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Research Reactor Modernization and Refurbishment



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FOREWORD

Many recent, high profile research reactor unplanned shutdowns can be directly linked to different challenges which have evolved over time. The concept of ageing management is certainly nothing new to nuclear facilities, however, these events are highlighting the direct impact unplanned shutdowns at research reactors have on various stakeholders who depend on research reactor goods and services. Provided the demand for these goods and services remains strong, large capital projects are anticipated to continue in order to sustain future operation of many research reactors.

It is within this context that the IAEA organized a Technical Workshop to launch a broader Agency activity on research reactor modernization and refurbishment (M&R). The workshop was hosted by the operating organization of the HOR Research Reactor in Delft, the Netherlands, in October 2006. Forty participants from twenty-three countries participated in the meeting: with representation from Africa, Asia Pacific, Eastern Europe, North America, South America and Western Europe.

The specific objectives of this workshop were to present facility reports on completed, existing and planned M&R projects, including the project objectives, scope and main characteristics; and to specifically report on:

- the project impact (planned or actual) on the primary and key supporting motivation for the M&R project;
- the project impact (planned or actual) on the design basis, safety, and/or regulatory-related reports;
- the project impact (planned or actual) on facility utilization;
- significant lessons learned during or following the completion of M&R work.

Contributions from this workshop were reviewed by experts during a consultancy meeting held in Vienna in December 2007. The experts selected final contributions for inclusion in this report. Requests were also distributed to some authors for additional detail as well as new authors for known projects not submitted during the initial 2006 workshop.

The report reflects the final contributions from both the 2006 workshop as well as additional papers received following the 2007 consultancy. It marks the next step in a process that began with the higher level recommendations contained in NP-T-5.4 — Optimization of Research Reactor Availability and Reliability: Recommended Practices. Future meetings on the topic of research reactor ageing, modernization and refurbishment will permit the addition of more detailed guidance as well as the collection and distribution of more examples of facility ageing challenges and completed projects.

The IAEA wishes to thank all meeting participants and contributors. Furthermore, the IAEA wishes to express its gratitude to the Delft University of Technology (TU Delft) for hosting the 2006 workshop. Finally, the IAEA wishes to thank D.J. O'Kelly for editing the final papers in preparation for this report. The IAEA officer responsible for this publication was E. Bradley of the Division of Nuclear Fuel Cycle and Waste Technology.

EDITORIAL NOTE

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SUMMARY

1. INTRODUCTION

Over half of the world's operating research reactors (RRs) are now over 40 years old. All organizations must address deficiencies and new requirements that evolve over time. Reactor organizations undertake an array of work activities to either re-establish performance that has degraded over time, maintain performance in the face of changing conditions (such as System Structure or Component (SSC) obsolescence) or adapt to new customer or regulatory demands. Such work can be broken down into the following categories:

- Maintenance & Repair — maintaining existing equipment as part of a facility maintenance programme.
- Refurbishment — replacing or upgrading (to state of the art, available, more reliable, etc.) equipment to achieve the original design intent or service objectives.
- Modernization — modifications to improve reactor performance, increase utilization, accommodate requested training and education, adapt to changing customer demands, satisfy contemporary design criteria, or to achieve other strategic objectives.
- Enter into a regional or thematic research reactor coalition.
- Construct a new reactor.

The scope of this publication includes only modernization and refurbishment (M&R) activities as defined above. As they are defined, M&R work is typically implemented as a site capital project or series of smaller individual sub-projects to accomplish an overall objective or objectives. Many organizations have completed M&R projects. Many more are faced with the need to either complete M&R projects or decommission operable facilities.

Increasing interest in the application of a variety of peaceful nuclear technologies has resulted in an increasing demand for RR goods and services. Therefore, and in the context of advancing reactor age and often diminishing resources, pressure has increased to optimize large capital projects with respect to both cost and schedule. One way to optimize project implementation is to identify and implement lessons learned from the implementation of similar projects elsewhere.

In 2006, a workshop was organized and hosted by the Technical University Delft in Delft, the Netherlands. The intent of this workshop was to create a forum for the exchange of relevant project implementation, technical detail and lessons learned. To support this workshop, the IAEA identified and organized the participation of relevant experts, coordinated activities with the host institute, collected all relevant presentations and papers, coordinated the drafting of this report and agreed to consider additional workshops approximately every two years.

The outputs of the workshop included a CD with all presentations — distributed at the end of the event — and a report containing submitted and other invited papers i.e. this publication.

2. SCOPE

This publication has been developed for use by RR management teams and relevant stakeholders responsible to approve, fund and/or implement M&R projects. It assumes

individual facilities have developed a 5–10 year strategic plan (IAEA-TECDOC-1212) in consideration of customer and market trends.

Strategic assessments could also evaluate the cost/benefit of an extensive M&R programme versus entering into a coalition, construction of a new research reactor or even a ‘do nothing’ option. In addition to the cost/benefit review, M&R projects are often an attractive option because they achieve a desired change while greatly reducing less tangible challenges. Examples include:

- Licensing implications of M&R projects can be significantly less than for a new reactor.
- Public acceptance of a facility that has been well operated for several decades may be well established.
- The facility site and support infrastructure are already in place.

Such considerations should be part of the decision process.

This publication contains a collection of papers describing M&R project implementation at different RRs. The motivation, project scope, complexity, means of implementation, and lessons learned are diverse, as are the size, age, organizational complexity and utilization of the facilities involved. Participants in the related meetings also contributed to general discussions of RR specific considerations for large capital project management.

This publication assumes a decision to modernize or refurbish has been made with sound, strategic justification. For example, completing a project to address ageing or obsolescence is not considered the justification. In this case, the modification must only be completed if it maintains reactor availability to satisfy current and future customer/user demands such as training commitments, isotope production, research activities, etc.

The motivation to implement an M&R project has been specifically addressed in each contribution. Changing conditions — either plant degradation, obsolescence or evolving customer demands — are not enough by themselves to justify a significant capital investment in an operating facility. The forecasted need for an RR’s goods and services must be adequate to justify the project. A matrix has been included to organize papers by specific type of motivation.

In addition to motivation, the contribution matrix also organizes contributions by scope of project as well as specific project management areas of interest. This approach will also help readers find papers related to their specific challenges.

Numerous constraints placed on operating organizations by various internal and external stakeholders compel the careful development, planning and implementation of all M&R efforts. Obviously non-proliferation and safety-related changes must consider impacts to the design bases, related licensing reports, and submittals subject to regulatory and possibly public scrutiny. But in addition, even changes not impacting proliferation, the design basis or safety-related documentation must be thoroughly considered and optimized to add real value to an organization with potentially limited resources. The most evident examples include project planning, management and control.

Highly enriched uranium (HEU) to low enriched uranium (LEU) fuel conversions were deemed out of scope for this report. This decision was made because, in most cases, the

decision to convert was made based on incentives and pressure external to the operating organization and did not usually come about as part of a facility strategic plan. However, it is noted that several facilities took advantage of conversion efforts to implement other M&R projects in parallel. These projects have been included in the report and the links to conversions have been identified.

3. RR SPECIFIC CONSIDERATIONS FOR M&R PROJECT MANAGEMENT

General project management best practices and guidance are thoroughly detailed in a large number of books, guides and other literature. For example, a popular standard on the topic is ANSI/PMI 99-001-2004 — A Guide to the Project Management Body of Knowledge. This reference, and others like it, contain a wealth of generic information and recommended practices for any organization embarking on engineering projects. It is not the intent of this report to educate the reader on project management practices, but rather highlight areas of particular focus for research reactor organizations. An overview of a general project development, review, execution and closure process is presented in Figure 1.

The interpretation and implementation of the information presented below could vary considerably. For this reason every attempt has been made to keep the information scalable and therefore, applicable to a wide range of organizations. Differences could occur depending on the following:

- organizational size;
- organizational structure;
- available resources (financial, human, final disposition for all waste streams, etc.);
- principal and supporting mission(s) of the facility or organization (scientific, training/educational, commercial, etc.);
- size and scope of project;
- public acceptance and licensing environment.

4. LINK TO ORGANIZATIONAL STRATEGIC PLAN

IAEA-TECDOC-1212, Strategic Planning for Research Reactors – Guidance for reactor managers, provides the rationale for a strategic plan, outlines the methodology of developing such a plan and then gives a model that may be followed. A well-developed strategic plan details the projected demand for future facility operation, reflects organizational priorities and justifies the allocation of resources to address facility needs. Strategic planning is a fundamental prerequisite to M&R projects. The implemented plan provides the context and criteria against which all project proposals are reviewed.

5. M&R PROJECT JUSTIFICATION

M&R project justification must address why the proposed modification is required. Reasons such as the satisfaction of modern safety or regulatory requirements, or the replacement of obsolete equipment fail to fully justify a project. For example, obsolete or non-compliant equipment could also be decommissioned in lieu of modification. Adding ties to current or forecast customer demands — preferably via direct links to a Strategic Plan — strengthens the justification and should facilitate more efficient project reviews, particularly if those reviews involve external stakeholders.

6. M&R PROJECT PLANNING AND APPROVAL

The list presented below includes items relevant to research reactor projects that have the potential to add significant risk to project implementation (schedule delay, scope creep, cost overrun, etc.). Each should be considered and addressed in the project plan as appropriate, depending on the specific project details. It should be noted that project approval here means the project has been reviewed and authorized by the management, stakeholder or oversight authority responsible to allocate the human and financial resources required to progress the project. Regulatory and/or design approvals, for example, are considered to be part of project implementation (i.e. a deliverable from the project manager or project management team).

- Has this or similar work been completed before within the organization (or externally)? Are there any lessons learned to be applied to reduce project risk? Consider any available information (cost, experience base, vendor experience, regulatory/licence impact, etc.).
- Does the organization possess the expertise to manage and complete the work (and will the relevant resources be available as needed to support the project)? This includes access to both a project management organization as well as required support organizations (engineering, construction, etc.) and their related management systems.
- How can/should the project implementation, including equipment installation, be optimized to minimize the impact on reactor availability?
- How can the modification design be optimized for reactor availability, reliability and maintainability? (Redundancy to avoid inadvertent plant shutdowns; equipment easily operated and maintained during reactor operation; use of high quality, highly reliable sub-components; etc.)
- How can the project be planned to keep personnel doses as low as reasonably achievable (ALARA)?
- How can project-related communications be optimized (internal: project meetings, management meetings, project reports, etc., as well as external: regulatory, customer, supplier interface, etc.)?
- What are the specific project integration requirements (required internal or external safety/regulatory approvals, new or revised procedures, revised spare parts inventories, decommissioning and disposal of old equipment and active waste, staff training, additional or reduced staff, customer interface changes, etc.)?
- Does the work involve any unique evolutions or abnormal requirements (confined spaces, special shielding, infrequently performed lifts, plant configurations, etc.)?
- How are proposed changes to the project scope assessed for their impact on the project schedule and budget? Whose approval is required to modify the project scope?
- Whose authorization is required to allocate project contingency funds?
- What is the sensitivity of project implementation to reactor disruptions or changes in customer demands (or vice-versa)? Given a conflict, how are priorities agreed?

7. MATRIX AND SPECIFIC PROJECT EXAMPLES

The matrix presented in Table 1 reflects the contributions to this report organized by content. Each paper has been indicated as to the general motivation for the project, the scope of the project and different project management details contained therein. The table is

provided to assist the reader to find specific project examples of interest. The specific contributions follow, organized by country.

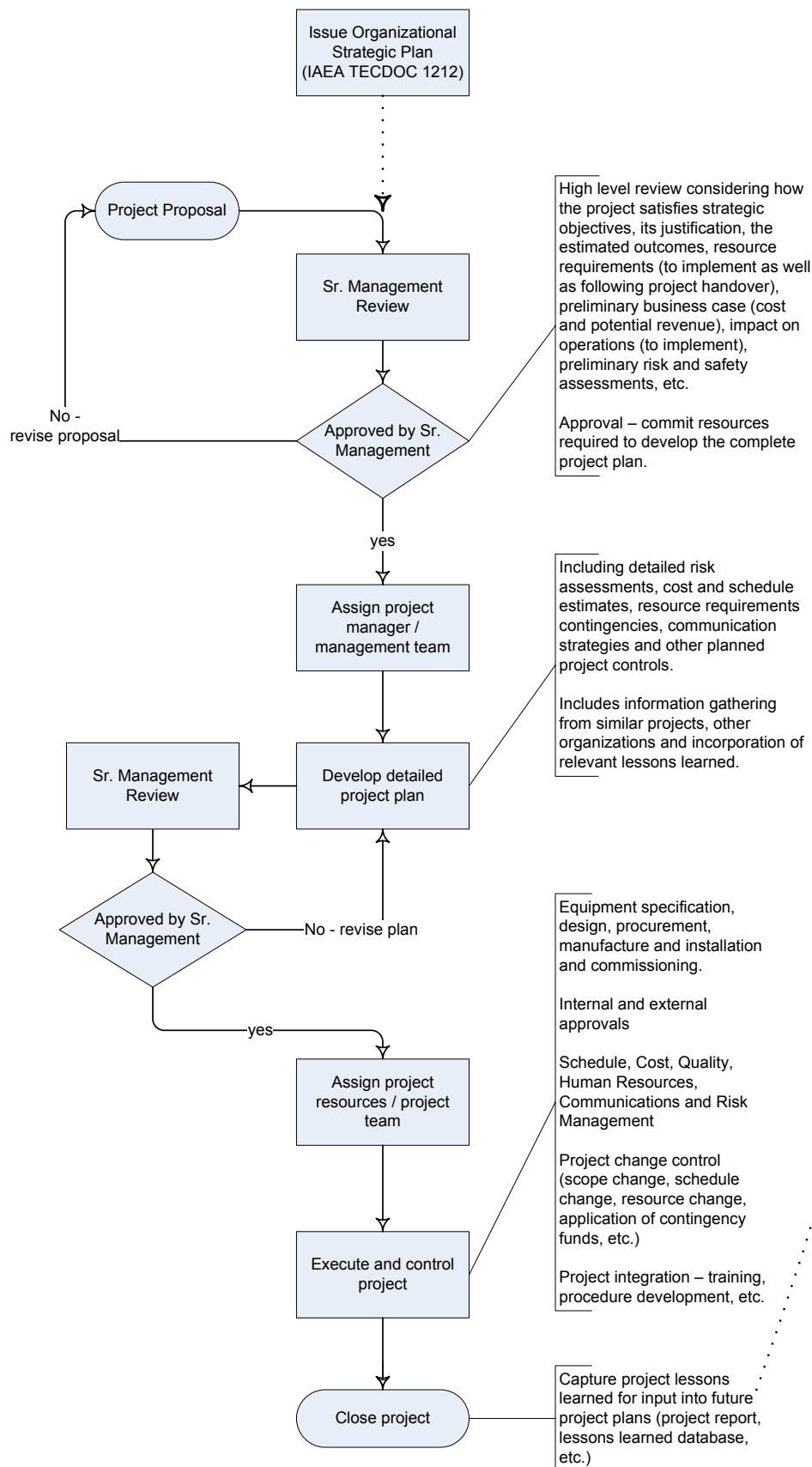


FIG. 1 Project flowchart.

TABLE 1. PROJECT EXAMPLES

Contribution	Basis for M&R / Project					What was modified				M&R Project Management					
	Customer demands	Training	Performance restoration or improvement	Safety	Availability	Linked to HEU/LEU conversion	Reactor & reactor support system M&R	Utilization / experimental facilities	Building infrastructure M&R (seismic, fire, security)	I&C system modification	Project plan	M&R management group / team	M&R budget / financial management	Licensing / regulator interface	Communicating lessons learned
Argentina RA-6			X			X	X				X				
Austria TRIGA-II	X									X					X
Bangladesh 3 MW TRIGA			X	X	X		X		X	X	X			X	X
Brazil IEA-R1	X		X	X	X		X	X	X	X	X		X		X
Czech Republic VR-1 TD		X		X		X	X		X	X	X				
France ILL HFR	X		X	X	X		X	X	X	X	X		X		X
Germany AKR, Dresden				X			X		X	X					
Germany FRG-1			X					X							
Hungary KFKI-BRR			X	X	X		X	X						X	X
India FBTR & KAMINI					X					X					X

TABLE 1. PROJECT EXAMPLES (cont.)

Contribution	Basis for M&R / Project					What was modified				M&R Project Management					
	Customer demands	Training	Performance restoration or improvement	Safety	Availability	Linked to HEU/LEU conversion	Reactor & reactor support system M&R	Utilization / experimental facilities	Building infrastructure M&R (seismic, fire, security)	I&C system modification	Project plan	M&R management group / team	M&R budget / financial management	Licensing / regulator interface	Communicating lessons learned
Indonesia Bandung TRIGA 2000	X		X	X			X				X	X	X		X
Netherlands Delft															X
Norway JEEP				X	X		X					X			X
Pakistan PARR-1			X	X	X	X			X						X
Portugal RPI					X					X	X		X		
Romania TRIGA-INR RR			X	X		X			X	X					
Ukraine WWR-M					X						X				
USA UT TRIGA		X	X				X	X			X	X		X	X
USA ORNL HIFR	X		X					X			X	X			X
Vietnam DNRR					X					X					X

REFURBISHMENT AND MODERNIZATION OF THE RA-6 RESEARCH REACTOR

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Abstract

The RA-6 reactor, located at the Bariloche Atomic Centre, is owned and operated by the Argentinean National Atomic Energy Commission (CNEA) and was commissioned in October 1982. It is a 500 kW, MTR, open pool-type reactor with highly enriched fuel elements. The main goal of the present project is to upgrade the reactor power to 3 MW in order to improve current applications and to develop new ones. To achieve this objective, a new core with LEU fuel elements was developed. Based on previous experience in the certification of a similar fuel element originally designed for the OPAL reactor (Australia), the new design is based on low enriched (19.7%) uranium silicides, including burnable poisons. The proposed core configurations are focussed on installing in-core irradiation facilities as well as for maximizing the performance of the external beam tubes facilities. The reactor modifications include the primary cooling system, the secondary cooling system, the electrical system, the instrumentation system, the shielding of some external facilities, etc. Safety studies related to the project are being performed in order to obtain a new licence for the entire facility, not only because of the modifications but also to generate updated regulatory documents; including the safety analysis report, the code of practice, the operation manual, the maintenance manual, etc. The funds for the core conversion are being provided by the United States Department of Energy.

1. INTRODUCTION

The reactor was designed and built entirely in Argentina and has been in operation since October 1982. The main design goal was to be a school reactor to support the field of nuclear engineering and to be a nuclear experiment facility for the Nuclear Engineering Department.

2. REACTOR DESCRIPTION

The RA-6 is an open pool-type reactor, cooled and moderated by light water with 500 kW nominal power. Figure 2 shows a view of the reactor block. The reactor core configuration is variable inside an 8×10 grid and consists of about 28–32 fuel assemblies with 90% enriched ^{235}U and a number of graphite reflectors. There are two different fuel assemblies:

- The standard assembly (horizontal cross-section $81 \text{ mm} \times 77 \text{ mm}$, height 750 mm) has 19 plates with approximately 8 g of ^{235}U per plate.
- The new control assembly is similar to the previous assembly; however, four plates (numbers 2, 3 and 17, 18) were removed and replaced by two channels in which a forked control device can move up and down.

The control rods are made of cadmium with stainless steel clad. Normally, the core configuration has 5 control assemblies, 4 in the central area and one in the periphery. Their drive mechanisms are installed on the top of the pool at the mechanisms bridge. Each reflector element is inside an aluminium box with the same outside dimensions as a fuel element. The open pool itself is a stainless steel tank 2.4 m diameter and 10 m depth.

Generally, two kinds of irradiation devices may be used to irradiate different specimen types or samples. For long period irradiations, hermetic aluminium cans are

normally used and positioned in special boxes inside the core. A pneumatic irradiation facility is usually installed inside the core. The plastic rabbit capsule can be easily inserted and removed during reactor operations and is used for short period irradiation of samples. Internal and external graphite thermal columns were positioned in one lateral core face for 15 years; but in 1997, they were replaced by the internal filter and the external port of the BNCT facility.

There are 5 irradiation beam tubes, 2 beam tubes run from the external side of the reactor block to the core crossing the concrete shield and pool water. The other 3 beam tubes go up to the external pool liner and special devices are needed to reach the core.

Other important reactor features are:

- The instrumentation and the safety and control logic system are more than adequate for our small reactor. However, it is a power tool for introducing students and trainees to more complex systems.
- The situation described above also applies to the ventilation system.
- A second control room that provides all the information about the reactor's parameters is used to perform supervised experiments.

3. REFURBISHMENT AND MODERNIZATION PROJECT

To take advantage of the reactor's special features, it was decided to increase reactor power to 3 MW. This decision was taken in order to:

- Improve current experimental facilities associated with CNEA Projects. These facilities include boron neutron capture therapy, instrumental neutron activation analysis, prompt gamma neutron activation analysis, and neutron radiography.
- Develop new applications, primarily to increase the capability of bulk radioisotope production to serve as a backup to the RA-3 reactor and to begin neutron beam development for neutron diffraction.

An agreement between Argentina and the USA was signed to make the RA-6 conversion from HEU to LEU. This agreement states that the funds for the new core design and construction will be provided by the US Department of Energy and other costs, like reactor modifications and licensing, will be supported by the Argentinean National Atomic Commission.

3.1. Fuel elements

The new fuel element design has been completed. It is based on high density (4.8 g U/cm^3) 19.7% enriched uranium silicides, and is based on previous experience by the OPAL Reactor (Australia) in the certification of a new fuel element.

The design includes the use of cadmium wires located at the external fuel elements structure along 50 cm, centred on the 61.9 cm fuel element active length. This burnable poison is used to decrease the reactivity induced for the fresh fuels and to help minimize the power peak factor of the core configurations. In Figures 1 and 2, the standard fuel assembly and the control fuel assembly horizontal cross-sections are shown.

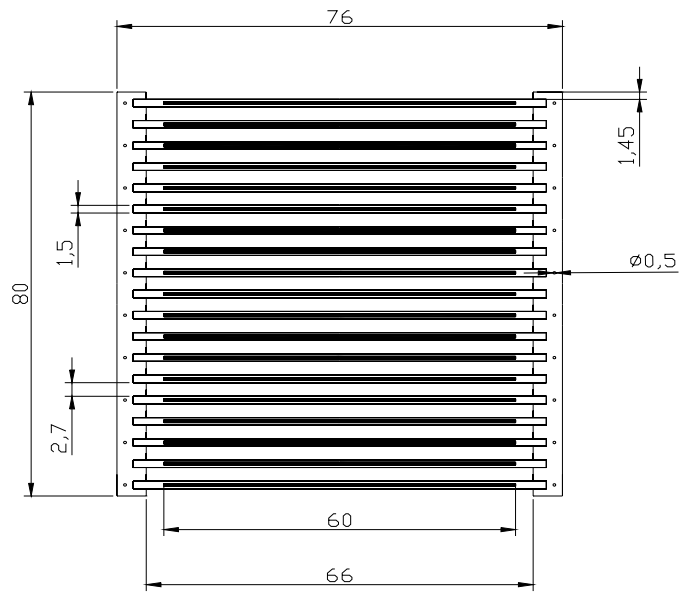


FIG. 1. Standard fuel assembly horizontal cross-section (dimensions in mm).

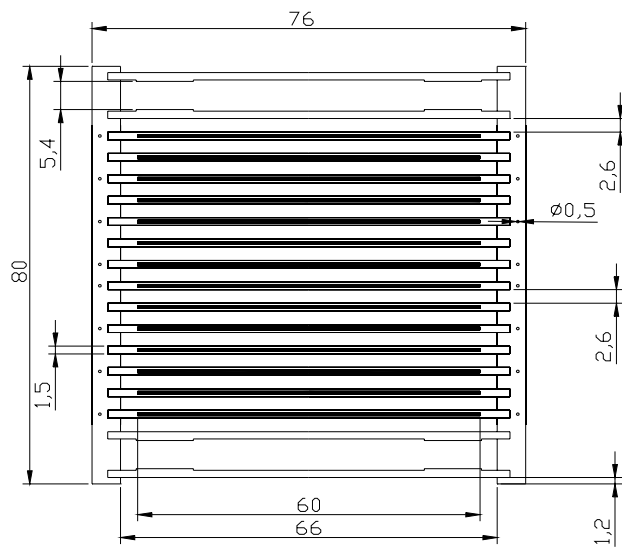


FIG. 2. Control fuel assembly horizontal cross-section (dimensions in mm).

3.2. Core configuration

Figure 3 shows the planned core configuration.

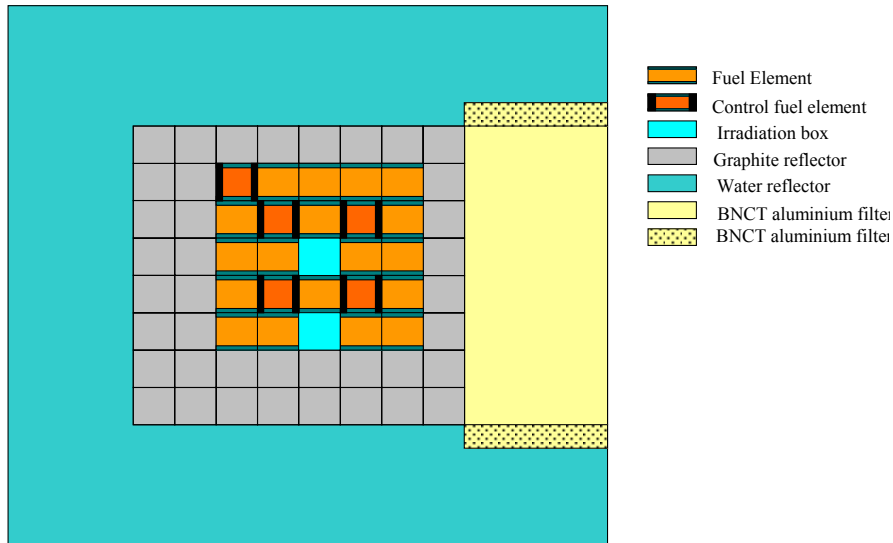


FIG. 3. New core configuration.

The main modifications related to the current core configuration are:

- Two in-core irradiation facilities to improve the irradiation performance.
- One graphite reflector row between the core and the BNCT filter in order to decrease fast neutron beam contamination.
- Two graphite reflector rows between the core and the neutron radiography and PGNAA facilities to improve the beam spectrum.

3.3. Primary and secondary circuits modifications

The primary and secondary cooling systems will be modified in order to dissipate 3 MW. In the primary cooling system, the water flow rate will be increased from 150 m³/h to 340 m³/h. This implies changes in the main pump and heat exchanger and a modification of the syphon breaker and flapper valve. Figure 4 shows the present system and Figure 5 shows engineering of the new system. Figures 4 and 5 also show the current and the modified secondary cooling systems, respectively. Figure 6 shows the new cooling tower system connected to the secondary circuit.

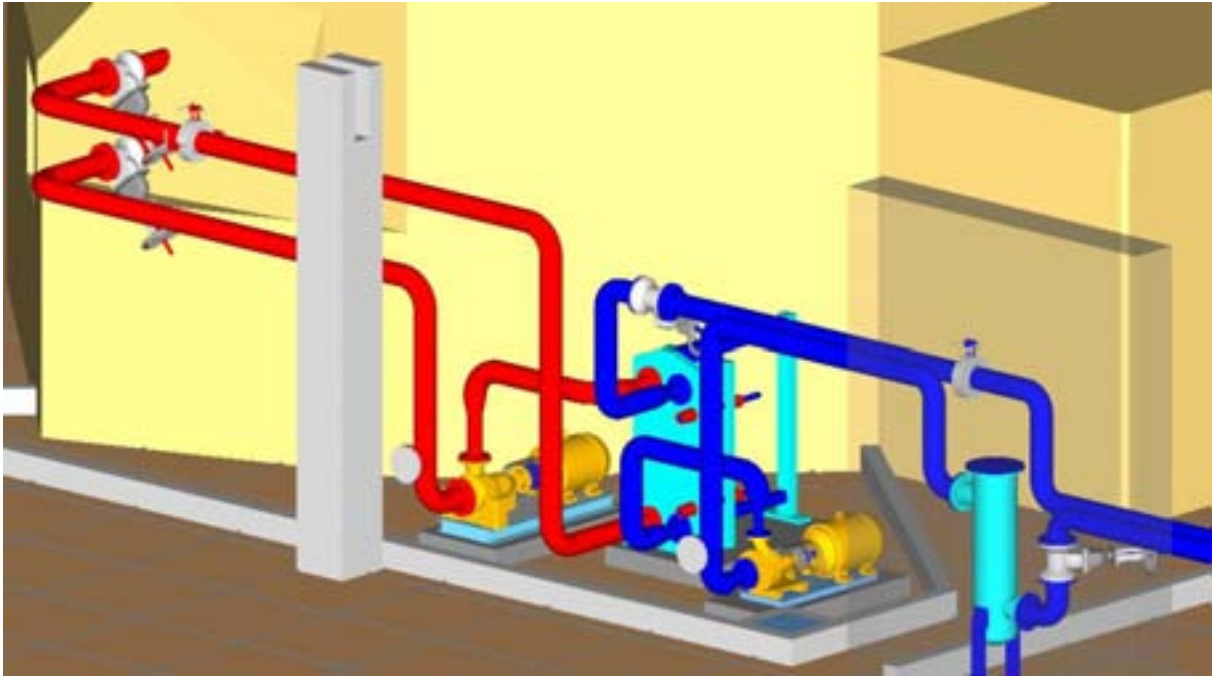


FIG. 4. Current primary and secondary cooling systems.

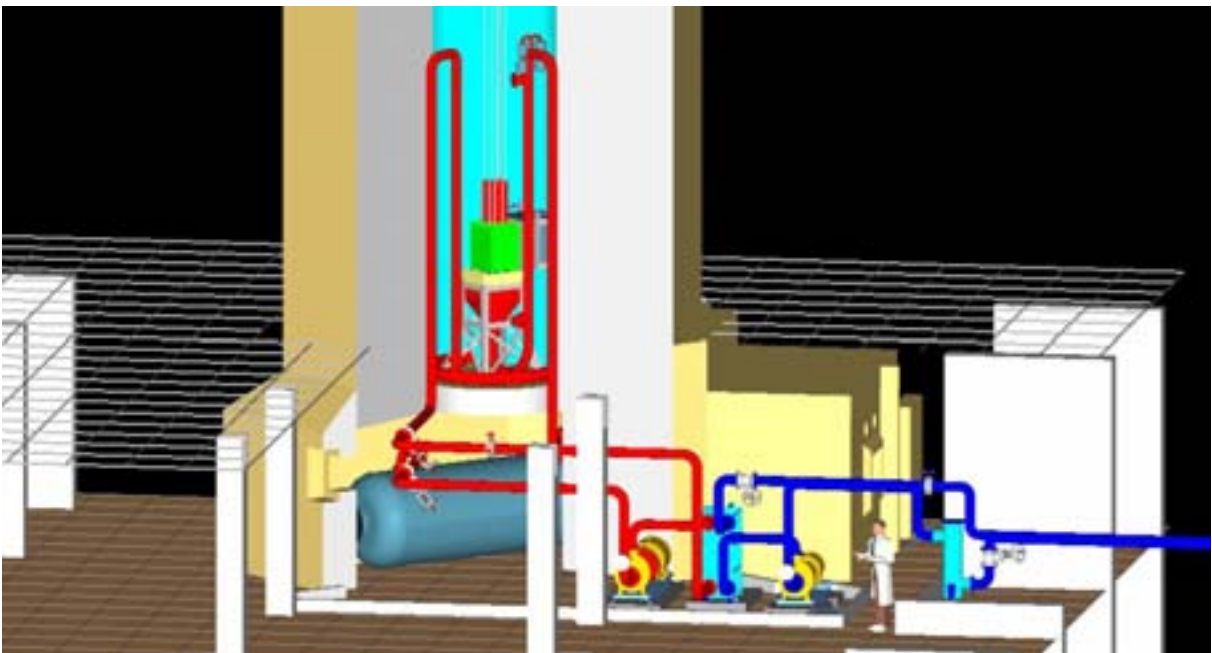


FIG. 5. Modified primary and secondary cooling systems.

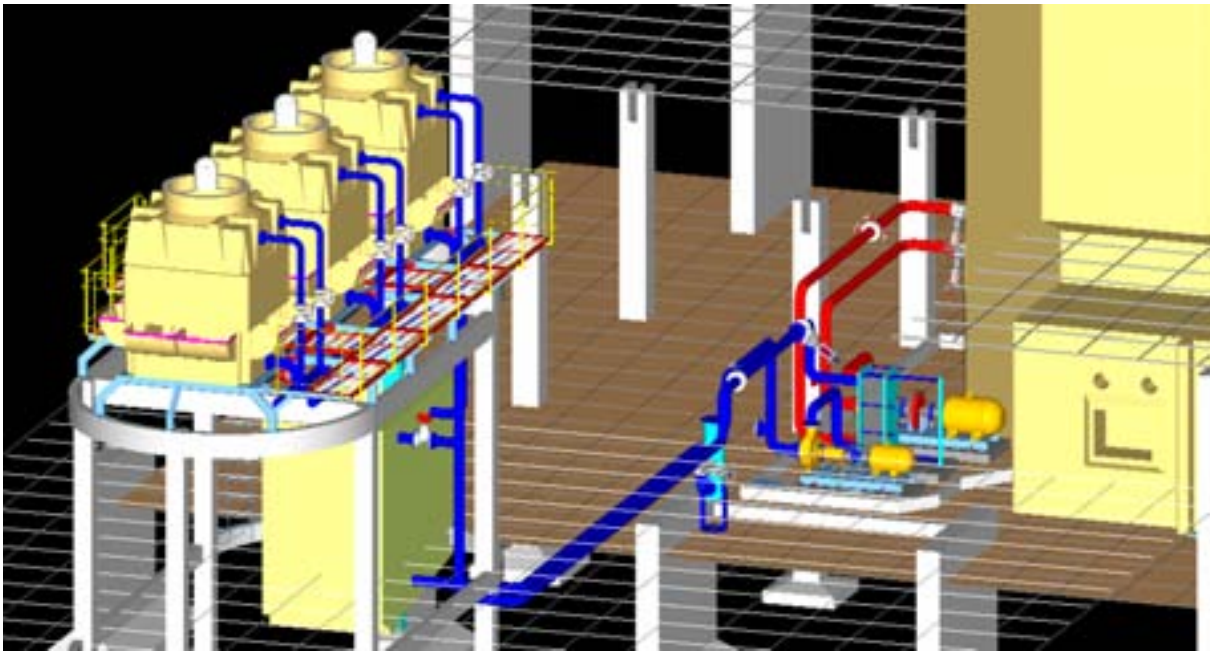


FIG. 6. Cooling tower system.

3.4. Main project tasks

The main engineering tasks that will be performed during the project include:

- Neutronics: Fuel element definition, critical core configuration, fuel assembly technical specifications, peak factor, reactivity feedback coefficients, kinetics parameters, startup and transition core configurations, etc.
- Radiological safety: Shielding, liquid wastes, gaseous release evaluation for the new operating conditions engineering and implementation of a continuous gaseous release monitoring system, modification of the pneumatic irradiation facility, relocation of area monitors, etc.
- Thermohydraulics: Maximum flow rate admissible in the primary circuit, flow rate determination in the different core channels (hydraulic experiment), convection coefficients determination (thermal experiment), hot channel characterization, technical specifications for the heat exchanger, pumps and cooling towers, etc.
- Nuclear safety: Selection of initiating events, evaluation of individual event sequences, probabilistic safety assessment (PSA) Levels 1, 2 and 3, comparison between PSA results and National Regulatory Authority (ARN) acceptance criteria, etc.
- Mechanics: Primary and secondary circuits layout, siphon breaker design, core pressure drop monitoring system, etc.
- Electricity: Revision and fitting of the electrical system for the new operating conditions, etc.
- Instrumentation and control: Revision and fitting of the instrumentation and control system for the new operating conditions, etc.
- Licensing: Rewrite all the mandatory documentation including the safety analysis report, etc.

3.5. Accomplished goals

Major plant modifications will be completed by the end of April 2008 and the new core will be available by mid-May 2008. The safety analysis is currently being completed, but creation of the safety analysis report will take an additional month. The remainder of the required documentation will be updated after the commissioning stage.

Thermal experiments have not yet been completed, so during the commissioning stage, the reactor power will not exceed 2 MW. After that, a comprehensive in-core programme will be implemented with the objective to assess the margin to nucleate boiling and assess the thermal transfer conditions in order to evaluate the feasibility of increasing power up to 3 MW.

The project schedule was modified several times due to consecutive delays in the implementation of different tasks. Project planning had to consider the fact that personnel involved with the project also had to support other government-related nuclear power projects.

EXPERIENCE WITH MODERNIZATION AND REFURBISHMENT OF THE VIENNA TRIGA MARK II REACTOR I&C SYSTEM

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Abstract

The rapid development in electronics and computer software and hardware is an important incentive for many research reactor operators to verify the present state of the art of its instrumentation and control (I&C) system (including reactor protection) and to carefully plan the replacement of this system. For this task, both the international and the national safety requirements have to be respected, the future strategic plans for the reactor have to be considered, the financial situation has to be resolved, and proper timing of the replacement has to be made. Although some I&C providers offer I&C packages, the reality is that every research reactor is different and the I&C package has to be carefully tailored to local requirements and needs. This makes the whole replacement procedure costly and lengthy. One major reason to upgrade and modernize an I&C system at a facility is the obsolescence of its I&C system, the unavailability of spare parts, an increased failure rate of the I&C system leading to frequent reactor shutdowns, and long repair periods that result in a high unavailability of the facility. Additional aspects that support a decision to modernize are the technological progress in I&C systems during the past decades that have led to higher reliability of I&C systems, improvement of man-machine interface, and extensive and fast data collection and processing. In addition to these technically-based decisions, changes in technical specifications and/or regulatory requirements may also influence the final decision for modernization of the I&C system of a given facility. An I&C modernization process might also be accomplished in conjunction with a facility power increase. The following paper summarizes the replacement procedure and the reactor-specific modifications to the standard reactor instrumentation offered by the supplier and the operation experience since its installation in 1992. Further, it compiles the benefits and other issues to be considered when changing from an analogue to a digital I&C system.

1. HISTORICAL DEVELOPMENT

The Vienna TRIGA reactor went critical on 7 March 1962. At that time, the reactor operated with the original General Atomics (GA) electronic tube-type console with a very simple design and incorporated all necessary features of an I&C system required at that time. In 1968, this console was replaced by transistorized instrumentation that worked very satisfactorily until 1992 when spare parts became unavailable. In 1990, it was decided to replace this aged instrumentation with a state of the art digital console. At that time, three quotations were received. Of the three quotations, one company offered an I&C system that existed only in drawings and could not supply a reference system. Another company had references for research reactor I&C systems, but could not process the transient mode operation of a typical TRIGA reactor. Therefore, General Atomics (GA) remained as they had already supplied a few digital I&C systems in the USA and their system was designed to handle transient operation. The new computerized I&C system was ordered from GA in 1990 and was installed and tested during summer 1992. The reactor went critical on 10 November 1992 after a two-month shutdown period. This paper summarizes the reactor-specific additions to the standard GA I&C package and operation experience during the past 15 years.

2. SYSTEM DESIGN

The original instrumentation design proposed by GA (Fig. 1) followed the US requirements of the time. At that time, five reference instrumentation systems had been installed, four of them in the USA and one of them in the UK. Due to differences in safety requirements between US and European authorities, the proposed system had to be

extensively modified to fulfil the Austrian requirements, which are almost identical to German requirements.

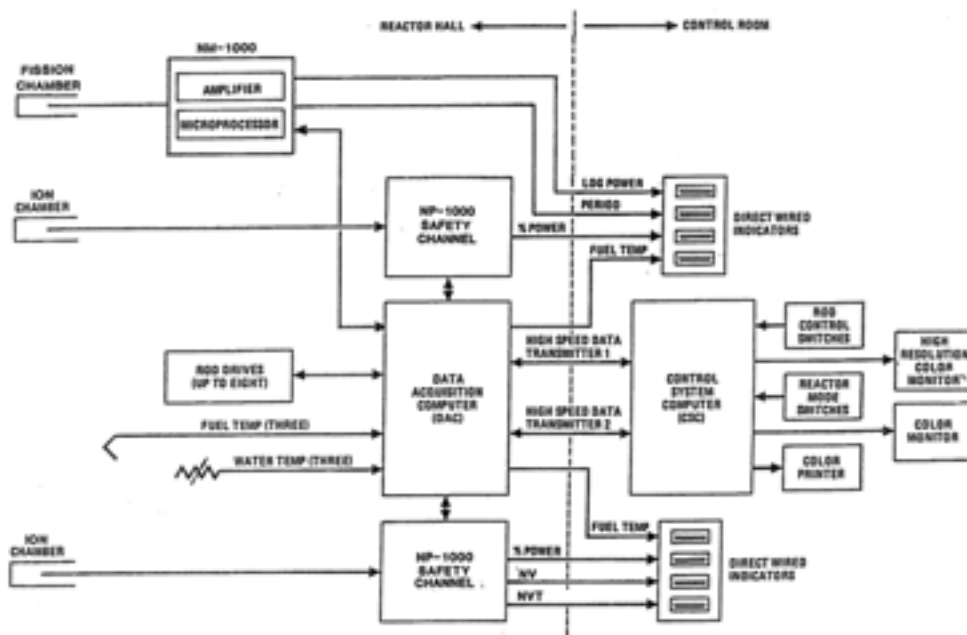


FIG. 1. I&C system as proposed by the supplier.

The originally offered I&C system was considerably modified to meet the stringent requirements of the licensing authority. Although one wide range channel was microprocessor-controlled, it was considered to only be an operational channel with no safety relevance. The data display is shown on one graphic monitor (Fig. 2) and one status monitor (Fig. 3). After a couple of early failures, the instrumentation operated satisfactorily and was successfully used in training courses for students and guest scientists. Due to the rapid development in computer software and hardware, as well as the suspected Millennium computer bug, it was decided in autumn 1999 to replace both software and hardware parts with a new design, also supplied by GA. This new version was installed in April 2000.

The main component of the new instrumentation system is a wide range channel (NM-1000) that operates at low power in the pulse mode and in the Campbell mode at higher power. From this channel, important parameters are extracted and displayed on a colour graphic screen, such as: linear power, log power, period, count rate, and percent power. In this channel, a microprocessor is involved using specially developed software that is very difficult to verify. As a redundant safety channel, GA offered a one percent power channel (NP-1000) that created a hard-wired shutdown signal at nominal power plus 10%. This meant that only one hard-wired safety channel was available in the proposed system. This design was unacceptable according to European standards and had to be modified. The modified system design is displayed in Figure 4. Besides the NM-1000 channel, which is considered only as an operational channel with no safety relevance, two identical hard-wired linear channels (NMP-1 and NMP-2) have been installed. Both channels are constantly compared and any discrepancy exceeding 10% triggers an alarm. Further, both channels are switched together by a range switch, resulting in a reactor scram signal in each power range at 10% overpower. In

addition, two strip chart recorders for linear power and for fuel temperature recording during pulse operation are provided.



FIG. 2. Graphic window of the TRIGA Vienna.



FIG. 3. Status window of the TRIGA Vienna.

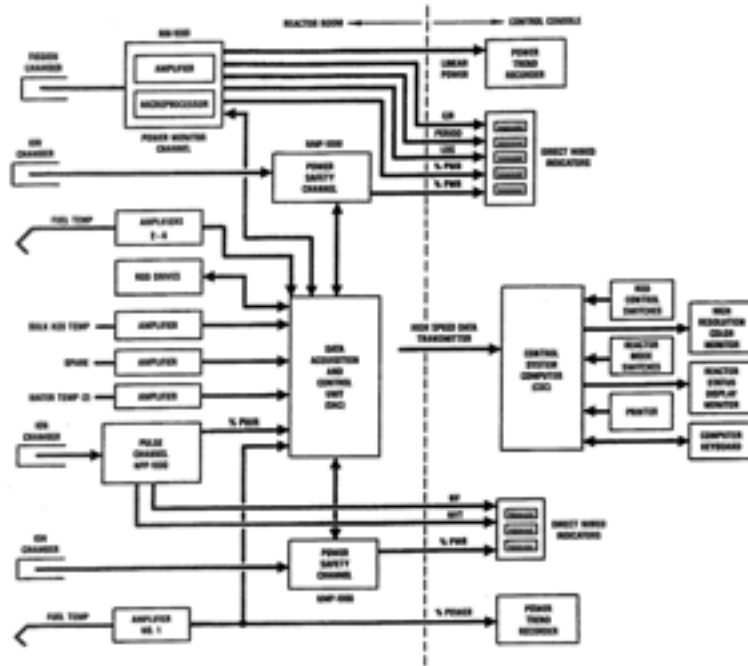


FIG. 4. I&C system as installed.

3. OPERATION EXPERIENCE

During the past years of operation, a number of failures occurred that made the reactor inoperable for a total of about 60 days. Some errors were localized within a short time; other errors were more difficult to overcome. For example, in February 1994 a computer drive in the data acquisition computer (DAC) failed and a new drive loaded with the Austrian DAC software was ordered from GA. In June 1994, the drive in the control system computer (CSC) failed and again a new drive loaded with the Austrian CSC software was ordered from GA. Apparently, both the DAC and the CSC software that had been used in Vienna before those problems were not the same as the new software loaded by GA. Therefore, problems arose with pulse operation. In fact, after June 1994 pulse operation was not possible because of this problem. Finally, the problems were solved and the system was fully operational again. No major problems with the instrumentation have occurred since the installation of the new version in April 2000.

The system is quite sensitive to room temperature. During the summer, control room temperatures of 35°C are sometimes reached and some channels in the DAC cabinet have surface temperatures above 55°C. Also, the microprocessor of the NM-1000 is very temperature-sensitive and the channel constants are erased above 30°C. The front and back doors of the DAC cabinet were removed and two ventilators were installed on top of the cabinet and greatly improved air circulation in the DAC.

Another problem that occurs from time to time is as follows: The automatic pre-start test fails with error information. An investigation showed that some components in the Campbell channel have changed their characteristics due to ageing. Replacement of these components solved this problem. Lightning may also influence the constants set in the NM-1000.

In the past five years, both CRT monitors were replaced with flat screens. However, special conversion software had to be purchased to adapt the flat screens to the existing hardware. Table 1 summarizes the problems experienced with the digital instrumentation in the past 15 years.

TABLE 1. DIGITAL REACTOR INSTRUMENTATION TRIGA VIENNA – MAJOR PROBLEMS SINCE 1992

Date	Cause	Consequence
Sep. 92	Thunderstorm	ROM of wide range channel destroyed
Oct. 92	Microswitch corrosion	Rod drive signals wrong
Jan. 93	Loose contact	Period scram
Mar. 93	High temperature in DAC	Reactor scram
Mar. 93	Broken capacitor	Transformer dead
Mar. 93	Loose wire	No magnet current
Apr. 93	Short circuit	Fuel temperature low
Jun. 93	Console power 'On' switch	Power supply DAC failed
Jan. 94	Automatic control dead	Connector at hard disk loose
Jun. 94	CSC computer dead	Hard disk to be replaced
Dec. 94	CSC computer problems	Configurator at CSC lost
Jul. 95	High temperature in control room	NM-1000 constants lost
Aug. 95	NM-1000 wrong constants	Li-Cd batteries replaced
Dec. 95	Spurious scrams NM-1000	Noise pick-up by Campbell channel
Aug. 96	Noise pick-up by NM-1000 channel	Simulation of several Watts of power in shutdown mode
Jan. 97–Sep. 97	Noise problems	Spurious scrams
Jan 98	Loss of constants at NM-1000 channel	Change of potentiometer
Sep. 98	NMP 1 voltage low	Replacement of compensated ion chamber at NMP 1
Feb. 99	NMP 2 voltage low	Replacement of compensated ion chamber at NMP 2
Aug. 99	Hard disk problems at CSC	Replacement of CSC hard disk
Oct. 99	Hard disk problems at DAC	Replacement of DAC hard disk
Nov. 99		New CSC and DAC computers plus software ordered
Jun. 2003	Failure of Action Pak	New Action Pak from local supplier
Sep. 2003	Failure of CIC	Reactor scram, new CIC
Dec. 2003	CSC problems: Computer not registered with IBM-Austria	No local spare parts
Sep. 2004	Low contrast of graphic monitor	New graphic monitor
Jan. 2005	Failure of UIC	No pulse operation
Aug. 2005	Microswitch shim rod broken	New switch
Jan. 2006	Noise pick-up by NM 1000 channel	Simulation of several Watts of power in shutdown mode
Aug. 2007	Graphic display monitor	New graphic monitor defect

The instrumentation design originates from the mid-1980s and most of the electronic equipment, especially the console computer and the DAC computer, were already outdated at the time of installation. Due to the rapid development in data acquisition technology, the problem of spare parts was imminent. However, with careful planning and the cooperation of the supplier, many problems have been recently resolved. The present instrumentation design fulfils both the requirements of the national authorities and a university training reactor that is mainly used in academic education. The rapid development in electronics and data acquisition accelerates design ageing of the system. It is also a challenge to keep the system in good and state of the art condition.

4. SUMMARY REMARKS TO BE CONSIDERED WHEN CHANGING FROM ANALOGUE TO DIGITAL I&C SYSTEMS

Modern digital technology has a wide variety of beneficial capabilities compared to analogue technology. Therefore, in many cases, making a 1:1 replacement (just replicating the capabilities of the old system with new technology) is not the best solution for the facility. When a new system is being implemented, the potential beneficial capabilities of the new technology should be evaluated to determine which are appropriate for inclusion into the system to achieve facility goals, such as increased reliability, availability, power output, etc. The following provides a list of some of the important benefits and issues to be considered when changing to a digital I&C system.

4.1. Benefits

- Measurement precision: Digital instruments do not have the drift problems associated with analogue instruments. It is also possible to use digital technology to more accurately measure parameters than was possible with analogue technology.
- Reduced equipment volume: Digital technology has the ability to process a large amount of data in one processor enabling the reduction of the volume of the whole system.
- Improved reliability: Digital technology can be used to achieve higher system reliability, e.g. by including a redundant processor that is in a standby state. In the case of a failure in the active system, the function of the system could be switched to the redundant standby processor with no interruption of the system function.
- Simplification of fault detection: Digital technology can incorporate self-testing and self-diagnosis for fault detection.
- Complex function capability: Digital technology can easily implement complex functions since the software does not have the same limitations as hardware since it is more versatile and does not require the addition of more and more components.
- Adaptability and ease of modification: Digital technology allows easy modification of existing algorithms and the incorporation of new capabilities in the system since the implementation is in the software and not the hardware.
- System monitoring capability: Digital technology can easily incorporate self-monitoring into a system so that it can observe its own performance.
- Operator support: Digital technology is easier to process all the data as long as the data are incorporated into the system so that modification of necessary information as required by the operators is easily accomplished.
- Installation ease: Digital equipment is easily installed owing to the reduction in the number of cables through multiplexing, data buses, and fibre optics.
- Self testing capability: Digital technology can easily incorporate self-testing functionality.

- Simplification of cabling: Digital equipment has a great advantage of reducing cabling once all the necessary inputs are incorporated into the system.
- Ease of system upgrading: The system architecture of digital technology easily accommodates future version updates.
- More attractive for young engineers: Digital technology is a current technology of interest to young engineers because they are familiar with it and is technology they learned in school. It is very difficult to get new engineers to work in nuclear power plants with an analogue technology they do not understand and have no interest in learning.
- Maintenance costs: Maintenance costs for digital systems will soon be cheaper than maintenance costs of analogue systems (price of digital technology going down, no offset setting, self-testing and self-diagnostics, etc.)

4.2. Issues to be considered

- High development costs: The development costs of new systems may be high due to the verification, validation (V&V) and licensing processes of expensive software-based safety systems.
- Software common mode failure risk: Without suitable hardware and software architectures and proper development processes in the development of the new systems, there is a risk of introducing common mode failures through the software. This risk can be reduced through the proper use of verification and validation (V&V) and diversity.
- Quantified assessment of reliability: If a quantified assessment of the reliability is required, for example, for PSA purposes, it should be noted that it might be very difficult to come up with defensible reliability estimates for software-based systems.
- Retraining of operating and maintenance staff: New systems may introduce the need for new training and skills in both the operations and the maintenance staff. On the other hand, these skills may be easier to find on the open market than skills in the old analogue systems.
- Absence of standards: There is an emerging body of standards available for digital systems, but it may be difficult to match the old standards with the new ones. There also seems to be less international consensus among licensing bodies on how to treat digital systems.
- Acceptance by regulatory bodies: Experience has shown that national safety committees are sometimes reluctant to accept that a computerized I&C system can guarantee safety.
- Verification and validation: Experience has shown that digital systems need a considerable amount of effort to ensure that they are working properly in all operational modes and that they are not exhibiting unintentional functionality in any operational mode.
- Difficulty of identifying all possible defects: Due to the complexity of digital software systems, it is almost impossible to provide proof of completeness in all operational modes.
- Short technological lifetimes: Digital systems often exhibit rather short technological lifetimes. Therefore, it may be necessary to more proactively combat obsolescence when using digital systems as compared with the old analogue systems.
- Qualification of tools: There are many computer-based tools available for the design and V&V of digital systems. The benefit of these tools may, however, be reduced due to difficulty of proving they are producing correct results.

- Problems with staff acceptance and retraining: The change of technology is sometimes considered to be very large and, therefore, may be difficult to get staff acceptance of the new systems. Early involvement of the staff when considering new digital technology usually helps in this regard.

5. FINAL REMARKS

The refurbishment of the I&C system of a research reactor is a major task that needs careful planning taking many aspects into account as described above. At an early planning stage, the future of the facility has to be shown to the national authorities by providing a detailed business plan because the costly I&C replacement will be compared by financial authorities against the cost of decommissioning the facility.

EXPERIENCE ON THE REFURBISHMENT OF THE COOLING SYSTEM OF THE 3 MW TRIGA MARK II RESEARCH REACTOR OF BANGLADESH AND THE MODERNIZATION PLAN OF THE REACTOR CONTROL CONSOLE

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Abstract

The 3 MW TRIGA Mark II research reactor of the Bangladesh Atomic Energy Commission (BAEC) achieved its first criticality on 14 September 1986. Since then, the reactor has been used for manpower training, radioisotope production, and various R&D activities in the field of neutron activation analysis (NAA), neutron radiography (NR), and neutron scattering. Full power reactor operations remained suspended from 1997–2001 when a corrosion leakage problem in the ^{16}N decay tank threatened the integrity of the primary cooling loop. The new tank was installed in 2001 and some modification and upgrades were carried out in the reactor cooling system such that the operational safety of the reactor could be strengthened. The cooling system upgrade mainly included replacement of the fouled shell and tube-type heat exchanger by a new plate-type one, modification of the cooling system piping layout, installation of isolation valves, installation of a chemical injection system for the secondary cooling system, modification of the Emergency Core Cooling System (ECCS), etc. After successful completion of all these modifications, the reactor was made operational again at full power of 3 MW in August 2001. BAEC, the operating organization, is now implementing a government-funded project to replace the old analogue control console of the research reactor with a digital control console. This paper focuses on the modification of the cooling system as well as the I&C system and the upcoming control console upgrade of the 3 MW TRIGA Mark II research reactor of Bangladesh. It also presents short descriptions of major incidents encountered so far in the reactor facility.

1. INTRODUCTION

The TRIGA Mark II research reactor of BAEC is a light water cooled reactor with a graphite reflector. It was designed for steady state and square wave operation up to a power level of 3 MW(th) and for pulsing operation with a maximum pulse power of 852 MW. The reactor is designed for multipurpose uses, such as training, education, radioisotope production, and various R&D activities in nuclear science and technology. The reactor achieved its first criticality on 14 September 1986 and was commissioned at a full power of 3 MW in the same year. Since then, it has been used for manpower training, radioisotope production, and various R&D activities in NAA, NR and neutron scattering. During this time, reactor operations were interrupted several times due to various incidents, mostly in the reactor cooling system. The most severe of these incidents was the ^{16}N decay tank leakage incident that took place in 1997 (Fig. 1). As a consequence of this incident, full power operation of the reactor remained suspended for several years. During that period, the reactor was operated at 250 kW under natural convection cooling mode, so as to meet the needs of experiments that required lower neutron flux (e.g. NAA, NR, etc.). Operation of the reactor at a lower power level was made possible by establishing a temporary bypass connection across the decay tank using local technology. To make the reactor operational again at full power, BAEC implemented a government-funded Annual Development Programme (ADP). Under the project, renovation and upgrading of the entire reactor cooling system was carried out. The renovated cooling system was successfully commissioned in June 2002 and it was

possible to restore the reactor to full power operation after five years of reduced power operations.

Since July 2004, the reactor has been used for production of ^{131}I on a routine basis. Since replacement of the damaged central dry irradiation tube in June 2003, a total of 129 batches of ^{131}I (1 764 GBq) have been produced to September 2006. The reactor is also regularly used to irradiate samples under various programmes in the areas of NAA, NRR, and neutron scattering. The annual operation data of the reactor are graphically shown in Figure 2.



FIG. 1. Removal of the damaged ^{16}N decay tank and marks of corrosion pitting on its surface.

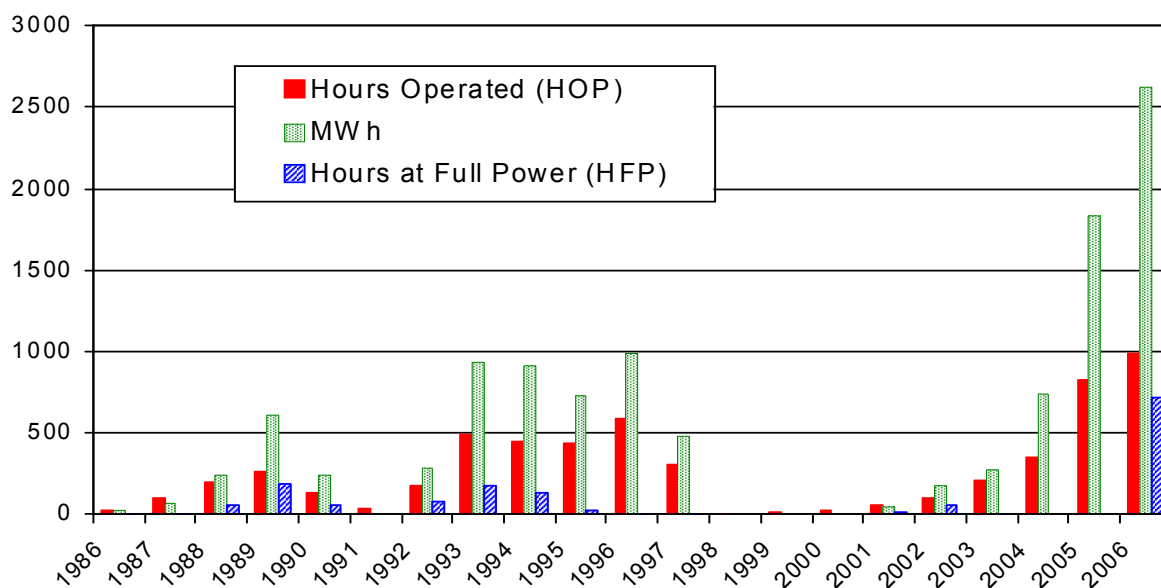


FIG. 2. Operation data of the reactor during Sep. 1986–Sep. 2006.

2. INCIDENTS ENCOUNTERED IN THE REACTOR FACILITY

From Figure 2, it can be seen that distinct rises and falls exist in the profile of the operational data of the BAEC research reactor. The cause of the fluctuations is related to some major incidents that hampered normal reactor operations. So far, five incidents have occurred in the BAEC TRIGA reactor facility. Of these, two incidents had a prolonged impact on the normal functioning of the reactor. Brief descriptions of the reportable incidents are given below:

Incident-1: A crack was discovered in the welding joint of the exi-check valve of the primary cooling system on 4 September 1990. A thorough study of the failure showed that the crack developed primarily due to a faulty primary pump foundation and the pipe support system. The system design partly contributed to the excessive vibrations. The system was repaired, tested, and cleared for normal operations.

Incident-2: A leak was detected in the decay tank of the primary cooling system on 14 July 1997. About 45 000 litres of demineralised water, with an activity concentration of about 28 Bq/L due to the presence of ^{58}Co , leaked from the primary cooling loop. The water was collected and contained in special storage and plastic containers. A number of independent investigations and assessments on the cause of the leakage and possible remedies have been conducted. The tank was isolated and later removed and analysed. Extensive corrosion and pitting were found in a particular area where rainwater had seeped in during the monsoon and accumulated over a long period, perhaps over a number of seasons. Corrosion and pitting were also observed on the inner walls/baffles of the decay tank. On 27 July 1998, the reactor was made operational at a power level of 250 kW (high power operations were not possible without a ^{16}N decay tank) by installing a temporary decay tank bypass connection. In 2000–2001, a new decay tank was installed. At the same time, several modifications in the primary cooling loop, such as replacement of the shell & tube-type heat exchanger with a plate-type heat exchanger, replacement of the T-connection at the discharge of the pumps with a modified Y-connection, etc. were implemented in order to make the reactor operational at full power.

3. MODIFICATION OF THE COOLING SYSTEMS

Since commissioning of the reactor, several modifications have been made in the reactor cooling system. Some of the recent modifications, which were carried out during 2000–2001 after the decay tank leakage incident of 1997 (mentioned above as Incident-2) are described in the following sections.

3.1. Installation of a new decay tank and a plate-type heat exchanger

A new decay tank with four aluminium saddles welded to its body was installed in the decay tank room. Each saddle was anchored to the floor with 4 steel-roued bolts (3.175 cm dia., 22.86 cm long). The saddles were bolted to the floor in a way such that one of the saddles remained fixed and the other three could slide on the floor. Sliding saddles having a length of about 10 m and a diameter of about 2.5 m were used in order to allow thermal expansion of the large decay tank. The new decay tank with the aluminium saddle is shown in Figure 3.

A new plate-type heat exchanger was installed to replace the old shell and tube-type heat exchanger. The new heat exchanger required several changes in the piping layouts of both the primary and the secondary cooling loops. The new heat exchanger with modified piping arrangements is shown in Figure 4.



FIG. 3. Installation of the new ^{16}N decay tank in the decay tank pit.



FIG. 4. Plate-type heat exchanger with modified piping arrangements.

3.2. Installation of a chemical injection system for the secondary cooling loop

A microprocessor-based chemical controller was installed for the secondary cooling loop. The controller injects three chemicals into the secondary loop water so as to maintain the secondary water chemistry parameters (conductivity, total alkalinity, chlorides, total hardness, silica, phosphonate) within permissible limits.

3.3. Modification of the cooling system piping arrangements

A Y-connection was introduced in place of the T-connection at the discharge side of the primary pumps and a butterfly valve was installed at the inlet of the decay tank. Two 25.4 cm butterfly valves were installed in the primary piping adjacent to the inlet and the outlet of the plate-type heat exchanger. Design of the secondary cooling system piping arrangement at the inlet of the heat exchanger was changed to facilitate the maintenance of the Y-strainers. A paddlewheel-type flow sensor with a digital readout panel was installed in the suction line of the secondary pumps to measure the flow rate of the secondary cooling loop.

3.4. Modification of piping supports

Necessary pipe supports were provided at different locations of the primary and secondary cooling loops in order to reduce the vibration of the piping to a minimum level. Three types of mild steel (MS) pipe supports were used for this purpose and include adjustable floor-mounted type, adjustable roof-mounted type, and wall-mounted type. In addition, a few supports that were simultaneously connected simultaneously to the floor and the wall were also installed.

3.5. Modification of the shielding arrangements around the decay tank

A concrete shielding wall with a thickness of about 1.12 m was constructed between the decay tank room and the primary pump room in order to protect personnel working in the primary pump room and adjacent areas from radiation hazards. A mild steel door (203.2 cm \times 63.5 cm \times 3.175 cm) was provided in the shielding wall to facilitate periodic inspection of the decay tank room. The top shielding in the decay tank room was also raised 50.8 cm from its previous position. A room with a corrugated iron (CI) sheet roof was constructed on the top of the decay tank room so that rainwater could not enter the decay tank room.

3.6. Modification of the emergency core cooling system (ECCS)

The ECCS is the single most important engineered safety system of the BAEC 3 MW TRIGA reactor and plays a key role in protecting the reactor fuel in the event of a loss of coolant accident (LOCA). The initial installation of the ECCS had several deficiencies, such as improper routing of the piping, defective installation of the battery, battery charger, and pump motor unit, etc. Therefore, in order to improve the operational safety of the ECCS, several modifications were needed in the system. These modifications were implemented after the installation of the new decay tank and associated components of the reactor cooling system, which comprises the plate-type heat exchanger, modified Y-connection, new isolation valves, etc. Modifications of the ECCS included the following:

- modification of the ECCS piping layout;
- shifting of the ECCS mounting block containing the ECCS pump-motor, battery and battery charger unit to a safe height;
- modification of the ECCS mounting block;
- replacement of the old ECCS lead-acid battery by new Ni-Cd battery.

After satisfactory completion of all the above modification works, the ECCS was tested and commissioned on 8 April 2003.

4. MODIFICATIONS AND UPGRADING OF THE REACTOR I&C SYSTEM

Modification of the I&C system started at the very beginning when the system was being commissioned in 1986. Some of the modifications carried out so far on the various I&C systems of the reactor are briefly described in the following sections.

4.1. High reactor pool level scram system (1986)

The primary cooling pump-motor trip system designed by General Atomics (GA), the reactor supplier, did not include any provision for a reactor scram under an abnormally high reactor pool level condition. It should be mentioned that the reactor pool level can increase when the return flow rate of the primary coolant exceeds the exit flow rate. This can happen because of partial closure of the exit motor operating valve (MOV) with respect to the return MOV. Therefore, in order to avoid any reactor tank overflow accident, two mechanisms were incorporated into the pump-motor trip system. One of these mechanisms uses a simple float switch (FS). When the FS is activated because of high pool level, the magnetic contactors of the primary pump motor starters are de-energized and the pumps stop functioning; thus, overflow of the reactor tank is prevented. The other mechanism uses a bi-stable trip circuit with a capacitance level probe as the sensor. These mechanisms can independently trip the primary pumps when the pool level increases beyond a preset value (about 7.5 cm above normal level). Tripping of the pumps causes the reactor to scram while it is operating under a forced convection mode of operation. However, none of these mechanisms can trip the reactor while it is operating under a natural convection mode of operation.

4.2. Instrument air interlock (1995)

The original design of the reactor hall ventilation system control circuit did not have an instrument air interlock to prevent operation of the ventilation system when the instrument air supply had been cut off by any means. As a result of the non-availability of this interlock,

the ventilation system could be started from the remote control panel located at the reactor control room, even when the pneumatic dampers installed in the ventilation ducts remained closed because of loss of instrument air supply. Under such conditions, the ventilation system would run without rendering any functional benefit. In order to avoid such a situation, the instrument air interlock system was incorporated into the control circuit of the reactor hall ventilation system in 1995. The interlock significantly improved the operational safety of the ventilation system.

4.3. Automatic trip system for the primary pump room exhaust blowers (1997)

This system provides automatic tripping of the primary pump room exhaust blowers in the event the stack monitor picks up a high radiation alarm either in its gaseous or particulate channels. The automatic trip system for the primary pump room exhaust blower, which was designed, developed and installed in 1997, helps prevent unwanted purging of the reactor hall by the primary pump room exhaust blowers in the event of an accidental particulate release or gaseous activity in the reactor hall.

4.4. Data acquisition system (1991)

A microcomputer-based data acquisition system (DAS) was designed, developed and connected to the reactor control console with a view to automatically monitor and record all operational parameters and information of the reactor. The system was developed by the Reactor Engineering and Control Division (RECD) of the Institute of Nuclear Science and Technology (INST) under close collaboration with the Reactor Operation and Maintenance Unit (ROMU). The software required for the DAS was developed by RECD under the guidance of a Japanese expert under the MEXT Scientist Exchange programme of the Japanese government. DAS was developed using TURBO Basic as the programming language and system performance was found to be quite satisfactory. The system is now being upgraded by RECD personnel. The RECD also developed a PC-based reactivity meter, which is now being used in the research reactor.

4.5. Plan for upgrade of the I&C system

The BAEC research reactor is currently operated with an analogue instrumentation and control system (I&C system) that was designed by GA almost 25 years ago. Since analogue I&C systems are becoming obsolete, BAEC decided to replace the present console with a new digital console. For this purpose, a supply contract has recently been signed with the General Atomics Electronic Systems Inc. of the USA. It is expected that by the end of 2009, installation of the new digital console and I&C system will be completed. It should be noted that the digital control system will have dedicated hardwire displays and controls. This will ensure continuation of safe operation of the reactor even when the computer becomes unavailable.

5. LESSONS LEARNED

The refurbishment work was performed to sustain safe and reliable operation of the reactor. After satisfactory installation of the cooling system refurbishment work, some of the parameters of the cooling systems were found to be improved. Tables 1 and 2 show the improvements in the cooling system and the primary pumps and motors.

TABLE 1. COOLING SYSTEM PARAMETERS (3 MW)

Parameters	Previous system	Present system
Flow rate	13 250 LPM	13 250 GPM
Core inlet temp.	40°C	35.8°F
Core outlet temp.	43°C	38.8°F
Fuel temp.	620°C	520°C

After refurbishment of the cooling system, the vibration of the primary and secondary system significantly improved. The following table shows the comparison between the present and previous vibration data.

TABLE 2. VIBRATION DATA OF PRIMARY PUMPS AND MOTORS

Location	Vertical component of vibration velocity (cm/s)	
	Previous system	Present system
Top of motor (Non drive end)	0.53	0.35
Top of motor (drive end)	0.60	0.33
Pump (suction end)	0.40	0.30
Pump (coupling end)	0.96	0.50

Table 1 shows that after installation of the plate-type heat exchanger, reactor core inlet, and reactor core outlet, fuel temperatures decreased compared to their previous values. This indicates that the safety margins have improved after the modification work. It should be mentioned that the old shell and tube-type heat exchanger was severely fouled and so it was not possible to operate the reactor at 3 MW on hot and humid days.

From Table 2, it is observed that after installation of the Y-connection and the appropriate pipe supports, the vibration level of the primary pumps and motors has been significantly minimized thus ensuring safer operation of the reactor cooling system.

In addition to these modifications, other modifications implemented in the reactor cooling system (ECCS, decay tank, etc.) have also enhanced the overall safety of the reactor.

6. CONCLUSION

The reactor has been safely operated for various peaceful applications in the field of nuclear science and technology with the exception of a few incidents as mentioned in this paper. To the extent possible, most of the modifications, repairs, and upgrades of the reactor facility were carried out with local resources. The reactor is now being used to produce ^{131}I , to conduct various R&D activities, and assists in the country's manpower training programme. Initiatives have been taken to install additional dry irradiation tubes in the core such that radioisotope production could be significantly increased. There is also a plan to develop the unused experimental facilities such as the thermal column and radial beam ports to strengthen the R&D activities associated with the reactor. It is expected that after replacement of the

present analogue control console and I&C system by a digital console, the operational safety of the BAEC research reactor will be significantly improved.

ACKNOWLEDGEMENTS

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BIBLIOGRAPHY

Safety Analysis Report of the BAEC 3 MW TRIGA MARK-II Research Reactor, BAEC, Dhaka (2006).

GENERAL ATOMICS (San Diego, CA), System Operation and Maintenance Manual, Document No. E 161 1114, January 1983 Revised (1987).

HAQUE, A., et al., Report on Commissioning of the Cooling System of the BAEC 3 MW TRIGA MK-II Research Reactor with New Decay Tank and Other Components, ROMU Report No. ROMU/TRN-02/02, Dhaka, Bangladesh (June 2003).

INTERNATIONAL ATOMIC ENERGY AGENCY, Code on the Safety of Nuclear Research Reactors: Design, Safety Series No. 35-S1, IAEA, Vienna (1992).

ZULQUARNAIN, M.A., et al., Report on Replacement of the Dry Central Thimble of the BAEC TRIGA Reactor with a Wet Central Thimble, ROMU Report No. ROMU/TRN-01/02 Revision No. 1.1, Dhaka, Bangladesh (June 2003).

ISLAM, M.S., et al., "Operation experience with the 3 MW TRIGA Mark-II research reactor of Bangladesh", Proceedings of the World TRIGA Conference, Vienna (2004).

MODERNIZATION OF THE IEA-R1 RESEARCH REACTOR TO SECURE SAFE AND SUSTAINABLE OPERATIONS FOR RADIOISOTOPE PRODUCTION

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Abstract

This paper describes the IPEN IEA-R1 research reactor modernization and upgrade programme over three decades and the valuable experience gained from the programme. The modernization programme and refurbishment of various systems and components has been a long-term and continuous activity at IPEN that started in the mid-1970s. After operating the reactor for about 15 years, several modifications were introduced during the 1970s and mid-1980s. These modifications included a stainless steel reactor pool liner, isolation chambers for access between the reactor building and pool area as well as hot and cold areas, a new ventilation and exhaust system, and a new reactor control console to cite only a few examples. This was also the period when the fuel conversion programme from HEU to LEU began. During the last fifteen years however, a much more concerted effort was made in order to refurbish the old components and systems to upgrade the reactor power to 5 MW(th). One of the reasons for this decision was to produce ^{99}Mo to alleviate part of the demand and ^{131}I to alleviate all of the national demand for these medical isotopes to minimize importation costs. During the early and mid-1990s, a large number of modifications were introduced in the reactor systems and components, particularly those related to improvement in reactor safety. In 1997, after these modifications were implemented, the reactor was authorized for operation at 5 MW(th). All the spent fuel elements were shipped to the USA in 1999 under a bilateral agreement. Several aspects of these implementations are reviewed including the important lessons learned. Some additional equipment, such as a pool water treatment and purification system and the safety and control elements was replaced during 2003–2005. A new primary heat exchanger replaced the old one in 2007.

1. INTRODUCTION

A nuclear reactor is a strong neutron source for both thermal and fast neutrons and can be efficiently used for the production of radioisotopes with numerous applications in medicine, agriculture, and industry and for other irradiation services. It could also be used for academic and applied research in nuclear and neutron-related sciences and engineering. The extent of reactor use is basically determined by its power level, which in turn determines the neutron flux. Reactor utilization also depends on the operations schedule. Low power levels and short operational cycles (only few hours of operation each day) are particularly inconvenient, both for high quality research and for the production of useful radioisotopes. On the other hand, high power levels, in conjunction with a prolonged operational cycle (several weeks of continuous operation), result in high fuel consumption and require an expensive maintenance programme. Since small research reactors are generally operated and maintained through government funding that is usually scarce, it is obvious that efforts must be made to optimize the utilization of such reactors and to determine the power level and operational regime on a cost and benefit basis. Social benefits derived from reactor utilization are usually given priority; however, the cost of such benefits must be determined in a realistic manner.

The IEA-R1 counts among the oldest research reactors in its category in the world with an operational history of more than 50 years and has an excellent safety record. The reactor is currently used for basic and applied research in the nuclear and neutron-related sciences, for the production of radioisotopes for medical and industrial applications, and for providing neutron activation analysis, real time neutron radiography, and neutron transmutation silicon doping. In order to upgrade the reactor for safe and sustainable operations to produce radioisotopes, IPEN undertook a major task of reactor system

modernization, funded by Brazilian government agencies and more recently through a technical cooperation programme funded by the International Atomic Energy Agency.

Brazil has four operational research reactors, all under the responsibility of the Brazilian National Nuclear Energy Commission (CNEN). Some details of the characteristics of these reactors are summarized in Table 1.

TABLE 1. BRAZILIAN RESEARCH REACTORS

	IEA-R1	IPR-R1	ARGONAUT	IPEN/MB-01
Criticality	September 1957	November 1960	February 1965	November 1988
Operator	IPEN-CNEN/SP	CDTN-CNEN/MG	IEN-CNEN/RJ	IPEN-CNEN/SP
Location	São Paulo	Minas Gerais	Rio de Janeiro	São Paulo
Type	Swimming Pool	Triga Mark I	Argonaut	Critical assembly
Power level	2–5 MW	250 kW	200 W	100 W
Enrichment	20%	20%	20%	4.3%
Supplier	Babcock & Wilcox	General Atomics	USDOE	Brazil

IEA-R1 is the only research reactor in Brazil operating at a substantial power level suitable for its utilization in wide areas of research such as physics, chemistry, biology, and engineering, as well as in the production of some useful radioisotopes for medical and other applications. The general features of this reactor, its past operational experience, and current utilization, are briefly reviewed. Future plans to optimize its performance in the areas of research and development, specifically its role as a radioisotope producer, are described.

2. IEA-R1 RESEARCH REACTOR

The IEA-R1 is the largest power research reactor in Brazil with a maximum power rating of 5 MW(th). The reactor was commissioned and achieved its first criticality on 16 September 1957. Although designed to operate at 5 MW, the reactor only operated at 2 MW from the early 1960s to the mid-1980s on an operational cycle of 8 hours a day, 5 days a week. It currently operates at 3.5 MW(th) with a 64-hour cycle per week. The reactor originally used 93% enriched U-Al fuel elements. Currently, it uses 20% enriched uranium (U_3O_8 -Al and U_3Si_2 -Al) fuel that is produced and fabricated at IPEN [1, 2]. The reactor is operated and maintained by the Research Reactor Centre (CRPq) at IPEN São Paulo, which is also responsible for irradiation and other services.

The IEA-R1 reactor is a multidisciplinary facility and is being used extensively for basic and applied research in nuclear and neutron-related sciences and engineering. The reactor has also been used for training, for producing radioisotopes with applications in industry and nuclear medicine, and for miscellaneous irradiation services. Several departments of IPEN routinely use the reactor for their research and development work. Many scientists and students at universities and other research institutions in Brazil also use it quite often for academic and technological research. However, the largest user of the reactor is the staff of the Research Reactor Centre.

The research programmes at CRPq include nuclear and solid-state physics, nuclear metrology, and radiochemistry, covering both fundamental questions and applied sciences. Most of the programmes have strong ties to universities, other national research institutes, and research laboratories. The CRPq takes its role very seriously as one of the major research reactor facilities in the country providing educational opportunities to students enrolled in nuclear science programmes. A large part of the research carried out at the CRPq is the active participation of many graduate students, who are working for their MSc and PhD degrees as well as some undergraduate students beginning their scientific activities.

The scientific programmes at CRPq span several multidisciplinary, fundamental, and applied research areas. Specific research programmes include nuclear structure study from the beta and gamma decay of radioactive nuclei and nuclear reactions, nuclear and neutron metrology, neutron diffraction and neutron multiple-diffraction studies for crystalline and magnetic structure determination, and perturbed $\gamma\gamma$ -angular correlation (PAC) using radioactive nuclear probes to study hyperfine interactions in solids. Additional research programmes include neutron activation analysis, both instrumental as well as radiochemical separation as applied to the fields of health, agriculture, environment, geology and industry. Research in applied physics includes neutron radiography and instrumentation.

3. NEUTRON IRRADIATION AND OTHER SERVICES

We firmly believe that no matter how important the academic research may be in its own right, it does not do any good if the results do not make their way to the outside world. The CRPq is making an enormous effort to enlarge the scope of services and applications resulting from reactor utilization so that more of the benefits of these applications can be offered to society. Some of the products and services offered by the reactor centre find their way to the petroleum industry, the aeronautical and space industry, medical clinics and hospitals, the semiconductor industry, environmental agencies, universities and research institutions. The reactor produces special radioisotopes such as ^{41}Ar and ^{82}Br for industrial processes inspections, ^{192}Ir and ^{198}Au radiation sources for use in brachytherapy, ^{153}Sm (EDTMP) for pain palliation in bone metastases, calibrated gamma sources of ^{133}Ba , ^{137}Cs , ^{57}Co , ^{60}Co , ^{241}Am , and ^{152}Eu used in clinics and hospitals practicing nuclear medicine and research laboratories. Routine non-destructive testing by real time neutron radiography, multielement trace analysis and miscellaneous neutron irradiation of samples for research applications are also offered. Regular irradiations are carried out to produce some primary radioisotopes for the Radiopharmaceuticals Centre IPEN.

Neutron irradiation of silicon single crystals for doping with phosphorus was developed at IPEN in the early 1990s. A simple device for irradiating silicon crystals with up to 12.7 cm diameter and 50.8 cm long, located in the graphite reflector, was installed in the reactor for commercial irradiation. Details about the design of the irradiation rig and its performance may be found elsewhere [3].

4. PRODUCTION OF RADIOISOTOPES

Until 1980, all $^{99\text{m}}\text{Tc}$ generators used in the country were imported. Due to the rapidly increasing demand and high cost of importation, the Radiopharmaceuticals Centre IPEN started to produce its own $^{99\text{m}}\text{Tc}$ generator kits with an automatic elution system using fission ^{99}Mo purchased from Canada. Today, ^{99}Mo - $^{99\text{m}}\text{Tc}$ generators in several different radiopharmaceutical forms, represent more than 80% of all the radioisotopes distributed to hospitals and clinics in the country. In 2007, more than 610.5 GBq Ci $^{99\text{m}}\text{Tc}$ generators were

produced with individual kit activity ranging from 9.25 GBq to 74 GBq and distributed to about 300 hospitals and clinics all over the country, benefiting more than 3 million patients. In addition to ^{99m}Tc generators, the Radiopharmaceuticals Centre also makes and distributes radioactive preparations for medical use based on ^{131}I (48.1 TBq/yr), as well as smaller quantities of ^{51}Cr , ^{32}P , ^{177}Lu , and ^{90}Y that form the imported radioisotopes. The shorter lived radioisotope ^{153}Sm (1.48 TBq/year) is regularly produced in the IEA-R1 reactor and distributed in the form of $^{153}\text{Sm-EDTMP}$ for pain palliation in bone metastases. These reactor-produced radioisotopes, in addition to some of the cyclotron-produced radioisotopes, such as ^{18}F , ^{67}Ga , ^{111}In , ^{201}Tl , and radiopharmaceuticals used in nuclear medicine, represents annual revenue from sales on the order of US \$25 million for IPEN (2007).

The importation costs for the reactor-produced primary radioisotopes are on the order of US \$5-6 million. A recent survey showed that the demand for ^{99m}Tc generators and ^{131}I preparations was steadily increasing at the rate of 8% and 20% per year, respectively, and was expected to continue to increase in the years to come. In order to meet the continuously increasing demand for these radioisotopes in Brazil, the IPEN will face increasing importation costs easily reaching US \$7-8 million in just a couple of years. This is quite cumbersome for an institution like the IPEN whose main funding comes from a federally-approved budget. Importation cost factors aside, increasing reliance on only one or two foreign suppliers of vital radionuclides, such as ^{99}Mo is certainly a strategic disadvantage for the country.

Considering these factors, as well as the possibility of producing radioisotopes, such as ^{99}Mo , ^{192}Ir , ^{131}I , and ^{125}I among others in the IEA-R1 reactor, an important decision was made several years ago to upgrade the reactor power to 5 MW(th) and to gradually increase the operational cycle to 120 h continuous per week. Initial plans to produce ^{99}Mo from fission were abandoned primarily due to the lack of necessary funds required for the processing plant but also because of the technical problems associated with the complex radiochemical processing and the management and storage of the highly radioactive waste generated during processing. It was decided to produce ^{99}Mo by the $^{98}\text{Mo}(n,\gamma)^{99}\text{Mo}$ reaction with a natural or enriched molybdenum target and to prepare the ^{99m}Tc generator using the gel process. In order to produce high specific activity sources of ^{99}Mo , resulting in ^{99m}Tc generators in the range of 9.25–37 GBq from the IEA-R1, it was necessary to raise the reactor power to 5 MW(th) and to adopt an operational cycle of at least 120 h continuous per week. Efforts were also made to modernize other related infrastructure. Four main areas received particular attention: (i) optimization of reactor systems, structures and components; (ii) optimization of reactor fuel element production; (iii) optimization of radiochemical processing facilities for radiopharmaceutical production; and (iv) an effective programme for spent fuel management. The IPEN assigned definite priorities for these projects at the institutional level.

5. REACTOR UPGRADE AND MODERNIZATION

The IEA-R1 is one of the oldest reactors of its kind in the world and has been operating for over 50 years and had an excellent safety record. However, during the last fifteen years, many changes in the reactor system and components have been made in an effort to upgrade the reactor power and extend the operational cycle. Special attention has been given to components and systems related to operational safety. The main objective has been to extend the lifetime of the reactor for several more years and secure its safe and continuous operation.

The reactor modernization programme and refurbishment of its systems and components has been a long-term and continuous activity at the IPEN that started in the

mid-1970s. After operating the reactor for about 15 years, several modifications were introduced in the basic structure of the reactor in the 1970s and mid-1980s. These modifications included:

- installation of a new ventilation and exhaust system;
- installation of a second primary heat exchanger and a cooling tower;
- renovation of the electrical system with installation of a no-break system using diesel power generators;
- substitution of reactor control console;
- substitution of old CB_4 type control rods with Cd, In, Ag (fourchette type);
- construction of isolation chambers for access to the reactor building and pool area and the separation of hot and cold areas;
- replacement of ceramic tile lining of the reactor pool with stainless steel;
- implementation of fuel conversion programme from HEU to LEU.

Some of the important improvements made in the reactor systems and components were completed in the last fifteen years and were motivated by the decision to upgrade the reactor power and to extend the operational cycle to produce important primary radioisotopes. The recent refurbishment and modernization activity began in the early 1990s and include:

- modification of the reactor core from 6×5 to 5×5 using LEU fuel elements;
- installation of a central beryllium irradiating element;
- replacement of 10 graphite reflectors with beryllium reflectors;
- installation of 4 isolation valves in the primary cooling system;
- repairs in the cooling tower and pipelines;
- installation of a new ventilation and air conditioning system;
- improvement in the control instrumentation;
- replacement of some of the old radiation monitoring system;
- installation of emergency spray cooling system for the core;
- improvement of the fire-detection and fire-fighting equipment with installation of smoke detectors, sprinklers and new fire hydrants.

With these modifications introduced in the reactor, a new safety analysis report was prepared and submitted to the regulatory body. An authorization from the regulatory body was obtained in September 1997 for commissioning of the reactor at 5 MW(th). The reactor operated at 5 MW(th) for six months. However, other projects like the optimization of fuel element production, the chemical processing facilities for ^{99m}Tc using the gel process, and the spent fuel storage facility had not been completed implemented and the reactor power was reduced back to 2 MW(th). In 1999, all the spent fuel elements stored in the reactor pool since its first criticality (a total of 127 elements) were transferred to the USA under a bilateral agreement (DOE-IPEN/CNEN) [4]. In 2007, another batch of 33 spent fuel elements were returned to the USA under a similar agreement.

The reactor power is gradually being increased. It has operated at 3.5 MW(th) on a continuous 64 h/week cycle since 2000. The IEA-R1 reactor has been ISO-9001:2000 certified since 2002 in Reactor Operation and Irradiation Services. From 2002–2007, substantial progress was made in implementing the reactor fuel fabrication programme. The Fuel Fabrication Centre IPEN acquired the knowledge and capacity to produce 15–16 fuel elements of the type U_3Si_2 (3.0 g/cm^3) per year. This facility has recently moved to another site within IPEN with better equipment and infrastructure and is awaiting an operating licence

from the regulatory board. The licensing process has caused considerable delay in the fabrication of fresh fuel elements. As a consequence, plans to upgrade the reactor power to 5 MW(th) was postponed until the end of 2008 when the fuel element supply is likely to normalize. A chemical method for producing ^{99m}Tc generator by the gel process has already been developed and the infrastructure, such as the hot cell facility and automation of the chemical processing has been completed. At present, the reactor pool has storage space for more than 150 spent fuel elements. The available pool storage space should be sufficient for about 8–10 years of reactor operation at 5 MW for 120 h/week. The fuel consumption is estimated to be 12–15 elements per year in this operation regime. A new project for spent fuel management and storage was initiated at IPEN to investigate the possibility of an alternate dry storage space. A project for a shielded cask for transportation and storage of the spent fuel in semi-dry conditions has been completed and a prototype will be constructed soon.

The ageing management and refurbishment programme for the IEA-R1 reactor components and systems is an ongoing activity of the CRPq. For example, the reactor pool water treatment and purification system was replaced in 2004. The older control and safety elements of the reactor began to show signs of ageing and were replaced in 2004 with identical elements (fork-type Ag-In-Cd) fabricated at the IPEN. A new primary heat exchanger was acquired and installed at the beginning of 2007. After commissioning tests, regular use of the heat exchanger was authorized by the regulatory board in December 2007. A completely new pneumatic rabbit facility was constructed and installed in one of the reactor grid positions for short in-core irradiations of samples for neutron activation analysis. A separate pneumatic transfer line (about 150 m long) was installed for the quick transport of irradiated capsules from the reactor to the radioisotope processing area in the pharmaceutical centre. A small hot cell is being constructed poolside to handle capsules containing highly radioactive samples after irradiation in the reactor. It is anticipated that these and other measures taken in past years to refurbish the reactor systems and components will assure the safe and sustainable operation of the IEA-R1 reactor for several more years and permit the production of important primary radioisotopes such as ^{99}Mo , ^{131}I , and ^{153}Sm in sufficient quantities. The higher neutron flux and extended operational cycle will also permit the production of other radioisotopes, such as ^{125}I , ^{177}Lu , ^{188}Re , and ^{192}Ir .

The CRPq received support from IAEA during 2005–2006 through a Technical Cooperation (TC) project BRA/04/056 — Modernization of the IEA-R1 research reactor to secure safe and sustainable operation for radioisotope production. The project permitted several training programmes for the reactor operations and maintenance personnel as well as improving the technical infrastructure of the reactor. Some of the goals achieved through the TC project are:

- HPGe gamma spectrometer system for radiometric analysis of water and air samples;
- several radiation monitoring and detection equipment;
- a neutron flux mapping facility using self-powered neutron detectors (SPNDs);
- an improved computational facility for neutronic calculations;
- a highly radioactive sample handling facility;
- installation of a continuous vibration monitoring system for rotating machinery;
- training of personnel engaged in electrical and mechanical maintenance, water chemistry, irradiation services and quality management system.

The rotating machinery in the IEA-R1 reactor system is primarily the water circulating pumps. As part of the reactor upgrade plan, a continuous vibration monitoring system has been installed. This will provide accelerometer data to a central processing unit

that will monitor the changes in the vibration levels of the pump-motor system. Defects such as imbalance, misalignment, looseness, and bearing faults can be detected before a catastrophic failure occurs. Thus, incipient fault detection and diagnosis of rotating machinery is an important feature of this upgrade.

6. LESSONS LEARNED

Our experience has shown that modernization and refurbishment programmes for small research reactors must be a continuous activity where small steps must be taken to improve the performance of the reactor with moderate budgets and shorter shutdown periods rather than very extensive refurbishment programmes that require large sums and long shutdown times. However, in both cases, very well-planned and skilful management of these activities are required. In this context, development of a strategic plan for the effective utilization of the research reactor is an essential step. The strategic plan should be presented to the major stakeholders to convince them that the modernization and refurbishment programmes for the reactor are a good investment. Continuous efforts should be made to maintain the reactor utilization index as high as possible. The production of radioisotopes for medical and industrial application is essential, as it provides immediate and visible social benefits, but reactor utilization in academic and applied research and education are equally important and cannot be ignored, particularly in developing countries. Our experience has also shown that the implantation of a good integrated management system including quality, safety culture, and environmental consciousness in the business of reactor operation, maintenance and irradiation services can help a great deal in raising self esteem and a sense of collective responsibility of the reactor staff. This ultimately reduces the need for unplanned maintenance and refurbishment in the reactor systems and components. Planning, efficient management, and continuous improvement are the key words in a quality management system.

7. CONCLUSION

It is anticipated that the past and current modernization and refurbishment programmes will assure the safe and sustainable operation of the IEA-R1 reactor for several more years and help produce important primary radioisotopes such as ^{99}Mo , ^{131}I , ^{153}Sm , and ^{192}Ir . Important economic benefits in foreign exchange are expected to result from the reduced need for radioactive imports. Estimated figures are: US \$400 000–600 000 for ^{99}Mo , US \$600 000–700 000 for ^{131}I , and US \$50 000–80 000 for ^{192}Ir . Reactor operations under the new conditions will also permit the production of ^{125}I seeds (which are currently imported) used for the treatment of prostate cancer.

In order to achieve the goals of modernization and reactor ageing management, the IPEN has implemented a number of tasks under a technical cooperation programme funded by the International Atomic Energy Agency and Brazilian government agencies. Currently, all aspects of dealing with fuel element fabrication, fuel transportation, radioisotope processing, and spent fuel storage are handled by IPEN at the site. Spent fuel assemblies stored in the reactor pool are visually inspected on a routine basis using underwater cameras. A leak analysis is performed if there is an indication of fission product release in the pool water. A continuous vibration monitoring system has been installed for incipient fault detection and diagnosis of rotating machinery such as pump-motor system.

As a consequence of an increased neutron flux (maximum thermal neutron flux of $2 \times 10^{14} \text{ n}\cdot\text{cm}^{-2}\cdot\text{s}^{-1}$ at 5 MW(th) power) and an extended operation period, other applications

and services such as silicon doping with phosphorus by neutron irradiation, neutron radiography, and neutron activation analysis will have better performance. The improved operational regime of the reactor will stimulate renewed interests in other applications, which are currently in experimental stages, such as boron neutron capture therapy (BNCT) and coloration of topaz. Neutron beam research will benefit due to the availability of more intense neutron beams. The existing neutron diffractometer has been modernized by adding a bank of 9 position sensitive detectors, a rotating oscillating collimator, and an elastically bent silicon single crystal monochromator. A viability study was recently carried out for the possibility of installing a small angle neutron scattering (SANS) facility at the reactor. This activity was supported by the IAEA through a research contract. It should be emphasized that academic research and postgraduate teaching at the Reactor Centre at the IPEN are very important programmes in the effective utilization of the reactor. Research scientists, students, and professors from universities and other research institutions and their students have free access to the research facilities at the Reactor Centre.

REFERENCES

- [1] MAIORINO, J.R., "The Brazilian research reactor IEA-R1", IAEA Advisory Group Meeting on Optimization of Research Reactor Utilization for Production of Radioisotopes, JAERI, Tokai-Mura, Japan (1995).
- [2] SAXENA, R.N., FULFARO, R., "Brazilian research reactor IEA-R1: enhancement in its utilization in basic and applied research and radioisotopes production", IAEA International Conference on Research Reactor Utilization, Safety, Decommissioning, Fuel and Waste Management, Santiago, Chile (2003).
- [3] CARBONARI, A.W., et al., An irradiation rig for neutron transmutation doping of silicon in the IEA-R1 research reactor, Nucl. Instr. Meth. in Physics Research B, **83** (1993) 157.
- [4] FRAJNDLICH, R., MAOIRINO, J.R., "The operational and logistic experience on transportation of Brazilian spent fuel to USA", presented in Technical Committee Meeting on National Experience on Return of Research Reactor Spent Fuel to the Country of Origin, IAEA, Vienna (2006).

MODERNIZATION OF THE VR-1 TRAINING REACTOR

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Abstract

The VR-1 training reactor has been in operation since 1990. The reactor is successfully used for training students in Czech universities and for training experts for the Czech nuclear programme. The VR-1 reactor is also used for basic research even though it operates at a low power. During the last 7 years, a large number of improvements in operation conditions were implemented. From a safety-related point of view, the most important are the upgrades of the instrumentation and control (I&C) system and the radiation monitoring system. Also, the core conversion from the HEU to LEU fuel took place. The upgrade of the I&C system began in 2001 and was completed at the end of 2007. Upgrade of the radiation monitoring system was carried out during 2004.

1. INTRODUCTION

The operation of the VR-1 training reactor was started in 1990 by the Department of Nuclear Reactors of the Faculty of Nuclear Sciences and Physical Engineering, Czech Technical University in Prague. The VR-1 reactor is a pool-type light-water reactor utilizing 20% enriched uranium. Its thermal power is up to 5 kW. The neutron moderator is light demineralised water, which is also used as a reflector, biological shielding, and coolant. Heat is removed from the core by natural convection. The pool design of the reactor facilitates access to the core, setting and removing of various experimental samples and detectors, and easy and safe handling of fuel assemblies. The control rods have an integral performance and their composition is identical. They differ only in function (safety, reactivity compensation or control) according to their connection to the instrumentation and control system. The absorber material is cadmium. A neutron source is used to start up the reactor. The source ensures a sufficient signal level at the output of the power measuring channels from the deepest subcriticality and thus guarantees a reliable check of the power during reactor startup. The reactor is equipped with several experimental devices; e.g. horizontal, radial and tangential channels, which are used to provide a neutron beam.

The VR-1 reactor is particularly utilized for the training of university students and nuclear power plant staff. The training at the VR-1 reactor is directed to reactor and neutron physics, dosimetry, nuclear safety, and control of nuclear installations. Students, not only from technical universities but also from universities of natural science, participate in reactor training. Scientific research is conducted within the reactor parameters and requirements of the so-called clean reactor core (i.e. free from any major effect of the fission products). Research at the VR-1 reactor is aimed at the preparation and testing of new educational methodologies, investigation of reactor lattice parameters, reactor dynamics studies, research in the field of control equipment, and neutron detector calibration, etc.

2. UPGRADE OF THE INSTRUMENTATION AND CONTROL SYSTEM

The original reactor I&C system of the VR-1 training reactor was developed in the mid-1980s. Even if the original I&C system completely met the demands that were put on it, its technical design was obsolete. There were also difficulties with maintenance due to a lack

of spare parts. Furthermore, during development and manufacturing of the original I&C system, some new internationally imposed demands on quality and qualification (e.g. the IAEA, IEC, and IEEE recommendations and standards) were not considered. Therefore, it was decided to upgrade the existing I&C system with the goal of applying the latest available recommendations and standards.

The principal upgrade of the I&C system began in 2001 [1]. Because of the frequent utilization of the VR-1 training reactor during the academic terms, it was decided to gradually carry out the upgrade, i.e. in stages during holidays so as not to affect training at the reactor. The first stage was the human-machine interface and the control room upgrade in 2001 and the control rod drives and safety circuits in 2002. Upgrades of the control system continued in 2003 and the independent power protection system was upgraded in 2005. The last stage was the upgrade of the operational power measuring system and the installation of the complete I&C in a new I&C room with up-to-date cabinets and with much better access to the systems for inspections and maintenance than in the original I&C location. The entire project was finished at the end of 2007.

2.1. Description of the system structure

A block diagram of the I&C system is shown in Figure 1. The system structure has to meet the requirements of the Czech State Office for Nuclear Safety.

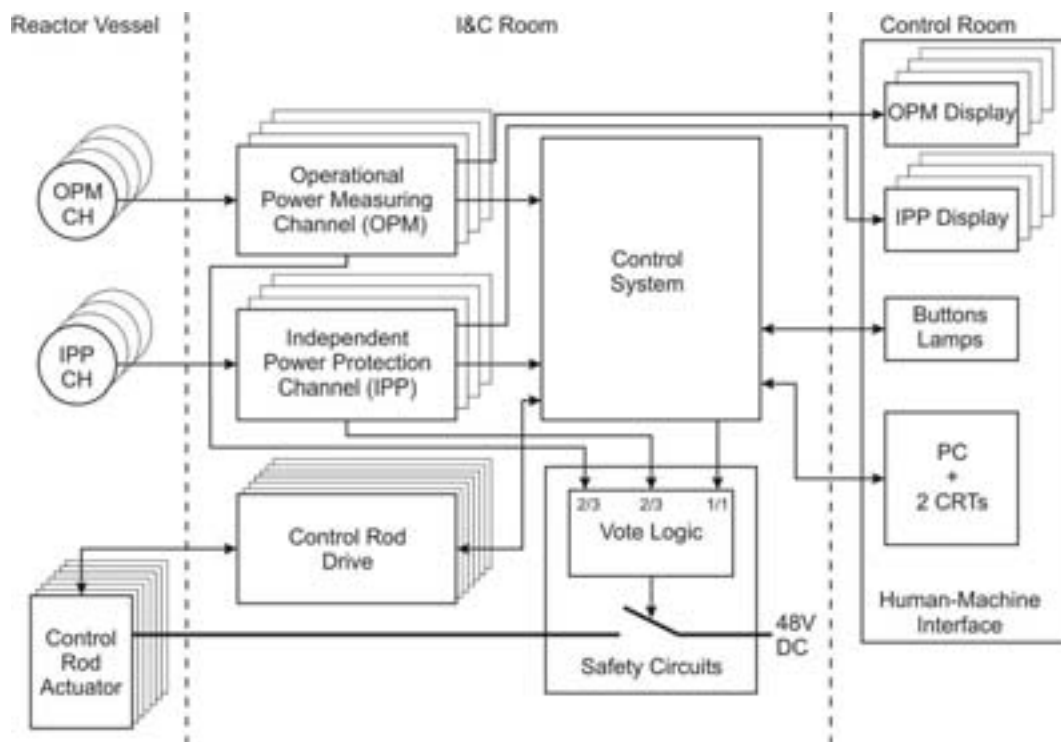


FIG. 1. Block diagram of the upgraded reactor I&C system.

First, the safety part of the system is described. This is the most important part of the I&C system for nuclear safety. The four operational power measuring channels (OPM) receive signals from wide range fission chambers (OPMCH), evaluate them, calculate the reactor power and the power rate, and send the values to the control system and to adjacent individual displays on the operator's desk of the human-machine interface (HMI) in the control room. Four channels equipped with boron chambers (IPPCH) work as an independent power protection system (IPP). They also evaluate the power and the power rate, send data to

the control system and to their displays, and initiate a safety action if the safety limits are exceeded.

The control system receives data from the OPM and IPP channels, checks received values with one another and with the safety limits. The control system calculates the average values of the reactor power and the power rate; next, it evaluates the deviation between the real power and the demanded power value set by the operator. The control system sends data to the HMI and receives commands from there. If the commands are permitted, the control system carries them out. The control system also serves as an automatic power regulator system and controls the movement of the control rods to achieve the required reactor power. The control rod movement is actuated by control rod drives.

The HMI enables communication between the I&C system and the operator. It consists of a computer with CRT displays and indicators to show the reactor operational status, as well as a keyboard and buttons to control the reactor. The HMI also stores data about the reactor operation history.

A vote logic receives the safety signals. The vote logic evaluates the inputs from the OPM channels in 2/3 logic, from the IPP channels independently in 2/3 logic, and the safety signal from the control system is evaluated in 1/1 logic. If the conditions for the safety action request are met in at least one group (OPM, IPP or control system), the power supply (48 V DC) to the control rods will be interrupted by the safety circuits, the rods fall down and stop the chain reaction (reactor SCRAM). Because of the low power of the VR-1 reactor, it is not necessary to remove residual heat from the active core.

2.2. New human-machine interface (HMI)

The human-machine interface replacement as the first stage of the control and safety systems upgrade was carried out in the summer 2001. The aim of the upgrade was to improve ergonomic and aesthetic properties of the operator's desk (see Fig. 2) and the control room, to enhance the operator's comfort and thus to improve the conditions for utilization of the reactor and nuclear safety [2].



FIG. 2. New operator's desk of the VR-1 reactor.

The software for the new HMI system was prepared utilizing the InTouch system development tool. This development tool, produced by WonderWare Company and working in the Microsoft Windows 2000 environment, is intended for data acquisition and visualization. The software recognizes commands from the keyboard and sends them to the control and safety system of the reactor, receives messages and data from the control system, and displays them on the monitors. The software is also responsible for the graphic presentation of the reactor status. The user interface of the HMI can be operated in Czech or English.

2.3. Control rod drives and safety circuits upgrade

The control rod drives, motors, and safety circuits were upgraded in 2002. The rod motors were replaced with the new motors that have the required properties and dimensions. Necessary mechanical changes on the control rod mechanism, induced by the utilization of the new motor, were done by the Škoda Company. High quality connectors were utilized for the connection of the cables to the motors. PLC Simatic S7-200, equipped with a proper power electronic board, serves as a drive of the motors. Appropriate software to control the PLC has been developed. The PLCs communicate with the control system via RS485 (Profibus) lines. New safety circuits utilize high quality relays with forced contacts to guarantee high reliability of their operation. The safety circuits have been installed in a 48.3 cm rack for easy installation in new cabinets of the new control and safety system.

2.4. New control system

The control system replacement was carried out in 2003. The control system is based on the industrial personal computer of the Nexcom Company mounted in a 48.3 cm rack with a redundant power supply system. The operating system of the PC is Microsoft Windows XP with the real time support RTX of the VentureCom Company. The computer is equipped with 8 fiber optic lines for communication with the operational power measuring and independent power protection channels (see Section 2.5), with the RS485 (Profibus) line for communication, with the Simatic control rod and I/O PLCs, and with the Ethernet line for data transfer to the human-machine interface. Because of the importance of the control system to nuclear safety, high quality hardware and software was required. Intensive verification and validation was carried out during the manufacturing and after the delivery.

2.5. Power protection system and operational power measuring system upgrades

The independent power protection (IPP) system upgrade was started in 2004 and was finished in 2005. The analogue section of the IPP channel processes the signal from the boron neutron chamber, amplifies it, and provides proper discrimination of neutrons. Then, it counts pulses from the neutron chamber, evaluates the reactor power and the power rate. Next, it compares gained data with the safety limits and sends the safety signal. The IPP channel also communicates with the reactor control system via fiber optic lines, controls the individual display at the operator's desk, and provides the channel testing. The IPP channel consists of 5 microcomputer units. The reason for the utilization of more microcomputers was to divide single functions to separate microcomputers to guarantee easier structure of the system hardware and, in particular, of system software. The communication among individual microcomputers is provided via buffer in a Field-Programmable Gate Array. The IPP channel software was developed in accordance with nuclear standards. The software design was coded in C language under NRC (Nuclear Regulatory Commission) restrictions. The reputable μ Vision 2 software development system of Keil Software Company was utilized.

Configuration management, verification, and validation accompanied the development process.

The beginning of the operational power measuring (OPM) system upgrade started in 2006 and was finalized in 2007. The OPM channel analogue section processes the signal from the fission neutron chamber, amplifies and evaluates it, and provides proper discrimination of neutrons. Depending on the reactor power, the signal from the chamber is evaluated either in pulse or in current (DC current and Campbell) mode. The OPM channel receives digitized signals from the analogue section and evaluates the reactor power and power rate. Next, it compares gained data with safety limits and sends the safety signal. It also communicates with the reactor control system via fiber optic lines, controls the individual display on the operator's desk, and provides the channel testing. The digital section consists of a high quality industrial PC. The OPM channel software has to fulfil quality requirements for the safety (protection) systems of nuclear facilities — quality assurance, configuration management, verification and validation activities in accordance with respective standards and guidelines. For a computer operating system, the proven system Phar Lap has been selected. The software has been coded in the C language with respect to NRC documents. Corresponding software life cycle documentation has been produced together with the software.

2.6. Installation of new I&C system

The previous I&C system was originally installed in the reactor control room. Space in the control room was very limited and only few students could be present in the control room during training. Also, accessibility to the I&C for inspections and maintenance was restrictive. A new room for the reactor I&C near the reactor hall was found with enough room for the complete I&C with sufficient room for inspections and maintenance. The I&C components have been arranged into 48.3 racks of the Rittal company as seen in Figure 3. These racks can be opened at the front and rear sides and access to the I&C electronics is very easy.

3. UPGRADE OF THE RADIATION MONITORING SYSTEM

The radiation monitoring system (RMS) at the VR-1 reactor has been derived from MS2000 monitoring system made by BQM in the Czech Republic. The control part contains a notebook computer and a 48.3 cm LCD touch screen. The touch screen is used to provide current information about the radiation state of the VR-1 reactor. The system is specially designed for BQM radiation detectors: GMS3, GMS3-V, VRS2. The probes are connected to the system by direct serial ports. The RMS system also includes other devices, such as a JKA300n neutron monitor and a KOPR06 Alfa Beta Continuous Air Monitor.

The RMS VR-1 system contains 6 GMS3 gamma probes, 3 GMS3-V gamma probes, 1 VRS2 gamma probe, 2 JKA300n neutron monitors, and 2 KOPR06 air monitors. The upgrade of the RMS system was made in 2004.



FIG. 3. New I&C room.

4. FUEL CONVERSION

Within the scope of the RERTR programme, CTU has successfully converted the VR-1 reactor from the Russian made IRT-3M highly-enriched fuel containing 36% ^{235}U to the Russian IRT-4M low-enriched fuel containing 19.7% ^{235}U in 2005. On 18 October 2005 at 14:10 UTC, the VR-1 reactor went critical with LEU fuel for the first time.

5. CONCLUSION

This paper described the comprehensive modernization of the VR-1 research reactor. From a safety point of view, the most important upgrade has been the renewal of the instrumentation and control system. The upgrade began in 2001 and was completed in autumn 2007. The I&C system was installed in the new I&C room and is more accessible and easier to maintain than at its old location. After the installation, the I&C system was carefully tested. Compared to the old system, the new I&C system provides better testability and maintainability and uses up-to-date technology in both the hardware and the software. The neutron chamber signal processing with the Campbell technique for the gamma discrimination provides more accurate reactor power measurement. Also, the quality requirements in both hardware and software were fulfilled during the I&C upgrade according to the respective international guidelines and standards. The most important experience during the I&C upgrade has been the significance of the complete, correct and unambiguous requirements and the thorough testing of safety and operational features of the upgraded control and safety system within real reactor operation. The upgrade of the radiation monitoring system was carried out during 2004.

REFERENCES

- [1] KROPÍK, M., MATĚJKA, K., JUŘÍČKOVÁ, M., “New I&C at VR-1 training reactor”, 15th International Conference on Nuclear Engineering, Nagoya, Japan (2007).
- [2] SKLENKA, L., et al., “Improving the safe operation and new open policy at VR-1 reactor” (IAEA-CN-100/58), Proc. International Conference on Research Reactor Utilization, Safety, Decommissioning, Fuel and Waste Management, Santiago, Chile, November 2003, IAEA, Vienna (2004).

BIBLIOGRAPHY

SKODA, R., et al., “Full replacement of Russian-origin HEU fuel by LEU fuel in the research reactor VR-1”, 46th INMM Annual Meeting, Phoenix, USA (2005).

RATAJ, J., “Criticality calculations and transient analyses for the VR-1 reactor with IRT-4M LEU Fuel Assemblies, RERTR International Meeting, Boston, USA (2005).

SKLENKA, L., MATĚJKA, K., “The first critical experiment with a LEU Russian fuel IRT-4M at the training reactor VR-1”, RERTR International Meeting, Boston, USA (2005).

SKODA, R., “Sparrow flies lower”, RERTR International Meeting, Boston, USA (2005).

MATĚJKA, K., et al., Training Reactor VR-1, Czech Technical University, Prague (2003).

KOLROS, A., STARÝ, R., “Innovation of dosimetric system on reactor VR-1 Sparrow”, Proceedings of Workshop 2003, CTU Publishing House, Prague, **B** (2003) 848.

MODERNIZATION OF THE HIGH FLUX REACTOR (HFR)

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Abstract

The Institut Laue-Langevin's High Flux Reactor is an extremely high performance neutron source. The source was commissioned in 1971, producing a neutron flux of the order of $1.5 \times 10^{15} \text{ n}\cdot\text{cm}^{-2}\cdot\text{s}^{-1}$. It is expected to operate until 2030. It has recently undergone major refurbishment and improvement within a programme of continuous upgrade that aims to guarantee reliability and service. The recent overhaul reinforced the facility's performance under Safe Shutdown Earthquake conditions, given that the horizontal acceleration to be taken into account is now on the order of 0.6 g at 3 Hz. Two major programmes are currently in progress. The Millennium Programme was launched in 2001 to improve instrument performance and detector efficiency; it has already increased neutron signals by a factor of 15. The Key Reactor Components programme was launched in 2005 and aims to ensure the renewal and upgrade of essential reactor equipment thus ensuring safe and trouble-free operations to the 2030 horizon.

1. BRIEF HISTORY OF THE FACILITY

1.1. Origins and membership

The Institut Laue-Langevin (ILL) was founded in 1967 under the impetus of the Franco-German reconciliation. It is operated by its founder countries, France and Germany, together with the United Kingdom, which joined the partnership in 1973. Its location is Grenoble, France (see Fig. 1).



FIG. 1. Joint ILL-ESRF site.

The ILL is a private company governed by French civil law ('société civile'). Between 1987 and the present day, it has gradually extended the scope of its activities (see Fig. 2) and has gained an additional 9 scientific members: Spain, Switzerland, Austria, Italy, the Czech Republic, Sweden, Hungary, Belgium and Poland.



FIG. 2. Utilization of the ILL in 2007.

1.2. Operational history

ILL's High Flux Reactor (HFR) started up in 1971 and became available for scientific experiments in 1972, producing a thermal power of 58 MW and a neutron flux on the order of $1.5 \times 10^{15} \text{ n}\cdot\text{cm}^{-2}\cdot\text{s}^{-1}$ in the reflector vessel.

Since going critical for the first time, the reactor has operated for around 150 cycles of 50 days each. Its history has been marked by the following events:

- 1985: construction of a new vertical cold source enabling the production of ultra-cold neutrons;
- 1987: construction of a second cold source to supply neutrons to the second neutron guide hall;
- 1990–1994: replacement of the reactor block;
- Since 2001: scientific instrument upgrade programme (Millennium Programme);
- 2003: replacement of the hot source;
- 2003–2007: seismic reinforcement of the facility while maintaining 3 fifty-day reactor cycles per year;
- Since 2005: Key Reactor Components programme for the renewal and upgrade of essential equipment to enable the facility to operate safely until 2030.

1.3. Principal areas of utilisation

ILL's HFR is entirely devoted to fundamental research in fields (see Fig. 3) as varying as:

- solid-state physics;

- magnetolectronics;
- pharmacology and biotechnology;
- polymer plastics;
- engineering sciences;
- food chemistry;
- cosmology;
- process engineering;
- geology and the environment.

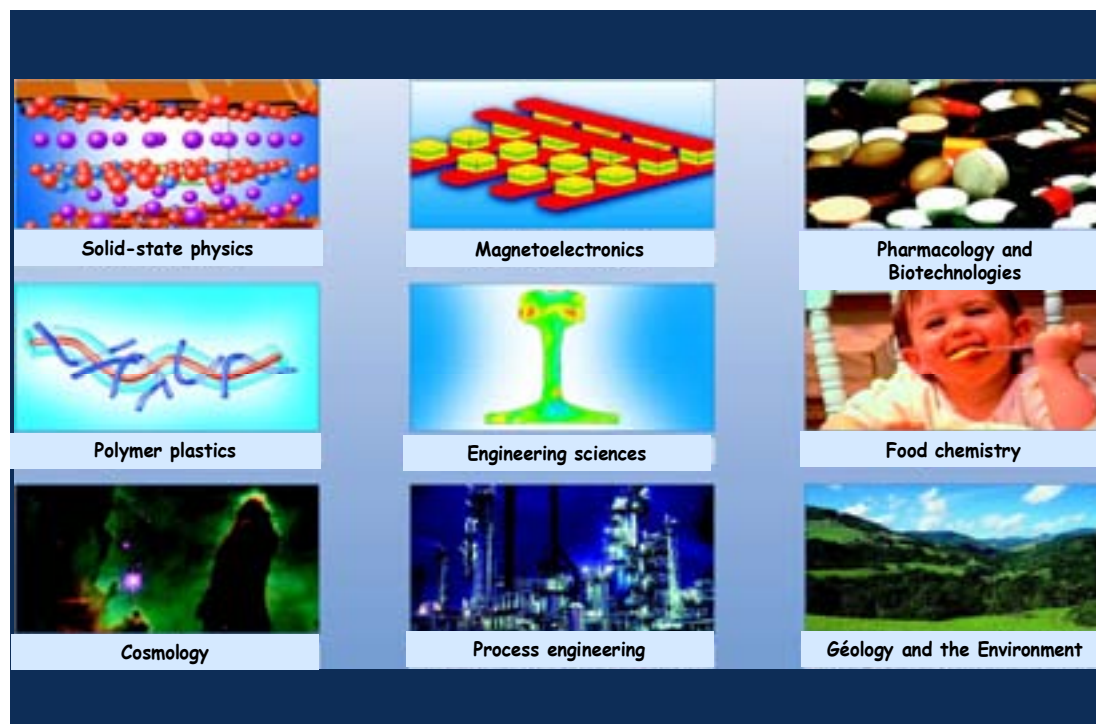


FIG. 3. Fields of application.

Generally, the study of condensed matter by neutron scattering:

- provides information about the structure and dynamics of matter (metals, alloys, ceramics, polymers, soft matter, proteins, liquids, glasses, etc.);
- enhances our understanding of physical/chemical phenomena (magnetism, superconductivity, etc.).

Thanks to the incomparable performance levels of the HFR and its scientific instruments, highly specific research can be conducted, for example:

- determining magnetic structures under very high pressures (above 7 GPa);
- studying neutron diffraction of very small samples (smaller than 0.01 mm^3);
- physically measuring atoms containing a large excess of neutrons;
- studying the process of nuclear transmutation, etc.

Since its launch in 2001, the Millennium Programme for improving instrument performance levels and increasing detector efficiency has made it possible to increase neutron signals by a factor of 15 (see also Fig. 4).

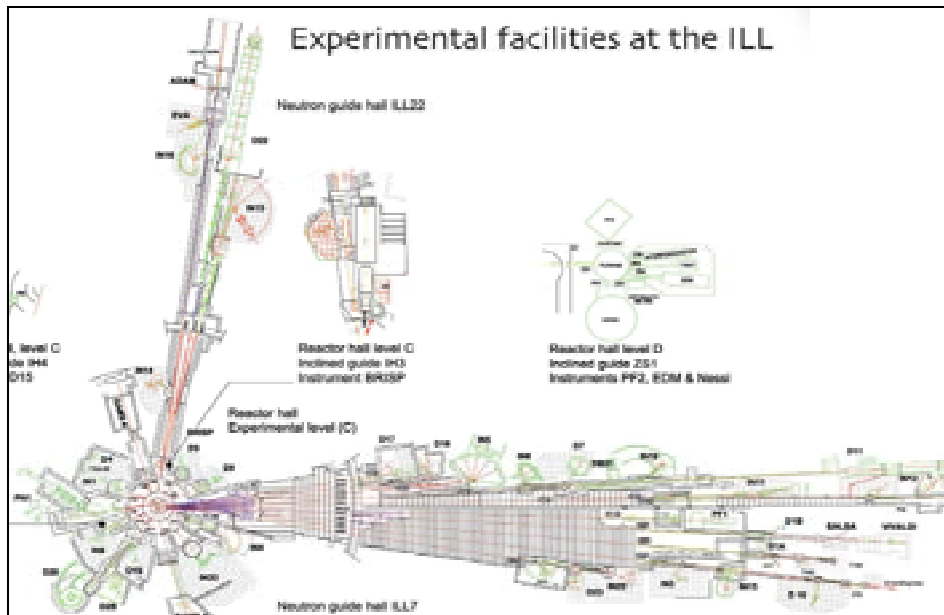


FIG. 4. The ILL's mission: Set-up of instruments.

2. SCOPE OF THE MODERNIZATION AND REFURBISHMENT WORK

2.1. Chief motives and justification

A number of modernization programmes have been completed or are currently being carried out at the HFR. The decision to carry out these major activities was primarily prompted by 3 different factors:

- The need to comply with safety standards and any future changes to these standards (new legislation and statutory guidelines), but also the need to reinforce the containment barriers. This is why the reactor block was replaced and its design modified so that the components most exposed to neutron radiation (in particular the antiturbulence grid) could be replaced. By replacing the aluminium beam tubes with beam tubes made of zircaloy, it became possible to reduce the frequency of maintenance operations on irradiated equipment and, as a result, staff exposure levels, etc.

The seismic reinforcement work (Refit Programme) has contributed to meeting the objective of complying with safety standards that had been changed. The update of the core neutronic studies using modern calculation codes and the planned reinforcement of the second and third barriers in the event of an accident also fall within this context.

- The request for increasing the performance levels:

This request has prompted the following work: the construction of a second cold source, the modification of the first cold source to allow the production of ultra-cold neutrons, and the scientific instrument upgrade programme (Millennium Programme).

- Equipment obsolescence/ageing:

In the long term, the management of equipment obsolescence is particularly critical for items such as instrumentation and control systems and electrical equipment, but also mechanical equipment (helium compressors, vacuum pumps, etc.).

These items are particularly relevant for facilities that plan to operate over a relatively long time scale. This is the case of the HFR, which has been able to maintain the highest level of safety and technical standards and must assure safe operation until around 2030.

2.2. Systems, structures and components impacted

The HFR equipment that has been, is, or will be impacted by these modernization programmes varies extremely:

- reactor block and its equipment (control rods, valves, etc.);
- scientific equipment and instruments;
- high and low-voltage electrical panels;
- reactor monitoring equipment;
- instrumentation and control systems for the reactor and scientific equipment;
- civil engineering structures (seismic reinforcement, creation of new experimental areas, etc.);
- primary circuit;
- gaseous effluents circuits, etc.

2.3. Safety/licensing-related analyses and documentation

At HFR, a major constraint exists: If the facility is shut down for more than 2 years, a decree relating to the creation of a new facility must be issued. In particular, this triggers a public benefit inquiry.

Further modifications concern items of equipment or functions important for the safety of the facility. They require formal authorization by the safety authorities and therefore, the submission of supporting documentation and analyses by the operator of the HFR. Some work also results from commitments made by the operator and others from recommendations issued by the safety authorities during safety reviews of the facility, which take place each decade. Of course, the facility's technical documentation must be updated (change requests, quality summaries, instructions, technical data sheets), together with the safety files affected (safety analysis report, and general operating rules (Règles Générales d'Exploitation)), which are usually submitted at the same time as the change request.

2.4. Operational/maintenance/decommissioning impact

Change requests include details of the operating modes of the new equipment and the schedules for periodic inspections and testing to which they will be subjected.

In general, the maintenance of new equipment can be reduced subsequent to a trial period. On the other hand, the number of items of equipment being subjected to inspections is increasing. As a result, the number of inspections and maintenance operations is also increasing.

The impact on decommissioning is outlined in detail in the documents sent to the safety authorities. In compliance with legislation, it is reassessed (including a financial evaluation) every three years.

2.5. Technical capabilities

The modernization of equipment increases the availability of the HFR; at the same time, the ageing of other equipment slightly reduces this availability. In addition, the number of statutory requirements is increasing as are plant internal technical demands. This means that HFR requires a slightly larger workforce not only to guarantee the operation of a growing amount of equipment but also to plan, manage, and implement the modernization programmes.

3. PROJECT MANAGEMENT

Under this headline and as an example for refurbishment, the seismic reinforcement project (known as the Refit Programme) will be described. The project ran from 2002–2007, during which a significant level of operation was maintained with 150 days of scientific activity per year.

3.1. Project organization

The project, which was initiated in 2002 for a duration initially estimated at 5 years, comprised setting up a temporary structure (the Refit Management Committee) in addition to the existing divisions in the ILL organization chart, bolstered by extra staff with skills and experience that did not necessarily exist at the ILL.

The Refit Management Committee (RMC) was made up of a team representing ILL Management (ILL Director, Head of the Reactor Division, Head of the ILL Finance Service, Chairman of the Millennium Programme Management Committee) and of an operational team comprising a unit with managerial, organizational, and control functions (RMC/DIR) and six task groups whose specific areas of operation are outlined below:

- group BAAN responsible for the auxiliary buildings ILL4, ILL7, ILL22 and the air-intake building, and for modifications to the areas immediately surrounding these buildings;
- group ETUS responsible for safety studies and the drafting of safety and radiation protection documents;
- group DETRI responsible for the detritiation facility;
- group RICC responsible for the instrumentation and control systems and protection against the risk of fire;
- group GEVE responsible for the ventilation and confinement system of the reactor building (containment and equipment), as well as for seismic civil engineering work and studies;
- group SEME responsible for the mechanical equipment inside the reactor building.

The 6 groups were responsible for the follow-up of the recommendations of the ‘Groupe Permanent’ of Experts for Nuclear Reactors dating from 2 May 2002, ILL's commitments concerning this ‘Groupe Permanent’, and the action to be taken with respect to the recommendations of the ‘Groupe Restreint d'Experts’ (select group of experts). The organization chart of the Refit Programme is shown in Figure 5.

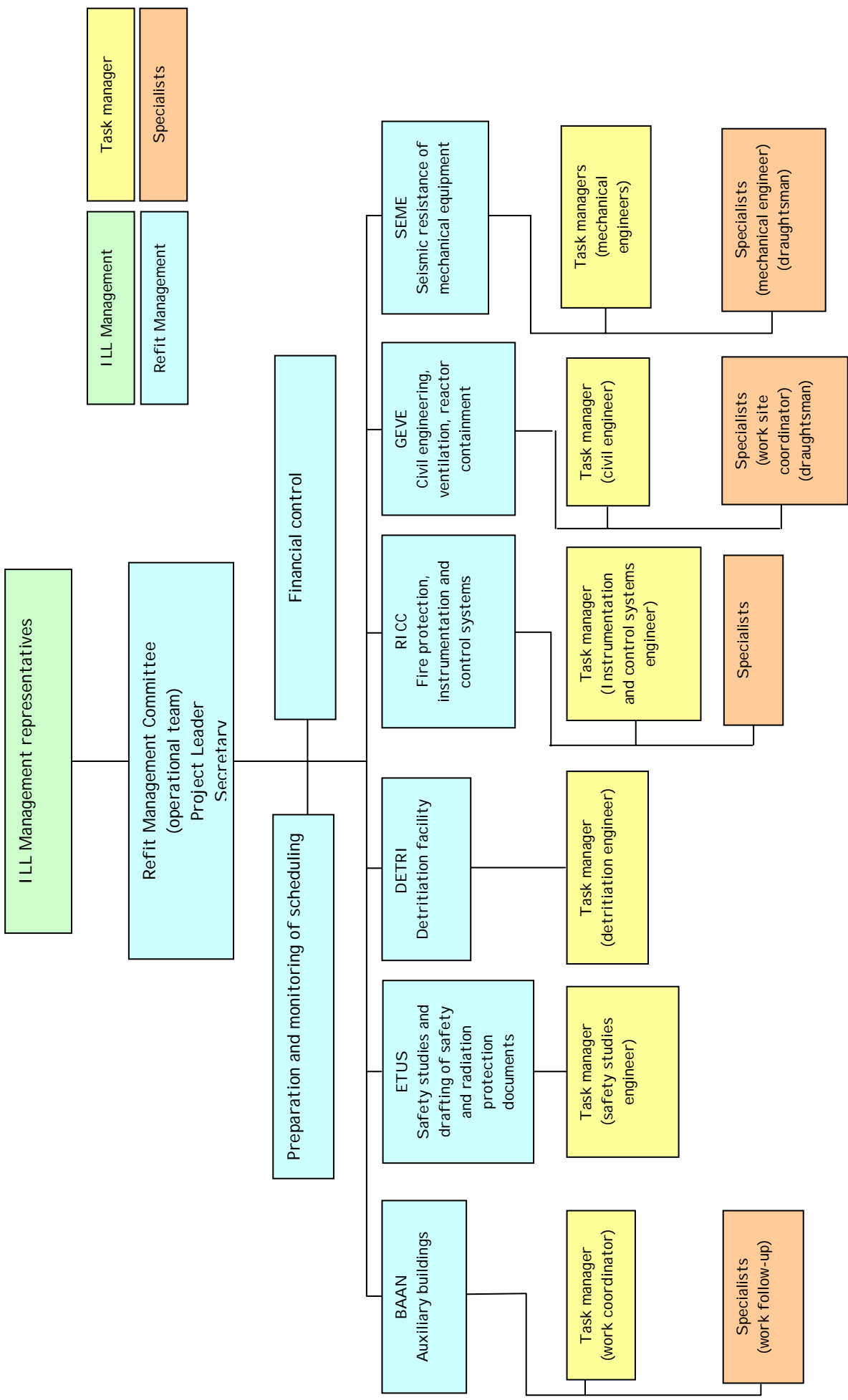


FIG. 5. Refit Project organization chart.

3.2. Resources

The operational teams were made up of ILL staff seconded to the Project Group (on a full or part-time basis) and staff from outside the ILL who took part in the project within the framework of technical assistance contracts.

The human resources within the Project Group were managed by the Project Leader.

The human resources from ILL needed for the Project Group were managed between the group leaders and the ILL heads of service. In cases of conflicts, any necessary arbitration was performed by the Project Leader and the division heads. Final decisions were made by the ILL Director.

3.3. Integration and risk management

In the ILL hierarchy, the Project Group was assigned to the ILL Management. Any ILL staff being a member of the Project Group reported on their activities within the group to the Project Leader. However, they remained assigned to their respective divisions.

Concerning technical choices/decisions, the RMC was advised and monitored by the Expert Advisory Committee (EAC), a group being set up on behalf of ILL Management and made up of experts from outside the institute. Twice a year, the RMC presented a progress report on the status of the project to the EAC and the EAC delivered an opinion thereto.

The Project Leader ensured that the analysis and management of the risks associated with the work to be carried out as a result of the RMC's activities were taken into account. Four types of risk were involved: nuclear safety, radiation protection, conventional safety, and industrial hazards.

3.4. Schedule

Initially, the project was planned for 5 years (2002–2006). Actually, the studies and work got underway at the end of 2002. In addition, in view of the operational requirements for the HFR (at least three 50-day cycles per year), the shutdown periods available for the work were relatively limited and could not be altered (one 6-month shutdown at the end of 2004 and the beginning of 2005, and one almost 11-month shutdown at the end of 2005 to the beginning of 2006). In the end, the project lasted from the end of 2002 all the way to 2008 (completion of studies and work).

A work schedule was drawn up for each of the RMC groups and was regularly updated by the relevant group leader. In collaboration with the group leaders, an overall time schedule for the project was prepared and updated by the planning officer for the Project Leader. To this end, the planning officer ensured that this time schedule complied with the normal reactor operating schedule and, if necessary, with those of the other divisions of the ILL. On the basis of the above schedules, deadlines were monitored by each of the group leaders and supervised by the Project Leader.

3.5. Budget

At the start of the Refit programme, the overall budget for external expenditure (excluding taxes and ILL staff costs) was estimated at €20 million ± 20%, over a period of 5 years (2002–2006).

In order to guarantee independent financial control of the project, ILL entrusted the role of financial controller to an outside specialist responsible for implementing a cost accounting structure for monitoring expenditure, producing a monthly financial report (statement of orders in progress, monitoring of actual expenditure compared with the budget, cash flow monitoring) and forecasting budget changes in the short and medium term.

As of 2002, the studies and technical choices had not been completely finalized. This fact explains in part the divergence with the final budget figure of approximately €29.7 million over a 7-year period (2002–2008) (see also Fig. 6).

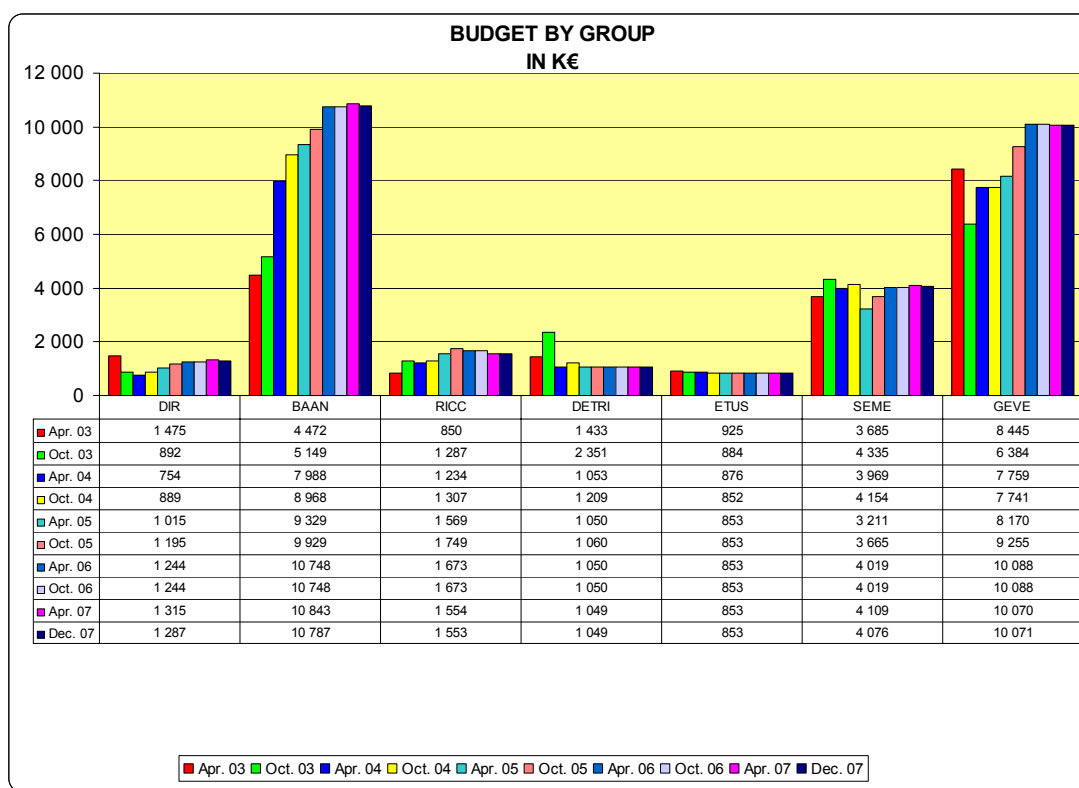


FIG. 6. Budget changes per group for the Refit Programme between 2002 and 2007.

3.6. Procurement

The general purchasing rules were those mandatory at the ILL. In particular, the commercial aspects were monitored by the relevant administrative services, primarily the Purchasing Service, in close collaboration with the financial controller of the RMC.

Only the authorization limits for purchase requests were changed compared with the ILL rules.

All invoices by the suppliers were addressed to the Finance Service. A copy was forwarded to the financial controller for payment authorization ('bon à payer' stamp) following technical validation of the order and thorough administrative checks.

3.7. Communication

3.7.1. External communication

The RMC had no direct relations with the nuclear safety authorities. Contacts took place via ILL's usual representatives in dealings with these bodies (Director of ILL, Head of the Reactor Division). It was, therefore, the role of the RMC to prepare for these representatives the responses to the recommendations and the documents needed to fulfil ILL's commitments to the safety authorities.

Nevertheless, the RMC structure, its organization and the skills and know-how of its staff were presented in external relations by a separate body, set up within the ILL to satisfy the recommendations and opinions issued by the safety authorities.

In connection with the project, ILL attended and submitted papers to international conferences, such as the ENC (European Nuclear Conference) in December 2005 and IGORR (International Group of Research Reactors) in September 2005 and March 2007.

3.7.2. Internal communication

The Project Group organized staff information sessions throughout the course of the project.

A dedicated page on the ILL intranet was also used to inform staff on the progress of the work and any possible disruptions of operation schedules that might be caused by it.

3.8. Quality

The quality of the project was managed by the group leaders and the Project Leader. The key items for the management of the project were the verified and approved written documents. The standard rules of Quality Assurance at ILL were applied for the project.

Any changes or deviations from the initial objectives of the project have been justified in writing.

4. LESSONS LEARNED

4.1. Safety-related lessons

It is clear that it is in the operator's interest to anticipate changes that need to be made, in particular for safety reasons or due to the reinforcement of safety standards.

This anticipation serves to facilitate relations with the safety authorities and above all to ensure better supervision and managing of the facility, which is the operator's prime responsibility.

4.2. Risk management lessons

The adoption of a risk management approach reduces the risk of an interruption of operation due to the shutdown of the facility for outages or, generally for a longer period, for safety reasons.

It is therefore obvious that all risks must be prioritized as follows:

- safety;
- extended outages;
- performance levels – short outages.

4.3. Project-related lessons

Major modifications inevitably disrupt operation and the organization of shift teams. On the other hand, they produce obvious benefits in terms of staff motivation, safety, and availability. At the same time, care must be taken to avoid overloading staff with work.

In order to guarantee the success of major modernization projects, it is vital to have access to the best possible skills in the field concerned. This is why for the Refit Programme (2002–2007), the ILL decided to set up a project group involving many external specialists. Obviously, for normal operations the ILL does not need staff, who are highly specialized in civil engineering or the seismic behaviour of ground and structures.

On the other hand, for projects which require internal know-how (deuterium, tritium, etc.), the ILL will, in the future, use an integrated project structure based on internal staff, and will then look for outside support for those activities for which it has no internal expertise, in order to reduce the workload on its own staff.

Experience has shown that, although necessary, these major modernization projects place a considerable stress on the staff. This is all the more true because it is difficult to increase the workforce, given that at the end of the work (which is expected to last for approximately another 10 years at the ILL) staff numbers will have to fall again to the level required for normal operations.

5. BRIEF OUTLINE OF FUTURE PLANS

Two medium-term programmes are already underway at the ILL. One is a programme for the renewal of scientific instruments that is scheduled to run until 2012 and beyond: the Millennium Programme Phase M-1, which covers the period from 2007–2012 with a budget of €39 million, is the continuation of the Millennium Programme Phase M-0, which has comprised an investment of €34 million between 2001–2008. The other programme is the upgrading of the key reactor components at a cost of around €20 million between 2007–2013.

For programmes of this kind to succeed, the following conditions must be satisfied:

- the acceptance of the facility by the safety authorities and the public;
- the determination of management;
- the firm will and conviction plus the ability of the associates to invest;
- the existence of strong demand and support from the scientific community in the long term.

Because these conditions are met, ILL can plan to operate the HFR until 2030 (Fig. 7 shows the reactor at work).

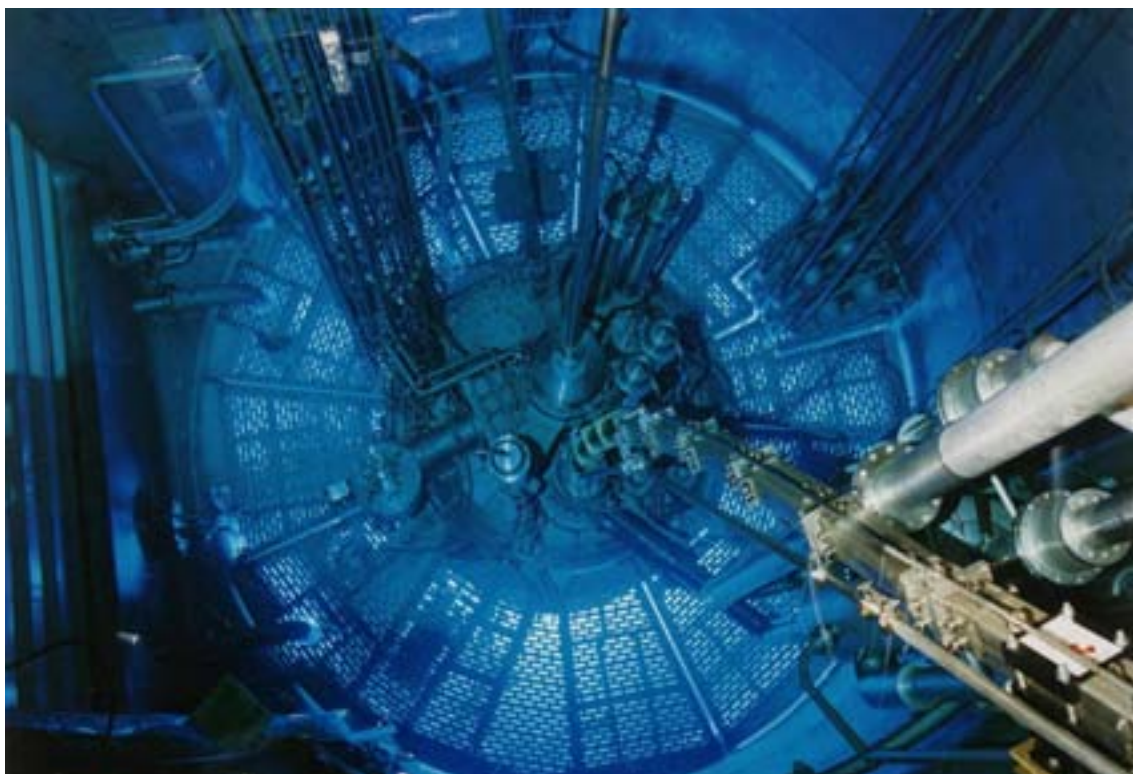


FIG. 7. HFR at work.

COMPLETE REFURBISHMENT OF THE AKR TRAINING REACTOR OF THE TECHNICAL UNIVERSITY DRESDEN

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Abstract

In 2004–2005, the AKR-2 training reactor of the Technical University Dresden (TUD) was completely upgraded in a refurbishment comprising civil work as well as new electrical and I&C equipment. One of the main steps was the complete modernization of the entire I&C system by AREVA NP GmbH. The digital safety system, TELEPERM XS, that had been implemented or contracted in more than 50 NPPs in 9 countries by 9 different manufacturers served as the basis for the technical solution. Currently, AKR-2 is the most up-to-date training reactor in Germany and is used for the training of students and professionals, for research, and as an information centre for the public.

1. INTRODUCTION

The AKR-1 training and research reactor (from the German Ausbildungskernreaktor) of the TUD was put into operation in 1978. For more than 25 years, the facility was successfully used for the training and education of students, for nuclear research projects, and as an information centre for the public. Among these groups, there are a high percentage of grammar school students in special courses on nuclear and reactor physics [1]. In the past, about 1 000 visitors each year took advantage of the AKR's capabilities.

In 1998, a new licensing procedure began with the goal of completing a comprehensive refurbishment of the reactor facility. The refurbishment took place in 2004 and included civil work as well as new electrical and I&C equipment. After having successfully completed all startup requirements, the regulatory authority authorized normal operations of the new AKR-2 reactor in April 2005.

This was a remarkable event because for more than 20 years only two licences were granted by the German regulatory authorities under Part §7(1) of the Atomic Energy Act for construction and operation of new nuclear plants. One of these licences is the AKR-2 training reactor of the TUD.

Training reactors like the AKR contribute to the maintenance and enhance nuclear know-how and competence. Even today with the current political circumstances in Germany (characterized by the intention to phase out nuclear power), there is considerable demand for young engineers and scientists in the nuclear sector. The demand is driven by retirement of staff members at NPPs and nuclear industrial suppliers, continued work in the field of basic nuclear physics research, nuclear engineering, radiation protection, waste disposal, nuclear medicine, and the administration and technical surveillance organizations. A special demand has arisen in the nuclear industry due to modernization of existing NPPs or new projects in other countries of Europe and global regions like Asia.

2. FIELDS OF APPLICATION OF THE AKR

2.1. Education and training

The main purpose of the AKR and its design basis was and is the education of students in nuclear and reactor physics, nuclear engineering, as well as in teaching basic knowledge and rules in radiation protection and radiation dosimetry. Basic experiments are provided and carried out in practical exercises for:

- nuclear engineering students;
- physics students;
- physics and mathematics lectureship students;
- interested students of any faculty of the university.

Since most universities and colleges do not have corresponding facilities to combine their lectures with practical exercises, students from other universities all over the country are welcome to participate in practical courses at the reactor. The course duration and selection of exercises are individually tailored to the needs of the student.

Specialized training courses in reactor physics have also been developed for personnel in the nuclear industry [2]. The AKR offers special training to young staff members from nuclear companies to provide them with the opportunity to get on-the-job training in order to help solve complex technical problems. The AKR training reactor is ideal for such a programme because of its education programme, experience, and equipment. The courses are primarily aimed at newly recruited staff members, but also experienced staff members as a refresher course in reactor physics. The current number of university graduates in this field does not meet the recent demand for qualified specialists in the nuclear industry. Consequently, the majority of recently recruited personnel go into industry with miscellaneous professional specialization and experience and without a nuclear background. These courses are one week long with a maximum number of eight participants and are a combination of lectures in reactor physics basics and subsequent practical exercises at the reactor. It is a unique advantage of the course that participants can immediately combine theoretical knowledge with practical experience in reactor operation and its behaviour. This synergistic effect is highly appreciated by course participants.

2.2. Application in research projects

Due to its physical characteristics, a zero power reactor, such as the AKR, offers limited research possibilities. However, it can be used in projects where high neutron fluxes are not required, but where variable operational conditions and low costs are requested. As a result, the AKR is involved in research projects, such as investigations on sophisticated neutron detectors, development of measuring techniques for safeguards purposes, radiation spectrometry in mixed neutron-photon fields, experiment-to-calculation comparison of neutron and gamma energy spectra in benchmark arrangements for reactor material dosimetry purposes.

2.3. Information centre for the public

Advantages of small, low power training reactors on a university campus are their central location and the relaxed admittance requirements not allowed in other nuclear installations. These facilities are suitable for use as information centres for groups or for private individuals. In addition to the transfer of basic knowledge, it is very important to

contribute to the public discussion of nuclear energy. Where else is almost any member of the public allowed (supervised by the reactor staff) to operate a real nuclear reactor in order to get an impression of its physical behaviour and an understanding of how the nuclear chain reaction is controlled?

3. BASIC TECHNICAL DESIGN IDEAS AND SAFETY FEATURES OF THE AKR

The TUD has educated students in nuclear technology since the foundation of a nuclear faculty in 1955. In subsequent years, additional courses were introduced to educate nuclear engineers in order to fulfil demands of industry, science, and administration. Students were given lectures, but theoretical knowledge had to be combined with practical experience based on an extensive programme of fundamental experiments in reactor physics, neutron physics, nuclear technology, radiation measurement techniques, radiation protection, radiation dosimetry, and others.

The full scale of this experimental programme was transferable to small training reactors that could be operated with great diversity in terms of experimental intentions and without commercial restrictions. For these reasons, the AKR training and research reactor was constructed in the heart of the Dresden university campus as an attractive experimental device for students. Its design and requirements for sitting on a university campus are as follows:

- Sitting on campus inside the area of the university requires high safety and reliability.
- Simple construction allows students to study the most important reactor parameters.
- Simplified operational procedures allows students to operate the reactor themselves (supervised by the reactor staff).
- Use of additional equipment extends the experimental capabilities of the facility.
- Simple construction allows for ease of use and of maintenance.
- Fewer restrictions on reactor access for students as compared to larger nuclear installations.

The AKR reactor is a thermal, homogeneous, solid material-moderated, zero power research reactor with maximum continuous power of 2 W. Safe operation of the reactor is guaranteed by a combination of inherent safety features, engineered safeguards and administrative procedures that allow inexperienced students to operate the facility. The AKR-1 achieved its first criticality on 28 July 1978. It was developed analogously to the well-known types AGN201 [3] and SUR100 [4], and has proved to be efficient and reliable in practice.

Technical advantages of the AKR design are:

- application of low enriched uranium (LEU), i.e. ^{235}U content in the fuel is < 20%;
- low absolute amount of nuclear fuel (total mass of ^{235}U in the reactor < 1 kg);
- inherent safety features (negative temperature coefficient of reactivity, integrity of the core maintained even in hypothetical case of power excursion);
- simple construction resulting in high reliability, low costs for inspections, maintenance and operation;
- strict avoidance of any liquid in the facility (no corrosion, no danger of contamination in case of leakage, no maintenance of water systems, no substitution or cleaning procedures of liquids);
- miscellaneous irradiation capabilities in the experimental channels;

- free space above the core for possible installation of additional experimental equipment.

The structure of the training reactor is shown in Figures 1 and 2. The cylindrical core has a diameter of 250 mm and a critical height of 275 mm. The disk-shaped fuel elements consist of a homogeneous dispersion of polyethylene and uranium oxide (19.8% ^{235}U enrichment, O/U ratio 2.27). The ^{235}U density in the fuel elements amounts to 0.060 g/cm^3 .

The core is completely surrounded by a graphite reflector (density 1.75 g/cm^3). Its axial and radial thickness is 200 mm and 320 mm, respectively. Therefore, the critical mass is relatively small (about $790\text{ g }^{235}\text{U}$). Within certain restrictions, the AKR is a minimum critical mass reactor.

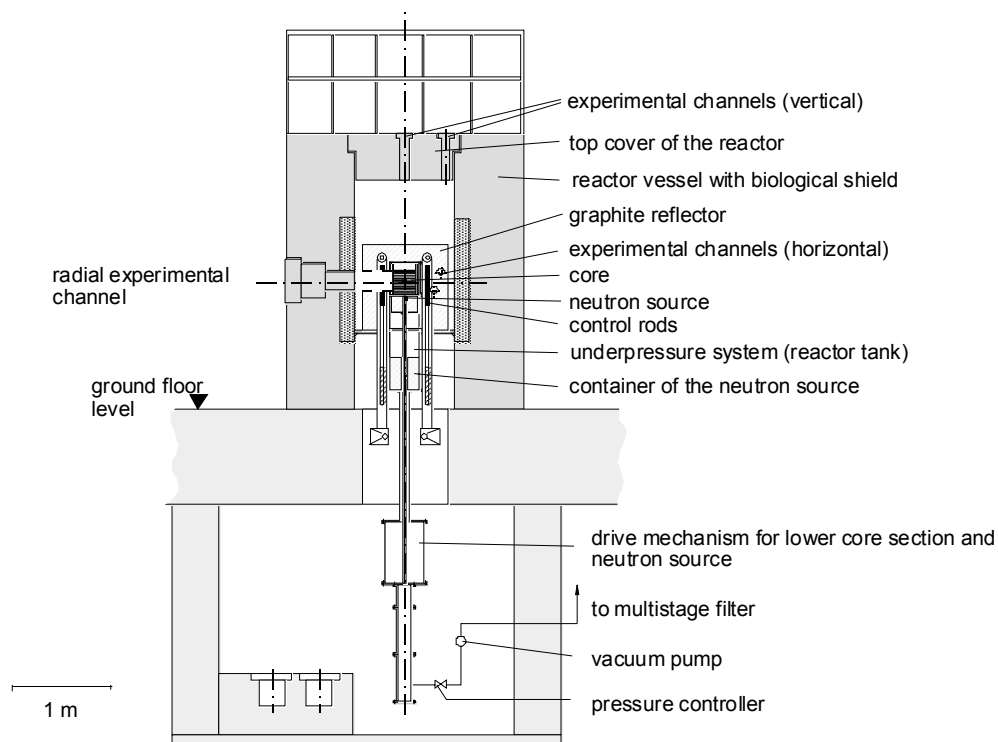


FIG. 1. Vertical cross-section of the AKR.

For safety reasons, the core consists of two separable sections. The fuel elements in each section are enclosed in a hermetically sealed aluminum container. A second and larger gastight tank encloses both the core sections and the parts of the reflector. The pressure inside this larger tank is slightly reduced compared with the environment. This subpressure acts as a barrier to prevent an uncontrolled leakage of radioactive fission products even in the unlikely case that all the other internal retention barriers fail. A pre-vacuum pump continuously maintains the tank subpressure, its control is automatically performed by means of a pressure controller even when the AKR is shut down. In the shutdown position, the distance between the lower and the upper core section is about 50 mm, i.e. the core is separated into two subcritical masses. The lower section is lifted by means of a core drive mechanism including an electromagnetic holding device. The startup neutron source (AmBe, neutron yield

$2.2 \times 10^6 \text{ n}\cdot\text{cm}^{-2}\cdot\text{s}^{-1}$) is moved through a tube within this mechanism from the source container to the bottom side of the core.

Three cadmium absorber plates control the AKR. These plates are moved vertically within the reflector outside the reactor tank. They are designed as combined control and safety rods. The lower core section and the control rods are held in their ‘working’ positions by solenoids. Any scram signal releases the rods and the lower core section, which then fall by gravity into their shutdown positions. The top cover of the AKR is removable. The free space above the core can be used for installing a thermal column or a subcritical assembly. There are six horizontal and vertical experimental channels with different diameters. They provide adequate in-pile irradiation volume with different neutron spectra.

The permissible continuous power level is limited only by the effectiveness of the biological shield. It consists of paraffin and barite heavy concrete with a total thickness of 750 mm. With the condition that the equivalent dose just outside the shield should never exceed 20 mSv/a (limit of the effective dose during any one calendar year for occupationally exposed persons in Category A according to the German Radiation Protection Ordinance), even under the worst circumstances, continuous 2 W operations are possible. At 2 W, the maximum thermal neutron flux density in the central experimental channel is about $5 \times 10^7 \text{ cm}^{-2}\cdot\text{s}^{-1}$.

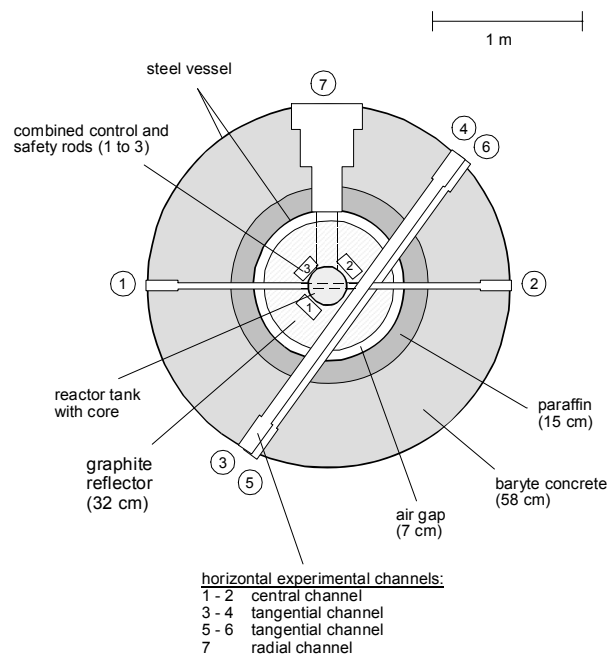


FIG. 2. Horizontal cross-section of the AKR.

Safe and foolproof reactor operations are guaranteed by a combination of inherent safety, engineered safety, and administrative procedures. The design and operation conditions of the AKR are based on the requirement that prompt criticality must not occur at all and an undue increase in power must not endanger the operators, the environment, or the reactor itself. Thus, safe operation of the reactor is ensured by the following measures:

- Excess reactivity is restricted to a maximum of 0.3%, i.e. prompt criticality is definitively excluded.
- All three control rods are designed as combined control and safety rods. The reactivity value of each rod is sufficiently high to shut down the reactor and keep it subcritical. Thus, even in the case of failure of any two rods, the AKR will be reliably shut down.
- Independent of any control rod movement, a separation of the two core sections by 50 mm reduces the reactivity by 5.8%. In case of a scram, this negative reactivity becomes effective within about 100 ms, thus ensuring a diverse fast shutdown of the reactor and a high degree of nuclear safety in the shutdown state.
- The temperature coefficient of reactivity is negative. From measurements, a value of $(2.90 \pm 0.05) \text{ } \beta/\text{K}$ was found. Estimates have shown that power excursions would be self-limiting before damages occur in the reactor or in its environment. This high inherent safety results from the physical properties of the fuel elements.
- By monitoring the core temperature and by including the measured value into the safety circuit, the possibility that additional positive reactivity could be introduced due to decreasing core temperature is prevented.
- All drives are dimensioned such that the rates of reactivity changes are lower than $\Delta k/k = 0.0001 \text{ s}^{-1}$.

Due to these features, a power excursion with harmful consequences can be virtually excluded.

4. PROJECT PLANNING DETAILS AND LICENSING PROCEDURE FOR THE REACTOR REFURBISHMENT

After commissioning in 1978, the AKR-1 was continuously operated in accordance with the operational licence granted by the regulatory authority of the former GDR (East Germany). In 1989, after 10 years of successful operation, a comprehensive periodic safety review of all operational and safety systems of the facility and of the reactor administration procedures were performed. As a result, the regulatory authority renewed the licence for an unlimited period.

After German reunification in 1990, a new paragraph (§57a) was introduced into the German Atomic Energy Act (Act on the Peaceful Utilization of Atomic Energy and the Protection Against its Hazards). The new paragraph stated that existing atomic licences in the East German states expired on 30 June 2005. Consequently, a new licence would be required based on the German Atomic Energy Act, paragraph 7(1) if the reactor were to continue operations. It was the mutual intent of the University and the Saxonian State Ministry of Science and Culture to continue student education at the AKR-1 reactor. This intention was in agreement with the recommendation by an expert group of specialists from all over Germany.

In 1998, a new licensing procedure began with the aim to fulfil all prerequisites to obtain a new operational licence based on requirements of the German Atomic Energy Act §7(1) with the Saxonian State Ministry of Environment and Agriculture as the responsible regulatory authority.

Within the framework of this licensing procedure, the technical equipment of the facility was assessed. As a result of the assessment, it was determined that an extensive refurbishment of the reactor facility, including auxiliary technical systems and the reactor

building, as well as I&C equipment was needed. The refurbishment costs were comparable to the construction and commissioning of a new facility.

All required application documents were initially prepared by the university's reactor group in accordance with the Ordinance on the Procedure for Licensing of Installations under §7 of the Atomic Energy Act (Nuclear Licensing Procedure Ordinance). Since reactor design and safety criteria in the German KTA-Safety Standards are defined exclusively for nuclear power plants, they could be applied to a research reactor only in a general sense. Therefore, the licensing procedure was strongly based on the guidelines and recommendations published in IAEA Safety Series No. 35, e.g.:

- Code on the Safety of Nuclear Research Reactors: Design (Safety Series No. 35-S1);
- Code on the Safety of Nuclear Research Reactors: Operation (Safety Series No. 35-S2);
- Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report Safety Guide (Safety Series No. 35-G1);
- Safety in the Utilization and Modification of Research Reactors Safety Guide (Safety Series No. 35-G2).

Since the facility is small with a maximum continuous power of 2 W, no official public announcement of the project according to §4(1) of the Nuclear Licensing Procedure Ordinance was required in advance, but the university voluntarily distributed information to the public. Indeed, a variety of third parties and other authorities of the region and of the city of Dresden were involved in the licensing procedure. The planning of the refurbishment project was prepared and supervised by a company specializing in the nuclear field (RWE NUKEM GmbH, regional business unit) in cooperation with other companies.

In October 2002, the Saxonian State Ministry of Finance provided financial support for the project and has, hereby, guaranteed the feasibility of the refurbishment. After the positive assessment of the planning project documents by the Technical Inspection Agency (TUEV) in May 2003, a revision of the planning details and of the safety report was made according to the recommendations of the independent experts, and the operating and testing manuals for the AKR-2 were worked out in detail.

All required prerequisites were met so that the Saxonian State Ministry of Environment and Agriculture (SMUL), as the responsible regulatory authority could grant the atomic licence based on §7(1) of the German Atomic Energy Act on 22 March 2004 for:

- interim storage of nuclear fuel and other radioactive material from operation of the old AKR-1 reactor;
- construction of the new facility AKR-2;
- commissioning and specified normal operation of AKR-2.

The combination of licensing, commissioning and operation in only one administrative decision was a great advantage for the university and a warranty that the reactor would also be operated after construction without danger of additional unexpected legal problems. The licence announcement was published in the regional press and in a reference in the Federal Bulletin (Bundesanzeiger). There were no objections against the AKR-2 project by either public groups or individuals.

5. CONSTRUCTION PHASE

The construction phase took place between April–December 2004. In preparation, all radioactive material was removed from the facility, i.e. the reactor fuel was unloaded and transferred together with all other radioactive substances and sealed sources into an interim storage within the same building. Remaining radioactivity due to the activation of reactor components was negligible because of low neutron fluences; contamination was also extremely unlikely. Independent verification measurements validated that neither activation nor contamination were detectable above the values given in Appendix III of the German Radiation Protection Ordinance — Ordinance on the Protection against Damage and Injuries Caused by Ionizing Radiation. Hence, the controlled area of the reactor hall could be classified into the lower level of a supervised area making the construction work considerably easier but still within the scope of the Radiation Protection Ordinance.

Refurbishment of the reactor facility did was composed of civil work as well as new electrical and I&C equipment, including:

- construction work at reactor building (Fig. 3);
- new construction of the personnel lock;
- complete new installation of electrical systems and room ventilation (Fig. 4);
- structural and equipment-related fire protection measures;
- new radiation monitoring equipment;
- systems for physical protection;
- new crane bridge and crane travelling gear;
- complete modernization of the entire I&C system based on the digital reactor protection instrumentation and control system TELEPERM XS by AREVA NP (Fig. 5) and new installation of 3 channels for the neutron flux monitoring: two (logarithmic) wide range and one (linear) power range channel (see Chapter 7).

On the other hand, the nuclear design of the reactor itself remained the same and no internal components or structures (reactor core, control rods, reflector, shielding, neutron detectors etc.) were replaced with the exception of a few switches and drives.



FIG. 3. Construction work in the reactor hall.



FIG. 4. Filter system of room ventilation.



FIG. 5. Installation of I&C systems.

6. COMMISSIONING OF AKR-2

The final report on the termination of construction of the new AKR-2 was sent to the regulatory authority on 16 March 2005. From 22–24 March 2005, field installation checkout, startup procedures, and nuclear commissioning of the plant according to confirmed commissioning procedures (including reloading of the reactor core with nuclear fuel and the critical experiment on 22 March 2005) were successfully performed in the presence of the regulatory authority and Technical Inspection Agency (TUEV).

After confirmation of the startup test report, the Saxonian State Ministry of Environment and Agriculture, as the regulatory authority, granted the approval for the start of

specified normal reactor operation on 7 April 2005. Training exercises for the students started the next day at the new AKR-2 facility as seen in Figure 6.



FIG. 6. The new AKR-2 training and research reactor of the Technical University Dresden.

7. THE NEW I&C SYSTEM WITH TELEPERM XS

7.1. Architecture of the I&C system

One of the most important steps in reactor refurbishment was the complete modernization of the entire I&C system. The digital safety system TELEPERM XS (by AREVA NP GmbH Erlangen, Germany) which has already been implemented or contracted in more than 50 NPPs in 9 countries by 9 different manufacturers served as the basis for the technical solution (reactor protection as well as operational instrumentation).

The nuclear instrumentation consists of three independent redundant channels for neutron flux measurement (digital system TK250, MGP Instruments GmbH Munich, Germany):

- two (logarithmic) wide range channels (DAK 250-i, pulse measurement with fission chamber detectors);
- one (linear) power range channel (DAK 250-g, DC measurement with γ -compensated ionization chamber detector).

The neutron flux channels of the TK250 system were developed for BWR and PWR applications and are designed and certified according to the requirements of the German KTA-3501 safety standards.

The architecture of the AKR I&C system is shown in Figure 7. The TELEPERM XS instrumentation rack and the reactor operator's control desk are presented in Figures 8 and 9.

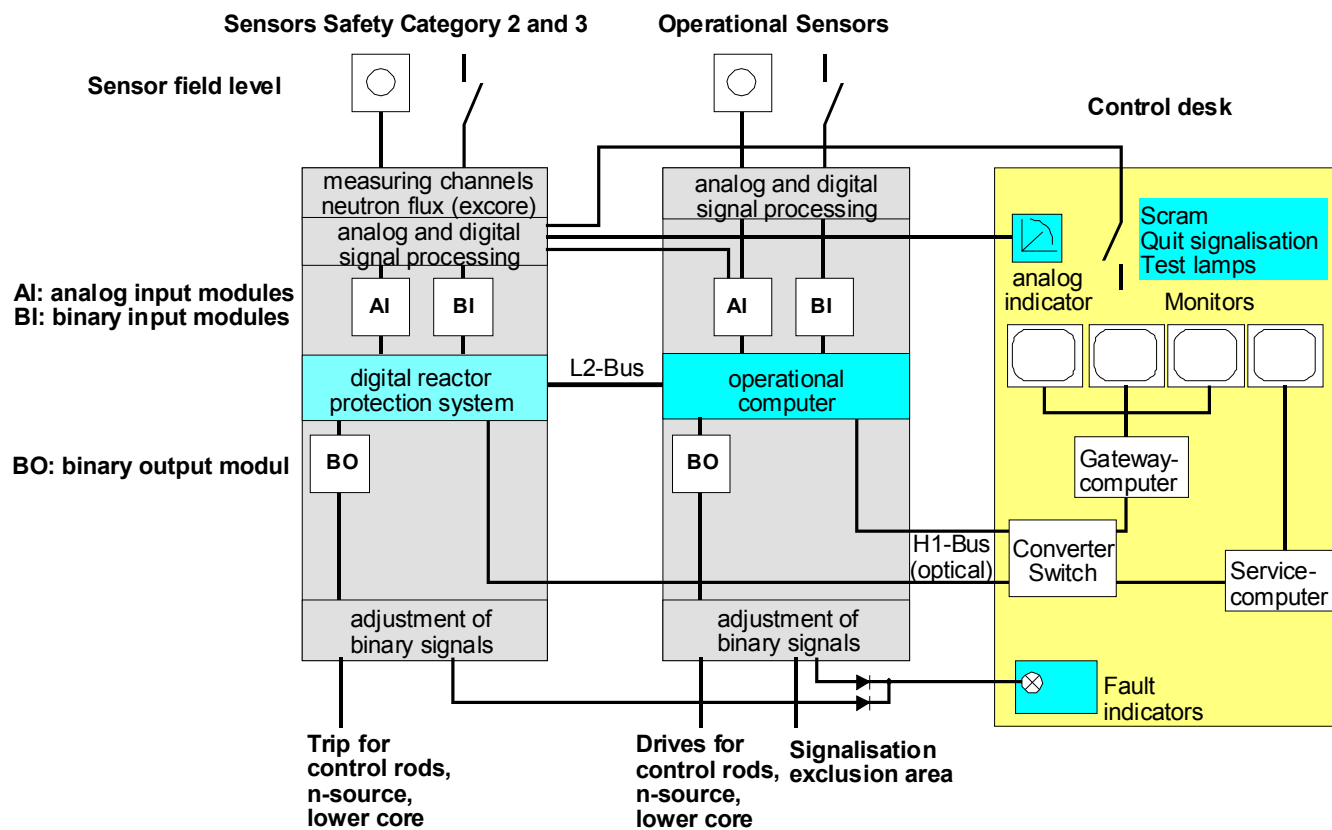


FIG. 7. Architecture of the I&C system of the AKR-2.

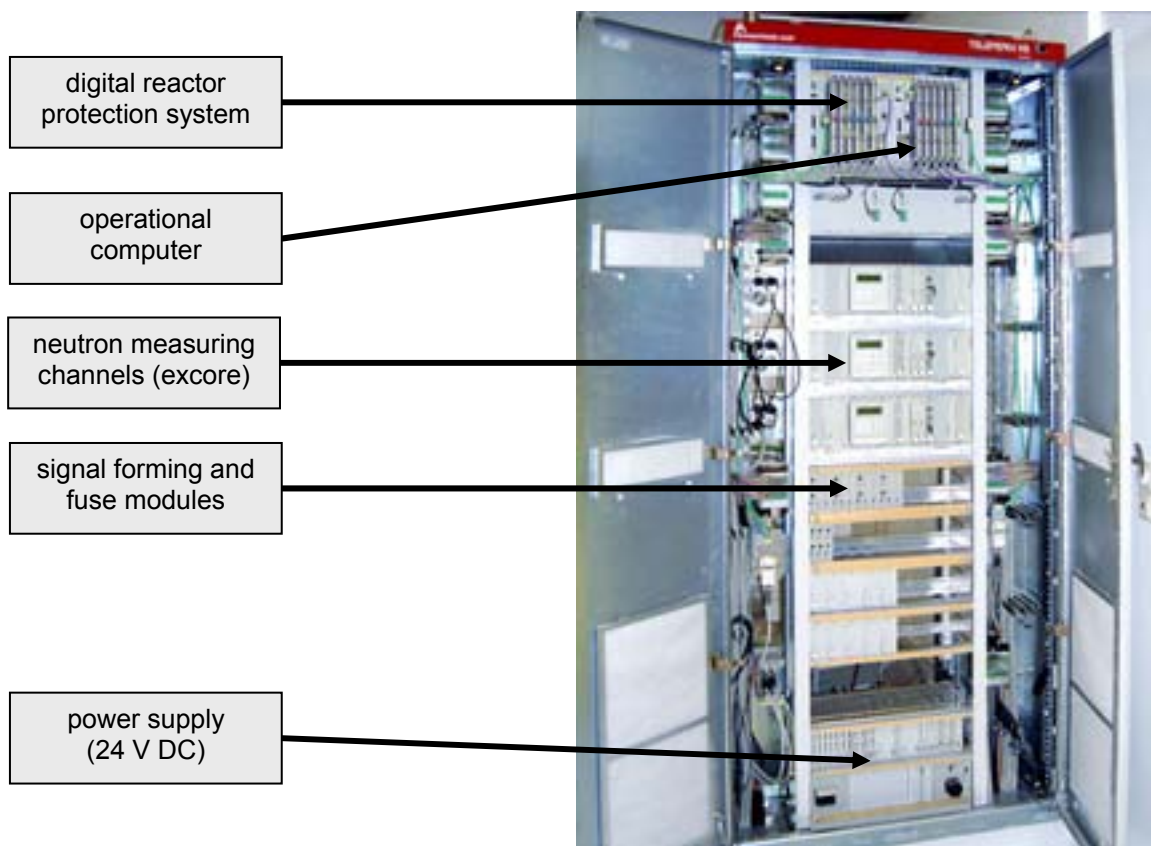


FIG. 8. TELEPERM XS rack of AKR-2.

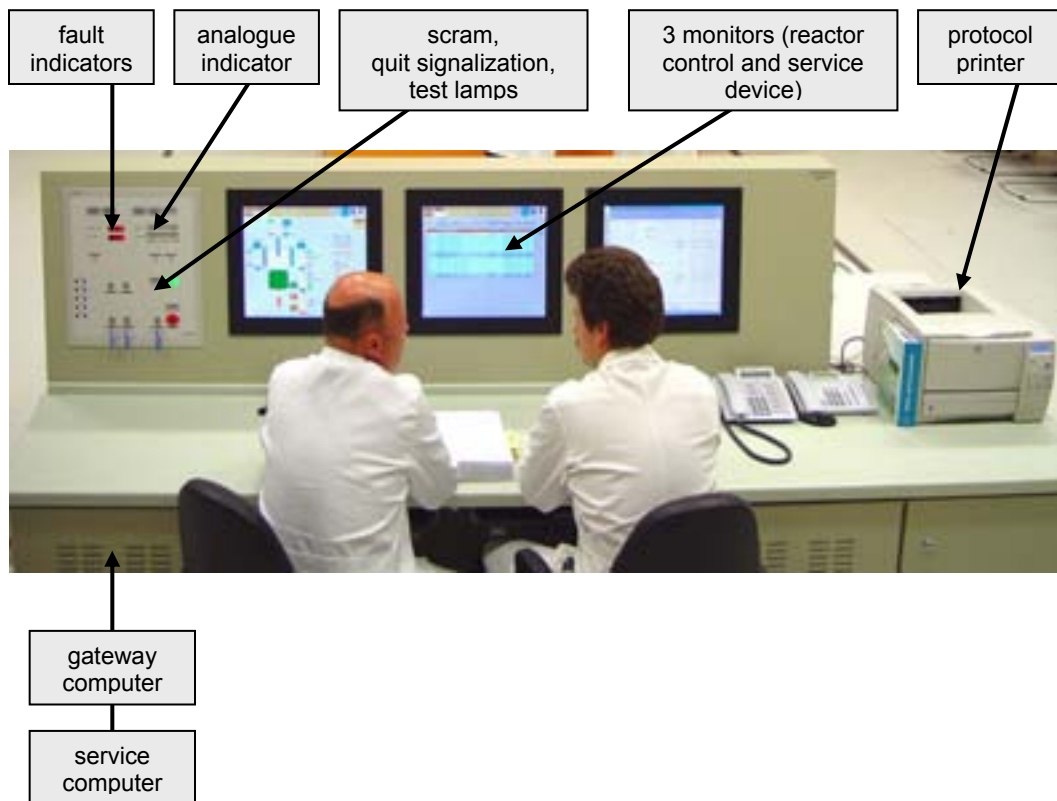


FIG. 9. Reactor control desk.

7.2. Wide range channels (Startup/Middle/Power Range)

The logarithmic wide range (startup/middle/power range) monitors with two redundant measuring channels DAK 250-i cover 6 decades (approximately $2-2 \times 10^6 \text{ n cm}^{-2}\cdot\text{s}^{-1}$) of neutron flux density at the measuring location outside the graphite reflector. Fission chambers (type 9R100) are used as detectors.

In the safety system, the absolute fission chamber signals and the so-called RELFAEG signals (being reciprocal to the reactor period) are evaluated for the following limits (prevention of the reactor start as well as release SCRAM):

- minimum neutron flux required for reactor startup (1/2 logic);
- SCRAM signal for exceeding maximum reactor power (1/3 logic together with the signal of the power range channel);
- SCRAM signal for too small reactor period (1/2 logic) or too high relative neutron flux exchange speed (RELFAEG), respectively.

7.3. Power range channel

The linear power range channel, DAK 250-g, measures the upper 2–3 decades (approximately $5 \times 10^3 - 2 \times 10^6 \text{ n}\cdot\text{cm}^{-2}\cdot\text{s}^{-1}$) of the neutron flux density at a detector position outside the graphite reflector. A γ -compensated ionization chamber is used as a neutron detector. In the safety system, the chamber signal is evaluated for following limits (release SCRAM):

- SCRAM signal for exceeding maximum reactor power (1/3 together with the signals of the wide range channels).

7.4. Safety and control system

The safety and control system should protect the facility from inadmissible demands and shall minimize the effects on the staff, the surroundings and the reactor in case of accidents. The safety and control system causes automatic reactor shutdown (SCRAM) in the following cases:

- malfunction of drives of control rods, neutron source, or core;
- too fast power increase (reactor doubling time < 10 s, warning signal at 20 s);
- too high absolute reactor power (> 2.4 W, warning signal at 2.2 W);
- too low moderator temperature (< 18°C);
- too high pressure in the reactor tank (> -8 kPa);
- in case of malfunctions in the safety and control system;
- non-availability of the I&C computer;
- in case of malfunction in an external experiment (optional);
- in case of malfunctions in the I&C rack;
- an implicit SCRAM is initiated in case of a safety computer failure (fail-safe by closed-circuit connection).

7.5. Reactor signaling system

Three categories of signals are initiated according to their safety relevance:

- Status signals inform the operator about the condition of the system. They are displayed on the AKR-2 reactor control panel.
- Fault signals inform the operator of an abnormal condition in the operating system. They are displayed on the reactor panel and must be acknowledged by the operator. Fault signals will be saved in the reactor log file.
- Alarm signals indicate an abnormal condition in the safety system with audible and/or visual signals. Alarm signals will be displayed on the conventional operation and control panel and are saved in the reactor log file.

7.6. Service equipment

The service equipment is used for planning, configuration, monitoring, maintenance and modification of the reactor safety system such as:

- diagnostics;
- modification of I&C components;
- modification of parameters;
- documentation and saving of hardware and software specifications.

It can be distinguished between services without required communication with TELEPERM XS and those with access to the safety units. The following properties characterize the service equipment:

- It is not part of the safety I&C; therefore, it can be switched off.
- It is never used for reactor operation process management.
- A hierarchically arranged operating surface is provided for user prompts.
- Unauthorized access protection to the service equipment is given by administrative measures (login/logout function) and additionally by key-operated switches.

- The software is protected against modifications by setting read-only features and additionally by backup procedures on data storage (CD-ROM).
- The service computer (see Figs 7 and 9) is located in the reactor control desk behind a locked door.

7.7. Gateway computer

The gateway computer (see Figs 7 and 9) sends information from the reactor safety and protection system to the control and monitoring system (and never in the opposite direction!). The gateway computer is located in the reactor control desk behind a locked door.

7.8. Interconnection between I&C units

Communications between the reactor protection and the operational computers in the TELEPERM XS rack to the gateway and service computers in the reactor operator control desk takes place via a fiber optic cable. In a media converter, the signals are converted into electric signals and transmitted to the gateway and service computers via an Ethernet switch (Fig. 7).

Normally, all three screens of the reactor control desk are connected to the gateway computer. Optionally, the monitor on the right hand side of the control desk can be swapped by a switchbox to the service computer (Fig. 8).

8. SUMMARY OF IMPORTANT STEPS AND TIMELINE OF AKR-2 REACTOR REFURBISHMENT

A summary of the important steps and a timeline of the AKR-2 reactor refurbishment is given in Table 1.

9. CONCLUSIONS

Since the AKR refurbishment, the AKR-2 at the Technical University Dresden is the most advanced training reactor of its kind in Germany. The AKR-2 is operated for the education of students and professionals, for suitable research projects, and as an information centre for the public. Training is performed, in principle, using the same I&C equipment (nuclear, safety and operational units) found in larger research reactors or nuclear power plants. In addition, the ability to conduct a variety of experiments in nuclear and reactor physics and radiation protection makes the reactor an attractive facility for many users.

The experience of the reactor refurbishment shows that the AKR-2 could be designed and commissioned only because all parties, i.e. the Technical University Dresden as licensee and utility operator, the SMWK and the SMUL as regulatory authorities, the TUEV as the technical experts, the design planner, and all companies involved had sincere intentions to successfully complete this project even under unfavourable political conditions for nuclear energy in Germany.

TABLE 1. IMPORTANT STEPS AND TIMELINE OF THE AKR-2 REACTOR REFURBISHMENT

01/1998	Recommendation of the Saxonian State Ministry of Science and Culture for continued operation of the training reactor AKR beyond 2005	SMWK
09/1998	Safety report (version 1) and other documents according to the Nuclear Licensing Procedure Ordinance; start of licensing procedure	TUD, SMUL
11/2001	First steps in design layout planning	NUKEM
02/2002	Design layout planning completed	NUKEM
03/2002	Further detailed planning of licensing procedure	NUKEM
05/2002	Design layout planning for I&C completed	AREVA
08/2002	Safety report (version 2)	TUD
10/2002	Release of financial support	SMF
05/2003	Assessment of the design layout planning with positive result	TUEV
08/2003	Start of planning of the engineering project	NUKEM
09/2003	Contract for the I&C project	AREVA
11/2003	Planning of the engineering project completed	NUKEM
12/2003	Planning of the engineering project for I&C completed	AREVA
02/2004	Preliminary operating and testing manuals (version 1)	TUD
02/2004	Decommissioning of AKR-1; unloading of nuclear fuel	TUD
03/2004	Atomic licence for commissioning and operation of AKR-2 on basis of §7(1) of the German Atomic Energy Act	SMUL
04/2004	Shop acceptance test of the neutron measuring channels	MGPI
04/2004	Begin of civil work for reactor refurbishment	
08/2004	Shop acceptance test of the I&C system	AREVA
09/2004	Installation of I&C systems at AKR reactor site	AREVA
12/2004	End of civil work and electrical and I&C equipment installation	
12/2004	Petition for approval of commissioning of the refurbished AKR-2 to regulatory authority	TUD
01/2005	Release of technical reactor documentation	NUKEM
01/2005	Revision of testing manual (version 2)	TUD
03/2005	Revision of operating manual (version 2)	TUD
03/2005	Final report on reactor construction to regulatory authority (16 March)	TUD
03/2005	Approval of nuclear commissioning procedure of AKR-2 (18 March)	SMUL
03/2005	Nuclear commissioning (22–24 March)	
04/2005	Final report on nuclear commissioning (1 April)	TUD
04/2005	Approval of specified normal operation of AKR-2 (7 April)	SMUL
04/2005	Start of specified normal operation (8 April)	TUD
07/2005	Official scientific commencement operation of the AKR-2	

SMWK: Saxonian State Ministry of Science and Culture
SMUL: Saxonian State Ministry of Environment and Agriculture (regulatory authority)
SMF: Saxonian State Ministry of Finances
NUKEM: RWE NUKEM GmbH (at time of reactor refurbishment)
AREVA: AREVA NP GmbH (at time of reactor refurbishment FRAMATOME ANP GmbH)
MGPI: MGP Instruments GmbH
TUEV: Technical Inspection Agency GmbH

REFERENCES

- [1] HANSEN, W., KNORR, J., “The AKR training reactor of the University of Technology Dresden and its experimental programme for education”, Utilization Related Design Features Of Research Reactors: A Compendium, IAEA Technical Reports Series No. 455, IAEA, Vienna (2007) 553–566.
- [2] HURTADO, A., HANSEN, W., DRUSCHEL, R., HEYER, J., Cooperation Agreement between the Technical University Dresden and AREVA NP GmbH, Proc. of the Nucl. Eng. Sci. and Technol. Conf. (NESTet 2008), Budapest, Hungary (2008).
- [3] BIEHL, T., et al., Compact, low-cost reactor emphasizes safety, Nucleonics **14** (1956) 100.
- [4] HILDENBRAND, G., HÖHNE, P., Der Siemens-Unterrichtsreaktor SUR und seine Verwendung im Rahmen kerntechnischer Praktika, KERNTECHNIK **4** 4 (1962) 141.

THE FIRST EUROPEAN FOCUSSED COLD NEUTRON SOURCE – OPERATIONAL EXPERIENCE AND NEUTRONICS RESULTS

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Abstract

A cold neutron source is one of the most important components of a research reactor. For this reason, the GKSS installed a cold neutron source (CNS) at the FRG-1 in 1988. Around 60% of all neutron scattering instrumentations at the facility use cold neutrons. The principal component of the CNS is the disc-shaped moderator chamber. The moderator is supercritical gaseous hydrogen. In order to increase the cold neutron yield, a study was made for a new moderator chamber layout in 2003. The new fundamental design of the moderator chamber is based on a hemispherical shape, thereby increasing the cold neutron flux by approximately 60% with the use of focussing effects. The study of all relevant parameters was done by AREVA NP in early 2006. The licensing procedure, the fabrication, exchange of the moderator chamber, installation, and testing programme took place from May 2006–June 2007 with the participation of independent experts.

1. INTRODUCTION

Long wavelength (cold) neutrons with high intensity are an indispensable probe for the study of the microstructure and dynamics of condensed matter. Cold neutrons are necessary for macroscopic characterization in basic and applied areas of biological, polymer, and materials research. With the current CNS, the number of long wavelength neutrons with wavelength > 0.4 nm were increased by more than a factor of 20 compared to the thermal flux. To increase the cold neutron flux, which fed more than half of the neutron scattering experiments, the moderator chamber of an existing spare unit needed to be replaced. A model of the new layout was the focusing moderator chambers of the American research reactors at the University of Missouri (MURR) and Oak Ridge National Laboratories (ORNL). These new moderator chambers resulted in gain factors between 50–150%. The following conditions formed the basis for the design and licensing procedure of the GKSS moderator chamber:

- simple design (hemispherical shape) and fabrication;
- the same material specification for the new moderator chamber as for the existing one;
- the same technical inspection as for the existing one;
- the same incident conditions (pressure, melting etc.) as for the existing one;
- comparable nuclear heating for the new and existing chamber.

Consideration of these conditions led to a brisk licensing period of only 4 months.

2. OPTIMIZATION STUDIES OF THE NEW MODERATOR CHAMBER

The FRG-1 research reactor is operated with a reactor core of 12 fuel elements in a 3×4 matrix arrangement. On three sides, the core is surrounded by beryllium reflector

elements; the fourth side faces a block reflector of beryllium with several holes containing the tops of the azimuthally-arranged beam tubes SR6 through SR9.

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

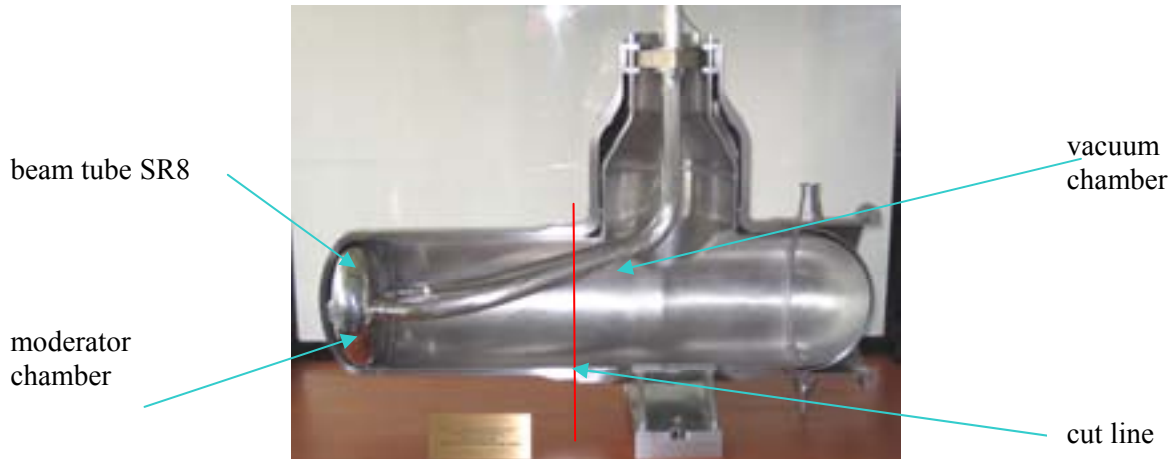


FIG. 1. Section of cold neutron source FRG-1, longitudinal cut through a prototype, located inside beam tube SR8 just a few millimetres from the core.

The moderator chamber is part of a CNS system operated with supercritical hydrogen at about 25 K and a pressure of 1 500 000 Pa. The hydrogen serves as a moderator for the thermal neutrons and as a coolant for the heat transport to the cryogenic helium refrigerator outside the reactor pool. The surrounding vacuum chamber provides good thermal insulation to the beam tube and the reactor pool. The advantage of this medium at these operating conditions is that it is always gaseous but with a density of about 90% that of liquid hydrogen.

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

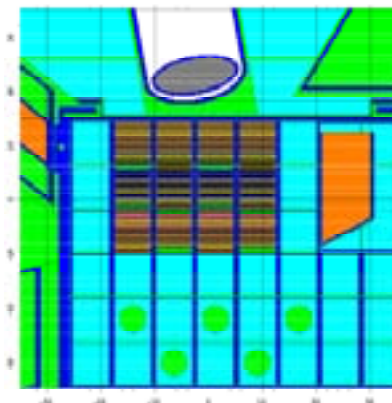


FIG. 2. Cross-section through core, reflector, and beam tubes.

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

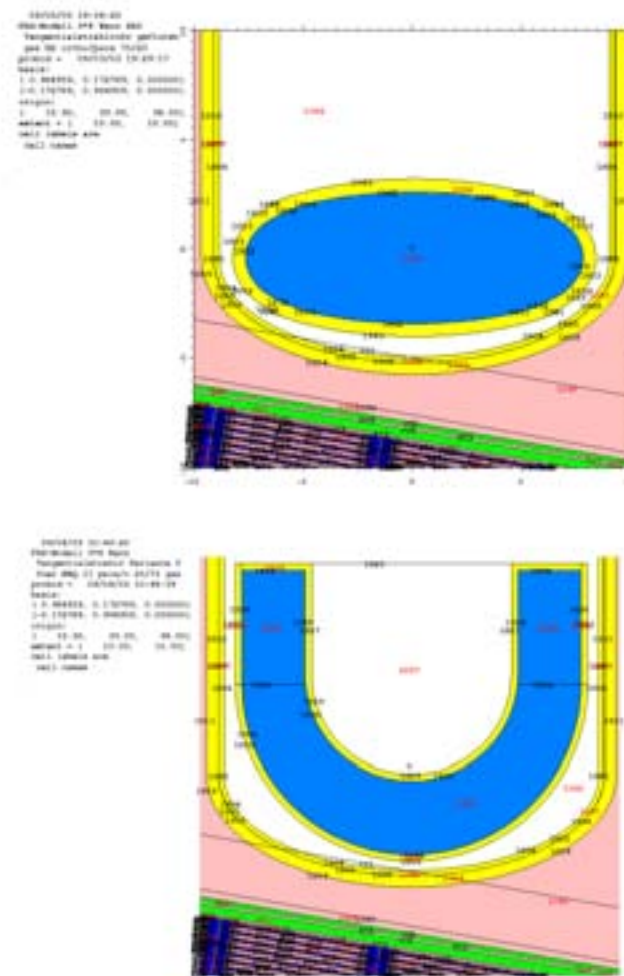


FIG. 3. MCNP model of moderator chamber for reference design (top figure) and optimised design.

For optimization of the moderator chamber geometry, a sequence of calculations was performed with MCNP for one reference burnup configuration by varying the moderator thickness and the length of the cylindrical part. The assessment of the results and the selection of appropriate moderator chamber geometry were made considering only those neutrons that had a chance to pass the neutron guide and to reach the experimental set-up outside the reactor pool. As a characteristic result, Figure 4 presents the calculated mean gain factors for all neutrons in the range of interest comparing both types of moderator chambers.

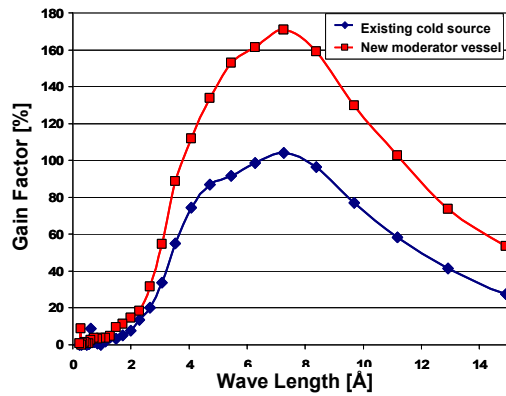


FIG. 4. Calculated gain factor.



FIG. 5. Installation of the new chamber.

3. FABRICATION AND INSTALLATION

The plan for the replacement of the old moderator chamber with the new one in the AREVA NP workshops in Erlangen was as follows (main steps):

- Cutting off the top of beam tube SR8, vacuum chamber and hydrogen pipes at the position indicated by the red line in Figure 2 (the radial gap between the beam tube and the vacuum chamber was only 0.15 mm).
- Fabrication and installation of new moderator chamber (in progress at the end of January 2007).
- Re-installation of tops of vacuum chamber and beam tube SR8 respecting the original dimensional requirements (Fig. 5).
- X rays of welds and pressure tests (end of April 2007).

After transport of the new in-pile part to the FRG-1, the installation of the in-pile part was done by the FRG-1 operations team in May–June 2007 (Fig. 6). An existing work instruction, which was examined during the first installation in 1988, was used for the installation.

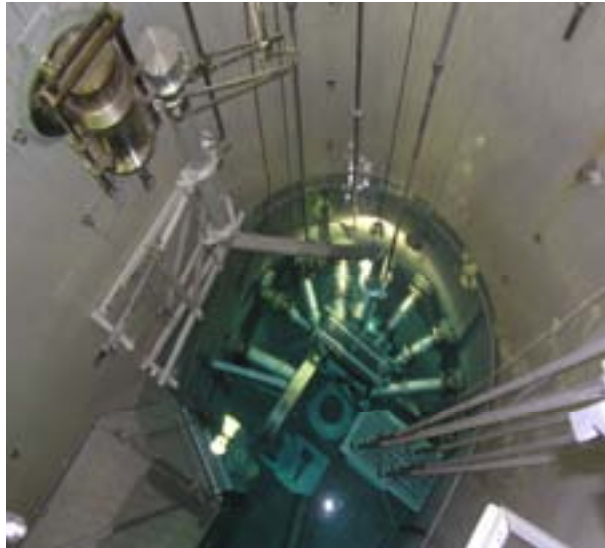


FIG. 6. Installation of the CNS in-pile liner.

4. COMMISSIONING AND VALIDATION

A part of the licensing procedure was the installation of a set in operation programme. This programme contains all the steps from the inspection of the spare unit before beginning of the work to operation of the CNS during full reactor power. After the installation of the in-pile part in the reactor pool, warm/cold leak tests were performed before hydrogen was filled into the plant. The operating parameters (cooling power) of the CNS with the new focussing moderator chamber were then determined by means of a heater in the helium refrigerator. The most important proof of the CNS was the determination of the operating parameters during reactor operation. For this test, the reactor was operated in different power ranges. These tests were successfully accomplished for the two operating conditions (standby operation $T = -35^{\circ}\text{C}$; normal operation $T = 25\text{ K}$). The final point of the commissioning programme was the release of the CNS for normal operation. A first measurement of the cold neutron gain factors for the Nero experiment yielded approximately 40% more cold neutrons, which was in good agreement with the MCNP calculations (Fig. 7).

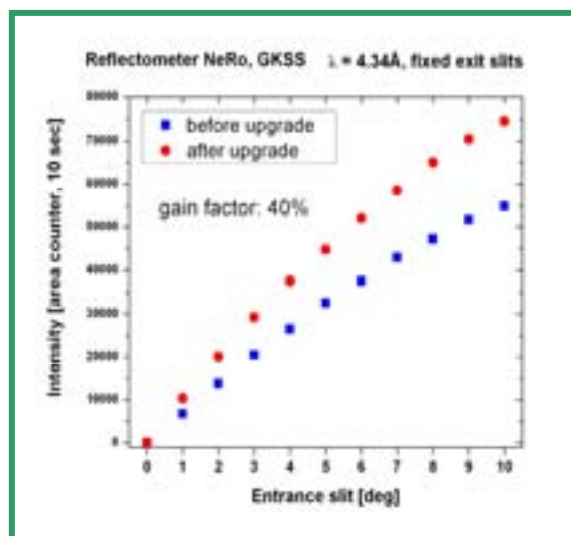


FIG. 7. Gain measurement at Nero.

5. SUMMARY

The GKSS had already increased the neutron flux with two core compactions and by installation of the first elliptical CNS. The installation of the focussing moderator chamber was a new step to further increase the cold neutron flux. With the additional 60% gain of cold neutrons, the FRG-1 provides a mid-range flux neutron source for use by the national and international community.

REFERENCES

- [1] BRIESMEISTER, J.F., MCNP – A General Monte Carlo N-Particle Transport Code, Los Alamos National Laboratory, Los Alamos, NM USA (2000).
- [2] SELBY, D.L., et al., High Flux Isotope Reactor Cold Neutron Source Reference Design concept, ORNL Report, ORNL/TM-13498, Oak Ridge, TN USA (1999).

FULL-SCALE RECONSTRUCTION AND UPGRADE OF THE BUDAPEST RESEARCH REACTOR

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Abstract

The BRR is a tank-type research reactor, moderated and cooled by light water. The reactor, which went critical in 1959, is of Soviet origin. The initial power was 2 MW(th). The first upgrade took place in 1967 when the power was increased from 2 MW to 5 MW, using a new type of fuel and a beryllium reflector. A full-scale reactor reconstruction and upgrade began in 1986. The upgraded 10 MW reactor received an operating licence in November 1993. Since that time, the reactor has been operating an average ≈ 3500 hours/year without any significant problems. In the early 1980s, about 8–10 years before the end of its 30-year service life, discussions began about whether to extend the life of the reactor or begin preparations for final shutdown. It was the consensus that further reactor operations were needed and the government made the decision in 1993 to go ahead with an upgrade and reconstruction. The design work started in 1984, while the reconstruction project began in 1986. According to the modernization plan, a partial decommissioning preceded the modernization, and during this time, with the exception of the civil engineering construction, all equipment was replaced. The reconstruction was essentially finished by the end of 1990; however, due to political changes in the country, the institute could only apply for a licence for reactor startup in 1992. In 1992, a consortium, namely Budapest Neutron Centre (BNC), was founded by four academic institutes to coordinate the reactor utilization and ensure a scientific background for managing the utilization strategy. With guidance from the BNC, the experimental facilities around the beam ports were put into operation continuously within 2–3 years of reactor startup, but investment in several other facilities was postponed for several years. One of the postponed experimental facilities was the cold neutron source (CNS), which was eventually commissioned in 2001. The installation of a multiframe time-of-flight (TOF) diffractometer was completed in 2004. Currently, 12 research facilities are in operation. A review of the last 15–20-year period with respect to reactor reconstruction milestones, including preparation phases for restarting regular operation and the launch of various experimental facilities, highlights many experiences. These experiences and lessons learned may prove useful to members of the research reactor community if faced with the dilemma of choosing between renewal or irreversible degradation.

1. FACILITY BACKGROUND

The Budapest Research Reactor (BRR) is operated by the Hungarian Academy of Sciences KFKI Atomic Energy Institute (AEKI). The BRR was built in 1957–1959 to a WWR-S design standard. It is a tank-type reactor with light water moderation and cooling. The reactor went critical in March 1959. In 1967, the original power of 2 MW was increased to 5 MW with a change of fuel type from EK-10 to WWR-SM. The core was also modified by the installation of a solid beryllium reflector surrounding the core. After this first upgrade, the reactor remained in operation until 1986 when it was a shutdown for a second upgrade.

During its first 27 years of operation, the reactor was used for neutron scattering, radiochemistry, shielding investigations, and radioisotope production. More importantly however, was its mission to establish and maintain an active nuclear research culture in the country. Table 1 provides operational record and pertinent data for the BRR.

2. RECONSTRUCTION AND UPGRADE SCOPE

The first study of the development goals of the BRR was completed in 1974 by the Hungarian Academy of Sciences. Following this, several feasibility studies were conducted by different organizations from the Hungarian and foreign scientific communities, including

industrial representatives and the Kurtsatov Institute in Moscow. In these studies, the utilization demands were considered primarily from the perspective of fundamental and applied research, but also stressed the importance of education, training, and even industrial applications. Although these studies were mostly undertaken from a scientific perspective, the technical possibilities were also considered.

TABLE 1. THE MAIN DATA AND OPERATIONAL RECORD OF THE BRR

Reactor type:	Light-water cooled and moderated tank-type reactor with beryllium reflector (the beryllium has been used since the time of 1st upgrade)
Physical start up:	25 March 1959
Fuel assembly:	EK-10 then WWR-SM after the 1st upgrade
Thermal power:	2 MW then 5 MW after the 1st upgrade
Shutdown for upgrade:	29 March 1967
1st upgrade:	Partial upgrade
Physical start up after 1st upgrade:	4 September 1967
Shutdown for reconstruction:	9 May 1986
❖ Operation record	
Average annual operation:	2 480 on 2 MW; 3 230 on 5 MW
Total operation time:	8 322 hours
Total MW-days:	12 647 MWday
Shutdown for upgrade:	9 May 1986

2.1. Strategic arguments for reconstruction and upgrade

As a result of the preliminary studies mentioned above, a general consensus was reached for further reactor operations and even a demand to increase the flux density. The strategic arguments for the necessity for further reactor operations and upgrades were grouped around four issues as follows:

- 1) **Radioisotope production:** The reactor should ensure future isotope production for medical and industrial application. The importance of the production of short-lifetime isotopes was emphasized.
- 2) **Basic and applied research:** Research activities in the field of condensed matter, materials science, activation analysis, radiochemistry, nuclear gamma spectroscopy, reactor safety, health physics, etc. would be highly desirable to continue and/or extend or, in some cases, start.
- 3) **Technological and commercial applications:** Demands for investigations relating to nuclear reactions based on neutron, neutron-induced embrittlement studies, and surveillance of a power plant's pressure vessels can only be met through the use of a research reactor. In addition, there are industrial and commercial applications, such as silicon doping, development of nuclear instrumentation, tests and certification for industry by neutron and gamma radiography, etc. These promote prosperous service applications for the reactor.
- 4) **Education and training:** Contribution to university and postgraduate education, training for nuclear engineering, and hosting international training courses (e.g. at the request and with the participation of the IAEA) are all duties of the reactor and an essential source of political interest.

2.2. Decision for reconstruction and upgrade

The first development concept [1] was formed in 1981 by the leading Hungarian power plant design company, ERÓTERV. The concept was based on new trends in nuclear research and applications, as well as modern reactor safety requirements including IAEA recommendations (e.g. as defence-in-depth). The concept took into account all feasible and available possibilities, such as fuel questions, core configuration with increased irradiation facilities, technical capabilities including utilization issues, as well as future operation and maintenance questions. The concept included a real cost–benefit analysis (comparable to a modern day SWOT analysis) and defined the design basis and design requirements including safety issues. Finally, on the basis of the considerations listed above, the reconstruction and upgrade project was drafted including an outline of the project schedule and the required resources, including financial and human demands. Due to the general consensus and the well-founded strategic arguments for the necessity of further reactor operation, the government made the decision in 1983 to proceed with the reconstruction and upgrade of the BRR.

3. RECONSTRUCTION AND UPGRADE PROJECT

After the political decision, the budget calculations and project scheduling began immediately. The technical design (conceptual, technical and workshop design) and licensing work were started in 1984. The implementation of the project began in 1986, following 27 years of operation since initial reactor criticality.

3.1. Budget and project schedule

A significant part of the investment grant was secured from governmental sources since the project was considered capital state investment. To obtain this nomination, which ensured a high priority for the project, was due in many respects to the strong support of the IAEA. However, domestic and foreign scientific public opinions also helped the project get a green light. Although the greatest part of the investment costs were covered by the governmental budget, the IAEA, in addition to continued moral support, contributed financially with direct procurement of beryllium elements, primary pumps, valves built in the primary loop and hot cell facilities (manipulators and associated accessories).

The financial resources covered the reactor reconstruction and reactor upgrade costs in all respects. The reactor technically was made according to the design. All reactor systems and auxiliary units were installed. However, it became obvious at the very beginning that the investment of some experimental facilities should be postponed to avoid delaying the project start or (more likely) the reactor restart.

Due to careful preparation and financial planning, the project was performed according to plan from the design stage right through to the implementation. Originally, the shutdown was estimated to take place by mid-1985. However, the design and licensing was delayed for one year; thus, the partial shutdown was performed in May 1986. The project implementation, together with partial decommissioning, was scheduled for a 24-month period. The reconstruction was finished (in the technical sense) by the end of 1990, but due to political changes in the country, the reorganization of the institute, and other non-technical considerations, the AEKI could only apply for a licence for reactor startup in 1992, after a 2-year period of uncertainty. During these 2 years of indecision, the reactor was essentially ownerless and the reactor came very close to being dismantled (this possibility was considered to be a real alternative).

3.2. Project management and human factors

It should be mentioned that good management is a key factor in a successful project. This was the roll of the reactor staff during the reconstruction. Although the reconstruction had a general contractor for design and implementation, the project was supervised and guided by the close cooperation of the institute director and the reactor manager. The specific design of the reactor (core design, thermohydraulic calculations, etc.) and safety assessments were also made by the nuclear physicist and engineers of the institute and these calculations were validated by experts of the Kurtchatov Institute. The panel reviews of the technical designs were also fulfilled by the reactor experts of the Institute.

While the implementation was the contractor's task, the majority of the work in the partial decommissioning was entirely carried out by the reactor staff. Nevertheless, the quality controls, factory tests, the assembling and testing of subsystems, general functional tests, verification and validation procedures, as well as the setting the baseline data (BLD: initial parameters as '0'-status) were done with the active participation of reactor staff as well.

It should be mentioned that all tasks undertaken by the reactor staff were managed with available manpower and there was no need for additional help. It should also be mentioned that during the 2-year period of uncertainty, several well-trained experts were hired by multinational companies coming into the country and left the reactor. In some cases, it was many years before their replacements could be found.

3.3. Technical and safety features of the reconstructed reactor

In our case, it was very important that the calculations and the outside expertise (Rossendorf, St. Petersburg) proved the biological shield of the reactor had sufficient strength to withstand the increased power. In order to satisfy the safety criteria, essential technical measurements were made during the reconstruction. These steps resulted in a significant increase in the safety of the facility. However, additional measurements have been made as a consequence of reactor development. On the basis of the PSAR [2], these features can be summarized as follows:

- Core. The core is built of WWR-SM fuel assemblies. The number of fuel elements was increased in order to increase the power, but to further increase the flux, the power produced by a single element was increased as well. This required more intensive cooling of the assemblies and was ensured by the adequate design of the primary cooling system (pipes, pumps, heat exchangers, etc.)
- Safety and control rods. The increased number of irradiation places, where the neutron flux is relatively high, resulted in a rather complicated core. To ensure a suitably long refuelling cycle, the number of control rods had to be increased as well. Whereas before 2 safety rods were sufficient, 3 rods are now built in to guarantee a safe shutdown.
- Measurement and control system. As the core became more sophisticated, control of the operational parameters became more important. As a result, the measurement and control system had to be more reliable. The measurements are generally redundant (triplicate) and the safety and warning signals of the nuclear instrumentation are evaluated using a majority vote (2/3).
- Material of vessel and primary loop. One of the most important safety criteria is to avoid loss of coolant accident (LOCA) failures. Thus, the materials of the primary loop were very carefully designed. The material of the vessel and grid is an aluminium alloy named R-AlMg_{2.5}, which is a modified version of the 5052 alloy. The material

composition is the same but the aluminium base material is purer. Pipe material is 18/8 type stabilized austenitic steel with nuclear grade certification.

- Safety systems. Two battery stations and two diesel generators were installed to ensure UPS for the emergency system. The safety logic is up-to-date and highly reliable. The fail-safe principle is realized in the actuation of the safety rods. Regarding emergency cooling, a passive cooling system (gravity tank) was constructed that provides core cooling for one minute after the loss of electric power. In addition to the make-up water system, which was also improved, an emergency water feedback system was constructed to collect and feed back the water from the reactor well and pump room in case of LOCA type incidents. In addition, a sprinkler system has been installed to cool the core in the case of extensive LOCA failure.
- Secondary cooling system. The previous open secondary circuit was replaced by a new, closed system, which contains two cooling towers. The secondary circuit is designed for 20 MW.
- Systems and measures to avoid radioactive release. The sealing of the reactor hall was strengthened to form a quasi-containment area. In addition, a recirculation ventilation system was commissioned that could be activated if the limit of radioactive release were to be violated.
- Nuclear spent fuel (NSF) storage. A new AR-pool was installed with a high-density grid with boron carbide absorbers. Also, new storage pipes were installed in the AFR-pool, which could be hermetically sealed.
- Waste management. Two new storage tanks with 150 m³ capacity each were installed with a renewed drainage network. The existing four storage wells were also renewed to ensure safe temporary storage for the collected solid waste.

3.4. Licensing-related analyses and obligatory documentations

Due to the three-level design procedure (conceptual, technical, and working designs) and the applied quality management and control system, the project has been well-documented and the certificates and records for demonstrating compliance with the requirements are settled and retained. The test records of commissioning procedures, as well as setting the so-called '0'-status of the systems are also retained. Prior to investment, SAR (for licensing the shutdown and reconstruction), preliminary SAR (prior to reactor startup for licensing the physical startup), then final SAR (for operations licensing after physical and energetic startup) were elaborated according to the status of the reconstruction project. According to nuclear safety regulations, the most important documents for reactor startup and continued operation were also elaborated.

Of course, the set of documents that served the licensing procedure on the basis of operational experiences were reviewed. In terms of the results of the periodic safety review (PSR), which was conducted in 2002–2003, it can be stated that the obligatory documents exist, that they are generally up-to-date and have been improved by experience.

3.5. Utilization

In 1992, during the period of uncertainty, a consortium, the Budapest Neutron Centre (BNC), was founded by four academic institutes to coordinate the reactor utilization and ensure a scientific background for managing the utilization strategy. With guidance from the BNC, the experimental facilities around the beam ports were put into continuous operation within the first 2–3 years after reactor startup. Later on, as the budget was guaranteed and/or available, the postponed experimental facilities were also put into operation. One of the

postponed utilizations was the cold neutron source (CNS), which was eventually commissioned in 2001 with the strong financial support of the IAEA and the EC. Within the framework of the utilization programme, a hot cell rehabilitation project was completed in 2003 with the financial support of a Hungarian government grant. The installation of a multi-frame time-of-flight (TOF) diffractometer was completed in 2004 with the sponsorship of the Hahn-Meitner Institute Berlin and the institutes of the BNC.

4. BRR AFTER THE RECONSTRUCTION

The newly independent AEKI applied for licensing in 1992. After receiving the licence for the first period, the reactor startup procedure began in 1992. The reactor reached first criticality on 12 December 1992. The licensing and testing period took almost a year (physical startup, zero power measurements, approaching nominal power, and at nominal power). The licence for regular operations at nominal power was issued on 25 November 1993, with no restrictions. Regular operations began immediately on 26 November 1993. An aerial view of the site is shown in Figure 1, the reactor hall is shown in Figure 2, and the control room can be seen in Figure 3. Table 2 provides the main technical data of the BRR after reconstruction.

TABLE 2. MAIN TECHNICAL DATA OF THE BRR AFTER RECONSTRUCTION

Vessel:	Aluminium alloy (height: 5 685 mm; \varnothing 2 300 mm)
Fuel:	WWR-SM(-M2); initial enrichment: 36% ^{235}U ; Average burnup: 60–65%
Core geometry:	Hexagonal (height: 600 mm; \varnothing 1 000 mm)
Equilibrium core:	227 fuel assemblies (in single equivalent)
Control:	• 3 safety rods (B4C); • 14 shim rods (B4C); • 1 automatic (fine) rod (SS)
Nominal thermal power:	10 MW
Mean power density in the core:	61.2 kW/L
Neutron flux density in the core:	• $2.5 \times 10^{14} \text{ n}\cdot\text{cm}^{-2}\cdot\text{s}^{-1}$ (thermal in the flux trap) • $1 \times 10^{14} \text{ n}\cdot\text{cm}^{-2}\cdot\text{s}^{-1}$ (approximate maximum fast flux in the fast channel)
Cooling system:	Two closed loops (primary and secondary loop)
Primary loop:	• Q_{nominal} : 1 650 m ³ /h • T_{inlet} : 45°C; • T_{outlet} : 50°C
Secondary loop:	• Q_{nominal} : 1 850 m ³ /h • Stack air flow rate: \approx 60 000 m ³ /h
Ventilation system:	• Pressure in technological rooms: gradually decreasing (-5 – -25 mm wh)
Experimental channels:	• Vertical irradiation places: \approx 40 channels • Horizontal irradiation beam ports: 8 radial & 2 tangential



FIG. 1: Aerial view of the BRR site. 1) Reactor building with the control room being on the 2nd floor. 2) Reactor hall. 3) Engine room with air filters of the air ventilation system. 4) Secondary pump room. 5) Cooling towers. 6) Auxiliary building accommodating the diesel generators and compressed air system. 7) Building of the liquid waste storage facilities (under the foreground area two storage tanks are accommodated). 8) AFR-pool (with solid waste storage wells in the foreground area). 9) Cold neutron source (CNS) measuring hall. 10) Cryo-system of CNS. 11) Time-of-flight measuring hall.



FIG. 2. Panoramic view of reactor hall.



FIG. 3. Control Room.

4.1. Operation record of the BRR

Since startup, the upgraded reactor has been operating on average $\approx 3\,500$ hours/year without any significant problem. The operation time record (scheduled and performed) is displayed in Figure 4, while the operation cycles performed in 2005 are shown in Figure 5.



FIG. 4. Operation time record.

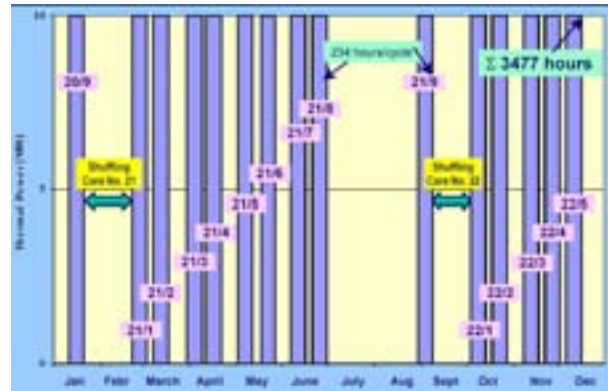


FIG. 5. Operation cycles in 2005.

In comparing the yearly operation data (see Fig. 4), it can be seen that the actual operations were close to the scheduled plan (coincidence $>93\%$). From the restart in 1993, the BRR fulfilled 22 refuelling cycles (campaigns), as indicated on the graphs of Figure 5 (the numbering on the graphs means: Number of campaign/Number of operation cycle). It can be seen from the Figure that the length of a typical operation cycle is 234 hours at 10 MW nominal power and a campaign consists of 9 operation cycles.

4.2. Periodic Safety Review (PSR) in 2002–2003

In line with Hungarian safety regulations [3], a periodic safety review (PSR) was conducted in 2002–2003. As a result, the operation licence was renewed in November 2003 and is valid until further notice. In the course of the PSR, validation test procedures carried out during the reactor upgrade were repeated. These were the most important parts of the validation test procedures of the reactor systems that were carried out during the system installation and commissioning during the reactor upgrade. The latest results were compared with the baseline data recorded 10–12 years ago during the same validation test procedures. Based on the results of these comparisons and the operation data and event audit, the most important findings were as follows [4]:

- There are no significant ageing problems, no unexpected degradation, and no abnormal phenomena appeared on any safety-critical system or component. The degradation is in accordance with the service life of the reactor.
- The 10-year service life of the reactor was safe and the operation passed without any OLC's violation.
- The assumptions of the PSAR are confirmed and its extreme conservatism justified by the experiences of operation and the reassessments of the FSAR.
- The operational environments, including human factors and safety culture appearing in the everyday practices, promote safe and reliable reactor operation.

4.3. Utilization

The BRR is used for various purposes. This includes, among other things, for irradiation and neutron research, the latter being the main utilization (to serve as a neutron source). Irradiations are performed in vertical channels (the reactor has more than 40 channels, including six flux traps that can be used for isotope production and material testing; in one of the channels there is a pneumatic rabbit system that is used for neutron activation analysis). In contrast, experiments are carried out at the horizontal neutron beam ports. The reactor has 10 beam ports (8 radial and 2 tangential) and almost all of them are in use. Currently, 12 research facilities are operating around the beam ports of the BRR. The present layout of the horizontal neutron beam facilities is shown in Figure 6.

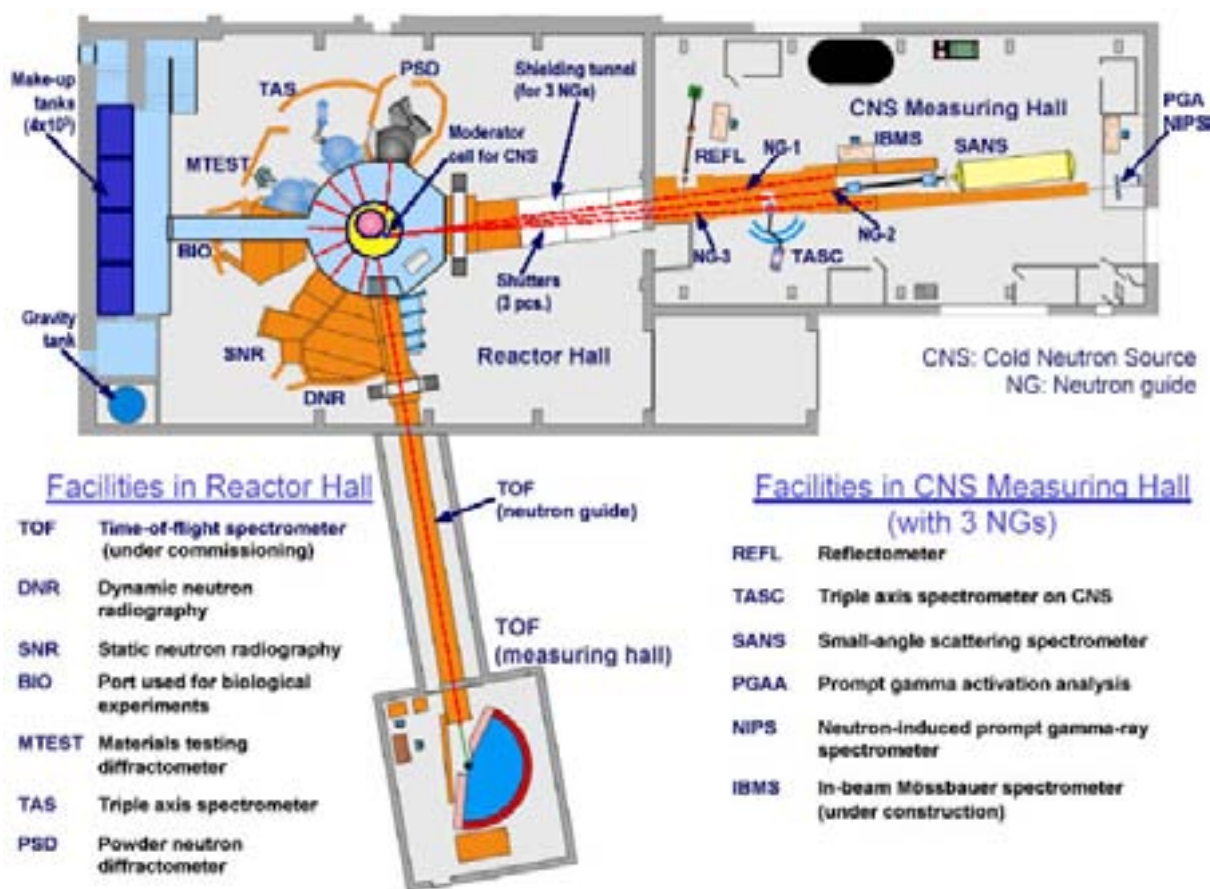


FIG. 6. Layout of the horizontal neutron beam facilities at the BRR.

The utilization of the reactor for basic and applied research is considered to be the primary application of the reactor. In particular, this includes the research fields of condensed matter, radiochemistry, biological irradiations, reactor physics and technology (with the commissioning the CNS, the material research possibilities of the reactor were significantly increased). The reactor is used for isotope production and also, as an industrial application, for neutron radiography and activation analysis.

The BRR's research facilities have been offered to the entire international user community, and in particular, to the EU and associated countries of the European Union in the Access to Research Infrastructures action of the 6th Framework Programme (FP6). It is anticipated the BRR will provide university and postgraduate education opportunities. In

addition, it should be used to provide training for specialists in the nuclear industry and those on international training courses.

5. LESSONS LEARNED

A review of the last 15–20-year period with respect to reactor reconstruction milestones, including preparation phases for restarting regular operation and the launch of various experimental facilities, highlights many experiences. These experiences and lessons learned may prove useful to members of the research reactor community if faced with the dilemma of choosing between renewal and irreversible degradation. The most important conclusions (with lessons learned) can be summarized as follows:

Conclusion #1. Looking back at the discussions to determine the reactor's future, final shutdown and decommissioning as an alternative solution, was really not considered. It could not have happened because the leader representatives of the institute (former KFKI) had an enthusiastic and realistic vision for further reactor operations and utilization. They viewed reactor operations from both the user and the operator perspectives, took the initiative in time and led the discussions throughout the preparation period.

Lesson learned: If the operators and users know what they want and have a realistic vision, the undecided public can be persuaded.

Conclusion #2. The preliminary discussion was begun halfway through the service life of the reactor. The real preparation began with the first official feasibility study, made about 8–10 years before the end of its 30-year service life. Due to this long preparation time, a general consensus was gained and well-founded strategic goals were defined.

Lesson learned: The preparation of the M&R project has to start early to give sufficient time to define good strategic goals with a general consensus of the scientific and industrial public.

Conclusion #3. On the basis of the general consensus and the well-founded utilization arguments, the Hungarian decision makers, as well as the IAEA were persuaded to support the reconstruction. As a result, the IAEA strongly sponsored the reconstruction with continued moral and financial support. In the preparation phase, the moral support of the IAEA was considerable and sufficient to obtain capital state investment status for the project. During the indecisive period, the moral support of the Agency based on the review of the PSAR, helped to win public opinion for reactor startup.

Lesson learned. The financial support of the Agency could be significant, but the moral support is absolutely necessary. The moral sponsorship of the Agency can be considerable and it could strongly maintain the M&R project up to the commissioning.

Conclusion #4. In spite of some budgetary deficiencies, the project was able to start more or less according to schedule and was completed in timely manner. In the case of some experimental facilities, this was due in part, to decisions made during the beginning phase to postpone investments.

Lesson learned. The method of postponed investment (or in other words sequential investment at utilization) can be applied. But it should be highlighted that this method can only be applied for the utilization!

Conclusion #5. The reactor management supervised the project from the beginning to the final stage and behaved like a real owner of the project. The partial decommissioning (that effectively achieved IAEA Stage 3 decommission level) was carried out and comprehensively documented by the staff while the on-site job of the reconstruction (in spite of turnkey investment) was also made with their active participation. The experiences gained from this job form a standalone database and opinion base for further reactor operation (everyday practice has already proven this statement).

Lesson learned. The reactor management has to own the M&R project while the staff should participate in the M&R project. Together they may personalize a proactive (foreseen and anticipated) sense of ownership.

Conclusion #6. Although the BNC was founded to manage the utilization, during the period of uncertainty, it could double the amount of public support for reactor startup. Also, later on, when the consortium put experimental facilities into operation around the reactor and started to manage the utilization strategy, it became obvious that the BNC could successfully represent the user interests, leaving the reactor management to focus on safe reactor operations. Thus, a decision was made to separate the reactor operation issues from the utilization issues. Due to this management system, the reactor manager was responsible for safe reactor operations almost from reactor startup, leaving utilization issues to be managed by the BNC. This management system is now highlighted as being one of the best operational practices.

Lesson learned. Separating the reactor operation issues from the utilization issues can increase the capability to enforce the reactor's interest (two parties could lobby for the same goal). On the other hand, this management solution can also promote safe reactor operations.

6. FUTURE PLANS

The factors determining the future operation of the BRR and likely to provide the M&R with tasks in the future are as follows:

BRR's lifetime. Due to the overall reconstruction and upgrade, the reactor lifetime was reset and is calculated from 1993. The lifetime of the deterministic elements (including the reactor vessel itself) is 30 years. This defines the service deadline of the reactor. Assuming that there will be no technical modernization to extend the lifetime, the deadline is 2023. Apart from the increasing and continued safety measures and modernisation, no life extension plans are currently proposed.

Utilization factor. Considering the short and long term research programmes, international obligations, and contract obligations for irradiation services (isotope production), users have a 40% annual availability factor ($\approx 3\,500$ operation hours in a year). Regarding the strategic issues of the research programmes, it is estimated that the reactor will operate until the end of its service life in 2023.

Technical and human factors. The PSR certified that there were no significant ageing problems, no unexpected degradation, and that no abnormal phenomena were detected in any safety-critical system or component of the reactor. However, few technical renewals, such as increased safety measures were prescribed by the licensing authority resolution closing the PSR (about 90% of which have already been completed). Regarding the human factor, the ageing of personnel was a real problem a few years ago, but a systematic

programme created after the PSR had led to the employment of 6 young colleagues in the last few years. This programme is expected to continue.

Fuel factors. As a result of a trilateral discussion between the USA, the Russian Federation, and the IAEA, a project to repatriate the Russian origin research reactor fuel was launched in 2004. The project, namely Russian Research Reactor Fuel Return Programme (RRFRP) is supported and coordinated by the US Department of Energy. The AEKI signed a contract for site preparation at the end of 2005. After signing this contract, two projects were started in the same year. The first contract was the site preparation for the transfer of Russian origin HEU SNF from the reactor, while the second contract was the preliminary calculation for core conversion from HEU to LEU. Both projects provide tasks for the next 3-5 years.

Licence factor. Although the operation licence, renewed in 2003, is valid until further notice, the Hungarian legal system requires operators of a nuclear facility to prepare a safety review every ten years. In practice, this means the licence validity has to be confirmed by a repeated PSR in 2013.

Financial factor. The basic annual operation costs are covered by the governmental budget. Other costs are paid for through grants and sponsorships, as well as income from industrial services. These sources together ensure sustainable safe reactor operations including waste management resulting from normal operations, investment in improved safety, and some modernization. The costs of core conversion and the procurement of fresh fuel incur extra charges.

ACKNOWLEDGEMENTS

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REFERENCES

- [1] ERŐTERV, Development Scope of Reactor Reconstruction, Budapest (1981).
- [2] AEKI, Budapest Research Reactor, Safety Report 4: Safety Criteria and Analysis of the Possible Accident Situation, Budapest (1995).
- [3] Hungarian Regulatory Body (ORSZÁGOS ATOMENERGIA HIVATAL), Nuclear Safety Regulation No. 5 (Research Reactors), Budapest, Hungary (1997).
- [4] TÖZSÉR, S., GADÓ, J., KÉSMÁRKY, K., VIDOVSZKY, I., “Periodic safety review of the Budapest Research Reactor” (IAEA-CN-120/4), International Conference on Topical Issues in Nuclear Installation Safety: Continuous Improvement of Nuclear Safety in a Changing World, Beijing, China (October 2004), 186.

PROPOSED MODERNIZATION AND REFURBISHMENT OF INSTRUMENTATION AND CONTROL SYSTEMS OF THE FBTR AND KAMINI REACTORS IN INDIA

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Abstract

The Fast Breeder Test Reactor (FBTR) is a 40 MW(th), loop-type sodium cooled fast reactor built at Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam as a fore-runner to the second stage of the Indian nuclear power programme. The reactor design is based on the French reactor Rapsodie with several modifications that include the provision of a steam-water circuit and a turbogenerator. The FBTR uses sodium both as a coolant and as the main heat transport medium to transfer heat from the reactor core to the feed water in the tertiary loop for producing superheated steam, which drives the turbogenerator. The reactor has completed 20 years of successful operation and many modifications were carried out to improve the performance. The current plan is to revamp the various systems to improve reactor availability. Currently, The FBTR is the only testbed available for our fast reactor programme and acts as an input to the design and development of the next generation fast reactor. Replacement of the computers of the Central Data Processing System (CDPS) of FBTR and Kamini reactors with state of the art embedded system and replacement of the first generation neutronic channels and reactor protection systems is planned. Modifications in the sodium leak detection system for on-line detection of any open sensor; upgrades of the steam dump system, augmentation of the Steam Generator (SG) leak detection system, revamping of the physical protection, tele-alarm, radiation monitoring systems, etc. for life extension and improved availability are planned. Kamini is a 30 kW, ^{235}U -fuelled, demineralised light water moderated and cooled, special purpose research reactor located at IGCAR. The reactor was designed and built jointly by Bhabha Atomic Research Centre (BARC) and IGCAR. The reactor works as a neutron source with a flux of $10^{12} \text{ n}\cdot\text{cm}^{-2}\cdot\text{s}^{-1}$ at the core centre and also facilitates the neutron radiography of both radioactive and non-radioactive objects and neutron activation analysis. This paper details the various measures taken and planned for the modernization and refurbishment of the instrumentation and control systems of the FBTR and Kamini reactors for improved availability and life extension of the plant.

1. FAST BREEDER TEST REACTOR (FBTR)

1.1. Introduction

The Fast Breeder Test Reactor (FBTR) is a 40 MW(th), sodium cooled, PuC-UC fuelled fast reactor, located at Kalpakkam, India. Construction of the reactor started in 1974 and was completed in 1984. The reactor went critical in October 1985 with a Mark I core rated for 10.5 MW(th) at a peak linear heat rating (LHR) of 320 W/cm. The reactor core was progressively enlarged and the turbogenerator was synchronized to the grid in July 1997. The present core has 43 fuel subassemblies rated for 16 MW(th) at a peak LHR of 320 W/cm. At present, the reactor has operated for 38 000 h and has seen 850 effective full power days (EFPD) of operation corresponding to a peak LHR of 320 W/cm. The peak burnup is 1.54 GWd/t with no fuel clad failure. Each of the four sodium pumps has satisfactorily operated for more than 125 000 hours.

Figure 1 gives the flow sheet of the reactor. Heat generated in the reactor is removed by two parallel hydro-dynamically coupled primary sodium loops and transferred to the corresponding secondary sodium loops. Each secondary sodium loop is provided with two once-through steam generator (SG) modules. The steam from the SG in both loops is fed to a common steam water circuit composed of a turbogenerator (TG) and a 100% dump condenser (DC). Generally, the turbine is in operation; in the case of its non-availability, reactor operation could be continued by dumping the steam in the DC. Stainless steel (SS316) is the

construction material for the reactor vessel and the primary and secondary sodium circuits. The major design parameters of the reactor are also given.

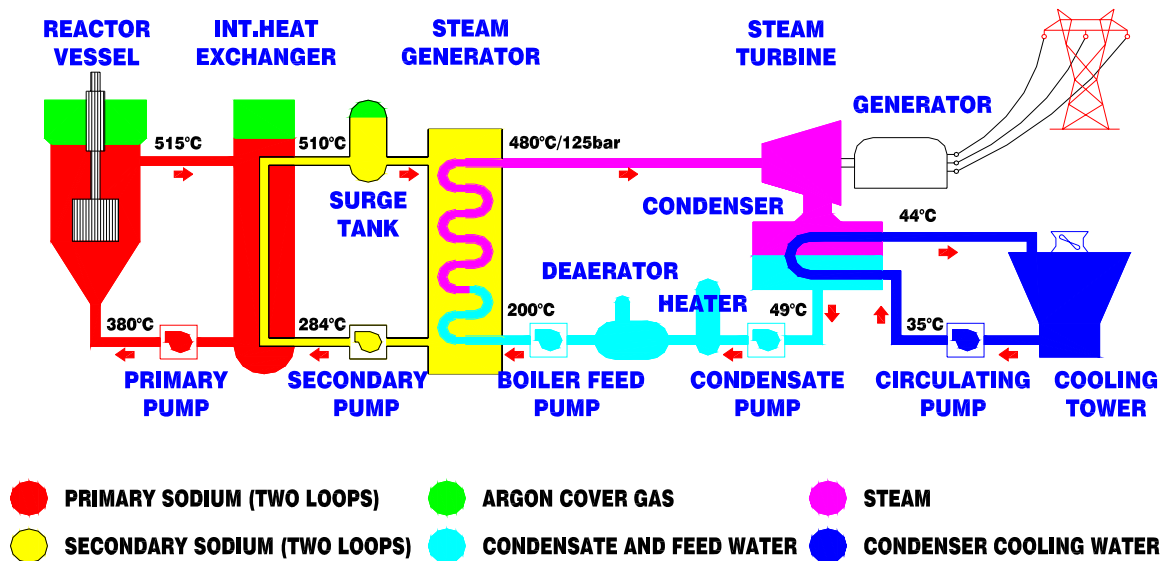


FIG. 1. Flow diagram of the FBTR.

1.2. Main characteristics

Reactor coolant	Sodium
Primary circuit concept	Loop
Thermal power	40 MW(th)
Electrical power	13.2 MW(e)
Fuel	70% PUC 30% UC [MARK I core]
Core height	320 mm
Sodium inlet temperatures	380°C
Sodium outlet temperature	515°C
Sodium inventory	150 t
Steam conditions at turbine inlet	490°C at 167 Bar
Control rods	6 Nos of Boron Carbide
Neutron flux	$3 \times 10^{15} \text{ n}\cdot\text{cm}^{-2}\cdot\text{s}^{-1}$

1.3. Scope of modification and refurbishment

The FBTR was commissioned in 1985. It has been the flagship of the IGCAR and the cradle for the development of fast reactor technology in the country. Fuelled with unique plutonium rich mixed carbide, the reactor performance has continuously improved each year. The FBTR completed its 14th irradiation campaign in October 2005.

As we continue to irradiate fuels and clad materials in the FBTR to gain knowledge of their behaviour, the residual life of the reactor components is also being assessed as a first step in extending the life of the reactor. The seismic qualification of FBTR in line with the current regulatory requirements has been initiated. The FBTR is currently the only test bed available for our fast reactor programme and its operating experience has been a major input in the design of Prototype Fast Breeder Reactor. Future plans of the FBTR include a series of experiments and an advanced fuel irradiation programme to aid the design and development

of the next generation fast reactor. It is also planned to demonstrate the breeding of ^{233}U from thorium in the FBTR. With this project, the life of the FBTR will be extended by fifteen years with improved plant availability to fulfil all these missions.

All the FBTR systems are 25 years old. Life assessment of the FBTR indicates that due to operations so far at lower power and temperature, the nuclear systems still have a residual life of more than 15 effective full power year; therefore, the reactor can be operated for more than twenty calendar years (depending on the availability factor) to fulfil all the intended mission. The aged instrumentation and control systems need to be replaced as part of ageing management of the reactor. Various modernization and refurbishing works have been initiated for improved availability and life extension of the plant.

1.3.1. Major refurbishments completed

1.3.1.1. Neutronic channels replacement

In the FBTR, the measurement of neutron flux ranging from 10^{-1} – 10^{10} nv at the detector location covers the measurement range from shutdown to full power at 40 MW(th). The entire flux range is covered by different neutronic channels (Fig. 2).

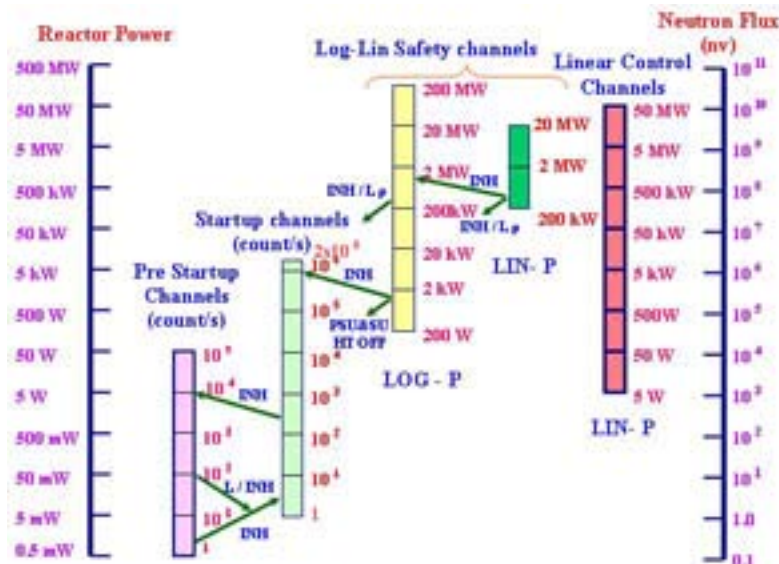


FIG. 2. Operating ranges of neutronic instrumentation.

To overcome the difficulty in monitoring and the obsolescence of components in older neutronic channels, they were replaced with new generation state of the art neutronic channels in 1998. The safety logic system orders safety actions (LOR or Scram) through the reactor protection system whenever any trip parameter exceeds the preset threshold. The trip parameters are derived from diverse and redundant inputs with adequate backup to enhance reliability. Based on operating experience, optimization of the trip parameters included various modifications, additions, and deletions of parameters in the Scram/LOR circuit to avoid spurious trips and achieve maximum availability without compromising safety.

The new neutronic channels were designed and developed with state of the art technology and all the prototype channels were designed, developed, fabricated, and tested at Electronics Division, BARC. The know-how of these channels was subsequently transferred to M/s. Electronics Corporation of India Limited (ECIL) for production and installation at the

FBTR. The new channels were successfully commissioned in June 1998. The salient features incorporated in the new generation channels have helped overcome various problems faced in the earlier channels, such as:

- The noise pick-up was overcome by optimizing the gain and disc bias in the channel along with improved grounding of the signal cable shield (G3) connected to chassis ground (G1) with a pig tail. After this modification, spurious alarms and trips on period due to noise pick-up were reduced to a minimum.
- The channel integration, such as combining different discrete circuits into a single module and bringing the trip circuits, Fine Impulse Test (FIT) circuits, scalar timer into single unit has resulted in reduced interconnections in the cabinets. As a result, spurious faults originating because of loose connections were eliminated. In addition, the 'optical isolation' provided between signal/FIT/SCRAM logic instead of 'pulse transformers' in earlier channels, annulled the FIT faults due to noise pick-up and stretching of pulses.
- All the trip circuit thresholds were provided with buffered output for distribution to subsystems I & II of the real time computer systems. This significantly improved the stability of the thresholds and eliminated spurious discordant alarms due to threshold variations and load impedance changes.
- The modular design of all the controls such as test selection switches, meter selection switches, test output trip jacks, test command switches, trip indications and channel fault indications were brought out on the front panel of the channel. This helps carry out the on-line maintenance with ease and less downtime. The calibration and testing of the channels is also very user friendly compared to the previous channels because there is no requirement for drawing out the chassis, thus avoiding the possibility of disconnecting the cable connectors as in the earlier channels.
- After commissioning of the new channels, very few electronic components failures were noticed during initial operations. From 2001–2006, no component failures in the modules were observed except the failure of a few power supply modules (+15 V). However, these failures were overcome by adequate de-rating of the components with the modified design. After replacing the old channels, the number of reactor trips from neutronic channels was reduced from seven in 1998 (prior to replacement) to zero in 2001.

1.3.1.2. Safety logic systems

The safety logic system consists of solid-state circuits and triplicate channels using 2/3 coincidence logic adopted for neutronic parameters. An on-line test facility (FIT) is provided to check for safe and unsafe failures in the scram logic. During reactor operations, various neutronic and process parameters and status of the systems/equipment are monitored. The safety logic system orders safety actions (LOR or Scram) through the reactor protection system whenever any parameter exceeds the preset threshold or the equipment/system is not available. It is of utmost importance to ensure that the postulated safety limits of the reactor are not reached.

The safety limits of the FBTR relate to the linear heat rating of the fuel, fuel cladding integrity, and primary system boundary integrity that are necessary to prevent fuel melting and also prevent release of radioactive fission products to the environment. The safety limits define an envelope within which the operation of the reactor is safe. These envelope settings of the trip parameters, also called as limiting safety system settings (LSSS), initiate safety actions to prevent reaching the safety limits. The trip parameters are derived from diverse and

redundant inputs with adequate backup to enhance reliability. Based on operating experience, optimization of the trip parameters included various modifications, additions, and deletions of parameters in the Scram/LOR circuit to avoid spurious trips and achieve maximum availability without compromising safety.

1.3.1.3. Addition/Modification of Trip Parameters

- Addition of Scram on high-count rate (Log C) from pre-startup channels after incorporation of these channels to facilitate reactor startup without an auxiliary neutron source.
- Addition of power range period (T_p) Scram to provide protection from 2–660 kW.
- Addition of Scram on minimum flow (Q_{min}) through the core to replace Scram from ratio of power to core flow (P/Q), since the ratio was found to remain unchanged during some incidents.
- Scram on ‘LOR ineffective’ was modified to convert unexecuted LOR order due to any reason into Scram. The LOR was converted to Scram only in the case of non-availability of the power supply to the control rod drive motors as per the original scheme.
- LOR from the entire sodium pump drives and feed water pump trips were backed up by the respective loop low flow signals.
- LOR on SG leak was modified to include diverse signals, namely rupture disc failure and expansion tank cover gas pressure. A signal from actuation of SG safe configuration was added to prevent reactor operation without SG.
- LOR on failure of CDPS was added to prevent reactor operation without operation of safety critical and safety-related programmes.
- LOR on control rod level discordance was added to prevent uncontrolled withdrawal of control rods.
- LOR on low current in EM coils of CRDM was included to avoid partial Scram.
- Scram on negative reactivity ($-\rho$) was added to safeguard against anomalous core reactivity incidents during stable operation along with positive reactivity ($+\rho$) Scram. Manual inhibition facility of reactivity Scram was provided to facilitate fast startup of the reactor.
- To avoid the recurrence of a few reactor trips which occurred due to the failure of the control power supply to the trip circuits of safety logics, active standby power supplies were provided with free wheeling diodes. The station uninterrupted power supply system was replaced with the state of the art system to improve the performance of CDPS and the safety logic.

1.3.2. Major refurbishments just completed or currently in progress

1.3.2.1. Central Data Processing System replacement

Initially, the CDPS in the FBTR consisted of two TDC-316 (III Generation) computers connected in a fault tolerant configuration and commissioned in 1983–1984. They were replaced with SS-I and SS-II systems in 1988 and 1992, respectively. Normally, the SS-I system supervised the plant, while SS-II system was a hot standby. The relay multiplexer cards of the SS-II system frequently failed due to leaky relays and aged capacitors. Also, both hard disks in the system had failed and were replaced with the SCSI–540 MB disks supplied with the computer due to non-availability of spare disks. There were no spares available for

the system as the system was no longer produced. Therefore, to improve the availability of the CDPS, it was decided to replace both systems.

While replacing the SS-II system, the safety critical, safety-related, and non-safety functions were segregated. Thus, the single SS-II system was replaced by three ED-20 based embedded systems as shown in Figure 3. The hardware and software of these systems was developed in-house to avoid obsolescence. These systems are commissioned and working satisfactorily.

The various functional modules running on the ED-20 systems are core temperature supervision, control rod level discordance supervision, steam generator leak detection, general supervision, discordance supervision of neutronic and thermal triplets, startup of reactor and startup of fuel handling conditions supervision, trip supervision, and a communication programme that sends ADC values analogue signals to the data server, human machine interface (HMI), etc.

The salient features of the new ED-20 systems are:

- In-house design.
- No operating system.
- No hard disk.
- Embedded system.
- Asynchronous VME bus-based system that is highly suitable for critical control application.
- Fail-safe design.
- All the field analogue signals are connected to the analogue input cards through 3 port isolators.
- I/O cards use industrial grade components and have undergone environmental testing. I/O cards have built-in diagnostic features.
- IEEE guidelines were followed for the entire software life cycle.

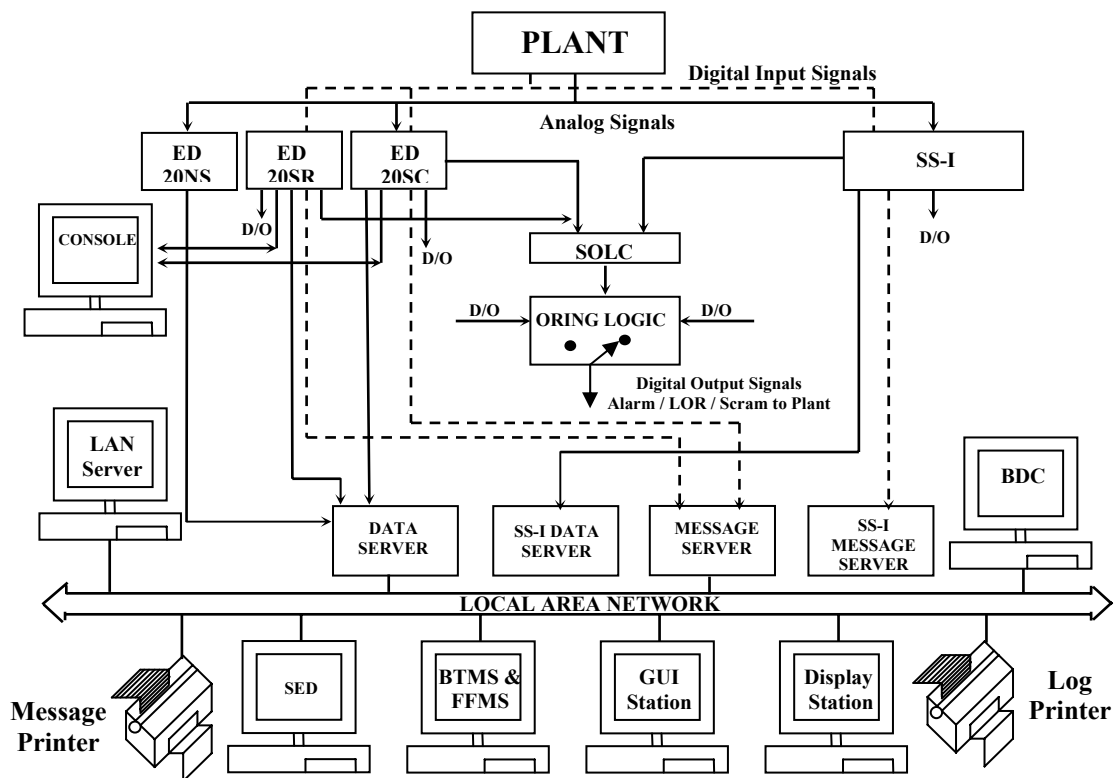


FIG. 3. Architecture of CDPS.

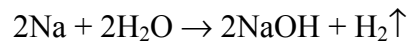
Apart from these two main systems, the following stand alone PC-based data acquisition systems for carrying out dedicated functions were also progressively added. These systems were added for real time logging of various plant parameters and history supervision.

- Sequential event detection system (SED) for scanning important trip contacts at 20 millisecond intervals. This system provides data for analyzing the cause of LOR/Scram.
- Radiation and air activity monitoring system for processing the signals from radiation monitors.
- SG and turbine data logger for logging signals from steam, water and turbine systems.
- Block pile temperature monitoring system (BTMS) is a remote data acquisition system that monitors 192 temperature signals from the block pile system and gives the temperature gradient at various elevations of the reactor vessel. The signals are terminated in the reactor containment building (RCB). Originally, the temperature was measured manually by connecting the signals to the sensor through patch cords. To avoid cabling from the RCB to the CDPS room, remote data acquisition modules are located in the RCB and the computer that scans these values is located in CDPS room.
- This system is also used to find the frictional force of control rods both on-line and off-line. During exercising of control rod on-power, this system gives the frictional force on each control rod to ensure safe shutdown of the plant on demand.

1.3.2.2. Improved steam generator leak detection system

The secondary sodium system of the FBTR has two steam generator (SG) modules connected in parallel in each loop (Fig. 4). The SG module is a once-through type heat exchanger in which sodium flows on the low pressure shell side (2–3 bars) whereas water/steam flows on the tube side at a high pressure (~125 bars). In case of a breach in the tube integrity, water/steam will leak into the sodium resulting in a violent reaction between the sodium and the water. The reaction products are of a corrosive nature and high pressure surges in the shell side of the SG will lead to undesirable consequences. To avoid these consequences, the steam generator leak detection system (SGLDS) is incorporated to detect a water/steam leak into the sodium at the initial stage by on-line measurement of hydrogen in sodium and to isolate the defective loop before the leak escalates. As well, this system serves to gauge the magnitude of the leak.

The leak is detected by monitoring the increase in the hydrogen concentration in sodium due to the dissolution of hydrogen produced during the reaction of sodium and water/steam, which is given by the reaction:



The sodium sample drawn from the outlet of the SG is passed through a nickel diffuser. Around the diffuser, a vacuum chamber is equipped with an ultra high vacuum maintained by a sputter-ion pump with the ion current indicating the equilibrium pressure in the chamber. In case of a leak due to a higher influx of hydrogen into the vacuum chamber, an increase in the equilibrium pressure is detected by an increase in the sputter-ion pump current. With the calibrated data, the magnitude of the leak can also be computed.

Though the SGLDS of the FBTR was commissioned in 1992, many modifications were required to improve the performance over the years. However, there was no provision to confirm the validity of a leak signal and the availability of the system was very dependent on a single system in each loop. To overcome these problems, the systems were duplicated in the first phase in July 2004 by adding an additional nickel diffuser vacuum system in each loop in series with the existing system. However, the safety action on the reactor continued to be on a single channel scheme with the duplicated channel as a manual standby. Hence, it was decided to add a third channel and effect safety action through 2/3 logic.

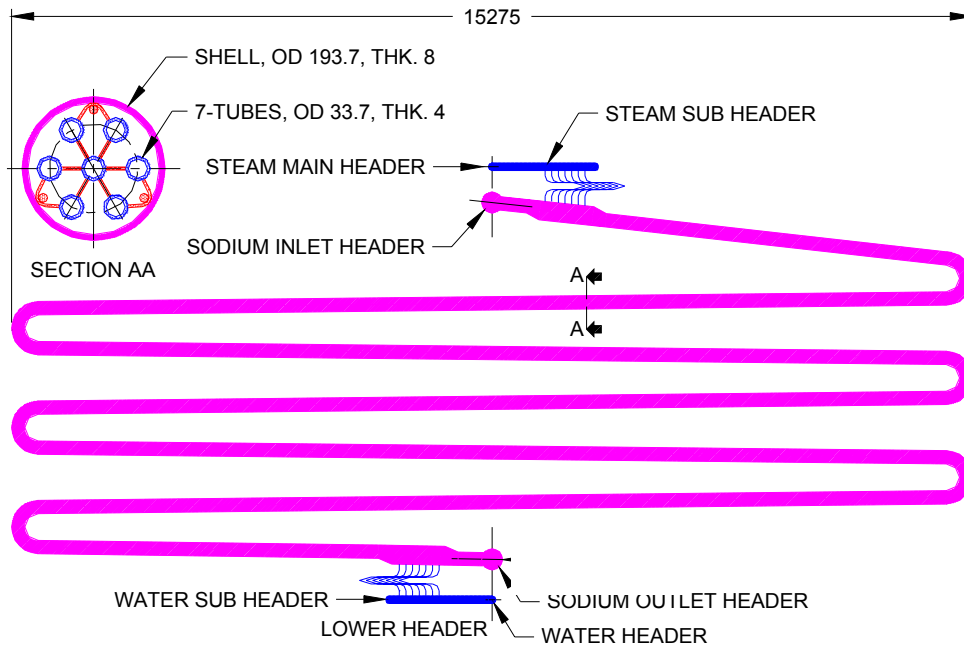


FIG. 4. FBTR steam generator.

The sodium sampling line was tapped off from the common outlet header of both SG modules. Sodium pipelines with an EM flow meter, reheater, nickel diffuser and associated valves were installed and tested as well as the necessary line heaters, sodium leak detection channels, temperature measurement and control channels, etc. An ultra high vacuum chamber consisting of a sputter-ion pump, a sorption pump, etc. was connected in the third channel and associated instrumentation and control logic were incorporated. A 19" panel was procured and installed in the extended portion of SGLDS cabin. The safety logic was modified to incorporate a 2/3 logic scheme to initiate safety actions on the reactor after getting the necessary approval from the safety authorities (Fig. 5). The retrofitting and commissioning of the third channel was completed for the east loop in June–September 2005. Incorporation of a third channel in the west secondary loop was completed in June 2006.

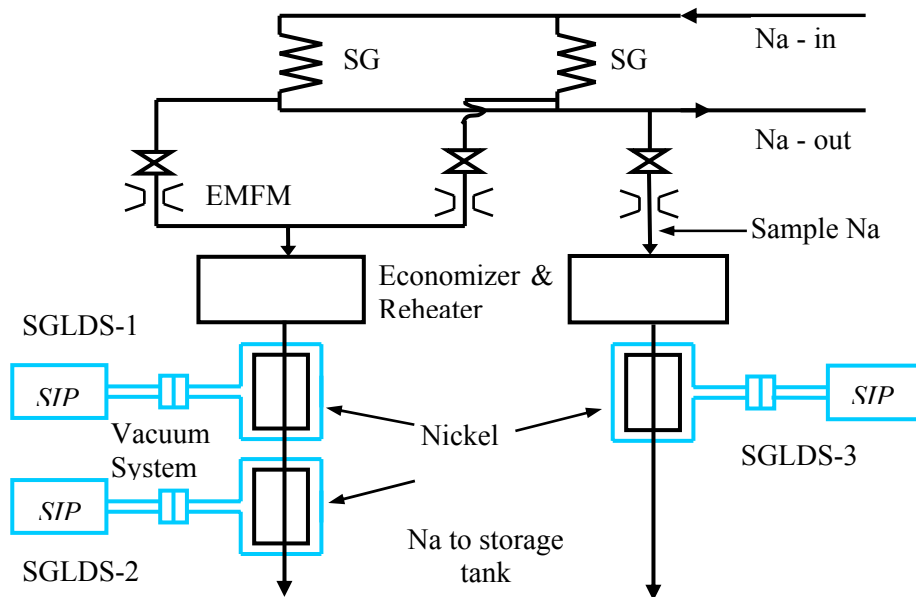


FIG. 5. Triplicataion of SGLDS.

1.3.2.3. Development of ion pump signal for safety action

The problems faced during initial commissioning were successfully resolved through suitable modifications to achieve satisfactory performance. However, the uninterrupted availability of the SGLDS remained a distant goal because the system faced a long downtime during the annual replacement of the quadrupole mass spectrometer (QMS) ion source filament as it required intervention in the vacuum system and subsequent calibration by actual hydrogen injections. In addition to this planned activity, premature failure of the filament forced reactor outages on a few occasions. The performance of the system was systematically studied and it was realized that certain modifications were required. The key to the modifications lay in developing an alternate technique to detect the hydrogen partial pressure (p/p) increase in the ultra high vacuum (UHV) chamber without the QMS. The concept of using the sputter ion pump ionization current, which is already in the circuit instead of the QMS, was explored since the current is a function of the total gas molecules in the UHV chamber.

Implementation of this concept required development of a technique to extract a microampere level current signal from the sputter-ion pump (SIP) that was riding over a very high DC voltage of 3.6 kV with adequate protection against high voltage and converting it into a 0–10 V DC signal. The signal had to be suitable for connecting to the CDPS for monitoring and carrying out safety functions and field experiments to meet the required performance standards.

The main task in designing a circuit to implement this technique was to make it fail-safe against SIP high voltage sneaking into the low voltage signal input to the monitoring CDPS. After designing and testing the circuitry, the prototype was fabricated in collaboration with M/s. ECIL, Hyderabad and extensively tested in a separate experimental set-up to study the performance pertaining to signal stability, noise immunity, repeatability, etc. After obtaining satisfactory results, the unit was installed in the secondary sodium east loop and the performance was observed via the QMS signal by carrying out hydrogen injection experiments at different loop sodium temperatures. The performance was excellent (Fig. 6) and the modification was incorporated after obtaining approval from the safety authorities.

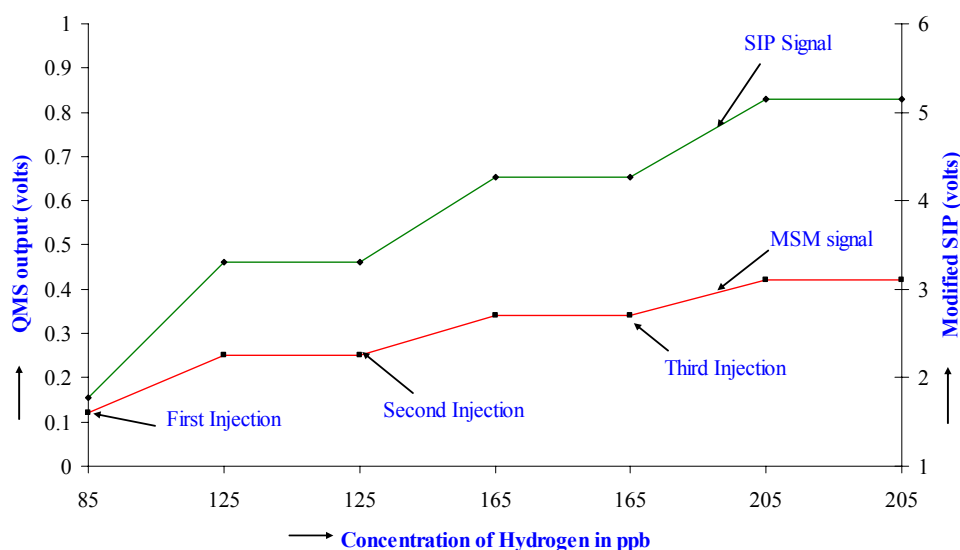


FIG. 6. Comparison of QMS and SIP signals.

These simplifications obviate the need for the QMS in the system with its associated problems of an expensive and yearly ion source filament replacement and calibration leading to large downtime. In addition, this also eliminates the potential for premature filament failure and maintenance of spares inventory, thus saving considerable foreign exchange and the hassles of embargo. This simplification in the system design will be of considerable importance in view of the requirements for an increased number of such systems for future fast breeder power reactors. Along with improved the surveillance method for validating the calibration on-line without requiring plant shut down, this will increase the reliability and availability of the SGLDS, thus increasing plant availability.

1.3.2.4. On-line calibration method

The calibration is essential to establish the relationship between meter output and the concentration of hydrogen in sodium. It is also necessary to periodically calibrate the system (yearly) as the permeability of the nickel membrane and capacity of the sputter-ion pump may change due to ageing. The calibration is carried out when the reactor is either shut down or operating at power level below 500 kW so that the SG is not valved in on the water side. The primary and secondary sodium temperature is maintained at $375 \pm 10^\circ\text{C}$ with the help of the reheater or reactor power. The calibration is achieved by injecting known quantities of hydrogen gas into sodium. For this purpose, an injection line with 20 mm o.d. joins the sodium header upstream of the steam generator. The quantity of hydrogen injected is calculated from the drop in pressure in the hydrogen tank. Signal evolutions in the QMS and SIP are monitored during hydrogen injection and the setpoints for safety actions are derived from the signal evolution observed. Since the calibration is carried out with the SG in drained condition on the water side, the reactor is not available for high power operation until the calibration is completed, which may take about 15 days. The possibility of calibrating the system with the reactor operating at high power with the SG valved in was explored for improving the reactor availability and found satisfactory.

1.3.2.5. Analysis of frequently failed components

The failure data of various systems were analysed in detail and the data of frequently failed components were prepared. The cause of failures was identified wherever possible and actions were taken to improve the reliability and availability of the systems by incorporating suitable modifications. With the improvements and modifications, the availability of these systems is expected to increase remarkably.

1.3.2.6. Modification in the sodium leak detection system

In the FBTR, wire type leak detectors are used to detect a sodium leak from the pipelines. Spark plug type leak detectors are used to detect sodium leaks from sodium capacities and the primary sodium piping which have double envelopes. Both these detectors exploit the high electrical conductivity of sodium.

The wire and spark plug type leak detectors had promptly detected the leak and alerted the operator on a number of occasions, including a medium size leak in the primary purification systems. However, spurious alarms were also observed many times and were mainly due to disturbances to the nickel-wire. Though many failures were in the safe direction, suitable logic modifications are being carried out so that unsafe failures, such as any discontinuity in wiring can be detected on-line. The discontinuity in the sodium leak detector channels can be detected on-line by modifying the logic circuit as shown in Figure 7.

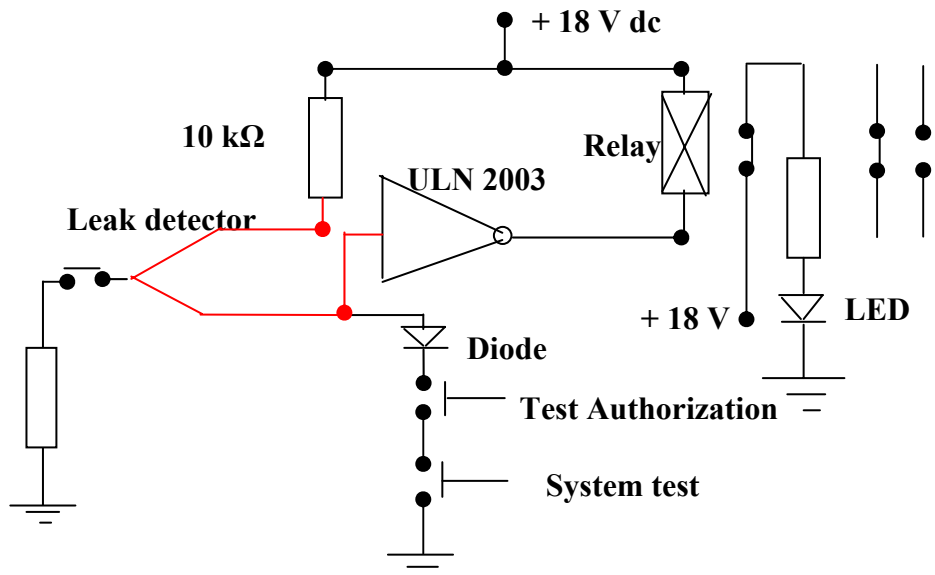


FIG. 7. Block schematic of sodium leak detection-logic circuit (modified).

1.3.2.7. Sodium temperature measurement

The temperature of sodium is measured in many locations by chromel-alumel thermocouples. The sodium temperature at the outlet of the fuel subassemblies in the reactor core plays a vital role in safeguarding the reactor from excessive temperatures that may arise from blockage of sodium flow, etc. The performance of these thermocouples has been satisfactory except for the failure of 1 mm thermocouples without thermowells meant for measuring the central fuel SA outlet temperature. Failure of these thermocouples was suspected to be direct impingement of high-velocity sodium on the bare thermocouples and the resulting vibration. To overcome this problem, it was decided to go for 2 mm o.d. thermocouples instead of 1 mm o.d. thermocouples.

1.3.2.8. Level monitors

There are two types of level sensors employed in the sodium capacities of the FBTR. One provides the level of sodium as a linear signal (continuous level monitors (CLM)) and the other detects whether the sodium level in a capacity is above or below a fixed level (discontinuous level monitors (DLM)). The CLM work on the principle of variation in the mutual inductance between two windings made of stainless steel sheathed copper wire wound on a stainless steel shaper when they are immersed in a conductive fluid like sodium. The DLM is based on the detection of the decrease in the resistance of a probe when short circuited by highly conductive sodium. The performance of the mutual inductance-based CLM is quite satisfactory. The performance of the resistance-type DLM was found to be unsatisfactory on many fronts. During initial commissioning, the appearance of the level signal was delayed in a few probes due to improper wetting of the probe by sodium. Further, spurious alarms were observed in some high level probes due to deposition of sodium aerosols around the probes, which were avoided by suitably modifying the thresholds. In some cases, the signal derived from the CLM is wired as a back-up. Hence, mutual inductance level sensors are also recommended for discrete level measurements.

1.3.2.9. Sodium Pump Drive System

In the past 13 years, about 50 modifications were implemented in this system to bring the performance of the system to the level of excellence. Improvements have been made to maintain the pump speed within 1 rpm. Timer relays were introduced for the Ward-Leonard (W/L) trip on the pump motor field low and armature overcurrent to avoid W/L trips due to spurious actuation of the 24 V relays in the electronic cards. Isolation amplifier-based electronic cards were introduced to monitor the armature voltage and armature current in control room.

The pump overspeed trip circuit, which extracts a small DC millivolt signal from the large DC voltage bus of the armature circuit and processes it through instrumentation amplifiers and optocouplers, were simplified by replacing this signal with a tachogenerator signal through simplified circuitry to achieve better reliability and easy maintenance and calibration. An on-line diagnostic circuit was incorporated to continuously monitor the healthiness of the auto mode speed control windings of the DC Generator to initiate an alarm to alert the operations staff to take timely action to prevent a reactor trip. An air-conditioned environment was provided for all the W/L control panels, which reduced the malfunctioning of electronic cards. Based on the analysis of various tests, 300 Ah and 80 Ah lead-acid tubular batteries were replaced with 150 Ah and 100 Ah Planté type batteries, respectively.

Associated speed control power relays of the sodium drive systems of the primary and secondary sodium pump drive systems were provided with a 330 Ω resistor in parallel with the existing 330 Ω resistor so that the effective resistance becomes 165 Ω . This ensures that the power relays associated with speed control of the sodium pump drives do not fail to pick-up during station blackout conditions. The effectiveness of this modification was confirmed by simulation. The transfer resistance cubicle temperature monitoring system was provided in all electronic panels of the sodium pump drive systems and temperatures above 45°C will be announced in the control room.

1.3.2.10. Experiences with conventional instrumentation

The steam water systems of the FBTR use extensive conventional instrumentation based on transmitters, controllers, and final control elements, such as valves. The performance of these instruments have been, by and large, satisfactory except for the 100% steam pressure control valves, which are brought into service when the thermogenerator set is not operational, during which time all steam produced in the steam generator is dumped into the dump condenser. These valves see a large pressure drop of around 105 kg/cm² when they are in service.

After commissioning, a number of problems were faced in the operation of these valves. The electronics for controlling the valve operation, which were mounted close to the valves, frequently failed due to the high temperatures observed during steam flow through the valve. Hence, the electronic circuits were relocated to cooler locations with improved ventilation. Also, the hydraulic pressure limit for closing the valve was increased from 34 kg/cm² to 80 kg/cm² so that the valves could be reliably closed against the high backpressure condition at closed position. Further, due to high differential pressure across the valve seat and the plug during steam service, it was difficult to open the valves once they were completely closed. To overcome this problem, the valve was not allowed to be in a fully closed condition. Since such a condition leads to other operational problems, it was decided to replace these valves with better rated valves.

1.3.3. Major refurbishments planned

- complete replacement of reactor protection system (RPS) with state of art circuits;
- revamping of the physical protection system, tele-alarm system, radiation and air activity monitoring system, etc.;
- development of a 14 L/s diode type sputter ion pump for use in place of the 35 L/s triode sputter ion pump in the SGLDS;
- commissioning of a large display station in control room;
- replacement 6.6 kV switch gear and replacement of protective relays with digital version;
- commissioning of air cooled service air compressors;
- construction of a new DM plant;
- replacement of all DC machines in sodium pump drive systems;
- replacement of 6.6 kV MOCB with vacuum circuit breaker and retrofitting in existing cubicle;
- replacement of AVR and excitation system in main alternator.

2. KAMINI

2.1. Introduction

Kamini (KAlpakkam MINI) is a 30 kW(th), ^{233}U -fuelled, light water moderated and cooled special purpose research reactor located at IGCAR. Beryllium oxide (BeO) is used as a reflector and cadmium is used as absorber material in the safety plates. The reactor was built jointly by Bhabha Atomic Research Centre (BARC) and IGCAR. The reactor functions as a neutron source with a flux of $10^{12} \text{ n}\cdot\text{cm}^{-2}\cdot\text{s}^{-1}$ at the core centre and facilitates neutron radiography of both radioactive and non-radioactive objects and neutron activation analysis. Facilities are also available in the reactor for radiation physics experiments and the calibration and testing of neutron detectors. The first criticality of Kamini was on 29 October 1996. Reactor power was raised to 30 kW on 17 September 1997 after successful completion of various reactor physics shielding and engineering experiments. Presently, the reactor is being used for irradiation experiments and neutron radiography. It is used for activation analysis for research, analytical, and forensic works, shielding experiments and neutron radiography of irradiated fuel from the FBTR. Kamini also helps the Indian space mission through the extensive use of neutron radiography of pyro-devices based in all space missions of the ISRO. The salient features of Kamini are given below.

2.2. Main characteristics of Kamini

Nominal power	30 kW(th)
Fuel	^{233}U (20 wt%) Al alloy
Number of fuel per subassemblies	9
Number of fuel plates per subassembly	8
Reflector material	200 mm thick BeO encased in Zircaloy
Moderator/ Coolant/ Shield material	Light water
Core cooling mode	By natural convection
Absorber	Cadmium
Beam tubes	3
Flux at outer end of beam tube	$10^6\text{--}10^7 \text{ n}\cdot\text{cm}^{-2}\cdot\text{s}^{-1}$
Flux at irradiation sites	$10^{11}\text{--}10^{12} \text{ n}\cdot\text{cm}^{-2}\cdot\text{s}^{-1}$
Core flux	$10^{12} \text{ n}\cdot\text{cm}^{-2}\cdot\text{s}^{-1}$

2.3. Neutronic instrumentation

The neutronic instrumentation system is provided for monitoring the neutron flux from the core and derives various parameters such as log count rate, log power, log rate and linear power for safety action. The neutronic instrumentation system consists of pulse and current channels. Two pulse channels using boron-coated counters monitor the flux during shutdown and startup and four current channels with uncompensated ion chambers are provided for power range control and safety. The reactor protection system is a relay logic-based hardwired system adopting redundancy, diversity, and fail-safe features. It consists of two channels and each channel consists of one group of redundant scram parameters and the protection system actuates scram based on 1/2 logic. The neutronic channels of Kamini were designed, developed and fabricated in the early eighties and were commissioned in 1996. The system has been in continuous operation for the last 10 years. During commissioning and in the initial few years of reactor operation, several problems were faced in the system, which in turn resulted in many spurious trips of the reactor. To overcome these problems, many modifications and improvements were extensively carried out to improve the performance of the neutronic channels.

2.3.1. *Problems faced with the neutronic instrumentation system*

Various problems were faced with the neutronic channels during initial commissioning and subsequent reactor operations. The problems are:

- 1) spurious trips on log count rate (LCRM) from pulse channels and log rate from DC channels due to noise pickup;
- 2) high leakage current from the detectors of the DC channels affecting reactor startup;
- 3) uneven range of measurement of all the DC channels and very low margin between signal output and the trip threshold of log-P signals causing reactor trips;
- 4) spurious reactor trips due to malfunction of trip relays;
- 5) frequent failure of channels due to ageing and PCB deterioration;
- 6) spurious alarms on nuclear channel unhealthy during shutdown;
- 7) difficulty in on-line adjustments and insufficient test facilities for channel calibration and component obsolescence.

2.3.2. *The improvements carried out*

Many spurious scrams occurred on different neutronic signals such as LCRM, log-P, lin-P, log rate and nuclear channel unhealthy parameters resulting in spurious reactor shutdown during reactor startup and during reactor operations. To overcome these problems, many modifications and improvements were carried out to enhance the performance of the channels.

- Pulse channels were provided with boron coated counters (BCC) with 4 cps/nv sensitivity in place of fission counters (FC) of 0.1 cps/nv sensitivity to meet the requirement of low shutdown counts for reactor startup and to get a high pulse amplitude signal from the detector for better noise discrimination and the field cables were shielded with flexible metallic conduits.
- All the power range DC channel detector signals and HV connecting cables showed low insulation resistance and high leakage current due to moisture ingress in the MI cables. They were replaced with soft coaxial cables to improve the cable insulation

and reduce the leakage current. The leakage was considerably reduced to enable reactor startup.

- Horizontally mounted trip relays in the reactor protection system were causing spurious trips due to fleeting loose connection between the relay base and its relay pins. All the trip relays were provided with a suitable clamp arrangement and no further spurious trips were noticed.
- Nuclear channel unhealthy alarms were occurring frequently whenever the field cables were disturbed near the Vault area. The HN connectors provided for the detector output cables were replaced with thin cable HN connectors. The detector output connectors and the field cable connectors along with the 'I' joints were fixed inside the junction box and were clamped to a nearby supporting beam in the vault area. Subsequently, no spurious alarms have occurred.

2.3.3. *Future plans*

During the initial phase of operation, there were many spurious trips from the channels. The modifications and improvements carried out in the channels have improved the performance and considerably reduced the spurious trips. The overall performance of the neutronic channels was satisfactory. However, the leakage current from the detectors and deterioration of PCBs of the channels causing failures, maintenance difficulty, and component obsolescence requires total revamping of the neutronic channels and the detectors. The detectors and neutronic channels need to be replaced by a state of the art system to enhance the performance of the reactor and improve operational safety.

3. PROJECT PLANNING, BUDGET AND SCHEDULE

It is planned to replace the instrumentation and control systems of Kamini reactor with a state of the art system. This involves procurement, installation and commissioning of the new systems. This is expected to improve the availability of the systems and reactor life will be extended with improved plant availability.

The project approval has been obtained and implementation is in progress with available manpower and resources without any external aid. The work is scheduled to be completed by March 2010. A reactor downtime of three months is required to complete the project on-site and reactor operations will be suitably scheduled.

It is also planned to replace the aged systems and equipment of the FBTR as part of life extension and ageing management. Since the nuclear systems have a residual life of more than 15 years, based on operational experience, condition monitoring, and maintenance frequency, replacement of some systems and equipment to extend the reactor life and improve plant availability is planned.

4. EFFECTS ON AVAILABILITY

4.1. **Improvement in plant availability**

The optimization study and the incorporated modifications in the trip parameters yielded good results in reducing the number of reactor trips and significantly improving the availability of the plant. The analysis of reactor trip data for the period of operation from 1993–2005 has indicated that the number of trips per year has decreased by a third. Works remains to further improve the system. The optimization of safety parameters is an important aspect in attaining the maximum availability of the plant without compromising safety.

The various modifications carried out in the neutronics improved the maintainability and the performance of the systems whereas the deletion of superfluous trip parameters from coolant pump drives has considerably reduced the trips from this system. The modification by which the quadrupole mass spectrometer-based steam generator leak detection system was replaced with sputter-ion pump-based system has yielded excellent performance and availability of the systems, eliminating the need for yearly ion source filament replacement along with its potential for premature failure. In a study undertaken to optimize the trip parameters, about 94 trip inputs (88 from RTDs of pump drive system) were deleted and about 13 trip inputs were added/modified. These modifications reduced the number of trips (both spurious and component failure) by an order in the coolant pump drive system and by half in the neutronic instrumentation. In the SGLDS, the downtime required for replacing the ion source filament and its premature failure potential was totally eliminated. The modifications in the reactor protection system helped to achieve significant increase in plant availability. Further improvements are planned and optimization studies will be continued in the future based on operational feedback.

The replacement of contact type LP heaters by shell and tube non-contact type heaters in the condensate system resulted in a reduction of feed water fluctuations that had caused cascade tripping of the pumps which in turn caused reactor trips. After the modification, the system has been trouble-free and there have been no reactor trips from the steam water system.

The replacement of single subsystem-II of the Central Data Processing System (CDPS) by three ED-20 based embedded systems has improved the availability of the CDPS. With the addition of two more channels in the steam generator leak detection system (triplication) in each secondary sodium loop and wiring of 2/3 logic for safety actions, the overall availability, maintainability, and reliability of SGLDS has been improved.

The modifications and improvements of the various systems significantly reduced the number of reactor trips, which reduced the thermal cycle, fatigue, etc. on many important safety-related components. These modifications were incorporated without affecting the overall safety of the plant.

4.2. Timing or constraints in implementation

Retrofitting a new channel in the existing plant in the SGLDS, in situ welding of stainless steel sodium pipe lines, the quality assurance in creating an ultrahigh vacuum environment, etc. was quite challenging and technically highly satisfying. All the teething problems faced during the commissioning were resolved and the third channel was successfully pumped down and an ultimate vacuum of $\sim 10^{-8}$ torr was achieved. System response was checked and calibration data was acquired by actual injection of hydrogen into sodium. The performance of the new channel was very good. With the addition of the third channel and wiring of 2/3 logic for safety actions, the reactor operation can be continued with the two healthy systems while the third system is in maintenance. Wherever possible, any parameter initiating safety action on reactor has to be triplicated and wired in 2/3 voting logic. This minimizes downtime due to spurious reactor trips on single channel and cumbersome investigations.

4.3. Management oversight

Management oversight did not affect refurbishment activity completed or in progress.

4.4. Resources used

Completed refurbishments could be made within current operating budget allocations and modifications were completed with the available manpower and plant resources.

However, for the proposed refurbishment in the FBTR & Kamini reactors, the capital budget of INR 80 million (~US \$2 000 000) under XI five-year plan is being used with the available manpower.

4.5. Lessons learned

The conclusions from various other modernization methods initiated for improved reliability and life extension of the plant are given below:

- An on-line calibration method should be explored wherever possible to avoid reactor shutdown for mandatory calibration purposes.
- The verification and validation of software codes and procedures to be followed for commissioning have become progressively stringent and effective planning is required before any software modifications.
- On-line detection of system component failures is required to effectively address frequent failures even with the compliance of mandatory surveillance tests.
- Reporting and analysis of events of lower significance is also required in order to avoid more serious events.
- Electronic component obsolescence is a serious problem in managing the maintenance of equipment.
- Computer codes developed in-house are preferred to provide adequate flexibility in modification and maintainability.

5. CONCLUSION

The capital cost of nuclear reactors, especially fast reactors, is very high. Nuclear power can be made economically viable only by designing them for longer lives and extending the lifetime by suitable requalification programmes at the end of design life. The FBTR is unique in the sense that the nuclear systems are relatively new, while conventional systems have undergone chronological ageing. It will be possible to extend the reactor life of the FBTR and Kamini reactors by modernization and refurbishing aged components and systems and by adopting suitable operational strategies. The lessons learned from the modernization already completed and currently in progress will act as an experience feedback in effectively planning and executing future projects without compromising plant safety.

BIBLIOGRAPHY

RAMANATHAN, V., BABU, B., RAJENDRAN, B., SAHU, H.K., "Nickel diffuser based instrumentation for real time detection of hydrogen concentration in liquid sodium in fast breeder test reactor", Proceedings from the National Symposium on Physics and Technology of Sensors-8 (NSPTS-8), Indira Gandhi Centre for Atomic Research, Kalpakkam, India, 27 February–1 March 2001 (2001).

MEENACHI, N.M., GOWRI, C., RAVI, M., VINOLIA, K., BABU, B., RAMANTHAN, V.,
“Distributed supervisory and control system for fast breeder test reactor”, Intl. Conf. on
Emerging Applications of IT, Calcutta, India (2006).

Software for Computers in Safety Systems of Nuclear Power Stations, IEC 880 (1986).

UPGRADE OF THE BANDUNG TRIGA 2000 REACTOR

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Abstract

Since 2000, the Bandung TRIGA 2000 reactor, formerly named the Bandung TRIGA Mark II reactor, was upgraded from 1 MW(th) to 2 MW(th). The main purpose of the upgrade was to increase the radioisotope production capabilities to support the Serpong Siwabessy Multipurpose Reactor, as well as to increase the safety features of the Bandung TRIGA reactor. The upgrade project began in 1994 with the establishment of an upgrade team that successfully released a Bid Invitation Specification (BIS) document in 1995. The actual upgrade started in April 1996. Originally, the upgrade was expected to be completed in one year; however, the upgrade was not completed until 2000. During the upgrade, the following systems were subject to either change, refurbish, addition or modification: reactor core configuration (from annular to hexagonal), number of control rods (from 4 to 5), chimney and emergency core cooling system (ECCS), graphite reflector, cooling system (primary and secondary, including: new heat exchanger and cooling tower), emergency ventilation system. Although the reactor tank was not part of the upgrade project at first, it was decided to replace the old tank after an inspection of the tank. The paper also describes some experiences and lessons learned during the removal of the old tank, installation of the new tank, disassembly and reassembly of the core structure, thermal- and thermalizing column and others.

1. HISTORICAL BACKGROUND

The nuclear era in Indonesia began in 1958 with establishment of the Atomic Energy Agency. The main goal of the agency was to promote the peaceful applications of nuclear energy. In 1952, the decision was made to build a reactor in Indonesia and on 10 October 1964, Indonesia's first nuclear reactor went critical. Since then, the Bandung TRIGA Mark II reactor has operated at a nominal power level of 250 kW(th). The reactor, made by General Atomics, USA, is located in Bandung city about 180 km from Jakarta, the capital of Indonesia. The main uses of the reactor are for radioisotope production for radiopharmaceutical and other purposes, research in neutron radiography, neutron activation analyses, neutron and gamma diffraction and spectrometry, and also development of some metal alloys. The reactor also serves to educate and train university students.

The Bandung reactor was upgraded from 250 kW to 1 000 kW in 1972 because of increased demands for radioisotope production. In this first upgrade, the reactor core was enlarged from an A-F annular configuration to an A-G annular configuration. The capacities of the primary and the secondary cooling system were also upgraded. However, the reactor tank was unchanged. Since then, the reactor has safely operated up to 1 000 kW.

The TRIGA stainless steel clad fuel elements were introduced in the core in 1978 to replace the aluminum clad fuel elements. The fuel replacement was performed gradually and by 1982, all fuel elements were stainless steel clad. Also, the core was loaded with two types of fuel elements, i.e. 8.5 wt% ^{235}U and 12 wt% ^{235}U . However, both fuel element types have the same enrichment of 20% ^{235}U and the same $\text{UZrH}_{1.6}$ material composition.

In 1992, the instrumentation and control (I&C) system was upgraded from analogue to digital instrumentation. The new I&C system, called the NM-1000 and NP/NPP 1000 system, was developed and made by General Atomics, USA.

A special task force, called the Upgrade Team, was established in 1994. The team had a task to formulate a Bid Invitation Specification (BIS) for the Bandung TRIGA Mark II reactor upgrade. Based on the BIS, the second upgrade of the Bandung TRIGA reactor started in 1996 after operating for 2 440 MWD since 1971. This second major upgrade will be described in the paper.

2. MODERNIZATION AND REFURBISHMENT SCOPE

2.1. Principal and supporting motivation

The increasing activities in nuclear physics research, such as neutron activation analysis and an increasing demand for radioisotopes, especially fission product molybdenum (FPM), were the major driving forces for upgrading the Bandung TRIGA Mark II reactor to a higher power level. The need to enhance the safety level of the reactor by addressing ageing reactor components was also one of our major concerns.

Therefore, the main objectives of the upgrade programme were:

- To enhance the safety level and features of the Bandung TRIGA Mark II reactor. This was accomplished by refurbishment, modification, replacement and addition of components and systems to improve safety and reliability.
- To upgrade the maximum thermal power of the reactor to 2 000 kW. This would increase the neutron and gamma flux in the core and other irradiation facilities, including the beam ports and rotary specimen rack.
- To provide in-core irradiation facilities capable of producing higher quantities of radioisotopes. These objectives would enable us to produce 1 000 Ci FPM, produced from the irradiation of 15 g of highly enriched ^{235}U targets.

2.2. Impacted systems, structures or components

During the upgrade, almost all reactor systems were refurbished, modified or changed. The only systems unchanged were the reactor building and reactor biological shielding. Most systems, components, and materials were fabricated and supplied by General Atomics, USA. However, all removal and installation activities were performed by the Center for Nuclear Techniques Research-BATAN (CNTR-BATAN).

2.2.1. Reactor tank

Initially, the upgrade project only involved a thorough inspection and testing of the reactor tank. However, the inspection results showed that the whole tank was heavily corroded and should be renewed.

For practical and radiation protection reasons, the old tank remained inside the biological shielding and the new tank was placed directly inside the old tank. However, the extruding parts of the reactor tank, such as the thermalizing column, thermal column, and beam ports had to be cut out and removed. Tank materials (aluminum 6061-T6) were supplied by General Atomics, but fabrication and installation was performed by CNTR-BATAN.

The height of the new tank was extended 1.5 m for a total height of 8 m in order to provide adequate shielding at the reactor deck. Therefore, a new platform was erected on the old reactor deck to make it easier to work on the reactor deck.

2.2.2. *Reactor core*

Originally, it was planned to use the old core structure, including the reflector. In the original core modification plan, the top and bottom grid plates were to be replaced in order to accommodate the new hexagonal core configuration. However, the old core structure needed to be disassembled because of the reactor tank replacement. During the removal of the old core structure it was found that even though the core structure was made of aluminum, it had high radiation levels, which made it very difficult, even impossible, to reassemble the core structure. At that point, it was decided that a new core structure needed to be fabricated and installed.

While the old core had an annular configuration, the new core had a hexagonal configuration with other parts remaining the same. A new system called the chimney was installed on top of the graphite reflector. The new core configuration and the chimney provide better natural convective cooling in the reactor core and appropriate for a power upgrade of 2 000 kW.

During the upgrade, 20 wt% ^{235}U fuel elements were introduced in the reactor core to compensate for the old and highly burned up fuel elements, which are still used in the present core. Currently, the core is loaded with a mixture of fuel element types, i.e. 8.5, 12, and 20 wt% ^{235}U .

2.2.3. *Control rods*

To increase the ability to control the core excess reactivity, a new control rod was installed in the reactor core for a total of five control rods. All the control rods are now located at the same relative distance from the core centre (in D-ring), i.e. in positions D-02, D-05, D-09, D-13, and D-16.

2.2.4. *Primary cooling system*

Even though the components and materials were supplied by General Atomics, the new primary cooling system was designed, built, and commissioned for a thermal capacity of 2 400 kW by CNTR-BATAN. The primary cooling system consists of the reactor pool, primary pump, heat exchanger, and the connection pipes. This system is also equipped with some measurement tools, such as a temperature measuring device, a pressure measuring device, a flow rate measurement tools, and a water conductivity meter.

2.2.5. *Secondary cooling system*

The new secondary cooling system was designed, built and commissioned for a thermal capacity of 2 400 kW solely by CNTR-BATAN. There was no involvement by General Atomics in this system. The system is located outside of the reactor building adjacent to the reactor hall. The piping circuit is made from carbon steel tube and includes two centrifugal pumps and two cooling towers. Commissioning of the secondary cooling circuit demonstrated the ability of the system to remove heat generated by reactor operations at 2 000 kW.

2.2.6. *Emergency core cooling system*

One of new engineered safety features of the upgraded reactor system is an emergency core cooling system (ECCS). This system prevents consequences associated with

a possible exposure of the reactor core during a loss of cooling accident (LOCA). During a LOCA, the system will spray water on the top of the reactor core in order to remove the decay heat for about 6 hours with 8 gpm averaged flow rate. The system was designed, built, and commissioned solely by CNTR-BATAN.

2.2.7. Emergency ventilation system

Another new engineered safety feature provided for the reactor system is an emergency ventilation system. This is to prevent release of gaseous fission products into the environment during abnormal reactor operations or a fuel cladding failure. The system consists of a ventilation path that has an active carbon (charcoal) filter. In normal mode, this ventilation path is bypassed, while during the emergency mode, the air flow in the ventilation system is directed to the emergency ventilation system manually. The system was also designed, built, and commissioned solely by CNTR-BATAN.

2.3. Safety/Licensing-related analyses and documentation impact

Due to the major modification discussed above, a new licensing application was submitted to the National Nuclear Regulatory Agency-BAPETEN. As a result, CNTR-BATAN has completely revised the Safety Analysis Report (SAR) to accommodate the new capabilities and features of Bandung TRIGA 2000 reactor. The new SAR, based on the format of IAEA Guidelines, was mainly prepared by CNTR-BATAN. However, General Atomics prepared the Safety Analysis section of the report.

3. PROJECT MANAGEMENT

3.1. Budget

The entire budget of the upgrade project was provided by the government of Indonesia. The total funding of US \$1 000 000 went to General Atomics for the following tasks:

- modifications of required reactor internal hardware and the core from 1 000 kW to 2 000 kW with proven and qualified LEU fuels, including safety evaluations;
- design modifications and requirements of the primary heat transport system and I&C system;
- design modifications and requirements for the reactor shutdown system;
- eight weeks training and guidance in tank inspection, commissioning, operation and maintenance;
- the necessary procedures, software, and technical data to enable CNTR-BATAN to perform verification, particularly for safety analysis;
- draft of reactor-related sections of the SAR.

The Indonesian government spent almost the same amount of money for funding the work provided by CNTR, which consisted of:

- removal of the old core using recommendations and other required assistance from General Atomics and installation of the new core components;
- design and modification of the ventilation system and secondary cooling system, installation of the new primary cooling system, and I&C system;
- design and installation of building and equipment and installation of new reactor protection system;

- installation of new fuel under General Atomics guidance;
- implementation of tank inspection, commissioning, operation and maintenance under General Atomics guidance;
- completion of final SAR for the entire facility, including reactor-related sections provided by General Atomics.

3.2. Schedule

The upgrade was scheduled to be completed in one year beginning in April 1996. However, due to the replacement of the reactor tank, which was not part of the original plan, completion of the upgrade was delayed. The 1998 economic crisis also played a part in the upgrade delay. The upgrade was finally completed in 2000.

4. LESSONS LEARNED

The following lessons were learned throughout the upgrade:

- Even though the core structure was made of aluminum material, some parts, such as nuts and bolts, were made of stainless steel. The high radiation levels of these stainless steel parts made their reuse difficult, or even impossible.
- Expect the reactor tank/liner to be in poor condition. This would help us manage and handle the ageing process of the reactor tank/liner.

BIBLIOGRAPHY

INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report, IAEA Safety Series No. 35-G1, IAEA, Vienna (1994).

INTERNATIONAL ATOMIC ENERGY AGENCY, Safety in the Utilization and Modification of Research Reactors, IAEA Safety Series No. 35-G2, IAEA, Vienna (1994).

CENTER FOR NUCLEAR TECHNIQUES RESEARCH, Term of Reference for Safety and Power System Upgrading of Bandung TRIGA Mark II Reactor, Document CNTR No. RE-01-01, Bandung, Indonesia (1993).

CENTER FOR NUCLEAR TECHNIQUES RESEARCH, Term of Reference for the Upgrading of Safety System and Thermal Power of The Bandung TRIGA Mark II Reactor, Document CNTR No. RE-01-02, Bandung, Indonesia (1995).

CENTER FOR NUCLEAR TECHNIQUES RESEARCH, Bid Invitation Specification (BIS) for the Upgrading of Bandung TRIGA Mark II Reactor, Document CNTR No. RE-010, Bandung, Indonesia (1995).

CENTER FOR NUCLEAR TECHNIQUES RESEARCH, Quality Assurance Programme of CNTR, Document No. QAIND-01-REV.01, Bandung, Indonesia (1995).

CENTER FOR RESEARCH AND DEVELOPMENT OF NUCLEAR TECHNIQUES, National Nuclear Energy Agency, Safety Analysis Report TRIGA 2000 Bandung Reactor, Document No. RE-01-06, Bandung, Indonesia (2001).

MODERNIZATION AND REFURBISHMENT AT THE HOGER ONDERWIJS REACTOR

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Abstract

This paper gives an overview of past and present modernization and refurbishment projects at the Hoger Onderwijs Reactor (HOR). To illustrate past project cases, two instrumentation and control projects are described in more detail, followed by the most important lessons learned. In addition, a present and a future project case are described, with a short overview of the contemporary challenges in realizing plant modernization and refurbishment in a university setting.

1. INTRODUCTION

The Hoger Onderwijs Reactor (HOR) is operated by the Reactor Institute Delft. The Reactor Institute Delft (RID) is part of the Faculty of Applied Physics, which belongs to the Delft University of Technology. The RID is the Dutch national centre for multidisciplinary research and education involving the HOR, nuclear radiation, and radionuclides. Until 2005, the institute was an interfaculty facility. In 2005, the interfaculty facility was split in a new research department, Radiation, Radionuclides and Reactors (R³), and the Reactor Institute Delft, caretaker of the research infrastructure. Together they form the national focal point of expertise in the fields of reactor physics, neutron and positron radiation (including detection), as well as radiochemistry. At the heart of the reactor institute is the HOR as a source of neutrons, positrons, and radioisotopes. Figure 1 shows a site view with the reactor building and adjacent experiment hall. The following sections will give a short facility description, the history of HOR modernization and refurbishment projects, case examples of modernization and refurbishment of the data acquisition system and neutron guide installation, the lessons learned, and a brief outline of two future modernization and refurbishment projects.



FIG. 1. RID site view.

2. FACILITY DESCRIPTION

The HOR is a pool-type research reactor. The basis of the HOR is the A-57 reactor originally built for demonstration purposes and designed by the American Machine & Foundry Company and exhibited at Amsterdam airport in 1957. Afterwards, this demonstration reactor was bought by the Dutch state for university and training purposes and was located at the Delft University of Technology. Construction of the HOR and laboratories began in November 1958 and first criticality was achieved on 25 April 1963. Figure 1 shows an overview of the RID site.

The reactor core set-up has been modified and relicensed several times, the latest major change being the core conversion from HEU to LEU with core compaction from 32 to 20 fuel assemblies. The 20 fuel assemblies are surrounded by beryllium reflector elements on three sides. Currently, the HOR is operated 24 hours/day, 5 days/week at a thermal power level of 2 MW, with an average thermal flux of about $2 \times 10^{13} \text{ n}\cdot\text{cm}^{-2}\cdot\text{s}^{-1}$.

3. HISTORY OF MODERNIZATION AND REFURBISHMENT

Since the HOR commissioning 1963, there has been a steady stream of modernization and refurbishment projects. Table 1 shows a retrospective listing of the most important projects since the first commissioning of the reactor. It is obvious that the projects differ very much in scope and impact. In 1957, when the plans for building a reactor facility in Delft were developed, the original purpose of the HOR was the training of nuclear engineers. For that purpose, a low flux research reactor was quite adequate. It turned out that the expected growth of nuclear power did not occur, and the purpose of the reactor shifted to the use of the HOR as a neutron source for fundamental science and basic research. Because of the original low flux design, the HOR was not suitable for this new task. In 1965, possibilities for increasing the power to get a higher flux were investigated and it was decided to increase the power to 3 MW(th). In 1968, a new forced flow cooling system was installed, including two cooling towers designed for 3 MW(th). However, the 2-section heat exchanger design was limited to a capacity of 2 MW(th). For 3 MW(th) operations, another heat exchanger section of 1 MW(th) capacity had to be installed at a later date. The safety analyses and safety reports were also updated for the 3 MW(th) licence application. The original plan was to operate the reactor at 2 MW for a certain period after installation and commissioning of the new cooling system. At a later date, based on experience and using a new fuel element design, the routine power level would be increased to 3 MW(th), after an increase of the heat exchanger capacity. The newly designed fuel elements were introduced, but the heat exchanger add-on was not realized and reactor power stayed at 2 MW(th) routine operations. Until now, this project had the biggest impact on our technical resources and was carried out internally at low cost and within a very limited time frame.

TABLE 1. OVERVIEW OF MODERNIZATION AND REFURBISHMENT PROJECTS AT THE HOR

Year	Description of modernization and refurbishment project	Main driving force
1963	Commissioning and start of 100 kW operation	Training of nuclear engineers
1968	Installation of 2 MW cooling system.	Utilization (scientists)
1971	Installation of the stainless steel lining in the large and small pool to reduce the leakage rate.	Safety
1981	Replacement of the old control room and upgrading and extension of reactor instrumentation.	Ageing
1985	Modification of the HOR ventilation system. The external piping circuit was modified and brought inside the building to prevent corrosion problems and replacement of isolation valves with a hermetic seal by a water column by butterfly valves.	Ageing
1992	Renewal of the isolation valves in the primary and secondary cooling system and replacement of the hand-operated control valves by electro pneumatic-operated control valves. The single-walled watertight feedthroughs of the pool were upgraded to double-walled feedthroughs.	Safety
1993	With reference to the safety analyses by Siemens, the primary and secondary cooling flows were increased by the installation of higher capacity pumps.	Safety
1995	To protect the containment against an inadmissible over- and under-pressure, a water seal was installed. Because of corrosion problems, the old water seal was replaced with a newly designed stainless steel water seal that was more compact and better accessible for maintenance.	Ageing
1997	The existing tubular in the wall of the containment was modified to lead the new n-guides from the reactor hall to the experiment hall, keeping the containment barrier.	Utilization (scientists)
1998	Completion of the new experiment hall and modification of existing feedthrough to get positrons to the new experiment hall.	Utilization (scientists)
2000	Replacement of the old HOR PDP 11/44 process computer and CAMAC-system by a network-based system.	Ageing
2000	Installation of a new n-guide from the reactor hall to the experiment hall.	Utilization (scientists)
2001	Extension of the butterfly valves of the HOR ventilation system by installation of an additional butterfly valve at the inlet and outlet of the HOR ventilation system.	Safety
2006	Full containment plate inspection and plate thickness measurements, conservation and corrosion spot treatment	Ageing
2006	Replacement of the primary cooling system suction head because of damage due to human error	Utilization (scientists)

In review, the project with the biggest impact on systems, structures and renewal of components was the replacement of the old control room and upgrading and extension of the reactor instrumentation which took place in 1981. In the early 1970s, it became clear that a strategy for upgrading and modernization of the instrumentation should be developed. At that time, the original instrumentation, using vacuum tube equipment, caused an increase in spurious trips and repair efforts and components became obsolete. In 1978, a task group was formed under the supervision of an external project manager to take care of the renewal project.

In the first stage of the project, the task group was mainly focused on the hardware and less on the requirements and system design criteria. It took some time to realize that this approach was inadequate. In 1979, it was decided that a total refit of the instrumentation, including the safety and control systems, should be performed. In practice, this implied a complete new system design, new sensors, detectors, cabling, control rod drives, cabinets, and even a newly built control room outside the reactor building due to space requirements. On the basis of general designs, terms, and quotes from the potential contractors and suppliers, a choice was made. This was followed by contract negotiations with the preferred contractor and/or supplier. The project was contracted as a fixed time/price and turnkey project. The target reactor shutdown/installation period was 8 months. The contractor/supplier was allowed a 24-month period to design, construct, and manufacture components before actual reactor shutdown. After the contract was signed and the design was finalized, a licence application was submitted in spring 1980. The licence was granted at the end of 1980. The actual shutdown period was approximately 10 months. The project organization is shown in Figure 2.

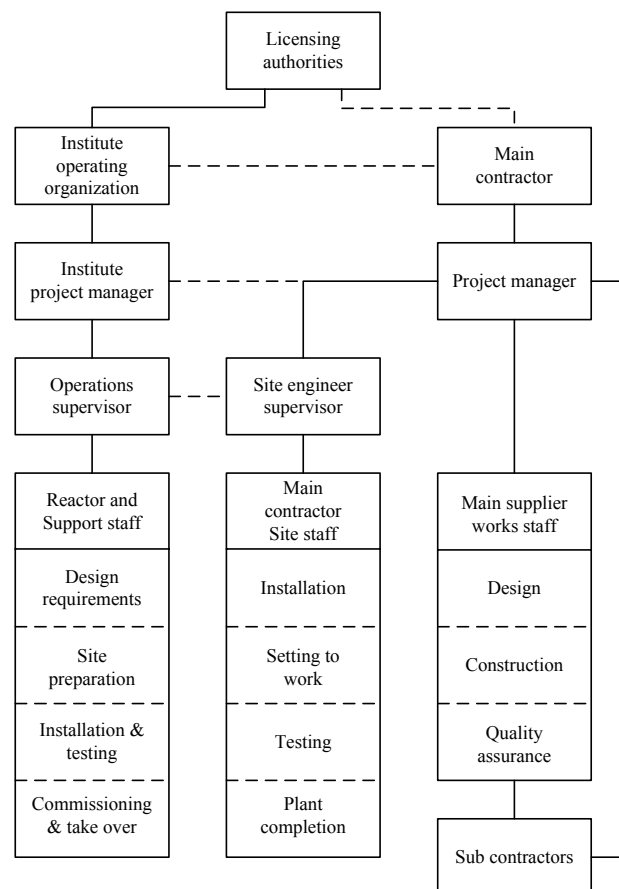


FIG. 2. Project organization control room replacement.

In this period, the reactor institute had the status of an interuniversity institute, which means that it was directly controlled and financed by the Ministry of Education and Science of the Netherlands. The final choice of the contractor/supplier was made by the Minister of Education and Science.

4. MODERNIZATION AND REFURBISHMENT OF THE DATA ACQUISITION SYSTEM

An example of a smaller and more recent modernization and refurbishment project was the installation of a new data acquisition system. This project was performed during the summer maintenance period of 2000. A computer-based data acquisition and control system was already installed during the replacement of the old control room and upgrade and extension of the reactor instrumentation. This system was based on a CAMAC processing interface and a PDP 11/44 host computer system.

There were two driving forces for starting a modernization project of the data acquisition system: ageing and utilization (scientists). The HOR Technical Specifications only allow the reactor to run for a limited time period without a functioning data acquisition system. In particular, due to very tedious repairs of the PDP 11/44 system at the end of the 1990s, reactor availability was deteriorating. There were several severe interruptions and our scientific customers were not amused.

System renewal was considered in 1993, but a lack of human resources along with competing projects delayed the project. In 1998, a project group consisting of internal reactor institute personnel was established and the project really took off. The addition of two outside experts to the project group added experience and knowledge with many different data acquisition systems. At the same time, the institute implemented an institute-wide 'how to' project guide for project management. The data acquisition project was set up according to this project guide.

Eleven possible suppliers were asked to provide a quote based on the software and hardware requirements. Some of the possible suppliers decided to withdraw from the project and 7 quotes were finally received. All budgetary quotations were assessed following a set of criteria set up in advance. From the first round of quotes, only one supplier was chosen to provide a more detailed design and quotation. The design approach and applied hardware and software of the top 2 suppliers were very different and incompatible with one another. It was decided that the most sensible approach was to proceed with the most appealing option. After this phase, the project was split in three task groups: hardware, software and a database group. An overview of the project organization is shown in Figure 3.

In the beginning of the project, very underestimated turned out to be the detailed description of the interfaces. The supplier needed, of course, an exact specification of all the analogue and digital input signals. Furthermore, the old system was implemented to use dedicated FORTRAN and PDP assembly code, e.g. for automatic control, on-line reactor power, and reactivity calculations. The supplier considered the lack of clear software specifications in use and the need for reverse engineering a big risk. It was decided that we would perform the reverse engineering of existing, specific, reactor-related programmes. It turned out to be a very time consuming and difficult task. The institute's project guide did not provide information on how to consider and handle risks at the beginning of the project.

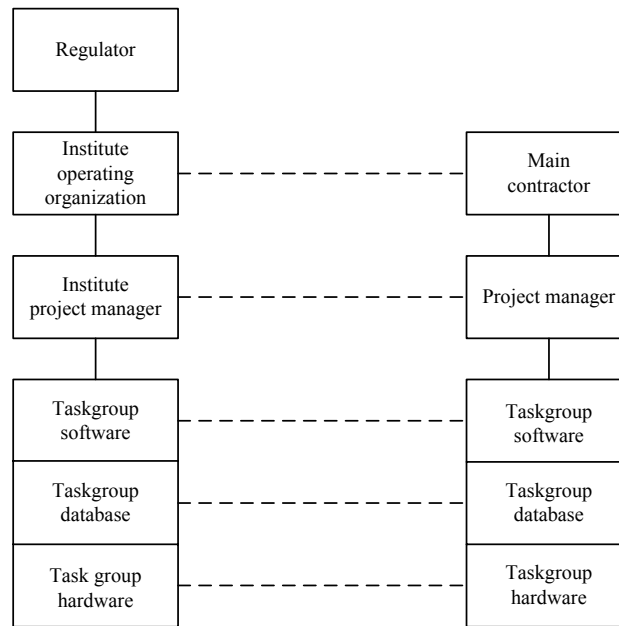


FIG. 3. Project organization data acquisition renewal.

Based on a procedure in our quality system dedicated to modifications of the reactor installation, we received approval from the nuclear inspectorate to begin the project. The procedure implied that for this type of modification, a formal licence procedure was not required. The procedure was based on a safety classification recognizing four (institute internal) safety classes:

- Safety class 1. Systems dealing with process safety and confinement, for example the control rods and the containment.
- Safety class 2. Systems within the second line of defence, for example the automatic run down actuating system (ARDAS) and the containment ventilation system.
- Safety class 3. Systems necessary for normal operation or maintenance like the secondary cooling system and the polar crane.
- No safety class. All other systems within the HOR configuration.

Depending on the safety level, the review process for each safety class has been specified. For safety classes 1 and 2, a review assessment and approval by the internal reactor safety committee and the Dutch regulator is required.

The data acquisition system incorporates the automatic rundown system, ARDAS, which is a limiting system. It must be emphasized that this system is not part of either the reactor protection or safety systems. The main function of the automatic rundown system is to command the insertion of all control rods with normal drive speed under certain, deviating process conditions. Since this specific function was classified in accordance with our internal procedure as Safety class 2, the assessment and approval of the complete project by our internal reactor safety committee and the Dutch regulator was required. The ARDAS uses digital inputs and is based on input tables for the different reactor operational modes, which are: automatic control and forced convection cooling, automatic control and natural convection cooling, manual control and forced convection cooling, manual control and natural convection cooling. The ARDAS programme is hardware resident and functions independently of the other tasks of the data acquisition system. This means that a malfunction

of the central data acquisition computer does not influence the automatic run down system. The system itself, however is single redundant.

During the factory acceptance tests, special attention was given to the ARDAS function. As well, during the installation phase, all aspects of the ARDAS function and possible unforeseen malfunctions were extensively tested. Despite these efforts, it took an additional three months after commissioning to debug an unexpected error in the ARDAS function. Since the new data acquisition system must interface with existing plant hardware, the interface or boundary descriptions should be very precise and as strict as possible. Of course, we intently tried to check the actual as-built status of the system to be replaced, but were forced to extend the installation phase for 4 weeks in order to adapt software and, in some cases, hardware to the remaining, existing plant hardware.

5. LESSONS LEARNED

The most important lessons learned from the two instrumentation and control projects described above, are:

- Actively stimulate communication (cross-horizontal, vertical top-down, and bottom-up) between all project members and layers. This means not only communication between project managers but also direct communication between the operation organization specialists on the one hand and the regulator/supervisor specialists and the contractor on the other hand.
- Start early in the preliminary project design stage with consultation sessions with the regulator specialists. This saves time later in the approval or licensing process.
- Make a sound modification procedure based on categorization (safety classes) and approval routes and have this modification procedure assessed and approved by the reactor safety committee and the regulator. This saves a lot of time in discussions on how to approve modernization and refurbishment projects.
- Implement a known project management system, such as PMI, IPMA or PRINCE2. Do not try to make your own design. By doing so, you force yourself to document and confirm that an acceptable business case exists for the project and an adequate project organization structure is ensured. And last, but not least, you are forced make available enough time and budget to consider project risks, such as a lack of information concerning the as-built status of older plants.
- A modernization & refurbishment project is a good opportunity to make improvements to the design, based on the plant operating experiences, technical developments, and safety design philosophy developments.

6. MODERNIZATION AND REFURBISHMENT IN THE FUTURE

Two large modernization and refurbishment projects are foreseen for the future. The biggest is the Optimized Yield for Science, Technology and Education of Radiation (OYSTER) project. The main driving force for this project is 'utilization' (scientists). A dedicated business plan has been made for this project. The objectives of the OYSTER project are three-fold:

- increasing the performance of the HOR as a source for neutron beam research to within one order of magnitude in comparison to the large international facilities;
- assuring the continued leading international forefront position of the positron source, POSH;

- obtaining irradiation facilities with high neutron flux and low gamma dose rates as required for novel production routes for radioisotopes and radio-pharmaceuticals.

The OYSTER project foresees a major modernization and refurbishment of the reactor and the neutron beam instruments. The basis for such a project has been partly motivated by the outcome of a feasibility study performed earlier. It comprises the installation of a cold neutron source, compacting the core using high density (4.8 g/cm^3 uranium silicide) fuel and a power increase from 2 MW to 3 MW.

The second project, which was officially kicked off in December 2007, is the modernization and refurbishment of the nuclear instrumentation and reactor protection system (trip logic) of the HOR. Ageing is the main driving force for this project. This project is closely related to future OYSTER project developments. In practice, this means that the new nuclear instrumentation and reactor protection system must fit to the current plant situation as well be adaptable to the requirements of the OYSTER project. This implies a risk because it is not known what changes will be made to the HOR design as a result of the OYSTER project. A new item and challenge in a HOR modernization and refurbishment project is the need for a European public procurement procedure. By European legislation, a strict public procurement procedure must be followed when the cost of a project exceeds a certain amount. In our opinion, this makes it difficult to meet with the vendors in the conceptual design phase. Meeting with vendors prior to the public procurement procedure could provide them with an unfair advantage during the procurement procedure, which is in conflict with European legislation. This means our latest challenge in the modernization and refurbishment project is to find a way to have consulting sessions with the vendors.

BIBLIOGRAPHY

DE VRIES, J.W., DE MEULEMEESTER, F.M., "Computer system for plant monitoring and control of the Hoger Onderwijs Reactor at Delft, The Netherlands", Proceedings ANS International Topical Meeting on Computer Applications for Nuclear Power Plant Operation and Control, ANS, Washington (1985).

INTERNATIONAL ATOMIC ENERGY AGENCY, Safety in the Utilization and Modification of Research Reactors, Safety Series No. 35-G2, IAEA, Vienna (1994).

GIBCUS, H.P.M., DE LEEGE, P.F.A., LABOHM, F., DE VRIES, J.W., VALKO, J., FELTES, W., HEINECKE, J., Upgrading the HOR: An Upgrading Study for the HOR, Proceedings International Conference Research Reactor Fuel Management (RRFM 2003), Aix-en-Provence, France (2003).

REFURBISHMENT OF SECONDARY CIRCUIT AND DISTRICT HEATING OF THE JEEP II REACTOR

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Abstract

This paper provides technical detail and related project management information related to a project to modify the Jeep II reactor cooling system to provide district heating. Specific information related to the primary and secondary cooling systems and modification project are explained. There are also specific sections on project management, lessons learned as well as plans for future projects.

1. INTRODUCTION

The JEEP II research reactor is located 20 km north-east of Oslo, Norway. It is a tank-type reactor and achieved criticality for the first time in December 1966. It replaced the first JEEP reactor that had been in operation since 1951.

Key data for the reactor are as follows:

- nominal power 2 MW(th);
- heavy water moderator and cooling;
- moderator temperature 55°C;
- atmospheric pressure;
- hexagonal core with 19 fuel elements;
- fuel 253 kg UO₂, 3.5% enrichment;
- maximum thermal flux $3 \times 10^{13} \text{ n}\cdot\text{cm}^{-2}\cdot\text{s}^{-1}$.

The main use of the reactor is for basic research in the material sciences and for irradiation purposes. There are 9 beam channels, 8 radial and 1 tangential, for a total of 10 beam ports. The tangential channel utilizes a cold moderator and another channel is used for neutron radiography. The remaining channels are used for different kinds of neutron diffractometers and spectrometers. In the reflector area, there are nine vertical channels for irradiation purposes. Seven of these channels are filled with heavy water and rotate and are used for the irradiation of silicon. The other two channels are dry and are used for isotope production and neutron activation analysis. In addition, there is one 'rabbit' channel in the core lattice which is used for activation analysis.

The reactor is operated 24 hours, 7 days per week. There are three main interruptions each year at Easter, summer, and Christmas for vacation and maintenance. On a daily basis, there may be several short interruptions for the loading and unloading of specimens for irradiation.

2. COOLING CIRCUITS

The original design of the reactor main cooling circuits consisted of a primary heavy water circuit and a light water secondary circuit. The two circuits were separated by a tube-type heat exchanger. The water in the secondary circuit was cooled in two air/water cooling towers. The secondary circuit was then open to the environment via the cooling tower.

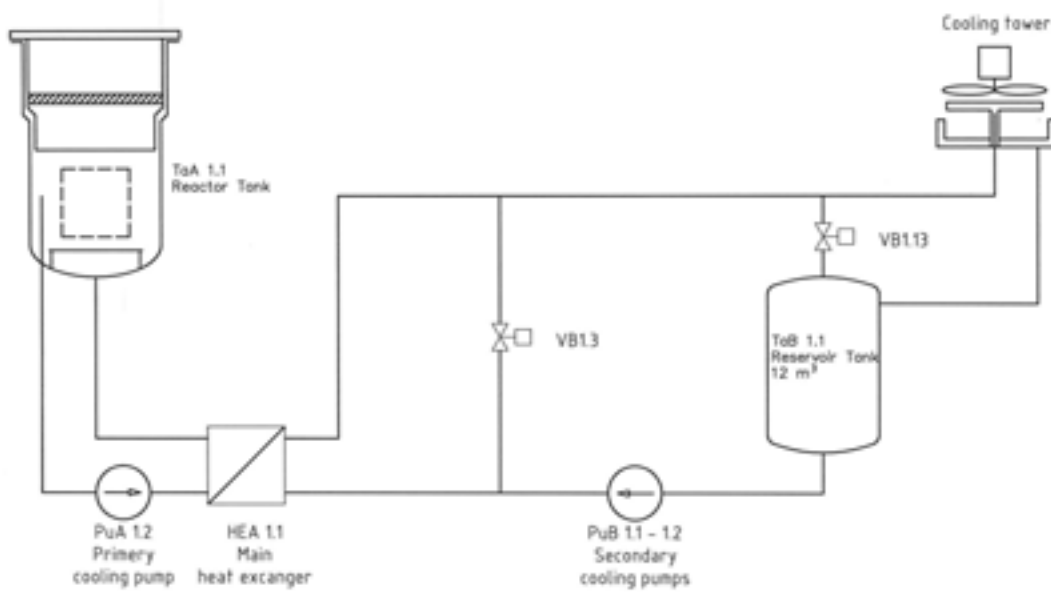


FIG. 1. Original cooling circuits (simplified).

In 1985, the secondary circuit was redesigned to deliver heat to a local district heating system. At this time, the cooling tower was separated as a tertiary circuit. The district heating circuits were part of the secondary cooling circuit. This system is shown in Figure 1. The district heating circuits and the cooling tower circuit are in parallel and the water from the main heat exchanger is distributed to the circuits by a regulating valve (VF 1 in Fig. 2).

The heat delivered to the district heating system constitutes approximately 25% of the total power consumption at the institute. In 2006, it was decided to replace the old tube-type heat exchanger. There was also a strong desire to isolate the district heating system as a tertiary circuit. This modification is described below.

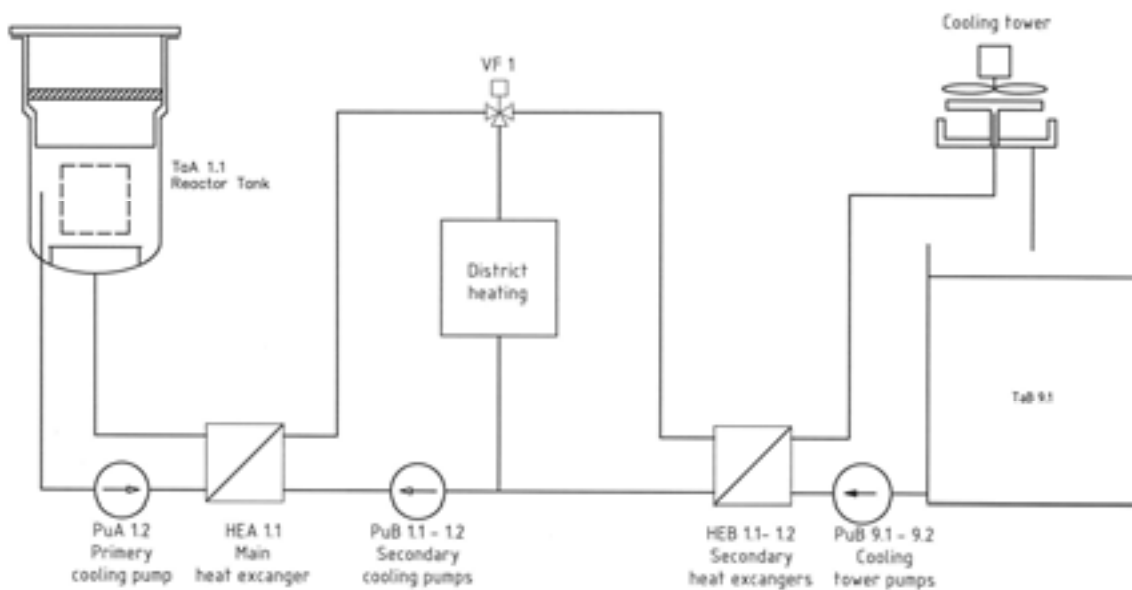


FIG. 2. Cooling circuits with district heating as designed in 1985 (simplified).

3. SYSTEM DESCRIPTION

3.1. Primary cooling circuit before modification

The circuit contains approximately 5 tonnes of heavy water and removes the 2 MW of heat produced in the reactor. The heavy water is pumped from the reactor vessel (TaA 1.1 in Fig. 2) through the main heat exchanger (HEA 1.1 in Fig. 2) and back to a distribution plenum in the reactor vessel by a centrifugal pump (PuA 1.2 in Fig. 2). The main heat exchanger is a conventional tube-type exchanger where the tubes go in loops. The heavy water circulates inside the tubes with a heat area is 129 m². The flow in the circuit is 235 m³/h and Δt is -6.15°C at an inlet temperature of 55°C and 2 MW power.

3.2. Secondary cooling circuit before modification

The secondary cooling water is circulated by one or two centrifugal pumps (PuB 1.1–1.2 in Fig. 2) through the main heat exchanger, the district heating circuit, and the cooling tower circuit heat exchangers (HEB 1.1–1.2 in Fig. 2). The district heating circuit and the cooling tower circuit are in parallel and the water is distributed between them by a regulating valve (VF 1 in Fig. 2). The flow in the circuit varies from 170 to 300 m³/h. The typical inlet temperature to the main heat exchanger is 37°C and $\Delta t = 9^\circ\text{C}$. The temperature of the water delivered to the district heating circuit is then approximately 46°C.

Requirements of the modified circuits are as follows:

- Flow and temperature conditions in the primary circuit shall remain the same as in the unmodified system.
- The pressure in the primary side of the main heat exchanger shall be higher than in the secondary at all operating conditions. This safety requirement is also valid for the unmodified system.
- The district heating circuit shall be isolated as a tertiary circuit.
- The temperature of the water delivered to the district heating circuit shall be the same or higher than in the unmodified system.

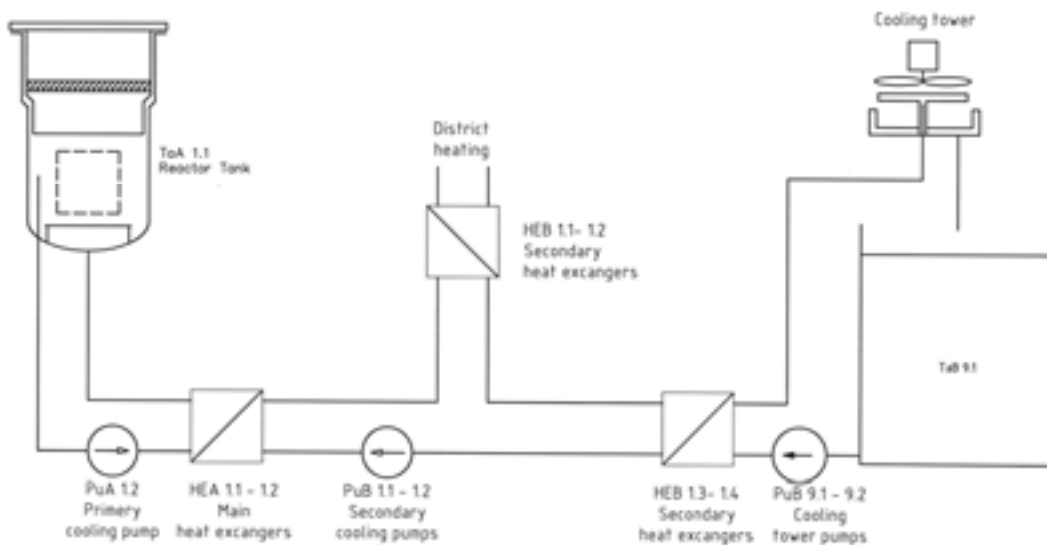


FIG. 3. Modified cooling circuits with district heating (simplified).

In the modified system, the district heating is separated as a tertiary circuit by two heat exchangers in parallel (HEB 1.1–1.2 in Fig. 3). These are in series with the cooling tower heat exchangers (HEB 1.3-1.4 in Fig. 3).

3.3. New main heat exchanger

The old tube-type heat exchanger was replaced with 2 plate heat exchangers in parallel (HEA 1.1–1.2 in Fig. 3). These are fusion-welded, stainless steel exchangers with 250 plates each. The heat transfer area is 74.2 m² in each exchanger. The outlet temperature on the secondary side of the exchangers is 54°C, which is 8°C higher than from the old exchanger. This increased temperature has resulted in a substantial increase in the amount of heat that can be delivered to the district heating system — up from around 1.5 MW to slightly more than 1.8 MW.

3.4. Heat exchangers for the district heating circuit

The heat exchangers for the district heating circuit se are similar to the main heat exchangers but with only 178 plates each. The outlet temperature on the tertiary side of the exchangers is 52°C, which is 6°C higher than the old exchanger.

4. PROJECT MANAGEMENT

The project was planned and accomplished as a maintenance task by the reactor staff. Consultants at the institute were used for the necessary calculations and dimensioning of new components. An external firm with certified craftsmen was used for all welding work.

A project group was established and worked for approximately six months to create a detailed scope of work for the project. The project was then presented to the institute's safety committee. After approval by the committee, the project was started as an ordinary maintenance task during the summer vacation and maintenance period and was planned to be finished in this period of time. Due to late delivery of two of the heat exchangers, this period had to be extended with several weeks.

5. LESSONS LEARNED

The reactor staff is fully capable of planning and accomplishing complicated modernization and refurbishment projects. Projects of this type help maintain and develop the staff's competence and motivation. When planning to accomplish this kind of project during a maintenance period, it is very important that procurement of all key components is completed prior to the start of the maintenance period.

6. FUTURE MODERNIZATION PLANS FOR THE COOLING CIRCUITS OF THE JEEP II REACTOR

To reduce the consumption of water and electric power, a replacement of the cooling tower by a dry cooler is planned. This will also eliminate the risk of development of harmful microbiological organisms. Regulating a dry cooler is also easier than a cooling tower. This modification is planned for the summer maintenance in 2008.

REFURBISHMENT AND POWER UPGRADE OF PAKISTAN RESEARCH REACTOR-1 (PARR-1)

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Abstract

The Pakistan Research Reactor-1 (PARR-1) was commissioned in 1965 at a power level of 5 MW. The reactor was originally designed for HEU fuel with 93% enrichment in the form of UAl_x-Al . The reactor equipment remained unchanged for the first 20 years of its operation. However, several factors, such as obsolescence of equipment and the non-availability of spare parts, forced a refurbishment programme of the facility. For this purpose, the old thermionic tube-based instrumentation and control system was replaced with new and modern instrumentation, the reactor pool was lined with stainless steel, and the old cooling system was removed and a new enhanced cooling system was installed for a power upgrade. In order to increase reactor safety, a new emergency core cooling system was installed, the reactor building was repaired, the HVAC system was improved, and a new compressor and additional diesel generators were installed. In order to meet international requirements, the reactor core was converted from HEU to LEU fuel. Due to changing experimental needs and demand for a higher neutron flux, reactor power was increased from 5 MW to 10 MW. The refurbished and upgraded PARR-1 achieved first criticality with LEU fuel on 31 October 1991 and full power of 9 MW on 7 May 1992.

1. INTRODUCTION

The Pakistan Research Reactor-1 (PARR-1) has operated at PINSTECH since December 1965 utilizing 93% ^{235}U HEU fuel. The reactor equipment remained essentially unchanged for the first 20 years of its operation. However, there were several factors that forced the facility to take up a renewal programme. These factors included the non-availability of spare parts, obsolescence of equipment, etc. It was proposed that the instrumentation and controls (I&C) system be modernized according to the current technology and to meet the current stringent safety and operation needs. The instrumentation needs were reassessed taking into account a higher power level, monitoring more parameters, and increasing the accuracy of display parameters to be measured. The number of neutron flux monitoring channels was increased to ensure reliable indication of all operating ranges and for the sake of redundancy and diversity.

The renovation work comprised a review of I&C system needs in light of the latest design practices in the field. As a result, a revised I&C configuration and logic was worked out. The procured instruments and locally developed modules and channels were first assembled and tested in the laboratory with signal sources. The instrument channels were then tested under actual operating conditions in the field and their performance compared with the old channels. After actual environmental testing and performance according to the requirements, the whole I&C system was replaced and commissioned in February 1986.

In 1976, the supply of HEU fuel was discontinued by the USA and other fuel exporting countries and it became essential to convert the core from HEU to LEU fuel. In view of changing experimental needs and isotope production at high neutron flux, it was also decided to upgrade the reactor power from 5 MW to 10 MW. A project on core conversion and power upgrade was initiated in 1988. Detailed neutronics, thermal hydraulics, and accident analysis studies were carried out for 19.99% enriched U_3Si_2-Al fuel. The reactor was shut down in September 1990 for core conversion to commercially available LEU fuel and

refurbishment of the reactor systems. Major modifications and additions were made in the heat removal systems. These included installation of new primary pumps, heat exchangers, and a cooling tower. To ensure safety in case of loss of coolant accident (LOCA), an emergency core cooling system was installed. The reactor pool and hold tank were lined with stainless steel to eliminate pool seepage. The reactor building was repaired to improve the containment and the HVAC system, diesel generators, and compressors were replaced. All the modified and renovated systems were re-commissioned. After renovation and upgrade, the reactor achieved its first criticality with LEU fuel on 31 October 1991, a power of 9 MW on 7 May 1992, and full power of 10 MW on 27 February 1998.

2. STRATEGIC JUSTIFICATION FOR THE MODERNIZATION AND REFURBISHMENT PROJECTS

2.1. Modernization of the I&C system

The original I&C system was based on the old thermionic tube instruments, non-compatible sensors, transducers, and transmitters. There were several factors that forced a renewal programme for the complete renovation of the I&C system. These factors included:

- Non-availability of spare parts and obsolescence of equipment.
- Reactor had to be shutdown whenever any of the startup or power monitoring channels became unavailable.
- Too many spurious scrams (shutdown) were experienced due to non-redundant logic for the protection channels.
- Process instrumentation was generally found to be inadequate and was the case with area radiation monitors.

The desirable features of the existing instrumentation were retained, while others were modified, improved, and supplemented to ensure safe and reliable operation of the reactor. The entire design was based on the principles of redundancy and diversity.

2.2. Stainless steel lining of the reactor pool

Water seepage through the pool walls was experienced soon after the commissioning of PARR-1 in 1966. The seepage areas were treated with water-proofing compounds. The seepage was controlled but it could not be eliminated in spite of various repair efforts and the conditions continued to deteriorate. Therefore, it became necessary to resolve this problem by lining the entire pool with stainless steel.

2.3. Core conversion and power upgrade from 5 MW to 10 MW

The reactor was originally designed for 93% HEU fuel in the form of UAl_x-Al at 5 MW(th) power. In the late 1970s, due to non-availability of HEU fuel and requirements of the IAEA, it became essential to convert the core from HEU to LEU fuel.

In view of changing experimental needs and isotope production at high neutron fluxes, it was decided to upgrade the reactor power from 5 MW to 10 MW. For this purpose, major modifications were made in the heat removal and other support systems.

3. RENOVATION AND MODERNIZATION OF INSTRUMENTATION AND CONTROLS

In early 1984, the reactor instrumentation development programme began to take up renovation of the I&C system of the PARR-1. The old system had obsolete, tube-type electronics and was posing operation problems and also problems in repair and maintenance. The I&C system was completely redesigned and replaced with solid state electronics. The new instrumentation was designed so that all probable accidents and initiating parameters were reliably covered.

The number of channels for measuring neutron flux and process variables was increased to provide requisite redundancy and diversity [1]. A number of channels were duplicated and a few new channels were added such as ^{16}N channels, secondary flow measuring channels, etc. Most of the process instrumentation was replaced and increased. The scope of radiation measurement and monitoring was extended. A CCTV system was installed for the surveillance of the building and reactor operations. Communication within the building and outside was improved.

The standard criteria for single failures — fail-safe design, redundancy, diversity, and channel independence were incorporated to the extent that they were advantageous and feasible. The new I&C system installed in 1986 has performed very well. Apart from the initial high failure rate, which is typical of a new system, instrument-related unscheduled scrams have been considerably reduced.

4. LEU CORE ANALYSIS (DESIGN PROCESS)

Conversion of PARR-1 from HEU to LEU fuel with a power upgrade, demanded detailed reactor design calculations pertaining to core neutronics, thermal hydraulics, accident analysis, and radiological consequence analysis [2–5]. The HEU core of PARR-1 utilized 93% enriched uranium fuel in the form of $\text{UAl}_x\text{-Al}$. The ^{235}U enrichment of the new fuel for PARR-1 was chosen to be 19.99%, which is regarded as upper limit for LEU fuel. The transition from HEU to LEU fuel for the same ^{235}U content involved an increase in the amount of ^{238}U , which enhanced neutron absorption. To compensate for this parasitic absorption and to achieve higher burnup, the ^{235}U content per fuel element had to be increased. The requirement of uranium loading, therefore, increased many times. A comparative study of several fuel element designs having different meat materials, uranium loadings, and number of fuel plates was carried out to optimize the neutron flux, fuel burnup and reactor power without compromising the safety of the reactor.

Initially, $\text{U}_3\text{O}_8\text{-Al}$ fuel was considered for the upgraded PARR-1. The fuel meat had a uranium density of 3.1 g/cm^3 , corresponding to 270 g of ^{235}U uniformly distributed in 23 plates of a standard fuel element. This was very close to the upper limit for $\text{U}_3\text{O}_8\text{-Al}$ fuel. However, to achieve a burnup of about 35% it was found necessary to increase uranium loading to about 3.3 g/cm^3 corresponding to 290 g of ^{235}U per standard fuel element. It was, therefore, necessary to use $\text{U}_3\text{Si}_2\text{-Al}$ fuel instead of $\text{U}_3\text{O}_8\text{-Al}$. For silicide fuel this density requirement was much below the limiting value. The neutronics, thermal hydraulics, and safety calculations were, therefore, repeated for silicide fuel. Based on these calculations and keeping in view the constraints and specific requirements of PARR-1, standard fuel element having 23 fuel plates and a loading of 290 g ^{235}U in the form of $\text{U}_3\text{Si}_2\text{-Al}$ was selected. The control fuel element was designed with 13 fuel plates plus 2 aluminum guide plates and contained 164 g ^{235}U .

5. CONSTRUCTION OF SPENT FUEL STORAGE BAY

In order to make the reactor pool area accessible for stainless steel lining, all the spent fuel elements and reactor components had to be removed and stored in a separate storage area. Therefore, a separate spent fuel storage bay was constructed. Lining of the bay with stainless steel was started in June 1990 and completed in November 1990. The bay was filled with demineralised water and commissioned for fuel storage.

6. FABRICATION OF FUEL CASK

A cask was designed and fabricated for transporting the irradiated spent fuel elements from the reactor pool to storage bay. It was designed on the basis that the dose rate outside the cask would not exceed $10\mu\text{Sv/h}$. The cask can contain 4 irradiated fuel elements at a time.

7. PARTIAL DECOMMISSIONING FOR CORE CONVERSION AND POWER UPGRADE

Partial decommissioning of PARR-1 was required before incorporating improvements in the facilities and installation of equipment for the core conversion and refurbishment activities. For the power upgrade, this activity was limited to cooling systems and process instrumentation. However, for the stainless steel lining on the interior of the reactor pool, decommissioning activities were extended to make the pool accessible. Decommissioning procedures were prepared for dismantling major components and radiation protection measures were taken [6]. Highly active reactor components (pool water, pool tiles) were analyzed for radiation and contamination and dose rates were measured on beam tubes, pneumatic rabbit tubes, thermal column extension, thermal shield, grid plate and graphite reflector elements.

The reactor was shut down in September 1990 for partial decommissioning. The decommissioning activities began with the removal of the through tube and beam tubes. After a 3-month cooling period, dismantling of the last HEU core began in November 1990. Spent fuel and graphite reflector elements were transferred to the wet storage bay via the fuel transfer cask. Control rod blades were also transferred to the wet storage bay in a special shielding arrangement made on a trailer shifting one blade at a time. The neutron source was transferred to wet storage bay in a specially made wax shielded container. Afterwards, the control room was electrically isolated and sealed to protect it from dust and contamination during the decommissioning and stainless steel lining of the pool.

Beam tubes, pneumatic rabbit system and thermal column were removed and stored in well-shielded enclosures. The core support frame, grid plate, and plenum were found to be highly radioactive. Due to its large size, high dose rate, and alignment problems, the removal of the core support tower was very carefully planned. The dose was determined and a shielded storage area was prepared in the reactor hall. The tower was disconnected at the upper portion near the bridge, lifted with small hoist, cleared from the pool top, and hanged vertically in a shielded storage area. Pool water was sent to a seepage pit. The entire primary cooling system was dismantled, except the embedded portion. In the pump room, all the piping, valves, and pumps were dismantled except the heat exchangers. Secondary piping was partially dismantled near the cooling tower. Pumps and valves were overhauled, packed, and stored in the pump room. Contaminated aluminum piping was placed in a separate location.

Adequate measures were taken to protect the personnel working around the reactor, the general public, and the environment from radiation. All workers were provided with TLDs, dosimeters, dungarees, overshoes, gloves, and masks as required. Additional ventilation was provided in the working areas. Outside radiological monitoring and periodic radiation surveys were carried out. After the completion of daily work, personnel were monitored for contamination and radiation exposure.

8. STAINLESS STEEL LINING OF THE REACTOR POOL

Seepage of water through the pool walls was experienced soon after the commissioning of PARR-1 in 1966. The seepage areas were treated with water-proofing compounds. The seepage was controlled, but in spite of various repair efforts, it could not be eliminated. Therefore, it became necessary to solve this problem by lining the entire pool with stainless steel. The work was planned during the shutdown period for core conversion and power upgrade. All the radioactive reactor components and fuel elements were removed from the reactor pool and the interior of the pool was made accessible for working personnel. The pool walls and the floor were lined with 3 mm and 6 mm stainless steel sheets, respectively. A leak check of the pool lining was carried out with dye penetrant and vacuum tests along the weld seams. The entire lining was found satisfactory. All the welding seams were thoroughly cleaned and passivated to enhance corrosion resistance. The reactor pool was filled with demineralised water and thoroughly checked for seepage.

9. REACTOR CONTAINMENT BUILDING

During the shutdown period for core conversion and power upgrade, special attention was given to minimize the leakages from the reactor building penetrations. Necessary building repairs were carried out to keep the leak rate within limits. There was seepage at a few points in the hemispherical part of the building. For this purpose, the upper insulating layer of the hemispherical part of the reactor building was removed. The structural concrete of the dome was coated with fibreglass reinforced neoprene, followed by a thick layer of insulating concrete blocks, and finished with steel reinforced plaster and painted. All airlock door seals were replaced and air supply and exhaust damper seals were improved.

10. REPLACEMENT OF EQUIPMENT FOR GENERAL SERVICES

During the reactor power upgrade project, some general service equipment, such as HVAC, electrical systems, diesel generator, and compressors needed replacement. Therefore, air handling units (air supply units), chillers, and cooling tower were replaced. An additional new compressor with an air dryer was installed. One emergency diesel generator was available and another emergency diesel generator was installed as a standby.

11. RENOVATION OF THE COOLING SYSTEM

The cooling system capacity of the PARR-1 was increased to meet the requirements of the planned power upgrade to 10 MW. Both the primary and the secondary cooling systems were redesigned and a major part of the piping was changed [2]. A locally fabricated set of heat exchangers with additional pumps was installed and a new cooling tower was constructed.

12. INSTALLATION OF THE EMERGENCY CORE COOLING SYSTEM

The emergency core cooling system (ECCS) was designed and installed to remove the core decay heat in the event of a loss of coolant accident (LOCA) and to spray water on the core before exposure to air to prevent a core meltdown. The system consists of two redundant water pumps located in the pump room and a set of spray headers with nozzles, mounted one meter above the normal pool surface level directly above the core. Both pumps are also powered by the emergency electrical supply system. During normal reactor operation and the reactor shutdown, the ECCS is in a ready to operate state with its pumps in auto mode. The system will automatically operate if the pool level falls one meter below the normal level. Only one pump operates at a time, while the other remains in standby. In the case of a low pool level signal, if one pump fails to operate, the second pump will begin to operate after a 10 s delay.

13. COMMISSIONING OF UPGRADED PARR-1

The reactor was partially decommissioned in November 1990 to make preparations for the installation of the upgraded LEU core. The reactor pool interior was lined with stainless steel. All the beam tubes, through tube, thermal column extension, thermal shields, and inlet/outlet frames were reinstalled. The core support tower along with the grid plate and plenum, which was removed from the reactor pool and stored in the reactor hall, was reinstalled. The first LEU core was loaded and made critical on 31 October 1991. Later, several core configurations were assembled in the open and stall ends of the pool and low power experiments, such as reactivity measurement, control rod calibration and flux mapping, were performed. Initial tests and measurements indicated that various systems of the reactor performed according to the design specifications and most of the experimental results were in agreement with the theoretical design. After completion of the low power tests, a core for normal power operation was assembled. Since most of the old systems had been improved and replaced with new and better equipment and more stringent safety conditions had been applied, the upgraded PARR-1 now serves as an improved research facility and will be good for about another two decades.

14. IMPROVEMENTS IN PERFORMANCE AFTER MODERNIZATION AND REFURBISHMENT WORK

14.1. Instrumentation and control system

The instrumentation and control system was redesigned and replaced with solid state electronics. The number of channels for measuring neutron flux and process variables was increased to improve redundancy and diversity. Safety systems set point limits and interlocks were kept according to the old instrumentation. However, scram for the ^{16}N channels were also provided at 120% nominal power. In the renovated I&C system, the startup channel, the linear-N channel, and the log-N channels have been duplicated. Normally, these channels are operated such that a safety action is initiated when a safety limit is exceeded on any of 2 channels. In the even one channel is faulty, the other is available and the reactor can be operated with one channel (faulty channel is bypassed). This has greatly improved the reactor availability factor. Nitrogen-16 and secondary flow measuring channels have been added in the system. The number of safety channels is the same as before but are operated in a 2/3 mode instead of a 1/3 mode; similarly, three ^{16}N channels are operated in a 2/3 mode (Table 1). This has almost eliminated spurious scrams and the number of unscheduled shutdowns has been considerably reduced.

TABLE 1. SAFETY CHANNELS

Channel	Number of Channels		Safety Action
	Old	New	
Startup	1	2	Rod withdrawal inhibit, Rod reverse
Linear-N	1	2	Rod reverse
Log-N	1	2	Interlock, Period scram
Safety	3	3	Scram
¹⁶ N	0	3	Scram
Pool water radioactivity	1	1	Scram
Coolant flow	1	2	Scram
Coolant temperature, ΔT	1	2	Rod reverse
Pool level	1	3	Scram
Primary pump failure	1	2	Scram
Reactor hall pressure	1	1	Scram
Earthquake	1	1	Scram
Reactor bridge unlocked	1	1	Scram

14.2. Stainless steel lining of the reactor pool

After installation of the stainless lining in the reactor pool, water seepage was completely eliminated. There is no water loss due to seepage and the water quality has improved. The number of regenerations of the pool recirculation demineraliser has been reduced. The ugly look of water drizzling from exterior of the pool and creating problems for the experimenters is gone.

14.3. Core conversion and power upgrade

Conversion of the reactor core from HEU to LEU fuel was an IAEA requirement and has been completed. Moreover, this opportunity was used to upgrade reactor power from 5 MW to 10 MW. The core was redesigned and a central irradiation facility was created in the centre of the core, where the thermal neutron flux is on the order of $1.5 \times 10^{14} \text{ n}\cdot\text{cm}^{-2}\cdot\text{s}^{-1}$. Previously, the samples were irradiated at the periphery of the core and required long irradiation times because of the low neutron flux at that location. The high neutron flux requirements for research and isotope production are being fulfilled.

15. LESSONS LEARNED

The difficulties in decommissioning and events after re-commissioning led to very important lessons:

- Before taking up any major activity or a series of activities, the prerequisites must be very carefully considered. Discussion among the experienced personnel and careful planning is essential before starting of the project.
- Drawings and data must be properly stored and readily available. Lack of information can make simple tasks difficult and time consuming. The utmost care must be taken to keep a good record of the data, especially drawings.

- Minor mistakes or ignorance can lead to major consequences and checklists must include checks to rule out such a possibility. Each single activity must be checked by at least one person higher in rank to the person performing the activity and this must be included in the procedures.
- Effective communication among the workforce and at all levels of management is vital. Procedures must ensure that such communication exists.
- Photography/telephotography must be done of important activities so that the systems and components can be easily understood in future. This becomes more important when drawings and data are missing.
- All activities carried out should be properly documented for future use.
- Installing a stainless steel lining in the reactor pool from the very beginning could have saved a lot of time.

16. CONCLUSION

The reactor power has been upgraded from 5 MW to 10 MW. The instrumentation and controls of PARR-1 are now based on solid state electronics and components are readily available. The core of PARR-1 has been converted from HEU to LEU fuel. After installation of a stainless steel lining in the reactor pool, leakage from the pool has been completely eliminated. Additional safety channels have been added. Spurious scrams and unscheduled shutdowns have been significantly reduced. Reactor availability factor has improved. A higher neutron flux in the irradiation facilities has helped the researchers and isotope production.

REFERENCES

- [1] KARIM, A., PERVEZ, S., Renovation and upgradation of PARR-1, The Nucleus 29 1, Nilore, Pakistan (1992) 7.
- [2] KHAN, L.A., ISRAR, M., ARSHAD, M., KARIM, A., AKHTAR, K.M., MOQUITE A., Pakistan Research Reactor-1, Final Safety Analysis Report for Conversion to LEU Fuel and Power Upgradation, Nilore, Pakistan (1992).
- [3] KHAN, L.A., ARSHAD, Design and engineering of LEU fuel for PARR-1, The Nucleus 29 1, Nilore, Pakistan (1992) 13.
- [4] KHAN, L.A., QAZI, M.K., BOKHARI, I.H., FAZAL, R., "Analysis of the Critical and First Full Power Operating Cores for PARR-1 Using LEU Oxide Fuel", Pakistan Institute of Nuclear Science and Technology, PINSTECH-114, Nilore, Pakistan (1988).
- [5] ARSHAD, M., BAKHTYAR, S., HAYAT, T., SALAHUDDIN, A., Neutronic calculations of PARR-1 cores using LEU silicide fuel, Pakistan Institute of Nuclear Science and Technology, PINSTECH-119, Nilore, Pakistan (1991).
- [6] ISRAR, M., SHAMI, Q.D., PERVEZ, S., Partial Decommissioning of PARR-1 (5 MW) for Core Conversion and Upgradation, NED-302, Nilore, Pakistan (1993).

PLANNING AND INITIATION OF AN I&C SYSTEM UPGRADE FOR THE PORTUGUESE RESEARCH REACTOR

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Abstract

The Portuguese research reactor (RPI) was converted to LEU fuel in 2007 and will continue to operate until at least 2016. A project to replace its nuclear instrumentation and control (I&C) system is currently underway. As in other nuclear installations, discussions were held on keeping an analogue system or migrating to a digital one. A final decision was made to acquire an analogue system that was functionally compatible with the current system and to develop a computer interface in-house for non-safety-related functions. The selection of a supplier was made and, as required by budget restrictions, a partial procurement of the main modules started in 2005. Several actions have been taken in recent years regarding the modernization of the radiological protection system of the RPI. In this paper, we will discuss the lessons learned with the major refurbishment done from 1987–1989 and plans for the modernization of the nuclear I&C system and the radiological protection system of the RPI.

1. INTRODUCTION

The Portuguese research reactor (RPI) is owned and operated by Instituto Tecnológico e Nuclear (ITN). It was built by AMF Atomics from 1959–1961 and its design closely follows that of the Battelle research reactor. The activities currently underway in the RPI cover a broad range from the irradiation of electronic circuits [1] to the calibration of detectors in the search for dark matter [2], with a strong emphasis on neutron activation analysis [3, 4]. The RPI was converted to LEU fuel in 2007 [5] and will continue to operate until at least 2016.

An extensive refurbishment of the reactor was done in the late 1980s [6]. The dominant reason for the refurbishment was the replacement of the primary circuit piping that had developed a leak and limited the maximum power of the reactor to 100 kW. Since this repair would interfere with the structure of the pool, it was decided to enlarge the scope of the intervention. Other changes were made, namely:

- introduction of a steel liner in the stall end of the pool;
- replacement of tiles and support epoxy in the open end of the pool;
- replacement of secondary piping and cooling tower;
- replacement of control rods and control rod drives;
- replacement of control rod magnets from underwater to out of water;
- introduction of a core drop pressure meter, providing functional redundancy for loss of primary flow;
- installation of an emergency diesel generator;
- replacement of the radioprotection monitoring system.

No changes were made to the nuclear instrumentation and control (I&C) system that had been installed in 1972. A project to replace the nuclear I&C system is currently underway to increase the reliability of this system.

2. EVOLUTION OF THE NUCLEAR I&C OF THE RPI

The initial I&C system of the RPI was based on vacuum tubes and relays. Neutron detection was provided by 5 channels: the startup channel using a fission chamber (FC), the log-N and period channel using a compensated ion chamber (CIC), the linear power level and automatic control channel using another CIC, and 2 power and safety channels using uncompensated ion chambers. The useful range of the power and safety channels was restricted to powers above 100 kW. The reactor period information provided by the log-N and period channel was used for safety actions.

Strip chart recorders were used for displaying the readings of the startup, log-N, and linear power channels. Some of the thresholds for safety actions were included in the recorders, e.g. 110% of nominal power in the log-N channel to initiate a reverse protective action. The thresholds for power and period levels that would cause a scram protective action were all handled by the safety amplifier.

From the early days, it was decided that this I&C should be upgraded and changed to transistors and electronic logic. The in-house development of the new system began in the late 1960s in collaboration with CEA, France. In essence, the modification replaced not only the vacuum tubes and relays, but also the safety philosophy. The startup and linear power level channels were maintained, while 3 log/lin power channels, using CIC, were introduced for both period and power information, with 2/3 coincidence logic to initiate safety actions.

The startup channel uses a CFUL01 fission chamber (1 g ²³⁵U) operated in pulse mode, supplied by Radiotechnique Compelec, France. The log/lin channels follow the reactor power over 7 decades, from 0.2 W to 2 MW, corresponding to a current range from 10 pA to 100 μA. The target current for 1 MW is 50 μA. Three WL-6377 high purity magnesium alloy, boron-lined ionization chambers are used for these channels. Their neutron sensitivity is approximately $4 \times 10^{-14} \text{ A}\cdot\text{n}^{-1}\cdot\text{cm}^{-2}\cdot\text{s}^{-1}$. The gamma sensitivity is approximately $3 \times 10^{-11} \text{ A}\cdot\text{R}^{-1}\cdot\text{hr}^{-1}$ when operated uncompensated and is reduced by two orders of magnitude when compensated. The WL-6377 was initially supplied by Westinghouse and later by Imaging & Sensing Technology Corporation, Horseheads, NY. The change of safety actions from the startup channel to the power channels is manually done after the log/lin channels show a power above 4 W.

Commercial modules from the Multibloc series of Merlin Gerin Provence, France (MGP) were used to process the signals from the neutron sensors and modules made in-house were used for analogue-to-digital conversion, display, and logic functions. Strip chart recorders were kept as central pieces of the control room, but no longer handled any safety-related functions. Discrete semiconductors were used to implement AND and OR functions in negative logic. A few relays were retained to perform NOT functions when required. Care was taken in the design phase to build simple circuits with few critical components that were easy to maintain.

The whole I&C is distributed into 18 vertical drawers. Each neutronic channel is split into 2 drawers; one housing the analogue treatment of the signal from the neutron detector (amplifiers, differentiator, etc.), and the other housing the processing and control of the signals. The safety logic is also housed in 2 drawers; one housing the interlock and inhibit actions, and the other housing the reverse and scram actions.

This system was installed in about two months in 1972. The layout of the initial AMF console remained essentially the same. For each drawer, there was a spare and an

exchange was performed every 6 months. Maintenance problems increased in the late 1980s due to a reorganization that integrated part of the reactor electronics group in another campus. This contributed to a decrease in preventive maintenance and a gradual increase in curative maintenance and downtime of the reactor. In the 1990s, this situation deteriorated when a further reorganization made ITN independent and weakened the connection with the group responsible for the I&C maintenance. This trend was stopped in 1997 when a small in-house team performed I&C maintenance and the preventive maintenance programme was revised [7]. Meanwhile new reactor operators were recruited and received I&C maintenance training, which had not been provided to the older operators.

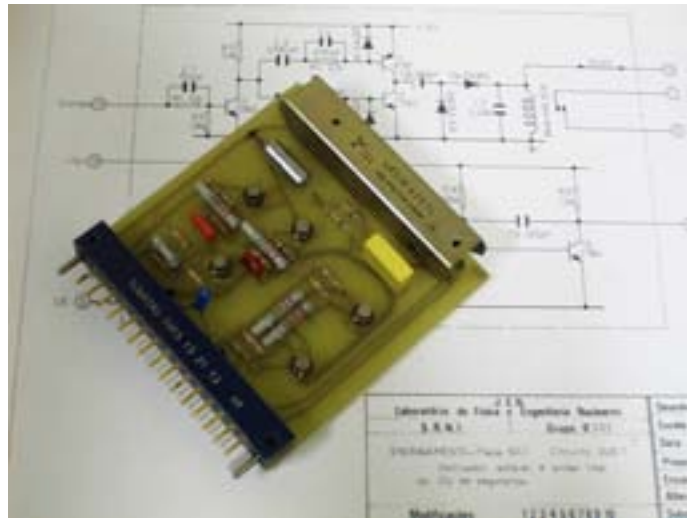


FIG. 1. Typical I&C in-house made board.

Figure 1 shows a typical in-house made board, using simple transistor-based circuits, which activate a reed relay at the final stage. Each logic circuit has a 20 kHz square wave generator. When all conditions are normal, this signal is transmitted via a cascade of NPN transistors to the final stage, rectified, and keeps the relay energized. Should any abnormal conditions occur (including a transistor malfunction), the amplitude of the rectified signal decreases and the relay is de-energized. The board shown above houses both the generator and the final stage.

The documentation prepared in the 1970s, although with some gaps, has proven to be sufficient to perform preventive and curative maintenance with a completely different team. Electronic components that were showing frequent failures (e.g. transformers) were replaced. Stocks of electronic components were re-established, with difficulty in some cases. In the case of the reed relays that were no longer manufactured, it was necessary to acquire similar ampoules and re-qualify the relay.

Several units were replaced without major problems. The obsolete strip chart recorders were replaced by DPR180 units from Honeywell. The Rosemount DP1151 pressure sensors used for measurement of primary flow and core drop pressure were replaced after approximately 15 years of use. In the first case, replacement was due to a deficient zero adjustment; in the second case, replacement was to install a sensor with a more appropriate range. A new Rosemount DP1151 ‘smart’ sensor with a range between 0–3.56 m H₂O is now used for measurement of primary flow (corresponding to a flow range of 0–227 m³/h) and a DP3051 sensor with a range of 0–60 cm H₂O is used for measurement of core drop pressure across the plenum. Given the fact that the DP1151 ‘smart’ sensor has a lot more embedded

electronics than previous models, a lead shield was installed in the pump room to decrease the gamma field by a factor of 5 at its location (on the order of 3–4 mGy/h before shielding).

A linear ^{16}N channel was recently added to the I&C system as a complement to the thermal channel of the RPI. This channel allows an easier adjustment of the power safety trip settings to compensate for the weekly ^{135}Xe cycle. No protective actions are derived from the ^{16}N linear channel. A gamma ionization chamber, model LND 50316 (0.05 m dia., 0.14 m active length), with a nominal sensitivity of $1.3 \times 10^{-9} \text{ A}\cdot\text{R}^{-1}\cdot\text{hr}^{-1}$ for ^{60}Co , was installed in the pump room next to the hot leg of the primary circuit. An electrometer, Keithley model 6514, was used to display directly the data in the control room. The current at 1 MW is approximately 500 pA. While the thermal channel is only usable from powers above 300 kW, the ^{16}N channel is usable from powers above 10 kW. The lower power limit was decreased by a factor of 5 through the installation of the signal cable inside a separate metal conduit throughout the reactor hall. A more sensitive ^{16}N channel would require a gamma ionization chamber with a much larger volume, such as model LND 50410 (0.18 m dia., 0.40 m active length), which has a nominal sensitivity of $1.3 \times 10^{-8} \text{ A}\cdot\text{R}^{-1}\cdot\text{hr}^{-1}$ for ^{60}Co [8].

3. UPGRADE OF THE NUCLEAR I&C OF THE RPI

The current nuclear I&C system is over 30 years old and its upgrade is under preparation. As in other nuclear installations, discussions were held on whether to keep an analogue system or migrate to a digital system. The general opinion was that it was the right time to introduce a digital system, as modern instrumentation must be microprocessor-based. Actual proposals from two representative companies were considered for a preliminary evaluation. One company proposed an analogue system functionally equivalent to the current one, while the other company proposed a digital system. Table 1 lists the main factors considered in the preliminary evaluation.

TABLE 1. FACTORS CONSIDERED WHEN COMPARING AN ANALOGUE AND A DIGITAL I&C SYSTEM AT THE RPI

Factor	Analogue system	Digital system
Changes in control room	Minimal	Significant
Retraining for operation	Minimal	Significant
Retraining for maintenance	Minimal	Significant
Local upgrade	Possible	Not possible
Integration with current system	Possible	Not possible
Obsolescence in 10 years	Not likely	Very likely
Cost	Moderate	High
Licensing effort	Minimal	Significant

At the time of the preliminary evaluation, all senior operators were close to retirement age, so factors such as extensive changes to the layout of the control room and the need for extensive retraining carried a big negative weight. These two factors clearly favoured the analogue option, as the digital option would entail radical changes in many aspects.

Other factors considered were the capabilities for local maintenance as well as for local development and upgrade. The local capabilities were judged to be appropriate for the maintenance and for limited upgrade in the analogue system only. Another issue addressed was the possibility for integration with the current system if budget restrictions would not

allow the purchase of a complete system. Although this would be a time consuming effort, it was considered possible only with the analogue system.

Cost and licensing efforts also favoured the analogue option. The direct cost of the digital option was about 3 times the cost of the analogue option. This ratio would significantly increase with the changes in the control room (e.g. new ventilation) and retraining of operators and maintenance staff. On the other hand, the licensing effort was considered to be significant in the case of a digital system, for which no previous experience exists. In contrast, the analogue option offers a function-by-function replacement at all levels, where it is easier to demonstrate that the new units or modules meet and normally exceed the specifications of the existing ones.

A final decision was made to acquire an analogue system, functionally compatible with the current system. A computer supervision interface for non-safety-related functions can be developed in-house and added later. This option combines a state of the art analogue control system with an easily upgradeable computer interface.

The only significant change that is planned in the control system is the introduction of a second startup channel. The two channels will be connected in a 1/2 logic. This action was recommended refurbishment in the 1980s [9], but was not implemented because changes in the control system would have been significant and the local engineering resources were already limited. The addition of yet another startup channel, connected in a 2/3 logic was not considered necessary.

Figure 2 represents the main blocks of the new I&C system, with 19 in drawers from Thermo Fisher Scientific (formerly known as GammaMetrics), USA, as well as the corresponding neutron detectors. All drawers are 1E qualified. For decades, this company has been a supplier of drawers and full systems for research and power reactors, including a full system (console control and safety system) in 2007 for the McMaster reactor (Canada), which is similar to the RPI.

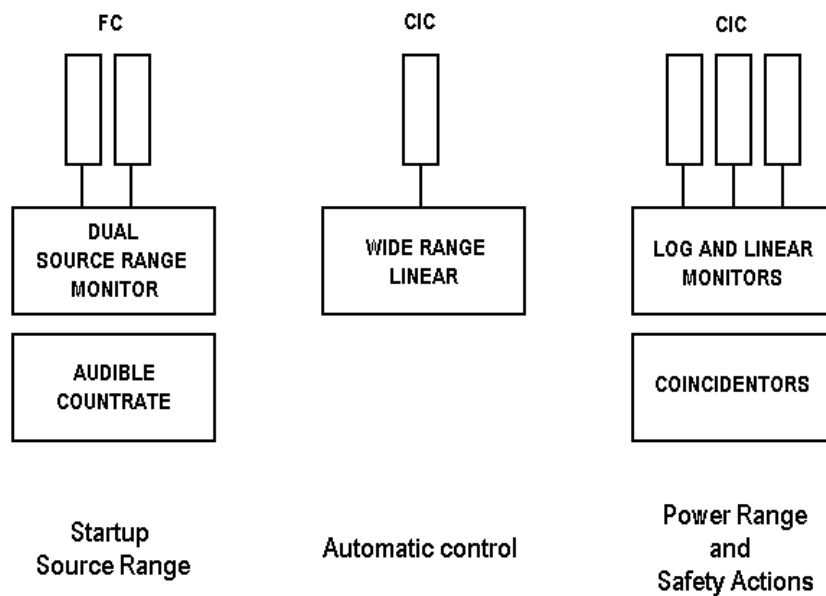


FIG. 2. Basic modules of the new I&C. A total of six neutron detectors will be used.

A custom-made Dual Source Range Monitor will be used in startup/source range. Two unguarded fission chambers, model WL-23110 (0.4 g ^{235}U , 0.18 cps/nv) will be used in this range. The fission chambers will be fixed (currently, the fission chamber is moveable). An audible count rate module will also be installed to keep this feature of the current system.

One TR40 Wide Range Linear Monitor, coupled to a CIC (type WL-6377 or equivalent), will be used for automatic control. The TR40 covers 8 decades, corresponding to a detector input range from 1 pA to 125 μA , i.e. an order of magnitude more sensitive than the current channel for automatic control. The target current for 1 MW is 100 μA (adjustable).

Three TR30 Log and Lin Monitor, coupled to 3 CIC (type WL-6377 or equivalent), provide power and period monitoring. The log/lin monitors also cover power over 8 decades on a log scale, corresponding to a detector input range from 1 pA to 100 μA , i.e. an order of magnitude more sensitive than the current log/lin channels. It covers a range from -30 s to 3 s, as well as power on a linear scale from 0–125%.

Trips for power and period will be combined in two coincidence modules for protective actions. The coincidence modules can handle 12 groups of signals in a 2/3 logic, while the current system requires only 7 groups. Some of the extra 5 groups can be used to provide duplication of specific safety actions. It is also possible to create a new group, composed of 3 core pressure drop sensors. The addition of 2 core pressure drop sensors to the existing one, connected in a 2/3 logic will increase the reliability of this system. The RPI has no record of spurious scrams from the existing sensor with the HEU cores, but the margin between the nominal core drop pressure and the safety trip setting was slightly decreased as a result of the safety studies for core conversion to LEU fuel. It is worth mentioning that as a conservative approach, a failure of the fastest acting device was considered in all postulated initiating events in the safety studies done for the LEU core conversion [10] so there are no special constraints on any of the protective actions.

The current I&C system has a small console placed in front of 4 equipment racks. It is planned that the new console will be integrated with the equipment racks. The main advantages of this approach are better access to all functions and a significant reduction in cabling. Figure 3 shows a preliminary layout of the console.

The preliminary layout shows the drawers distributed in 4 equipment racks, labelled A–D in Figure 3. Rack A houses the source range monitor, audible drawer, and the wide range linear monitor. A Yokogawa DX200 paperless recorder will complement the display in the source range monitor and maintain a vital link with the current console. Rack B houses a rod control panel, a rod position display panel, one of the log/lin channels, and a DX200 paperless recorder connected to the 3 log/lin channels. Rack C houses a plant control panel, two log/lin channels, two Yokogawa DX100 paperless recorders (one for the thermal and ^{16}N channels, the other for the temperatures on the primary and secondary cooling circuits), and 2 LFE2040 limit controllers (one for primary flow, the other for core drop pressure). Rack D houses two coincidence modules, the scram logic and magnet power supply, plus a plant status display panel, complementary to the one in the magnet power supply.

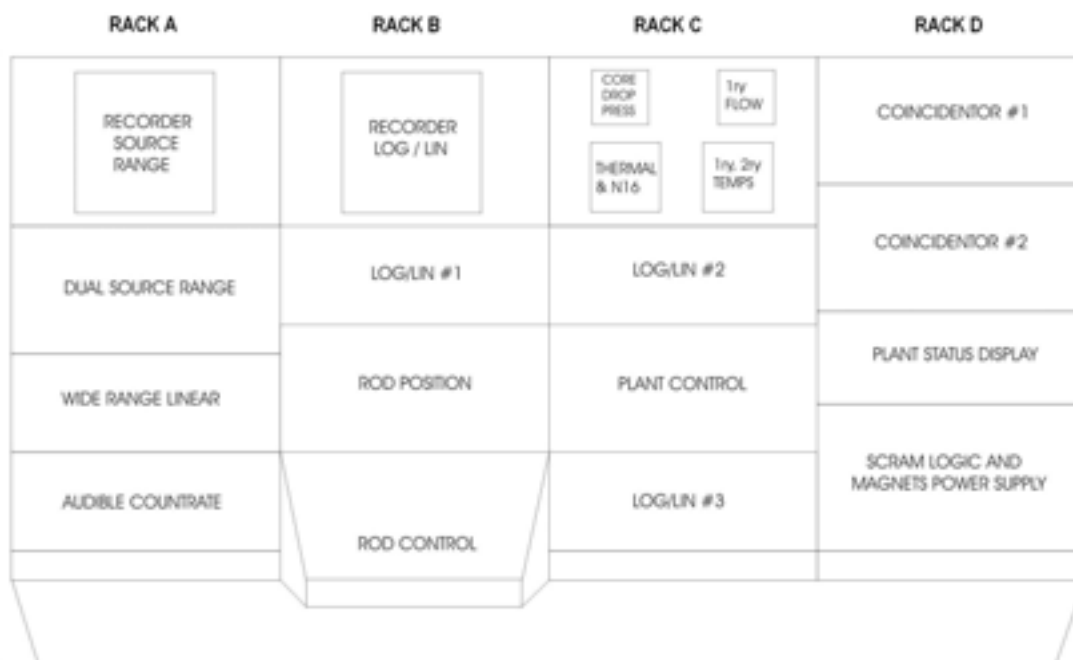


FIG. 3. Preliminary simplified layout of the console.

It is expected that the supervision interface will perform the following functions:

- alarm handling;
- report generation;
- event recording;
- plant status display;
- administrative data management;
- data communication;
- status and self-diagnostics.

Compared with a typical supervision and control system [11], it will not include plant control functions, limitation functions, and power regulation functions. All reactor protective actions will be hardwired and the supervision system will acquire data through isolated analogue outputs of the ThermoFisher Scientific drawers and through the Yokogawa paperless recorders. The paperless recorders include several communications options and will also function as data acquisition boards for items, such as the temperatures in the cooling circuits, which are not connected to protective actions.

The partial procurement imposed by budget restrictions has clear disadvantages:

- In principle, the overall price will be higher.
- Replacement takes longer.
- Factory acceptance test of the whole system is not possible, thus representing a higher risk for the facility.
- Regulatory approval may be delayed.

However, it may be the only possibility for facilities with limited budgets to get a state of the art I&C.

4. EVOLUTION OF THE RADIOPROTECTION SYSTEM OF THE RPI

The radioprotection system of the RPI has been in continuous evolution, including not only component obsolescence, but also new data display and storage needs, as well as improved calibration procedures. The original system consisted of 3 area ionization chambers. These were replaced at the time of the refurbishment in the 1980s with 5 C/EIP 51 chambers from MGP located in (i) core bridge, (ii) experimental area, left side, (iii) experimental area, right side, (iv) hot chemistry lab, and (v) control room. Their outputs are displayed in the control room using INR MV21 units, also from MGP. A CMDB beta measuring unit from MGP was installed in the western wall of the reactor hall at pool surface level to monitor aerosol atmospheric contamination using a thin plastic scintillator. Monitoring of the air activity concentration is performed on a continuous basis using a C/CAG 141 sensor from MGP for air contamination by radioactive gases located at the western wall of the reactor hall at pool surface level. An open ionization chamber with a volume of 10 L is used to detect the β -radiation of the passing air. All INR MV21 are read by a computer interface developed by CEA, Grenoble and installed in 2002. Data is archived and displayed in the control room and in the Health Physicist's office. Should the PC interface fail, a Honeywell DPR180 strip chart recorder is used for backup recording and display.

A stack monitor developed in the IAEA's laboratories in Seibersdorf was installed in 1991 under the Technical Cooperation programme. The AirMon91 [12] is a computer-based system designed to monitor gaseous effluent samples of radioactive particulates, iodine, and noble gases released from the stack. The system consists of 2 shielded sampling chambers, a noble gas detector mounted inside the chimney, a vacuum pump, an air flow meter, control valves, a programmable logic controller (PLC), amplifiers, timing single channel analyzer, and high voltage power supplies. A PC computer stores the acquired data, displays it in the control room, and writes periodic reports.

The sampled air first passes through a particulate chamber in which aerosols are caught by a paper filter and their β -activity is determined by an NE107A plastic scintillator. The air then passes through the iodine chamber where the elemental iodine is absorbed by a charcoal filter and the gamma radiation of ^{131}I is measured. A 0.18 m dia. NE102A plastic scintillator in the stack monitors the β -radiation emitted by the radioactive noble gas isotopes in the gaseous effluents. The detection limits are 0.034 Bq/m³ for particulates, 0.11 Bq/m³ for iodine, and 1000 Bq/m³ for noble gases.

Age has not been kind to the AirMon91, as:

- Calibration is cumbersome.
- Some components are obsolete.
- The IAEA can no longer provide replacement components.
- The ITN managed to secure only some spare components.

The PC was the first major component to become obsolete. The replacement of the XT-type computer (based on an Intel 8088 microprocessor) by a faster model made it necessary to significantly change the BASIC code provided by Seibersdorf, as timing was microprocessor-dependent through delay loops. The SPS400 PLC also had to be replaced by a newer model because ABB discontinued the product and its technical support. Although the hardware replacement was not a problem per se, the development of software emulating the communications protocol of the discontinued model was a resource-consuming task.

5. UPGRADE OF THE RADIOPROTECTION SYSTEM OF THE RPI

An inspection of the European Commission services in 2002 in the framework of Article 35 of the Euratom Treaty triggered a gradual upgrade of the radioprotection system of the RPI. The first action was the installation of a new ^{131}I monitor in the stack, followed by a new particulate monitor, also in the stack, and later by an additional particulate monitor in the reactor containment building. Redundant monitoring in the stack is now envisioned. Currently, the iodine and particulate channels of the AirMon91 provide redundancy for these quantities. Redundancy will also be installed for noble gas monitoring.

An IM 201S monitor (1E qualified) from MGP was installed in the stack for ^{131}I monitoring. The IM 201S uses a NaI scintillator facing an activated charcoal cartridge in which radioiodine is trapped. A radioactive source built into the NaI crystal allows compensation of any drift due to temperature changes. The spectrometry capability, based on a 1024 channel spectrum analysis, allows immediate and easy iodine isotope identification in case of an alarm. The measuring range is $3.7\text{--}3.7 \times 10^6 \text{ Bq/m}^3$. The system was provided with a calibration source and an easy-to-perform calibration procedure. Data from the IM 201S is transmitted to the reactor control room and to the Health Physics office.

The IM 201S removes several weak points in the iodine channel of the AirMon91:

- aerosols removed before iodine measurement;
- limitations in background subtraction;
- lack of a reliable and easy to perform periodic calibration procedure.

An ABPM 201L Alpha Beta Particulate Monitor, also from MGP, was installed for beta particulate monitoring. Calibration sources were also provided by the manufacturer. The ABPM 201L uses a double silicon detector that performs gamma compensation and a radial fin grid that limits the scattering of alpha particles (static compensation), which facilitates the compensation of the radon and thorium solid progenies by the processing algorithms (dynamic compensation). The measurement range is $1\text{--}10^7 \text{ Bq/m}^3$. A second ABPM 201L monitor was installed inside the containment building to replace the otherwise obsolete CMDB unit.

The noble gas monitor is a simple and reliable detector, essentially independent from the other components of the AirMon91. However, the used EMI 9623A photomultiplier tube is no longer manufactured and a suitable replacement could not be found making it necessary to change to a 0.13 m dia. tube with a corresponding reduction in efficiency by a factor of 2, which is still acceptable. A home built replacement is being built using a 0.13 m dia. BC400 plastic scintillator from Saint Gobain (equivalent to NE102) coupled to a Photonis XP4512B photomultiplier tube.

Improvements in two other systems are being evaluated:

- ionization chambers in the reactor hall;
- fission products monitor.

The MGP ionization chambers installed in the late 1980s are no longer manufactured. Additionally, the computer-based data acquisition system has shown that the chambers are noisy, especially at low power levels. The noise level could be reduced through the introduction of individual uninterruptible power supplies, but not to a satisfactory level. A

replacement by small digital survey meters, model FH 40GL, from ThermoElectron, is being evaluated. The FH 40GL uses a proportional counter, covering a range from 10 nSv/h to 0.1 Sv/h. It can be equipped with an external neutron probe for simultaneous gamma and neutron measurements in selected points. The FH 40GL allows direct local display and alarm, as well as full readout capability through a PC interface. The Atomic Institute of the Austrian Universities, Vienna, has recently installed such a system in its TRIGA reactor, using a locally developed software application for storage and display [13]. ITN has the full capability to perform a periodic calibration of this type of detector through its metrology lab.

The fission product monitor uses a NaI scintillator facing an anion exchange resin column. All signal processing modules, including the detector, were replaced during the 1980s refurbishment. It is planned to add spectrometry capability to the current system, for immediate and easy iodine isotope identification in case of an alarm. It is also planned to evaluate the performance of a lanthanum halide crystal [14], such as the BrillLanCe 350/380 from Saint-Gobain, which could advantageously replace the NaI scintillator, given its significantly better energy resolution (less than 4% for the 662 keV gamma ray from ^{137}Cs) and higher efficiency.

6. FINAL NOTES AND CONCLUSIONS

The main refurbishment in the 1980s was done at a critical time for the facility. As a result, main components of the RPI are less than 20 years old. This not only had an obvious technical impact on facility operations, but a beneficial effect on public opinion.

The decisions of which systems to modernize and replace were made after extensive consultations with local authorities and with independent experts provided by the IAEA. This was essential to prioritize the investments given the limited funds available. After 15–20 years later, it appears the best decisions were made for equipment replacement and modernization. One exception may be the primary piping system. At that time, it was rated for a maximum power of 10 MW, although no other components were rated for that power. The RPI has not had a power upgrade and there are no plans for a power upgrade, so it has not been possible to take advantage of this feature.

The radioprotection system of the RPI has been in continuous evolution, including not only component obsolescence, but also new data display and storage needs, as well as improved calibration procedures. Significant investments have been made in this system to comply with regulatory requirements.

A project for the complete replacement of the nuclear I&C system is currently underway, taking into account the age of the current system and planned operation of the RPI to at least 2016. A final decision was made to acquire an analogue system, functionally compatible with the current system and to develop a computer interface in-house for non-safety-related functions. The selection of a supplier was made and a partial procurement started in 2005 as required by budget restrictions.

REFERENCES

- [1] FRANCO, F.J., et al., Radiation tolerant D/A converters for the LHC cryogenic system, Nucl. Instr. and Meth. A **553** (2005) 604.
- [2] GIRARD, T.A., et al., SIMPLE dark matter search results, Phys. Lett. B **621** (2005) 233.
- [3] FREITAS, M.C., et al., Multianalytical determination of trace elements in atmospheric biomonitors by k α -INAA, ICP-MS and AAS, Nucl. Instr. and Meth. A **564** (2006) 733.

- [4] DIAS, M.I., PRUDÊNCIO, M.I., Neutron activation analysis of archaeological materials: an overview of the ITN NAA laboratory Portugal, *Archaeometry* **49** (2007) 381–391.
- [5] MARQUES, J.G., et al., “Commissioning of the new LEU core of the Portuguese Research Reactor”, Proc. 12th Topical Meeting on Research Reactor Fuel Management, RRFM 2008, Hamburg, 2008, European Nuclear Society, Brussels (2008).
- [6] CARDEIRA, F.M., MENEZES, J.B., “Modifications and modernization of the Portuguese Research Reactor”, Proc. Symp. Management of Aging of Research Reactors, Geesthacht, 1995, IAEA, Vienna and GKSS, Geesthacht (1995) 438–450.
- [7] MARQUES, J.G., RAMALHO, A.J.G., “Operating and maintenance experience of the C&I of the Portuguese Research Reactor”, Proc. Int. Conf. Control and Instrumentation in Nuclear Installations, Bristol, 2000, The Institution of Nuclear Engineers, London (2000).
- [8] LND, Nuclear Detector Manual, LND INC., Oceanside (NY), USA (2001).
- [9] PLOUJOUX, M., Portugal – RPI: Mission d’ Expertise AIEA, Report SIAAR-134-178/87, CEA, Grenoble (1987).
- [10] MATOS, J.E., et al., “Core conversion analyses for the Portuguese Research Reactor”, Proc. International Meeting on Reduced Enrichment for Research and Test Reactors, Cape Town, South Africa, 2006, Argonne National Laboratory (2007).
- [11] INTERNATIONAL ATOMIC ENERGY AGENCY, Research Reactor Instrumentation and Control Technology, IAEA-TECDOC-973, IAEA, Vienna (1997).
- [12] HOLLENTHONER, S., WIERZBINSKI, S., AIRMON 91 Stack Monitoring System for Portugal, IAEA, Vienna (1991).
- [13] BÖCK, H., Atomic Institute of the Austrian Universities, Vienna, Private Communication (2006).
- [14] VAN EIJK, C.W.E., New inorganic scintillators – Aspects of energy resolution, *Nucl. Instr. and Meth. A* **471** (2001) 244.

TRIGA-INR MODERNIZATION FOR LIFETIME EXTENSION

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Abstract

Research reactors have an important role in the world for creating and maintaining the advanced infrastructure necessary for the progress of energy programmes and also to offer support for the development of various research domains of each country. After 27 years of TRIGA research reactor operations, a modernization programme for the reactor systems was required in order to achieve two main objectives: safe operation of the reactor in accordance with the provisions of the updated safety standards and reactor utilization in a competitive environment to satisfy the current and future demands and requirements of nuclear research. The TRIGA research reactor modernization programme addresses facility improvements and upgrades based on new requirements issued by national and international partners and consists in the following: promotion of a new concept for reactivity control; modernization of the reactor instrumentation and control system; modernization of the dosimetry surveillance system; refurbishment of the ventilation system; refurbishment of the irradiation devices; refurbishment of secondary cooling system and reactor power increase from 14 MW to 21 MW. The financial resources for the modernization of the reactor systems were provided by the Ministry of Trade and Industry. Refurbishment of the reactor instrumentation and control (I&C) system was sustained by a trilateral contract IAEA–INVAP–INR Romania and co-financed by the government of Romania.

1. INTRODUCTION

The Institute for Nuclear Research (INR) is located in Pitesti, 100 km north-west of Bucharest. The TRIGA research reactor achieved its first criticality on 17 November 1979 and was commissioned in 1980.

A main objective of the TRIGA research reactor is the testing of nuclear fuel and nuclear materials. For these tests, specific irradiation devices that can perform several activities are used. These devices include:

- power ramp;
- overpower ramp;
- fission gas pressure measurements;
- fission gas analysis;
- fuel clad residual deformation;
- structural material (Zr–2.5%Nb) studies;
- power cycling on CANDU fuel elements.

Figure 1 illustrates the pool arrangement with the two TRIGA-type reactor cores. The experimental locations contain irradiation devices, such as loop-type and capsule-type devices, that allow PWR and PHWR irradiation tests. Other experimental devices include:

- The neutron radiography facility designed for radiographs of nuclear fuel elements, such as TRIGA-HEU, TRIGA-LEU, experimental CANDU, and other devices that can be placed inside the investigation chamber. It uses the transfer exposure method with indium and dysprosium foils.
- The neutron diffraction facility is used for nuclear materials structural analysis, e.g. texture or stress analysis.

- The Prompt Gamma Neutron Activation Analysis facility can be used to measure trace and major elements in samples. The facility is linked at the radial neutron beam tube at the ACPR-TRIGA reactor.
- A thermal column that will be used as national standard for neutron flux.

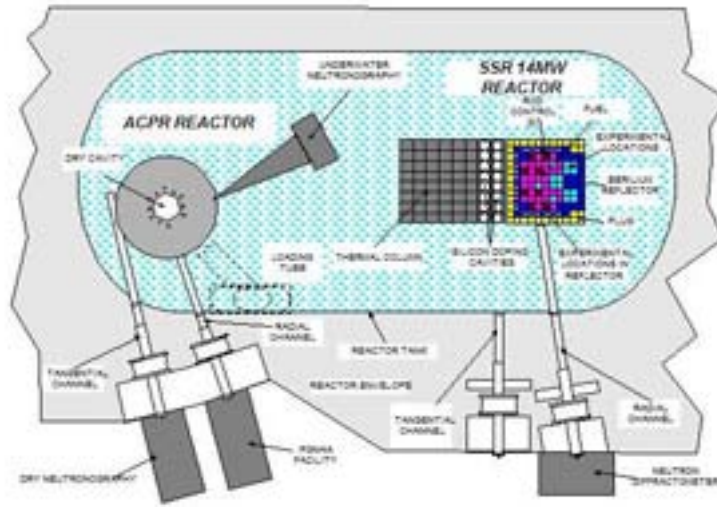


FIG. 1. TRIGA reactor pool.

The ACPR reactor is used for testing fresh CANDU-type nuclear fuel in the transient regime. Over the years, isotope production for domestic, industrial, and medical (^{192}Ir , ^{131}I and ^{99}Mo) needs was developed.

2. PLANNED AND ACCOMPLISHED UPGRADE AND/OR REFURBISHMENT ACTIVITIES

The need to prolong the reactor lifetime required a core conversion from HEU to LEU nuclear fuel. This process was successfully completed in June 2006. Considering the provisions of national laws and regulations, as well as international standards concerning the nuclear safety for research reactors, and the 14 MW TRIGA Safety Analysis Report, our institute proposed a project to upgrade and refurbish several main systems.

3. UPGRADED SYSTEMS

3.1. The instrumentation and control (I&C) system of the TRIGA reactor

- Modernization of the I&C system consists of a complete separation between the operation system and the nuclear safety system to achieve a higher degree of safety and reliability of the system and is based on regulatory body demands and IAEA recommendations.
- The reactor protection system is automatic and independent of other systems.
- The reactor protection system design requires that triggered automatic actions run to completion and cannot be interrupted or delayed. Also, the new control system design considers common mode failures.
- The new protection system must be able to control the process before the safety limits are reached.

Figure 2 is a simplified functional diagram of the new control system. As depicted in various blocks in Figure 2, the primary cooling system, the computer monitoring system, and

the irradiation devices are Romanian achievements. The reactor core upgrade was achieved with CERCA (France) LEU nuclear fuel elements and new Romanian control rods. The reactor console is to be designed, manufactured and installed by INVAP (Argentina).

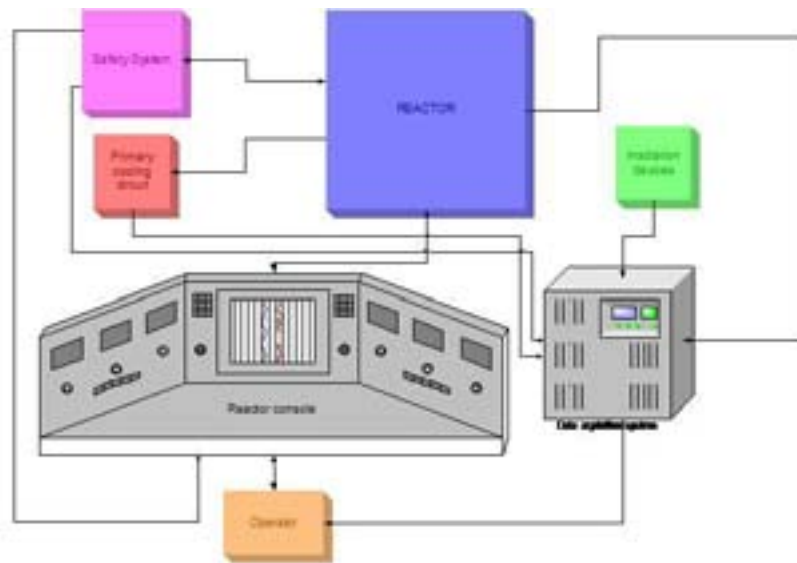


FIG. 2. New control system.

3.2. The reactivity control system

A new concept for the boron carbide control rods has been developed in the ICN and replacement of the old system has been completed. The main features of the new concept are redistribution of the boron carbide absorbent and its enclosure in zircaloy pins, which are practically corrosion-free at working temperatures (Fig. 3).

The primary goal of this concept was the preservation of the absorbent capacity of the control rod system in order to fulfil the requirements of the approved safety limits. It also preserves the hydraulic flow through the rod channels, so that the globally designed core flow-rate is not affected.



FIG. 3. Spatial views of the new control rod.

Based on these considerations, the mechanical project improved the structural stability of the control rod assembly. It has the advantage of easier maintenance and eliminates the helium purging system. The major drawback of the original design, i.e. the corrosion of the old aluminium square-pipe clad with subsequent water seepage into the absorbent section and clad swelling is resolved with the new design. Consequently, the non-operational periods are shortened and the system reliability is increased.

3.3. The primary cooling circuit modernization

The primary cooling system has the following features:

- Separation between operations and safety components.
- System redundancy.
- Operating parameters measurement.
- Acts in reactor scram logic whenever the safety limits are exceeded.

Also the logic implemented through redundant PLC's has the ability to perform the following:

- Display safety levels of process variables.
- Display the safety relays state.
- Display the alarm state.
- Transmit parameters and display to reactor console.

4. REFURBISHED SYSTEMS

4.1. Dosimetry system

According to basic radiological standards requirements that dictate the use of the International Units system and a decrease in personnel dose limits, the dosimetry system has been refurbished. The refurbishment included new configurations for a fixed measurement system with data transmission capabilities, on-line data processing and parameter evaluations. As a result of operational experience and national regulations and international recommendations, the new system includes supplementary sampling points. This ensures enhanced radiological monitoring in working areas, as well as in the environment and the population.

4.2. Cooling towers

The reactor secondary cooling system was refurbished to increase the performance of the heat exchangers in the cooling towers to allow heat removal for reactor operations up to 21 MW. Through this refurbishment, the secondary cooling circuit ensures a heat transfer of 4 MW/cell with $\Delta t_{\min} = 10^{\circ}\text{C}$ and a maximum cooling tower outlet temperature of 25°C .

4.3. Ventilation and air conditioning system

Rehabilitation of the ventilation and air conditioning system involves replacement of equipment with updated technology that satisfies system requirements. The updated equipment will work much better, but will be similar to existing equipment. The ambient conditions realized by the ventilation system offers an adequate environment for workers and for operation of systems and components. Once these systems have been upgraded and/or refurbished, it may be possible to increase the reactor power from 14 MW up to 21 MW, ensuring a competitive status, side-by-side with current and future perspectives.

4.4. Irradiation devices

The refurbishment of the irradiation facilities took into account IAEA recommendations, as well as national regulations in order to comply with internal and

external demands to ensure their continued operation in a safe and responsible manner, i.e. BWR type tests and ACR 1000.

This ambitious upgrade and refurbishment programme started is scheduled to be completed by the end of 2007. In 2008, all the pre-operational, operational and commissioning tests will be performed on all upgraded/refurbished systems, followed by the safety documentation completion and licensing by the National Regulatory Body. Therefore, with the upgrades and refurbishment, the reactor lifetime will be extended for another 20 years and it may be possible to perform irradiation experiments related to nuclear safety and fuel behaviour in nuclear power plants.

The refurbishment and upgrades do not modify the Limits and Conditions of Operation for systems and the safety functions remain unchanged. Installation and use of the new components and equipment will increase the reliability and safety of the systems. In the meantime, the maintenance and operation procedures in the Quality Management System will be thoroughly reviewed and amended to include the upgraded and refurbished systems and their integration into the facility.

For 27 years of reactor operation, no major reactor events occurred to impact the safety of the reactor, personnel, or the general population. This is because the reactor was operated in accordance with the following:

- a set of technical and administrative provisions;
- quality assurance manual;
- IAEA recommendations;
- the internal system of work procedures and instructions;
- personnel training and evaluation programme.

Adherence to the above conditions improves the reliability and safety of the reactor and decreases the likelihood of an accident. The two INSARR missions performed during 2000–2005 also confirmed that the level of nuclear safety is adequate and the facility is operation in a proper manner.

Following the modernization/refurbishment programme completion, we will be able and intend to perform activities, such as:

- power cycling and power ramp tests on CANDU fuel elements;
- irradiation of CANDU pressure tubes samples;
- irradiation of re-fabricated fuel;
- BWR/PWR fuel irradiation;
- CANFLEX fuel irradiation;
- ACR-1000 fuel irradiation.

5. CONCLUSION

Through this generous modernization programme, extension of the reactor lifetime along with an increase in power to 21 MW will enable our facility to participate in various projects as a reliable supplier of irradiation services and post-irradiation examination. Both fuel irradiation and material testing may be performed over long periods in high neutron flux, up to $3 \times 10^{14} \text{ n}\cdot\text{cm}^{-2}\cdot\text{s}^{-1}$, in loops and capsules fitted to customer specifications.

CURRENT STATUS OF MODERNIZATION WORKS AT THE KIEV WWR-M RESEARCH REACTOR

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Abstract

The WWR-M research reactor at the Institute for Nuclear Research is one of the first research reactors constructed and commissioned in the former USSR. The reactor was commissioned in 1960. At present, the reactor is a unique nuclear installation in the Ukraine due to its technical parameters and skilled staff. Future safe reactor operations require the replacement of obsolete equipment that was not upgraded during the last modernization. The current condition of the reactor allows for its safe operation for at least 8–10 more years if some systems and elements are upgraded. Extending reactor operations till 2015 is now considered to be the strategic goal. A coherent approach for the reactor modernization is in progress now. Previously, the Institute for Nuclear Research undertook several measures to improve both reactor and radiation safety. The current project includes the design of the control rod system equipment and renewal of an existing one; namely, the replacement of the current control rod system in the technological programme for automatic regulation, control, drive, and protection.

1. INTRODUCTION

The WWR-M research reactor at the Institute for Nuclear Research of the National Academy of Sciences of the Ukraine is one of the first research reactors constructed and commissioned in the former USSR. The WWR-M reactor was constructed 47 years ago to provide regional nuclear centres with research reactors on the initiative of academician I. Kurchatov. From the beginning, the reactor constituted the scientific and technical basis for research not only of scientists of the NASU, but also other organizations in the Ukraine and the former USSR. Various research activities in nuclear and radiation physics, radiation physical metallurgy, radioisotope production, and radiation biology have been conducted during more than forty years. At the beginning of 2002, the Institute for Nuclear Research obtained the licence for reactor operations and does not have a renewal date. The term of the licence is until the end of the reactor operation. The working life of the vessel and primary circuit is not determined by design documentation and negative changes outside the design limit were not found. The current condition of the reactor allows for its safe operation for at least 8–10 more years if some systems and elements are upgraded.

2. DESIGN AND LAYOUT OF THE WWR-M REACTOR

The WWR-M research reactor one of the last water-water reactors built. These reactors have a simple construction, easy experiment conduction, relatively small cost and operational expenditures, reliability and operational safety. The main goal of reactor is to provide neutron beams for investigations in different areas of physics and engineering. The reactor site is located in the south-eastern part of Kiev. The KINR, which employs about 1 000 people including support staff, is administrated by the NASU.

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

Main reactor elements (system) are:

- reactor vessel (tank) with the core;
- experimental equipment;
- cooling circuits (primary and secondary);
- water purification system for primary circuit;
- control rod system and system for control of reactor's parameters;
- power supply system for regular operations and as a backup for the main power supply source;
- radiation protection system;
- radiation control and protection system;
- special sewage system (collection, storage, and treatment of liquid radwaste);
- storage for fresh nuclear fuel;
- temporary storage for spent nuclear fuel;
- emergency cooling system;
- special ventilation system and filtration system for regular operations and in the case of accidents;
- reserve water supply system;
- fire control system;
- physical protection system.



FIG. 1. Common view of WWR-M reactor.

The main characteristics of the reactor are presented in Table 1 and the reactor's vertical section is shown in Figure 2.

TABLE 1. MAIN REACTOR CHARACTERISTICS

Reactor power	10 MW(th)
Number of fuel assemblies (WWR-M2 type)	262 (max), 156 (min)
Core volume	82 L
Maximal density of heat flow	490 kW/m ²
Water flow through primary circuit	1 200 m ³ /y
Water flow rate in the core	2.6 m ³ /s
Water pressure at the core input	1.35×10^5 Pa
Pressure difference in the core	1.5×10^5 Pa
Maximal water temperature at the core outlet	50°C
Maximal temperature of the fuel assemblies	95°C
Maximal density of the thermal neutron flux:	- in core – 10^{14} n/cm ² s - near reflector (isotope channels) – 6×10^{13} n·cm ⁻² ·s ⁻¹
Maximal density of the fast neutron flux (E > 0.8 MeV)	- at the bottom of hot cell – 4.8×10^{14} n·cm ⁻² ·s ⁻¹ - on supporting grate – 5.2×10^{12} n·cm ⁻² ·s ⁻¹

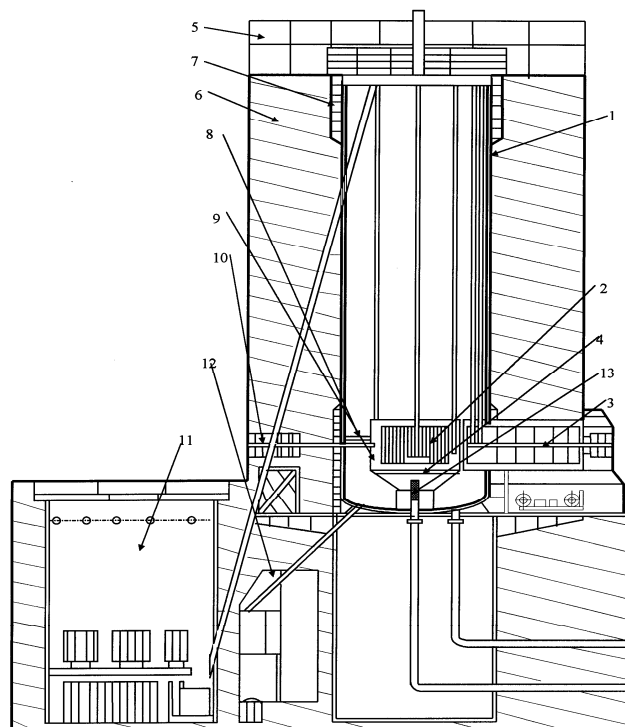


FIG. 2. Reactor's vertical section: 1) reactor vessel; 2) core; 3) thermal column; 4) core grids; 5) reactor cover; 6) biological shielding (heavy concrete); 7) iron; 8) grid for water flow; 9) Be-reflector; 10) horizontal experimental channel; 11) SF storage; 12) 'hot cell'; 13) filter.

3. OPERATIONAL HISTORY

The WWR-M reactor was commissioned on 12 February 1960. Until 1994, the reactor was primarily used to study the radiation behaviour of a variety of reactor materials. During this period, the reactor was operated about 100 hours per week for an annual total of 3 000–4 000 hours. The total operating time was equal to 104 500 hours (prior to 10 November 1993):

- capacity of 10 MW – 68.3%, 801.8 GW·h;
- capacity of 9.5 MW – 22.7%, 134.8 GW·h;
- transitional capacity – 9.0%, 43.4 GW·h.

The reactor was shutdown in 1993 and the core was entirely unloaded to the SNF wet storage facility. The reactor did not operate again until May 1998. From May 1998 until the end of 2001, the reactor was operated according to the interim permit issued by the regulatory body. Since May 2001, the INR has had a permanent licence for reactor operations.

From 1994–97, the INR undertook numerous measures to improve the nuclear and radiation safety of the reactor, in particular:

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

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

In accordance with the requirements of Article 8 of the Law of the Ukraine, On Permissive Activity in the Sphere of Nuclear Energy Utilization, which has been in place since 2000, the basis for activity, work, and operations related to the facility lifetime, is the licence issued by the state nuclear and radiation safety regulatory body. On 15 March 2002, The Board of the State Nuclear Regulatory Committee of Ukraine (SNRCU) issued a permanent licence to the INR to operate the WWR-M reactor. This was the first operating licence issued by the SNRCU. The licence states the term of operation; in this case, until the end of the reactor's lifetime. The term of operation was continued until the end of 2008 by a decree of the SNRCU Board. At that time, a new operating licence was issued.

The reactor operations schedule is determined by the requirements of the experimental programmes. The reactor usually operates in weekly cycles (24 hours per day from Monday to Friday). If necessary (in accordance with the conditions of experimental works), the reactor can operate continuously for two to three weeks. After the reactor was restarted in 1998, it was operated for 431 hours. In subsequent years, the reactor operated for the following number of hours: 1999 – 532; 2000 – 288; 2001 – 413; 2002 – 814; 2003 – 960;

2004 – 1503; and 2005 – 1492. Any incidental situations that occurred during 47 years of reactor operations have not resulted in a violation of the safe operation limits and conditions.

4. CURRENT STATUS OF REACTOR

The National Academy of Sciences of Ukraine approved the Strategic Plan for the Use of Research Reactor WWR-M of the Institute for Nuclear Research in July 2004. This multipurpose strategic plan for the use of the WWR-M reactor is directed at the effective use of logically defined and analyzed experiments on the reactor. The primary goal of the plan is the coordination of activities between operations, researchers, and users from different organizations; determination of user needs and installation capabilities; provision of the stable reactor operation by means of a stepwise implementation of the planned strategic tasks. The strategic plan will be reviewed annually for conformity and coordination of the current strategic tasks under the plan.

The plan defined the strategic goal as the **extension of reactor operations until 2015**. On the basis of this strategic goal, the following operational goals were determined:

- modernization of the reactor management, protection, and control system;
- modernization of the spent nuclear fuel storage facility;
- improvement of the fire prevention system;
- modernization of the emergency power supply system;
- modernization of the emergency core cooling system;
- conversion on low-enriched nuclear fuel;
- removal of the spent nuclear fuel.

The goal for the modernization of the reactor system and replacement of equipment was to increase the safety of reactor operations. All reactor systems were completely or partially modernized throughout many years of reactor operations. Important safety equipment, in operation since 1960, was the subject of periodic review in accordance with a special programme approved by the regulatory body; the operator determined the lifetime of the equipment and a final decision was coordinated with the SNRCU.

The reactor vessel and sections of the primary circuit pipeline are made from the aluminum alloy, CAV-1. The lifetime of the vessel and pipeline sections are not determined by the design documentation. For inspections of the reactor vessel, primary circuit and heat-exchangers in use for the last 20 years, the following methods are used:

- visual survey;
- hydraulic test;
- ultrasonic measurements of thickness;
- radiography of joint welds.

Investigations performed since 1988 provide evidence that there were no changes that negatively affected design limits of the reactor vessel and primary circuit components. Some parts of the reactor have been in operation since commissioning in 1960. The INR has developed a complex programme for the continuation of reactor operations, which includes:

- Perform inspection of equipment that cannot be replaced.
- Replace obsolete equipment, if technically feasible.

- Service, maintain, and repair equipment to remain in compliance with technical requirements.
 - Make alterations/changes in the operational licence.
5. REPLACEMENT OF THE CONTROL ROD SYSTEM AND SYSTEM FOR CONTROL OF REACTOR'S PARAMETERS

The control rod system (CRS) provides reactor power control (intensity of chain reaction) and performs the following functions:

- control and density change of the thermal neutron flux (power);
- control of density change rate of the thermal neutron flux;
- reactor launching and transition on the needed power level;
- automatic maintenance of the needed power level;
- compensation of reactivity changes caused by fuel burnup, poisoning, and thermal effects;
- alarm and warning signalling;
- reactor shutdown due to alarm signal.

The boron carbide rods (a neutron absorber) are used for manipulation of the chain reaction. There are nine rods — 1 automatic regulation rod, 5 compensative rods, and 3 independent emergency rods. The rods are located at reactor core level in separate channels that are made from aluminum alloy, CAV; the channel diameter is 35×2.5 mm. The rod shell is made from stainless steel of 27×1 mm diameter; the rod diameter is equal to 25 mm with a length of 600 mm).

The rod movement is carried out with servo-drives. Before reactor startup, the emergency rods are in the extreme top position (600 mm). In case of emergency or a planned reactor shutdown, they drop to the core in 0.3 s. The operator moves other rods, either manually or automatically, from the core to the level corresponding to the start of chain reaction. In an emergency, 3 emergency rods stop the chain reaction; the compensative rods are slowly moved to the core (600 mm in 17 s).

The power measurements are carried out by means of measure channels, which consist of KHK-53M, KHK-3, and KHK-4 type ionizing chambers and the electronic devices for signal formation. The power measurement range lies in the range from 10^{-5} –120% nominal power. There are 7 ionizing chambers; 3 of them are in the reactor power protection channel; 2 are in the reactor period measurement and protection channel; 2 are in the reactor power measurements channel, and 1 is in the automatic power regulation channel.

The lock system composed of control rod system is in operation. This system prevents reactor operator mistakes from occurring, prevents reactor startup when reactor systems are not ready for operation. The lock system also scrams the reactor (by means of emergency and compensative rods) when an important safety system breaks or when there are excursions in reactor parameters from established values, such as temperature, pressure, primary circuit heat-carrier discharge, water level in the reactor vessel, unauthorized power increase of 20%, decrease of reactor period less than 10 s, increase in radioactivity in the primary circuit water or ventilation air.

The reactor control system parameters (there are about 80 measuring points and channels altogether) provide measurements of the following parameters (see Table 2):

- primary circuit heat-carrier temperature – 6 measuring points;
- secondary circuit heat-carrier temperature – 5 measuring points;
- water discharge in the primary and secondary circuits – 4 measuring channels;
- temperature of water, air, and bearings – 17 measuring points;
- the water levels in the reactor vessel and fuel storage pond and others – 9 tank-level gages;
- water pressure in the primary and secondary circuits – 9 measuring points;
- air discharge in the ventilated volumes – 6 measuring channels;
- specific electro-conductivity of the primary circuit heat-carrier – 1 measuring channel.

TABLE 2. OPERATING PARAMETERS DURING REACTOR REGULAR OPERATIONS

Designation	Unit	Value
Heat-carrier temperature at the output	°C	≤ 50
Heat-carrier temperature at the input	°C	≤ 43
Water pressure in the primary circuit	kg/cm ²	1.0 ÷ 1.5
Water pressure in the secondary circuit	kg/cm ²	1.2 ÷ 3.5
Temperature drop at the input and output	°C	≤ 7.7
Water level in the reactor vessel	mm	4 800–5 010
Water level in the fuel storage pond	mm	3 500–4 000
Level of liquid radwaste in the storage tanks	mm	0–4 800
Release of radioactivity to the ventilation system:	Ci/year	
– rare gases		≤ 4 450
– iodine		≤ 1.5
Water temperature before the heat-exchangers (secondary circuit)	°C	≤ 25
Discharge of technical water (secondary circuit)	M ³ /hour	≤ 1 100

The capital modernization (replacement) of the control rod system was performed during 2006–2007. The reasons for the replacement are as follows:

- Equipment, tools and devices have been in operation since the start of reactor operations.
- Difficulty in procuring spare parts.
- Current control rod system is not in compliance with modern requirements, namely:
 - Reserve control panel was absent.
 - Cable types and layout do not meet fire safety requirements.
 - Installation of an automatic data recorder was not possible.

In accordance with the design, the old control rod system was replaced with a modern system that incorporated the best design solutions. This project was outlined in the programme–technological complex for the regulation, control, manipulation and protection of the WWR-M research reactor of INR NASU and was designed, manufactured and assembled by the Radiy Corporation (Kirovograd, Ukraine). The new system performs the following functions:

- control of neutron flux;
- automatic regulation of neutron flux;
- emergency protection;
- manipulation by the drivers;
- measurement, control and signalling of the technological parameters;
- remote manipulation by the drivers;
- information and diagnostic.

The following elements were a subject of replacement:

- equipment of the main control board (devices for the control of technological parameters, electrical equipment for the mechanisms driving, alarm system);
- relay switchboard of the control rods system as well as the workstations for the rods;
- control desks for the pumps and valve gates in the primary and secondary circuits, ventilator and cooling tower;
- sensors of the technological parameters of the reactor's systems (circulation rate, levels, temperature, pressure etc.) and impulse tubes;
- power and control cables of the operating mechanisms.

Requirements of new control rod system:

- New system should have control of the neutron flux via three independent channels. Each channel should be able to measure power means of neutron detectors. Emergency protection is carried out through the comparison of threshold values and pulse generation on the rods.
- Each channel should have an independent power supply and the tools necessary for detectors operation. Information should be displayed on both control boards (main and reserve) with recording/archiving.
- Control of following parameters should be provided:
 - water temperature in the primary circuit (10 points);
 - temperature difference at the core input and output (up to 10°C);
 - water pressure in the primary circuit;
 - water level in the pool;
 - water flow rate in the primary circuit.
- Emergency response signal should be automatically generated for excursions of reactor parameters from threshold values. In such an emergency, the reactor will scram.
- System should be able to measure reactor power via neutron detectors and provide automatic power regulation. The control of the rod's positions should be provided as well as the manipulation by 9 rods.
- The following elements should be located at the main control board:
 - monitoring devices of the safety system;
 - supervisor computer;
 - tools for the manual manipulation by rods.
- All equipment needed for the manipulation and protection should be located at the reserve control board. This equipment is necessary during emergency shutdowns where the main control board will be inaccessible (for example, in the case of fire).



FIG. 3. Reactor control panel: old (left) and new (right).

The complex includes:

- one cabinet for remote manipulation;
- one cabinet for converter and signalling;
- three cabinets for signal shaping;
- one cabinet for computer;
- seven modules for power direction of the control rods drives;
- ten cupboards for equipment manipulation.

After dismantling the old system, the new system was assembled and is currently under testing and commissioning is planned for the end of 2007. Figure 3 shows the reactor control panel before and after replacement.

6. CONCLUSION

The current technical condition of the reactor allows its safe operation for no less than 8–10 years if some systems and elements are upgraded. Extension of reactor operations until 2015 is the current strategic goal. The upgrade is now in progress and replacement of the control rod system and the reactor control system has been successfully completed.

CORRECTIVE REPLACEMENT OF THE TRIGA REFLECTOR AT THE UNIVERSITY OF TEXAS AT AUSTIN

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Abstract

The TRIGA research reactor in the Nuclear Engineering Teaching Laboratory (NETL) at the University of Texas at Austin (UT) was shut down for six months 1999–2000 due to a gas pressurization of the reactor graphite reflector. The gas mixture was vented in early 2000 by remotely drilling into the reflector and the NETL reactor was returned to operation. Over the next three years, the NETL received incremental funding from the US Department of Energy to replace the reflector. The NETL staff chose to perform the reflector replacement and component refurbishment with the reactor pool full and thus minimize the time the reactor needed to be out of service. A team of four research divers from the UT Applied Research Laboratories (ARL) was trained to perform the repairs and work in high radiation areas. The replacement of the reflector with the reactor pool filled and a two-month decay period reduced the total radiation dose to the divers while balancing the ALARA commitments with the operational need to return the facility to critical operations in a timely manner. The dive operations and successful repairs occurred over a two-week period in 2004. Lessons from this project include the benefits of good planning, directed and specific training programmes for radiation workers, and experiences in dealing with unexpected occurrences despite careful planning. In addition, this project showed that major research reactor component replacement or refurbishment projects are possible without extended shutdown periods or large personnel exposures.

1. INTRODUCTION

In late 1999, the TRIGA reactor reflector at the Nuclear Engineering Teaching Laboratory (NETL) at the University of Texas at Austin (UT) was discovered to have pressurized and bulged due to a small water leak through a welded area. The reflector was vented in early 2000 to allow for reactor operations and to prevent further deformation. The venting caused flooding of the reflector, saturating the graphite inside, and filling all voided areas in the reflector [1]. This event significantly affected several experiments and reduced the capability of a facility that had been operating for less than ten years. The damaged reflector required repair or replacement as soon as possible because the failure had disabled two neutron beam experiment areas. By 2003, sufficient funds were received through the US Department of Energy's University Reactor Instrumentation Grant Programme [2] to allow the contracted fabrication of a replacement reflector assembly.

The NETL had developed an active research reactor experiment and educational programme over several years and by 2003 operated an average of 170 days a year even with the reduced capabilities caused by the flooded reflector. There was concern that a lengthy shutdown period would prevent completion of several ongoing student research projects and reduce the number of NETL long term reactor users as they sought replacement facilities for their experiments. A decision was made to attempt a reflector replacement without draining the reactor pool to reduce the overall shutdown period while still maintaining a high level of personnel safety.

2. DESCRIPTION OF NETL TRIGA RESEARCH REACTOR AND POOL

The University of Texas at Austin's TRIGA research reactor attained criticality in March 1992, making it one of the newest research reactors in the world. Although this was not the first research reactor at UT, it was the first with neutron beam ports and sufficient

power to have a neutron and radiation beam experiment programme. The NETL reactor is a Mark II TRIGA reactor design fabricated by General Atomics of San Diego, California and licensed by the US Nuclear Regulatory Commission to operate at 1.1 MW(th). The reactor sits nears the bottom of an 8 m deep above ground pool. The pool cross-section is approximately a 2 m × 3 m oval in size with the reactor assembly off-centre leaving an open space on one side of the reactor tank (Fig. 1).

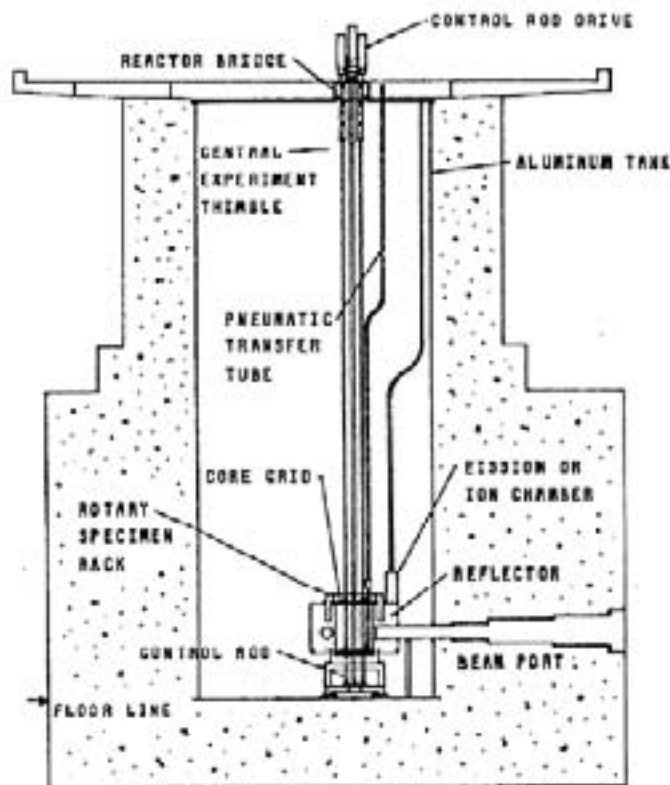


FIG. 1. Cross-section of NETL reactor pool.

The NETL reactor has five available neutron beam ports. Beam port 3 is unique because it is a piercing port with the reactor installed by sliding the reflector assembly horizontally 25 cm over the beam pipe. The reflector assembly supports the reactor core and is an aluminum-canned annular unit machined from a single block of graphite. As seen in Figure 2, beam ports 1 and 5 are actually a single through-tube assembly that passes adjacent to the reactor core and is welded into the reflector. This tangential tube is bolted to 15.24 cm i.d. beam pipes protruding from the pool walls. Movement of the reflector relative to the pool wall is permitted by installing flexible aluminum bellows assemblies into the beam pipe system. The thin bellows is protected from external damage by an aluminum shroud around the bellows area.

The reflector is attached to a support table on the pool floor with four bolts as shown in Figure 3. These 4 bolts and the 32 bolts on the bellows had to be removed prior to pulling the reflector off the beam port 3 pipe during the replacement. Draining the pool for the task would have required a long (up to six months) shutdown for radioactive reactor components to decay sufficiently for safety. The pool water (nearly 38 000 L) would need to be stored or disposed of during the project and then replaced with low conductivity and neutral pH water. In the past, this evolution took several weeks before the pool water conductivity and chemistry would allow fuel to be transferred into the pool. Finally, even with the long

radioactive decay period, the radiation doses expected during the project were much higher without the intervening radiation shielding of the water.

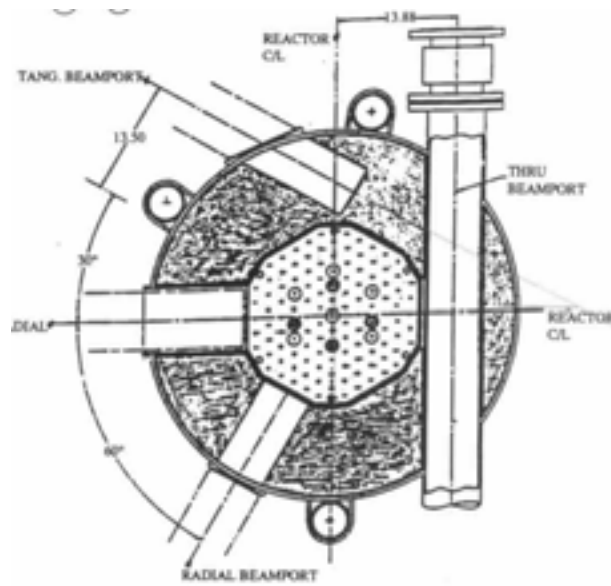


FIG. 2. Top cross-section view of reactor reflector assembly (dimensions in inches).

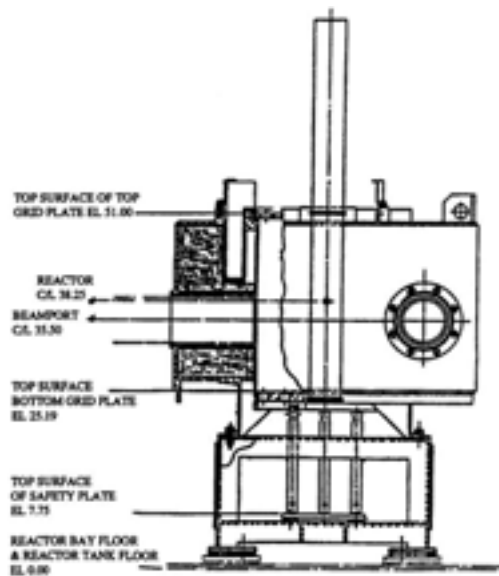


FIG. 3. Side view of TRIGA reflector assembly (dimensions in inches).

3. SELECTION AND TRAINING OF DIVE TEAM

Four experienced divers from the University of Texas Applied Research Laboratories (ARL) Scientific Research Dive Team (SRDT) were selected to perform the underwater removal and installation of the TRIGA reflector. The SRDT supports acoustic research and development programmes across the world for UT and other agencies. The two lead divers had over 20 years of experience in equipment installation projects and the other two divers had at least 10 years of experience. The ARL dive programmes had operated without a safety incident for over five decades with a high level of attention to detail, training and procedures.

The SRDT divers had no previous experience in radiation safety or working within high radiation areas and UT was reluctant to allow the divers to perform the project until their safety was assured. The NETL and UT health physics staff provided comprehensive radiological safety training to the divers well in advance of the expected project start date. The radiological safety training programme and the Radiation Work Permit (RWP) met the requirements of the UT safety programmes, US Nuclear Regulatory Commission Regulatory Guide 8.38 (App. A) [3], and the Electric Power Research Institute (EPRI) Underwater Maintenance Guide [4]. For training the divers, General Atomics loaned a full-size, unused reflector assembly to UT so the divers could perform mock-up training in their practice pool. This provided valuable lessons on the operation of the pneumatic tools underwater. An additional full-scale mock-up of the bellows and beam tube flange area permitted testing of several methods of flange installation underwater and saved many hours of time later.

During the actual reactor pool dive, two divers were in the reactor tank performing the task. A third scuba-equipped diver was in safety standby and assisting from the surface of the pool. The fourth diver (the dive master) was responsible for voice communications, diver air supply, dive supervision, and monitoring of dive activities using closed-circuit video from cameras on the diver's helmets. The divers wore dry dive suits and full helmets to minimize contact with the reactor pool water. The dive helmets provided continuous air supply from the pool surface and two-way voice communication with the safety monitor topside. The divers wore multiple dosimeters to fully monitor and evaluate doses received during the dive operations. These dosimeters were attached to all extremities, chest, and helmet. Additionally, each diver in the pool had a digital radiation monitor. The radiation monitors selected were Canberra UltraradicTM personal monitors because they had a bright, large backlit LCD display, an alarm function, and were watertight down to one meter. The Ultraradiac monitors were sealed in commercial diving bags to withstand the full dive depth without leaking.

4. CONTROLLING DIVER RADIATION EXPOSURE

Clearly, the TRIGA reactor fuel in the pool would need to be removed and stored for diver safety. The NETL was constructed with six fuel storage pits approximately 5 m deep in the floor of the reactor room. Fuel was transferred three elements at a time from the reactor pool to the storage pits using a small, lightly shielded (10 cm lead) cask and the building crane. The workers had to climb and work from a 7 m scaffold to unload the TRIGA fuel elements safely in air and transfer into the fuel storage pits. Over 100 elements were transferred from the reactor core and pool storage racks to the facility fuel storage pits. The reactor fuel in the pits was covered with reactor-quality water for cooling, radiation shielding and criticality control.

Calculations based on known percentages of elements in the reactor structural materials indicated that a significant amount of radioactive decay would occur after 60 days (Fig. 4). The bulk of the reflector was fabricated from aluminum (alloy 6061T) or carbon (in graphite). Trace isotopes in the aluminum include iron, silicon, copper, and chromium. All other fasteners and hardware were stainless steel (alloy 304). Chromium-51 represented the most significant source of radiation after a few weeks of decay (28-day half-life, 320 keV gamma). Waiting two months before replacing the reflector assembly provided adequate decay with diminishing returns if extending the decay time for the longer-lived isotopes.

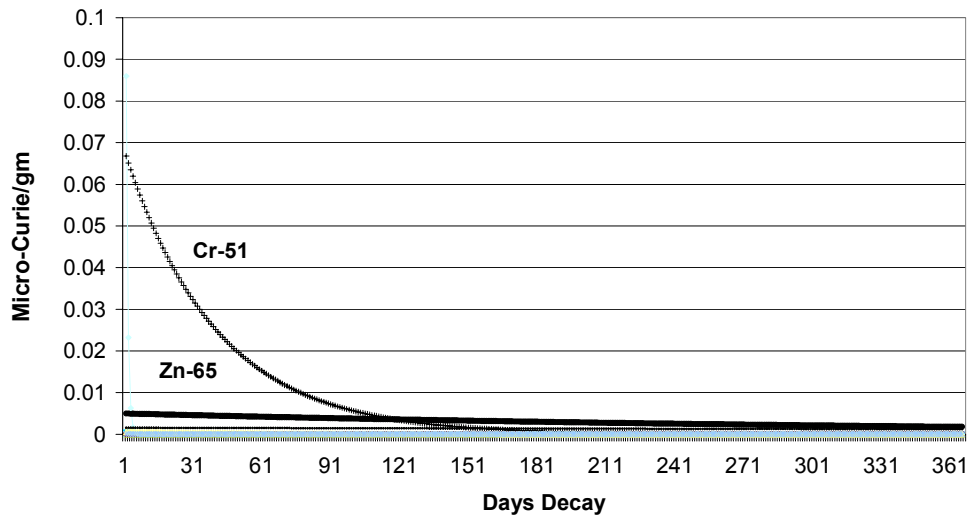


FIG. 4. Decay of isotopes in T6061 aluminum.

From the beginning and before any work commenced, the NETL Management was committed to evaluate dose rates and determine if it was safe for personnel to enter the pool and work around the reflector. Localized hotspots with a significant amount of ^{60}Co and ^{55}Fe were within the various stainless-steel components. Radiation exposure was controlled with shielding, remote tools and directed training. The only project period where the primary safety concern was radiation exposure was during the removal of the old reflector. The majority of the radioactive components in the pool (after the fuel was removed) were contained in the reflector assembly and the rotary specimen rack. When the new reflector was installed, all stainless steel fasteners were replaced to reduce the potential doses received during installation.

Preliminary calculations followed by confirmatory direct measurements indicated the primary localized sources of radiation (following removal of all reactor fuel) around the TRIGA reactor was the rotary specimen rack (the RSR or lazy Susan) as well as the stainless steel bolts attaching the flexible bellows assembly to the reflector and the beam pipe. The RSR operated by driving a 48 cm diameter stainless steel ring through a gear system. The measured dose rates underwater near the RSR were in the range of 60 R/hr. The RSR had to be removed from the reactor pool to perform the reflector replacement, but due to its vertical size, construction, and awkwardness, it was the most challenging object to remove from the pool. The ceiling height of the reactor room prevented bringing the RSR directly out of the pool because the experiment loading tubes of the RSR are approximately 7 m long. Additionally, a geared motor drives the RSR from the pool surface via a solid drive shaft through a long, straight access pipe. Large fittings on the three pipes attached to the RSR were disconnected to reduce the vertical height of the RSR. The large size of the fittings exceeded the maximum size of any locally available wrench and there was a concern that the commercial wrenches could produce contamination or corrosion from unknown materials in the wrenches. Special single-use wrenches were fabricated from stainless steel using water-jet cutting technology and sized to fit all RSR fittings. The high radioactivity and large size of the RSR required a unique storage and shielding solution. A 1 880 L round plastic storage tank was purchased and installed behind 61 cm thick concrete walls. The RSR assembly was immersed in demineralised water within the tank for shielded storage until it was returned to the reactor pool.

Measured dose rates on contact with the reflector assembly at the centreline of the reactor between the two bellows assemblies (Fig. 3) were 1.5 R/hr and dropped to 600 mR/hr approximately 30 cm away from the reflector. Between the flanges of the bellows assembly, the dose rate was 4.2 R/hr but rose quickly to 13.0 R/hr near the stainless steel bolts. An additional moveable shield was required so the divers could work for longer periods near the reflector when removing fasteners. The shield was constructed by hanging multiple layers of commercial-grade Herculite covered lead-wool sheeting from an aluminum beam. The final temporary shield was 122 cm × 25.4 cm × 91 cm and weighed approximately 590 kg. The underwater shield was lowered into place and supported by the facility overhead crane during reflector removal. The shield reduced the dose rates in the work area to approximately 5 mR/hr. The divers were also provided with several locally fabricated remote pneumatic wrenches and long-reach tools for additional distance from the radioactive stainless steel nuts and bolts.

5. BEAM PIPE PLUG SYSTEM AND BELLOWS ASSEMBLY

The neutron beam pipe through tube passes directly alongside the reactor core acting as a dual tangential neutron beam. The tube is watertight and attached to the reflector assembly using flexible bellows coupling. Sixteen sets of nuts and bolts hold each bellows assembly to the reflector and allow slight movement of the assembly relative to the pool walls. The beam ports had to be plugged in order to remove the reflector assembly with the pool still filled with water. This was accomplished using a specially designed pipe-plugging device. The beam port pipe is approximately 15 cm in diameter at the interior wall of the reactor pool. Beam port 1 increases to 20.3 cm at the outside wall and beam port 5 increases to 38 cm. The device needed to prevent pool draining by withstanding a pressure of 75 kPa from the expected pool depth. Mechanical Research and Design (MR&D) of Manitowoc, Wisconsin was selected to design and fabricate two beam pipe plug assemblies. The plugs were designed to expand two O-rings against the inside walls of the beam pipes and seal against the pressure of the reactor pool (Fig. 5).



FIG. 5. Beam port plug showing expanding seal area.

The final beam pipe plug design had to do more than just keep water from flowing from the pool. The reliability and flexibility of the pipe plugs was the key to the success of the entire project. The final plug design was required to do the following:

- The plugs had to reliably seal the pipe against an 8 m head of water without leaking or slipping. To verify the plugs were holding and prior to loosening the bolts on the flanges, the area between the two plugs, a volume of approximately 33 L, was

pressurized to 103 kPa to check for leakage. The plugs had a tube passing through the centre of the compression ring that permitted airflow.

- A significant quantity of water could remain trapped inside the pipe after the reflector was replaced and presented a contamination or corrosion hazard if it were allowed to spill uncontrolled from the beam port after the plug was removed. The centre air tube was also required to act as a drain for the internal reflector volume following replacement.
- Immediately following the proper torquing of all bellows nuts and bolts, the system needed to be checked for leakage prior to removing the pipe plugs. The air system also permitted a low pressure (to prevent bellows damage) check of the bellows to pipe seals at 103 kPa. This was the final quality assurance check and verification of proper bellows installation.

To meet these requirements, a special air-handling rig was built to control the system (Fig. 6) in addition to the design of the actual pipe plugs. The pipe plug system (PPS) was designed to reduce facility service air to low pressures for safe purging and pressure testing of the bellows pipe, drain water to a collection tank, monitor internal pressure to verify the system was holding pressure, and provides pressure relief to prevent overpressurizing the bellows assembly. The PPS was connected to the air tube of the pipe plugs using a quick disconnect fitting and three meters of rubber tubing.

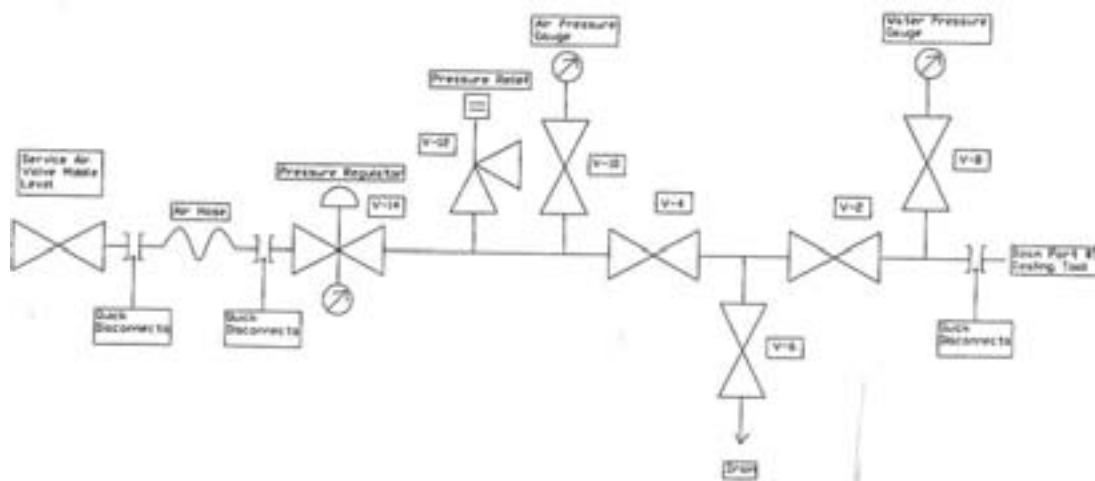


FIG. 6. Pipe Plug System (PPS).

The pipe plug assembly had to be held firmly in place against pool water pressure to prevent it from slipping out of the pipe. This was accomplished using a large backing plate held in the external pipe by a radial array of bolts (8 or 12 depending on o.d.). These bolts acted as set screws against the inner wall of the beam pipe. Although this was a reliable system, the NETL facility chose to install the normal beam port 0.625 cm steel cover plate as an additional backup method to prevent the pipe plugs from slipping completely out of the pipe.

The removal of the bellows flange bolts was primarily performed with pneumatic hand wrenches. The commercial off-the-shelf tools brought the diver's hands too close to the bolts so they were lengthened by the NETL and operated using an air-valve several feet away. Hardware removed from the flanges was collected using a long-handled scoop and transferred to the pool surface in a bucket. Upon reaching the pool surface, the bucket was quickly dumped into a disposal chute that dropped the radioactive hardware into a shielded storage

area on the building floor 9 meters below. Two nuts seized during removal and could not be removed by hand. These nuts were removed using a commercial grade nut-splitter (Fastorq Bolting Systems, Inc., Houston, TX) operated by a hydraulic hand-pump at the pool surface.

After removal from the reactor pool, small amounts of corrosion were found during examination of the bellows assemblies. It was believed that this corrosion was caused by moisture condensing within the pipe during initial construction and not leakage. The flange faces and bellows were cleaned and refurbished in an isolation glove box and then chemically anodized. It could not be guaranteed that the thin aluminum of the bellows had not been damaged during removal or subsequent cleaning. In order to test the integrity of the aluminum, the bellows were sealed with flanges and rubber O-rings and pressure-checked underwater as a final step before preparing the assemblies for reinstallation.

The bellows flange uses two concentric metal O-rings for a more reliable seal. Initially, the installation procedure was to install the O-rings and bellows with the system flooded and later drain the water from the pipe using the PPS. There was a concern that some amount of water could remain trapped between the O-rings while tightening the flange bolts. This water could potentially expand due to heat and radiolysis or cause corrosion later producing an unreparable leak into the beam port pipe. To prevent this situation, air was constantly forced through the pipe plugs during bellows installation. The airflow efficiently displaced the water and the divers initially tightened the bolts from the top down to force water out the bottom of the seals. The divers were expected to have difficulty installing and holding the O-rings while bolting the flanges of the bellows assembly.

This would be especially true with large volumes of air flowing past the seals during installation. To reduce the time the diver would need to manipulate the bellows, the O-rings were pre-installed into the bellows and held in place by thin plastic sheets prior to returning to the pool. The divers slipped the bellows into place and inserted several alignment pins (to prevent unnecessary damage to bolt threads) to hold the bellows and O-rings in place before slipping the plastic sheet out. As an additional aid to the divers, the bellows were slightly pre-compressed within a jig designed to support the bellows while slowly releasing the tension. The system worked so well that the O-ring metal seals never fell out of the grooves during the installation. This design and procedure performed far better than the initial dry installation that was recalled to take several days.

The divers could only install one bellows assembly at a time so a blank flange with a rubber gasket was temporarily bolted to the other side of the reflector to permit pressurization of the bellows and through tube during installation of the first bellows assembly. The installation of the second bellows required significant prying to align bolts a few millimetres but the installation went relatively smoothly. The air forced through the gap of the flange effectively emptied the majority of the water from the pipe interior and it was not necessary to drain water through the PPS as originally planned.

6. DIVE OPERATIONS AND SAFETY

The radioactive material in the reactor pool water had been evaluated by sampling and counting a 0.5 L Marinelli beaker. The only isotope detected after two months of reactor shutdown was ^{124}Sb at concentrations 3% of the allowable release limits. Swipe samples were taken using a sample of adhesive tape pressed against objects in the pool with a long aluminum pole. These contamination surveys indicated there was significant but not extreme levels (~5 K dpm) of removable contamination on the pool floor and the reflector surface.

The control of diver contamination followed a typical contaminated work area strategy. Large area rubber mats were placed in advance on the pool floor in the expected location for the divers to stand, kneel, or sit while working. The diver's feet and hands were covered with standard nuclear industry anti-contamination rubber gloves and overshoes. Later, the standard rubber gloves were replaced with diver gloves over rubber gloves for durability during tool manipulations. The divers frequently kneeled while working under the reflector so rubber kneepads used by carpenters were used to protect the dive suit. Upon exiting the pool, the divers immediately stepped into small, shallow decontamination pools for frisking and decontamination. Generally, the divers did not have detectable contamination through the period of dive operations.

The reactor pool water generally has high visibility and one can easily see an object on the pool floor through eight meters of water. Similar clarity was expected for the dive operations but this did not occur due to unanticipated cloudiness produced from new nylon ropes. The nylon ropes were used to lower and retrieve tools and parts for the divers and were newly purchased to prevent the introduction of unknown chemicals or materials into the reactor pool. The new ropes tended to release an inert material or small particles upon use in the pool water. The fine particles severely reduced the visibility in the pool and the divers could not be observed from the pool surface during the first few days of dive operations. This cloudiness increased and the pool pH decreased the longer the divers were in the pool. This was caused by the large amount of air added to the pool by the divers' breathing and the pneumatic tools. The divers were continuously observed at the local work area by a camera mounted on one diver's helmet and a separate camera mounted in a watertight tube that was manipulated from the pool surface. The separate camera provided close-in observation of areas within high radiation fields and a static observation of the divers working together. An ion chamber installed in the watertight camera housing mapped the local radiation field and hot spots were visually identified from the pool surface.

7. CONCLUSION

The NETL reflector was effectively replaced on schedule over a two-week period at the end of May 2004. The ARL dive team set new operational endurance records for sustained underwater work during the project and completed the removal and installation in about 40 hours of actual dive time. Subsequent facility tasks included reinstalling neutron detectors, coolant piping, RSR and control rod drives, moving over 100 fuel elements back into the reactor pool from storage, reloading the core, and performing all annual calibrations and retests. The entire project from the reactor shutdown to the reactor return to full power operation took just four months. Diver radiation doses were considered extremely low for the tasks performed and the environment [5]. The highest whole body dose received was 43 mrem to one diver and 340 mrem extremity dose to another diver. Careful planning prior to the shutdown period enabled all research projects in progress to be completed early or placed on hold in a controlled manner with minimal impact to deliverable schedules [6].

REFERENCES

- [1] O'KELLY, D.S., Hydrogen and oxygen gas production in the UT TRIGA reflector, 2000 International Conference on Nuclear Science and Technology: Supporting Sustainable Development Worldwide (2000 ANS Winter Meeting), Washington, DC, 12–16 November 2000; Trans. of the Amer. Nuc. Society, **83** (2001).
- [2] US DEPARTMENT OF ENERGY, 2002 University Reactor Instrumentation Programme, Contract No. DE-FG07-03ID14473.

- [3] US NUCLEAR REGULATORY COMMISSION, Regulatory Guide 8.83 (June, 1993), “Control of Access to High and Very High Radiation Areas of Nuclear Plants (Original draft guide was issued as DG-8006, 10/1991) (Rev. 0, ML003739558, 06/1993) (proposed Rev. 1, DG-8028, ML052590173, 09/2005) (Rev. 1, ML061350096, 05/2006).
- [4] JENCO, J.M., “Underwater Maintenance Guide: Revision 2, A Guide to Diving and Remotely Operated Vehicle Operations for Nuclear Maintenance Personnel”, EPRI Research Project 2814-24, NP-7088 Rev. 2, (2004).
- [5] O’KELLY, D.S., O’KELLY, D.J., PENNINGTON, W.S., TIPPING, T., “Minimizing personnel radiation dose during major reactor repairs at The University of Texas at Austin”, 2004 International Topical Meeting on Operating Nuclear Facility Safety, (2004 Winter ANS Meeting), Washington, DC (2004).
- [6] O’KELLY, D.S., “Successful replacement of the UT TRIGA reactor reflector,” 2004 Annual Meeting of the Organization of Test, Research and Training Reactors, Columbia, Missouri (2004).

MODERNIZATION OF THE HIGH FLUX ISOTOPE REACTOR (HFIR) TO PROVIDE A COLD NEUTRON SOURCE AND EXPERIMENTATION FACILITY

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1. BACKGROUND

In June 1961, construction began on the High Flux Isotope Reactor (HFIR) facility inside the Oak Ridge National Laboratory (ORNL) at the recommendation of the US Atomic Energy Commission (AEC), Division of Research. Construction was completed in early 1965 with criticality achieved on 25 August 1965 [1]. From the first full power operating cycle beginning in September 1966, the HFIR has achieved an outstanding record of service to the scientific community.

The design of the HFIR is based on the flux trap principle, with an inner moderating region, surrounded by an annular region of fuel, which is, in turn, surrounded by a beryllium reflector [1]. Such a configuration permits fast neutrons escaping from the fuel to be moderated in the un-fuelled space in the centre producing a region of very high thermal and epithermal neutron flux. This reservoir of thermalized neutrons is trapped within the reactor, making it available for isotope production and materials irradiation studies. Some neutrons leak into the reflector surrounding the fuel. These neutrons are available for neutron scattering experimentation by extending an empty tube into the reflector region. The HFIR has four such access tubes. Neutrons scattered into the empty tube travel outside the reactor shielding to highly specialized instruments for use in experimentation. Additionally, a variety of vertical holes are provided in the reflector in which to irradiate materials for isotope production, materials irradiation studies, and neutron activation analysis. A cross-sectional view can be found in Figure 1 [2].

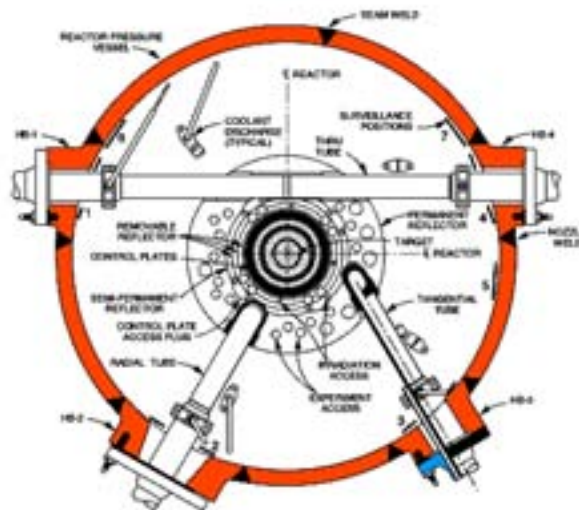


FIG. 1. HFIR cross-sectional view.

The mission of the HFIR is, “Safe, reliable, predictable, and efficient HFIR operation to support the neutron science mission.” [1]. The primary focus of neutron science at the HFIR is currently neutron scattering research exploring fundamental and applied research on the structure and dynamics of matter. Neutron scattering is a useful source of information about the positions, motions, and magnetic properties of solids. When a beam of neutrons is aimed at a sample, many neutrons will pass through the material. But some will interact directly with atomic nuclei and bounce away at an angle, like colliding balls in a game of pool. This behaviour is called neutron diffraction, or neutron scattering. Using detectors, scientists can count scattered neutrons, measure their energies and the angles at which they scatter, and map their final position (shown as a diffraction pattern of dots with varying intensities). In this way, scientists can glean details about the nature of materials ranging from liquid crystals to superconducting ceramics, from proteins to plastics [2].

The original HFIR neutron science mission of medical, industrial, and research isotope production is now a secondary, but still active, focus. The most notable isotope produced by the HFIR is ^{252}Cf . This neutron-emitting isotope is used for reactor startup sources, scanners for measuring the fissile content of fuel rods, neutron activation analysis, and fissile isotope safeguards measuring systems. In addition, ^{252}Cf is used as a medical isotope to treat several types of cancer. The production of ^{252}Cf and other trans-plutonium isotopes is a unique capability for which the reactor design was originally optimized [2]. The high flux regions in the HFIR also provide for the production of lighter isotopes that have a high specific activity and specialty isotopes that cannot be produced in lower neutron fluxes.

The HFIR neutron science mission is also carried out through neutron activation analysis using two pneumatic ‘rabbit’ facilities that shuttle small samples in and out of the reactor reflector region. These samples are measured in the HFIR Neutron Activation Laboratory using gamma spectroscopy to detect trace elements that cannot be precisely measured through any other technique. The HFIR neutron activation capabilities have been used for environmental remediation operations, forensic studies, geological studies, and for studies performed by the Food and Drug Administration and the semiconductor industry [2].

2. MODERNIZATION & REFURBISHMENT SCOPE

In early 1995, the ORNL deputy director formed a group to examine the need for upgrades to the HFIR following the cancellation of the Advanced Neutron Source Project by the DOE. This group indicated that there was an immediate need for the installation of a cold neutron source facility in the HFIR to produce cold neutrons for neutron scattering research uses [3]. Cold neutrons have long wavelengths in the range of 4–12 ångströms (Å). Cold neutrons are ideal for research applications with long length-scale molecular structures such as polymers, nanophase materials, and biological samples. These materials require large scale examination (and therefore, require a longer wavelength neutron). These materials represent particular areas of science at the forefront of current research initiatives that have a potentially significant impact on the materials we use in our everyday lives and our knowledge of biology and medicine.

A team was formed to examine the feasibility of retrofitting a cold source facility into one of the four existing HFIR beam tubes. This feasibility study determined the HB-4 beam tube to be the most practical location for this cold source. Although all four beam tubes are capable of providing a high flux environment for a cold moderator, the HB-4 location was chosen based on the readily available space outside the reactor building in-line with the beam tube for a future cold neutron guide hall [3]. A pre-conceptual design study was completed

in late 1995 that identified a supercritical hydrogen system as the best type of cold source for the application. In this design concept, a moderator vessel containing supercritical hydrogen at approximately 18 K is located in the tip of the beam tube where the thermal neutron flux is highest. Because hydrogen interacts strongly with neutrons and because neutrons have a relatively similar mass, the hydrogen absorbs much of their energy, lowering the neutron temperature. This results in a neutron with less energy and a longer wavelength than originally produced in the reactor environment. It was predicted that a cold neutron beam produced by this design would be comparable in cold neutron brightness to the best facilities in the world. Given this information, the HFIR cold source initiative was approved by the DOE Office of Basic Energy Sciences (BES) following a recommendation by the Basic Energy Sciences Advisory Committee (BESAC). The BESAC also recommended that a spallation neutron source be constructed at ORNL in addition to the HFIR cold source [4]. These two facilities would combine to respond to a worldwide demand for increased cold neutron scattering research capacity.

As part of the HFIR cold neutron source initiative, ORNL also provided reactor upgrades to enhance neutron scattering capabilities. This task involved the redesign of the beryllium reflector and other reactor components to provide a larger diameter beam tube for the cold source and also to enlarge the other three thermal neutron scattering beam tubes to accommodate more thermal neutron scattering instruments and to improve the flux-on-target for the existing instruments. These new reactor components were installed during a scheduled reflector replacement outage in 2001, followed by the installation of a suite of all new thermal neutron scattering instruments to take advantage of the larger neutron beams [5]. The reactor resumed operation with the three thermal neutron scattering beam tubes while design, analysis, fabrication, and installation work continued on the cold source.

Design of the cold source continued throughout 2001–2003 with only a small number of personnel dedicated to this initiative. In late 2003, the periodic BES Peer Review of the HFIR concluded that “The cold source development must become a project with a dedicated core staff having sufficient priority to get additional help as needed”. Following this review, the HFIR cold source initiative was projectized by BES under the direction of the Research Reactors Division at ORNL in order to expedite the completion of the effort [6]. The culmination of this project was two reactor outages to install the cold neutron source. The first, followed fuel cycle 406 and lasted from 23 July 2005 to 14 December 2005. The second outage following fuel cycle 407 on 13 January 2006, placed HFIR in an extended outage for the remaining duration of the installation of the cold source. Following an extensive readiness review, the DOE approved re-start of the reactor with the new cold source installed. The HFIR then resumed operation with the cold neutron source on 17 May 2007 [5]. Since that time, the HFIR has maintained near 100% predictability in operations and has operated 6 cycles over the following 10 months [5]. The two cold neutron SANS instruments have now been used for over 32 experiments as of the end of January 2008. These include studies of various composite materials and studies of biological samples for protein structure and interactions. There are over 70 requests for time on these instruments totalling approximately 300 days for the remainder of the fiscal year (anticipated 92 days at power remaining as of end of January; per the experimentation coordinator, Dr. Gregory Smith) [7].

3. COLD SOURCE DESIGN

The mission of the HFIR cold source team was to provide a world-class cold neutron source in the HB-4 beam tube that is both safe and reliable. To this end, it was necessary to design a moderator to be located in the HB-4 beam tube that would enhance the production

of 4–12 Å neutrons available to the neutron scattering instruments. This goal required a high density hydrogen moderator at temperatures of 14–20 K be located in the tip of the HB-4 beam tube where the thermal neutron flux is greatest. The tip of the HB-4 beam tube is approximately 0.10 m away from the reactor fuel, so nuclear heating of the moderator was the most difficult design constraint to overcome. A thorough analysis of this design indicated that the nuclear heating rate would be just over 2 kW. The relatively small size of the HB-4 beam tube (0.12 m o.d. at the tip), and the long, horizontal orientation, dictated that the moderator be maintained by positive forced circulation rather than the more common natural convection design of other facilities [3]. In order to avoid steady state operation in the sub-cooled nucleate boiling regime and to address potential system instabilities inside the moderator vessel due to the density change of hydrogen in the high heat flux, the design employs hydrogen in the supercritical fluid regime. This means that the cold source support systems had to be designed to circulate supercritical hydrogen pressurized to 14.15 bar (abs), at temperatures between 14–20 K, at a rate of approximately one litre per second [3]. A system of supercritical hydrogen fluid at cryogenic temperatures requires a very complex set of integrated systems. Many new systems, structures, and components and the modification of many existing HFIR systems, structures, and components were required to support these design requirements (a simplistic flow diagram can be found in Fig. 2).

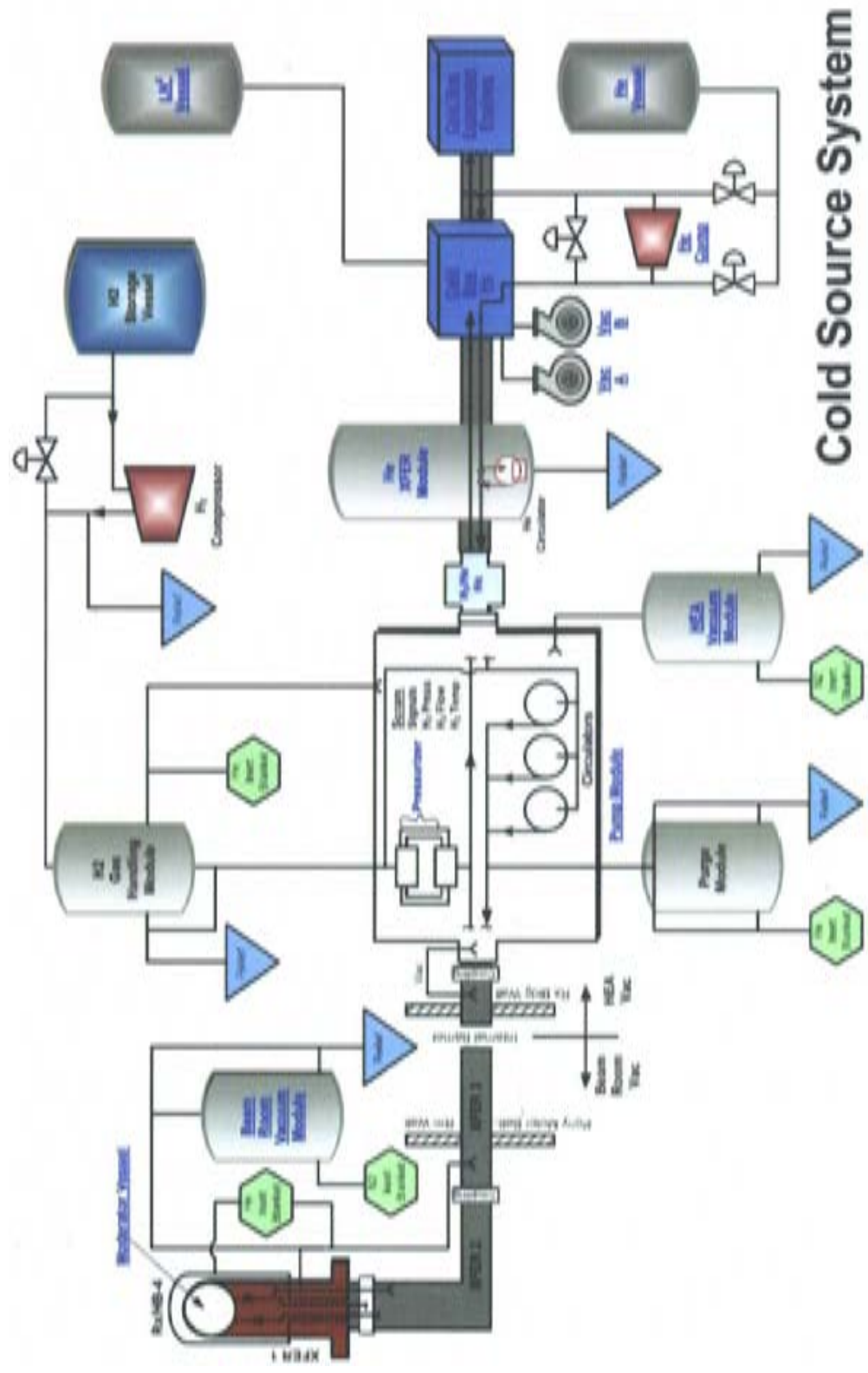


FIG. 2. Cold source simplistic flow diagram.

The existing HB-4 beam tube had to be re-designed in order to maximize the size of the moderator vessel and the viewing angle available to neutron scattering instruments. Also, the new beam tube design needed to support all of the various functions required by the containment of a cryogenic hydrogen system along with its normal function as reactor coolant boundary and pool coolant boundary. Internal to the beam tube vessel, a uniquely designed moderator vessel was designed to be placed in the tip of the beam tube to ensure optimum neutron brightness. Neutron brightness varies by thickness of the thermal moderating fluid film and its surface area visible within the instrument viewing angle. The moderator design also considered the thermal and normal stresses, and the heat transfer characteristics required to minimize localized boiling should the system pressure be reduced to liquid state [3]. Images of the moderator vessel tip are given below in Figures 3 and 4 [8]. The material of choice for the moderator vessel is aluminum. The aluminum allows free neutron movements between the reactor core and the neutron guides. A denatured aluminum alloy, T-6061, was used for its higher strength of material properties as well as the highly desirable characteristic of limiting hydrogen diffusion as a result of ingrained silicone in the aluminum crystalline structure. Exposure to radiation at cryogenic temperatures, along with the high heat flux, tends to stratify the impurities of the moderator vessel over time. This requires periodic annealing of the moderator vessel. Annealing is performed by simply warming the moderator vessel up to slightly above ambient temperature after each reactor operations cycle [9].



FIG. 3. HB-4 moderator vessel tip.

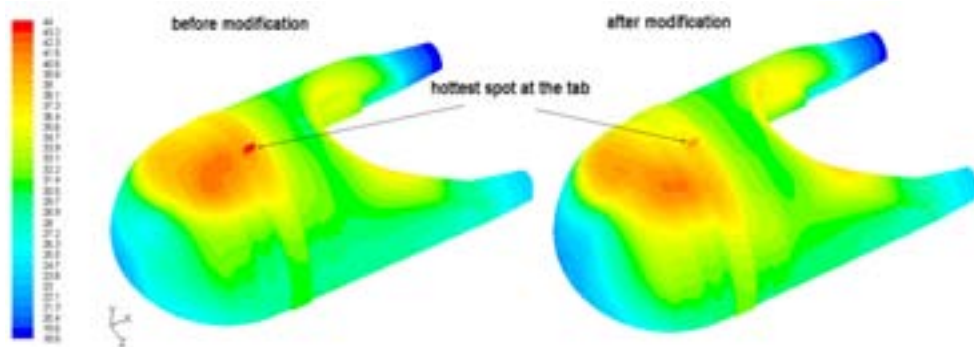


FIG. 4. Moderator vessel tip thermal analysis.

Hydrogen, when exposed to air in the proper concentrations, can result in a fire or explosion. Also, cryogenic lines that are exposed to air or water have been known to break from ice loading as moisture freezes on the exterior of the line. To address these design concerns, each cryogenic hydrogen line should be covered by one or more vacuum regions for insulation as well as one region filled with helium. It was also agreed that warm hydrogen lines would be covered by one blanket of inert gas of either helium or nitrogen. Nitrogen is

used as the inert gas blanket only in non-cryogenic applications, as it can freeze at temperatures within the cryogenic design temperature range of the system.

The moderator vessel tip is enclosed within a vacuum cavity for insulation. An additional outer layer surrounds the vacuum cavity with helium. Two separate inlet and outlet tubes circulate the cryogenic hydrogen through the moderator vessel. To minimize special requirements, these tubes connect to a single hydrogen transfer line. This transfer line is designed with five concentric piping lines in keeping with the hydrogen safety design philosophy. The innermost line routes the hydrogen supply. The annuli around this line contain insulating vacuum, hydrogen return, insulating vacuum, and an inert helium blanket, respectively. This unique transfer line is designed to minimize heat loss in the beam tube. It connects at the back of the beam tube to a second transfer line that has separate supply and return lines, each covered by insulating vacuum and then helium. These lines travel through the beam room and out of the reactor building. At the beam room wall, there is a barrier in the vacuum and helium lines. This design partitions the vacuum and helium service for the beam tube so that it does not communicate with the vacuum and helium systems outside the reactor building [10]. This design confines any radioactive contamination that might be generated in these lines to the reactor building. Additionally, since the cryogenic hydrogen circulates to the cold source systems outside the reactor building by design, the transfer line is constantly monitored for radioactivity, providing an alarm to warn the cold source operators in the event of radioactive contamination in the hydrogen line. An image of the concentric tube design as seen during fabrication is provided in Figure 5.



FIG. 5. Fabrication of transfer line Section 3.

Outside the reactor building, the two transfer lines are connected to the pump module. The pump module is a vessel within a vessel that houses three parallel circulators, a pressurizer, and a flow venturi. The three circulators have characteristics similar to a fan, but have specially designed impellers to move the high-density supercritical hydrogen through the system. The high density of hydrogen allows the circulator to develop the discharge pressure necessary to move the hydrogen through the circulation loop. The circulators are powered via an industrial uninterruptible power supply (UPS) to provide continued operation in the event of a loss of off-site power. The pressurizer, just upstream of the circulators, is a set of two small stainless steel vessels stacked vertically and linked by multiple small diameter tubes the mid-line of the each of the vessels. The bottom of the lower vessel connects to the hydrogen circulation line. Ambient temperature hydrogen is fed into the top of the upper vessel. A

braided copper strand strap is brazed to the lower vessel and hydrogen circulation line to thermally link it to the two. This allows the lower vessel to remain in a high density supercritical fluid state while the upper vessel remains in a warm gaseous state. This arrangement provides a stable interface between the cryogenic supercritical fluid of the circulating loop and the warm gas of the hydrogen pressure control system. A specially designed and calibrated flow venturi is located downstream of the pressurizer and circulators. The pump module also contains instruments to monitor differential pressure, absolute pressure, and temperature to ensure adequate cooling is maintained to the moderator vessel during reactor operation. An automatic reactor shutdown (referred to as a SCRAM) signal is generated based on these parameters in order to protect the integrity of the moderator vessel. Safety analysis has demonstrated that these automatic shutdowns are not necessary for reactor safety, but are prudent in protecting the substantial investment in the beam tube and other hydrogen containing components. Images of the pump module, circulators, and pressurizer can be found in Figures 6–8, respectively.



FIG. 6. Pump module.



FIG. 7. Hydrogen circulator.

The heat gained by the cryogenic hydrogen through nuclear heating and through intrinsic heat gain around the loop is transferred to the cryogenic helium system through the hydrogen/helium heat exchanger. This heat exchanger is located in the heat exchanger module

that is connected to the pump module. Like the pump module, the heat exchanger and its cryogenic components are contained in a vessel within a vessel (seen in Fig. 9). The heat exchanger itself is a commercially available, brazed aluminum, cross-flow channel heat exchanger. As a result of the heat exchanger materials and construction, its rate of temperature change must be controlled to prevent possible thermal stress that could fail the brazing. Strict limits are imposed on heat exchanger differential temperature and the rate of temperature change, in order to prevent damage. Also, the small channel design of the heat exchanger makes it easy to become blocked. Stringent hydrogen and cryogenic helium system cleanliness requirements are imposed to preclude heat exchanger channel blockage. The hydrogen flow channels can also become blocked by frozen hydrogen, which is mitigated by design and control aspects incorporated into the cryogenic helium system.



FIG. 8. Hydrogen pressurizer and temperature element location map.



FIG. 9. Hydrogen-Helium Heat Exchanger Module.

Three separate vacuum systems provide the vacuum insulation for the hydrogen circulation system and the cryogenic helium system. One vacuum system is integral to the cryogenic refrigerator and provides insulating vacuum for the cryogenic helium system from the refrigerator cold box up to the cryogenic helium transfer line connections to the heat exchanger module. The next system provides vacuum insulation for the cryogenic hydrogen lines in the beam tube including the moderator vessel and the associated hydrogen transfer lines. The third system provides vacuum on the remainder of the transfer lines, the inner vessel of the pump module, and the inner vessel of the heat exchanger module. Both of these vacuum systems are serviced by interchangeable vacuum modules that contain a roughing pump, turbo pump, turbo pump controller, vacuum gauges, residual gas analyzer (RGA), and a communications computer. This design allows for the quick replacement of either module with a spare. Each module contains a fast acting valve to isolate it from the system in the event of inadvertent system pressurization or of a vacuum pump failure. The vacuum pump effluent is constantly monitored by the RGA for the presence of hydrogen, helium, oxygen, and nitrogen. Additional piping and valves connect the hydrogen system to the vacuum line for using the RGA to determine hydrogen concentration.

A hydrogen gas management system is required to maintain and control pressure in the hydrogen circulation system. This is achieved via independently controlled feed and bleeds valves. These valves control the flow of ambient temperature hydrogen to and from the connection at the top vessel of the pressurizer. These valves are located in the gas handling module which maintains an inert helium environment around the valves and associated components. Hydrogen that has been bled off from the system is returned to a low pressure storage vessel located away from the reactor building. The bleed valve seat was specifically designed to handle liquid hydrogen in upset conditions. An additional process relief valve provides for faster system response around the bleed valve in upset conditions.

The pump module, heat exchanger module, gas handling module, and one vacuum module are located outside the reactor building inside the Hydrogen Equipment Area (HEA). The HEA is an enclosed space with specialized posting for hydrogen hazards, as well as oxygen deficiency monitoring, hydrogen detection monitoring, and a fire detection and suppression system. A dedicated exhaust system provides normal ventilation to the area, and provides emergency, high volume flow when oxygen deficiency or the presence of hydrogen (above the lower flammability limit) is detected. Also, a special louvered cupola on the HEA is designed to relieve the overpressure of a hydrogen fuelled explosion and deflect the pressure wave away from the reactor building.

Supply and return transfer lines connect to the Gas Handling Module and carry the warm hydrogen gas to and from the remainder of the hydrogen pressurization system located on concrete pads away from the reactor building. Hydrogen bled off from the circulation system flows to a large, double-walled, low pressure storage vessel (seen in Fig. 10). This vessel supplies the hydrogen compressor and is sized to store the entire cold source inventory of hydrogen at ambient temperature. The hydrogen compressor pressurizes a small vessel (called the hydrogen feed vessel) which feeds back to the circulation system through the feed valve in the gas handling module. The hydrogen feed vessel is sized to contain sufficient hydrogen inventory to stabilize the hydrogen circulation loop should the thermal heat load of the reactor be lost (a reactor SCRAM). Pictures of the hydrogen compressor and hydrogen feed vessel can be found in Figures 11 and 12.



FIG. 10. Hydrogen storage vessel.



FIG. 11. Hydrogen compressor.



FIG. 12. Hydrogen feed vessel.

The cryogenic helium system provides the heat sink for the hydrogen circulation system through the helium/hydrogen heat exchanger. The hydrogen circulation system temperature is controlled by the mass flow rate and temperature of the helium entering the heat exchanger while the flow rate of the hydrogen circulation system is held relatively constant. The helium refrigerator is located in a building adjacent to the HEA. Helium from a storage vessel located outside the refrigerator building is compressed to a high pressure by any one of up to five sealed helium compressors. The high pressure helium then travels through four heat exchangers (the first two heat exchangers being enclosed in a liquid nitrogen bath). The resultant pre-cooled, high-pressure helium is then routed to any one of four expansion engines where the helium loses additional heat through work, cooling the helium to approximately 14 K. The helium then travels to the helium transfer module. The normal heat power of the reactor, including base heat load of the hydrogen and helium systems is approximately 2 kW. This requires a normal operating configuration of three helium compressors and three expansion engines. The four refrigerator heat exchangers are located internal to a vacuum insulated vessel (commonly referred to as a cold box), and the expansion engines are mounted on top of an adjacent vacuum insulated vessel (called the expansion engine pod).



FIG. 13. Helium compressor skid.



FIG. 14. Helium refrigerator expansion engine pod.

From the helium refrigerator, the cryogenic helium flows to the helium transfer module. The helium transfer module is a vacuum insulated vessel that houses a specially designed helium circulator, two immersion heaters, and two control valves. During routine operations, the cryogenic helium flows through the first immersion heater (called the trim heater) in the helium transfer module. The trim heater controls the helium temperature entering the helium/hydrogen heat exchanger. This in turn controls the hydrogen circulation system temperature at steady state conditions. It also provides safety protection to prevent a helium supply temperature of below 14 K, which is the freezing point of hydrogen at normal operating pressure. Two control valves throttle to control helium flow to the hydrogen/helium heat exchanger. The helium flow returning from the helium/hydrogen heat exchanger then flows through the second immersion heater in the helium transfer module called the compensating heater. This heater is intended to provide a heat load to the returning helium that is equivalent to the nuclear heating in the event of a loss or reduction of reactor heat power. This provides stability in the control of the cryogenic refrigerator during reactor transients. In the event of loss of off-site power, a battery powered helium circulator is provided in the helium transfer module to provide continued helium flow to the helium/hydrogen heat exchanger until diesel backup power would support operation of a helium compressor.



FIG. 15. Helium refrigerator cold box.

A primary concern with the design of any cryogenic system is the potential for trapped cryogenics to warm and overpressure the system. To address this concern, an extensive relief system is required. Furthermore, the potential for the relief lines to contain hydrogen or other cryogenics requires that the relief lines be filled with inert gas to preclude reaction with air or freeze plugging caused by the moisture present in air. Relief valves are connected to each isolable section of the hydrogen circulation system and discharge to a relief accumulation vessel (RAV) located on a concrete pad away from the reactor building. The RAV, like the

relief lines, is filled with helium. It is sized to contain the maximum possible hydrogen relief volume without activating its own relief valve that vents to the atmosphere. The intent of this design is to mitigate the potential for a large uncontrolled hydrogen release to the atmosphere. Once hydrogen has relieved to the RAV, cold source staff members can vent the hydrogen to atmosphere in a controlled evolution. In addition to the cryogenic hydrogen system, all other cavities that contain or could potentially contain a cryogen are provided with relief valves, inert relief lines, and independent relief stacks located adjacent to the RAV. The relief stacks are sized and elevated to appropriate hydrogen safety standards with redundant relief valves.

The HFIR cold source consists of a multitude of components located in many locations that must work together in order for the system to reliably operate. In some cases, these components and their controls are inaccessible during operations since they are confined in inert and vacuum cavities. As has been discussed, the operation of the cold source and the reactor are inextricably linked as the moderator vessel requires cryogenic cooling anytime the reactor operates. To ensure that the many systems of the cold source operate efficiently and reliably, a sophisticated control system is required. A digital network distributive control system was chosen as the best method to integrate the many control functions. The distributive control system (DCS) consists of a network of programmable logic controllers (PLCs) with a common programmed code. The inputs are constantly monitored and most are recorded for trending and analysis. A large system overview screen is mounted adjacent to control access screens. These screens are continuously updated to provide input for operator control. The control screens provide the cold source staff members with touch screen control of all important functions and parameters. Control stations are located in the refrigerator cold box room adjacent to the HEA and in the reactor building experiment room. An image of the large system overview screen is provided in Figure 17.



FIG. 16. Relief accumulation vessel and relief stacks.

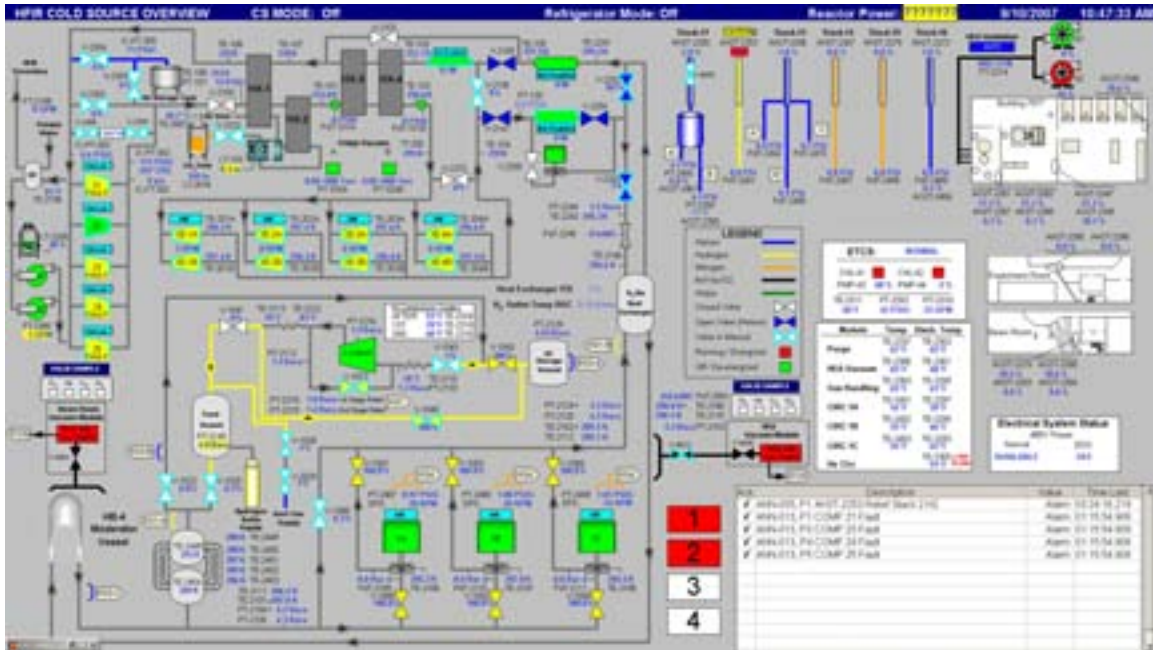


FIG. 17. Cold source overview screen.

A cold source is only useful if sufficient experimentation facilities are available. To that end, a new cold neutron guide hall was constructed outside the reactor building in line with the neutron beam emanating from the HB-4 cold source. The HB-4 cold source illuminates four super-mirror cold neutron guides that channel the neutrons into this facility for neutron scattering research. Currently, this facility houses two small angle neutron scattering (SANS) instruments that are located at the end of cold guides CG-2 and CG-3. The CG-2 SANS is dedicated to materials research while the CG-3 SANS is dedicated for biological research. Cold guides CG-1 and CG-4 are capable of supporting more instruments that are in varying stages of development. Seven instruments are planned for application on the available HB-4 beam line [2]. The most mature of these is a joint USA/Japan cold triple axis spectrometer that is now being installed on CG-4. A picture of the facility is provided in Figure 18.



FIG. 18. SANS-1 and SANS-2 facilities.

4. SAFETY ANALYSIS

The HFIR cold source involved the development of many new structures, systems, and components, and the modification of many existing structures, systems, and components, as well as the introduction of hazards such as cryogenics, oxygen displacement gases, and hydrogen. Therefore, the operation of the Cold Source at the HFIR was judged to be an unreviewed safety question during the conceptual design process. To address the new hazards introduced by the cold source, a Documented Safety Analysis (DSA) was prepared following the format of Department of Energy (DOE) DOE-STD-3009, Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Documented Safety Analyses. The cold source DSA was developed as a companion to the existing HFIR Updated Safety Analysis Report (USAR) as part of the safety basis for the HFIR. To accomplish this, the cold source DSA addresses [9]:

- Cold source accident scenarios with the potential to initiate or exacerbate an existing USAR event having off-site radiological consequences.
- Cold source accident scenarios with the potential to create a new event.
- Cold source accident scenarios that could affect the performance of safety-class or safety-significant SSCs on the Safety-Related Equipment List.
- Cold source accident scenarios that could affect currently required HFIR operator actions.

In addition to reactor safety, the cold source DSA addresses non-radioactive hazards to determine if any of them are unusual hazards that could impact facility and co-located workers. The only unusual hazards carried forward from hazard screening for evaluation for their intrinsic hazards are the HFIR radioactive inventory and the disposition of the activated beam tube and components at the end of their service life. Several other hazards, notably hydrogen combustion and explosion, cryogenic temperature impacts, and cryogenic fluid pressurization, are considered as potential accident initiators involving the HFIR radioactive inventory. The design basis accidents evaluated in the cold source DSA include [9]:

- beam tube pressure boundary (with reactor coolant system and reactor pool) failures (from causes other than internal cold source system failures);
- gas releases outside the reactor building (explosions and oxygen deficiencies);
- gas releases in the beam tube;
- gas releases in the reactor building (explosions and oxygen deficiencies);
- hydrogen releases in the HEA;
- natural phenomena (earthquake, lightning, wind, and tornado);
- external man-made events;
- increased frequency of a reactor trip;
- dropped heavy loads.

Operation of the HFIR is closely coupled with operation of the cold source because the reactor cannot operate at power unless the cold source is operating to circulate cryogenic fluid to remove the heat transferred to the moderator vessel. However, detailed analyses of the HB-4 beam tube have shown that it retains the ability to perform its reactor coolant system pressure boundary safety function even after loss of heat removal to the internal moderator vessel. Therefore, the reactor SCRAM function from cold source inputs, although important to continued operability of the Cold Source, is not required for reactor nuclear safety [9]. The interface with the HFIR SCRAM system is designed to ensure that cold source SCRAM failures do not prevent the normal reactor SCRAM from functioning.

The safety of the cold source facility is enhanced by its design, which has taken advantage of lessons learned at similar facilities around the world. Industrial standards for handling hydrogen and cryogenic fluids have also been applied. Some examples of design features that incorporate best practices from similar facilities include:

- Cold source cryogenic systems are insulated by high vacuum.
- Vacuum spaces around cryogenic hydrogen pipelines are monitored by residual gas analyzers for in-leakage into the vacuum spaces.
- Cryogenic hydrogen sections of the system are provided with slightly pressurized blankets of helium to further preclude outside air from mixing with the hydrogen.
- Blanketing regions are also protected from over-pressurization by relief devices.
- A ventilation system has been provided in the HEA that is designed to increase exhaust flow upon detection of hydrogen in the area.

Cold source operating requirements identified in the DSA have been integrated into the HFIR Technical Safety Requirements (TSRs). New Limiting Conditions for Operation (LCOs) have been added to the HFIR TSRs to address operability of the vacuum and hydrogen circulation system pressure relief systems and the beam room vacuum isolation valves and to control loads transported by hoisting and rigging over the HB-4 beam tube [9].

The cold source DSA was approached in two phases. The first addressed the final system design and the hazards identified with the facility to allow major system testing and the presence of hydrogen on-site, but did not address reactor operation with hydrogen. The second phase relied on the safety analysis modelling and the system testing results that confirmed the safety analysis modelling to address the case of reactor operation with hydrogen. The completed DSA was submitted to the DOE for review and approval prior to reactor operation with the cold source installed.

In preparing the cold source DSA, a systematic approach was taken to identify hazards, starting with hydrogen at the core and working through the flow backwards to the relief stacks. All possible failure scenarios of the HB-4 beam tube were addressed, including thermal cold shock, internal over-pressurization, and overheating. The results of an HB-4 failure scenario analysis drove the design process to include exhaustive testing throughout the fabrication process. The net result is an HB-4 beam tube that retains primary coolant system and pool boundary for all cold source initiated accident scenarios. From HB-4, failure analysis continued through guillotine break of the hydrogen transfer lines at multiple locations inside the reactor building. Finally, explosion calculations were performed for both the beam room, HEA, and hydrogen storage vessel pad based on a guillotine break scenario to ensure nuclear reactor safety is maintained. The cold source DSA was integral to the cold source design by providing the complex analysis required to evaluate proposed system and component design alternatives. Therefore, the time and personnel resources required to create the cold source DSA were comparable to those required to fabricate and install the cold source.

5. OPERATIONAL BASES ANALYSIS AND TESTING

Several advanced computational tools and modelling were used to design the cold source system. First, typical nuclear heating computations and models were developed through Monte Carlo calculations using the MCNP code [3]. Additional thermal modelling and fluid flow calculations were required in designing the moderator vessel and the required pressure, temperature, and fluid flow rates. A version of the Advanced Thermal Hydraulic Energy Network Analyzer (ATHENA) software code was developed specifically for

cryogenic hydrogen with specific emphasis on its supercritical phase in order to model the thermodynamics of the hydrogen system [9]. This required an extensive literature research and benchmarking since no other thermohydraulic code exists for supercritical hydrogen. Many other computational codes, including CFX-4, and COMSOL, were used to model the cold source for design and safety analysis [11]. Of specific concern was the possibility of thermoacoustic oscillations in the hydrogen at the surfaces of the moderator vessel where heat flux is highest. Thermoacoustic oscillations are initiated by thermal oscillation of the cryogenic hydrogen as the exciting force for a standing acoustic wave. This acoustic wave can result in severe pressure oscillations as well as additional thermal load to the system. This phenomenon has been known to be devastating to vessels at heat fluxes higher than those projected for the HFIR cold source, however, no data existed in the range of the HFIR heat fluxes [12]. Further literature search and modelling of previous experiments involving supercritical hydrogen demonstrated that this phenomenon was not likely; however, test objectives were included in the integrated test plan to collect data that might be indicative of thermoacoustic oscillations [13].

Several mock-up tests were conducted to verify the results of the mathematical models generated for the safety analysis and to provide empirical data where modelling was unreliable. One such test included a piping loop and mock-up moderator vessel to determine the estimated pressure drop of the system and flow induced vibration at the moderator vessel tip. The test medium was xenon gas at about 0°C. In these test conditions, xenon provides correlated hydrodynamic characteristics with cryogenic supercritical hydrogen (based on calculated fluid defining properties such as Reynolds number). Also, multiple circulator designs and a mock-up of the moderator vessel were tested using cryogenic hydrogen for performance curve verification in a large vacuum chamber at the Tullahoma space centre [11]. Lastly, prior to operation of the cold source with the reactor, multiple tests were performed for commissioning and safety analysis model validation. In particular, a specially designed test heater assembly (seen in Fig. 19) was fabricated with an immersion heater and outfitted with multiple instruments and installed into the hydrogen circulation loop. The test heater allowed the ability to apply a controlled thermal heat load to the system to mimic the reactor power and provided detailed information on the system response for comparison to the computational analysis model. This included data to resolve the important questions surrounding the possibility of thermoacoustic oscillations.

Complete system testing with the test heater also provided invaluable operating experience for the cold source staff members who operate the system. Cold source staff members were intentionally chosen from staff with previous reactor operations experience. This ensured that they had the conduct-of-operations experience and a basic knowledge of the interaction between the cold source and the reactor safety requirements. Testing actually provided operating experience that contradicted the operating philosophies developed through conceptual studies. Specifically, during testing it was apparent that the cryogenic helium and the hydrogen circulation system could not maintain thermal coupling during cooldown. The helium system being much smaller and not experiencing the same density change as the hydrogen system allowed it to run away from the hydrogen system. This situation created the potential for localized freezing of the hydrogen system inside the heat exchanger. This test led to the development of the current cooldown process that brings the helium system to its ultimate temperature first, and then relies on control valves in the helium transfer module to slowly cool the hydrogen system to the point of thermal coupling.



FIG. 19. Test heater assembly.

Finally, neutron time of flight testing was conducted on the HFIR cold source in October 2007 to measure the brightness of the HB-4 cold neutron beam. Brightness is a measure of the number of neutrons of a given wavelength, per steradian (solid angle) from the source per square centimetre per second. Brightness is expressed as $n \cdot \text{cm}^{-2} \cdot \text{s}^{-1} \cdot \text{sr}^{-1} \cdot \text{\AA}^{-1}$. The preliminary results of this measurement showed that the HFIR cold source brightness was approximately twice that reported for the previous world record holder, France's Institut Laue Langevin in Grenoble (these brightness results are dependent upon comparison of reporting techniques which are under review) [14]. Brighter neutron sources can provide a higher resolution in experimental data and can reduce the time required to conduct an experiment.

6. REACTOR OPERATION WITH THE COLD SOURCE

Introduction of the new cold source in the HFIR has presented significant operational changes. Most significantly, the aluminum moderator vessel, while the best material for cold neutron production, cannot withstand the reactor heat at power without significant cooling. Therefore, the reactor cannot run without the cold source in operation and at cryogenic temperatures. The cold source must remain at cryogenic temperatures until the reactor is de-fuelled or decayed to the point that decay heat will not damage the moderator vessel. Once the reactor is de-fuelled, the hydrogen circulation system must be warmed to ambient temperature to anneal the moderator vessel. Following refuelling and initial startup, the reactor must be brought up to power in deliberate stages to allow the cold source to stabilize at each power increase. Careful communication is required between cold source and reactor operations staff during reactor startup, operation and shutdown. Hydrogen system alarms are annunciated in the reactor control room as well as in the cold source control room so that appropriate response actions can be taken by both parties.

Operation of the cold source has also proved to have significant impact on the HFIR resources. The cold source consumes approximately one fifth of the entire HFIR operating and maintenance budget. This includes labour, utilities, spare and replacement parts, and consumable materials (liquid nitrogen and a large supply of helium being two of the larger cost items) [15]. Maintaining such a large, interconnected piping system requires a significant increase in labour costs for pipe fitters and refrigeration mechanics. The cold source facility is approximately one third the size of the reactor in physical terms and has almost an equivalent linear length of piping [16]. Additionally, the cold source has added a large number of instruments that must be maintained and calibrated; almost the same number of instruments as installed to support the reactor facility [17]. This increase in equipment and maintenance

scope requires the attention of system engineers, safety analysts, and other maintenance personnel. The cold source maintains an operation staff of one shift staff member, one additional day shift staff member, and one operations lead. The reactor continues to maintain a staff of one operations engineer, a shift supervisor, two weekday shift supervisors, and three operators per shift [18].

7. PROJECT MANAGEMENT

The cold source initiative as advanced in 1996 assumed that the cold source could be designed and constructed using existing technologies; with a graded approach to applying regulatory requirements since the majority of the facility was to exist outside of the reactor building; and employing many of the existing resources at the HFIR used in the routine operation and maintenance of the reactor. This represented the approach used to design and construct cold sources at other reactors and accelerators around the world up to that point in time. Based on these assumptions, a budget of US \$5 000 000 and a schedule of 2 years, 5 months were estimated at the time of the pre-conceptual design report [3]. As the initiative proceeded, it became evident that these assumptions were incorrect and that the technical scope of the cold source would be much more of a challenge than had originally been conceived. The HFIR reflector replacement and upgrades outage concluded in 2001 without an installed HB-4 beam tube since cold source development was still in progress. Following the 2003 BES Peer Review, the HFIR cold source initiative was projectized by BES under the direction of the Research Reactors Division at ORNL in order to expedite the completion of the effort. By August 2005, nearly all project risks identified in the 2004 project plan had been realized. All available reactor staff was redirected to the cold source project in order to address the concerns of the DOE regarding the cold source project completion. An aggressive schedule was established, and the final result was installation of all the major components of the cold source for testing in September 2006, continued installation and testing from September 2006–May 2007, and combined cold source and reactor operation in May 2007. The estimated total cost of the HFIR cold source was approximately US \$70 000 000.

Many of the project risks encountered once the cold source initiative was formally projectized were the direct results of design and procurement decisions made during the early stages of the initiative under a much less stringent set of base assumptions. One particular example was the hydrogen circulator design. Soon after work began on the cold source in the late 1990s, a variable speed circulator design with magnetic bearings was developed with very high circulator shaft speeds to allow more developed mass flow at lower density conditions. This very desirable attribute would potentially allow for higher mass flow rates at low density conditions, thus potentially enabling the reactor to run without the cold source being at cryogenic state. The design and fabrication of a full set of circulators was performed in collaboration with two specialty companies. Testing in cryogenic hydrogen was not economically feasible, so all testing was performed with nitrogen and helium. Once a cryogenic hydrogen loop was made available at the Spallation Neutron Source in 2005, one of the circulators was tested. The basic design was found to be unreliable at its expected operating conditions. Modification of the existing circulator design was possible, but would not guarantee a positive solution and could potentially cause significant project delay. Quickly this project risk was evaluated resulting in a decision to procure new circulators of a standard, already proven design. This change also resulted in a cascading effect on the analysis of system response to transients and to the development of operating philosophies and procedures.

An evaluation of the scope and status of the project during August 2005 revealed that significant work remained. In order to address this need, all available HFIR resources were redirected to this project effort. Each engineer and technician was assigned specific systems and components for the duration of the project. All totalled, over 15 engineers, 12 safety analysts, over 30 other technical staff resources, over 25 engineering contractors, and approximately 150 different craft personnel were utilized to finish design, fabrication, and installation of the cold source [19]. Throughout the installation phase of the project, weekly meetings were held to coordinate efforts between engineers, and daily meetings were held to coordinate efforts between engineering and craft resources. These meetings were vital to minimizing impact from design changes, maximizing resources, and ensuring continued progress on a daily basis. These meetings would be used to discuss design concerns or changes, planned field work for the day and week, and address high risk milestones.

Procurement administration and control also played a significant role in this large and complex construction project. The cold source system is a combination of nuclear grade components, cryogenic components designed for hydrogen, and standard industrial components. Many components required a high level of quality control and quality assurance. Clearly, with the large number of items, fabrications, and services being procured, a tremendous effort of multiple quality assurance personnel was required. Source inspections were performed for many components and each individual component was inspected by the installation task leader as well as a secondary quality assurance. Also, as components were installed, functional testing of individual components and portions of systems proved essential to establishing quality control. The most frequent functional test was a helium leak test. Baseline leak testing provided invaluable information that allowed identification of individual quality suspect parts or connections. Each system has a total system leak test baseline to which it is still measured against in maintenance troubleshooting today. To date, from the operation of its first operating cycle, the cold source has not had any significant failures that have resulted in unscheduled reactor shutdown or delayed startup. This is a testament to the quality and attention to detail provided by everyone involved in the HFIR cold source project.

8. LESSONS LEARNED

In any large scale construction project, personnel safety is of foremost concern. The necessary proximity of ongoing construction and personnel increases the risk of injury. Careful focus on personnel safety was maintained throughout the project. A job hazard evaluation is required by HFIR procedures and is inherent in the HFIR safety culture. Common practice is to review the safety requirements with the associated personnel (craft, operations, and engineering) prior to daily work execution, as well as anytime assigned personnel change or safety conditions change. Communication during the daily work status meetings as well as between craft and field personnel became the best vehicle for ensuring that a focus on personnel safety was maintained at all times. Over 1 000 000 man-hours of work were achieved with no lost time injury. Personnel safety remains a focus of the HFIR and ORNL throughout all activities [19].

Operations personnel were fully included in the HFIR cold source design team from its inception in 1996. This ensured that an operations perspective was applied to all designs. Operations also lead the integrated systems testing effort with close collaboration with the design engineering and safety analysis leads. While the testing requirements and test sequences were generated by all three groups, cold source operations had the final decision in moving forward with any given test. This approach ensured an emphasis on personnel safety

and protection of unique equipment. It also had the added benefit of a final result of systems interfaces that are more intuitive and more feature-rich to the HFIR operations personnel.

One of the successes of the HFIR cold source design was the decision to modularize the various cryogenic components by function. This allows maintenance to be performed on internal components without affecting the entire system. The vacuum modules are a good example of this design application. Three identical modules were fabricated for two service locations. These modules can be interchanged so that maintenance on the internal vacuum components can be performed off-line while the other two modules support continued cold source operation. Redundant, parallel hydrogen circulators are another similar example of this modular design. The system is designed to allow maintenance on an individual circulator without disruption of the remainder of the system.

The primary lesson learned from a project management perspective was that such a complex project should be performed strictly applying accepted project management principles from the beginning. The original assumption that a graded approach to cold source requirements could not be realized because of the high degree of integration of the cold source structures, systems, and components with those of the reactor. Requirements should be clearly defined and adhered to through the duration of the project. Requirements should include not only the end use requirements such as brightness and neutron beam size, but should also address nuclear safety, personnel safety, operability, and reliability. Significant technical issues should be fully addressed prior to finalization of design and the beginning of fabrication. Prototypical components that are unproven should be tested in operating conditions prior to fabrication as in the case of the hydrogen circulators. Lastly, the project schedule and budget should contain sufficient contingency to absorb any uncertainties or identified project risks.

Very few reactor-based cold neutron sources have been designed and operated to date. The creation of each cold source then, is a unique project. The heat loads, geometry, regulatory requirements, and existing support facilities vary greatly. Therefore, the experience acquired from prior cold source projects alone cannot be used to establish cost and complete scope assumptions for a new cold source proposal. While previous experience is a useful reference, it must be combined with the specific requirements of the facility in which the cold source is to operate. Multiple considerations of site-specific applications are required when applying experience from previous cold source projects. This is best handled from the beginning via application of modern project management processes.

9. FUTURE MODERNIZATION & REFURBISHMENT SCOPE

The HFIR is facing an exciting time in its history. While the original facility was commissioned over 40 years ago, upgrades such as the cold source have provided new life for an ageing reactor. Several upgrades were already installed or under development prior to the cold source; however, several more have presented themselves as a direct result of the success enjoyed by this project.

While modernization of the HFIR facility has enabled an expanded offering of neutron user capabilities, an ageing facility requires significant refurbishment in order to continue to meet the demands for experimentation time. The HFIR already has undergone significant refurbishment scope. Complete replacement of the cooling tower, replacement or refurbishment of existing rotating equipment, modernization of the electrical distribution system, and significant changes in the reactor instrument and controls are some of the areas that have already addressed. Most recently, prior to the cold source, the HFIR had already

initiated a massive refurbishment of the reactor primary system. The primary system is maintained by any three of four large, vertical pumps. These pumps circulate the primary system water through associated heat exchangers. The pump and associated heat exchanger are located in a cell with several valves and other associated equipment. The vast majority of the equipment in the cells is original to the HFIR. The refurbishment scope included removal of the primary pump, including complete disassembly, inspection, and maintenance. The associated motor was sent out for disassembly and maintenance as well. An inspection of the heat exchanger was included, as well as inspection, maintenance, and replacement of several of the associated valves and equipment. Additional lighting, cleaning and painting, and a new access platform were also added to each heat exchanger cell. One heat exchanger cell is 90% complete, and future HFIR refurbishment scope includes the same overhaul of each remaining heat exchanger cell.

The success of the HB-4 cold source has also resulted in an internal review for a feasibility of a second cold source to be installed in HB-2. HB-2, by comparison to HB-4, provides a significantly higher neutron flux, and therefore, would be expected to provide higher neutron brightness than the existing HB-4. HB-2 is in fact, a larger access port than HB-4, allowing for a natural circulation design to be considered. Installation of this new cold source would be best achieved during the next beryllium reflector replacement, scheduled for 2020 [2].

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REFERENCES

- [1] OAK RIDGE NATIONAL LABORATORIES, HFIR Informational Brochure, ORNL Publication No. ORNL-2006-G00199/jcp, Oak Ridge, TN (2006).
- [2] OAK RIDGE NATIONAL LABORATORIES, Neutron Sciences, *About; Neutron Science; Facilities*, (2008), <http://neutrons.ornl.gov/>
- [3] OAK RIDGE NATIONAL LABORATORIES, High Flux Isotope Reactor Cold Source Preconceptual Design Study Report, ORNL Publication No. ORNL/TM-13136, Oak Ridge, TN (1995).
- [4] UNITED STATES DEPARTMENT OF ENERGY, Basic Energy Sciences Committee (February, 1996), BESAC History, (2008), <http://www.sc.doe.gov/bes/besac/history.htm>
- [5] OAK RIDGE NATIONAL LABORATORIES, CMMS (February 2008), HFIR Cycle Dates, (2008), <http://home.rrd.ornl.gov/cfpro/CMMS/index.cfm?fuseaction=s.CyclesAll>
- [6] MONTANO, P.A., Letter Correspondence to Dr. James Roberto Re: BES Review Findings (Department of Energy), Germantown, MD (2003).
- [7] SMITH, G.S., Monthly Experimentor's Report, Oak Ridge National Laboratories, Oak Ridge, TN (2008).
- [8] FREELS, J., Calculation Independent Review, HFIR Calculation No. 2005-228 C, Oak Ridge, TN (2006).
- [9] OAK RIDGE NATIONAL LABORATORIES, High Flux Isotope Reactor Cold Neutron Source Documented Safety Analysis-Phase 2, ORNL Document No. ORNL/RRD/INT-138/V2 R1, Oak Ridge, TN (2007).
- [10] OAK RIDGE NATIONAL LABORATORIES, Cold Source LH2 Transfer Line Section 3A & 3B, Input/Return Assembly & Parts List, HFIR Drawing No. NEXANS21013303000A, Oak Ridge, TN (2006).

- [11] OAK RIDGE NATIONAL LABORATORIES, High Flux Isotope Reactor Cold Neutron Source Reference Design Concept, ORNL Publication No. ORNL/TM-13498, Oak Ridge, TN (1998).
- [12] FRIEDMAN, R., GRAHAM, R., HENDRICKS, R., HSU, Y., Experimental Heat-Transfer Results for Cryogenic Hydrogen Flowing in Tubes at Subcritical and Supercritical Pressures to 800 Pounds Per Square Inch Absolute, NASA Document No. D-3095, Washington, DC (1966).
- [13] OAK RIDGE NATIONAL LABORATORIES, HFIR Cold Source Testing Plan, ORNL Document No. HFIR-CST-5230 R2, Oak Ridge, TN (2007).
- [14] IVERSON, E.B., ROBERTSON, J.L., Spectral Brightness of the HB4 Supercritical Hydrogen Cold Source at HFIR, ORNL Document No. SNS-106100200-TR0161-R00, Oak Ridge, TN (2007).
- [15] OAK RIDGE NATIONAL LABORATORIES, High Flux Isotope Reactor Fiscal Year 2008 Integrated Work Plan, Oak Ridge, TN (2007).
- [16] OAK RIDGE NATIONAL LABORATORIES, Architectural and Piping Layout Drawings, HFIR Drawing Database, Oak Ridge, TN (2008).
- [17] OAK RIDGE NATIONAL LABORATORIES, Instrument Records, HFIR Instrument Index Database, Oak Ridge, TN (2008).
- [18] OAK RIDGE NATIONAL LABORATORIES, March 2006 and February 2008 Organizational Chart, <http://home.rrd.ornl.gov/orgcharts/orgchart.pdf>
- [19] OAK RIDGE NATIONAL LABORATORIES, SAP Report for Hours Charged to an Account, ORNL SAP Database, Oak Ridge, TN (2008).

DALAT RESEARCH REACTOR AND ITS CONTROL AND INSTRUMENTATION SYSTEM RENOVATION

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Abstract

The Dalat Nuclear Research Reactor (DNRR) was reconstructed from the 250 kW TRIGA Mark II reactor built in early 1960s. The reconstruction work was started in March 1982 and the first criticality of the reconstructed reactor was achieved on 1 November 1983. The reactor has been operating at a nominal power of 500 kW since March 1984. From 1992–1996, a national project for inspection and refurbishment of the reactor and its technological systems was implemented. The reactor instrumentation and control (I&C) system was also renovated during 1992–1993 under IAEA TC project VIE/4/010. A main goal of the VIE/4/010 project was to redesign and construct a number of electronic systems and modules, which play a key role in enhancing the reliability of the I&C system. The first project on the I&C renovation in the 1992–1993 period was focused on the process and instrumentation system, but not on the neutron measurement and data processing parts. The second modernization and refurbishment programme began within the framework of the national project and was joined with the IAEA TC project VIE/4/014 during 2005–2007. The new I&C system was successfully installed and the DNRR has operated with the new I&C system since April 2007. Some results of the project for modernization of the reactor I&C system are presented in this paper.

1. INTRODUCTION

The DNRR core is loaded with Soviet WWR-M2 fuel assemblies. Until now, six core configurations have been used. The first five configurations were loaded from 1984–2006 with HEU of 36% enrichment. The sixth core configuration was loaded in September 2007 with a mixed core of 98 HEU fuel assemblies and 6 LEU fuel assemblies of 19.75% enrichment. The number of fresh LEU fuel assemblies in storage is enough for more than 15 years of reactor operation. In this respect, safe operation and effective utilization of the DNRR to at least 2020 is a long term objective.

The original DNRR I&C system was designed and manufactured by the former Soviet Union and put into operation in November 1983. In general, the system was reliable and had proven its capability to ensure safe reactor operations. However, it was difficult to find spare parts and the 1970s technology of the system, with its discrete and low level integrated electronic components, had become obsolete and was not well-adapted to the tropical climate. The reliability of the I&C system was degraded because of ageing of equipment and electronic components, which were the main reasons for its renovation and modernization.

The first renovation was implemented in 1992–1993 within the framework of the IAEA TC project VIE/4/010 [1]. The main renovation task was to redesign and construct a number of electronic systems and modules that played a key role in enhancing the reliability of the system. Because the first renovation was mainly focused on the process and instrumentation system, it was necessary to propose a plan for the second modernization and refurbishment during 2005–2007 with the replacement of neutron measurement and signal processing parts of the original I&C system. The shared budget of the national project and the IAEA TC project VIE/4/014 has been received for this modernization.

2. DESCRIPTION OF THE ORIGINAL DNRR I&C SYSTEM

The reactor I&C system can be divided into four main parts: neutron flux control system (NFCS), reactor data display system (RDDS), control logic system (CLS), and process and instrumentation system (PIS).

2.1. Neutron Flux Control System (NFCS)

The NFCS system measures reactor power and period; giving out analogue signal proportional to the unbalance between reactor and setting powers; and giving out alarm and scram signals on power and period for each measuring channel.

The NFCS system consists of three identical and independent electronic units namely, the information acquisition and processing unit (IAPU). The neutron flux is measured over ten decades ($10^{-8} - 1.2 \times 10^2\%P_n$, $P_n = 500$ kW) and is covered by three overlapping measure ranges by nine individual measuring channels (3 channels in each IAPU): source range (SR), — 10^{-8} – $10^{-2}\%P_n$; intermediate range (IR) — 10^{-3} – $10\%P_n$; and power range (PR) — 1 – $120\%P_n$.

To control the neutron flux of the reactor, nine neutron detectors with gamma compensation (6 fission chambers type KNK-15 and 3 ionization chambers type KNK-3) were used. The KNK-15 chambers are operated in the pulse mode for the SR and IR ranges, while the KNK-3 chambers are used in current mode for the PR range.

2.2. Reactor Data Display System (RDDS)

The designed system was used for (i) using a computer monitor to display important reactor parameters, such as power and period values of 9 measuring channels in 3 ranges, the averaged values of reactor power and period in each range, beginning and ending values of each range, safety threshold values of reactor power and period, reactor negative period, etc.; (ii) recording of reactor power and period on a 2-pen recorder; (iii) sending alarm signal if any parameter is abnormal; and (iv) managing and saving data in the PC hard drive. Because of low reliability, low quality of indication, the RDDS system was totally replaced in the framework of the VIE/4/010 project programme.

2.3. Control Logic System (CLS)

The CLS system controls the reactor in manual and in automatic modes, shuts down the reactor at power or period SCRAM signals, SCRAM signal from the PIS system, or failure of the city electricity network; and provides control console status information of the whole reactor I&C system. To control the reactor, 7 control rods are used (2 safety rods (SR), 4 compensation rods (CS), and an automatic regulation rod (AR)).

In order to increase accuracy and reliability of the rod position indicator and the rod drop time measurement, the rod position indicators and related electronic boards were replaced within the framework of the VIE/4/010 project. The indicators are of a digit and bar-graph type. In addition, all relays in the intermediate relay boards and all transistors in the amplifier boards were replaced with higher quality and more reliable items.

2.4. Process and Instrumentation System (PIS)

The PIS system has been totally renovated within the framework of the VIE/4/010 project. The system has the following functions:

- Measures and records/indicates the reactor technological parameters, such as the temperatures in various locations in the reactor tank, such as the inlet and outlet temperatures of the primary and secondary cooling loops at the heat exchanger, low water levels in the reactor and other tanks and in the reactor hall sump, water flow rates of the primary and secondary loops, air flow rate from the reactor tank space and in the reactor stack, pressure at various points on the primary and secondary loops, and conductivity of reactor water before and after the primary loop water purification system.
- Creating alarm and SCRAM signals on water level of the reactor tank and water flow rates of the primary and secondary loops and sending these signals to the CLS system to shut down the reactor.

3. MAIN RESULTS OF THE FIRST RENOVATION PROJECT

The main results obtained from the first renovation during 1992–1993 are as follows:

- Redesign and rearrangement of the control panels and reactor control console in the reactor control room.
- Replacement of the reactor operation parameter's monitoring system. The new system was built using an industrial grade PC and advanced data acquisition add-on cards.
- Design of a position indication block and a drop time measuring block for the control rods.
- Design of a new PC-based 12-channel dosimetry system. Radiation levels are displayed digitally and graphically on the PC monitor.
- Replacement of some equipment in the reactor process and instrumentation system (PIS). The main reactor technological parameters are measured, recorded, and displayed on the reactor control console.

4. MAIN RESULTS OF THE SECOND MODERNIZATION AND REFURBISHMENT PROJECT

In the first renovation project, the electronic part for neutron measurement and data processing (including the NFCS and the CLS system) was unchanged. However, it was totally replaced under the second modernization project.

To modernize the equipment of the original DNRR I&C system, the SNIIP Systematom Company from the Russian Federation was chosen as the supplier for the updated integrated control and protection system (CPS) called the ASUZ-14R complex [2]. The channel structure is used in the ASUZ-14R complex to generate control signals of the protection safety system and to ensure monitoring; the bus structure is used to implement monitoring functions and information transfer to the top level (Fig. 1). In accordance with IAEA requirements and Russian standards for research reactors, the ASUZ-14R complex ensures that the emergency protection control in several cabinets have similar functional purposes. The composition of each cabinet includes neutron detectors, equipment for

information processing and control signal generation, video check and registering devices, communication lines, etc.

The ASUZ-14R complex ensures the safety, control, check and monitoring of the reactor facility by means of the following channels and equipment:

- channels for monitoring reactor power and period by thermal neutron flux density (NFME channels);
- channels for monitoring process parameters (PPME channels);
- channels for logical processing of signals from NFME channels, from technological and supporting systems and generation of control signals for protection safety system and normal operation system (SLPE channels);
- channel for automatic power regulation (APR channel);
- channels for reactivity monitoring (RME channels);
- channel for monitoring control rod (CR) positions (RPME channel);
- information channels for displaying operative information at the control panel;
- buttons and keys on the control panel;
- equipment for archiving, diagnostics, and recording (ADR equipment).

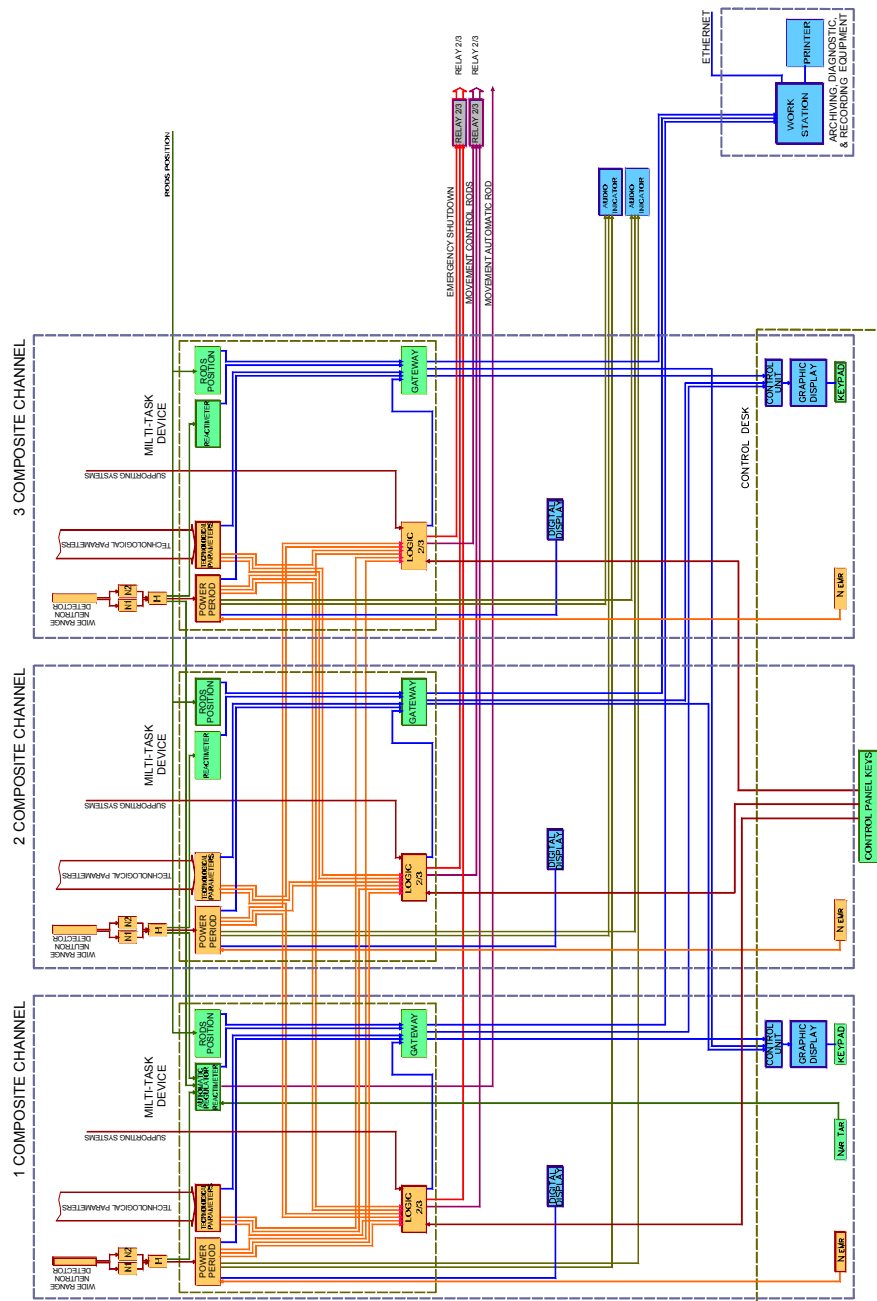


FIG. 1. Structural diagram of ASUZ-14R complex.

The designed structure of the ASUZ-14R complex is based on a functionally and structurally designed integrated unit of the control safety system (CSS), also called the UNO-251R1 cabinet, consisting of a combination of independent different channels fulfilling the functions of emergency protection control, check, and monitoring. The UNO-251R1 is built according to the principle of reasonable software and hardware redundancy of technical means (minimum 25%) for unification, subsequent modernizations and functional escalation of equipment; the redundancy is increased with as the number of tasks executed is increased (but not more than 50%). The UNO-251R1 consists of a structurally completed multitask, multiprocessor device with connected units for detection, control, displaying, and recording. The UNO-251R1 is functionally independent of other UNO-251R1s in the ASUZ-14R complex and provides an automatic check of equipment in the operation mode (up to check of discrete communication lines). At optimum configuration, the ASUZ-14R complex ensures

the functions of control of emergency protection and normal operation mode of the reactor by means of three UNO-251R1 cabinets, control panel, and ADR equipment.

At optimum configuration, the logic of the ASUZ-14R complex collects signals when exceeding the permissible values by all monitored parameters of each UNO-251R1 according to a 2/3 logic. The output signals of the emergency protection control from each UNO-251R1 are also sent to the control devices of the actuator drives (usually with electromagnetic relays) according to 2/3 logic. This structural logic of generating the emergency protection signal is dictated by increased reliability factors of the ASUZ-14R complex.

The integrated UNO-251R1 device is designed for receiving frequency, analogue, and discrete signals from communication lines of the detection units, the control panel, and other systems important to safety; for generation of discrete control signals for control safety system; for information transfer via standard interfaces RS-485 for displays, archives, protocols, and additional analysis.

The UNO-251R1 device consists of functionally and structurally completed units — independent sets of equipment fulfilling rigidly specified tasks. The UNO-251R1 is an integral part of the ASUZ-14R complex and generates a generalized emergency protection control signal of the actuators depending on value and rate of change in the neutron flux and on process parameter values. Three main channels, NFME, PPME, and SLPE participate in the generation of emergency protection control signals.

A structural diagram of the UNO-251R1 device is shown in Figure 2 and consists of the following parts:

- BPM-107R1 – neutron parameter protection unit;
- BPM-108R1 – process parameter protection unit;
- BFM-29R1 – signal logical processing unit;
- BNO-102R1 – reactivity monitoring and automatic power regulation unit;
- BNN-400R1 – supply unit.

The UNO-251R1 device executes the following main functions:

- generation of emergency protection signals and warning by reactor power level and by inverse value of reactor power increase speed (unit BPM-107R1);
- generation of emergency protection signals and warning by safety system process parameters, and monitoring of normal reactor operation process parameters important to safety (unit BPM-108R1);
- generation of generalized emergency protection signals according to 2/3 logic by neutron parameters and 1/1 logic by process parameters, generation of control signals of technological and experimental systems according to 2/3 logic on the basis of process parameter information (unit BFM-29R1, function SLPE-EMR);
- monitoring of control rod positions (unit BFM-29R1, function SLPE-CNTR);
- monitoring of reactivity and automatic power regulation (unit BNO-102R1);
- control of flow of input and output data from units located in the UNO device, transfer of data to the BKC-76R2 unit for display on FPM-3171GA control panel monitor and to ADR equipment for archiving and recording (unit BFM-29R1, function GATEWAY).

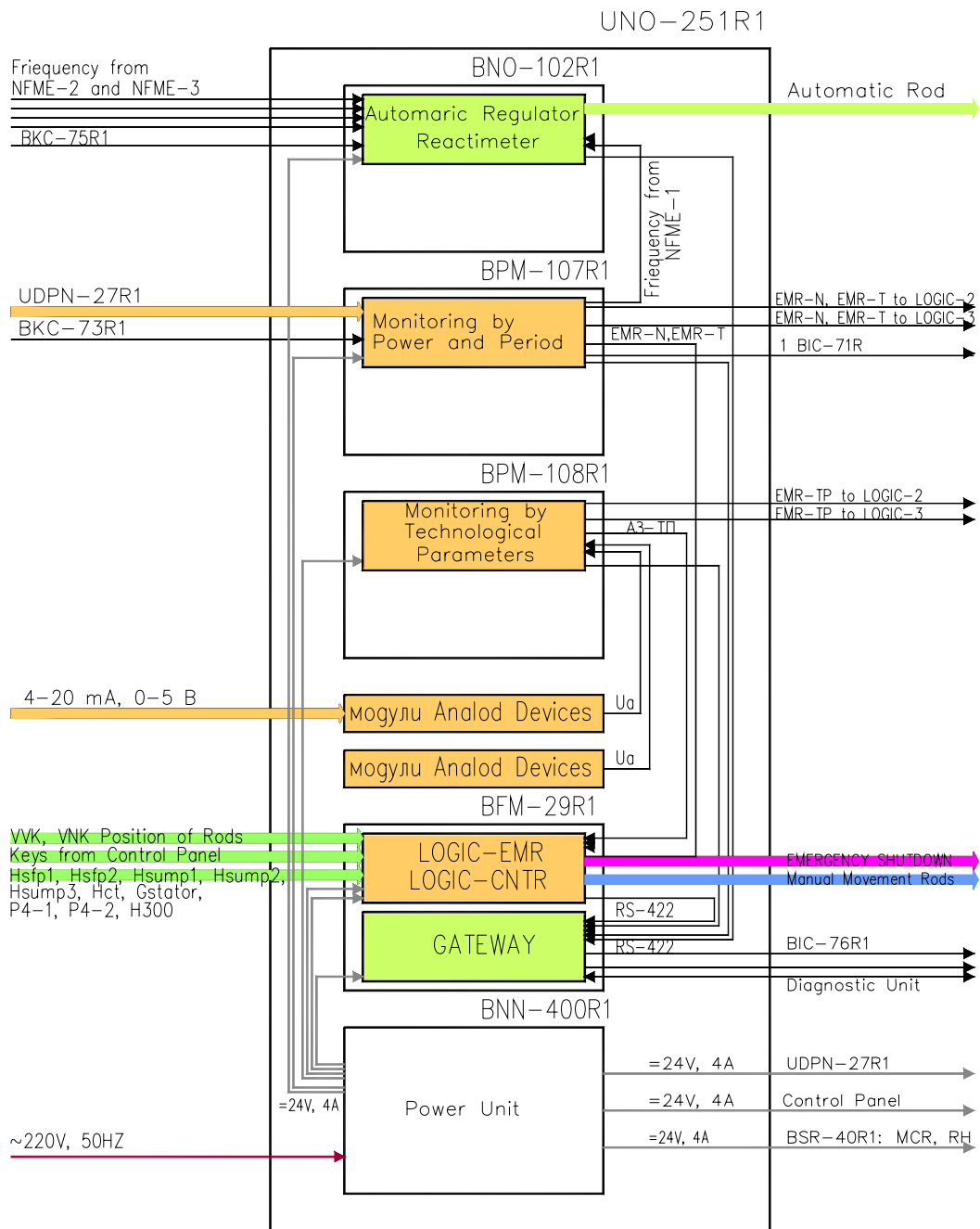


FIG. 2. Structural diagram of UNO-251R1 device referred as CSS device (on the left side the source of the information is presented, the right side shows the output commands and information presentation).

The ASUZ-14R complex has been in operation since April 2007 and licensed in October 2007 after more than 6 long cycle reactor runs.

5. LESSONS LEARNED FROM THE SECOND MODERNIZATION AND REFURBISHMENT PROJECT

- The project was delayed because it was difficult to keep the budget approvals of the two independent funding resources, national and agency, in sync.

- It was important that the supplier understands the buyer's technical specification requirements prior to contract award. In order to satisfy this issue, technical discussions should be as detailed as possible.
- The contract should be clear to both supplier and buyer. There was a misunderstanding by the buyer that training for operations staff was automatically included in the contract. The system was already installed and put into operation, but operations staff had not been trained because of ambiguity in the contract.
- Language barriers were encountered by both the buyer and the supplier. The buyer had difficulty with the Russian language and the supplier had difficulty with the English language.
- Geographical distance between supplier and buyer was also a consideration when considering potential suppliers.

6. CONCLUSION

The DNRR has safely operated for more than two decades [3]. To achieve this long life, modernization and refurbishment of the reactor technological systems were made.

The second modernization and refurbishment project of the reactor I&C system was successfully implemented in 2005–2007. As planned, the neutron measurement and data processing parts of the original I&C system were totally replaced by the new and modern system designed by Russian SNIIP Systematom Company. The modernization and refurbishment project was finished and commissioned in April 2007 and the new I&C system was licensed in October 2007.

It must be mentioned that although the reactor operates at low power and has experimental facilities, the DNRR has played, and will continue to play, an important role as a main tool for the development of a nuclear power programme in Vietnam. Therefore, a strategic plan for the existing reactor is needed and a plan for the refurbishment, upgrade, and modernization of its technological systems is still necessary in order to continue safe operation and effective utilization of the DNRR to at least 2020.

REFERENCES

- [1] Safety Analysis Report (SAR) for the Dalat Research Reactor, Second Edition, Dalat (2003).
- [2] Complex of Control and Protection System Equipment ASUZ-14R (for IVV-9 reactor), Operating Manual RUNK.506319.004 RE-E, (2006).
- [3] NGUYEN, N.D., Dalat Nuclear Research Reactor—Twenty years of safe operation and efficient exploitation, Dalat (2004).

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