RRFM IGORR 2007

LYON, France 11.3. - 15.3. 2007

Transactions

11th International Topical Meeting

Research Reactor Fuel Management (RRFM)
and
Meeting of the International Group on Reactor Research (IGORR)

Centre de Congrès, Lyon, France
11– 15 March 2007

Organised by the
European Nuclear Society (ENS)

and

IGORR: International Group on Research Reactors

in co-operation with the
International Atomic Energy Agency (IAEA)

European Nuclear Society
Rue de la Loi 57
1040 Brussels, Belgium
Phone + 32 2 505 30 54
Fax +32 2 502 39 02
E-mail ens@euronuclear.org
Internet www.euronuclear.org

These transactions contain all contributions submitted by 9 March 2007.

Session II

New Projects and Upgrades

Session II - New Projects and Upgrades

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

D. L. SELBY
Neutron Facilities Development Division
Oak Ridge National Laboratory
Building 7962-I, Oak Ridge, Tennessee 37831-6430

ABSTRACT

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

1. Introduction

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

2. Description of HB-4 Beamline Upgrades

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

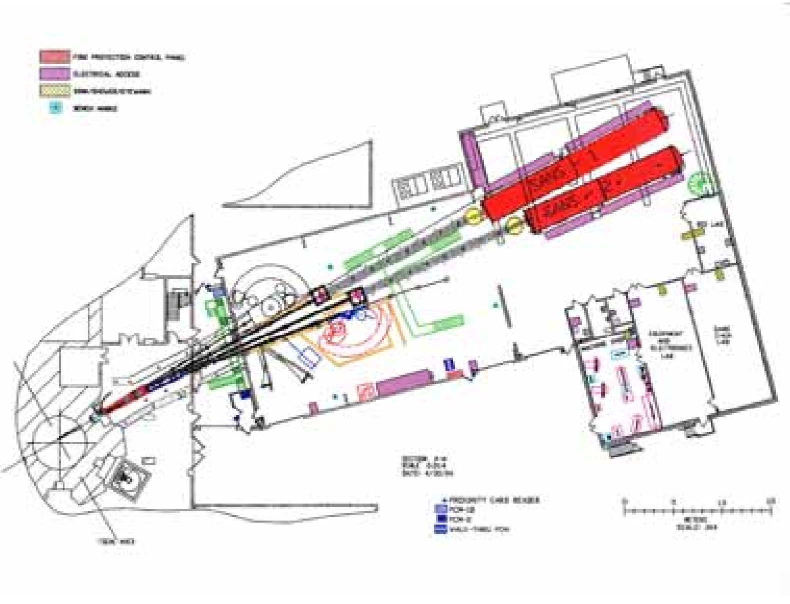

Figure 1: HB-4 Guide and Instrument Layout

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

2.1 HB-4 Shielding Structures

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

Figure 2: Open Shield Tunnel Door in Beam Room

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

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

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

Figure 3: Guidehall Shielding Layout

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

2.2 Neutron Guides

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

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

Figure 4: Guides installed in HB-4 Guidehall

2.3 HB-4 Guidehall and Instruments

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

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

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

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

3. Schedule for Completion of HB-4 Projects

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

4. Plans for Future Upgrades

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

Figure 6: Proposed New User Support Facility

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

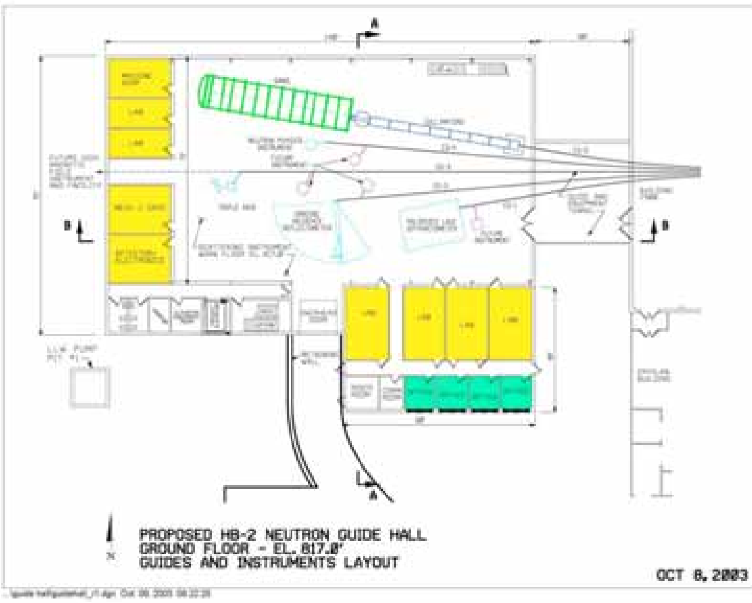

Figure 7: Early Conceptual Layout for an HB-2 Guidehall

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

P. SCHREINER, W. KNOP
GKSS-Forschungszentrum
Max-Planck-Str. 1, 21502 Geesthacht - Germany

D. COORS, D. VANVOR
AREVA NP
Freyeslebenstr. 1, 91058 Erlangen – Germany

ABSTRACT

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

1 Introduction

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

➔ Simple design (hemispherical shape) and fabrication

➔ The same material specification for the moderator chamber as for the existing chamber.

➔ The same technical inspection as for the existing one

➔ The same incident conditions (pressure, melting etc.) as for the existing one.

➔ Comparable nuclear heating for the new and exiting chamber.

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

2 Optimisation studies

2.1 Neutronic studies for improvement of performance of moderator chamber

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

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

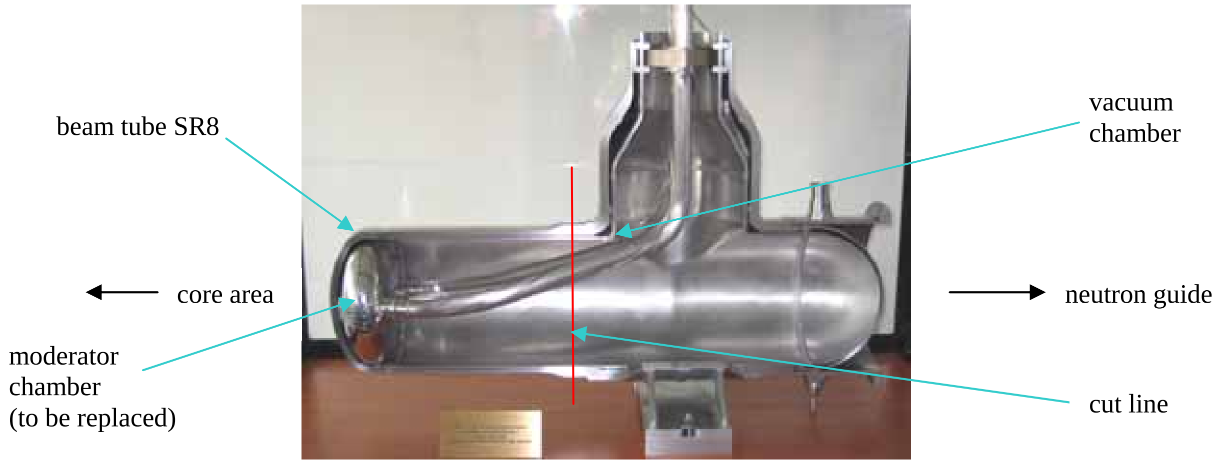

Fig 1. Inpile section of cold neutron source FRG-1, longitudinal cut through a prototype

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

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

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

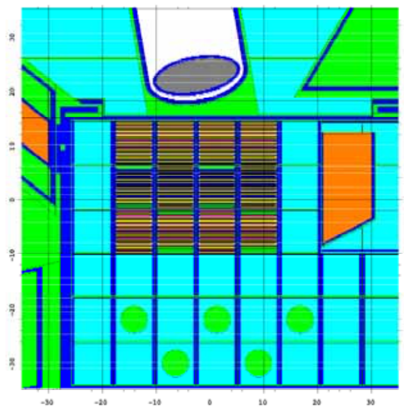

Fig 2. Cross section through core and reflector at the level of beam tube SR8

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

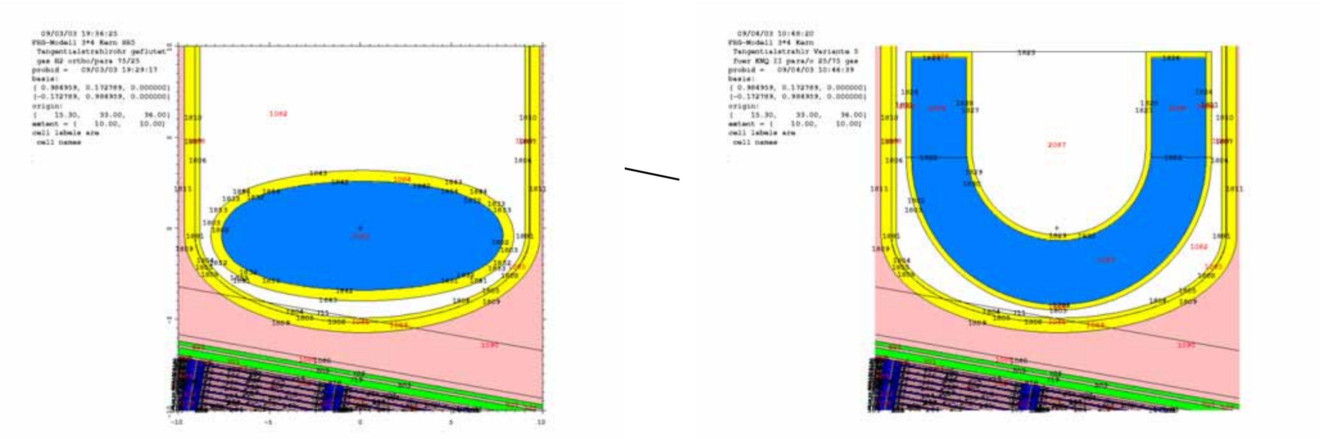

Fig 3. MCNP model of moderator chamber for reference design (left figure) and optimised design

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

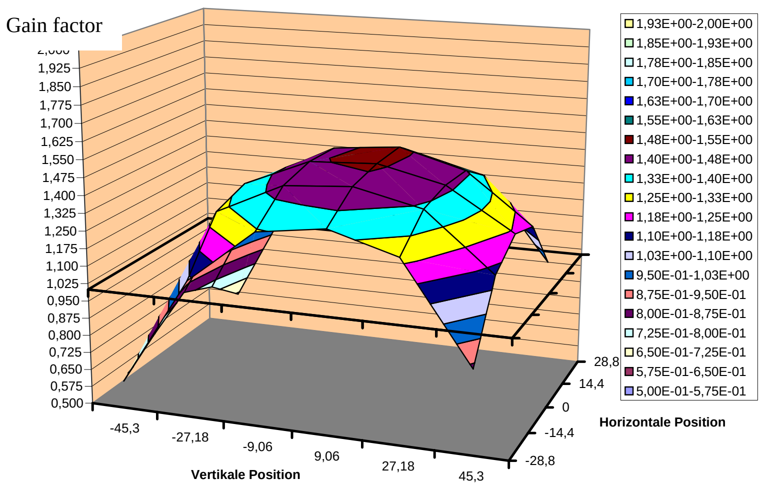

Fig 4. Gain factors in the range of 2 to 10 Å at the entrance to the neutron guide

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

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

2.2 Verification of moderator chamber design

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

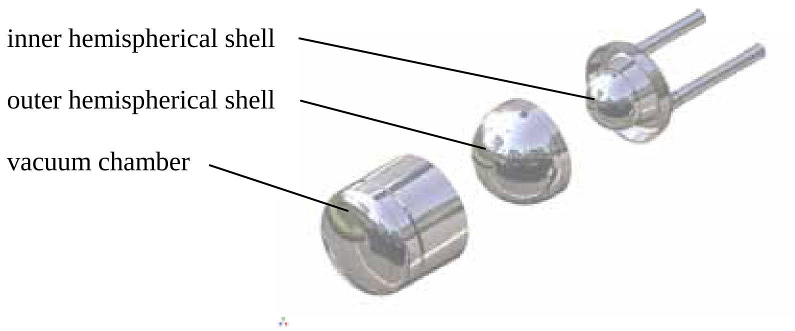

Fig 5. Exploded view of new focusing cold neutron source

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

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

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

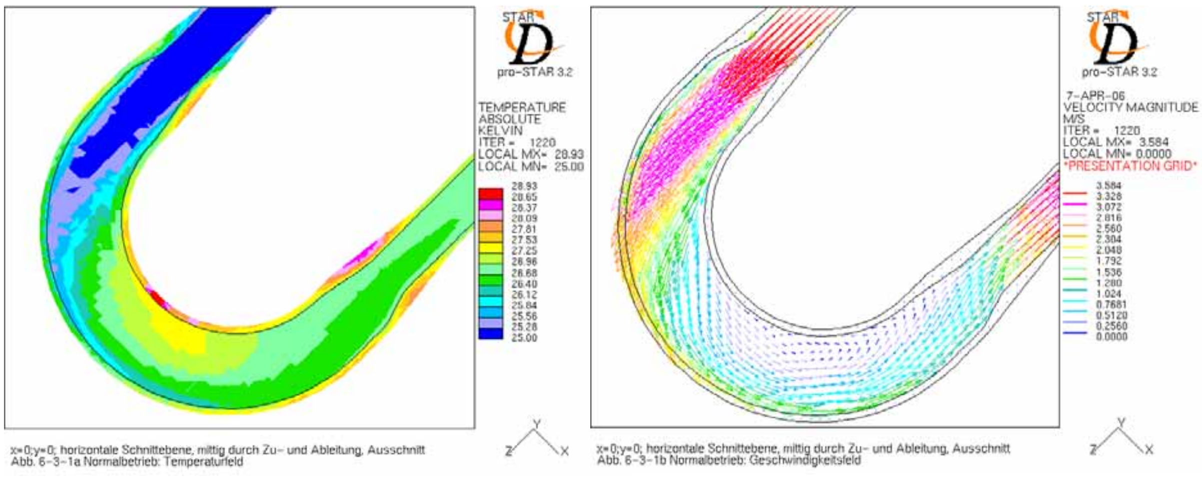

Fig 7. Temperatures (left figure) and coolant velocities in the moderator chamber at normal operation

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

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

3 Fabrication and installation

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

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

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

4 Commissioning and validation

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

5. Summary

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

6.References

[1] J. F. Briesmeister (Ed.), MCNP – A General Monte Carlo N-Particle Transport Code
[2] D. L. Selby et. al., High Flux Isotope Reactor Cold Neutron Source Reference Design Concept ORNL-Report ORNL/TM-13498, May 1998

Acknowlegments:
We thank W. Bernnat (IKE Stuttgart), H.J. Roegler (advisor to AREVA NP GmbH) and W. Feltes, K.F. Freudenstein, C. Röhlich (AREVA NP GmbH) for useful discussions and technical support.

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

T.APOSTOLOV, D.DRENSKI, I.DIMITROV

Nuclear Scientific Experimental Centre,
Institute for Nuclear Research and Nuclear Energy, BAS
72 Tzarigradsko chausse Blvd., 1784 Sofia, Bulgaria

ABSTRACT

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

1. Introduction

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

The following experimental channels are planned:

- two vertical channels in the fuel assemblies to supply fast neutron flux 3.10^{12} n/cm^{2}s;
- two vertical channels in beryllium blocks to supply thermal neutron flux 8.10^{12} n/cm^{2}s;
- seven horizontal channels outside the aluminium vessel of the reactor core with fast neutron flux $1,6.10^{12}$ n/cm^{2}s, and thermal neutron flux 5.10^{11} n/cm^{2}s on the core vessel;
- six vertical channels outside the aluminium vessel of the reactor core with fast neutron flux 2.10^{12} n/ cm^{2}s, and thermal neutron flux 7.10^{10} n/ cm^{2}s on the core vessel;
- channel for BNCT with epithermal neutron flux $0,9.10^9$ n/ cm^{2}s.

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

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

2. General Modernization & Refurbishment Scope

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

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

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

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

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

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

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

The following other activities concerning reconstruction have been fulfilled:

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

3. Modernization & Refurbishment Scope of Instrumentation and Control System

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

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

3.1. CPSE complex.

The most important parameters in CPSE are the reliability and the fast response, determining the safety and failure-free operation of the reactor, so the CPSE will be built by “independent channels” principle and will satisfy the requirements for fast-execution (minimum delay) of control signal for emergency protection and of permissible probability of non-actuation of control signal for emergency protection by the requests for reactor shutdown. The system will ensure the control of emergency protection and displacement of actuators by means of duplicated channels for monitoring and protection by power, period and process parameters. The logic of CPSE operation is majority voting according to logic 2 out of 3 signals by each of the following parameters:

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

The functions of the system will be:

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

3.2. Actuators of Control and Protection System - ACPS

The actuators are divided according to their function in two types [2]:
-compensative actuators for regulation of reactor power and for reactivity compensation
-safety actuators for shut-down of reactor in case of accident

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

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

The ACPS is composed from five following main parts:

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

3.3. Information-computing system

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

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

3.4. Control rooms

There will be build two new control rooms :

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

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

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

D. MOULIN [1], B. POUCHIN [1], D. PARRAT [2], N. SCHMIDT [1]

[1] *Department of Reactor Studies, DEN, Building 212,*
CEA Cadarache, BP1, F-13108 Saint Paul Lez Durance Cedex
[2] *Department of Fuel Studies, DEN, Building 315,*
CEA Cadarache, BP1, F-13108 Saint Paul Lez Durance Cedex

ABSTRACT

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

1. Introduction

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

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

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

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

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

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

2. Main operating principle

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

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

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

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

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

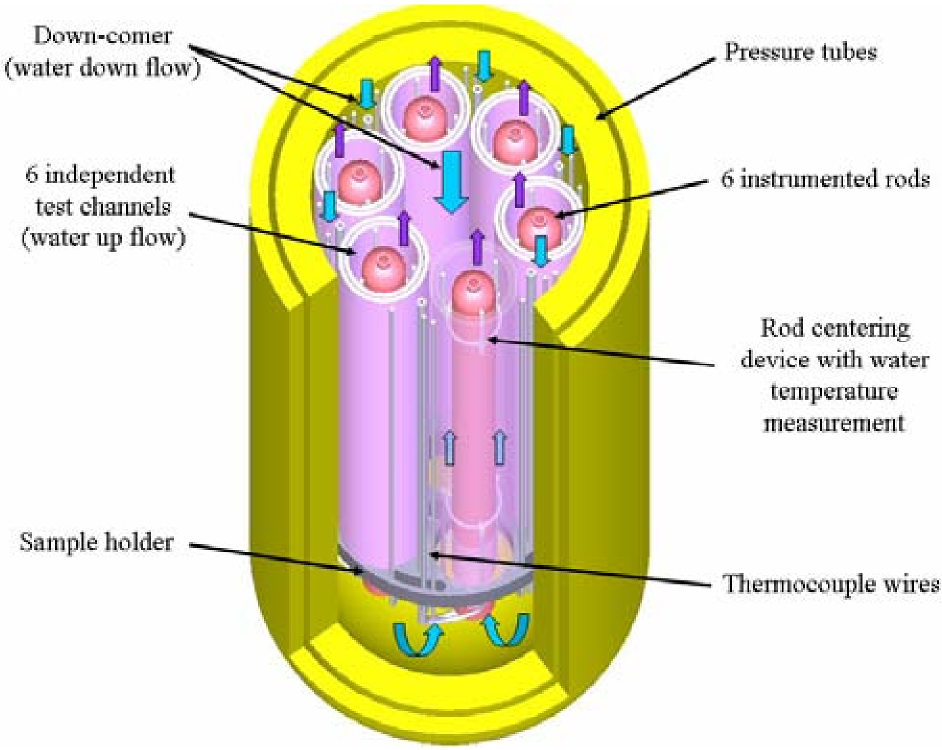

Figure 1 : Radial cut of the loop's in-pile rig

3. Main in-pile instrumentation

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

The classical Self Powered Neutron Detectors (SPND) could be placed in the fissile zone close to each fuel rod to measure neutron flux, and neutronic modelling allows LHGR calculation of each fuel rod. However, this process is not very accurate, especially when irradiation history has to be taken into account with increasing burn-up. Moreover those detectors made of materials with large thermal neutron cross section may induce local shadow effects too high for such experiments.

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

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

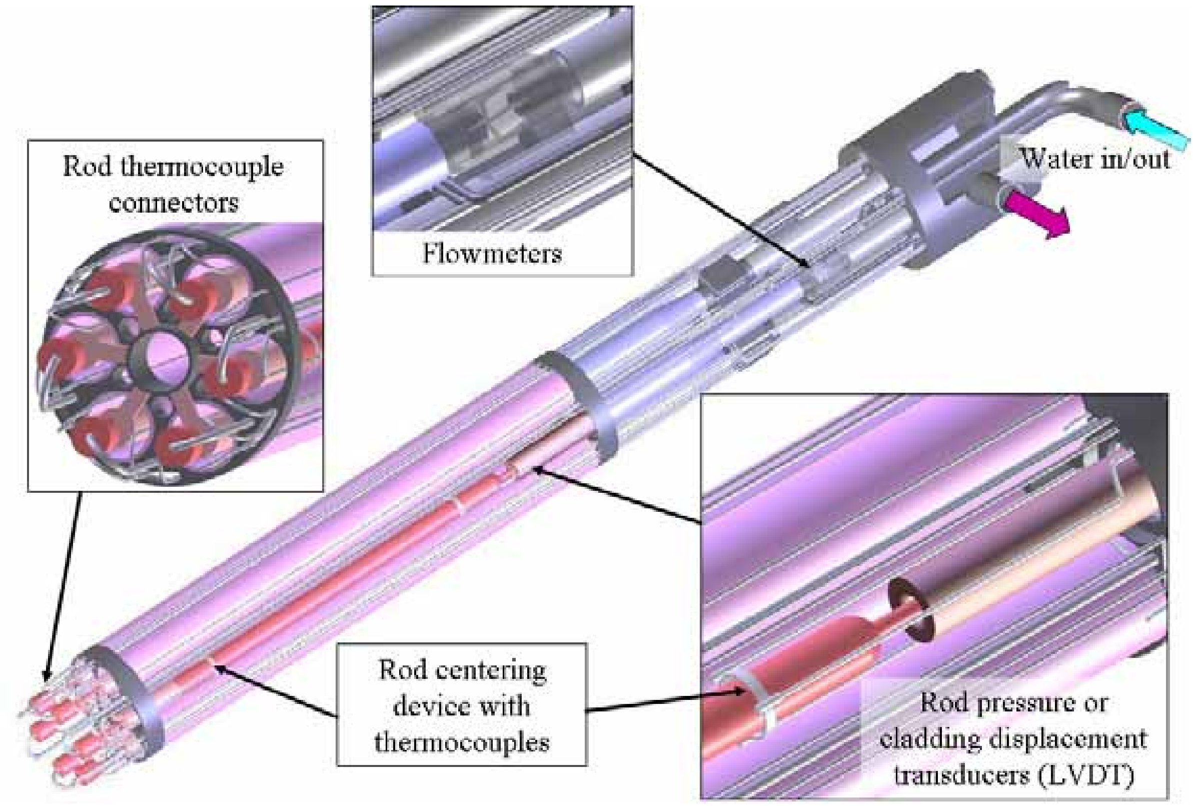

Figure 2 : Schematic view of the sample holder with the in-pile instrumentation

4. Thermal and mechanical studies

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

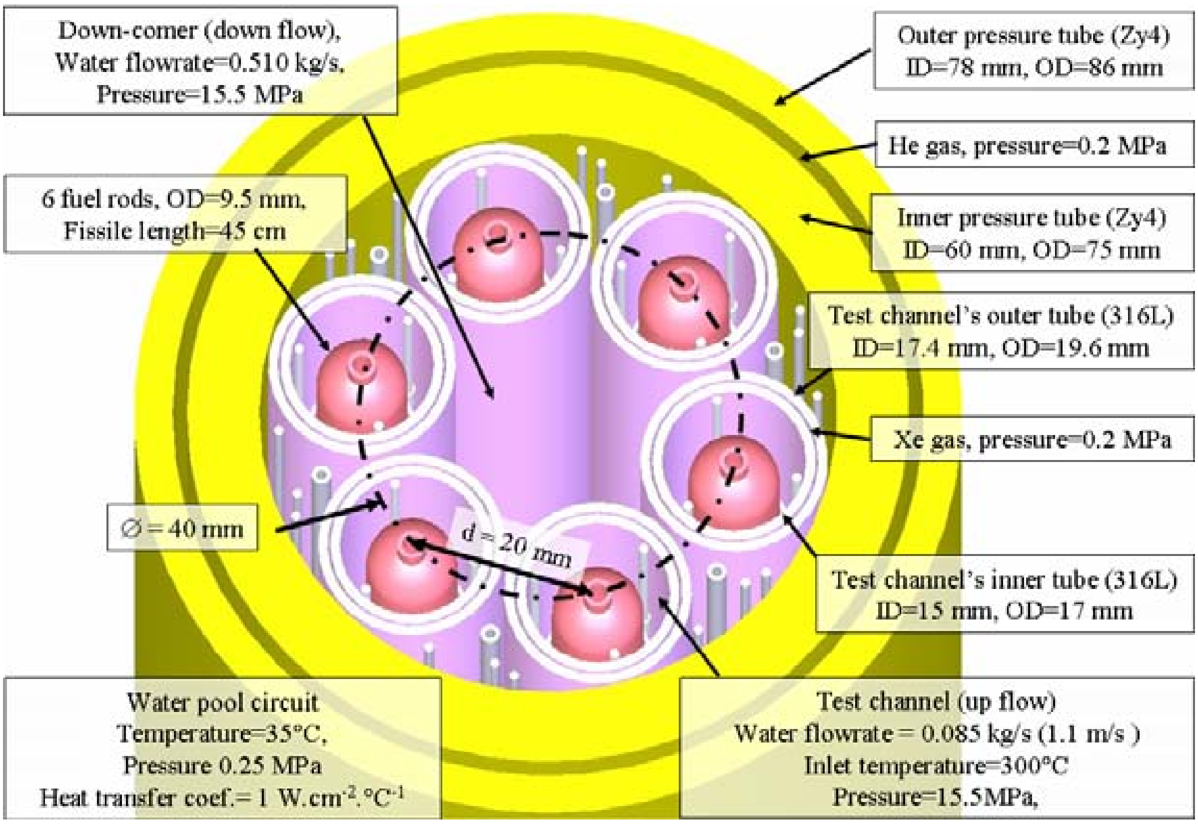

Figure 3 : Main characteristics of the loop's in-pile part

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

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

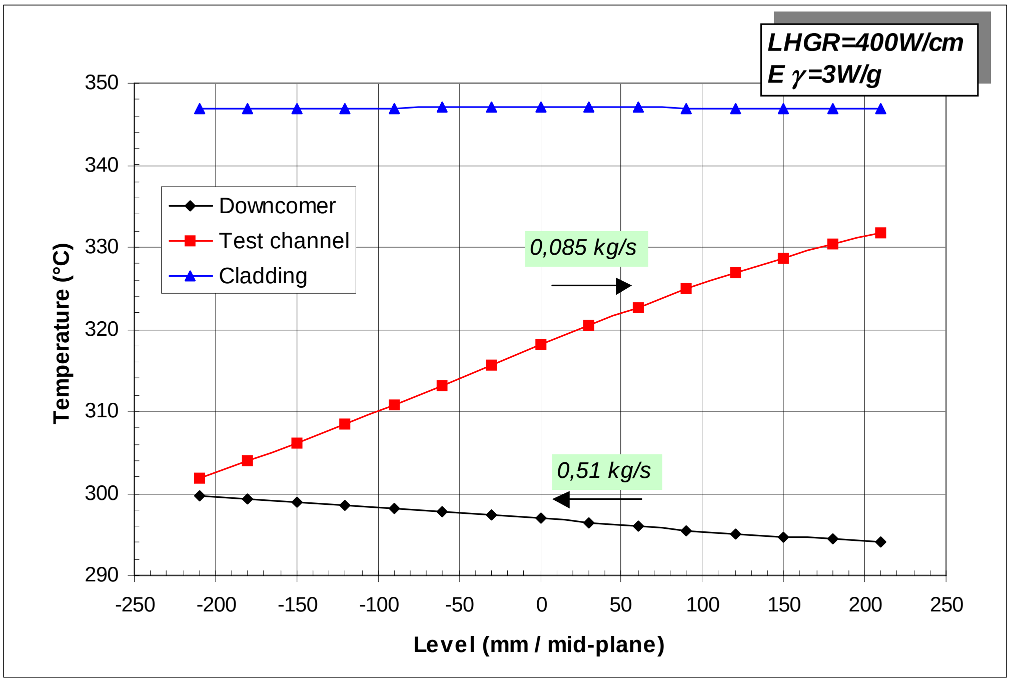

Figure 4 : Axial temperature profiles, LHGR=400 W/cm, Eγ=3 W/g

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

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

A finite element model was developed to compute the mechanical stress within tubes, taking into account the local properties versus temperature of the tubes and pressure values (Table 1).
Then, the calculated stresses were compared to the allowable stresses. This analysis does not highlight any operating problem in nominal conditions: the calculated stresses are smaller than the allowable stresses (Table 2).

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

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

5. Neutronic calculations

Neutronic simulations were carried out with the 3D TRIPOLI4 Monte Carlo code, using 10 million neutron histories. The 100 MW core is described with 34 fresh fuel assemblies of UMo7 20%-enriched 3 × 8 plates. Neither control system nor reloading has been taken into account. The pressurized water loop is located in a water row dedicated to displacement systems in the beryllium reflector. The experimental fuel samples are 2% enriched fresh UO_2 fuel rods. In the configurations presented in this paper the LHGR of experimental rods is given at reactor mid-plane with a statistical accuracy of ± 4% (2σ).
Different studies were performed by varying the distance of the loop's axis from the core rack and the best homogeneity of LHGR is obtained near the core in the peak of thermal neutron flux (Figure 5). To reduce the LHGR values and minimize the difference between rods, a solution consists in keeping the test device near the core rack and adding a Nickel screen of varying thickness. In the two examples presented figures 6 and 7, the gradient is kept lower than 10 %. Indeed this solution needs to be optimized: for instance, a specific Nickel screen could be designed for a given operating point and the displacement system would be kept in order to make small changes around this operating point.

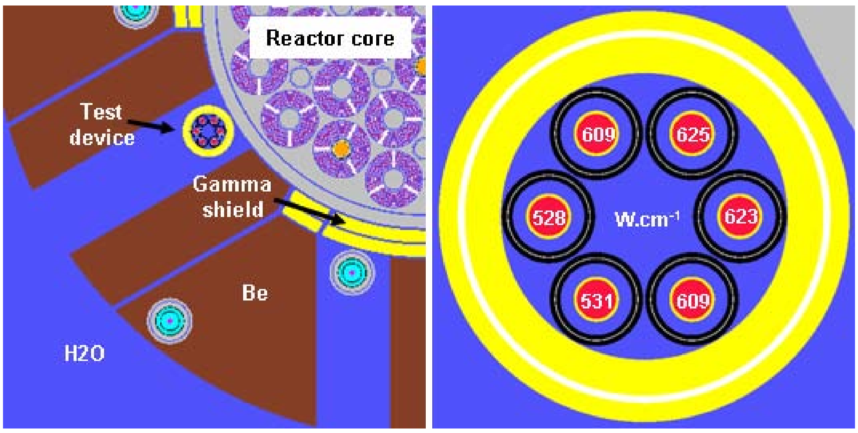

Figure 5 : Test device without specific screen – distance from the core 7 cm.
Fuel rod LHGR (core mid-plane): mean value 588 W.cm^{-1} – grad. 17 %

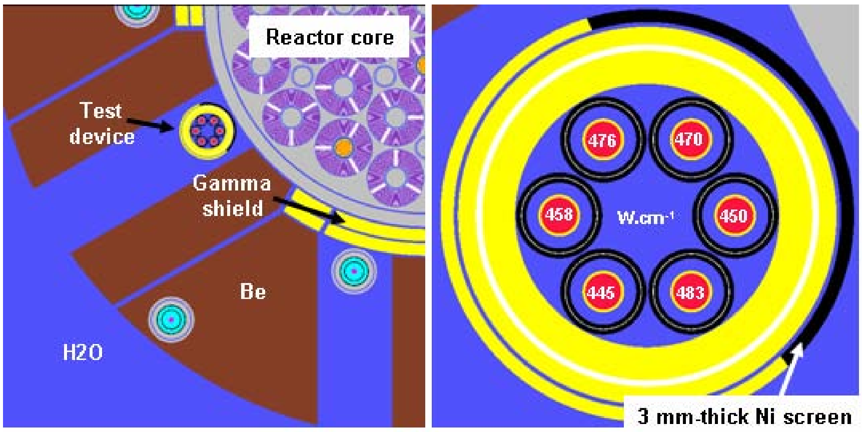

Figure 6 : Test device with a 3 mm-thick Nickel screen – distance from the core 7.4 cm.

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

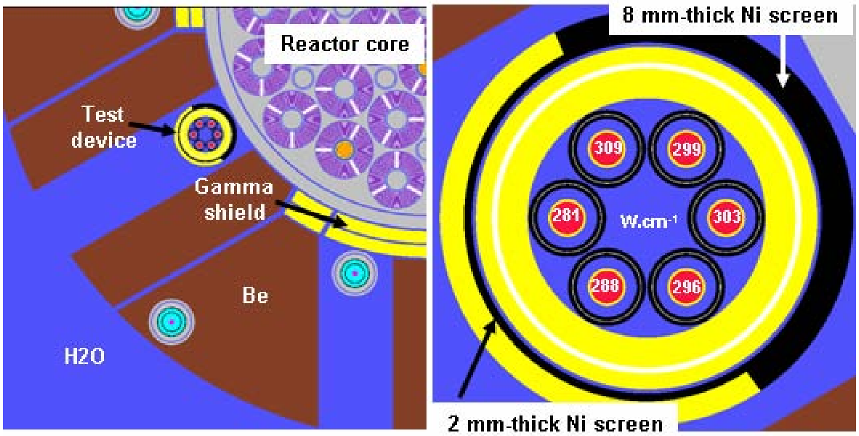

Figure 7 : Test device with an 8 mm-thick, 2 mm-thick Nickel screen – distance from the core 7.9 cm.

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

6. Conclusion

The main characteristics of a water loop dedicated to the irradiation of 6 rods with pressure and temperature conditions identical to PWRs have been presented in the frame of a conceptual design study. One important feature of the concept lies in its specific sample-holder which provides 6 instrumented test channels to improve on-line monitoring of linear heat generation rate of each fuel rod by independent heat balance. Another important characteristic is the possibility to load fuel rods equipped with two sensors at both ends. Based on neutronic calculations, solutions with Nickel screens have been proposed to minimize the LHGR differences between rods.
In a next stage, the main components of the loop should be defined and all its circuit should be taken into account, not only in steady state operating conditions, but also in transient and incidental operating conditions. This proposal of loop has to be discussed with fuel R&D teams as well as end-users to confirm the interest in multi-rod irradiation experiments and thus to carry on the study and the development of such a test device.

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

T. SUZUKI

Department of Research Reactor and Tandem Accelerator, Japan Atomic Energy Agency
Tokai-mura, Ibaraki-ken 319-1195 – Japan

M. UMEDA

Nuclear Safety Research Center, Japan Atomic Energy Agency
Tokai-mura, Ibaraki-ken 319-1195 – Japan

ABSTRACT

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

1. Introduction

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

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

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

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

2. Specifications of NSRR

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

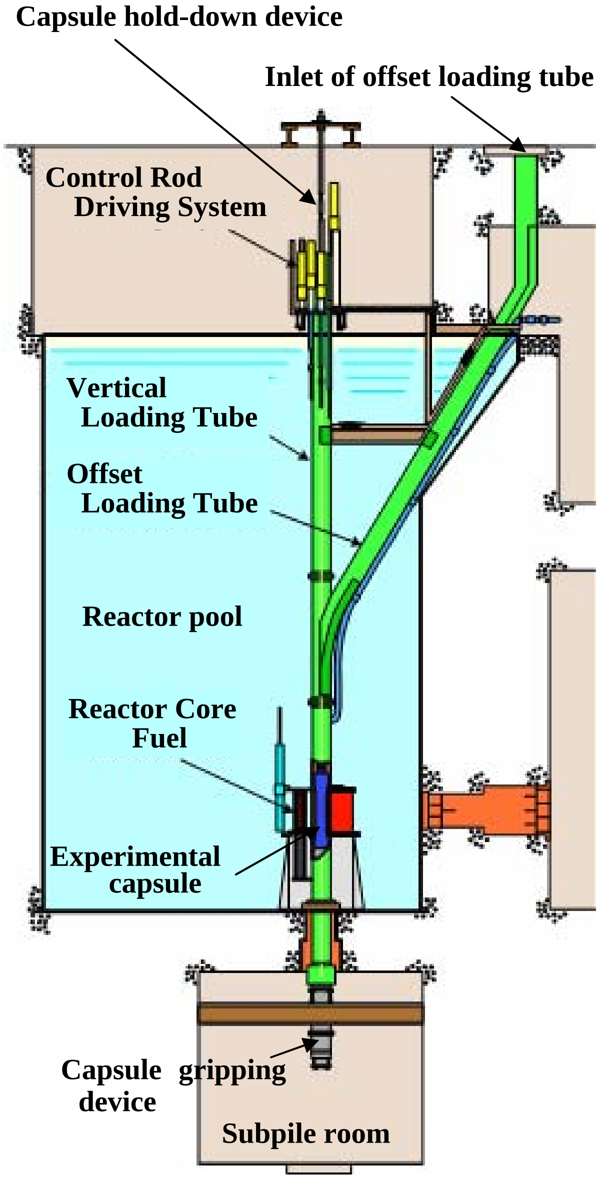

Figure 1: Vertical Cross section of the NSRR

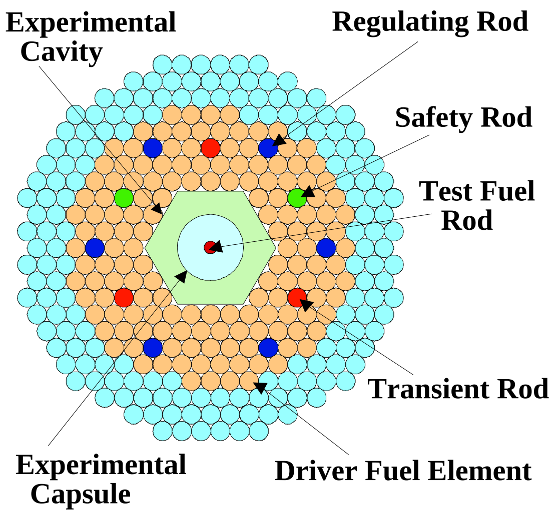

Figure 2: Horizontal Cross section of the NSRR core

Table 1: Specifications of the NSRR

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

The operation for pulse irradiation is made by a quick withdrawal of enriched boron carbide transient rods by pressurized air. Since the hydrogen in the fuel elements is heated instantaneously by rapid power escalation, the NSRR has a large prompt negative reactivity coefficient. This safety features provides a high self-controllability of the NSRR. Figure 3 shows typical transient power history and integrated power history. The maximum reactor power of the NSRR is 23 GW. In this case, the full width at half maximum of the power pulse is 4 ms and the total energy release reaches 130 MJ.

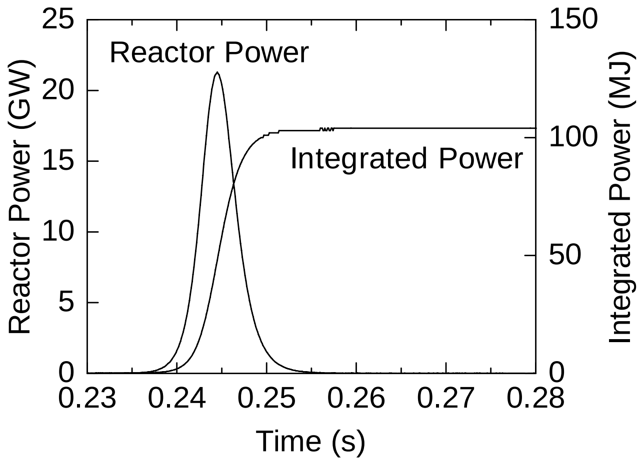

Figure 3: Typical power history of the NSRR

3. Results of NSRR experiments

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

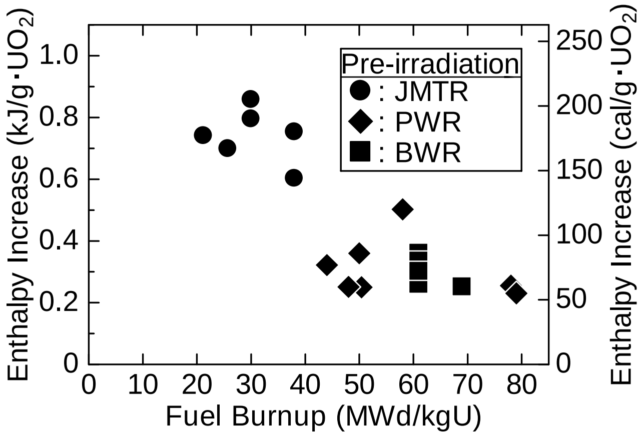

Figure 4: Enthalpy increase at fuel failure as a function of fuel burnup

Experiments with pre-irradiated fuels indicated that fuel failure occurred at low enthalpy before a cladding temperature rise. The failure mode is the pellet/cladding mechanical interaction (PCMI). Figure 4 shows fuel enthalpies at failure in the NSRR experiments as a function of burnup. The fuel enthalpy at failure decreases when burnup becomes high. This is because high burnup makes cladding brittle due to corrosion such as surface oxidation and hydrogen absorption. Figure 5 shows a cross section of PCMI-failed high burnup fuel. The fracture is composed of two parts, one of which is ductile in the inner region of cladding and the other is brittle in the outer region. Some cracks beside the fracture are also observed in the peripheral region. The process of PCMI-failure is thought that incipient cracks through oxide layer and hydride-precipitated layer are formed in the peripheral region, hoop stress concentrates on the tip of the cracks, one of the cracks develops and penetrates the cladding, i.e. fuel failure. The cladding ductility, or the amount of hydride precipitation, is an important factor of the PCMI-failure of high burnup fuels.

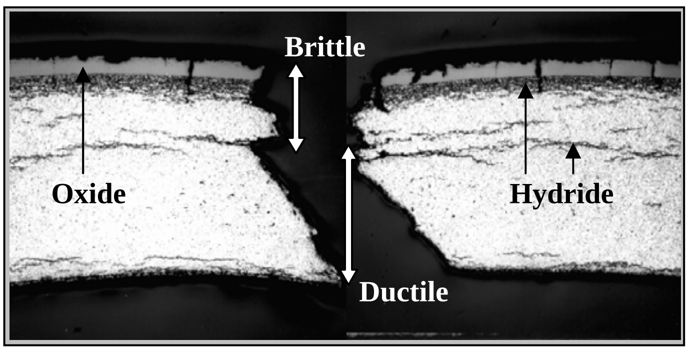

Figure 5: Cross section of PCMI-failed fuel

4. Development of new capsule

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

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

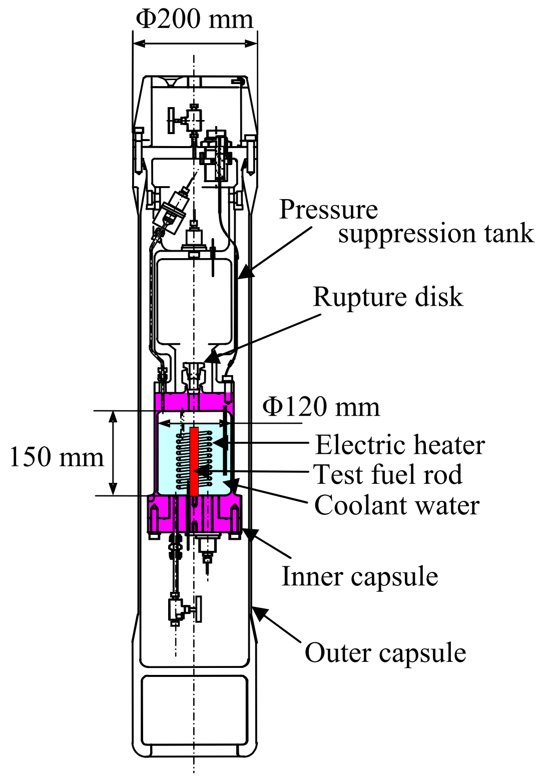

Figure 6: Schematic diagram of new capsule

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

5. Summary

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

Table 2: Specifications of new capsule

Inner capsule	Material	SUS304

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

Table 3: Instrumentations of new capsule

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

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

S. Anim-Sampong, E. Amponsah-Abu, A.G. Ampong, E.H.K. Akaho, B.T. Maakuu, J.K. Gbadago, B.J.B. Nyarko, R.E. Quagraine, I. Ennison, M.A. Addo, H. A. Affum

Ghana Research Reactor-1 Centre
Department of Nuclear Engineering & Materials Science
National Nuclear Research Institute
Ghana Atomic Energy Commission
P.O. Box LG 80, Legon, Accra, Ghana

Abstract

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

1.0. Introduction

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

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

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

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

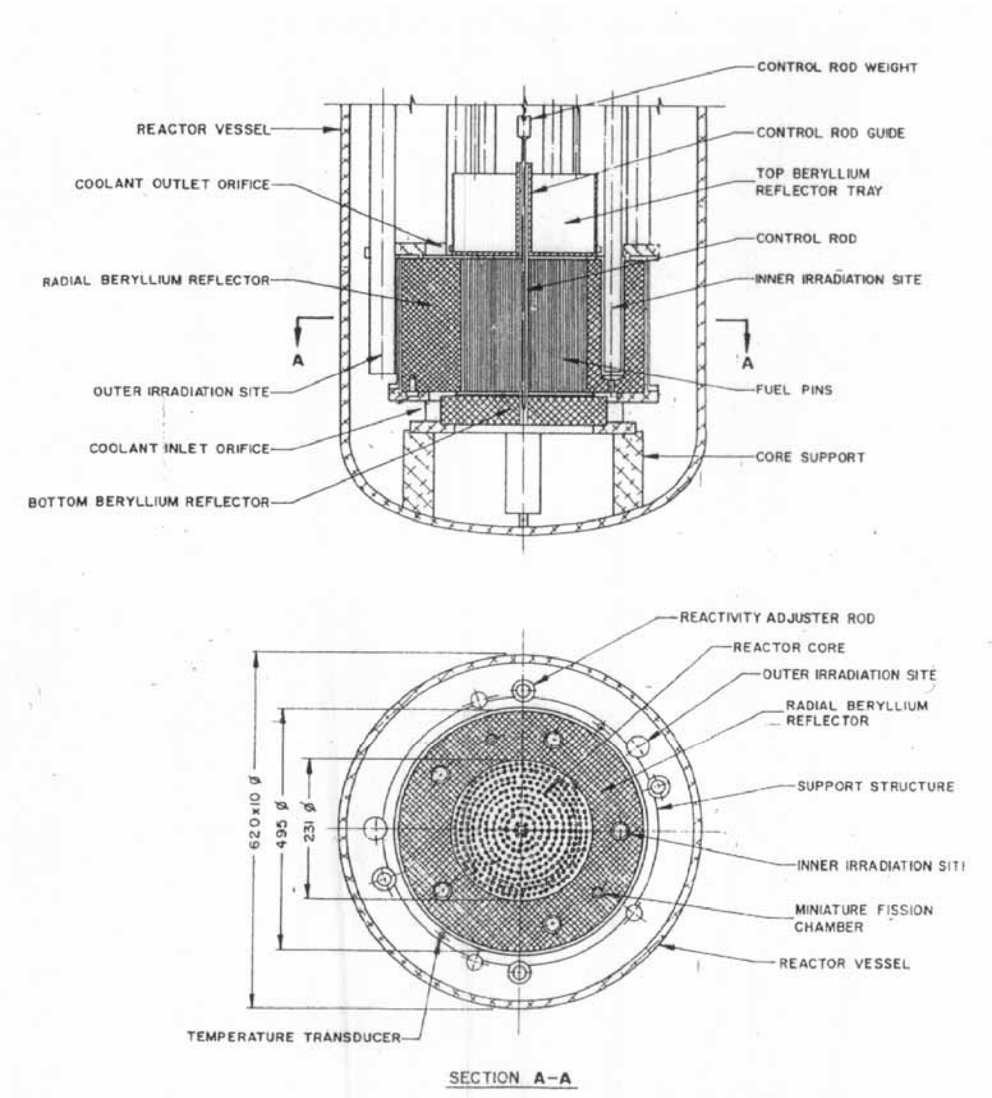
At the request of the Government of Ghana, an IAEA Reserve Fund Project was established to assist in upgrading the GHARR-1 to enhance its utilization for socio-economic development. Extra-budgetary contribution from the Government of China will provide for expenses related to equipment procurement and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

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

2. The GHARR-1 Facility

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

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



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

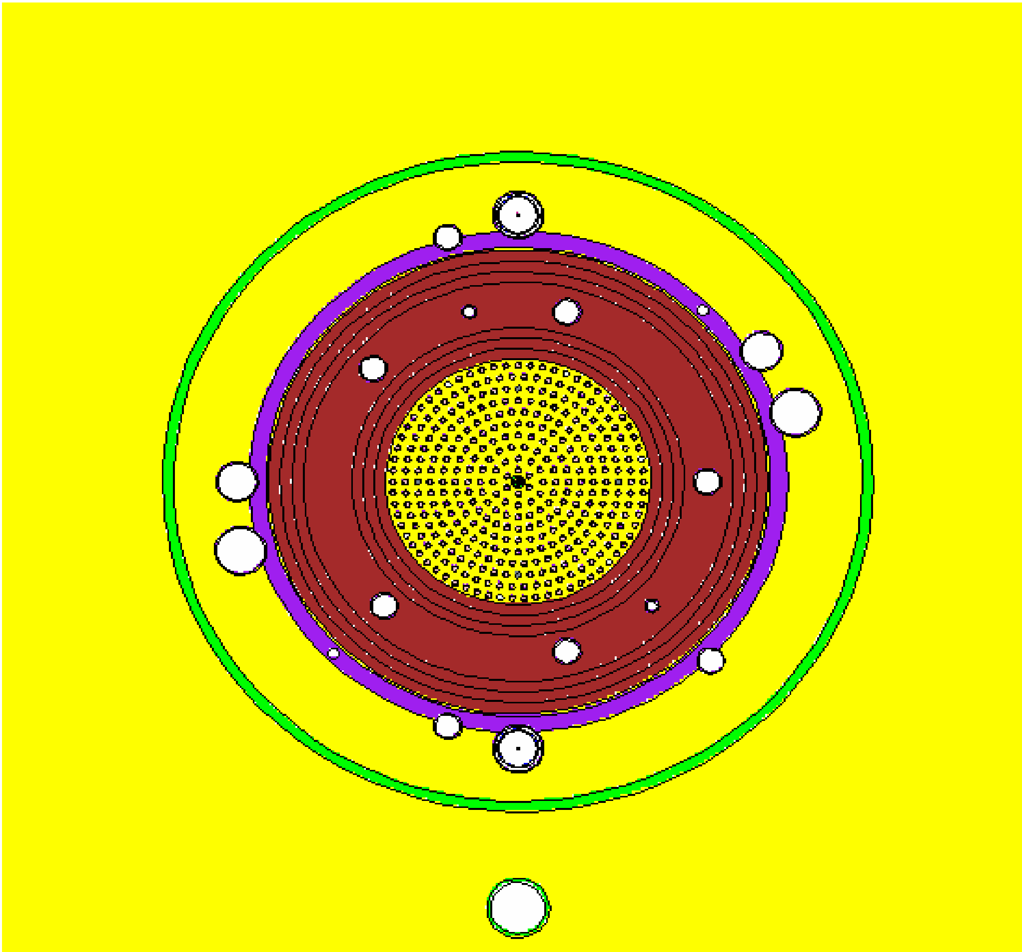


Fig. 2: MCNP5 plot of GHARR-1 core configuration showing fuel region (reactor core), irradiation channels, fission chamber, regulating and slant channels and annular beryllium reflector.

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

3.0. GHARR-1 Reactor Control

The GHARR-1 control system includes control console and computerised control system, neutron detectors, control rod and its drive mechanism and associated electronics. The systems provide information on conditions for reactor operations [10-12]. The principal functions of the control system are; to start up the reactor in a safe and reliable manner, to maintain the reactor at a selected power, and eventually to shut down the reactor. Under-moderation of the core contributes to high negative temperature coefficient (-0.1mk/^{o}C). In addition the excess reactivity of the reactor is limited to $\rho_{ex} \leq ½ \beta_{eff}$ [6]. This ensures that prompt criticality is not possible. Because of the safety provided by the combination of the reactor's limited excess reactivity (4mk under normal conditions) and its self-limiting power excursion response (due to its negative temperature coefficient), it is inconceivable that any situation could arise for reasons of safety that the reactor be quickly shutdown. For this reason, the control system is simplified.

Control is achieved either automatically or manually. The range of the reactor control mechanism for automatic control covers the range $10^8 – 10^{12}$ n/cm^2.s of neutron flux. The working range for NAA is limited to 10^{11}-10^{12} n/cm^2.s while reactor physics experiments are performed in the range of 10^8-10^{12} n/cm^2.s. In particular, since the facility is primarily utilized for NAA, accurate control of the neutron flux is required. To achieve this, a set of control modes are adopted. The MCCLS is particularly designed to ensure a stable flux accuracy of ± 1.0%. Thus, no regulation is made by the system when a flux deviation < ±0.5.0% from the setting value is recorded. Quick responses and accuracy of the control systems ensure a stable flux within ±0.5.0% deviation.

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

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

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

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

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

Further reviews of same were done by the IAEA to ensure that all safety aspects of the project were provided [13-16]. The safety aspects of the project implementation report included key issues such procedures approved by the national regulatory authority for modifications of the research reactor, reasons for the modifications initial safety analysis conducted for the modifications and impact of the project on the safety of the research reactor, the safety analysis, the safety analysis report, conceptual design of the modified system and verification of design criteria the training and qualification of operating and maintenance personnel as reasons for equipment modification and upgrade, technical specifications and functions of components, and quality assurance, installation, commissioning plan for the modification

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

4.0. Advances in Project Implementation

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

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

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

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

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

5.0. Expected Outcomes

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

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

6.0 Conclusion

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

Acknowledgement

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

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RRFM IGORR 2007

LYON, France 11.3. - 15.3. 2007

European Nuclear Society
Rue de la Loi 57
1040 Brussels, Belgium
Telephone +32 2 505 30 54
Fax + 32 2 502 39 02
rrfm2007@euronuclear.org
www.euronuclear.org

Layout and Design: Marion Brünglinghaus, ENS