables.	2000	Upgraded ventilation control systems to PLC control.
1994	Increased enrichment of SAFARI-1fuel from 46% to 90%.	2000	Installed Isotope production thimble in core position F6.
1994	Replaced drift eliminators in cooling tower (wood to plastic).	2000/2001	Installed access control system and turnstiles.
1994	Installed Video Surveillance system of reactor building	2000/2001	Cooling tower refurbishment: changed water distribution system and improved spray and rain density.
1995	Fission Molybdenum production upgraded for reactor operations at 20 MW using 7-plate holders.	2000	Cooling tower automation: automatic valves; full PLC control for inlet/outlet valves, make up and blow down.
1995	Isotope irradiation in 99Mo thimbles authorized.	2000	Upgraded compressors (new standby compressor – old standby compressor converted to emergency compressor – old emergency compressor scrapped).
1995	Upgraded lightning protection system. Added Dehnventil units, finials and additional earth points.	2001	Further upgrade of Video Surveillance system of reactor building.
1995	Replaced temperature and delta temperature transmitters for primary and secondary water.	2001	Replaced sections of ventilation ducting.
1995	Upgraded all differential and pressure transmitters to smart transmitters with Hart protocol.	2001	Automatic TDS control added for cooling towers.
1996	Replaced secondary pumps.	2001	Installed isotope production thimble in core position D6.
1996	Modified anti-vibration skirt in reactor vessel to accommodate a thimble in core position B8 and 99Mo production positions expanded to six.	2003	Commissioned an in-pool gammametry facility originally installed in ~1983 as part of a Jetpompe facility that was never utilized.
1996	Developed and installed a hot-cell radiation monitor.	2003/2004	Replaced the obsolete reactor primary coolant pumps with modern off-the-shelf equivalent.
1996	Upgraded rabbit transfer system to PLC control	2004	Installed isotope production thimble in core position B6.
1996	Replaced secondary piping	2004/2009	Install security fence and perimeter monitoring camera network. Upgraded access and egress control throughout building

DATE	DESCRIPTION	DATE	DESCRIPTION
1996-2000	Upgraded all fluid drive couplings to soft starters, ventilation fans, cooling tower fans, pool pump.	2007	Upgraded the standby compressor from a two stage piston compressor to a worm compressor with built in cooling and moisture removal.
1996	High-density storage racks for spent fuel modified to overcome electrolytic corrosion problem.	2007	Upgraded the air drier to a modern automated system.
1997	Installed new 16N channel for testing.	2007	Stripped and re-coated the process wing floors with an industrial epoxy coating, This upgrade removed the need to polish and strip the floors regularly, thus reducing the quantity of chemical waste.
1997	Installed a “Multilink” SCADA system.	2007	An industrial network on a fibre-optic backbone which is separated from the campus network was installed. This network will be commissioned in 2008, and will consist of a fully redundant server system and will enable the automation of various manual tasks operations.
1997	Installed a ring intercom system throughout the plant.	2007	The Adroit Scada system was upgraded from a 750 scan point license to unlimited scan points. This was required for the Industrial network installation and implementation.
1997	Installed isotope production (IPR) thimble in core position B8.	2002-2006	Stuck collimator in BT no 1. Equipment developed and collimator successfully removed. Deformation noticed in beam tube. BT still sound
1998	Upgraded reactor hall crane to PLC control.	2003	‘Filter house’ was erected over filter pits on top of exhaust fan yard to facilitate all-weather filter replacement and improved contamination control
1998	Upgraded clock displays and synchronized all clocks and PC’s to national time standard.	2006-2007	All accesses to radiological controlled areas fitted with automated full-body contamination monitors.
1998	Instrumented primary pumps for temperature and pressure monitoring.	2003	A second, back-up degasifier pump installed.
1998	Upgraded water chemistry monitors.	2007-2008	Stuck collimator BT no 5 removed successfully. Deformation in beam tube noticed. Beam tube still sound.
1998	Replaced Isotope Production Thimble in core position B8 with a 99Mo production Thimble.	2008	Replaced control drives motors and refurbishes electrical panels.
1998	Conducted qualification tests on 300g fuel elements.	2008	Refurbishment of cooling tower fill and structure.
1998	Upgraded cooling tower inlet/outlet valves to pneumatic actuators.	2008	Replacement of two Neutron Safety Channel detectors and cables. Purchased spares
1999	Upgraded “Multilink” data log system to “Adroit” system with remote access and up to 750 parameters.	2008/9	Core conversion to LEU fuel started. Still in progress.

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