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Utilization Related Design Features of Research Reactors: A Compendium

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UTILIZATION RELATED DESIGN FEATURES OF RESEARCH REACTORS: A COMPENDIUM

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UTILIZATION RELATED DESIGN FEATURES OF RESEARCH REACTORS: A COMPENDIUM

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2007

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FOREWORD

For more than 50 years, research reactors have played an important role in the development of nuclear science and technology. They have made significant contributions to a large number of disciplines, as well as to the educational and research programmes of about 70 countries worldwide.

In the recent past, however, the utilization patterns of research reactors have changed remarkably. At present, new and upgraded research reactors are either facilities specialized in education, materials research and radioisotope production, or state of the art machines designed and equipped to carry out cutting edge research involving neutrons. A significant number of operating research reactors have become service-for-fee facilities producing radioisotopes, and performing neutron radiography, semiconductor doping and neutron activation analysis for a wide range of users while continuing their traditional role in education and training. At the same time, high quality basic research is the driving force for the few new, state of the art and high performance research reactors.

There are significant utilization issues being faced by the research reactor community, one being the selection, design and operation of various types of devices in research reactors. Early in 2002, in order to facilitate the exchange of ideas, concepts and experience, the IAEA decided to prepare a publication on facilities and associated devices for selected fields of utilization of research reactors, including constraints and restrictions imposed on design and operation.

Pursuing that objective, in December 2002 the IAEA convened a meeting to consider updating the existing documentation on multipurpose research reactors, which was produced in 1988. It was agreed at that meeting that updating the original material, and preserving its organization and contents was not the best response to the actual needs of the research reactor community worldwide. Instead, the recommendation was to prepare a guide on the utilization of research reactors. This publication was designed to focus on the wide variety of facilities that exist at research reactors to carry out research, education and the production of goods and services.

The IAEA convened a technical meeting in July 2003 to prepare this compendium by integrating current information in the design, construction and operation of research reactor facilities and their associated devices. The information presented at that meeting formed the basis of this publication. The meeting also identified gaps in the information presented and suggested approaching other experts in the research reactor community to supply written

contributions on purpose designed features which were not covered in the meeting.

The IAEA wishes to thank the authors of the contributions in this compendium, as well as the expert consultants. In particular, A. Ballagny (France), J. Razvi (United States of America) and H.-J. Roegler (Germany) contributed significantly to the preparation of this publication. Special thanks are due to H.-J. Roegler for compiling and editing the text. The IAEA officer responsible for this publication was P. Adelfang of the Division of Nuclear Fuel Cycle and Waste Management.

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INTRODUCTION

INTRODUCTION

1. BACKGROUND

For more than 50 years, research reactors have played an important role in the development of nuclear science and technology. They have made significant contributions to a large number of disciplines, as well as to the educational and research programmes of about 70 countries worldwide. There are significant utilization issues being faced by the research reactor community, one being selecting, designing and operating various types of devices in research reactors. In order to facilitate the exchange of ideas, concepts and experience, the IAEA prepared this compendium of facilities and associated devices for selected fields of utilization of research reactors, including constraints and restrictions for the design and operation of the research reactors.

In December 2002, the IAEA held a meeting to discuss the revision of an existing IAEA publication on multipurpose research reactors. The participants concluded that rather than updating the previous report, a compendium focusing on a wide variety of existing research reactor facilities would better serve the research reactor community.

To help prepare this compendium, the IAEA held a technical meeting in Vienna from 30 June to 2 July 2003. This was immediately followed by a meeting of experts involved in various new and planned multipurpose reactor projects who analysed the papers presented at the technical meeting and produced the initial draft of the current compendium.

2. OBJECTIVES

Over the past several decades, trends in nuclear research activities have resulted in many research reactors becoming service for fee facilities. In this more competitive environment, many research reactor organizations are struggling to address several threats to their continued operation, one being the selection, design and operation of various utilization related devices. Within this context, the compendium has been developed to:

— Assimilate current information on the design, construction and operation of research reactors, facilities and their associated devices;
— Serve as reference material during reviews to consider new or additional facility features;

— Provide details about the variety of services available at currently operating research reactors.

3. SCOPE

This publication focuses on the wide variety of facilities that exists at research reactors to carry out research, education and the production of goods and services. Within these facilities, the scope has been directed towards systems, structures or components whose principal function is to support research reactor utilization — as opposed to, for example, those systems, structures or components (such as reactor protection systems) whose principal function is safety system actuation, accident mitigation or transient response.

4. STRUCTURE

This compendium includes a collection of contributions from representatives of various research reactor facilities. Authors are listed at the beginning of each paper and their respective institutes can be found in the Contributors to Drafting and Review.

Devices and methods for fuel and material testing at research reactors are addressed in Part 2. These tests are usually related to the operation of existing nuclear power plants, improvement of their performance, studies on new fuel, and development of new power reactors and fuel cycle technologies. To successfully carry out fuel and material irradiation experiments, a wide range of research reactor facilities is needed. This chapter presents a general description of fuel and material testing experimental facilities at research reactors and its associated instrumentation. Special attention is given to operational requirements and reactor interfaces.

Part 3 discusses irradiation facilities for the production of medical and industrial isotopes, illustrating issues related to radioisotope production at research reactors, especially facility designs. The production of radioisotopes that are employed in industry and medicine is one of the main utilization activities in many research reactors. To properly carry out radioisotope production, specific devices have been developed and are being operated. Part 3 provides a description of some of these devices and their interaction with the reactor.

The achievement of low resistivity silicon wafers by neutron irradiation is currently the preferred method of production of such material. Facilities for the irradiation of silicon ingots require high throughput irradiation sites, which

permit frequent handling and have the ability to achieve a uniform flux field. To that end, the successful utilization of research reactors for neutron transmutation doping presents opportunities and challenges to existing reactors, as well as to the design of new facilities. Part 4 presents examples of facility designs enabling the irradiation of silicon ingots of various sizes using a uniform flux.

Boron neutron capture therapy (BNCT) is a promising and possibly curative method of treating currently incurable brain tumours, but at present this procedure is far from perfect. Because of the lack of selectivity of the boron carriers, it appears so far that radiation toxicity limits the radiation dose, so that tumour damage is modest. Current investigations and developments are aimed at targeting the boron carriers to the tumour, in order to limit the damage to the healthy, surrounding tissue.[1] If, in spite of the above constraints and potentially significant funding required, a research reactor facility chooses to become involved in BNCT treatment using neutron beams, the associated risk would be considerable. Aspects of medical therapy facilities at a research reactor include proper neutron beam design for various energies, beam purity to achieve therapy with minimum damage to healthy tissues, and treatment room design. Part 5 addresses design and operation of such facilities, to illustrate the demands and constraints on the utilization of neutron beams at research reactors for medical therapy.

Beam tubes are the interface between the reactor core, as the neutron source, and the neutron based instruments. Issues such as neutron beam design, utilization and upgrading are described in Part 6. This includes the design of beam tube arrangements for new research reactors and modification of existing beam tubes to improve performance or safety. Beam tube arrangements to supply neutron beams for scattering experiments and for neutron radiography are also described.

Secondary neutron sources are designed and installed at research reactors to generate neutrons with energies different from the thermal spectrum of the reactor. Different methods are used to prevent moderation, for example, to use or to extract fast (fission) or epithermal neutrons prior to their slowing down, or to moderate the thermal neutrons in the reflector region with moderators of different temperatures, such as hot graphite, known as hot neutron source (HNS), or cold hydrogen rich materials, such as liquid hydrogen and deuterium, methane, etc., known as cool neutron sources (CoNSs) and cold neutron sources (CNSs), or even cold helium or deuterium ice, the latter

[1] VAN RIJ, C.M., WILHELM, A.J., SAUERWEIN, W.A., VAN LOENEN, A.C., Boron neutron capture therapy for glioblastoma multiforme, Pharm. World Sci. **27** (2005) 92–95.

especially for ultra cold neutron (UCN) sources. Besides the latter methods, nuclear reactions are applied to shift neutron spectra or to generate neutrons of specific energy, such as secondary fission sources. These devices and methods are covered in Part 7.

Nuclear safety tests and dedicated facilities are covered in Part 8, which presents a comprehensive overview of research reactor devices used to perform safety related tests. This part focuses mostly on devices designed to study post-accident consequences. The experimental study of post-accident consequences enables more accurate safety incident analyses and accident scenario predictions.

There is no clear separation between fuel testing and its related facilities as described in Part 2 and the issue of nuclear safety tests in Part 8. In particular, Section 2.5 could have been placed in Part 8 as well. It represents a comprehensive overview of most issues which are relevant when developing and testing fuel for power reactors, but it is restricted to those safety issues and tests which are part of the normal qualification of fuel, whereas Part 8 is directed mostly to post-accident consequences, the knowledge of which is necessary when evaluating incident and accident scenarios in safety analyses. However, it is recommended, when reading Part 8, to include Section 2.5 when considering safety related tests and the facilities necessary to perform such tests.

Part 9 deals with neutron activation analysis, which continues to be a useful technique in the analysis of a wide variety of materials, despite the development and improvements of other analytical techniques, such as inductively coupled plasma mass spectrometry (ICP-MS). Various implementations of the method are presented, with special attention to devices operated in specific research reactors where this technique is applied.

Part 10 illustrates the utilization of research reactors as training and educational tools. Examples of integrating different research reactors with a joint goal of fulfilling the training and educational needs of the next generation of nuclear scientists and engineers are presented.

Part 2

FUEL AND MATERIAL TESTING

2.1. INTRODUCTION TO FUEL AND MATERIAL TESTING

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1. CONTEXT

The first reactor, whose name reflected its primary task, was the Material Testing Reactor at Idaho Falls, United States of America. The Material Testing Reactor was commissioned in the early 1950s. The majority of other material testing reactor type reactors were built between 1960 and 1965. Today, they still represent the primary means to carry out neutron irradiation for studying materials and fuels of existing and future nuclear power plants. These material testing reactors are between 40 and 50 years old and the issue of their replacement is of fundamental importance to the future of nuclear energy.

However, a material testing reactor type research reactor is only one building block when it comes to material and fuel irradiation. The study of the behaviour of fuels and materials under irradiation requires the simultaneous availability of:

— A research reactor as neutron source with a power level of at least 10 MW;
— Cold laboratories for the preparation of specimens and measuring of pre-irradiation characteristic data;
— High activity laboratories for the examination of irradiated specimens and the measurement of characteristic data after irradiation (post-irradiation examination, known as PIE), including internal transport means, such as shielded casks or transfer hot cells;
— Irradiation devices making it possible to adjust the conditions (temperature, pressure, physical and chemical environment, etc.) and to monitor the properties which change during irradiation with suitable instrumentation for in situ measurement;
— Skilled staff able to analyse the results and to develop reliable predictive models for the behaviour of materials and fuel under irradiation;
— Logistical resources for the management of nuclear materials (transport and storage of radioactive materials, etc.).

The resources required are substantial and can only be justified by the:

— Strong interest of the nuclear power plant component manufacturers and/or the utilities operating nuclear power plants, in programmes to establish suitable component lifetimes, extend the lifetime of existing plants, or to increase fuel burnup, etc.;
— Strategic position of some countries concerning the application of nuclear technology as is the case for those which take part in the Generation IV International Forum;
— Constraints originating from licensing requirements, as well as from the increasing need to gain public acceptance of nuclear power.

2. FUTURE REQUIREMENTS

With the large number of irradiation tests already performed at the numerous material test reactors, one may assume that there is sufficient accumulated knowledge worldwide already about the materials and the fuel in use at nuclear power plants. However, in Europe the Future European Union Needs in Material Research Reactors (FEUNMARR) project, carried out in 2002 as part of the European 5th Framework Programme, reviewed all the medium and long term irradiation requirements to establish a European policy in this field. This survey indicated that:

— There will be a need for materials irradiation capabilities as long as nuclear power retains a significant share in the mix of energy production sources;
— For the current generation of nuclear power reactors, material testing reactors represent an invaluable resource to assess new materials, explore modified operating conditions and address design basis safety and unanticipated operating issues that can develop;
— There is a large and growing demand in terms of hard spectra for irradiation originating from new plant ideas, such as accelerator driven systems (ADSs), fast burners for fission products and fusion technology.

The short and medium term tasks resulting from utilities' requirements are:

— An increase in the discharge burnup, which goes hand in hand with an increase of the cumulated fission product inventory and which brings about R&D into future cladding materials, mainly to minimize waterside

corrosion. Regarding the fuel, for two types of events, i.e. the reactivity insertion accident (RIA) and the loss of coolant accident (LOCA), consequences resulting from rod overpressure due to the high inventory of fission gas and from mechanical properties, such as the countervailing of the remaining ductility at higher cladding temperatures plus higher inner pressure, all require experimental proof, as the existing codes do not allow reliable predictions in that extended range of burnup;

— Plant life extension (PLEX), which needs ongoing surveillance programmes in material testing reactors related to the pressure vessel and structural components, and their embrittlement with irradiation by fast neutrons;
— Introduction of MOX fuel for nuclear power plants, which calls for developments in terms of the modified microstructure of that fuel to enable its use under conditions identical to those of uranium dioxide fuels. That development comprises optimization of the MOX fuel and, specifically, its microstructure to the current status regarding UO_2 fuel, in terms of issues such as comparable fission product retention.

To explore materials and components, including the fuel for any new generation of nuclear power plants, the availability of various flux levels and neutron energy spectra provided by material testing reactor type research reactors are necessary to plan and to enable any type of irradiation required for the selection and qualification of the fuels considered by the Generation IV International Forum. In particular, it has to be feasible to create physical and chemical conditions and to achieve temperature levels specific to each reactor design. Furthermore, 'neutron converter' devices have to be provided at the research reactors to adjust locally the neutron spectrum in order to better reproduce the spectrum characterizing each design under study.

3. FUTURE TRENDS

Nuclear power plants are currently making a substantial contribution in terms of the qualification of improved fuel designs and of new fuels for future use in such plants. However, these plants are not suitable for:

— Study of fuel behaviour under incident and accident conditions and sequences;
— Studies which require on-line measurements, as it is not possible to instrument inside the fuels studied which are used in nuclear power plants;

— Timely acceleration of irradiation damage of materials which are under investigation in terms of their lifetime and for the prediction of their behaviour;
— Study of innovative fuels.

In other words, research reactors which are totally dedicated to nuclear safety tests and designed for studying the overall behaviour of nuclear fuel in accident conditions and sequences are highly necessary and require costly programmes, the results of which are difficult to extrapolate to other types of fuels or to other comparable conditions. As a consequence, it seems reasonable to assume that there is a continuous need for material testing reactors capable of:

— Hosting irradiation experiments equipped with sensors for in situ measurement of changes in materials and fuels. This requirement can only be met if a complete range of new sensors can be developed (or can be derived by modifying existing sensors and be qualified) which comply with the requirements of the tests at extended ranges of burnup, pressure, temperature, neutron fluxes, gamma heating, strain, dimensional changes, etc.;
— Providing technically adequate experiments, within the bounds of the facility licence conditions, leading to potential failures and even to part melting of fuel samples. This type of experiment can lead to requirements, such as reinforcing the containment of the facility and putting additional barriers into place for the protection of personnel. Furthermore, the full potential of such tests can only be exploited if laboratories are installed close to the experiment for on-line measurement of the gaseous and solid fission products;
— Broadening the typical material test reactor scope for irradiation (i.e. irradiation in the thermal spectrum) to the field of fast neutrons (hard spectrum), at least locally around the experiments, by converters and/or filters of a nature suitable to adjusting the neutron spectrum to one characterizing the concept of the nuclear power plant technology under study.

Neutron sources are being designed which aim to meet the requirements described, either based on research reactor fission processes or on proton accelerators with suitable targets, e.g. subcritical arrangements of research reactor fuel.

2.2. IRRADIATION OF MATERIALS AND FUELS IN A FRENCH RESEARCH REACTOR

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1. INTRODUCTION

Research reactors are necessary tools for nuclear energy development and evolution. They have paved the way for the use of nuclear power as an energy source and are considered still to be essential for its further development.

Irradiation of materials and components carried out in research reactors must meet the needs of the industry in terms of both the existing fleet of nuclear power plants and their further development, as well as the demands for new power reactor types capable of accomplishing the fundamental goals of sustainable development by reconciling reliability, economy, nuclear safety and ecology. Such experimental work contributes to understanding the local phenomena involved and to confirming the solutions adopted for power reactors. One of the main advantages of research reactors is the possibility to perform experiments up to limits which would not be feasible at a power reactor, as research reactors can be used for instrumented irradiation with an adjustment of the experimental parameters, such as temperature, pressure and neutron flux.

In France, the OSIRIS research reactor located at the Commissariat à l'énergie atomique (CEA) site at Saclay is currently available, while design work is in progress to build the Réacteur Jules Horowitz (RJH) at the Cadarache site which is intended to replace the current research reactor plants.

In order to make use of the results of the irradiation tests, the research reactor must be surrounded by other 'facilities':

— Hot laboratories for post-irradiation examinations (PIE);
— Scientific teams for modelling the phenomena involved;
— Technical teams in charge of developing the test devices and their associated instrumentation.

Together, this forms a scientific experimental platform.

After mentioning the industrial issues and the requirements for R&D in terms of irradiation, Section 2 describes the experimental platform, the irradiation devices for the main applications and the associated instrumentation, followed by an analysis of their interfaces with the research reactor and its operation, as well as of the nuclear safety of experimental devices. As a conclusion, Section 2 considers possible roads to improvement.

2. INDUSTRIAL ISSUES: R&D REQUIREMENTS

The short and medium term industrial issues relate to the operation of existing nuclear power plants and to the improvement of their performance. In the longer term, the main issue is the development of new power reactor technologies.

2.1. Operation of existing nuclear power plants

To manage the operation of nuclear power plants and to improve their competitiveness under optimal conditions, the industry and researchers as partners must:

— Optimize performance and consumption of nuclear power plant fuel;
— Assess a possible lifetime extension of the nuclear power plant;
— Anticipate requirements of the safety authority on the nuclear power plants.

The main corresponding R&D requirements are indicated in Table 1.

2.2. Development of new types of nuclear power plants

Apart from the fundamental goals that the technologies for new nuclear power plants must accomplish, the main issues relate to both the operation of the fuel and the resistance of the structural materials. R&D requirements, therefore, relate to studies of the behaviour of fuels and their qualification, as well as to the qualification of structural materials under the conditions required for future types of nuclear power plants as proposed by the Generation IV International Forum.

New requirements have to be taken into account, such as:

— Screening tests to select new materials to be used in fast reactors which require specific devices to increase the fast flux locally. Typically, an

TABLE 1. R&D REQUIREMENTS FOR EXISTING NUCLEAR POWER PLANTS

Issue	R&D requirements
Optimization of fuel performance	Behaviour of fuel during power transients Behaviour of fuel at high burnup: core temperature, cladding deformation, release of fission gases, densification, corrosion of cladding Qualification and characterization of new fuels Development of cladding alloys optimized for mechanical behaviour and corrosion resistance
Plant life extension	Behaviour of fuel assembly structures and internals: kinetics of corrosion, swelling and creep Behaviour of materials of the pressure vessels and vessel internals in extended operation: changes in mechanical properties of pressure vessel steels, resistance against corrosion
Nuclear safety	Behaviour of fuel in incident situations Behaviour of materials at particular areas (joints, welds, etc.)

accumulated damaging effect of 100 dpa is required to make sure the swelling rate is acceptable;

— Spectrum adjustment to optimize the burnup of minor actinides and to test the in-pile behaviour of related fuels under realistic conditions.

The irradiation programmes associated with these requirements must be designed to allow access to major parameters of the tests, as well as to be representative of both the phenomena and the conditions prevailing in a power reactor. Decoupling of phenomena is, therefore, frequently considered necessary to determine the intrinsic properties of irradiated materials.

3. THE EXPERIMENTAL PLATFORM

To be successfully completed, an irradiation experiment calls for a wide range of skills and facilities: the irradiation phase itself is only one aspect within a broader scientific context. The skills and facilities are provided by the experimental platform, which includes the reactor, the hot laboratories and the modelling tools.

As an example, the OSIRIS research reactor is rated at 70 MW thermal power and is of the open pool type with a core accessible from the top. The core can accommodate up to 16 experimental devices in positions where the fast

neutron flux (n-energy greater than 1 MeV) is between 1 and 2×10^{14} n·cm^{-2}·s^{-1}. Structures placed outside the core can simultaneously receive around 20 devices at the first perimeter where the maximum fast neutron flux is ten times lower than inside the core, and many more in the second and third perimeters.

The OSIRIS research reactor is associated with the ISIS neutron mock-up, which consists of a core of the same geometry but 100 times less powerful. This mock-up is especially intended for acquiring knowledge of reactor and irradiation neutronics. Determination of the neutronics parameters of the irradiation is based on the activation measurement by dosimeters installed inside the devices. Measurement of the damage rate is performed by means of electrical ohmic resistance variation dosimeters. Calorimetric measurements serve to determine the contribution of gamma heating and to measure the power levels in fuels. Some of these measurements are performed on mock-ups of the devices irradiated in the ISIS research reactor.

Elsewhere, the research reactor can accommodate non-destructive examination installations, such as neutron radiography and gamma spectrometry, making it possible to carry out in situ checks between irradiation phases. To recover the irradiated samples, there must be hot cells associated with the reactor. A minimum of two cells is necessary, while the ideal is to have a number of cells each dedicated to one type of the subsequently listed work, thus facilitating their performance:

— Alpha compliant hot cell for recovery and examination of any failed fuel;
— Hot cell for unloading and reloading samples of materials with provision for dimensional measurement, which avoids transfer to a hot laboratory between the different phases of an irradiation experiment;
— Hot cells for dismantling devices, discarding waste and the treatment of samples from isotope irradiation, to avoid disturbing or interrupting the operation of the research reactor.

The hot laboratories contain a variety of scientific equipment necessary for:

— Preparation of samples taken from previously irradiated materials;
— Refabrication of experimental short fuel rods from sections of irradiated nuclear power plant rods;
— Post-irradiation examination of fuel (sampling of gases, metallography, etc.) and materials (mechanical tests, metallography, metrology, etc.).

Moreover, there is a high degree of synergy between the different experimental facilities and the applied simulation tools. Simulation of complex

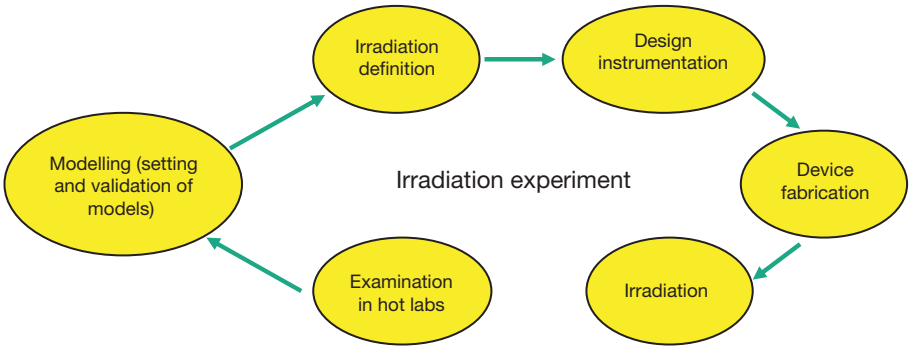

FIG. 1. The experiment loop.

systems indeed uses theoretical and physical models for describing variation over time and multidimensional variations of phenomena for the integration of multiscale and multiparametric approaches. The results of irradiation experiments are used to establish and to validate the models of the simulation tools.

Meanwhile, the multiparametric approach to simulation requires decoupling of some parameters and the design of new irradiation experiments. New types of devices with an increased requirement for specific measurements under flux are generally demanded in these new experiments. Figure 1 shows the experiment loop.

4. IRRADIATION EXPERIMENT INSTALLATION

An irradiation experiment installation typically consists of:

— An in-pool part installed in the reactor core or nearby, depending on the neutron spectrum and flux level required. Samples to be irradiated are placed there;
— An out-of-pool part which combines the installations for treatment of the fluids and all the control and instrumentation, with the data acquisition and data processing of the signals of the measurements performed in the in-pool part;
— Links between these two parts, to transfer the fluids and the measurement signals.

The experimental devices can be divided into two categories:

— Capsules in which the heat transfer medium is static, of which there are different subtypes:
  • Passive capsules in which the neutron flux and the temperature are measured for determination of the fluence at the location of the material and in which the temperature is controlled only;
  • Capsules in which, in addition to the features of the previous type, the samples are subjected to stresses and strains during irradiation;
  • Metrology capsules to measure dimension changes of samples in situ.
— Loops replicating the thermal hydraulics, the neutronics and the chemical conditions encountered in nuclear power plants.

For the in-pool part, the geometry of the devices is fixed so as to optimally exploit the capacity of the reactor. As a general rule, to achieve high neutron flux levels at the samples, the device must have a diameter that is as small as possible and must be surrounded with high density fuel. However, the device is generally longer than the useful zone subjected to neutron flux. This enables sensitive equipment, such as connectors and sensors, which are placed in casings, to be located in the low irradiation zones.

Examples of irradiation installations at the OSIRIS research reactor are described below.

4.1. Analytical irradiation of fuel

The examinations carried out after irradiation of the fuel rods in a nuclear power plant give an overview of the behaviour of the fuel. Analytical irradiation in a research reactor makes it possible to single out the effects by separating the desired phenomena or the parameters, while measuring quantified data resulting from the variation of the different variables, or by subjecting the fuel to non-nominal irradiation conditions.

4.1.1. Devices for studying the behaviour of new fuels

There are devices designed for the irradiation of fuel within a programme of developing new fuels with microstructures and compositions that are optimized to limit the release of fission gases. Such a device is placed at the periphery of the reactor core and is cooled by forced convection directly with reactor pool water. Its design enables the device to:

— Achieve, as a result of the small dimensions of the fuel pellets, high burnups in reasonable time periods under thermal conditions that are close to those in power reactors;
— Adjust the power level by moving the device inwards/outwards perpendicularly to the reactor core;
— Be loaded or removed during operation of the reactor.

The irradiation temperature is supervised by means of thermocouples located in the centres of the fuel rods and the temperature is adjusted by varying the power level, i.e. moving the device in the gradient of the flux outside the core.

Each fuel rod is enclosed in a container around its cladding which acts as a barrier against the pool water. Each such container is equipped with a thermocouple and is filled with helium, and provides the means for sampling between periods of irradiation. The gas pressure inside the container is permanently monitored by a pressure sensor.

Each container can be connected to a removable auxiliary monitoring circuit, which can be used to pressurize and depressurize the container and to check the pressure sensors, as well as the gas sampling systems.

The arrangement consisting of the fuel rod and the container constitutes one experiment load. One device can accommodate six experiment loads arranged in a cylinder. The device can be rotated to gain uniform fluence for each fuel rod in the device. If required, the experiment loads can be changed between reactor cycles. The cylinder of the device with its six experiment loads is placed in a sample holder. A shroud around the sample holder ensures both cooling of the experiment loads by channelling the cooling water and support for the neutron detectors (collectrons) for permanent measurement of the neutron flux.

The entire device is placed on a mechanical displacement system for adjusting the power level by moving it inwards/outwards in the flux gradient of the out-of-core region (Fig. 2).

4.1.2. Pressurized water loops

Pressurized water loops are designed for the irradiation of fuel under thermal hydraulic and chemical conditions which are representative for the environment of fuel in a PWR. There are two devices:

— The first one is especially suitable for power ramp tests. The experiment load consists of a short fuel rod that is either new or refabricated in the hot laboratories from a fuel rod irradiated in a PWR;

FIG. 2. Displacement system for adjusting power levels at OSIRIS.

FIG. 3. Test channel of a pressurized water loop used at OSIRIS.

— The second loop is more specifically devoted to qualification experiments which may include power cycles alternating between two power levels. The loop can receive one to four fuel rods, as well as fuel plate(s) or absorber elements.

The rod geometry is similar for both types of loops. Figure 3 shows the test channel of such a loop.

The design of these loops and their positioning around the core enables their loading and discharge while the reactor is operating. The devices are placed on mobile supports, which are moved inwards/outwards perpendicularly

to the core border to either adjust the power level of the fuel rod under irradiation or perform power ramping or power cycling at varying rates by automatic control coupled to the neutron flux. There are also other methods available for varying the power of the fuel rod under irradiation without changing the power level of the core of a research reactor, for example, by using neutron absorbing gases as a screen around the fuel rod and by varying the pressure of that screening gas (e.g. devices applying ^{3}He as gas at the R2 research reactor at Studsvik, Sweden, and performing the screening with BF_3 at the HFR research reactor at Petten, Netherlands). Moreover, it is possible to use solid fast moving screens for changing the power level at the fuel rod under irradiation.

The samples are cooled by a forced convection water flow induced by an injector system with a nozzle, amplifying the flow generated by the pumping set of the loop supply system located outside the pool. This design makes it possible to reduce the size of the connecting pipes and thus provide flexibility to allow movement of the device inside the pool. Moreover, an electric heater at the top of the loop and a heat exchanger in the reactor pool enable overall control of the coolant temperature.

The heat generated inside the experiment load during its irradiation is measured in real time by means of the heat balance applying the signals of sensors for flow, pressure and temperature, as well as by a neutron balance from the collectrons. Any fuel cladding failure is picked up by a gamma activity detector in the cooling water of the devices associated with a delayed neutron detector. The chemistry of the water used in the loop is established by a conditioning facility. During operation of the device, it is monitored by an analysis of cooling water samples taken at different positions of the system. The loop intended for power ramp tests is additionally equipped with a linear variable differential transformer (LVDT) type sensor in order to continuously monitor the elongation of the rod during power ramping.

4.2. Irradiation of materials

At OSIRIS, two basic rigs are operational and can accommodate samples of material for irradiation either inside the core or at its periphery. In-core irradiation is generally representative for the neutron flux conditions experienced by core internals and fuel assembly structures in nuclear power plants, while irradiation at the periphery makes it possible to accelerate the irradiation damage by enhancing the fluence applied to the pressure vessel steel in a given irradiation time, while remaining representative of other conditions at the pressure vessel of nuclear power plants.

These rigs basically consist of two concentric tubes enclosing a layer of gas, making it possible to:

— Increase/decrease the temperature inside the inner tube;
— Adjust the inner temperature as a function of the position and the loading of samples by modifying the nature of the gas enclosed;
— Check for leaks from either tube containing the sample, if necessary for safety reasons.

A cutaway view of the capsule of such a rig for irradiation at the core periphery is given in Fig. 4.

The inner tube is provided with electrical heating elements sunk in sprayed on metal particles. The spray layer thickness is calibrated by subsequent machining. The heating system allows an accurate control of the temperature of the samples during irradiation by instantaneously reacting to the variation in nuclear heating by the radiation from the core, for example, varied locally by the movement of the control rods. Another method applied elsewhere for the adjustment of the temperature of the sample is to vertically move the capsule within the axial flux profile, as applied at the HFR at Petten, Netherlands.

These material irradiation capsules are connected to a glovebox for the management of the media and to a control rack for control of the electrical heating elements. Sample holders specific to each type of experiment are inserted inside the capsules. Such experiments are irradiation:

— With the accumulation of fluence under a controlled temperature;
— Of samples under stress;
— Of samples under stress with dimensional monitoring during irradiation.

The sample holders containing the specimens (as an example, see Fig. 5) are immersed in a static and inert heat transfer fluid to keep the temperature uniform by ensuring heat exchange. For irradiation at temperatures below 450°C, alloys of sodium and potassium are used.

These sample holders require the availability of suitable sensors for performing the required measurements in the special environment of the reactor consisting of neutron bombardment, gamma heating, vibrations due to the coolant flow and very limited space.

FIG. 4. Cutaway view of a capsule for irradiation at the core periphery.

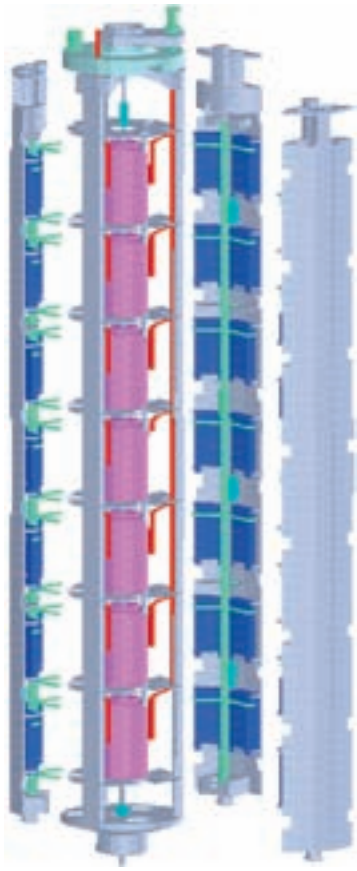

FIG. 5. Sample holder of an in-core irradiation capsule.

5. INSTRUMENTATION OF DEVICES

To meet requirements, the irradiation devices are equipped with systems for the reliable measurement of the irradiation parameters:

— Neutron flux and neutron spectra;
— Temperature, specimen dimensions;
— Applied force;
— Status of physical and chemical environment;
— Fission gas pressure.

5.1. Measurement of neutrons

Two types of equipment are applied for the measurement of neutrons:

— Continuous measurement giving access to instantaneous values;
— Global measurement for integrated dose.

Continuous measurement uses collectrons or fission chambers. As a general rule, this equipment measures a flux of a given spectrum or a reaction rate of an isotope relative to a specific energy, but it is also extremely sensitive to neutron spectrum and to gamma heating. Thus, interpretation of the results by calibration and correction is necessary.

The global measurement intends to determine the fluence. It is carried out using dose integrators consisting of a calibrated wire of pure metal with a specific capture profile for a flux of neutrons of a given energy. The conversion from count rate to fluence is obtained by neutron mock-ups of the devices and determination of the transfer coefficients.

5.2. Measurement of temperatures

The measurement of temperatures is continuously performed by thermocouples calibrated for the measurement range applied and installed in the device. As for neutronic measurements, some types of thermocouples are sensitive to that nuclear environment and can be used only in some cases, for example, for short transients, or must be corrected permanently, which remains difficult to achieve with the required accuracy.

The thermocouples used are mainly of the K type and of tungsten–rhenium for high temperatures. For some applications in safety transient experiments, ultrasonic thermocouples may also be used despite their bulk.

5.3. Measurement of specimen dimensions

The need to know the material or fuel behaviour during irradiation necessitates in situ measurements of sample geometry, such as:

— Elongation of a fuel rod during a power transient, in order to detect the precise moment of any cladding failure;
— Elongation of specimens subjected to stresses;
— Variation of the diameter of a sample subjected to stresses, for example, by monitoring primary creep strain in fuel rod cladding subjected to high internal pressure simultaneously monitored.

To perform those measurements, industrial laboratory techniques are used which, however, need to be adapted to the subsequent three constraints in terms of irradiation:

— Resistance to gamma and neutron radiation;
— Reduced available space due to the geometry of the devices;
— Location of the data acquisition electronics some tens of metres from the measurement location.

At present, the technologies most commonly used are:

— LVDT type sensors, which are widely used in research reactors, such as the HBWR at Halden and the OSIRIS at Saclay for sensing cladding elongation;
— Strain gauges making measurements outside the intense flux area of strain by means of remote mechanical sensing of the region of interest. Figure 6 shows the system for measuring the diameter of cladding subjected to primary creep.

These systems have the disadvantage of being in contact with the sample to be measured and thus generate thermal and mechanical disturbances locally. Studies are currently being conducted to develop a measuring system in which there is no contact with the sample.

FIG. 6. Creep measurement system (detail).

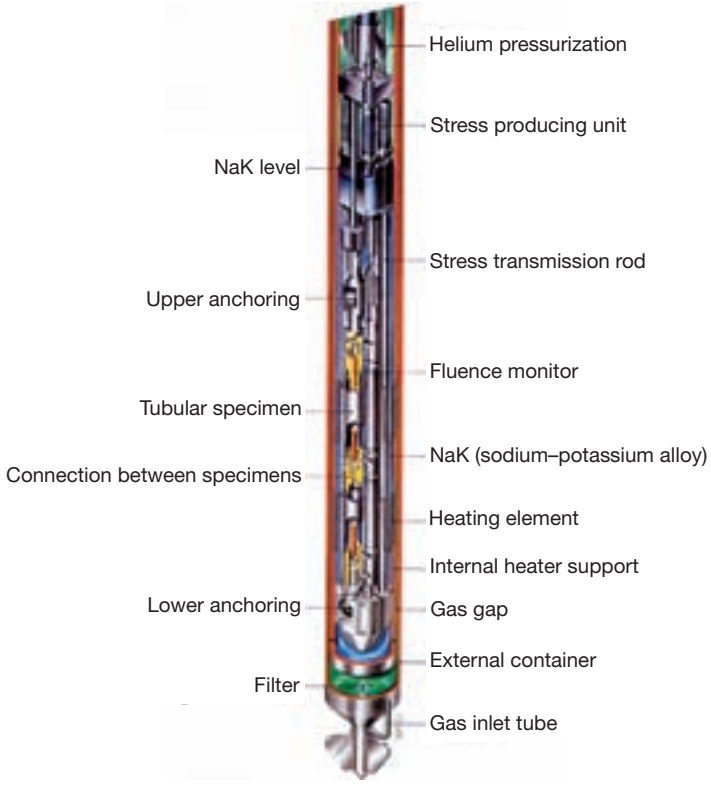

FIG. 7. Measurement of creep and growth under irradiation.

5.4. Measurement of force

Stress measurements at samples are performed in a number of ways by:

— Devices installed before irradiation, for example, by inner core swelling under irradiation, applying U-bend systems, etc.;
— Conventional mechanical systems which are generally difficult to install due to their size, if significant forces are to be generated. They can be used more easily in irradiation devices located at the periphery of the reactor where larger dimensions can be tolerated;
— Bellow systems using pressure controlled from outside the device or by the pressure of the coolant; Fig. 7 shows the system used for applying tension to specimens.

Forces can be measured continuously by means of force sensors or strain gauges. As the properties of such measuring systems vary under irradiation, it is necessary to regularly carry out calibration of those systems.

5.5. Measurements associated with the quality of the physical and chemical environment

Measurements are related to the monitoring of the medium in which the specimens are irradiated. For irradiation under the conditions prevailing in a PWR, a chemical treatment of water is necessary before introducing it into the device. This conditioning guarantees the quality of the water used and ensures the correct concentrations of additives, such as boron, lithium hydroxide and hydrogen. During irradiation, samples are taken to supervise the stability of the water chemistry. Control and measurements of the amounts of admixtures are carried out in a laboratory that can accept contaminated and possibly radioactive probes. In the event of concentration changes, adjustments are made by the injection of pretreated water.

For irradiation in a gaseous environment, measurements of the purity of the gas can be carried out by spectrometry or by the use of an ionization chamber enabling, for example, measurement of the tritium content or to detect any fission product in the medium of interest.

5.6. Measurement of fission gas pressure

Understanding the fission gas pressure variation in the fuel rod plenum is important for analysing the overall behaviour of a fuel rod. This measurement can be performed by a backpressure sensor installed directly at the end of the fuel rod. This has the advantage of being reliable and accurate, as the sensor is not subject to any variation over time despite the hostile environment at the fuel rod itself.

Figure 8 is a basic diagram of an operating internal pressure sensor.

6. PRINCIPLES OF DESIGN OF AN IRRADIATION DEVICE

First and foremost, an irradiation device must meet the customer's performance requirements and enable irradiation that can provide results of the requested scientific quality. An installation must also meet requirements concerning operation, interfacing with the reactor and nuclear safety consistent with the design basis.

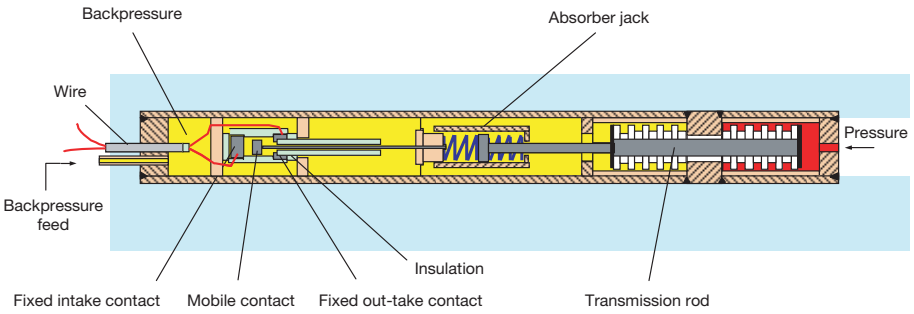

FIG. 8. Sensor for measuring internal pressure in a fuel rod.

When a device is designed, it must always be verified that the requirements resulting from the following are addressed:

- An accurate expression of the scientific requirement, which necessitates full understanding between the designer of the device and the scientist intending to use the irradiation results (as a device can be used for a number of successive irradiation experiments, it is appropriate to establish its limiting properties from the start: nature of specimens, temperatures, environment, etc.);
- A preliminary safety analysis, carried out at the design stage, making it possible to integrate nuclear safety requirements into the design and to operate the device consistent with the requirements of the safety regulations and with the rules for the radiological protection of workers;
- Proper considerations of experience feedback from previously used devices, in order to prevent errors and to improve the quality and reliability of the device under design;
- Establishing a thermal and/or thermal hydraulic design basis using computer codes approved by the safety authority and integrating safety related construction requirements, such as barriers and shielding. In most cases, the thermal requirements for maintaining the required temperature conditions of the specimens and the required mechanical strength as imposed by the accepted design codes are not compatible. As a result, the design frequently is an appropriate compromise which reconciles the customer requirements relating to thermal aspects (temperature level, gradients within samples, etc.) with the designer's demands in terms of safety, i.e. meeting the mechanical strength requirements;
- A functional analysis of the device and its components to ensure maintainability at the design stage, i.e. to guarantee that maintenance will be technically possible;

— A comprehensive design review incorporating all those concerned: scientists, system designers, device designers, reactor operators, radiation protection experts, shielding specialists, etc. This design review shall verify that the device as designed is consistent with the predefined requirements.

During the construction phase of the device, particular care must be paid throughout to any modification of, or deviation from, the design; this must be dealt with globally and not locally only, as a modification or deviation can have indirect consequences on other systems or devices than the device or part of it which is directly concerned. During this phase, the main difficulties are:

— Selecting suppliers for specific work which is highly technical;
— Managing the interfaces between the different suppliers involved in manufacturing the device.

7. INTERFACES WITH THE REACTOR

As a result of its design and operation, the reactor imposes constraints on the devices, which in turn complicate the operation of the research reactor. The reactor hosts a large number of experimental devices which all have their own characteristics. The experimental devices must be designed to coexist with others. Coexistence requires tolerating disturbances, such as local reduction of flux, shadow effects and modification of cooling conditions. In addition, the operation of experimental devices must not induce any notable operational change and/or any reduction of the safety of the research reactor.

The irradiation devices must comply with some rules specific to the operation of the research reactor, enabling the latter to operate properly with its maximum availability. Maintenance of devices often becomes complicated by the simultaneous presence of a number of them in a narrow space (see Fig. 9).

Therefore the research reactor must possess dedicated means for facilitating maintenance, such as:

— Removable access platforms enabling operators to approach working areas, for example, to handle connecting cables;
— Accessible areas where connecting boxes are concentrated;
— Penetrations of housings or enclosures inside the reactor pool made accessible by straightforward action by the operating staff, for example,

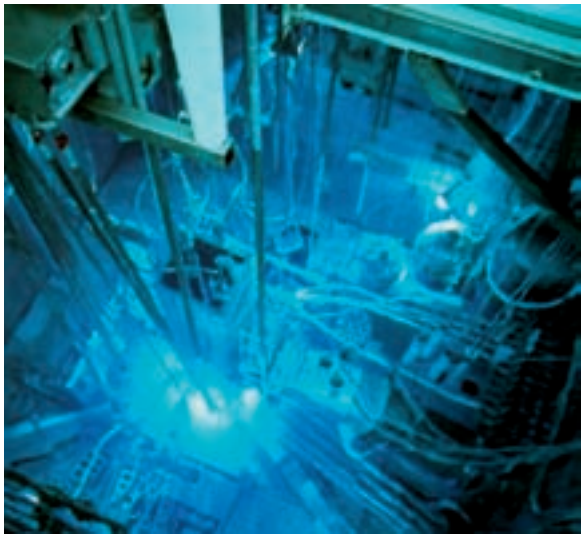

FIG. 9. Irradiation devices at OSIRIS.

the lowest accessible level inside the pool for access to pool wall penetrations (which are submerged for reasons of radiological protection when the reactor is in operation);

— Transferring devices in to and out of the pool to non-destructive testing stations and/or to hot cells (management of devices having been subjected to irradiation must be facilitated and be as flexible as possible, in particular, it must be possible to install or remove devices located at the periphery of the reactor while the reactor is operating);
— Dedicated areas for the preparation of devices inside the research reactor prior to irradiation.

In addition, to ensure the proper operation of both the reactor and the irradiation devices, provision for action on both shall be made. For example, designs must shut down the reactor when a parameter exceeds a value prejudicial to the quality of the results of an irradiation sequence. This type of action is performed using 'two out of three' logic, in which two out of the three independent measurements made under the same conditions must indicate that a limit has been exceeded.

To test all the electrical and mechanical connections, checks are always carried out for the experiments prior to starting the reactor.

8. NUCLEAR SAFETY OF IRRADIATION DEVICES

The implementation of a new irradiation device is subject to an authorization by the regulatory body (in France, the Ministry of Industry). The safety analysis is reviewed by the Institut de radioprotection et de sûreté nucléaire (IRSN) as technical adviser of the regulatory body.

Therefore, the mechanical design of the devices has to be based on rules, such as the French code RCC-M or others, such as ASME, and operational conditions must be calculated by using qualified neutronics and thermal hydraulics codes.

The presence of an experimental device inside the core or at its periphery can give rise to incidents and/or accidents, as well as to risks with consequences for the reactor and the device. Therefore, it must be made sure that, at any time, it is possible to transfer the reactor and the device into a safe state. In addition, the consequences of the risks, as well as of incidents or accidents originating from the device, must be restricted to the reactor hall.

To achieve these objectives, safety related functions must be provided (supervision of containment, of energy release, of radioactivity release, etc.) for the experimental device under consideration, for the reactor and for the other experimental devices. The continuous availability of the safety functions is ensured by:

— Identifying the dangerous products, such as irradiated structures of the devices and irradiated fluids placed in the neutron flux zone, specimens, pressurization gases of the device container, and sodium–potassium alloy contained in the device;
— Identifying the potential risks associated with dangerous products, such as nuclear risks, radiological risks, conventional risks (e.g. thermal, mechanical, chemical, electrical);
— Designing barriers to confine the dangerous substances as well as their associated auxiliary systems. A number of means acting on the device as well as on the reactor can be used in a suitable order, depending on the variation of a parameter. At a first threshold, for example, the electric power supply of the heaters of the device can be switched off and, if the endangering parameter still is not within its limits, at a second threshold the reactor may be automatically shut down. Each trigger applies the ‘two out of three’ logic;
— Covering the operational status (normal, incident and accident) within the design basis of the device by investigating such states or events which are not covered by the design basis but the consequences of which have been studied;

— Making a nuclear safety analysis comprising verification of the proper operation (acceptable consequences in terms of design limits or overall safety of the device and the reactor) in certain predetermined states or events;
— Indicating preventive measures.

Finally, consideration will be given to the end of the device's service life. The conditions of its dismantling are evaluated, the resulting waste is identified and the means of its treatment discussed.

9. CONCLUSION

Irradiation of materials and reactor fuel in a research reactor is generally a complex action requiring the availability of special skills and substantial resources.

Despite the large number of devices for irradiation applied since research reactors were introduced, there is scope for the improvement of irradiation devices and the manner in which they are used. Areas of improvement of devices are mainly revealed by experience feedback. Changes in requirements are demanding better monitoring of the parameters, as close as possible to the specimens.

Experience feedback shows that it is useful to have generic devices and to standardize the equipment. Those features make it easier to manage a multitude of devices and to carry out the necessary maintenance. Moreover, such features limit the number of applications for authorization approval prior to proceeding with irradiation.

Finally, there is a growing requirement for irradiation devices with increasingly sophisticated instrumentation able to achieve the results as obtained in cold laboratories. Even though this may be somewhat utopian in view of the constraints associated with the environment of the devices, research reactor staff must continuously improve the technology in place and investigate new technologies that are available but not yet used in this field.

2.3. IRRADIATION FACILITIES AT THE ROMANIAN 14 MW TRIGA TYPE RESEARCH REACTOR

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1. INTRODUCTION

During almost 25 years of operation, the multipurpose 14 MW Triga type research reactor at the Institute for Nuclear Research (INR-RR), located at Pitesti, Romania, succeeded in achieving the goals for which the reactor itself and its irradiation facilities had been built. Utilization of the reactor focused mainly on the irradiation/testing of CANDU type nuclear fuel within the framework of a research programmme related to fuel technology development and modelling of fuel behaviour under normal and anticipated transient conditions. To accomplish these goals, the basic irradiation facilities were designed, built and operated in accordance with the principles of nuclear installation safety, licensed by the Romanian regulatory authority.

This paper presents the irradiation facilities with their utilization and production features and the performance of in-pile capsules and loops, with minimum constraints concerning safe installation and operation. Their utilization programme is illustrated in some detail. The research reactor itself is briefly described as the plant housing the irradiation facilities.

2. TRIGA 14 MW MATERIALS TESTING REACTOR AND TRIGA ACPR

The INR-RR represents a uniquely designed neutron source. Its large pool accommodates two reactor cores (see Fig. 1) plus a large set of experimental facilities. Its cores use light water as coolant and moderator. The maximum operating power is 14 MW, but all structures and systems are designed for a power of up to 28 MW [1].

The grid plate housing the steady state reactor (SSR) core comprises positions for up to 35 fuel assemblies (FA), 8 control rods and 5 in-core positions for experimental facilities plus several beryllium reflector elements (Fig. 2 shows 15 LEU FAs, 20 HEU FAs and 6 control rods).

The INR-RR's fuel assemblies consist of 25 fuel pins in a 5 × 5 array. The fuel is a uranium zirconium hydride mixture with erbium as a burnable poison.

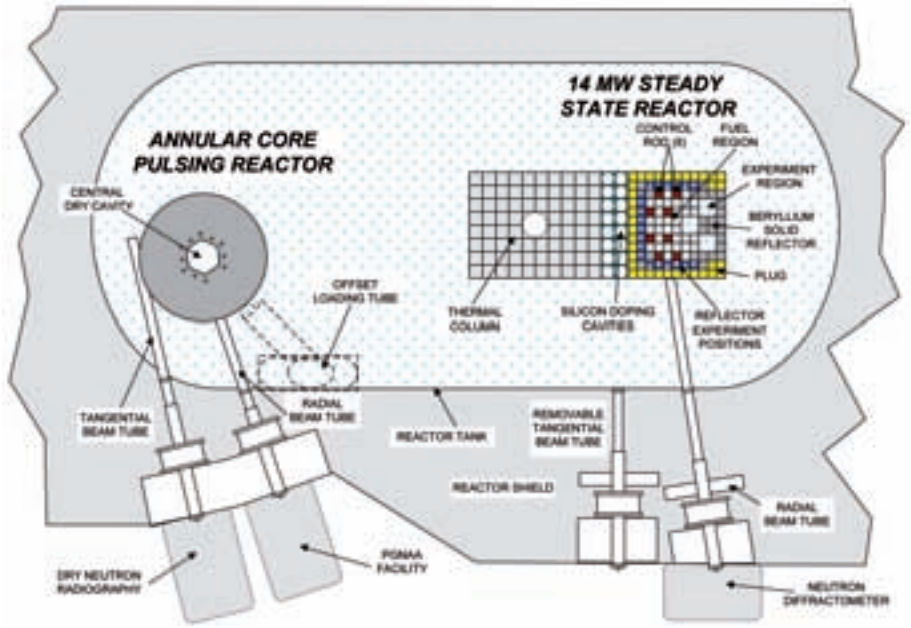

FIG. 1. In-pool arrangement of steady state reactor and pulsing reactor.

This fuel design allows an exceptionally high burnup and thus generates a reduced amount of spent fuel.

The structure of the core is simple and flexible, enabling, by relocation of fuel assemblies and reflector elements, an easy access and installation of irradiation devices and in-pool facilities plus their correlation to desired levels

FIG. 2. INR-RR core arrangement (horizontal cross-section).

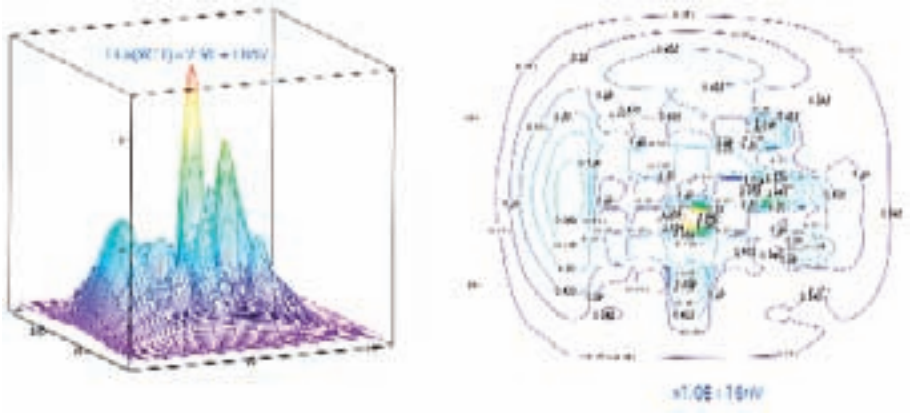

FIG. 3. Thermal flux distribution at 10 MW.

of flux and fluence. A typical thermal flux shaping across the core (at 10 MW) is presented in Fig. 3.

The main parameters of the steady state 14 MW reactor and of the annular core pulse reactor (ACPR) are compiled in Tables 1 and 2, respectively. The operating schedule allows cycles ranging between 26 and 60 days.

3. SERVICES FROM THE REACTOR STAFF

The data acquisition system of the reactor and of the experiments has been upgraded every 5–6 years. At present, the reactor and the irradiation devices have a distributed architecture which enables fast data acquisition for transient experiments in the ACPR for 6–10 channels at 104 measurements per second. Each significant experimental facility is equipped with its own data acquisition system (Fig. 4) connected within a network comprising the computers of other experimental facilities and of the reactor.

The following services are available at the INR-RR to provide and to sustain nuclear fuel and material testing for the steady state and for transient/ accident conditions:

— Technical information on the SSR and the ACPR capabilities, design, operation, access and limitations;
— Advice on the design of irradiation devices and other facilities using the free neutrons, including assistance forexperimental facility related licensing, in-reactor installation and operational surveillance, in order to meet licensing commitments involving technical andsafety requirements;

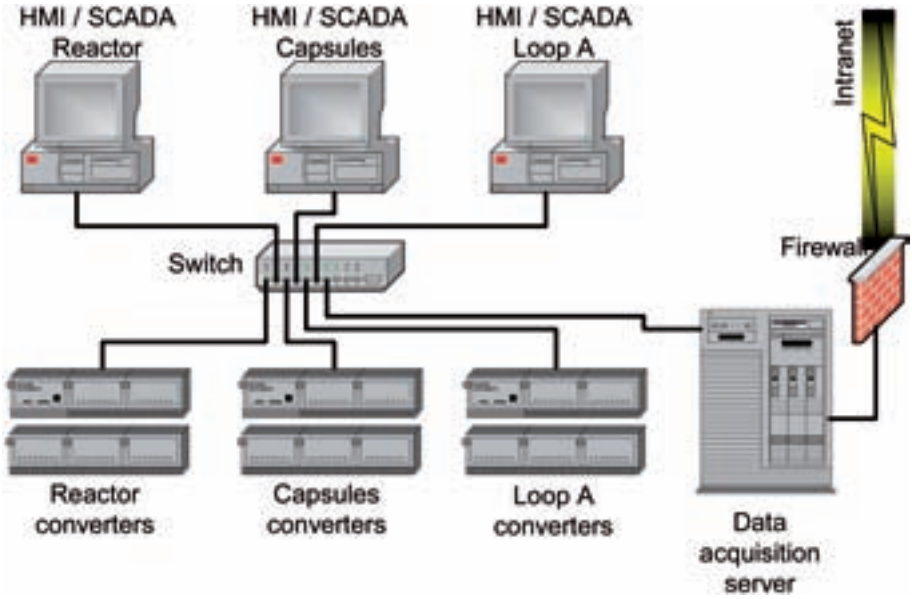

FIG. 4. Data acquisition system for reactor and experimental facilities.

— Connection/disconnection of experimental facilities to and from reactor systems, and handling of irradiation devices, including their transfer to hot cells for inspections and (interim) examinations of irradiated samples;
— Data acquisition, data recording, data processing, as well as irradiation reports by the reactor operation staff and by the facility experts.

For special experiments or facilities, the reactor staff may contribute by calculating the neutronics and/or the thermal hydraulics, by designing the in-core parts of the facility, by writing the preliminary safety analysis report for the facility or the special experiment performed using that facility, by preparing the final safety analysis report to allow the installation and operation of the facility and/or the performance of the special experiment. Such support may comprise:

— Neutron metrology, determination of thermal, epithermal and fast flux and fluence;
— Determination of integral neutron spectrum;
— Determination of local and axial flux shaping;
— Continuous neutron flux and local power monitoring with self-powered in-core neutron detectors.

TABLE 1. STEADY STATE REACTOR — MAIN CHARACTERISTICS

Nominal thermal power		14 MW
Fuel type		Mixed, HEU and LEU
Number/bundle dimensions		35/89 mm square cross-section
Number of fuel pins/assembly		25
Maximum thermal/fast flux		2.9×10^{14} n·cm^{-2}·s^{-1} / 3.6×10^{13} n·cm^{-2}·s^{-1}
Reflector	Material	Beryllium, square cross-section identical with fuel assembly
	Numbers	20 with 33 mm diameter central hole 24 without hole
Control rods	Material	Hot pressed compacts of boron carbide (B_4C) with aluminium clad, natural ^{10}B content; square cross-section identical with fuel assembly dimensions
	Number	8
	Control rod drives	Electrical motor with rack and pinion; electromagnetic connection of drive and absorber
Pool		4.5 m × 9 m, 10 m deep, aluminium tank, light water filled (300 m^3)

TABLE 2. ANNULAR CORE PULSING REACTOR — MAIN CHARACTERISTICS

Steady state power		Up to 500 kW; wide steady state power range (starting from mW); suitable for tests and calibration measurements at very low power levels
Pulse operation		Suitable for testing of experimental fuel in accident conditions
Maximum pulse power		20 000 MW
Minimum period/pulse width		1.2 ms/4.6 ms (width at half maximum)
Fuel	Type	12 wt% U–ZrH fuel
	Enrichment	20 wt% ^{235}U
	Cladding material	Stainless steel with dimples
	Diameter	35.6 mm
	Number of rods	146 + 6 fuelled followers
	Number/type	6/fuelled follower type
	Poison material	B_4C from natural boron

TABLE 2. ANNULAR CORE PULSING REACTOR — MAIN CHARACTERISTICS (cont.)

Control rods	Rod drive	Rack and pinion; electromagnetic connection of drive and absorber
Transient rods	Number/type	2 fast transient rods and 1 adjustable transient rod/ air follower
	Poison material	B_4C from 92% enriched boron
	Rod drive	Fast pneumatic adjustable rack and pinion drive
Maximum integrated thermal flux per pulse		1.5×10^{17} nvt

4. NUCLEAR FUEL AND MATERIAL TESTING FACILITIES AT THE REACTOR

Assimilation of a nuclear fuel technology for power reactors supposes a phase of technology development and qualification following technical specifications of the product established by the design. A special chapter of those technical specifications concerns the behaviour of nuclear fuel in normal operating conditions, during anticipated transients and, sometimes, under accident conditions. In order to demonstrate the fuel capability to sustain the conditions mentioned, tests to be performed within an irradiation programme should be established.

By design, the INR-RR is connected to an underwater transfer channel leading to hot cells at the post-irradiation examination laboratory (PIEL), a building adjacent to the reactor. That system allows the transfer of the in-pile part of the irradiation devices for interim examination, the loading with pre-irradiated samples or loading with a fresh set of samples (i.e. experimental fuel rods or materials). This facility increases the experimental capability and reduces the cost of experiments as well as the amount of radwaste by reutilization of devices and parts. The PIEL is equipped with a transfer cell; two working stations; a large examination cell (9 m × 6 m × 6 m) with seven working places; two steel shielded cells for destructive examination, metallography, chromatography and burnup determination; and one lead shielded cell for mechanical testing. All testing equipment is computer aided. A radwaste treatment/conditioning facility operates in the vicinity, at the same site, for the management of radwaste from the reactor, from irradiation devices and from hot cell operation in support of the INR-RR and the PIEL.

Irradiation of nuclear fuel in an irradiation device is an expensive and time consuming part of the INR-RR's broad programmes oriented towards the

TABLE 3. IRRADIATION DEVICES AT THE INR-RR, PITESTI, ROMANIA

Specification	Reactor	Irradiation device
Normal conditions	14 MW SSR	Capsule C1: on-line fission gas sweep analysis Capsule C2: fuel internal clad pressure evolution and central temperature Capsule C7: corrosion under irradiation Capsule C9: load follow-up Loop A: overpower
Transient conditions	14 MW SSR	Capsule C2: ND loop A ramp tests
Accident conditions	ACPR	Capsule C6: (RIA simulation)
Future plans in terms of accident conditions	14 MW SSR	Modified capsule C2: for LOCA type accident, using fresh and irradiated fuel Modified loop A: to simulate LOCA at PWRs
	ACPR	Design a new irradiation device to allow testing of pre-irradiated fuel in RIA conditions
Material irradiation	14 MW SSR	Capsule C5: for structural materials Loop A: pressure tube material

demonstration of a new fuel fabrication technology, increase of fuel performances, plant life extension, and safety directed fuel behaviour under accident conditions.

The irradiation devices (Table 3) operating at the SSR for fuel testing are designed and licensed for CANDU 600 thermal and chemistry parameters, i.e. 110 bar, 310°C, demineralized water with a pH value of 7.5–10.5; O_2 content: max. 100 ppb.

4.1. Loop A

The irradiation facility loop A (Figs 5 and 6) serves for investigating the behaviour of the CANDU type fuel at the INR-RR [2]. Its main features are as follows:

— Total power 100 kW
— Water flow rate at the samples 3–7 m^3/h
— Maximum pressure in the primary circuit 135 bar
— Maximum water temperature 310°C
— Useful internal diameter 54 mm

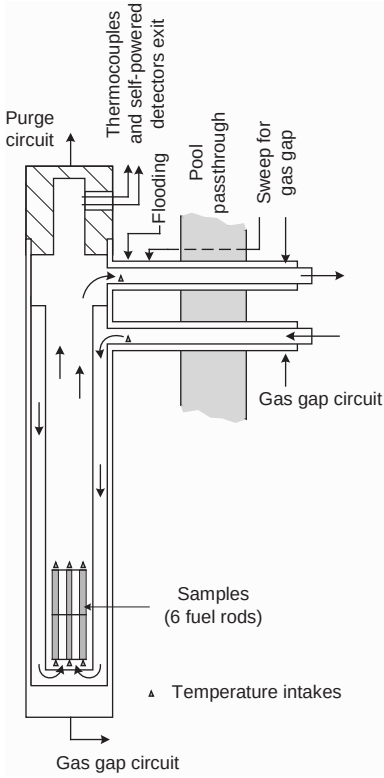

FIG. 5. The in-pile section of loop A.

The in-pile section of the loop consists of two main elements: the 3.80 m long tubular envelope and the sample holder basket fitted at the top with a plug for the passage of instrumentation cables. Except at the top, the in-pile section is double walled and the two tubes, which form the wall, confine a low pressure gas.

The out-of-pile section mainly includes a primary circuit to maintain nominal fuel rod cooling conditions (pressure, temperature, flow rate, water quality), a gas circuit to vary heat transfer, and an effluent circuit to collect drainage from the primary circuit and either send it to the effluent tanks after condensation or process it before evacuation. The out-of-pile section is made of AISI304 stainless steel and is located in a tight cylindrical containment shielded with 150 mm lead walls (35 t) to ensure the protection of its surroundings in case of an experimental clad rupture.

Loop A allows the performance of overpower tests as well as ramp tests, for three or six fuel rods simultaneously (Figs 5 and 6). Table 4 summarizes the history of the experiments performed.

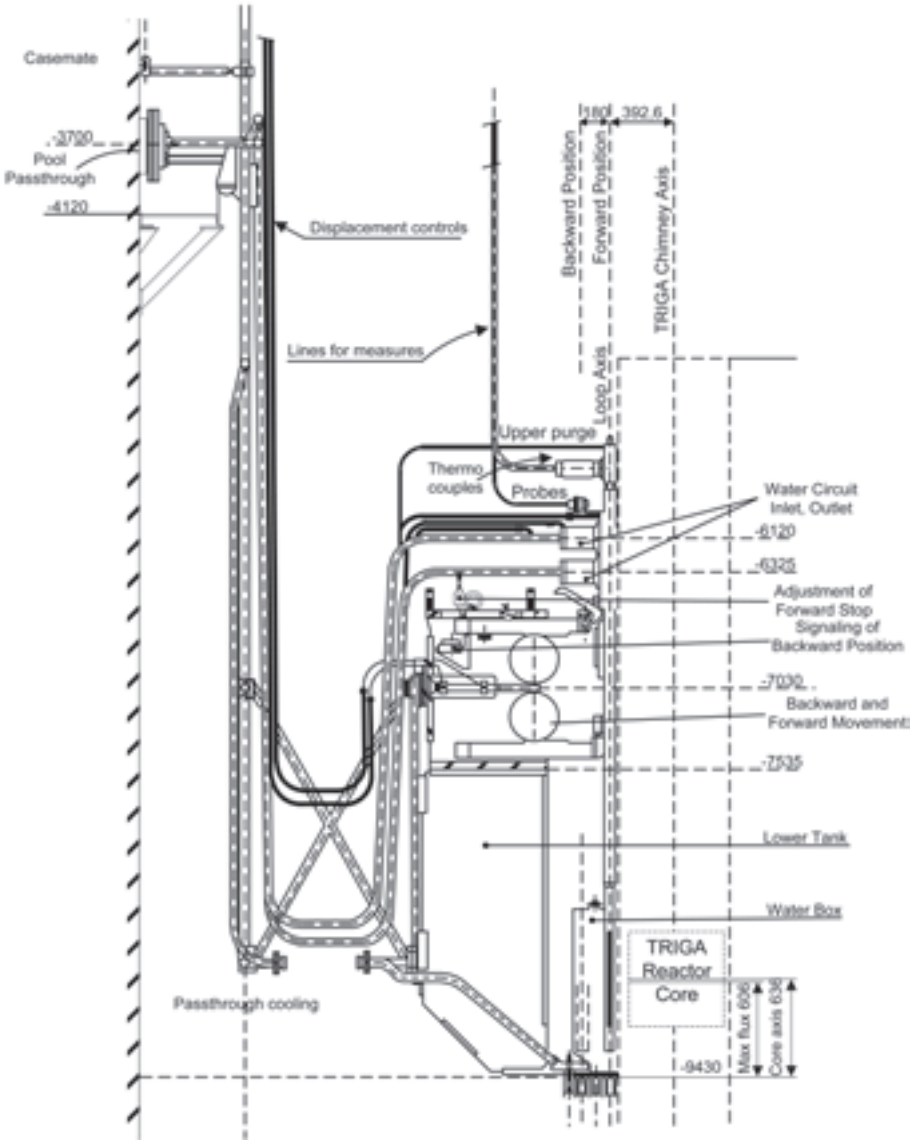

FIG. 6. The in-pile section of Loop A.

The high pressure primary circuit of loop A (Fig. 7) contains the elements of a power reactor primary circuit at reduced scale, allowing the simulation of thermal hydraulic parameters, a specified forced flow across the in-pile section and the water chemistry established through demineralizing bad resins (DEBR), water purification by mixed bed ion exchange filters.

TABLE 4. IRRADIATION TESTS PERFORMED APPLYING LOOP A

Irradiation test	Samples	Irradiation period
Overpower test	3 fuel rods	20.08.83–05.11.83
Overpower test	3 fuel rods	01.08.84–31.05.85
Power ramp test	6 fuel rods	01.11.85–07.06.86
Power ramp multiple test	6 fuel rods	03.12.86–18.09.87
Power ramp multiple test	6 fuel rods	13.10.87–27.07.88
High burnup power ramp multiple test	6 fuel rods	16.02.89–21.09.94
Corrosion test for NPP[a] Zr-2.5% Nb pressure tube samples	48 samples (24 in-flux, 24 out-of-flux)	03.05.96–07.02.99
Mechanical properties test for NPP Zr-2.5% Nb pressure tube samples	24 samples (12 in-flux, 12 out-of-flux)	03.05.96–07.02.99

[a] The Cernavoda nuclear power plant is a CANDU type nuclear power plant.

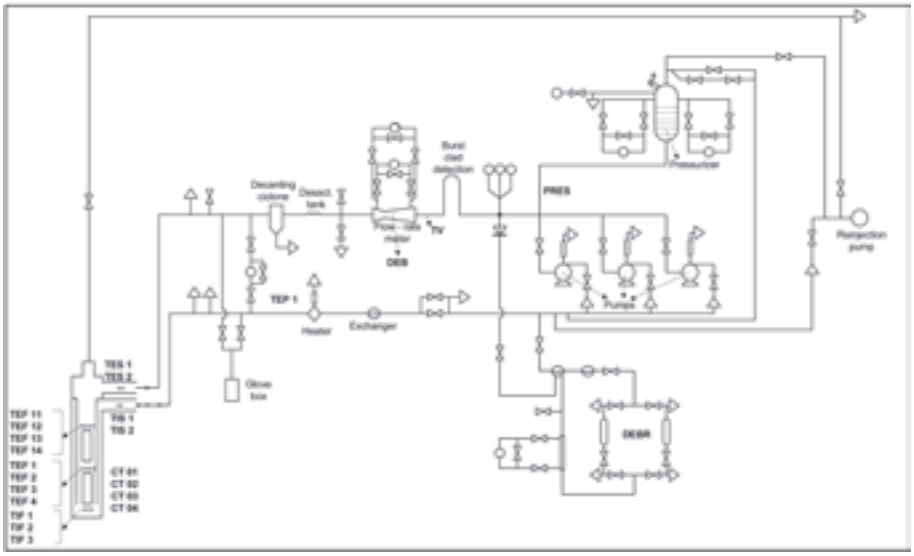

FIG. 7. Loop A flow scheme.

4.2. Capsules C1 and C2

Capsules C1 and C2 are almost similar irradiation devices designed to meet the main parameters, as follows [3]:

— Maximum pressure 150 bar
— Maximum outer fuel temperature 330°C
— Cooling water flow rate 300 L/h
— Purification flow rate 2.5 L/h
— Useful inner diameter 29.5 mm
— Maximum fuel element diameter 15 mm

Capsules C1 and C2 allow the performance of instrumented fuel rod irradiation in order to:

— Measure the central temperature of one fuel rod in correlation with several ranges of power and burnup (see Figs 8 and 9);
— Measure the internal fission gas pressure evolution versus varying burnup and power originating from power ramps;
— Measure the composition of fission gas release during irradiation (see Table 5);
— Determine the elongation of one fuel rod during irradiation.

Each type of experiment performed in the capsules is under natural convection cooling inside the in-pile section. The heat generated by one experimental fuel rod, i.e. 35 kW, is dissipated to the pool through the external

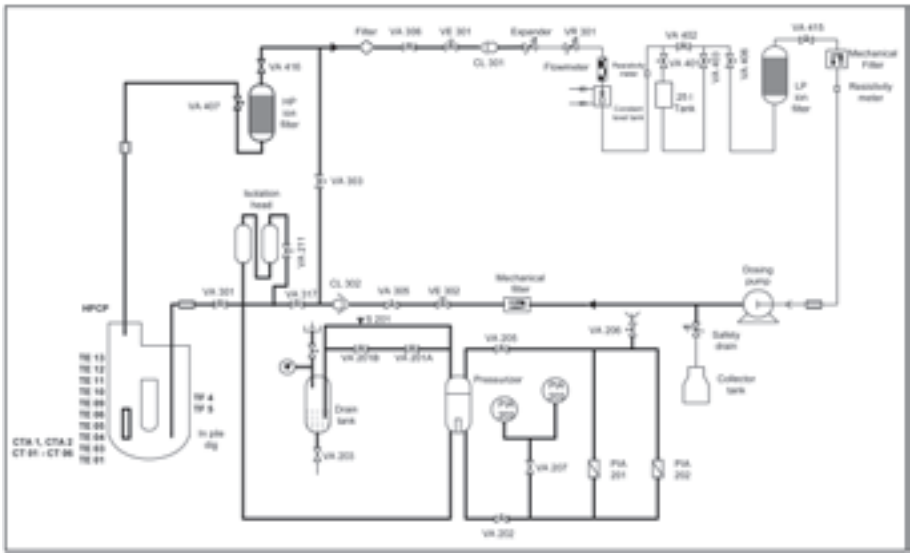

FIG. 8. Capsule C1/C2 flow scheme.

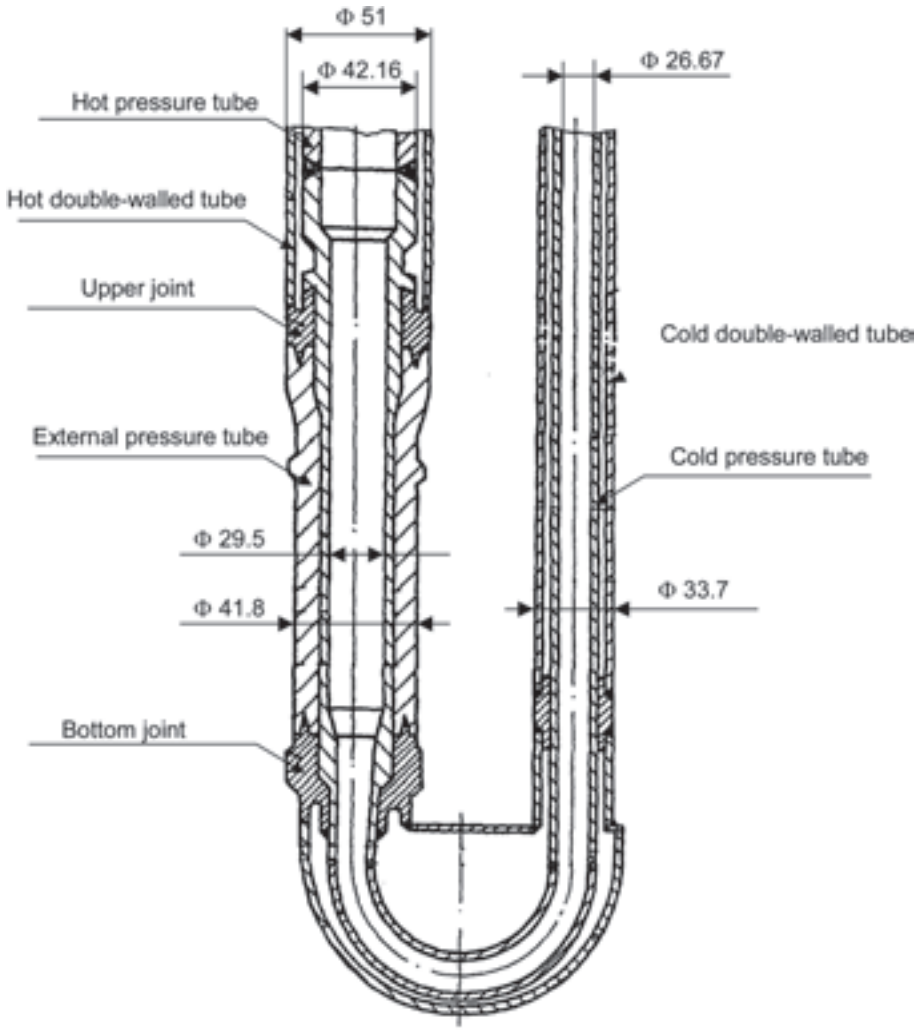

FIG. 9. Capsule C1/C2 in-pile section.

pressure tube. The out-of-pile section is shielded using lead bricks together with remote control valves. For each type of test listed in Table 5, dedicated instrumentation of the fuel element and the in-pile part was installed to collect the test data during irradiation.

4.3. Capsule C5

Capsule C5 [2] was designed and used for the irradiation of steel samples and zirconium alloys in helium atmosphere at temperatures between 200 and

TABLE 5. TESTS PERFORMED IN CAPSULES C1, C2 AND C5

Capsule	Experiment	Irradiation period
C1	Fission gas composition for CANDU type fuel element	20.03.84–15.08.86
C1	Densification tests for CANDU type fuel element	15.06.87–18.09.87
C1	Power ramping tests	13.10.87–19.07.89
C2	Dimensional measurement	10.81– 10.84
C2	Fission gas pressure	26.03.84–21.09.94
C2	Power ramp	03.12.85–30.05.87
C2	Irradiation conditions calibrations	26.02.92–26.06.92
C2	Residual deformation of cladding determination	12.12.96–05.12.01
C2	Central temperature measurement in fuel	13.05.98–08.10.98
C2	Fission gases release effects on temperature	15.05.00–07.11.00
C5	SS 403 M irradiation (irradiation time: 1983 h)	18.06.85–04.10.85
C5	SS 403 M irradiation (irradiation time: 1121 h)	30.05.86–19.08.86
C5	Zircaloy-4 tube irradiation (irradiation time: 6603 h)	22.12.87–28.11.88
C5	Zircaloy-4 tube irradiation (irradiation time: 6660 h)	23.07.92–21.07.94
C5	Zr-2.5% Nb alloy irradiation in inactive atmosphere (helium) (irradiation time: 11400 h)	11.10.95–10.12.01

300°C (Fig. 10). Table 5 contains the experiments performed using C5. The C5 in-pile section was designed to irradiate several types of samples of cladding as well as pressure tubes. The specified test temperature is obtained by gamma heating and/or heating by an electrical furnace.

4.4. Capsule C9

Capsule C9 was designed and operated to simulate the behaviour of one fuel rod in load follow-up mode under power plant conditions. Figure 11 shows the flow diagram with its specific features to allow continuous power variation while the pressure is kept constant inside the device. Figure 12 adds the main mechanical parts: the single walled pressure tube in a calorimeter tube and the mechanical displacement system.

4.5. Capsule C6

Capsule C6 [4] was designed to simulate reactivity insertion accidents (RIA), inserting it into the ACPR with a pulse amplitude of up to 20 000 MW

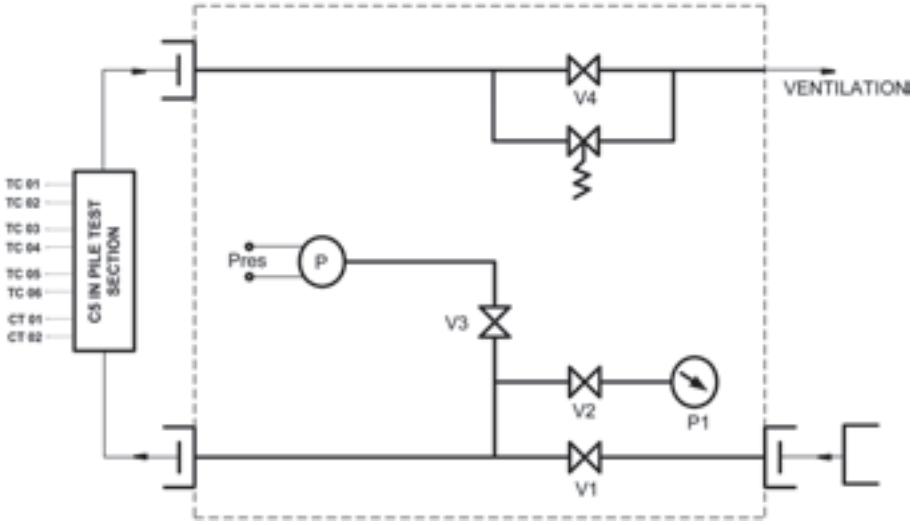

FIG. 10. Capsule C5 flow scheme.

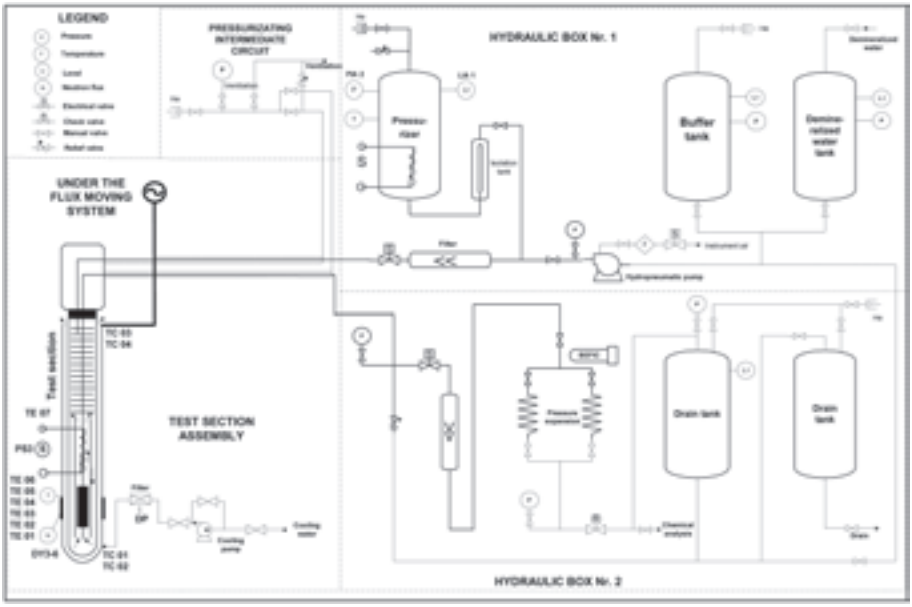

FIG. 11. Capsule C9 flow scheme.

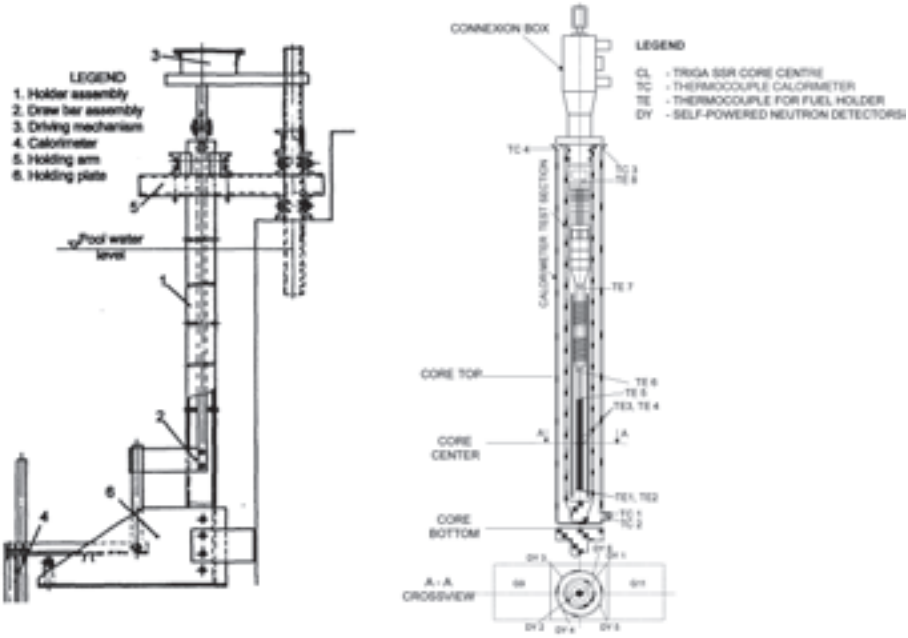

FIG. 12. Mechanical structure and instrumentation of capsule C9.

(Figs 13 and 14, the latter showing a fast data recording example). Based on the specific requirements of the fuel manufacturer, a safety research programme on CANDU type power reactor fuel was run between 1984 and 1989. At the beginning of 1997, the experiments were resumed and four more samples have been tested.

4.6. Post-irradiation treatment

All fuel rods irradiated in these loops and capsules were transferred to the hot cells in the post-irradiation examination laboratory, where non-destructive and destructive tests were conducted. The irradiation and post-irradiation data were used by the fuel performance analysis laboratory to improve or confirm some technological features of the fuel fabricated by the institute before 1992. Those data were also used for certain computer code validation and development.

The experimental data obtained from irradiation of fuel elements in 'fission gas pressure tubes' were used in the intercomparison of fuel code prediction, an activity led by an IAEA technical committee meeting on fuel performance, in Vienna, 2001.

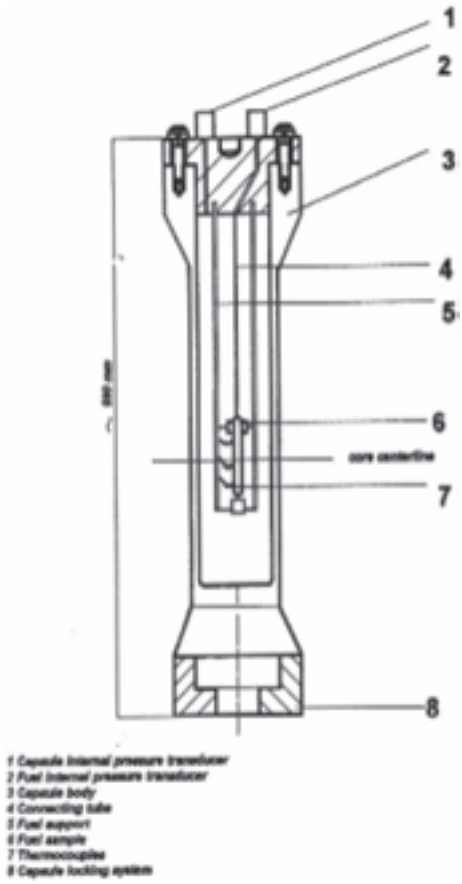

FIG. 13. Mechanical structure of capsule C6.

5. FUTURE OBJECTIVES

The future of fuel testing at the INR-RR will see tests on pre-irradiated CANDU type power reactor fuel in an outer ring at the ACPR core, for which a new concept of irradiation device will be developed with the schedule: design 2003, component manufacture 2004, and commissioning 2005.

The future activities focus on the utilization of the research reactor and of the irradiation devices/facilities for the development of a new fuel type, for international cooperation following a similar external request, as well as within an IAEA regional project. These activities will be based on design modifications of an irradiation device in order to operate under LOCA conditions [3] and refabrication of experimental fuel rods from fuel operated and transferred

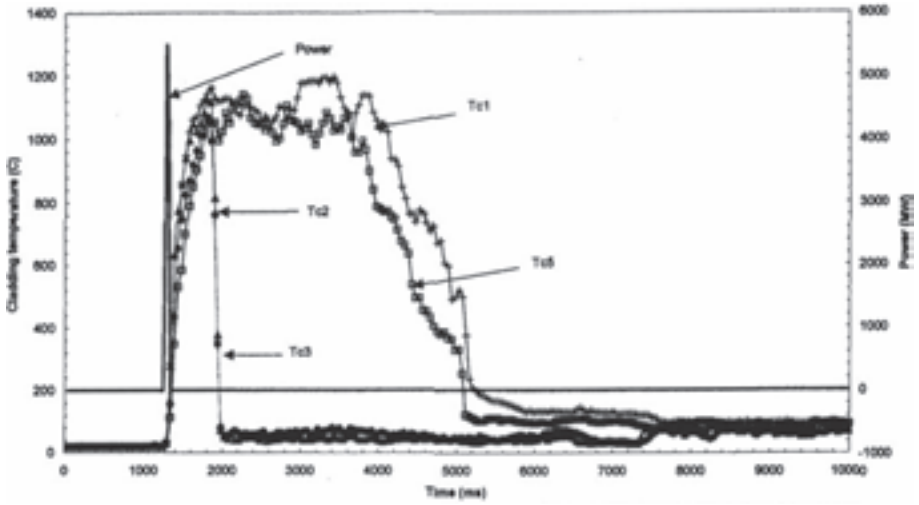

FIG. 14. Cladding surface temperature versus reactor power at capsule C6.

from the WWER type nuclear power plant and from the CANDU type nuclear power plant. LOCA and RIA tests using refabricated fuel rods are planned for 2005.

6. MINIMUM PREREQUISITES AND CONSTRAINTS CONCERNING SAFE INSTALLATION AND OPERATION OF IRRADIATION DEVICES AT THE INR-RR

6.1. Minimum prerequisites

In order to maintain safe conditions for reactor and irradiation devices, an efficient utilization and reliable generation of experimental data, the following prerequisites were settled as a consequence of the experience at the INR-RR, partially repeating what has been said already in Section 1:

- Ensure by neutronics and thermal hydraulics evaluations that the design of the experiment/test linked to the research reactor core complies with the irradiation specifications and the safety limits;
- Provide a mechanical workshop for fabrication/manufacture of special items of irradiation devices (specific for each test);
- Make available a laboratory, laboratory techniques, components and materials, as well as specialists for the instrumentation of irradiation devices and of fuel samples;

— Provide a data acquisition system with adequate staff support;
— Employ specially trained and licensed personnel for irradiation devices operation and maintenance, independent from the staff performing the reactor operation;
— Establish safety provisions for normal and emergency operation of the irradiation devices;
— Make available hot cells for handling of devices and facilities and their sample loadings in the irradiation sample holders;
— Provide a radwaste facility for the decontamination of irradiation devices, following normal operation or accident conditions, as well as for the handling of radwaste produced when fuel irradiation experiments are abandoned.

6.2. Constraints imposed on design and operation of irradiation devices

The following conditions and constraints are imposed on the design and the operation of irradiation devices and facilities at the INR-RR:

— Power reactor parameters inside the pressure tube located in or near the research reactor core entail the assembly of requirements for design, fabrication, testing, licensing, operation and maintenance as nuclear equipments, which meet the ASME Class 1 and Class 8 demands;
— High increases of internal irradiation device contamination should be considered and handled, to prevent long term shutdown of the reactor and of other facilities under development and use, avoiding cross-contamination and releases;
— The overall reliability of the irradiation devices should be higher than that of the research reactor hosting them, in order to achieve the designed availability of the reactor;
— The safety analysis of the irradiation devices becomes a part or an addendum of the safety report of the research reactor plant, with the main requirement that all safety issues of the irradiation device remain within the limits and conditions previously approved for the research reactor.

7. QUALITY ASSURANCE AT THE INR-RR

Two years prior to commissioning the reactor in 1979, the operator was requested by the Romanian regulatory authority to develop and apply a quality assurance (QA) programme linked to the reactor commissioning activities and,

later on, after a successful startup, to apply QA in the operation and utilization of the research reactor. Broad experience in this field was accumulated and subsequently used in other departments of the INR. At present, the INR is accredited for ISO 9001-2000 by the Lloyd's Register, comprising reactor operation, as well as the supply of products and services.

8. CONCLUSIONS

The TRIGA research reactor at the INR is integrated into an environment of installations, i.e. several irradiation facilities, a hot cell post-irradiation examination laboratory, and a radwaste conditioning plant. This research infrastructure sustains activities of a number of research programmes developed by other departments of the INR for the INR's own needs and in the interests of international cooperation.

The irradiation devices for power reactor fuel testing, such as those in operation at the INR-RR are key purpose built facilities, which allow the performance of the requested research activities.

The good records on nuclear safety, the extensive utilization in the past, the continuous efforts for upgrading, and the core conversion entirely to LEU fuel will make this relatively young research reactor plant available for research programmes and for technological development over the next two decades.

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2.4. FUEL AND MATERIAL IRRADIATION AT THE HANARO FACILITY

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1. INTRODUCTION

In the Republic of Korea, 16 nuclear power reactors, 12 PWR plants and 4 PHWR plants are operated commercially. The nuclear share is above 43% of the total electric power generation. Therefore an intensive effort is now under way to develop advanced fuel, such as the direct use of spent PWR fuel in CANDU reactors (known as DUPIC fuel) and advanced PWR fuel to be made available for use in nuclear power plants in the Republic of Korea. As a part of the national nuclear R&D programmes, a series of in-pile tests are being carried out to examine the performance of advanced fuel compared with standard fuel. One of the important points among the main issues of concern is thermal behaviour. At the Highly Advanced Neutron Application Reactor (HANARO), the irradiation needed to verify the pellet manufacturing process and product characteristics can be performed. To this end, two types of irradiation tests were planned at the HANARO: capsule tests and loop tests. A fuel test loop is currently not available; however, seven in-core test holes for irradiation of small specimens, such as fuel rods or pellets, are on hand at the HANARO. Most of the separate tests for the development of advanced PWR fuel, DUPIC fuel and metallic fuel carried out since 1998 were using a non-instrumented capsule [1–5]. Hence, since 2000, new irradiation technology development using an instrumented capsule has become highly desirable, to provide more qualified data to fuel designers [6–11].

2. GENERAL CHARACTERISTICS OF THE HANARO

The HANARO is an up-flow light water cooled, heavy water moderated tank in open pool type research reactor [12]. It was designed to meet the growing need for a high intensity neutron source in the area of national nuclear R&D. In order to satisfy the needs, the following major performance requirements were adopted in the design:

— The maximum thermal neutron flux should be greater than 5.0×10^{14} n·cm^{-2}·s^{-1};
— The axial neutron flux distribution should be ±20% over 500 mm in length;
— The refuelling cycle should be at least four weeks and the discharge burnup should be greater than 50%;
— The reactor must be able to operate at 25 mk of excess reactivity;
— The reactor should have a capability of performing various experiments without interference;
— The reactor must have inherent and passive safety features.

The reactor assembly consists of the inlet plenum, the lower grid plate, the reflector tank, the flow tubes and the chimney. The stainless steel inlet plenum (2 m in diameter and 0.6 m in height) supports other structural components and distributes the coolant at the inlet to each flow channel. The reflector tank of 2 m in diameter and 1.2 m in height, which contains D_2O, surrounds the reactor core and accommodates various vertical and horizontal experiment holes. In the chimney, the heated coolant is mixed with cold bypass flow coming down from the chimney top and is finally sucked out through two outlets. A total of 39 flow tubes are available for fuel assembly loading and for irradiation rigs. This structure is submerged to the bottom of the open reactor pool, which is 4 m in diameter and 13.4 m in depth.

The reactor core consists of an inner core inside the inner shell of the reflector tank and an outer core outside the inner shell as shown in Fig. 1. The inner core provides 23 hexagonal and 8 circular flow channels, while the outer core has just 8 circular flow channels; 36 element and 18 element fuel assemblies are loaded into the hexagonal and circular flow channels, respectively. The fuel element in each assembly is a rod of 6.35 mm diameter and 700 mm length, containing as fuel 19.75 w/o enriched U_3Si–Al 'meat' enclosed in aluminium cladding. The cladding has eight longitudinal fins to enhance the heat removal from the fuel rod surface.

The reactor regulating system (RRS) consists of four cylindrical hafnium shrouds, fission chambers, programmable controllers, stepping motors and magnetic clutches. The fine and coarse control of the reactor power is provided by the RRS such that the maximum power increase rate is less than 5% present power per second (PP/s). In the case of an RRS trip, the four shrouds of the RRS are inserted into the core by gravity since their magnetic clutches are de-energized.

The reactor protection system (RPS) also has four absorber rods, which are identical in their neutron absorbing parts to those of the RRS; however, they are operated by hydraulic pumps. For reliability of the RPS, the system

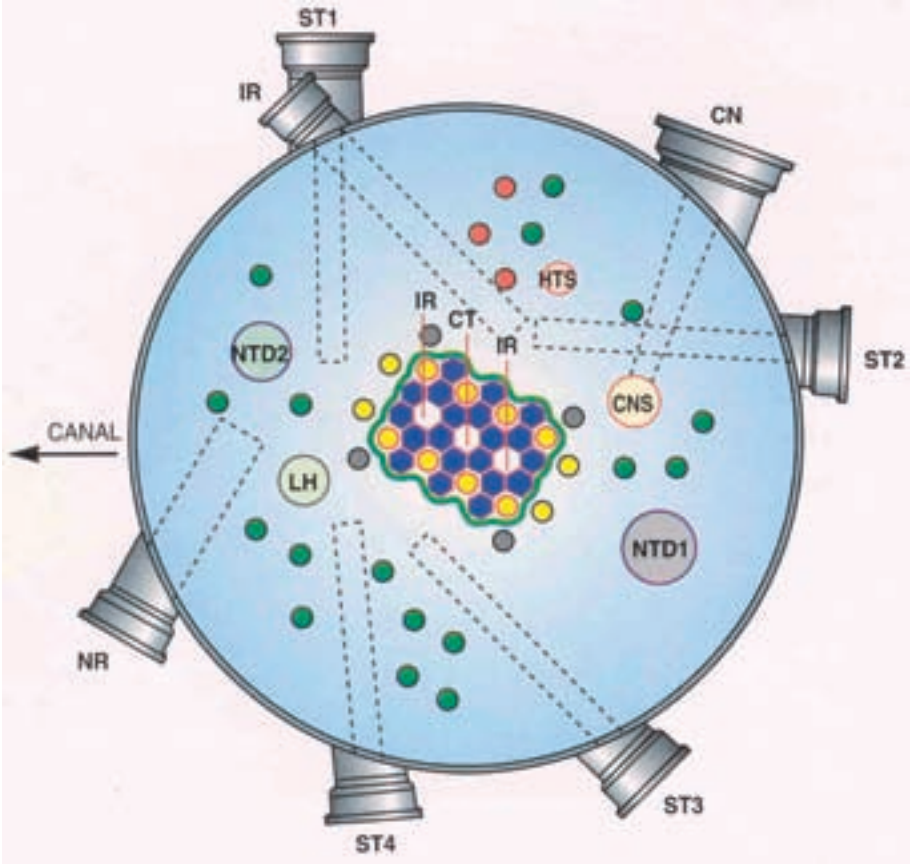

FIG. 1. Horizontal section of the HANARO core and reflector. The vertical experimental holes comprise IR, OR, CT: capsule irradiation, RI production, fuel test loop; LH: RI production, fuel test loop; HTS, IP: RI production; PTS: neutron activation analysis; NTD: neutron transmutation doping. The horizontal experimental tubes comprise ST1: polarized neutron spectrometer; ST2: high resolution diffractometer, four cycle diffractometer; ST3: reflectometer; ST4: triple axis spectrometer; IR: boron neutron capture therapy; CN: small angle neutron spectrometer; and NR: neutron radiography.

was designed to be physically separated and functionally diverse from the RRS. When the RPS trip signal is generated, solenoid valves actuate the hydraulics to initiate rapid shutdown.

At normal operation, the pumps press the light water coolant into the lower plenum, so generating an upward flow of 7.3 m/s velocity through the flow tubes. Finally, the coolant exits the core at its top. Additionally, about 10% of the total primary cooling system (PCS) flow of 780 kg/s is fed into the bottom of

the pool and rises slowly along the outside of the reactor structure. Thereafter it is drawn into the chimney and meets the core flow at the chimney bottom. Both the core and bypass flows are sucked into the chimney nozzles. During a reactor shutdown, the core decay heat can be removed through the PCS loop by natural convection using the same path as the PCS during a normal operation, as long as the heat sink provided by the secondary flow is available. Otherwise, the decay heat is dissipated into the pool by buoyancy driven natural convection flow enabled by opening pressure–difference–balanced flap valves. The reflector is cooled by a separate cooling system to keep the D_2O temperature at an adequate level for the neutron economy, and so are the experimental facilities.

The secondary cooling system removes the heat from the primary coolant, from the heavy water of the reflector, from the water of the spent fuel storage pool and from the coolant of the experimental facilities. It consists of three pumps (one on standby) and a four cell cooling tower.

3. CURRENT IRRADIATION TESTING ACTIVITIES FOR FUELS AND MATERIALS

The primary experimental facilities are installed in the flux trap positions within the in-core region of the HANARO. In addition, several capsule irradiation tests can be performed in the out-of-core region. Many of the irradiation experiments in the HANARO are capsule experiments because of the ease of construction and handling. Non-instrumented and instrumented capsules for fuel and materials irradiation experiments are open to internal and external users. The main activities of the capsule development and utilization programme are focused on in-reactor material tests, new and advanced fuel R&D, and basic research. Major irradiation facilities, as shown in Fig. 2, are composed of the capsule, the temperature control system, the capsule supporting system, and the capsule cutter.

The current application programme of the capsules in the HANARO is as follows:

— Reactor materials tests:
  • Reactor vessel materials: SA508 and weld material, SA533 (KAERI programmes);
  • Reactor pressure tube material: Zr–2.5Nb;
  • Structural materials: STS304, STS316, STS321, Cr–Mo alloy;
— Fuel tests:
  • Advanced PWR fuel;
  • DUPIC fuel;

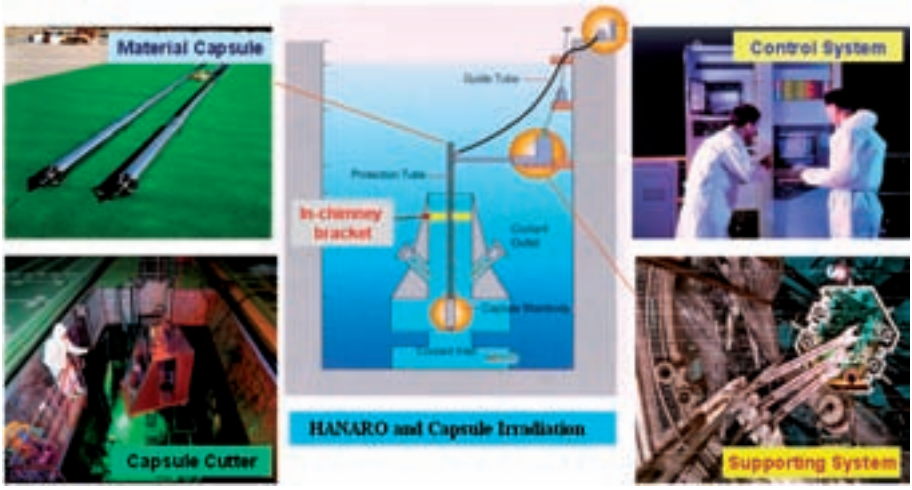

FIG. 2. The HANARO's capsule irradiation facilities.

- Burnable poison materials: Gd_2O_3–TiO_2, Dy_2O_3–TiO_2;
- U–Zr alloy;

— Tests for materials applied by industry:
- Reactor vessel materials: SA508, SA533 (industry programmes);

— Fundamental research:
- Zr–1Nb–1Sn–X alloy, Zircaloy-4, STS350, STS 304, STS 309;
- Semiconductor and magnetic materials;
- UO_2 fuel;
- Neutron dosimeter, etc.

4. CONSTRAINTS IMPOSED ON THE DESIGN OF THE CAPSULES

Currently, a small number of the operating research reactors have a power large enough to perform irradiation of nuclear fuels and materials. The HANARO was specifically designed to provide essential information for the development of nuclear fuels and materials in the Republic of Korea. The design requirements for the fuel and materials irradiation tests considered during the HANARO's design phase [12] were:

— Vertical in-core hole with diameter ≥10 cm, thermal neutron flux $\geq 2 \times 10^{14}$ n·cm^{-2}·s^{-1};
— Vertical flux trap with diameter ≥5 cm, thermal and fast neutron fluxes $\geq 5 \times 10^{14}$ n·cm^{-2}·s^{-1} for radiation damage.

Also, there are some limitations for the experimental requirements of the user. In order to meet all the demands which are related to the reactor design features, the following specifications are reviewed at the design stage of the irradiation capsule:

— Neutron spectrum/flux level to enlarge the possibility to irradiate samples up to the required fluence or burnup;
— Excess reactivity;
— Accessibility of the test holes to minimize the interface with nearby structures;
— Structural integrity during irradiation;
— Thermal balancing of the device under the thermal hydraulic conditions of its operation;
— Specimen volumes, etc.

4.1. Required analysis

The entire tasks in the design of the irradiation devices, such as a capsule, definitely include a detailed analysis of the thermal hydraulics and of the structure related stress, along with shielding and vibration analysis (see Refs [8–11]). Particularly, the capsule structure is a slender corpus having an L/r ratio of more than 15. Thus, the buckling and impact analyses are important to evaluate the structural integrity of the capsules at their early design stage [7]. Seismic analysis including vibration is also required; based on the ASME B&PV code, with parameters such as displacements, reaction forces and stresses (Section III, Div. 1, subsection NF-3322).

4.2. Required tests

Any new capsule should be structurally integrated and reconciled with the HANARO's hydraulic conditions. For design verification of a new capsule, major tests required are a pressure drop test, vibration test, endurance test, etc. These tests will be performed using a mock-up capsule in the out-of-pile test facility, which has the same flow tubes as those in the HANARO. These tests, if necessary, will be performed using a mock-up capsule, in accordance with the following operating requirements in terms of the HANARO:

— Pressure drop requirement: pressure difference >200 kPa, mass flow < 12.5 kg/s in the test hole;
— Vibration test: the allowable total displacement should be less than 300 μm under the HANARO hydraulic condition. As a result, from the

displacement data it is ensured that the capsule does not interfere with any other structure;
— Endurance test requirement: one tenth of the total time of habitation in the reactor, coolant temperature 40°C, 110% of the normal flow rate. After completion of the endurance test, a visual examination will be performed and the dimensional changes will be checked.

5. REQUIREMENTS FOR IRRADIATION AT THE HANARO

5.1. Limits on reactivity

Several limitations of the reactivity are applied. The total reactivity of all the irradiation samples must be lower than 25 mk [13]. The reactivity worth of any stationary sample must be less than 12.5 mk. Reactivity insertions or removal rate must be less than 0.125 mk/s.

5.2. Thermal hydraulic requirements

Absolutely no surface boiling is allowed in the HANARO. To guarantee the thermal margin for the fuel sites within the actual core, the flow in the test channel must not be larger than in the fuel channels. In the out-of-core tests, any user must check the differential pressure and the flow rates.

5.3. Structural interface with the HANARO

The interface between the experimental inserts and the reactor structures must be checked. That checking includes the interface of the insert support structure as installed in the chimney and the flow induced vibration and any resulting fretting wear. In the case of a long lasting irradiation, the safety review committee may require a special endurance test.

5.4. Safety analysis

In the case of fuel irradiation, the maximum linear power of the test sample for the entire irradiation period must be estimated. Based on that estimation, an accident analysis is performed to evaluate the integrity of the sample (maximum temperature) and of the surface of the cladding of the sample minimum departure of nucleate boiling ratio (MDNBR). Two design basis accidents, i.e. the locked rotor of one PCS pump (loss of flow) and the

spurious withdrawal of the control absorber rod (CAR) (reactivity induced accident), are checked.

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2.5. FUEL SAFETY TESTS IN MATERIAL TESTING REACTORS

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1. INTRODUCTION

Nuclear research reactors are facilities around a fission neutron source. They are operated for fundamental and applied research. Among their potential activities are the so-called safety related tests. These tests are performed in the framework of fuel qualification (for fuel licensing) or fuel safety R&D (for reactor licensing, reactor complementary safety analysis or reactor safety reassessment).

These tests are performed by submitting, under adequate irradiation conditions, fuel samples to separate effect tests or integral trials representative of transient type, incident or accident type phenomena, each for the corresponding reactor type.

Depending on need, these tests can be performed on non-irradiated or pre-irradiated fuel samples, with the possibility of testing either tight or pre-failed fuel samples. The experimental protocol may involve a deliberate significant failure of the samples up to sample melt.

Constraints of such tests may lead to the construction of dedicated research reactors to host a single family of safety tests (e.g. for R&D dedicated to severe accidents). A consequence of such dedicated plants is that they offer limited flexibility for evolution to other experimental programmes.

Therefore, as a natural complement to dedicated safety research reactors, a significant interest exists in assessing the possibilities to perform more limited tests in multipurpose research reactors, especially in material testing reactors (MTRs). Nonetheless, due to intrinsic safety aspects of such experiments, this goal has to be approached very cautiously. In particular, such programmes should only be conducted in multipurpose research reactors presenting adequate provisions to respond to the technical, operational and/or safety requirements of such tests.

2. GENERAL SURVEY OF FUEL R&D NEEDS

A key challenge for any application of nuclear power is that the reactors be run safely and economically. This is especially fundamental for power reactors linked to the electric power grid.

Fuel behaviour considerations represent a major issue within this challenge. It is essential to accumulate a sufficient level of understanding of fuel behaviour and to bring this knowledge into codes to subsequently provide adequate predictions. This in turn leads to a better understanding of fuel performance, to a possibility of optimization in operation margins, to flexibility in fuel management and to improved operation economics for the corresponding reactor types.

For reactor licensing, reliable predictions of fuel behaviour constitute a basic demand. Design issues in connection with justified predictions of fuel performance and safety are fundamental. The ultimate goal is a characterization of fuel behaviour in both normal and off-normal conditions. From this knowledge, design and operating rules can be derived to prevent fuel failures and the subsequent release of fission products to the environment. Moreover, for incident or accident conditions, the target is to prevent or mitigate fuel damage by the definition of (and, subsequently, the check of compliance with) a set of appropriate technical decoupling criteria.

Furthermore, beyond initial reactor licensing, there exists a strategic field for which technical solutions to improve the economics of operating reactors are investigated and provided. In this field, the possibility of fuel upgrades in terms of burnup increases or other improvements in the technical–economical performance of fuel is a key R&D topic. Such upgrades require the generation of a relevant safety file for the fuel as well as an (at least partial) re-examination of the safety file of the host reactor. This leads to ad hoc programmes to establish an adequate description of the behaviour of the upgraded fuel in both normal and off-normal conditions.

The basis of the necessary experimental data typically comes from three main types of tests, which are complementary and necessary:

— Out-of-reactor tests on small scale samples, in order to identify and quantify the basic mechanisms;
— In-reactor separate effect tests on short fuel rods or fuel rod sections;
— In-reactor integral tests, taking into account a maximum of mechanisms simultaneously, and in any case the fuel bundle geometry.

Corresponding fuel test programmes can be applied to existing fuel designs, to evolutionary ones or, especially for innovative reactor concepts, to

designs realizing a complete technological breakthrough. The R&D plan has to take into account those different realities. In particular, the volume of necessary precharacterization, in parallel with the separate effects for non-irradiated fuel, have to be considered, as well as semi-integral trials in research reactors, all drastically increasing with the degree of innovation of the fuel concept under testing.

3. MAIN CLASSICAL EXAMPLES OF FUEL TESTS UNDER NEUTRON FLUX: CASES OF NORMAL, TRANSIENT AND INCIDENT TYPE

This section succinctly reviews typical examples of fuel tests under irradiation conditions representative of normal operation, transients and incident sequences. The following generic items are reviewed (independently of the technology of corresponding nuclear power reactors): steady state analyses, ramp conditions and fission gas release analyses.

For each generic example reviewed here, a few technical hints are provided in view of the test conditions at the research reactor plant.

3.1. Steady state conditions

Steady state tests are mentioned for the sake of completeness although they do not, strictly speaking, constitute fuel safety tests.

Steady state tests correspond to a set of irradiation tests of fuel samples at various power levels, typically up to the maximum power level planned for steady state operation of the fuel under testing. These tests can be directed towards parametric studies, systematic studies or long period irradiation. They can provide useful information in terms of the technological power limits of the fuel tested.

Steady state irradiation tests are typically performed at multipurpose research reactors (see Fig. 1), which have the flexibility to adjust sample power levels, to adjust sample average temperatures (in experimental devices) and, up to a certain extent, to adjust the neutron flux spectrum at the fuel samples (through adjustment of the core/reflector layout or of the experimental device layout, e.g. by neutron absorbing screens). It has to be noted that a very useful means to adapt the sample power level is to manufacture the sample with a suitable enrichment level for the test, different from that for the final application in the power reactor.

In a further step, the fuel can be tested in lead test assemblies (LTAs) at nuclear power plants. In this test phase, the number of fuel assemblies and their

FIG. 1. TANOXOS irradiation device at the OSIRIS material testing reactor.

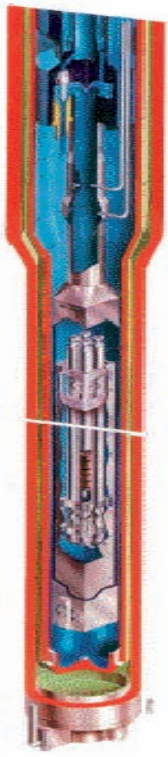

FIG. 2. The ISABELLE four rod irradiation device designed for power ramp studies.

burnup is progressively increased. At each step, fuel examinations are carried out to check the behaviour of the fuel in view of licensing requirements.

3.2. Transient and ramp conditions

These tests represent the border case between fuel standard tests and fuel safety tests.

Ramp conditions consist of submitting fuel samples to controlled power variations up to significant power gradients (e.g. 100 W/cm per minute). These variations are intended to be representative of load following (at low power gradients) up to incident type power transients (at significant power gradient).

Such tests are notably used to check the correct behaviour of fuel during such transients (especially in connection with the burnup as a parameter for the tests). Safety criteria can be checked, for example, for water cooled nuclear

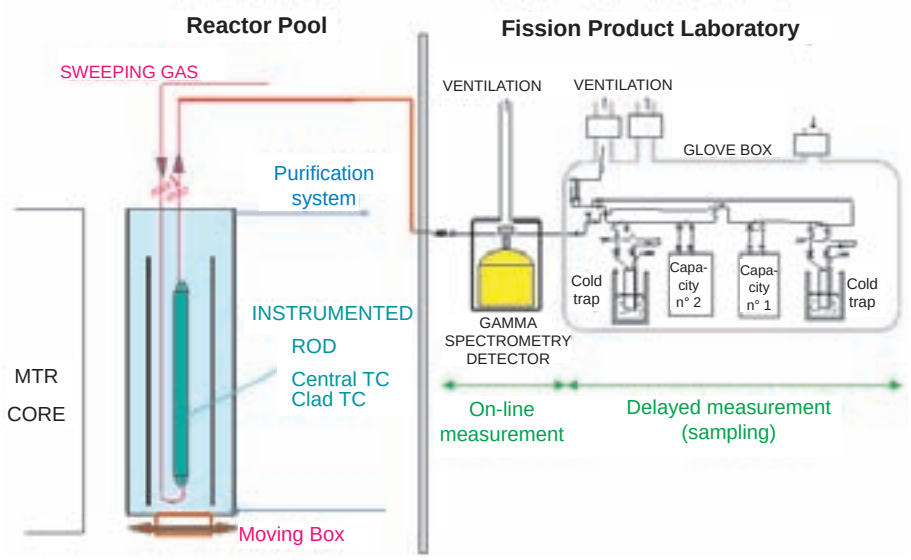

FIG. 3. Schematic set-up of a fission gas release experiment.

power plants, the pellet clad mechanical interaction (PCMI), which is related to the stress on the cladding produced by pellet expansion during a short period of time. This situation can result in an interaction and, if the stress is sufficiently high and the cladding ductility low enough, it can lead to a clad failure.

This kind of experiment is typically carried out in multipurpose research reactors also, which provide the necessary flexibility to accommodate power variations and host test devices which enable the testing of samples in different thermal hydraulic conditions from those of the multipurpose research reactor core. Power variations can be performed at constant core power by variations of the conditions at the test device (see Fig. 2). Corresponding technical solutions are variable neutron screens (solid or gaseous) or the use of displacement systems (especially at the core periphery).

3.3. Fission gas release tests in steady state and/or ramp conditions

The objective of this type of test is the parametric study of fission gas release (FRG) from fuel samples according to various parameters, such as temperature, ramp rate and atmosphere (oxidizing, reducing, steam, etc.), in order to simulate the conditions encountered during various events.

This type of test permits the identification of basis mechanisms for FGR (such as gas located in inter- or intra-granular positions) and assessment of the influence of the design and irradiation power history on FGR. The scheme of a typical set-up is shown in Fig. 3. These tests greatly help identify and quantify

models used in codes and thus improve their prediction capability. A complementary application is to quantify the source term for transients in safety studies.

A correct achievement of this type of test relies on the capability to build up short half-life fission products and to measure, within a very short time frame, the release of the different fission gases. Therefore, it requires that the FGR test phase (either steady state or ramping) be performed very soon after the irradiation phase or directly under irradiation conditions. Furthermore, due to the nature of products measured, there is a clear interest to provide on-line fission gas sampling and measurement. This can be achieved through implementation of a sampling line from the fuel sample to a fission product analysis device.

MTRs, provided they have satisfactory design and operation conditions, are adequate tools for such tests. Nonetheless, correct provisions have to be implemented, as such tests involve sampling and transportation of gaseous fission products with possible irradiation and contamination risks by the test device or the sampling and measurement device. Furthermore, purification and storage of released fission products must be provided.

Proper design and operation measures need to be implemented to prevent short term or long term contamination risks for the facility due to the different aspects of the test life cycle (e.g. test device preparation and retrieval, connecting and disconnecting at the irradiation position, sampling process) and to ensure the ad hoc radiological protection of facility personnel.

4. ACCIDENT TYPE TESTS

The main classical examples of fuel testing under neutron flux conditions are the accident type tests. Fuel behaviour under accident conditions has to be studied in tests representative of the addressed mechanisms. For the present industrial power reactors, the most common accidents studied are the loss of coolant accident (LOCA) and the reactivity insertion accident (RIA). Furthermore, special investigations are performed in terms of the simulation of severe accident conditions.

4.1. LOCA conditions

Loss of coolant accidents are notably studied for LWRs. The test sequences typically consist of a blowdown, a cladding heat-up under steam cooling conditions provoking clad burst and oxidation, a quench and a

post-quench phase. The power level is normally very moderate, of the order of a few W/g (corresponding to 10–20 W/cm at PWRs).

The current regulatory safety criteria for LOCA, still in use in most countries, are derived from the emergency core cooling system (ECCS) acceptance criteria that were issued by the US Atomic Energy Commission in December 1973 and published in the US Code of Federal Regulations (10.CFR50, part 50.46). Among the relevant criteria, it is emphasized that the calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

This accident is characterized by the assumption of a certain number of fuel failures, of the non-fragmentation of rods (the rods keep their geometry), and of ensuring long term core cooling.

Technical issues associated with LOCA can be investigated either through separate effect or integral experiments to be performed, depending on constraints, under non-irradiated conditions or under irradiation in dedicated research reactors (see Part 8), or in multipurpose research reactors.

Historically, separate effect programmes have been undertaken on the kinetics of cladding oxidation at a temperature of around 1200°C and on the assessment of the ductile–fragile limit. Integral tests in research reactors have also been performed and are foreseen in several countries. Such experiments involve submitting fuel samples to accident type thermal hydraulic transients, with coolant restriction and/or evacuation according to a given scenario (typically involving sample deflooding and final reflooding). The temperature finally reached and the final sample state depend on the test input.

The very stringent test conditions require that these tests be performed in dedicated research reactors with special safety features. However, using standard multipurpose research reactors can be envisioned (and has been effected) for less stringent test conditions (typically, with automatic power transient shutdown when clad decoupling temperature criteria are reached). Key aspects of such tests are data acquisition during the transient (sample temperature monitoring, fission product release monitoring, etc.) and management of the test device after test sequence completion. Figure 4 shows the design concept of a device for fuel testing under LOCA conditions, Fig. 5 adds the flow diagram of a real device for LOCA tests.

In this context, specific attention has to be paid to contamination and safety aspects. With regard to contamination, the main issue is the involvement of fission product release (in any form, not only gaseous) into the environment of the sample (notably following clad rupture), with consequential contamination of the coolant in the test device and of the sampling lines. Furthermore, the problem of contaminated sampling lines as for gas sampling in the FGR tests is extended to liquid sampling here. Specific technical solutions have to be

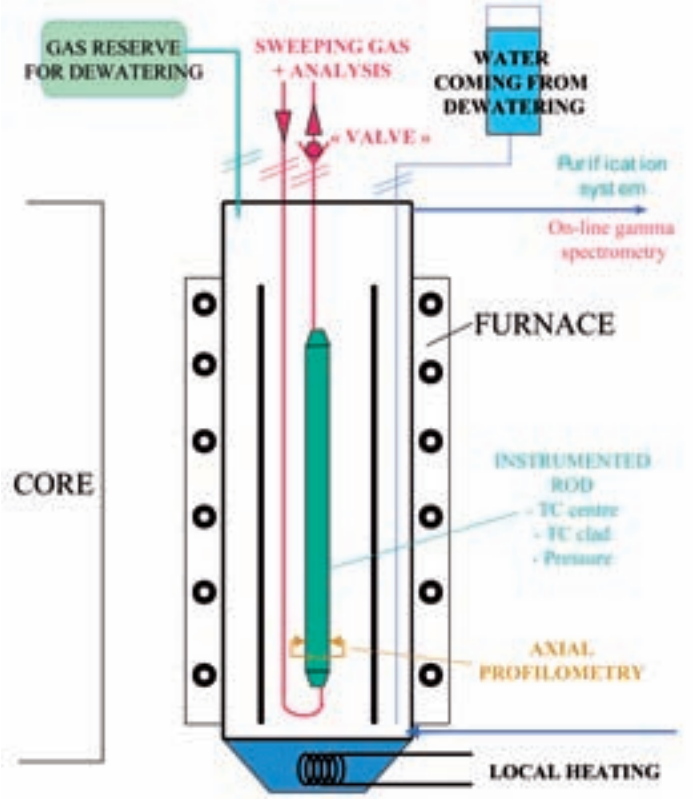

FIG. 4. Conceptual design of an in-pile separate effect test in a multipurpose research reactor for studying LOCA type phenomena.

provided in order to prevent short term as well as long term contamination in the test facility originating from the performance of such tests, and to ensure radiological protection of the staff.

With regard to safety issues, by definition, such tests involve fast thermal hydraulic power transients on fuel samples. Therefore a key aspect is to prevent the propagation (due to any design or operation failure) from standard test input conditions to more degraded fuel conditions (e.g. with regard to fuel temperature) during the test, which could generate sample coolant interaction phenomena beyond those taken into account in the test input conditions. This aspect requires very specific design and operation considerations before any such test is undertaken. These considerations have to address both the test device as well as the hosting reactor design and operation, with particular regard to their robustness against active or passive failures or their accumulation.

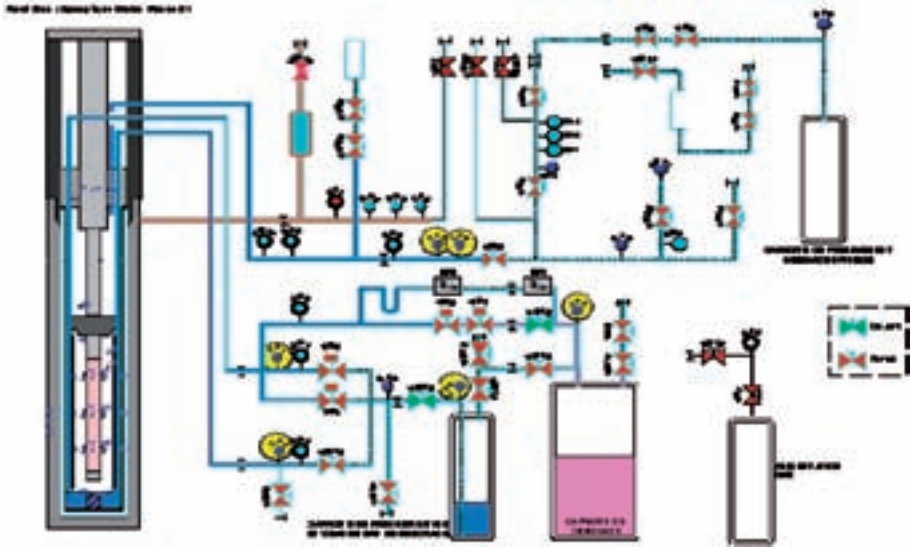

FIG. 5. Flow diagram of a LOCA type experiment: the FLASHMOX programmme.

In principle, due to the constraints recalled above, such tests should not be undertaken in multipurpose research reactors (using irradiation devices in those MTRs) without relevant design and safety provisions to handle the risks. Furthermore, the size of sample tested can be minimized deliberately and, apart from generic design and operating conditions, analyses can result in the implementation of temporary supplementary provisions in those MTRs when such tests are performed.

4.2. Reactivity insertion accident conditions

RIAs are very low probability events characterized by accidental injections of extra reactivity in reactor cores. The time period of the transient may vary from 10 ms up to a few hundreds of milliseconds and the power level during the pulse can be very high, e.g. more than 1000 times greater than the rated power.

RIA tests in research reactors are important for determining a fuel safety margin in view of rupture and for assessing potential failure consequences, such as fuel degradation and dispersal. Such tests are essential also for understanding the mechanisms acting during the transient sequence, prior to and after the failure, and for writing and testing reliable transient codes.

Two approaches are typically used in safety analyses. The first one is based on energy deposited during the test (at the time of failure). The second

approach takes into account a criterion based on the correlation between the strain level and the occurrence of a failure. Sometimes, fission gas release can be an important issue. In any case, the objective is to determine a safety domain in which there is no fuel failure or degradation, and/or dispersion and the function of cooling the fuel are preserved.

By design, RIA experiments involve submitting limited quantities of fuel samples to representative power scenarios. For integral tests, that condition can only be met by an adequate reactivity pulse provided by a suitable reactor core, which feeds the excess neutrons into the test fuel samples.

The neutronics characteristics of that neutron source have to be such that they provide enough feedback to achieve the neutronics transient by pure physical feedback while respecting decoupling criteria acceptable for the normal operation of that source. Nonetheless, the final reactor shutdown is achieved by a reactor scram at the end of the test sequence, to ensure a sufficient shutdown margin on the long term.

Therefore, integral tests require dedicated reactor cores (pulsed cores) (see Part 8). The test conditions can also be very demanding on the test device features, especially when considering that representative coolant conditions prevail during the test.

When MTR core features are not designed to perform such neutron pulses, such pulsed experiments have to be excluded for safety reasons.

Nonetheless, from a theoretical point of view, an alternative for multi-purpose MTRs would be to simulate fast power transients by experimental device design with the core remaining at a stable core power level (no need for control rod action; an automatic preprogrammed reactor scram can terminate the test). A theoretical solution could involve fast transit (e.g. electromagnetic motion) of a small experimental sample in a guide tube throughout the core at nominal power.

In any case, achievable performances would be much more limited (separate effect experiments with access to restricted levels of power on samples which could, however, enable the reproduction of mechanisms that are active during a real RIA sequence).

Specific precautions in terms of safety issues should be carried out before envisioning any such tests, and only multipurpose research reactors with relevant design and safety provisions should be considered. In any case, parametric tests in multipurpose research reactors should be performed during a dedicated reactor cycle with no other irradiation test, to prove specific design and operation provisions.

At this moment, it seems that no practical implementation of RIA type separate effects experiments in MTRs, as described above, has been effected.

4.3. Severe accident conditions

Severe accident conditions in LWRs are multiple failure sequences beyond design basis during which the core would undergo a partial or total loss of its integrity and initial geometry. During such sequences, fuel assemblies will suffer a dramatic degradation and release a significant part of their material in airborne or particle form.

In case of severe accident sequences, the main concerns for reactor safety are core coolability and the parameters of the source term as determined by fission product behaviour and containment integrity.

Severe accident studies require integral trial runs in research reactors (e.g. quantitative studies of core melting under irradiation conditions in terms of assessing radiological releases and dissemination).

Corresponding technical requirements for the design of a related test device, insertion of such a device into a research reactor and operation of the reactor plus the device during the different phases of the test cycle (preparation, re-irradiation, accident sequence management, post-irradiation operation) impose peculiar technical and safety constraints.

In accordance with the technical and safety analyses, those constraints lead to dedicated test devices and often even to single experiment dedicated (research) reactors (see also Part 8).

5. NECESSARY PLANT PROVISIONS AND EXPERIMENTAL SUPPORTS FOR FUEL SAFETY TEST OPERATION

The test cycle must be considered as a whole, from the sample entry in the facility to the sample discharge out of the facility for further analyses after irradiation, including evacuation of the waste generated by the irradiation (up to the ultimate test device dismantling). Especially for fuel safety tests, this cannot be achieved properly without the existence of an adequate means of support.

Special prevention in terms of the risk of buildup of an unavoidable contamination level in the facility due to repeated operation of fuel safety tests must be considered by design in advance. This generic concern in terms of multipurpose research reactors is significantly reinforced by the specific constraints of fuel tests. For such purposes, hot cells have to be available. There must be at least one hot cell inside a nuclear building close to the irradiation facility, and accessible without transport constraints and time losses. Depending on the nature and number of test programmes simultaneously addressed, enhanced requirements may result in the need for a sufficient

FIG. 6. View of the SILOE MTR fission product laboratory (FP lab).

number of hot cells and/or workplaces at hot cells. Figure 6 provides an impression of such an arrangement at the SILOE MTR.

Notably, for tests involving leaking fuel sample cans or the risk of failing cans of fuel samples, the existence of a dedicated alpha hot cell, offering enough flexibility to treat different types of test devices becomes a necessity. This hot cell should be provided in addition to the other boxes or cells used for standard test device operation, in order to avoid cross-contamination effects and to enable the implementation of specific interfaces and/or tools to manage the corresponding safety tests.

The sampling and analysis aspects must also be taken into due consideration. To perform sound data acquisition during such tests, permanent laboratories with on-line measurement capacity have to be implemented. Of special interest for these tests is a fission product analysis laboratory. It must be noted that such a laboratory and the corresponding connecting/sampling lines to the test devices require specific layout and radiological protection provisions. In connection with these different topics, the connection and disconnection aspects of test device lines potentially contaminated by the experiments must be given special consideration.

Finally, the plant where the irradiation tests are taking place must be in a position to store irradiation devices and other test equipment used for these experiments, when no such test programmes are being performed, in conditions allowing adequate preservation, monitoring and easy retrieval. At the end of the life cycle of test devices or equipment, the plant must be able to retrieve, dismantle and dispose of waste equipment, including the specific equipment and/or test device used for the safety directed fuel tests.

When such aspects have not been considered in the initial design of a multipurpose research reactor plant, it is a fundamental requirement to check, when fuel tests are envisioned, that the plant effectively provides (or can effectively be upgraded to provide) sufficient auxiliary means and systems to really be in the position to cope with the needs resulting from those tests, up to final disposal of all equipment and materials.

6. INTRINSIC SAFETY ISSUES OF FUEL SAFETY TESTS

From a general point of view, a comprehensive assessment of the defence in depth philosophy of the multipurpose research reactor plant must be performed, in the context of a potential test device operation. It is a reasonable goal to assess and manage risks linked to test devices and their related experiments with the same rules, as far as applicable, and as sufficient as those used for risks linked to the intrinsic reactor process.

In this context, special consideration has to be given to fuel safety tests, since:

— The involved fuel samples (possibly leaking at the beginning or end of the experiment) represent both a radiation source and a radiological source term of their own. This is an issue both for safety and for radiological protection plus waste management since it represents an additional contribution to the radiological risks of the plant, which depend on the nature of experiments, on the design of test devices and on the design of the test device life cycle inside the plant. There exists a non-negligible risk of contamination spreading in the experimental device and beyond into the plant if no provisions are implemented to contain this risk and provide adequate decontamination, waste recuperation and confinement means. The repeated operation of fuel tests can quickly lead to lasting contamination in the plant;
— They are likely to be performed with thermal hydraulic conditions in the test device which may significantly differ from those of the hosting reactor core; therefore, without any consideration of the test

programmes, test devices may represent an energetic source term that has to be managed as a potential hazard for plant safety (e.g. protecting the core and its main safety functions in case of accidents at test devices, such as leaks or breaks);
— The most stringent of such tests may include the necessity to prevent the risk of occurrence and, when required, to manage such a risk, of energetic reactions between fuel samples and the environment, and even more, it may require management of the consequences originating from such risks.

Thus, for fuel safety tests to be performed at multipurpose research reactor plants, it must be checked whether the conditions imposed by such tests are indeed enveloped by the plant safety case. If there is no clear technical demonstration that they are enveloped by that safety case, a very cautious assessment must be performed in order that these tests would not result in increased safety risks beyond the design capabilities of the reactor plant systems and features. Furthermore and in any case, a very cautious evaluation has to take place to ensure that these tests will not give rise to any risk of accident more severe or more frequent than those taken into account in the plant safety case.

It is obvious that no tests and experiments should be envisaged when it cannot be demonstrated that the plant offers adequate design provisions to prevent and to mitigate risks and consequences linked to those tests and experiments. Extremely cautious evaluations have to be made if there exists a non-zero probability that an experiment could lead to core accident conditions or, in an extreme case, to severe damage of the core of the plant.

As these tests and experiments involve specific activities or support functions, the evaluation has to be extended to the whole testing process (e.g. experiment preparation and disposal, handling operations and potentially contaminated connecting lines).

Several intrinsic design features of the multipurpose research reactor plant in question can facilitate the establishment of acceptable safety conditions for performing fuel safety tests by:

— Implementation of a strong confinement of the plant;
— Implementation of adequate margins on reactor core design to improve reactor robustness towards potential irradiation device failure;
— Facilities to implement temporary additional protection provisions for specific experimental programmes.

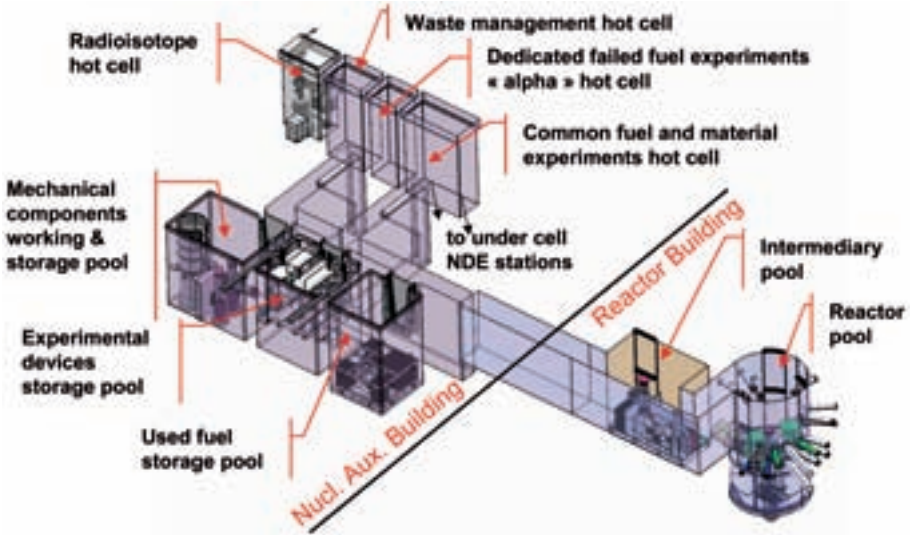

FIG. 7. Plant provisions and experimental supports for fuel safety tests at the Réacteur Jules Horowitz.

Assessment of the safety constraints of newly planned experiments can be facilitated by the existence of a generic evaluation of the test device, including its potential impact on the structure of the multipurpose research reactor plant. Figure 7 demonstrates that such conditions need to be envisaged during the early design stages of an MTR as was performed for the Réacteur Jules Horowitz.

Finally, operating rules for the reactor plant adapted to fuel test conditions and requirements provide extra guidance to help perform these tests under adequate safety conditions.

7. CONCLUSIONS

The most stringent fuel safety related tests can lead to the installation of dedicated test devices in research reactors designed specifically for safety tests and experiments.

Apart from that specific border case, multipurpose research reactors often provide flexible tools capable of hosting a very large spectrum of devices for fuel safety related tests.

The utilizable flexibility of the plant for irradiation tests strongly depends on the characteristics of the plant design, especially its support functions and its connected tools, its safety file and its rules and constraints for operation.

Ideally, those aspects have to be anticipated early in the design of new multipurpose research reactors when they are intended for fuel safety test utilization.

With regard to existing multipurpose research reactors, the enhancement of an existing capability of performing fuel safety tests should imply a careful assessment of the technical features, the operation and the safety constraints to identify the eventual need for facility upgrades.

Part 3

IRRADIATION FACILITIES FOR THE PRODUCTION OF RADIOISOTOPES

3.1. INTRODUCTION TO MEDICAL AND INDUSTRIAL ISOTOPES*

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Germany

1. INTRODUCTION

Nearly all research reactors, during the course of their lifetime, are used to produce radioisotopes. Research reactors differ widely in type, quality and quantity of radioisotope generation and in terms of the purpose for which these radioisotopes are produced. The focus of Part 3 is on those research reactors and their related facilities for which radioisotope production is a major utilization task and which contribute substantially to the market for such isotopes that are utilized to a significant extent in industry and medicine.

2. OBJECTIVES

A research reactor which is not solely generating radioisotopes for facility internal needs, such as training courses (see Part 10), or research use, but use in industrial and medical applications, must have design features and the supporting infrastructure to facilitate large scale production processes. It also must be able to cope with the constraints originating from such production on facility operation, as well as from the conditions of a market that demands quality and delivery schedules. The presence of competitors also adds to the need to successfully meet such demands on a reliable basis. Therefore, commercial applications result in constraints that may not be present with radioisotope production for research projects and, furthermore, cannot be negotiated internally to resolve conflicts or problems as they arise.

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* This paper has been prepared on the basis of an oral presentation by A. Lee, Atomic Energy of Canada Limited (AECL) at an IAEA Technical Meeting held in Vienna on 2 July 2003. The paper was revised by J. Razvi, General Atomics, USA.

3. RADIOISOTOPES UNDER CONSIDERATION

The commercial production of radioisotopes concentrates on two main areas: medicine and industrial applications. The radioisotopes for medical purposes, such as ^{99}Mo, ^{125}I and ^{131}I, ^{133}Xe and ^{89}Sr, are better known generally than those for industrial purposes, such as ^{75}Se, ^{192}Ir, ^{169}Yb and ^{60}Co. The production is dominated by ^{99}Mo in medicine and by ^{60}Co in industry. Whereas ^{60}Co, applied in large quantities in food irradiation and sterilization plants, is mostly the produced in nuclear power plants, the production of ^{99}Mo is the radioisotope which determines the operation of medium to high power research reactors (NRU, HFR, SAFARI, etc.) engaged in the commercial production of radioisotopes. At such facilities, large scale production at research reactors for commercial purposes has to be considered, as well as small scale generation for internal purposes and research projects.

4. TARGETS

All radioisotope production needs targets. A distinction must be made between production applying the fission process using fissile material as a target, and activation of a substance containing the mother isotope of the radioisotope in sufficient quantity and purity (neutron activation targets). Six examples of some of the conditions and constraints of the latter are compiled in Table 1.

Fission targets are mostly plates or small tubes rolled from plates or extruded, consisting of a 'meat' layer covered on both sides by cladding. The picture frame technique is frequently applied, as for the fuel plates of MTR type research reactor fuel. Conditions and data of fission targets of six radioisotopes are provided in Table 2.

Activation targets are far more frequent in their application, as they do not require reprocessing to separate the desired radioisotopes. All types of targets exist, from tiny amounts used in small sample carriers at low power research reactors to sophisticated gas filled ampoules or cuvettes.

5. IRRADIATION FACILITIES AND DESIGN REQUIREMENTS

All targets need facilities in order for them to be placed inside or near the research reactor core during irradiation. For low activity requirements, existing beam tubes (radial, tangential), as well as reflector thimbles are often sufficient. Their loading and discharge is performed on demand during reactor operation.

TABLE 1. NEUTRON ABSORPTION TARGETS[a]

^{99}Mo – medical	— Low specific activity from MoO_3 targets; — Specific activity directly proportional to available thermal neutron flux; — 5–15 d irradiation time at steady continuous power: • ~71% of saturation at 5 d irradiation; • Product quality degrades with decay.
^{131}I – medical	— TeO_2 targets: • Purity is important; • Exothermic reaction with Al if overheated; — Acidity directly proportional to available thermal neutron flux; — 14–28 d irradiation at steady continuous power for saturation: • 90% of saturation at 28 d.
^{125}I – medical	— Enriched ^{124}Xe target: • Enrichment requirement inversely proportional to available thermal neutron flux; • Minimum enrichment ~5 wt% of ^{124}Xe; • Cost increases exponentially with enrichment; — Up to 1 week irradiation time at steady continuous power; — Care needed in handling irradiated targets: • Volatile Xe gas and I aerosols; • Requires good vacuum and cryogenic processing; • ^{126}I impurities depend on irradiation time and flux (longer irradiation produces more ^{126}I); • Gamma ray similar to ^{137}Cs (orders of magnitude higher activity); • Activated capsules; • I extraction requires high temperature (~650°C); • High dose and contamination hazards.
^{192}Ir – industrial	— Requires high flux ($>2 \times 10^{14}$ n·cm^{-2}·s^{-1}); — Very strong neutron absorption; — Long irradiation times (several months); — High level wastes generated.
^{60}Co – industrial	— Requires $>5 \times 10^{13}$ n·cm^{-2}·s^{-1} for large sources; — Very strong neutron absorption; — Large quantity in target; — Long irradiation times (5–7 a or more); — High level wastes generated.
^{60}Co – medical	— Requires high flux ($>2 \times 10^{14}$ n·cm^{-2}·s^{-1}); — High specific activity needed; — Small quantity in target; — Very strong neutron absorption; — Long irradiation times (2–3 a or more); — High level wastes generated.

[a] Table courtesy of A. Lee, AECL.

TABLE 2. FISSION PRODUCT TARGETS[a]

^{99}Mo – medical	— Typical 2–4 kW/g ^{235}U specific power for large scale production; — Typical 1–2 kW/g ^{235}U specific power for small scale production; — 10–15 d irradiation time at steady power: • Optimum is 13–14 d; • ~71% of saturation at 5 d irradiation; • Need to account for decay during processing and delivery; • Need to account for recovery efficiency; • 'Just in time' delivery; • 23% decay loss in 24 h; • ~78% decay loss in 6 d; • Schedule matched to customer requirements.
^{133}Xe – medical	— Produced along with ^{99}Mo; — Requires at least 7 d decay after processing to decay impurities: • 60% of product lost in decay; • Shielded storage needed; • Storage capacity depends on customer schedule.
^{131}I – medical	— Produced along with ^{99}Mo; — Not available to be extracted from all target forms; — Requires at least 12 d decay after processing to decay impurities: • 65% of product lost in decay; • Shielded storage needed; • Storage capacity depends on customer schedule.

[a] Table courtesy of A. Lee, AECL.

For high activity requirements, specially designed facilities exist in an enormous variety for manual, semi-automatic and fully automated loading. The design must satisfy many requirements and constraints. Typical issues to be considered during the design are:

— Adequate heat removal, as target samples, as well as sample containers are heated by the absorption of neutrons and gamma rays plus — in the case of fission targets — by the fission energy. This includes consideration of the specific heat of the sample itself and of its container. Sophisticated considerations of natural convection, bypass cooling or forced flow cooling may be necessary.
— Available excess reactivity during the entire fuel cycle of the research reactor, which often determines the number of facilities for radioisotope generation and their production frequency at a given plant.

— Reactivity effects during loading and discharge, as net reactivity changes but also as reactivity changes per unit time, which the research reactor reactivity control system has been able to compensate.
— Flux perturbations caused by a target sample containing neutron absorbing material (activation target) depresses the flux locally and may even give rise to inclined flux distributions across the core.
— Flux perturbations caused by a target sample containing fissile material that enhances the flux locally and may lead to local power peaks at the nearby fuel assemblies. Also, new local sources of gamma rays may have to be considered.
— Careful consideration of sample container integrity under all impacts and stresses, to avoid inadvertent contamination.
— Adequate shielding during target handling.
— Disturbances to the operational parameters of the research reactor, such as both local and general power and flux levels, control rod positions and their impact on local neutron fluxes.
— Reactor operation requirements, such as long periods of stable power to obtain the desired specific activities for the target material under consideration. Often the 'just in time' delivery issue is a high priority.
— Careful consideration of the relevant facility infrastructure necessary to support all preparation, handling and supervision under the highest safety standards, as well as sufficient and experienced staff. Such infrastructure may include hot cells, gloveboxes, shielded handling flasks, lifting devices, remote welding, remote chemistry, etc.
— The waste issue and potentially higher personnel dose rates are two of the most serious obstacles to overcome when considering large scale radioisotope production.

Not all the named constraints and requirements are directly linked to the design and construction of a facility for radioisotope production. However, all must be carefully considered when developing or modifying the application of a research reactor for radioisotope production. They will also be included in the safety considerations linked to a new radioisotope facility.

6. COMMERCIAL REQUIREMENTS

In addition to the technical and licensing requirements described above, a research reactor operator interested in commercial ventures must consider product quality and product purity requirements. Those demands, however, are

TABLE 3. COMMERCIAL REQUIREMENTS[a]

^{99}Mo – medical ($T_{1/2}$ = 65.94 h)	— Adjust to customer demand within 3–5 d notice; — Delivery at same time of day; — Delivery on same days of the week.
^{133}Xe – medical ($T_{1/2}$ = 5.24 d)	— Adjust to customer demand within 1 week's notice; — Delivery at same time of day; — Delivery on same days of the week.
^{131}I – medical ($T_{1/2}$ = 8.02 d)	— Adjust to customer demand within 1 week's notice; — Delivery at same time of day; — Delivery on same days of the week.
^{125}I – industrial ($T_{1/2}$ = 59.4 d)	— Adjust to customer demand within 1 week's notice; — Delivery at same time of day; — Delivery on same days of the week.
^{60}Co – medical ($T_{1/2}$ = 5.27 a)	— Adjust to customer demand within 6 months' notice; — Delivery on schedule in 1–3 month intervals.
^{192}Ir – industrial ($T_{1/2}$ = 73.8 d)	— Adjust to customer demand within 6 months' notice; — Delivery on schedule in 1–3 month intervals.

[a] Table courtesy of A. Lee, AECL.

outside the irradiation facility as such. Nevertheless, Table 3 lists the commercial requirements, data and standards for six different radioisotopes.

7. EXAMPLES OF RADIOISOTOPE PRODUCING RESEARCH REACTORS

Some prominent examples of research reactors involved in the commercial production of radioisotopes are provided in Table 4. These facilities are located all over the world. It must be pointed out that there are other research reactor plants with specific facilities generating specific isotopes, such as Pu isotopes, ^{3}H or transuranic elements, which are not included for discussion here.

TABLE 4. EXAMPLES OF RESEARCH REACTORS PRODUCING RADIOISOTOPES

NRU – Chalk River, Canada	— 135 MW; — Continuous operation for as long as practicable, typically 3 weeks; — Relatively short shutdown times, 4 d max., for maintenance; — ^{99}Mo, ^{133}Xe, ^{192}Ir, ^{60}Co medical, ^{131}I, ^{125}I.
HFR – Petten, Netherlands	— 45 MW; — Continuous operation for 28 d; — Requires backup ^{99}Mo production during refuelling outage; — ^{99}Mo, ^{133}Xe.
R-2 – Studsvik, Sweden	— 50 MW; — Continuous operation for 28 d; — Requires backup ^{99}Mo production during refuelling outage; — ^{99}Mo, ^{133}Xe; — ^{192}Ir.
SAFARI-1 – Pelindaba, South Africa	— 20 MW; — ^{99}Mo; — ^{125}I.
MNR – Hamilton, Canada	— Adjust to customer demand with 6 months' notice; — Delivery on schedule in 1–3 month intervals.

3.2. RADIOISOTOPE PRODUCTION AT THE HIGH FLUX REACTOR AT PETTEN

D. Bergmans
High Flux Reactor,
Petten, Netherlands

1. HISTORY OF THE HIGH FLUX REACTOR

A combined initiative of the Dutch administration and industry led to the construction of the High Flux Reactor (HFR) in Petten, Netherlands. The design was based on the Oak Ridge research reactor in the United States of America. Construction was contracted to the American Car and Foundry, Inc. Work commenced 27 August 1957 and ended with the HFR's first criticality on 11 November 1961. After a successful 'hot commissioning' phase, the research reactor was transferred to the European Committee, contributing to the Euratom mission and to the Joint Research Centre (JRC) research programmes in particular. Since then, the research reactor has been operated by the Nuclear Research and Consultancy Group (NRG) (in the past, the Reactor Centre Nederland and the Netherlands Energy Research Foundation).

In the ensuing 40 years, the research reactor and its equipment have been modernized continuously. The focus of the research programme changed from pure material and fundamental physics research via applied research for, among other things, gas cooled reactors and fast breeders, as well as PWR and BWR power plants; into the real multipurpose research reactor of today.

2. WORKLOAD AT THE HFR

Today, the HFR can be considered as a multipurpose neutron source for:

— Fissile material testing;
— Structural material testing;
— Solid state physics;
— Activation analysis;
— Training and education;
— Radioisotope production.

During the last decade, the importance of radioisotope production increased strongly and was directly connected to the isotope computed tomography (ICT) developments in medical diagnosis. Availability, punctually executed timetables, and fair and reproducible fluxes put the HFR in an outstanding position with regard to (nuclear) medical health care. At the present time, approximately 45% of the HFR capacity supports isotope production. NRG supplies 20–25% of the total world radioisotope market.

3. TECHNICAL DATA OF THE HFR

The HFR is a light water cooled and moderated tank-in-pool type research reactor. The normal operating thermal power is 45 MW. The core lattice is a 9 × 9 array containing 33 fuel assemblies, 6 control assemblies, 23 beryllium reflector elements and 9 in-core experimental positions. A row of 9 beryllium reflector elements is arranged inside a rectangular reactor vessel located on the east, exterior side of the core box. A unique pool side facility (PSF), offering 12 experimental positions, is located on the west, exterior side of the reactor vessel. That PSF (see Fig. 1) is perfectly suited for dynamic fuel tests

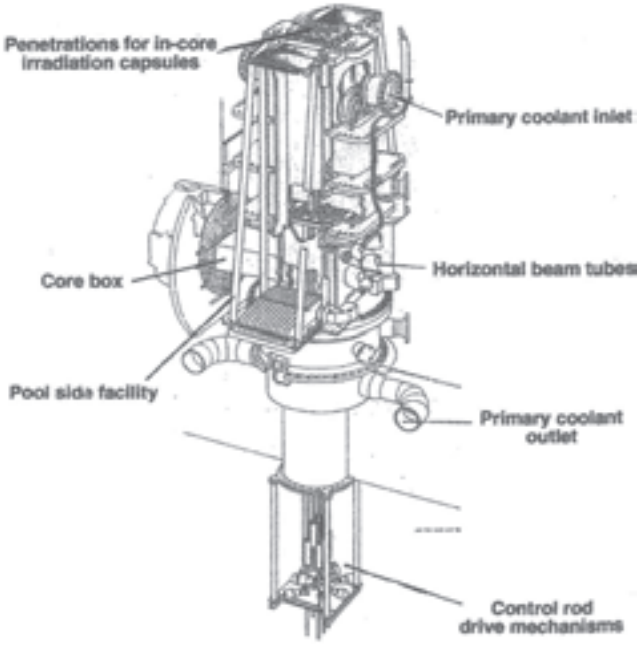

FIG. 1. The HFR interior with the core box and supporting components, in particular, the pool side facility.

as well as for radioisotope production. During power operation, loading and unloading can be performed quickly, safely and easily. The operating schedule of the HFR consists of 24.7 d of operation followed by a 3.3 d shutdown for maintenance, refuelling and rearrangement of experiments. In this scheme, the HFR has two extended shutdown periods: one for major maintenance procedures and modifications, and a second one which combines staff training with the major staff holiday period. Those periods have a maximum duration of four weeks each. In total, the HFR is operated at more than 280 full power days per year. For supplementary information see Table 1 and Fig. 2.

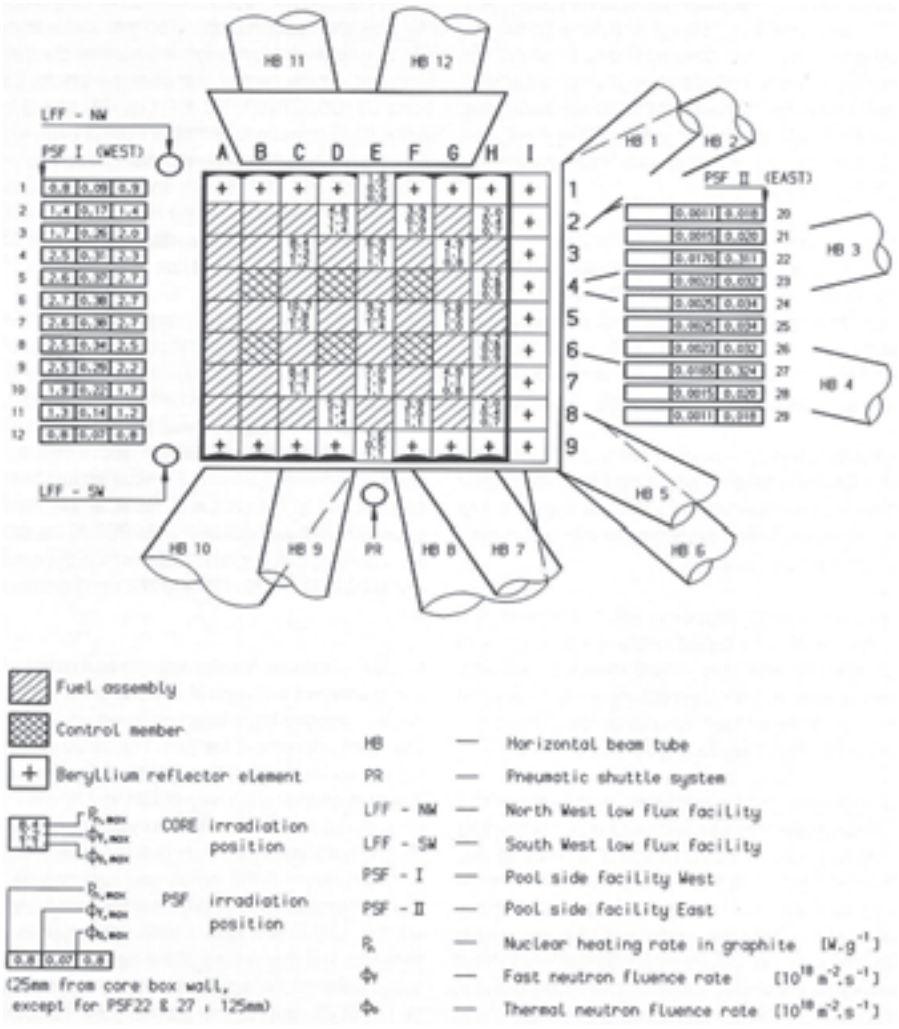

FIG. 2. HFR core cross-section with positioning of the various irradiation devices.

TABLE 1. MAIN PARAMETERS OF THE HFR PLANT AND OPERATION, INCLUDING THE BOUNDARY CONDITIONS FOR IRRADIATION DEVICES

Reactor thermal power		45 MW
Specific thermal power (averaged over active core volume)		310 MW/m^3
Number of fuel assemblies (FAs)		33
Number of control assemblies		6
Number of in-core irradiation positions		9
Number of reflector irradiation positions		10
Number of horizontal beam tubes		12
Number of pool side facility (PSF) positions		22
Uranium loading of fresh fuel assembly		450 g U-235
Uranium loading of fresh control assembly		310 g U-235
Boron charge in side plates of fresh FAs		1000 mg B-10 in Al
Total fissile uranium charge per fresh core		12 kg U-235
Total core volume		0.2 m^3
Average thermal flux at an in-core FA position		1.0×10^{15} $n \cdot cm^{-2} \cdot s^{-1}$
Maximum thermal flux at an in-core FA position		1.6×10^{15} $n \cdot cm^{-2} \cdot s^{-1}$
Maximum fast flux at an in-core irradiation position		2.5×10^{15} $n \cdot cm^{-2} \cdot s^{-1}$
Maximum fast flux at PSF I		3.8×10^{14} $n \cdot cm^{-2} \cdot s^{-1}$
Maximum fast flux at PSF II		1.7×10^{13} $n \cdot cm^{-2} \cdot s^{-1}$
Maximum thermal flux at an in-core irradiation position		1.4×10^{15} $n \cdot cm^{-2} \cdot s^{-1}$
Maximum thermal flux at PSF I		2.7×10^{15} $n \cdot cm^{-2} \cdot s^{-1}$
Maximum thermal flux at PSF II		3.2×10^{14} $n \cdot cm^{-2} \cdot s^{-1}$
Radiation heating in graphite:	in-core position	6–12 W/g
	reflector position	2–6 W/g
	PSF I	<3 W/g
Volume flow of primary coolant through the core	(4100 m^3/h)	1.14 m^3/s
Coolant velocity inside an FA		7 m/s
Coolant velocity inside a dummy element		0.2–7 m/s
Primary coolant inlet temperature	(45°C)	318 K
Primary coolant outlet temperature	(55°C)	328 K
Coolant temperature difference across core		10 K
Average heat flux at an FA in central position	(100 W/cm^2)	1.00 MW/m^2
Maximum heat flux at an FA in central position	(160 W/cm^2)	1.60 MW/m^2
Absolute pressure at top of core	(3.4 bar)	0.34 MPa
Pressure difference across core	(1.1 bar)	0.11 MPa

4. IRRADIATION DEVICES FOR RADIOISOTOPE PRODUCTION

In a multipurpose research reactor, it is recommended to standardize the experimental facilities as much as possible. One reason for such a philosophy is to achieve short loading and unloading times. These reductions of handling times also have a positive effect on health physics aspects, especially on the collected personnel doses. This issue is valid also for radioisotope production facilities.

Many of the in-core irradiations are performed using standardized reloadable irradiation capsules. Four principal inserts (see Fig. 3) offer irradiation channels of different sizes. If necessary, in-pile instrumentation may be connected to out-of-pile measurement equipment through gastight standard heads. For radioisotope generation, the reloadable isotope facility (RIF) and the high flux facility for isotopes (HIFI) devices are inserted at such in-core subchannels.

In addition to the core region, PSF I is used for positioning various irradiation devices into the neutron field outside the core and the core tank. The majority in number and the most space consuming facilities are the rigs for the production of the radioisotope ^{99}Mo. The HFR plant possesses several custom-built facilities for in-core irradiation as well as for insertion in the PSFs. These irradiation devices are designed for the shape of the fissile ^{99}Mo targets, which are either tubular or plate shaped. Special devices have also been designed for cobalt production (COBI, CORRI). Additionally, a rotary device (PROF) and two conveyor systems (FASY and PRS) support isotope production. For comprehensive information on the devices and on the radioisotope services, see the JRC publication EUR 15151 EN.[1]

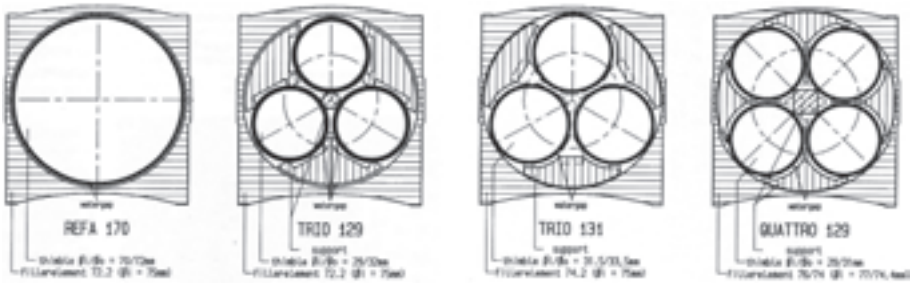

FIG. 3. The four main configurations for in-core irradiation channels at the HFR.

[1] AHLF, I., ZURITA, A. (Eds), "Characteristics of the installation and the irradiation facilities", High Flux Reactor (HFR) Petten, Joint Research Centre, Institute for Advanced Materials – Petten Site, Rep. EUR 15151 EN, Commission of the European Communities, Luxembourg (1993).

TABLE 2. FLUXES ALONG THE CENTRE LINE OF THE RIF DEVICE AT C7 (HFR AT 45 MW)

Capsule height No.	$\Phi_{thermal}$ (n·cm^{-2}·s^{-1})	Φ_{fast} (n·cm^{-2}·s^{-1})
1	1.8×10^{14}	1.5×10^{14}
2	2.0×10^{14}	1.6×10^{14}
3	2.1×10^{14}	1.7×10^{14}
4	2.1×10^{14}	1.7×10^{14}
5	2.1×10^{14}	1.7×10^{14}
6	1.3×10^{14}	1.5×10^{14}

4.1. Reloadable isotope facility

The reloadable isotope facility (RIF) is a simple capsule holder which is placed in a beryllium reflector assembly. It is made of aluminium with four cylindrical holes, two for accommodating 10 mm diameter capsules and two for 15 mm diameter capsules, with a total usable height of 250 mm in each hole. The normal capsule external length is 50 mm. The capsules are cooled with circulating water at about 40°C. The device is shown in Fig. 4.

The irradiation positions are located near the reactor midplane, in a relatively flat part of the vertical flux distribution. The irradiation time is usually a multiple of 24 h. Neutron flux values are given in Table 2.

The standard design of the RIF was supplemented later by a high flux version of the RIF making available more space for capsule holders. Its new position in the core grid is D8, with a maximum radiation heating of 5.3 W/g.

4.2. High flux facility for isotopes

In order to meet the constant demand for higher neutron fluxes at the positions of radioisotope facilities and for enhancing the possibility to irradiate ^{192}Ir in the HFR, a special HIFI device (see Fig. 5) was developed in the 1990s, maximizing reliance on the moderator (light water) and minimizing the use of structural material. By the principle of flux trap technique, a considerable increase in the specific activity of iridium irradiated within HIFI was achieved.

The HIFI device is completely manufactured from aluminium and is placed into a filler element of 72.2 mm diameter. The design of this device is such that five thin aluminium tubes are fixed around a central spine. Each tube can receive five capsules with a standard length of 80 mm each and diameters of

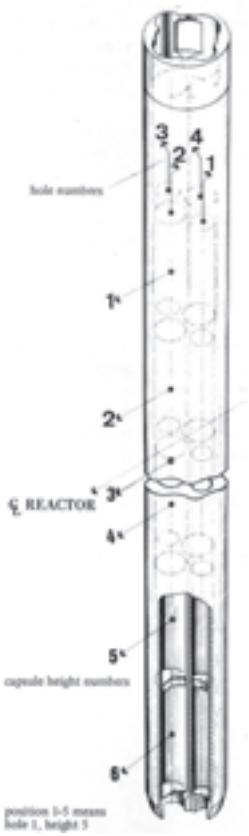

FIG. 4. RIF device.

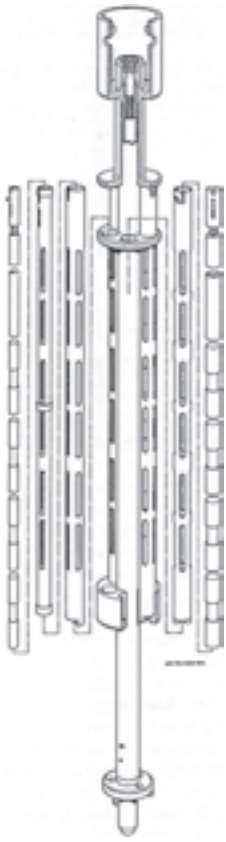

FIG. 5. HIFI device.

10–15 mm. Long vertical holes are machined into the vertical tubes to maximize the water around the capsules.

A movable cover holds the capsules in position. The capsule holder has to be removed with a standard filler tool. Positioning in the filler element is performed by a stop pin which fits into the lower part of the filler element. The distribution of the apertures in the base of the filler element aligns three positions of the holder with the core. For thermal and fast neutron flux values, see Table 3. The maximum radiation heating in the HIFI device is 5.3 W/g at grid position D8.

4.3. High flux pool side isotope facility

The high flux pool side isotope facility (HFPIF) can be installed on any rail of the PSF I table (see Fig. 2). In principle, the HFPIF consists of two tubes

TABLE 3. NON-PERTURBED MAXIMUM FLUXES IN THE HIFI DEVICE AT D8 (HFR AT 45 MW)

Tube No.	$\Phi_{thermal}$ (n·cm^{-2}·s^{-1})	Φ_{fast} (n·cm^{-2}·s^{-1})
1	4.3×10^{14}	1.4×10^{14}
2	4.5×10^{14}	1.7×10^{14}
3	4.4×10^{14}	1.7×10^{14}
4	4.3×10^{14}	1.4×10^{14}
5	4.2×10^{14}	1.1×10^{14}

TABLE 4. FLUXES IN ROW 1 OF THE HFPIF DEVICE ON RAIL 5 OF THE PSF (HFR AT 45 MW)

Capsule No.	$\Phi_{thermal}$ (n·cm^{-2}·s^{-1})	Φ_{fast} (n·cm^{-2}·s^{-1})
Row 1, capsule 1	2.1×10^{14}	2.9×10^{13}
Row 1, capsule 2	2.4×10^{14}	3.4×10^{13}
Row 1, capsule 3	2.5×10^{14}	3.5×10^{13}
Row 1, capsule 4	2.5×10^{14}	3.5×10^{13}
Row 1, capsule 5	1.9×10^{14}	2.3×10^{13}

arranged behind each other. Samples are loaded using aluminium capsules with a 25 mm external diameter and 80–160 mm length. The facility can be loaded and unloaded during reactor operation. Normally, reloading takes place once a day. Measured values of thermal and fast flux in the tube adjacent to the core box wall are given in Table 4. The device is shown in Fig. 6.

FIG. 6. HFPIF device.

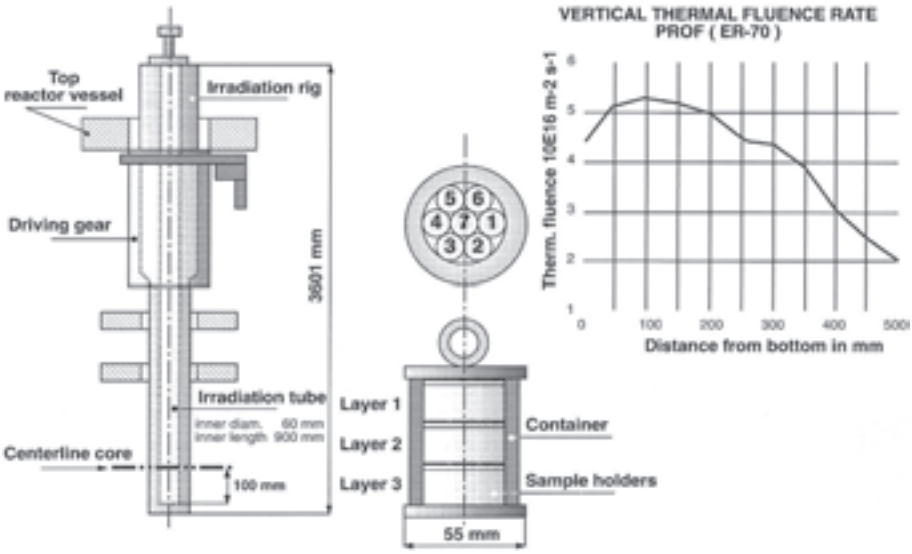

FIG. 7. The HFR device PROF.

4.4. Pool side rotating facility

The pool side rotating facility consists of a driving gear and an irradiation rig and is installed near the reactor core in the vertical irradiation position LFF-NW (see Fig. 2). To flatten the radial neutron flux, the rig rotates around the vertical axis at 1 rev/min.

The irradiation tube is situated in the lower end of the rig. A shield of lead surrounds the tube to reduce the gamma heat. The irradiation tube has an internal diameter of 60 mm and a length of 900 mm. Probes are placed in sample holders fitted into a polythene container with an outer diameter of 55 mm. A total of 21 sample holders (with a volume of 1000 mm^3 each) can be loaded. Several containers can be placed into the irradiation tube. The irradiation times range from 30 min to 24 h. The irradiation is performed under N_2 atmosphere. Schematic impressions of the facility and a container, as well as the flux distribution above the bottom of the irradiation tube, are compiled in Fig. 7.

4.5. Pneumatic shuttle systems FASY and PRS

Two pneumatic conveyor systems are operated at the HFR: the fast pneumatic shuttle system (FASY) and the pneumatic shuttle system (PRS). Radioisotope generation in the probes of both systems principally supports

TABLE 5. FLUX VALUES AND GEOMETRICAL DATA OF THE STANDARD RADIOISOTOPE DEVICES

Device	$\Phi_{thermal}$ ($n \cdot cm^{-2} \cdot s^{-1}$)	Φ_{fast} ($n \cdot cm^{-2} \cdot s^{-1}$)	$\varnothing_{inner}$ (mm)	Inner length (mm)	$\varnothing_{outer}$ (mm)	Outer length (mm)	Irradiation time
RIF	2.1×10^{14}	1.7×10^{14}	9 or 14	70	10 or 15	80	Multiple of 25 d
HFPIF	2.5×10^{14}	3.5×10^{13}	9 or 14	45 or 90 or 145	10 or 15	50 or 100 or 150	As wished
HIFI	4.5×10^{14}	1.7×10^{14}	23	70	25	80	As wished
PRS	4.0×10^{13}	2.3×10^{12}	15	70	22	100	Max. 1 h
FASY-1	8.1×10^{13}	5.4×10^{12}	10	19	12	20	15 min
FASY-2	1.3×10^{12}	2.0×10^{11}	10	19	12	20	15 min

neutron activation analysis. The relevant data of the systems are contained in Table 5, together with the data of the standard radioisotope devices.

4.6. Fissile isotope target device

Tumour diagnosis has made increasing use of the short lived radioisotope ^{99m}Tc produced from the radioactive decay of ^{99}Mo. FIT (Fig. 8) was designed to enable the production of ^{99}Mo using fission.

Targets for the in-core FIT irradiation device consist of MTR type plates. The plates are manufactured using the picture frame technique and are subse-

FIG. 8. FIT device.

quently transformed by bending and welding into small cylinders of 22/19.5 mm diameter and 160 mm height. Up to three cylinders are threaded onto a central location pin with a centring plug. The cylinders may also be loaded and centred via a modified head of the TRIO capsule in an aluminium tube of 3314 mm length and 27 mm inner diameter. The tube is then loaded into a standard TRIO device.

Cooling of the targets is provided by a primary water system. Coolant enters the tube through two longitudinal slots, machined in the upper stainless steel section of the tube above the target positions. The water flows downwards through an outer and inner channel, formed by the tube plus the targets and by the targets plus the central location pin, respectively. The coolant exits the tube through a hole in the lower end cap.

The FIT construction enables loading and discharging of the targets during reactor operation, i.e. without stopping the cooling loop of the HFR. The maximum perturbed neutron flux is dependent on the core position selected for device insertion. A typical radially averaged perturbed (measured) thermal flux is 6×10^{13} n·cm^{-2}·s^{-1} (H4 position at the grid).

4.7. Medical use of the molybdenum device

As an alternative to the FIT device, another device has been designed to irradiate fissile targets. Medical use of the molybdenum (MEDUSE) device uses targets in the form of flat plates and irradiates them at the PSF in order to generate ^{99}Mo. MEDUSE consists of a standard PSF carrier which can be loaded in any PSF position, modified in the front to allow insertion of the sample holder.

The sample holder is constructed of aluminium and is welded to the carrier on the front side. Inside, a movable aluminium holder keeps the fuel plates in place. They are arranged in three columns, with two plates per column, comprising a total of up to six plates in a holder. The fuel plates are spaced to allow for the passage of coolant through them. The standard irradiation period per charge is 5 d. Thereafter the rig is transported to the dismantling cell where the targets are removed from the sample holder and fresh targets are loaded.

4.8. Molybdenum devices MYKONOS and INCOMODO

The device for plate type targets consists of one vertical hole which accommodates eight target plates in two stacks of four each (see Fig. 9, the PSF device for plate type targets). The tubular device is based on three vertical holes and accommodates three stacks with three cylindrical targets each (see Fig. 10, INCOMODO, the in-core device for tubular targets). All devices are

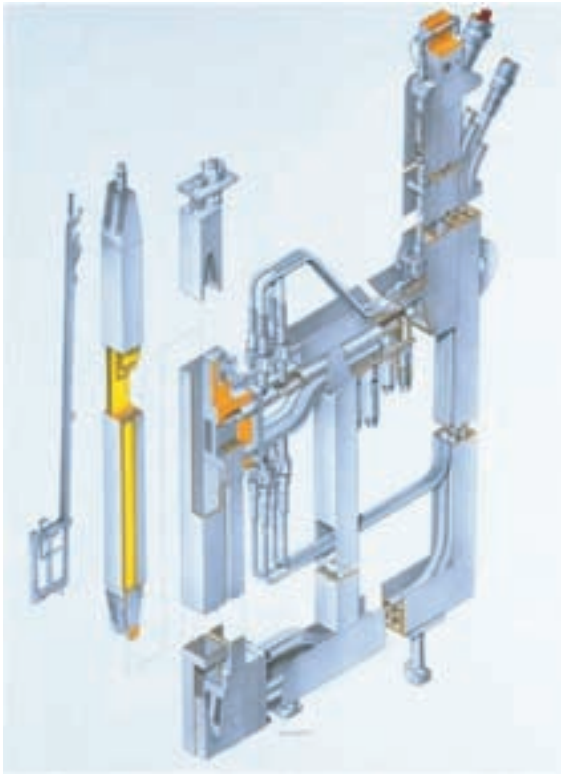

FIG. 9. MYKONOS, the actual pool side facility device for plate type targets applied for radioisotope production.

cooled with water supplied by special cooling circuits. The cooling water is extracted from the pool cooling system. The in-core device design allows loading and unloading during reactor operation. A standard ^{99}Mo irradiation sequence consumes 150 h of reactor operation time. However, due to fluctuations in the operational demands during an operation cycle, frequent changes in the loading pattern of the core and of the PSF are common at the HFR. To cope with these commercial requirements, extra efforts are required of the reactor operators group.

4.9. Cobalt isotope production and cobalt reflector irradiation

The cobalt isotope production (COBI) device (Fig. 11) is used to irradiate small cobalt chips (0.93 mm × 18 mm × 50 mm). COBI accommodates up to 120 cobalt chips divided over eight simple baskets stacked in an aluminium tube. Each chip is encased in a small argon arc-welded stainless steel box filled with

FIG. 10. INCOMODO, the pool side facility device for tubular targets applied for radioisotope production.

helium. The baskets themselves consist of two concentric aluminium tubes of 59 mm length.

Fifteen encased chips are radially arranged in the formed circular volume by putting them into small grooves on the inner and outer surfaces of the short tubes. A stainless steel spine runs through the central holes of the baskets. It is connected to a bottom plug welded into the long aluminium tube. A long nut with a lifting hook is screwed onto the top of the spine and secured with a split pin. HFR primary cooling water flowing through the facility cools the numerous components. A ring with vertical holes is welded around the lifting hook to restrict the flow inside the capsule.

The COBI device has no instrumentation and requires no external installations. It is placed into a filler element with a bore of 72 mm.

The cobalt reflector irradiation (CORRI) device is similar to COBI and facilitates the irradiation of 48 cobalt strips. The outer diameter of CORRI is 50 mm. CORRI is loaded into an in-core reflector position, whereas COBI can

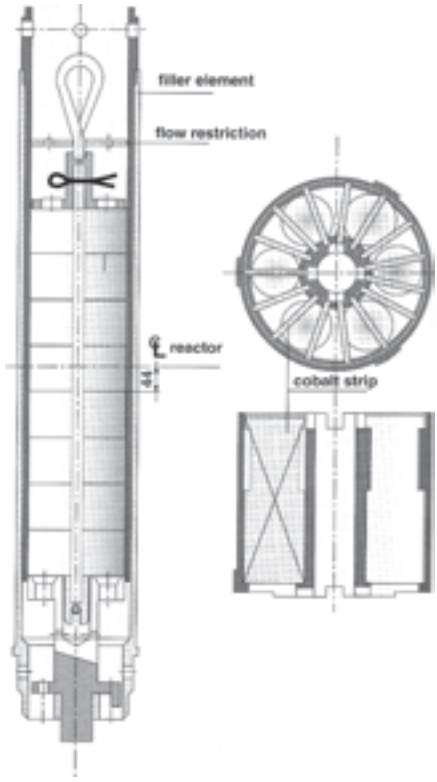

FIG. 11. The COBI device.

be used in practically all in-core positions. Due to often conflicting commercial requirements, special efforts are required of the reactor operator group.

5. OUTLOOK

In 2005, the HFR is due to start its HEU to LEU conversion process. The geometry of the core and PSF will not be affected and so, in principle, the design base of all devices and rigs will remain. From detailed MCNP calculations and simulations, it is assumed that the changes in the neutron spectra at the irradiation positions and the spatial neutron flux distribution will be marginal.

To improve communications with the competent authorities and to ensure that all safety, operational and, in particular, commercial requirements are satisfied, the new operation licence for the HFR has been transferred from the owner, European Community – JRC (EC–JRC), to the operating organization, NRG.

3.3. RADIOISOTOPE PRODUCTION FACILITIES IN ARGENTINA

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1. INTRODUCTION AND HISTORY

Radioisotopes and ionizing radiation are currently used routinely by many sectors of the economy, such as medicine, industry, agriculture and R&D, having been incorporated into numerous daily activities and contributing significantly to the improvement of the quality of life for the population. The use of radioisotopes and ionizing radiation contributes to medical investigation, prognosis, diagnosis and therapy, as well as to the industrial competitiveness and development of advanced technologies.

Most applications of radioisotopes and ionizing radiation are clearly established, peaceful uses. New applications are continuously appearing as a result of the progress of science and technology, generating new needs and requiring the resolution of new problems.

Activities related to the applications of radioisotopes in Argentina began about the same time that the National Atomic Energy Commission (CNEA) was founded in 1950.

In 1954, with the installation of a particle accelerator (synchrocyclotron) in Buenos Aires, Argentina [1], CNEA started its activities on radioisotope production. During the next years, 20 different new radioisotopes were discovered using this accelerator.

In 1958, the RA-1 research reactor (Argonaut type) was designed and built at the Constituyentes Atomic Center. Providing a maximum power of 140 kW (maximum thermal neutron flux 1×10^{12} n·cm^{-2}·s^{-1}), this research reactor enabled the local production of some of the radioisotopes used in nuclear medicine at that time, such as ^{131}I, ^{24}Na, ^{42}K and ^{198}Au.

In 1967, the RA-3 research and production reactor was put into operation at the Ezeiza Atomic Center [2], 40 km from Buenos Aires, with a maximum power of 5 MW and a maximum thermal neutron flux of 4×10^{14} n·cm^{-2}·s^{-1}. The RA-3 consolidated the position of CNEA as a radioisotope producer, covering most domestic needs and generating quantities exportable to other countries of the region.

Finally, the installation of a Radioisotopes Production Facility in 1971 (which is mainly a facility for processing irradiated samples to generate

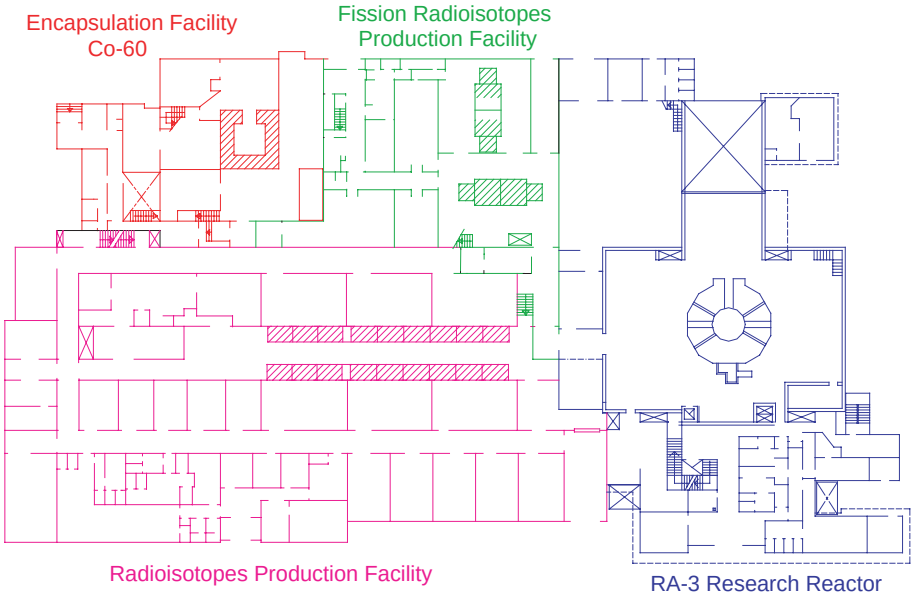

FIG. 1. Production facilities at the Ezeiza Atomic Center.

radioisotopes) [3], of a ^{60}Co Encapsulation Facility in 1978, of a ^{99}Mo Production Facility in 1985 and of a cyclotron in 1994 — all of them at the Ezeiza Atomic Center — complemented and completed a comprehensive infrastructure supporting the production of radioisotopes for applications in medicine and industry (see Fig. 1).

Starting in 1958, CNEA has been developing human resources, not only for its own activities, but also to enable the transfer of relevant technologies.

2. IRRADIATION FACILITIES AT THE RA-3 RESEARCH REACTOR

The RA-3 research reactor (see Figs 2–4) has been designed as an experimental facility to facilitate basic research experiments and technological applications, mainly on nuclear engineering, but for the production of radioisotopes as well. Its principal characteristics are:

— Startup/refurbishing: 1967 and 1987–1990;
— Type: pool type (pool diameter: 3.3 m, pool height: 12 m);
— Thermal power: 10 MW;
— Maximum thermal flux: 2.4×10^{14} n·cm^{-2}·s^{-1};

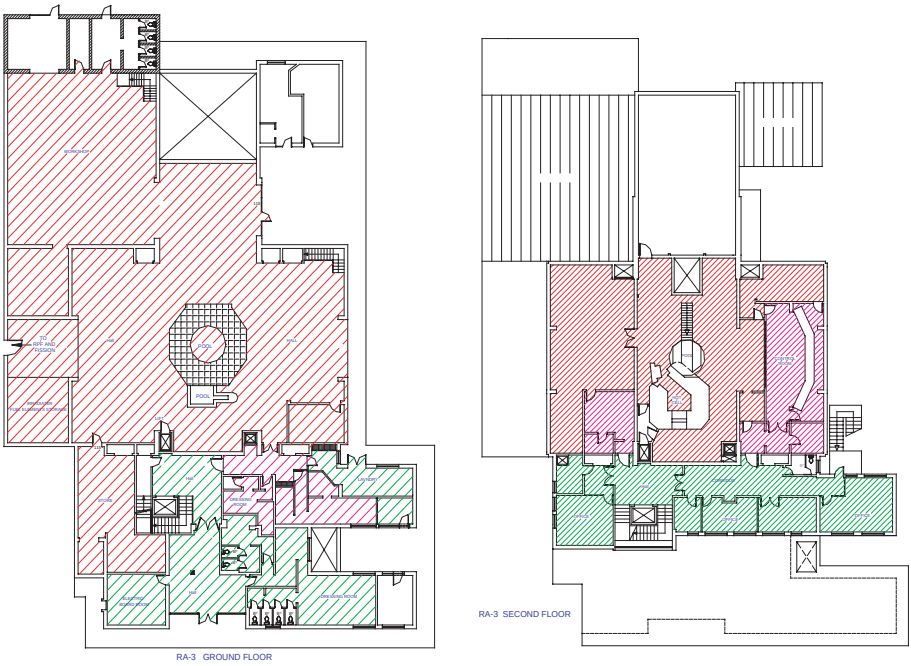

FIG. 2. RA-3 research reactor, section of ground floor and second floor.

FIG. 3. RA-3 research reactor, operations hall and core.

— Fuel assembly type: MTR, 290 g ^{235}U, 20% enriched;
— Moderator and coolant: light water;
— Heat removal: forced circulation;
— Number of standard fuel assemblies (FAs): 23;

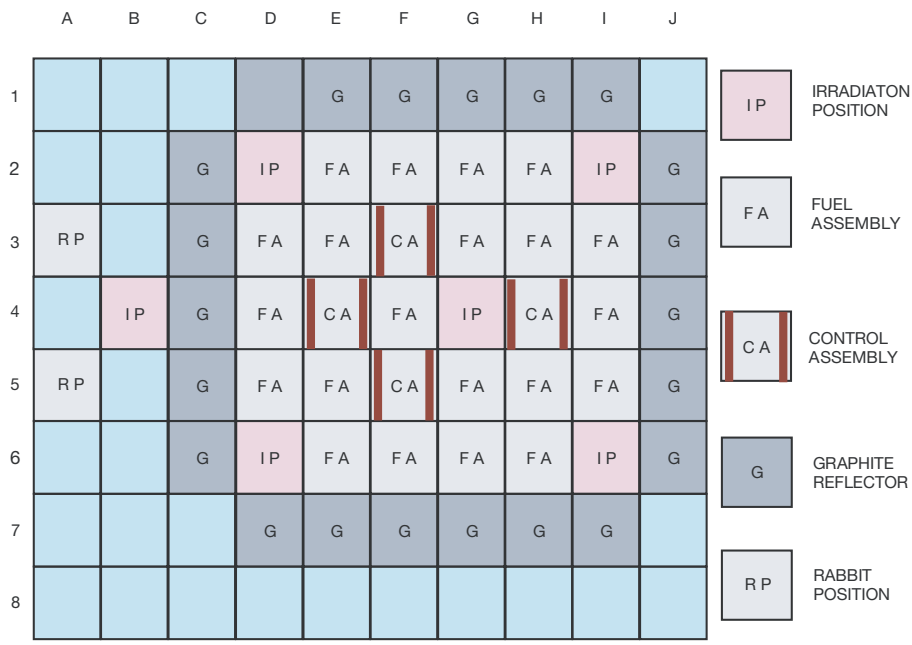

FIG. 4. RA-3 irradiation grid.

— Number of FAs with control rods (CAs): 4 (Ag–In–Cd absorbers);
— Number of irradiation rigs: 6 (92 positions);
— Pneumatic conveyor: 2 positions;
— Thermal column: 1 (15 irradiation positions);
— Operation cycle: 120 h/week; 45 weeks/a.

2.1. Irradiation facilities

The main devices for experimentation are:

— Irradiation positions in the core and the grid area as:
  • Irradiation rigs;
  • Pneumatic systems;
  • Test loops;
— Thermal column;
— Radial beam tubes for n-beam experiments;
— Radial tube for neutron radiography;
— Open tangential tube.

Of the devices listed above, those used for the generation of radioisotopes are briefly described in the following sections.

2.1.1. Irradiation rigs

The irradiation rigs (see Fig. 5) house the targets to be irradiated for the production of radioisotopes. At the same time, these rigs are used for transport between their irradiation positions and the hot cell, where the targets are loaded and discharged. The irradiation rigs are inserted at different irradiation positions in-core or adjacent to the core border.

A special rig is used for irradiation of the uranium targets for fission ^{99}Mo production.

The different irradiation positions selected for the irradiation processes are as follows:

— Central trap:
- Position: G-4;
- Use: for fission ^{99}Mo and ^{92}Ir production;
- Capacity: 12 targets with plate type geometry (uranium miniplates for fission ^{99}Mo production or samples for ^{192}Ir production);
- Maximum load of ^{235}U: 16.8 g;
- Δ_ρ: ≤800 pcm;

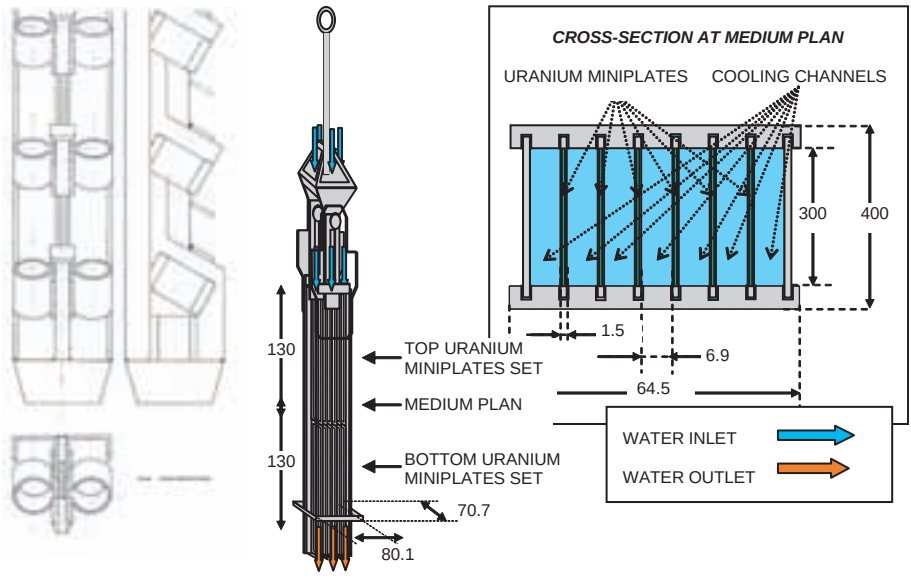

FIG. 5. Irradiation rigs for manual loading at the RA-3 (for capsules and miniplates).

- Thermal neutron flux at 10 MW operation: 2.1–2.4 × 10^{14} n·cm^{-2}·s^{-1};
- λ (Φ_{epi} /Φ_{th}): 0.04;
- Type of experiment: stationary (with the reactor shut down for loading/discharging);
- Applicable operation procedure: the irradiation rig is loaded into or removed from G-4 while the reactor is shut down. The ^{235}U load is limited by the maximum excess core reactivity;

— Internal rig (four, inside the active core) (see Fig. 5):
- Positions: D-2, D-6, I-2, I-6;
- Use: for the production of radioisotopes in general;
- Capacity per rig: 16 targets (canned by aluminium capsules);
- Δ_ρ: ≤40 pcm;
- Thermal neutron flux at 10 MW operation: 4 × 10^{13}–1.2 × 10^{14} n·cm^{-2}·s^{-1};
- λ (Φ_{epi}/Φ_{th}): 0.04;
- Fast neutron flux: 0.2Φ_{th};
- Type of experiment: movable;
- Applicable operation procedure: the irradiation rigs may be loaded, moved or discharged from the core during reactor power operation. The rig insertion speed is limited to 20 pcm/s;

— External rig (one, at the grid outside the active core):
- Position: B-4;
- Use: for ^{51}Cr production and neutron activation analysis;
- Capacity of the rig: 16 targets (canned by aluminium capsules);
- Δ_ρ: ≤100 pcm;
- Thermal neutron flux at 10 MW operation: 2–4 × 10^{13} n·cm^{-2}·s^{-1};
- Type of experiment: movable;
- Applicable operation procedure: the irradiation rigs may be loaded, moved or discharged from the core during reactor power operation.

2.1.2. Pneumatic systems

The pneumatic systems allow the transport of targets (encapsulated) from a loading station to different irradiation positions, returning them after a pre-established time. There are two identical systems, each of them for a determined grid position. Both systems can act simultaneously. The capsules have a cylindrical geometry and use screw caps (capsule diameter 25 mm, capsule total length 64 mm). Nitrogen is used for propelling the capsules.

A pneumatic system room, located on the first floor, houses two radiochemical hoods with the loading stations of the two pneumatic systems. Additionally, one of the pneumatic systems is connected to the hot cell.

Operation of the pneumatic systems, which is either manual or automatic, is initiated from the control room of the reactor:

— Pneumatic systems (two, at different grid positions):
  • Positions: A-3, A-5;
  • Use: mainly for neutron activation analysis;
  • Capacity: 1 capsule per position;
  • Thermal neutron flux (at 10 MW operation): 1.0×10^{13} n·cm^{-2}·s^{-1} (estimated);
  • Type of experiment: movable;
  • Applicable operation procedure: the irradiation rigs could be loaded, moved or discharged from the core during reactor power operation.

2.1.3. Thermal column

The thermal column is used for radioisotope production with low gamma radiation exposure and also for experiments with low energy neutrons. It is located in the ground floor hall of the reactor building. During the experiments, the samples to be irradiated are put in some holes or compartments in the block consisting of nuclear pure graphite. A lead sheet located between the core and the core directed face of the column reduces the gamma radiation to acceptable levels in order to minimize the heat of the graphite and to enable safe operation.

Sample irradiation at the thermal column:

— Capacity: 1 stringer in a central hole with 15 positions for capsules;
— Thermal neutron flux (at 10 MW operation): $5 \times 10^{10} – 2 \times 10^{13}$ n·cm^{-2}·s^{-1};
— Type of experiment: movable;
— Applicable operation procedure: the irradiation rig is loaded in or discharged from the core while the reactor is shut down.

2.2. Auxiliary facilities

The auxiliary facilities inside and outside the research reactor, which are involved in the irradiation process, and the radioisotope handling and processing are described in some detail here to demonstrate that radioisotopes cannot be produced in real quantities without a developed infrastructure.

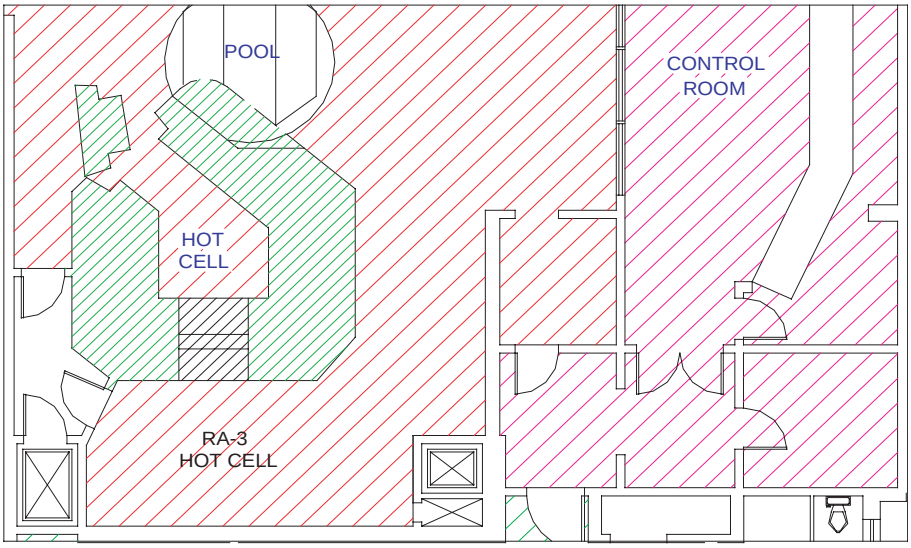

FIG. 6. RA-3 hot cell.

2.2.1. Hot cell

The hot cell (see Fig. 6), located at the second floor of the RA-3, supports the transfer of fresh targets into or adjacent to the core and also receives them after irradiation. It is also involved in spent fuel assembly receipt from the core before sending them to storage in the decay pool located in the ground floor hall of the RA-3.

This hot cell has one pair of 'master–slave' manipulators and a bridge crane to support all operations. A terminal station at the hot cell working table is used by the cart for transporting the irradiation rigs and fuel assemblies to and from the core and grid.

2.2.2. System for sending and receiving irradiation rigs and fuel assemblies

The system for sending and receiving irradiation rigs and fuel assemblies consists of one small guided cart which allows the transportation of irradiation rigs or fuel assemblies from the hot cell to the core, as well as to the storage and decay pool. For this purpose and in order to avoid high gamma exposure at the surface of the reactor pool, the cart moves within a tube embedded inside the concrete shielding, entering the pool below the water level. The capacity of the cart is one fuel assembly, with or without control rod, or one irradiation rig.

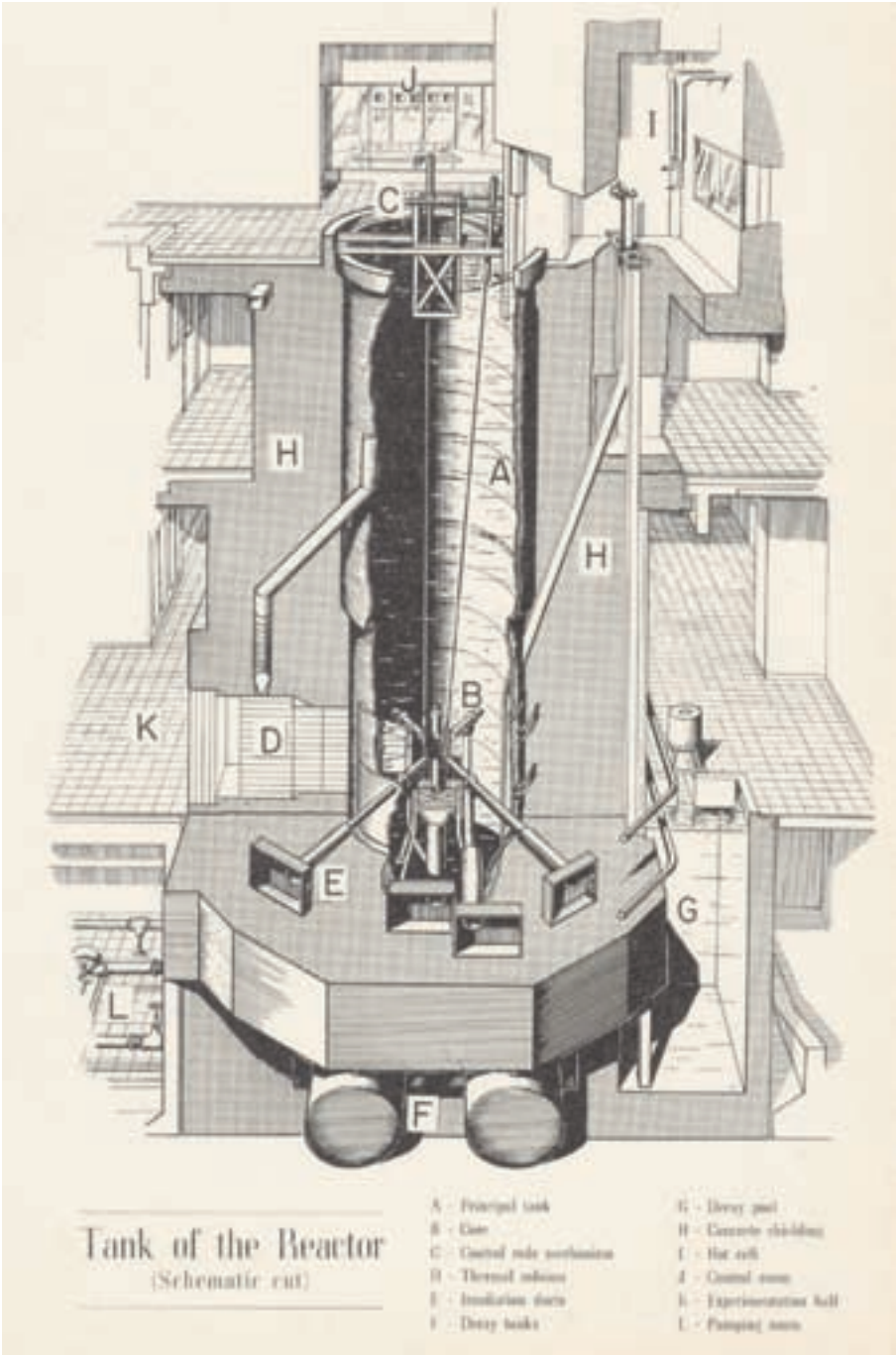

FIG. 7. Schematic view of the tank of the reactor.

2.3. Licensing procedures

The RA-3 research reactor has been licensed by the Argentine Nuclear Regulatory Authority (ARN), which is responsible for the requirements concerning organizational structure and responsibilities, staffing, training of personnel, operating procedures, modifications, experiment records and reports, and all required action following any potential violation of operational limits and conditions.

The head of the reactor facility is the responsible person for irradiation activities. Formal procedures have to be fulfilled for each irradiation. All experimental irradiation is authorized by the ARN with regard to the specific irradiation facility and the material to be irradiated. Prototypes of the irradiation rigs have to undergo thermal hydraulic tests in a cooling loop to satisfy ARN requirements.

In the case of new experiments, it is necessary to present a description of the experiment or the special irradiation, its objective and the expected results, calculations of the generated power, potential incidents (from the safety point of view), calculation or estimation of the reactivity increment of the experiment, calculation of the generated activity after irradiation, expected dose rate and shielding to be used for transport and storage, cost estimation and cost–benefit analysis. Then all information is evaluated by the RA-3 Internal Safety Committee according to the operation licence, and authorized if acceptable.

3. RADIOCHEMICAL PROCESSING FACILITIES

The main facilities used for the radiochemical processing of targets irradiated in the RA-3 research reactor are described below.

3.1. General criteria for design and safety

One of the main issues for radioisotope processing facilities is safety.

To prevent accidents in processing facilities such as the Radioisotopes Production Facility, which provides the infrastructure for processing the irradiated samples, a ‘defence in depth’ strategy compatible with international standards is adopted. This strategy requires multilevel protection and physical barriers to prevent the release of radioactivity inside the plant and to the environment, and employs at three levels:

— Prevention of accidents;

— Protection against accidents;
— Mitigation of the consequences of accidents.

Principal features of the facility design related to safety are:

— Prevention of spills and dispersion;
— Multiple containment features;
— Zoning of functional areas.

All facilities have been designed and are operated in accordance with international standards. Confinement structures have been designed to provide specific retention for radioactivity during normal operation and following any postulated release within the structures.

All facilities are operated under radiation protection considerations. During normal operation, the radiation exposure of site personnel and the public remains below prescribed limits and is kept as low as reasonably achievable (ALARA). Moreover, mitigation of radiological consequences from accidents has been ensured.

In accordance with the nature of the tasks to be performed and/or the level of hazard involved, all processing facilities are divided into clearly defined sectors. The laboratories are distributed according to a task diagram. The air circulation flows from lesser to greater potential radiological risk areas. Access to potentially contaminated zones is strictly regulated and supervised. Potentially contaminated zones and radioisotope processing areas are maintained at lower than atmospheric pressure. Transport, processing or manipulation systems are established in such a way as to meet the ALARA principle.

All active exhaust routes are equipped with measurement and/or sampling devices to quantify activity, and with appropriate filter systems. Special attention was placed on the discharge of gaseous effluents and their hazards within the boundaries of the Ezeiza Atomic Center itself and, outside those boundaries, to the environment in general.

All processing is performed within hot cells or glove boxes. In those areas where tasks outside hot cells or glove boxes cannot be avoided, specific procedures are followed. The procedures have been implemented to prevent unauthorized discharges. In normal operating conditions, there are at least two levels of confinement (static, dynamic) between any source of contamination and the environment. All equipment and all processing steps have been designed to minimize waste, to ease maintenance and subsequent decontamination, and for easy dismantling of the equipment.

3.2. Radioisotope processing facility

The Radioisotopes Production Facility was built primarily for the handling and processing of medium and short lived radioisotopes and for the preparation of radiopharmaceuticals and labelled molecules. It is also located at the Ezeiza Atomic Center. Due to the operations that are performed in different sections of the facility, it was necessary to classify the rooms according to different degrees of radiological risk. There is a central corridor in which the leaktight and shielded hot cells are installed, one for each type of radioisotope and related process. Also, beneath the central corridor in a basement, there is a service corridor that houses the tanks for the temporary storage of liquid waste and the pipelines to transport that waste out of the building. Both the central corridor and the basement are areas of greater contamination risk (hot corridors). The central corridor is internally connected, through a safety air system (SAS), with the RA-3 research reactor. The adjacent laboratories, designed for remote operation and control procedures, form an area of lower radiological risk. Lastly, the peripheral parts of the building house cold laboratories, offices, storage areas, a machine shop and changing rooms.

The total floor surface of the building is 2500 m^2, comprising 18 different hot cells on the ground floor, as well as processing and quality control laboratories with their glove boxes and radiochemical fume hoods, mechanical and electrical workshops, offices and changing rooms. The power supply, the air intake system and the service corridor may be found in the cellar. The first floor houses the ventilation system. The hot corridor at the rear side of the hot cells is connected to the RA-3 building through one appendix.

According to the radiological risk, the facility is divided into three areas. A free area includes the main access, offices, a cafeteria, a storeroom, electric switchboards, the air intake system, as well as changing rooms, etc. A supervised area includes radiochemical processing and quality control laboratories, packaging, storage and ventilation exhaust fans. Finally, the controlled area comprises the hot corridor with the 5 t bridge crane and the SAS, as well as the ventilation filter rooms, the entire cellar, and the rooms for decontamination, hot maintenance and repair. The different building zones have been detailed in Fig. 8.

The facility has a total of 18 hot cells, 9 of them with a shielding of 50 mm of lead and 9 with 100 mm. The cells provide separation between the hot corridor and the radiochemical laboratories. Each hot cell has an internal airtight box (1800 × 1500 × 900 mm) housing the radiochemical process equipment. Some of the housings are made from acrylic glass, others from stainless steel. The lower part of the boxes beneath the worktable houses the tanks for solid and liquid waste temporary storage. Airtight boxes, ventilated

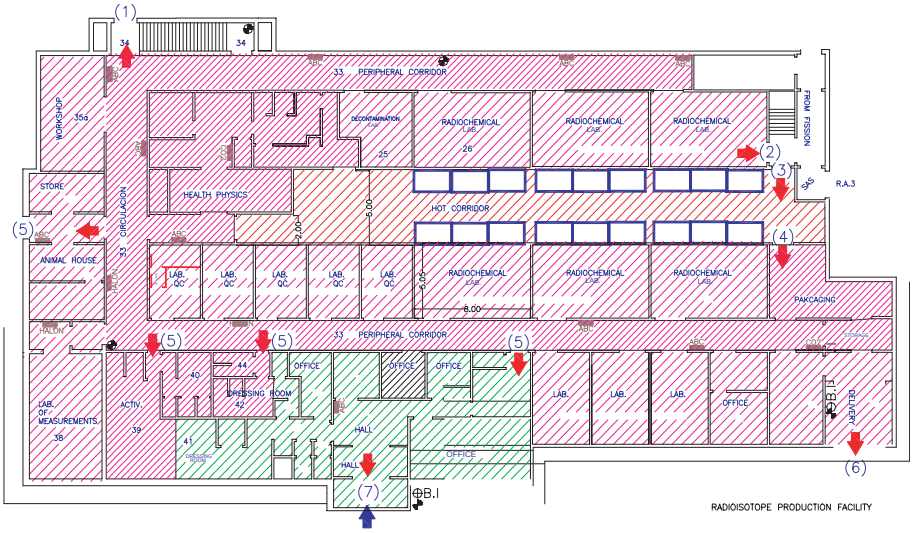

FIG. 8. Radioisotopes Production Facility.

from bottom to top, are continuously under negative pressure (25 mm water column) and the air is renewed 2–6 times per hour.

All the walls are of brick except the wall of the room used for gamma spectrometry. In this case, the wall is made from reinforced concrete (thickness 0.2 m and density 2400 kg/m^3). The walls are painted with 1 mm of epoxy resin. In addition, the floor is covered with paint from epoxy resin with a thickness of 3 mm.

The ventilation and air conditioning system maintains the air flow between connected areas, reduces the radioactivity level in the confined areas, maintains air comfort conditions in normally occupied rooms during normal operation and protects the outer environment. The air of the facility is renewed twice per hour. Except for offices, measurement rooms, etc., which are ventilated with individual equipment, the facility has a common ventilation system. The external air is filtered and conditioned (temperature and humidity), and is then injected into the peripheral rooms and corridors. From there, the air passes to the hot corridor, sweeps the radiochemical laboratories and is led to the cellar. Finally, the air arrives at the filter rooms located on the first floor and, after passing through absolute filters, is sent directly to the stack. All the ventilation system is controlled by a programmable logic controller (PLC).

The liquid waste generated by the radiochemical processes is temporarily stored in tanks located in the lower parts of the hot cells, inside an airtight box beneath the worktable. When the tanks are full, the liquid is, depending on the

total activity and the half-life of the radioisotopes, transferred by vacuum to an intermediate tank and sent to the Waste Storage and Treatment Facility located within the Ezeiza Atomic Center, or transferred into two external decay tanks, each with a volume of 15 m^3. The liquid waste coming from sinks or radiochemical hoods is sent directly to the external decay tanks. All transfer pipes are made of PVC. The solid waste generated during each process is generally stored for some time within the hot cell before being sent to the Waste Storage and Treatment Facility.

The Radioisotopes Production Facility has been licensed by the ARN, which maintains the requirements concerning organizational structure and responsibilities, staffing, personnel training, operational procedures, modifications, experiment records and reports, and all required action following the violation of operational limits and conditions. It is obvious that thorough quality control of all processing steps and all incoming and outgoing products, including their packaging, has to be performed throughout.

From its inauguration in 1973, the facility has been processing and producing medium and short lived radioisotopes and labelled compounds, mainly for nuclear medicine and industry, comprising ^{18}F, ^{24}Na, ^{32}P, ^{46}Sc, ^{51}Cr, ^{59}Fe, ^{64}Cu, ^{82}Br, ^{86}Rb, $^{99}Mo/^{99m}Tc$ and $^{113m}Sn/^{113}In$ generators, ^{131}I, ^{169}Yb, ^{188}Re, ^{198}Au, ^{197}Hg and ^{203}Hg.

3.3. Fission Radioisotopes Production Facility

The main purpose of the second radioisotope processing facility at Ezeiza is the separation of different radioisotopes of commercial interest (^{99}Mo, ^{131}I, ^{90}Sr, ^{137}Cs and ^{133}Xe) from irradiated uranium targets. The commercial operation of the Fission Radioisotopes Production Facility started in 1985. In 1995, that facility was enlarged to its present state (see Fig. 9).

The facility is located in a three-storey building neighbouring the RA-3 reactor. The present production capacity is licensed by the ARN to 65 TBq (1750 Ci) of ^{99}Mo per process, only 15–18 TBq of which are demanded by the local market.

Alpha–beta–gamma hot cells, located in a controlled area and divided into two blocks, are used for the radiochemical processes. This controlled area is connected, through an SAS, to the RA-3 research reactor. The transport of the irradiated targets is done through this SAS using a motorized cart with 230 mm lead shielding and a capacity for carrying up to four targets.

The first block contains two principal hot cells, shielded with 300 mm of lead, and two auxiliary hot cells. One of the principal hot cells is used for dissolution of the target and the first separation step. The other is used for storing the irradiated uranium after dissolution. Each hot cell has an internal

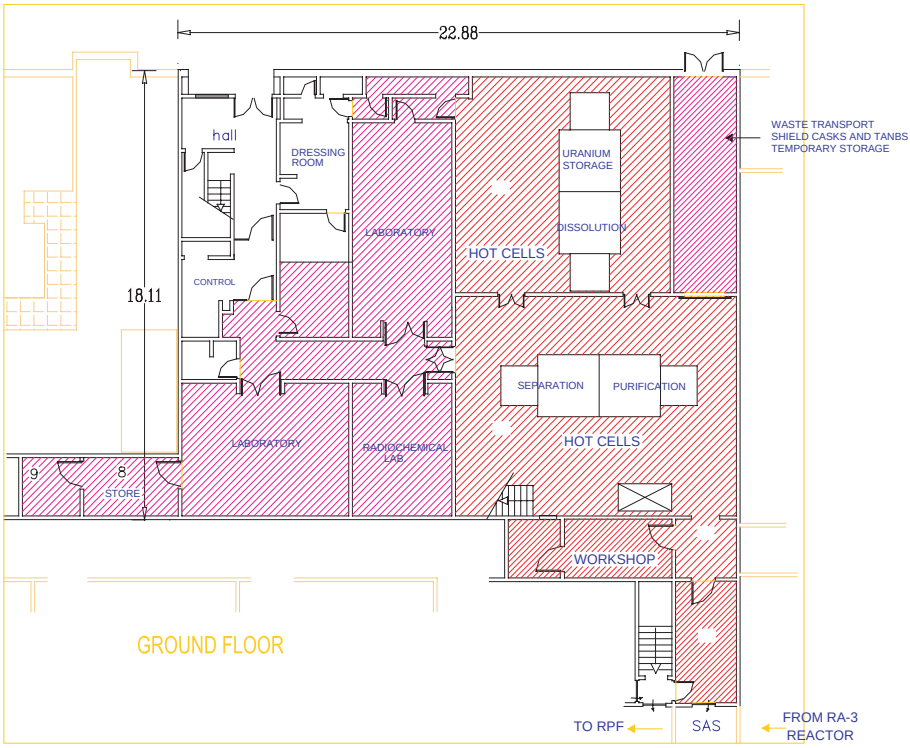

FIG. 9. The Fission Radioisotopes Production Facility.

airtight box (2000 × 1600 × 1600 mm), built from 3 mm thick stainless steel supported on an iron structure 1 m from floor level, housing the radiochemical process equipment. Liquid and gas decay tanks are located beneath each airtight box. Maintenance access doors are located on the back walls.

The second block is used for the separation and purification steps. It also contains two principal hot cells shielded with 200 mm of lead and two auxiliary hot cells. A stainless steel pipeline, located beneath the floor, connects the dissolution hot cell of the first block with the first purification hot cell.

The airtight boxes are continuously under negative pressure (25 mm water column) and the air is renewed 2–6 times per hour.

The auxiliary hot cells, used to support the process, are connected with the principal hot cells through a transfer tunnel.

The facility has been producing ^{99}Mo, continuously and routinely, since 1985. High enriched uranium (HEU, 90% ^{235}U) targets were used up to 2002. They consisted of a uranium–aluminium alloy (the 'meat') clad with aluminium. Due to international concerns about the proliferation of weapons

grade material, however, which restricted the supply of HEU, in 2002 the HEU targets were replaced by low enriched uranium (LEU, <20% ^{235}U) targets [4]. The radiochemical process was subsequently modified to handle five times more uranium [5]. A description of the implemented method and the quality control of the final product are given in Ref. [6].

The highly complex technology involved exceeds the subject of this paper and of the Compendium. It should be clear, however, that the production of radioisotopes at a research reactor in an industrial style requires many more facilities, buildings, processes, equipment, skills and know-how than pure generation at a research reactor in rigs or rabbit systems.

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3.4. IRRADIATION FACILITIES FOR THE PRODUCTION OF RADIOISOTOPES FOR MEDICAL PURPOSES AND FOR INDUSTRY AT THE ROSSENDORF RESEARCH REACTOR

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1. INTRODUCTION

In 1955, the Government of the German Democratic Republic initiated radioisotope production. With that decision, the following plants received their go ahead [1]:

— Research reactor with its user facilities;
— Cyclotron with its specific facilities;
— Institute for radiochemistry;
— Library, lecture hall, workshops and administration buildings supporting the necessary scientific and administrative environment.

The Zentralinstitut für Kerntechnik (ZfK), also known as the Central Institute for Nuclear Technology, was founded at Rossendorf near Dresden, Germany, to house all those plants. The Rossendorf Research Reactor (RFR) was constructed in 1956–1957. That endeavour was enabled by the technological support of the former USSR under a bilateral agreement which included the delivery of a 2 MW research reactor of the WWR-S design [2].

2. BRIEF CHARACTERIZATION OF THE RFR

The RFR was a heterogeneous tank type thermal research reactor (see Fig. 1). Desalinated water was applied as both moderator and coolant. During the first decade of operation, fuel with 10% enriched uranium was used in the Russian EK-10 type fuel assemblies with a 68 × 68 mm cross-section and 500 mm active length containing 16 fuel rods each.

The reactor tank consisted of a central vessel in which a separate basket enclosed the reactor core. Up to 52 fuel assemblies could be loaded; any reduction of that number by irradiation inserts resulted in a loss of excess reactivity.

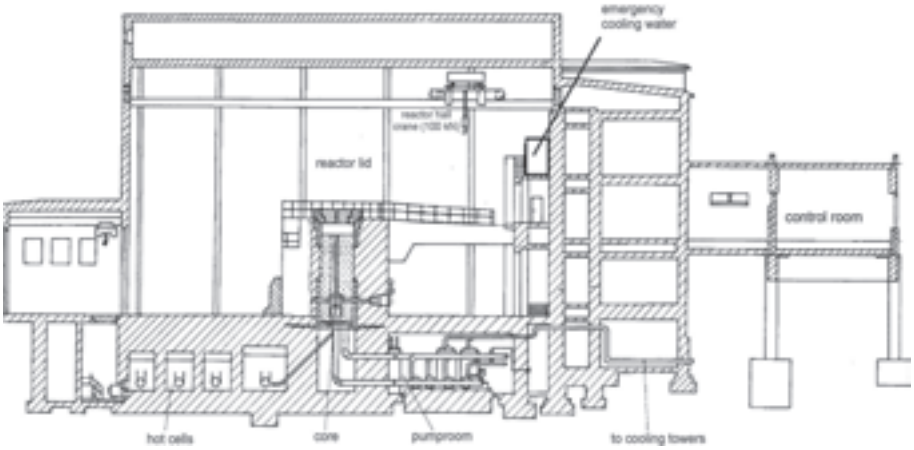

FIG. 1. Cross-section of the RFR plant (archives VKTA Rossendorf).

The central vessel set the boundary of the primary cooling loop at the core area. Coolant with an average temperature of 35°C flowed downwards through that vessel at 1000 m³/h. The experimental advantage of this concept required greater emergency cooling capability when reactor power was increased. A flooded, 2250 mm diameter, aluminium protection vessel enclosed the central, 1100 mm diameter aluminium vessel. The thermal column extended into the protection vessel up to the border of the reactor core. Above the core, the protection vessel had a walk-on aluminium platform, which could be accessed with the reactor lid open. That platform carried the facilities to supervise and to control the reactor core, as well as the mechanical installation for radioisotope production.

In 1967, the RFR was converted to higher performance fuel assemblies containing 36% enriched uranium (hexagonal Russian ECH-1 fuel assemblies with 35 mm width over the flats at the upper end piece and 32 mm through the 600 mm active height). That conversion was the result of the successful operation of similar fuel at the Gatchina WWR-M.

The new RFR core grid comprised 217 positions, two thirds of which were required for fuel assemblies. An arrangement of beryllium and aluminium elements at the remaining grid positions contributed to the performance improvement for irradiation applications.

TABLE 1. INITIAL PROGRAMME OF RADIOISOTOPES AT THE RFR

Nuclide	Half-life	Nuclear reaction	Target	Final product
^{18}F	1.83 h	$^{6}Li(n, \alpha)t$ $^{16}O(t, n)^{18}F$	Li_2CO_3	$[Na_3 Al]F_6$
^{24}Na	14.96 h	$^{23}Na(n, \gamma)^{24}Na$	Na_2CO_3	$[^{24}Na]Cl$
^{32}P	14.26 d	$^{31}P(n, \gamma)^{32}P$	P_2O_5	$[^{32}P]_2O_5$
^{42}K	12.36 h	$^{41}K(n, \gamma)^{42}K$	K_2CO_3	$[^{42}K]Cl$
^{64}Cu	12.70 h	$^{63}Cu(n, \gamma)^{64}Cu$	Cu	$[^{64}Cu]Cl_2$
^{72}Ga	14.10 h	$^{71}Ga(n, \gamma)^{72}Ga$	Ga_2O_3	$[^{72}Ga]Cl_3$
^{76}As	25.87 h	$^{75}As(n, \gamma)^{76}As$	As_2O_3	$[^{76}As]_2O_3$
^{80m}Br	4.42 h	$^{79}Br(n, \gamma)^{80m}Br$	$BaBr_2$	$[^{80m}Na]Br_2$
^{82}Br	1.47 d	$^{81}Br(n, \gamma)^{82}Br$	$BaBr_2$	$[^{82}Na]Br_2$
^{86}Rb	18.63 d	$^{85}Rb(n, \gamma)^{86}Rb$	Rb_2CO_3	$[^{86}Rb]Cl$
^{90}Y	2.67 d	$^{89}Y(n, \gamma)^{90}Y$	YO_3	$[^{90}Y]Cl_3$, $[^{90}Y]$ seeds
^{140}La	1.68 d	$^{139}La(n, \gamma)^{140}La$	La_2O_3	$[^{140}La]Cl_3$
^{198}Au	2.70 d	$^{197}Au(n, \gamma)^{198}Au$	Au	H $[^{198}Au]$ Cl_3, $[^{198}Au]$ seeds

3. RADIOISOTOPE PRODUCTION AT THE RFR

From 1953, the German Democratic Republic imported radioisotopes from the former USSR. Due to transport and administrative delays, importation was restricted to long lived isotopes. Improvement came in 1958, when the first radioisotope compound (ethyl bromide) was delivered by the ZfK. In 1959, the supply from the ZfK had grown to 300 deliveries totalling 700 GBq. The initial ZfK supply [3] comprised the nuclides shown in Table 1.

Customer requests grew in the following years. However, the remaining volume for irradiation inserts at the RFR grid decreased due to the growing number of fuel assemblies inserted to compensate for the reactivity loss from burnup. Despite multilateral support from Eastern States, enhancing the RFR irradiation capacity remained the major task. Capacity was mainly increased by three complementary measures: longer operation time (from 1606 h in 1958 to 4200 h in 1964), upgrading of reactor power (from 2 MW in 1958 to 4 MW in 1965) and improved irradiation inserts. Although this was adequate for the medium term requirements of the consumers, it was soon evident that these measures would be insufficient for the long term.

Cooperation between the German Democratic Republic (Rossendorf), Poland (Swierk) and Hungary (Budapest) was established to further increase the generation capacity at the three centres. At the RFR, the main measure was a power increase to 10 MW, which mostly served to enhance the thermal flux at the irradiation inserts. Increasing the ratio between thermal flux at those inserts and the thermal power was an important criterion for the power increase as such. Important measures to enhance that ratio were the installation of irradiation inserts, which were embedded into beryllium; the set-up of a fixed beryllium reflector at the core border; and the application of so-called beryllium cassettes inside the core, replacing single fuel assemblies. The positioning of those beryllium cassettes enabled optimization of the neutron flux at their locations for an intended purpose. As those cassettes had the exact geometry of the fuel assemblies, there were no drawbacks in terms of core cooling. The increase in radioactive deliveries to the customers over time is an indication of the strategic success of this work (see Fig. 2).

4. IRRADIATION FACILITIES OF THE RFR

4.1. Irradiation facilities of the RFR from 1957 to 1967

Initially, the grid plate of the RFR had 52 positions (see Fig. 1), 35 of which were filled with fuel assemblies only. The initial irradiation inserts

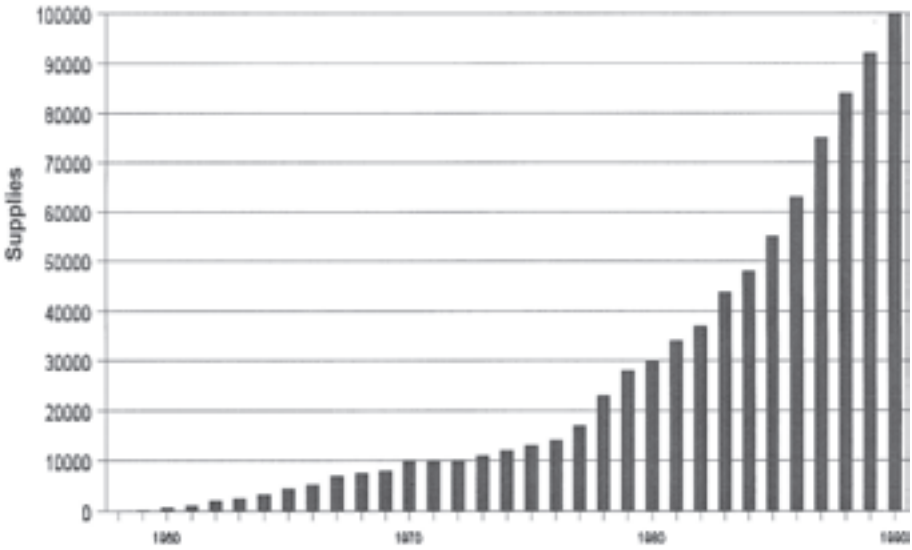

FIG. 2. Number of deliveries of radioisotopes from the ZfK.

consisted of simple displacement tubes, which were inserted at the positions not used for fuel assemblies. Those tubes displaced the water in front of the ten horizontal beam tubes of the RFR to avoid reducing the neutron flux at the entrances of those beam tubes from absorption in the rather thick water layer between the core border and the entrances. Inside the upper part of the displacement tubes, exposure of aluminium isotope irradiation capsules (IIC) took place. The irradiation volume made available that way was around 1.25 L and provided an average thermal neutron flux density of 1.2×10^{12} n·cm^{-2}·s^{-1}. Additionally, eight smaller irradiation tubes, four vertical air-filled channels inside the thermal column and three channels for irradiation inside the biological shield at rather low neutron flux density were available. During the initial period, the air-filled channels inside the thermal column were loaded/discharged with fishing lines. The channels inside the biological shield were used only rarely.

An important supplement of the irradiation facilities was the four hot cells (HK1–HK4) at the RFR, which existed below the floor of the reactor hall (at level 0.00 m) inside the cellar (at level –4.90 m). HK1 was designed for a fission product activity of 370 TBq and had dimensions of 2.0 m × 3.2 m × 2.3 m. HK1 included some tooling, such as a cutting device for the IICs. HK2–HK4 were designed for 135 TBq maximum and had dimensions of 1.2 m × 2.0 m × 2.0 m. Each hot cell was equipped with 'master–slave' manipulators of Russian design (M15 and M22, respectively). The manipulators permitted position dependent loading between 30 N and 150 N and were operated from one room per hot cell. The hot cells were linked to the above-core volume and between each other by transfer channels. A special interlock system allowed the transfer of irradiated material (quartz ampoules, packaging). The hot cells were not designed for alpha active materials.

The initial irradiation insert handling devices delivered with the reactor caused real problems [4]. Normally, the irradiation samples were molten into quartz ampoules which were packed into the IICs. The IICs were loaded manually into their respective positions with a simple loading rod prior to the start of the reactor. At the end of the irradiation process, discharge was performed using a crank driven discharge rod, which was mounted on the reactor tank lid, comprising two eccentrically revolving parts. The design of that lid enabled each position of the grid to be approached for discharge. Both, loading and discharge of IICs, were possible only with the primary cooling loop shut off. The problem was the unsecured gripping which, if it failed, was not detectable due to missing instrumentation. It was an uncertain and slow process overall. Additionally, the IICs could not be fully lowered down to the hot cells, so the transfer to the hot cells was performed by dropping the IICs

into a transfer channel, a process that contributed substantially to the failure rate of the ampoules.

Due to the supplementary fuel assemblies inserted to compensate for the loss of reactivity due to burnup, the available irradiation volume was continuously reduced. Therefore, fewer displacement tubes could be inserted. To compensate for that loss, displacement boxes without air cushions were installed in six positions in front of unused beam tubes. That compensation was equivalent to six tubes at the grid plate. Further measures were implemented in order to improvie the irradiation techniques at the RFR, including:

— Insertion of new IICs with smaller facility top ends to enlarge the irradiation volume and to reduce the active waste at the same time;
— Application of short period irradiation capsules from plastics;
— Application of gastight and watertight aluminium IICs (welded IICs);
— Installation of tube type elongations of the displacement tubes to above the water surface of the reactor tank, in order to make a part of the irradiation positions accessible during the ongoing operation of the primary cooling loop and thus during reactor operation at rated power;
— Application of a new gripper mechanism which could be slowly lowered into the hot cell, enabling the transfer of IICs from the hot cell to the irradiation positions of the RFR. This was a semi-automated system and included status indication.

The reactor power upgrade to 4 MW in 1965 as well as the prolonged annual operation increased radioisotope sample deliveries again. However, these increases were accompanied by an increase in radiation dose at the reactor top and inside the hot cells. The reserves of the RFR for producing radioisotopes were consumed. Figure 3 shows the reactor top together with the improved discharge rod.

The number of irradiation positions at the grid was not reduced following the power increase to 4 MW. Additionally, the required excess reactivity for operation of the RFR at 4 MW had increased, thus, the discharge burnup of the fuel assemblies dropped. Consequently, the demand for underwater storage positions for discharged fuel assemblies increased. To compensate for that demand, a new fuel storage pond (AB 2) was built in 1966, which increased the storage capacity for spent fuel assemblies from 77 kg heavy metal originally to 256 kg. Within that pond, a gamma irradiation facility (GAR 1) was integrated. GAR 1 was successfully operated as a gamma irradiation service by the effective cyclic exchange of freshly discharged, partially burned fuel assemblies from the RFR core with decayed, partially burned fuel assemblies from the AB 2 itself. As a standard, 2 × 17 partially burned fuel assemblies were discharged

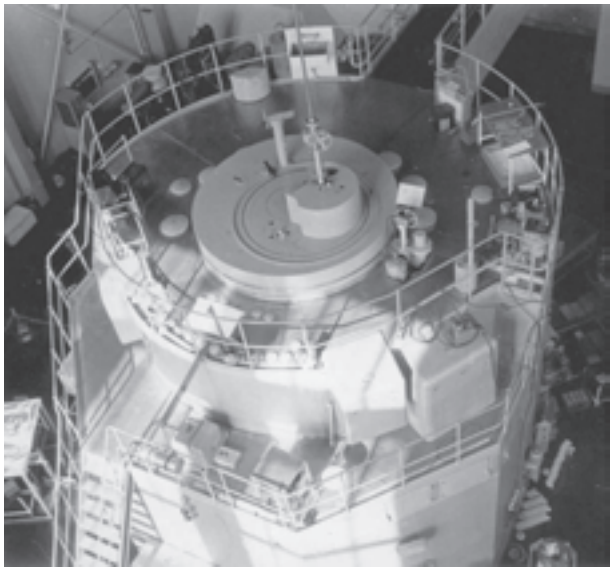

FIG. 3. RFR reactor top with improved discharge rod (archives VKTA Rossendorf).

from GAR 1, which used the gamma radiation of the fuel assemblies between the 5th and 60th day after their end of in-core operation period. Their irradiation dose rate averaged over the 55 day period resulted in about 100 Gy/h. The field of application of GAR 1 was predominantly the radiation sterilization of disposable medical products, such as catheters, blood transfusion equipment, syringes, dialysis tools, infusion tools and surgical protective gloves. Test irradiations were performed on cereals and foodstuffs, as well as on technical components, e.g. electric cables.

4.2. RFR irradiation facilities from 1967 to 1991

The irradiation capacity of the RFR before 1967 did not meet radioisotope production requirements. The mentioned conversion of the RFR from EK-10 to ECH-1 fuel assemblies resulted in new possibilities for the increase of that capacity, as the RFR core with its 217 grid positions had an effective beryllium reflector from 1967 on. New irradiation positions thus emerged (see Fig. 4). By the stepwise increase of reactor thermal power (from 1971: 6 MW; from 1974: 8 MW; from 1979: 10 MW), the irradiation techniques had to undergo fundamental change. Automation of the radiation processes became unavoidable, as the dose rate at the reactor top grew with the increase of reactor power and as the demands of the customers for radioisotopes steadily increased. Aims of the automation of processes were:

FIG. 4. Top view of the RFR core with new fuel assemblies (post-1967) (archives VKTA Rossendorf).

— Reduction of the radiation exposure of RFR personnel;
— Decrease of the staff per shift by performing the loading, reloading and discharge processes from the reactor control room;
— Improvement of the transport techniques of the IICs subsequent to irradiation and to the location of the radiochemical processing;
— Increase of the irradiation capacity and tapping reserves in capacity.

In 1971, the first automated loading and discharge facility for IICs, named BERTA, was put into operation. BERTA was based on forerunning, fast loading equipment, called ANTON, with which the requested manipulations were pretested. In 1973, with the commissioning of new nuclear instrumentation, the possibility of automation became real; the development of computer science and data processing derived at the ZfK was of decisive importance for that step. From 1978 on, the following devices were implemented one after the other:

— Reactor lid positioning automation;
— Both the reactivity controlled loading, reloading and discharge facilities DORA;
— The n-fluence controlled, upstroke turnaround irradiation channels;

— Loading, reloading and discharge facility (ERIKA) for serving the upstroke turnaround irradiation channels;
— Loading and discharge robot (BERO) as the central control system;
— Rabbit system for the pneumatic transport of IICs between the RFR core area and the RFR hot cells;
— Pneumatic irradiation facilities at the reactor grid plate and at the thermal column.

One important measure was increasing the distance between the walk-on aluminium platform at the reactor top from 400 to 900 mm. That greater distance enabled installation of the irradiation facilities at the reactor top substantially. An automated system to transport IICs between the reactor and the radiochemical laboratories radioisotope production was designed but never commissioned.

The reactor lid positioning system consisted of the control, an optical digital indicator and the combined power supply and signal transmission system. The control was operable from the irradiation related part of the reactor control console and was linked to the robot BERO. The optical–digital indicator relayed the position of the two eccentric reactor lids. A full 360° rotation was not possible, as the related trailing cable system was too complex.

With the DORA facilities, reactivity controlled loading, reloading and discharge processes of the IICs into and out of irradiation positions became possible during full load operation. One of the two DORA facilities was mounted to the reactor lid and linked to the reactor control circuit. The second DORA facility was used as a backup. The loading process of the DORA facility was stopped as soon as the reactivity change caused from the movement of the DORA gripper and from the IIC arrived at a preset limit. This limit was sliding, depending on the actual reactivity rate of the automatic control rod, and preset to a lower limit (e.g. 1.64×10^{-4} s^{-1}). Once the automated control circuit of the reactor had compensated for the reactivity change by moving the control rods, the handling process continued. The introduction of this automation was a substantial prerequisite for enhancement of the throughput of IICs at the RFR.

The n-fluence controlled upstroke turnaround irradiation channels were reserved for the homogeneous doping of silicon. Thus, they were used for the production of radioisotopes in cases of extreme urgency or at times when no silicon was to be doped. Those IICs (diameter 65 mm, length 200 mm) loaded into the upstroke turnaround irradiation channels differed in design and dimension from those inserted into the standard irradiation channels and handled by ERIKA. This different IIC was discharged from its irradiation channel after arriving at a given n-fluence instead of irradiating it for a given

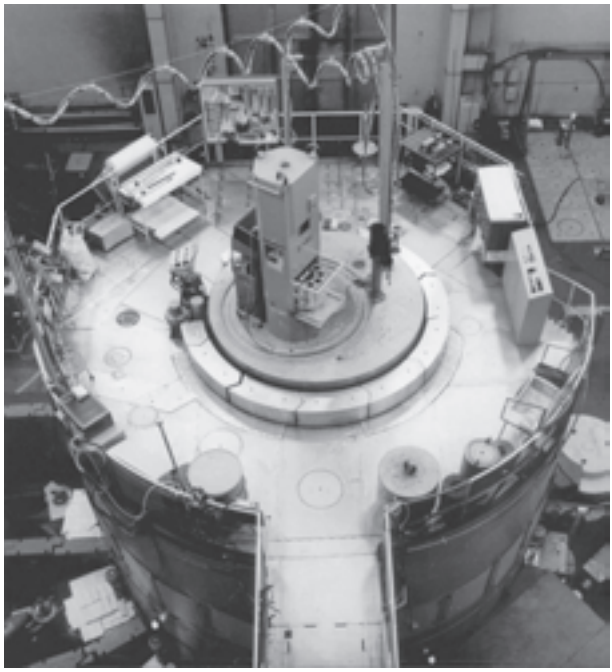

FIG. 5. View of the top of the reactor with the DORA and ERIKA facilities (archives VKTA Rossendorf).

preset time. This process was enabled by the application of n/ß detectors. They were mounted directly beside those irradiation channels. In order to terminate irradiation of the samples as precisely as possible, the entire channel, with its IIC inside, was removed from the radiation field of the reactor core at the end of the irradiation. Further, to minimize the effect originating from radial flux gradients, the irradiation channel was rotated at 30 rev./h. The effect of vertical flux gradients was weakened by an Ni shield with varying sheet thickness. Using these measures, the requested irradiation parameters could be accomplished within a margin of less than 5% for the entire sample. Due to the larger dimensions, the loading, reloading and discharge of these IICs were performed by the loading facility ERIKA (see Figs 3–5). Unlike DORA, ERIKA had one reloading position and several decay positions for 30 IICs in total inside the reactor top, as well as a magazine comprising ten storage positions. The discharge of the irradiated IICs was performed when their activity had decayed to 30 MBq maximum. It is obvious from the above that ERIKA was designed for silicon ingots, up to an ingot diameter of 100 mm.

From August 1981 on, the loading and discharge robot BERO [7] served for the automated loading, reloading and discharge of IICs for radioisotope

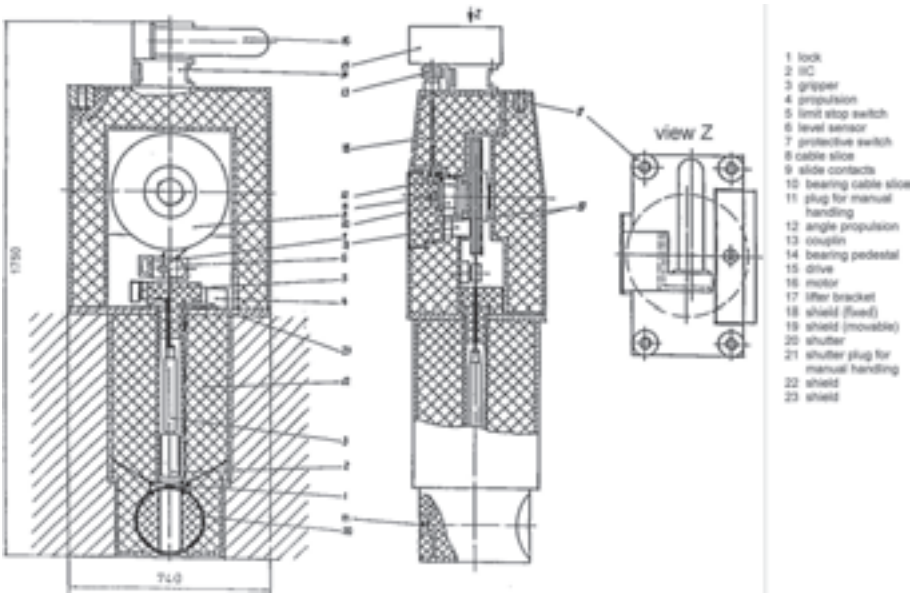

FIG. 6. Cross-section of DORA (archives VKTA Rossendorf).

production and for the irradiation cassettes for silicon doping. For that purpose, the existing control means for the basic handling processes were linked to a computing system, as developed at the ZfK, via specially designed interfaces. With the use of BERO, a substantial increase in automation of the reactor related part of radioisotope production could be accomplished.

The rabbit system for the pneumatic transfer of IICs consisted of a station placed underneath the reactor lid, comprising one line each to the hot cells HK 1 and HK 4. A retransfer of IICs out of the hot cells to the station beneath that lid could also be accomplished and was especially suitable following irradiation of IICs which had previously been irradiated. This facility was operated via a specific control unit either from the main control room or locally from the reactor top. The admissible mass of one IIC was limited to 650 g.

The pneumatic irradiation facilities inside the reactor core and inside the thermal column consisted of two irradiation positions each (one of which was clad with a Cd layer), the respective transfer tubing up to a laboratory where the activity was analysed, and the measuring apparatus at that laboratory. The transfer velocity was 6–10 m/s. The facility was operated at a pressure of 0.1 MPa. The facility predominantly served for short term NAA, i.e. in the region of minutes and seconds.

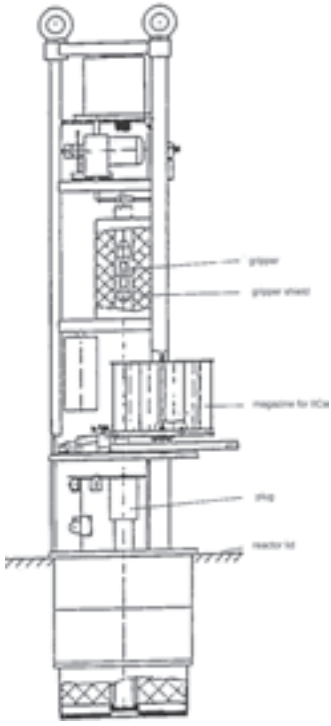

FIG. 7. Cross-section of ERIKA (archives VKTA Rossendorf).

As is typical for research reactors, within the RFR, improvements and modifications of the irradiation facilities and related sample and ingot carriers were an ongoing process, even near the end of reactor operation in 1991. At this point, the following facilities were available:

— 9 A channels[1] for IICs of type A
(ø = 44 mm, L = 150 mm), $\Phi_{th,\,max} = 2.3 \times 10^{13}$ n·cm^{-2}·s^{-1};
— 18 B channels for IICs of type B
(ø = 44 mm, L = 150 mm), $\Phi_{th,\,max} = 1.2 \times 10^{14}$ n·cm^{-2}·s^{-1};
— 5 C channels for IICs of type C
(ø = 59 mm, L = 150 mm), $\Phi_{th,\,max} = 1.6 \times 10^{14}$ n·cm^{-2}·s^{-1};
— 2 DE channels[2] for neutron transmutation doping (NTD) of silicon cassettes (ø > 79 mm, L = 200 mm), $_{th,\,max} = 2.2 \times 10^{13}$ n·cm^{-2}·s^{-1};

[1] Inside ripped tubes ($ø_{out}$ = 60 mm, $ø_{in.}$ = 47 mm).
[2] An enhancement to four DE channels was envisaged.

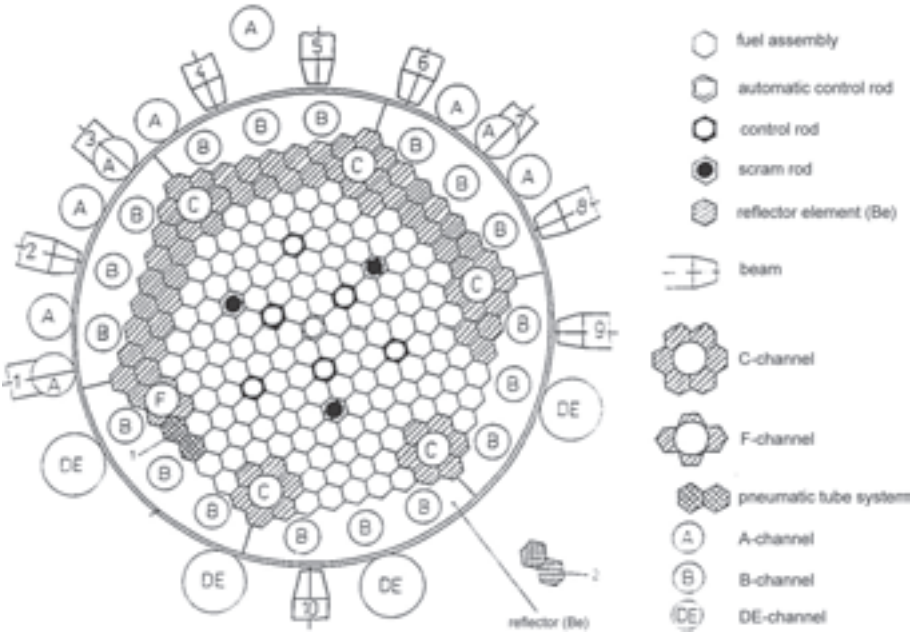

FIG. 8. Arrangement of the channels (archives VKTA Rossendorf).

— 2 DE channels under preparation;
— 1 F channel for IICs of type F
(ø = 44 mm, L = 150 mm), $_{th, max} = 1.7 \times 10^{14}$ n·cm^{-2}·s^{-1}.

In 1991, the total volume available for irradiation was about 40 L. The special IICs which were at a user's disposal included:

— Rabbits (ø = 25 mm, L = 45 mm and 70 mm) of polyethylene and aluminium;
— Ir irradiation IICs (ø = 36 mm, L = 70 mm) from AlMg1Si1;
— Xe irradiation IICs (ø = 36 mm, L = 150 mm) from AlMg1Si1;
— U irradiation IICs (ø = 59 mm, L = 150 mm) from AlMg1Si1.

The U irradiation capsules could also be used for non-fissile material and they lost importance when the new Mo production technology was applied.

5. GUARANTEE OF SAFETY FOR THE IRRADIATION PROCESS AND THE EQUIPMENT

Between 1959 and 1991, a total of 121 incidents were registered during the performance of irradiation. In the early years (2 MW operation), such incidents frequently were originating from technical deficiencies of the loading and discharging technology. In the period between 1965 and 1970 (4 MW operation), the number of incidents involving the IICs and the irradiation samples distinctly increased. This trend continued with every increase of reactor power and arrived at a peak in 1979–1980, the first two years of reactor operation with 10 MW. This trend emphasized the need for more stringent rules and regulations for the entire irradiation processes and facilities applied at the RFR.

Extensive procedural controls were introduced. Beyond theoretical considerations, such controls, rules and regulations were derived from years of operating experience and, in particular, from the subsequent investigation of any and all incidents. In terms of those incidents, in addition to understanding the n-flux relation at the irradiation positions, issues were considered, such as nuclear heating of the target material, the potential and real phase transformations from increasing sample temperatures, the release of gas from samples by various physical effects, the increase of the inner pressure in the IICs and within the capsules inside the IICs, as well as swelling effects of all materials involved.

The change of uranium irradiation technology from U_{nat} to enriched uranium was of special importance within these considerations. As the reactor power was increased, serious incidents appeared during the irradiation process for natural uranium. Those incidents led to several notifiable events and cumulated in a medium term shutdown of the RFR due to severe contamination of the primary cooling loop by molten uranium. Up to 5% of the U irradiation capsules were involved. The resulting risk for ongoing safe operation of the RFR could not be tolerated any longer. The solution was the development and introduction of the Anlage Molybdän Rossendorf (AMOR) process. Instead of U_{nat} pellets, fuel assemblies of the RFR were irradiated. The process relied on a 100 h irradiation of a fresh fuel assembly in the RFR. After a 20 h decay period, the irradiated and discharged fuel assembly was transferred, using a specialized fuel assembly transport cask, to the AMOR 1 plant for dissolution. Inside the AMOR 2 and AMOR 3 plants, the fuel cycle was closed by separating the fissile material from the molybdenum extraction waste solution and by subsequent production of new fuel assemblies (so-called target assemblies). The duty cycle of the entire process was about 90%; 5 TBq per week were provided at the respective delivery dates.

The incidents which had taken place during the performance of radioisotope production decreased substantially from 1981, despite increases in client deliveries (see Fig. 2). In addition to the mentioned arrangements and modifications, the reason for this positive gradient was the implementation of further provisions. To satisfy client expectations with respect to radiation services, the following had to be guaranteed:

— A minimum single wall hermetic encapsulation of the sample material was provided inside quartz ampoules or seal welded aluminium capsules, or seal welded austenitic steel capsules or IICs with a welded base;
— The target materials did not chemically react with each other or with the packaging materials;
— The quartz ampoules were packaged fail-safe against fracturing;
— The stability of the packaging was not reduced or endangered by the labelling;
— Boreholes in the IICs for cooling did not become clogged by Al foils or other similar material;
— No explosives or inflammable material were delivered for irradiation.

Inside the IICs, the target material was enveloped by an inner wrapping. For this wrapping, various quartz ampoules, Al capsules or CrNi steel capsules were used (Table 2).

Assuming a 20 W target heat generation and 60°C primary coolant temperature, the following target temperatures resulted:

— 150°C with single packaging of the target;
— 350°C with double packaging of the target;
— 600°C with triple packaging of the target.

Since an efficient emergency cooling system was available, a loss of primary coolant flow resulted in no problems for the IICs. Inside the C channels, the heat generation in the IICs from gamma radiation ranged from about 1.7 W/g to roughly 2.3 W/g for light targets and to about 3.3 W/g for heavy targets (Pb) when operating the RFR at 10 MW. The heat generation by n-scattering was negligible.

Special care was taken when a new type of irradiation, sample or capsule was to be licensed. The first irradiation of each had to undergo an assessment by the responsible head of department, the radiation protection officer of the RFR and the officer for reactor safety of the RFR. From 1985 on, all clients were briefed on all provisions by an irradiation checklist [10] or by performing, as before or on request, counselling interviews.

TABLE 2. INNER PACKAGING DIMENSIONS AND TOLERABLE INNER PRESSURE FOR IICs

Type of inner packaging	Diameter (mm)	Length (mm)	Wall thickness (mm)	Tolerable inner pressure (MPa)
Quartz ampoules	5	60	1	2
	7.5	60	1	2
	10	75	1	2
	12	80	1	2
	12	130	1	2
	15	100	1	2
	25	100	1	1
	30	90	1	1
	47	120	1	1
Al capsules	10	50	1	0.2
	12	105	1	0.2
	20	90	1	0.2
	30	110	1	0.2
	54	120	1	0.2
CrNi steel capsules	9	70	0.25	5
	11	80	1	10

Upon RFR final shutdown, its installations and facilities for radioisotope production became dispensable. Except for the DORA and ERIKA facilities, all others became nuclear waste. The two DORAs and ERIKA were transferred to Řež, Czech Republic, or Budapest, Hungary, for further use.

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3.5. COMPARISON OF REACTOR AND CYCLOTRON PRODUCTION OF MEDICALLY IMPORTANT RADIOISOTOPES, WITH SPECIAL REFERENCE TO ^{99}Mo/^{99m}Tc, ^{64,67}Cu AND ^{103}Pd

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1. INTRODUCTION

Radionuclides find application in many fields, however, their major medical use is for both diagnosis and therapy. Modern in vivo diagnostic studies are generally carried out via emission tomography, i.e. by introducing a radionuclide in the body and measuring its distribution in an organ from outside the body (in vivo studies). For this purpose, two imaging techniques are used, namely, single photon emission computed tomography (SPECT) and positron emission tomography (PET). The former involves the use of a radionuclide that emits either a single or a predominant gamma ray and imaging done via a movable gamma ray detector. Some of the important SPECT radionuclides are ^{99m}Tc ($T_{1/2}$ = 6.0 h), ^{123}I ($T_{1/2}$ = 13.3 h) and ^{201}Tl ($T_{1/2}$ = 72.9 h). ^{99m}Tc is produced using nuclear reactors and is most widely used. ^{123}I and ^{201}Tl are cyclotron products and are used for specific purposes. In PET, on the other hand, a positron emitting radionuclide is used and the gamma quanta emitted in the annihilation of the positron are registered using an array of detectors. The most important PET radionuclides are ^{11}C ($T_{1/2}$ = 20.4 min), ^{13}N ($T_{1/2}$ = 10.0 min), ^{15}O ($T_{1/2}$ = 2.0 min) and ^{18}F ($T_{1/2}$ = 109.8 min). They are mostly produced using a cyclotron. SPECT is rather slow and non-quantitative but is comparatively cheaper. It is therefore widely used, both in advanced and developing countries. In contrast, PET is fast and quantitative, but rather costly. As yet it is established only in developed countries.

Therapy with radionuclides involves both external and internal use. The externally used radionuclides, such as ^{60}Co ($T_{1/2}$ = 5.27 a), ^{137}Cs ($T_{1/2}$ = 30.1 a) and ^{192}Ir ($T_{1/2}$ = 73.8 d), are well established and are produced using nuclear reactors. Internal therapy with radionuclides is based on the use of low range but highly ionizing radiation, such as soft X rays, Auger electrons, low energy β^- rays and alpha particles. Some of the commonly used therapeutic radionuclides are ^{32}P ($T_{1/2}$ = 14.3 d), ^{89}Sr ($T_{1/2}$ = 50.5 d), ^{90}Y ($T_{1/2}$ = 64.0 h), ^{103}Pd ($T_{1/2}$ = 17 d), ^{131}I ($T_{1/2}$ = 8.0 d), ^{153}Sm ($T_{1/2}$ = 46.3 h) and ^{188}Re ($T_{1/2}$ = 17.0 h). Most of them are produced in a nuclear reactor. In recent years, the increasing success in tumour

targeting has placed more demands on the development of new therapeutic radionuclides [1].

As is well known, the production of radionuclides is carried out using nuclear reactors [2, 3] as well as cyclotrons [4]. The research reactor produced radionuclides generally have excess neutrons. They mostly decay by β^- emission and are particularly suitable for internal radiotherapy. Some of them decay by isomeric transition (IT) and are used in diagnostic studies. The cyclotron produced radionuclides, on the other hand, are often neutron deficient and decay mainly by electron capture (EC) or β^+ emission. They are thus suitable for diagnostic studies. In particular, the short lived positron emitters needed in PET investigations are obtained only via low energy charged particle induced reactions at a cyclotron, the reactor being incapable of delivering those radionuclides, except for ^{18}F, which can be produced in small amounts but is rather contaminated with tritium. It should be mentioned that both reactors and cyclotrons are now extensively used for medical radionuclide production. Today, more than 250 research reactors worldwide are in operation and most of them are at least partly utilized for radionuclide production purposes. Similarly, there exist more than 250 cyclotrons in various parts of the world, many of them in hospitals, which produce short lived radionuclides for medical use. Thus the radionuclide production science and technology, both at research reactors and cyclotrons, has become a very important feature of modern nuclear medicine.

Although the production of radionuclides at both reactors and cyclotrons has some common features, such as chemical processing of the irradiated material and quality control of the separated product, there are still several distinct aspects, specific to production at the two facilities. These relate to nuclear reaction data, targetry for high current irradiations at cyclotrons and the level of radioactivity involved in reactor irradiation. Since an accelerated charged particle impinging on a material loses energy very sharply, it is not meaningful to use an average cross-section of the nuclear reaction; instead, a complete understanding of the excitation function is required. A second characteristic is that the loss of energy of the charged particle generates significant heat in the target material. The high current production irradiations thus demand sophisticated methods of heat dissipation. As far as the level of radioactivity is concerned, it is often much higher in reactor production than in cyclotron production of radionuclides. Furthermore, reactor production often utilizes the fission process, which involves handling large amounts of radioactivity as well as a large variety of radionuclides.

In this section, the production of four important radionuclides at cyclotrons and research reactors is discussed. One of them (^{99m}Tc) is very well established, but the other three are emerging radionuclides with respect to

their medical application. This comparative study should demonstrate the complementarity of nuclear reactors and cyclotrons regarding the production of medically important radionuclides.

2. THE RADIONUCLIDES ^{99}Mo/^{99m}Tc

The most commonly used radionuclide in medicine in general, and in diagnosis via SPECT in particular, is ^{99m}Tc. It decays via IT to ^{99g}Tc, emitting a single gamma ray of 141 keV energy. Due to its relatively short half-life of 6.0 h and due to the absence of corpuscular radiation, the dose caused to the patient is minimized; thus this radionuclide is ideally suited for diagnostic examinations. It is readily available through a generator system, the parent radionuclide ^{99}Mo ($T_{1/2}$ = 65.9 h) having been previously produced and purified.

2.1. Production in a nuclear reactor

The radionuclide ^{99}Mo ($T_{1/2}$ = 65.9 h) is produced via two major routes:

— ^{98}Mo(n,γ)^{99}Mo;
— ^{235}U(n,f)^{99}Mo.

The (n,γ) process does not have a very high cross-section and the product formed is of low specific radioactivity. Consequently, an Al_2O_3 column, loaded with this irradiated material to form a generator, can be used only for a short time, since the breakthrough of Mo from the column becomes serious after a few elutions. A high flux reactor, if available for such irradiations, would give rise to much higher specific activities and thus would lead to much better performance of the generator.

The fission process is the best method of ^{99}Mo production, although it entails extensive radiochemical work. Since alpha particle emitting actinides are also involved, the demand on the purity of separated ^{99}Mo is very stringent. Nonetheless, the method is very well developed. Hundreds of curies (TBq quantities) of no-carrier-added ^{99}Mo are produced at a few reactor sites and distributed all over the world. The radioactive material is loaded on Al_2O_3 columns for making generators, which are then distributed to almost every corner of the globe. Because of its availability and high reliability, generator produced ^{99m}Tc has gained a very important place in diagnostic nuclear medicine. A periodic elution of the column with normal saline leads to the removal of the daughter ^{99m}Tc, formed during the waiting period, in the form of $Na^{99m}TcO_4$. The eluted solution is used for labelling several organic compounds

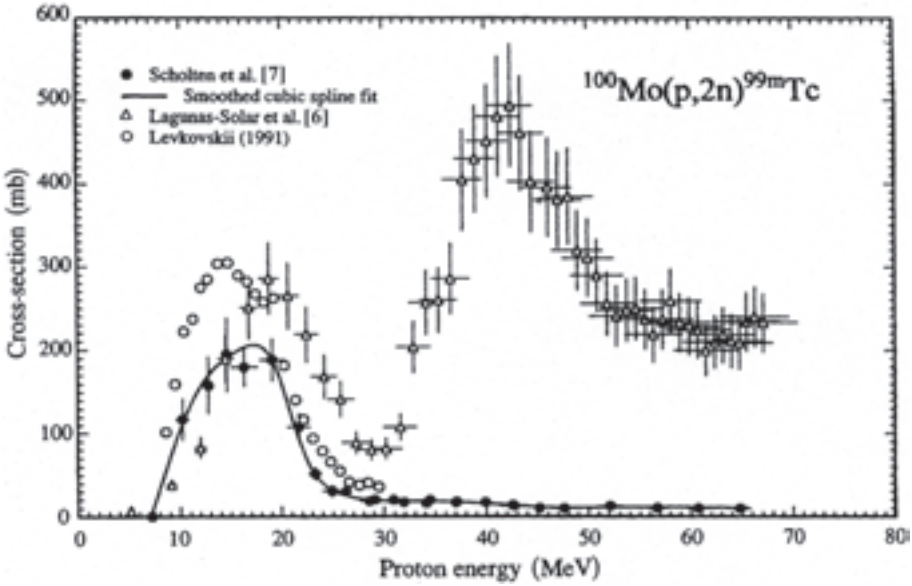

FIG. 1. Excitation function of the $^{100}Mo(p,2n)^{99m}Tc$ reaction [7]. The strong peak at about 42 MeV reported in Ref. [6] was not observed by the authors of Ref. [7].

with ^{99m}Tc, which find application in investigating several organ functions [5]. As mentioned, ^{99m}Tc is very widely used. Despite the great progress in PET studies in recent years, performed in several hundred PET centres established worldwide, in vivo scanning using ^{99m}Tc radiopharmaceuticals still constitutes more than 70% of all nuclear medical diagnostic investigations.

2.2. Possibility of production at a cyclotron

In view of the danger of closure of nuclear reactors, which would jeopardize the production of $^{99}Mo/^{99m}Tc$, some basic studies were done to ascertain whether this generator pair could be produced at a cyclotron. A group in California, United States of America, measured cross-sections [6] for proton induced reactions on molybdenum of natural isotopic composition, and the data were extrapolated to ^{100}Mo of high enrichment. The results are shown in Fig. 1. The formation of ^{99m}Tc was assumed to occur via two processes, i.e. $^{98}Mo(p,\gamma)^{99m}Tc$ plus $^{100}Mo(p,2n)^{99m}Tc$, and it was claimed that medium sized cyclotrons could partly or wholly replace reactors, as far as the production of $^{99}Mo/^{99m}Tc$ was concerned. In an IAEA sponsored study, done mostly at Jülich, Germany, some key measurements were performed [7] using several cyclotrons. The cross-section of the $^{98}Mo(p,\gamma)^{99m}Tc$ reaction, measured using

99.5% enriched ^{98}Mo, was found to be negligibly small (<0.2 mb) over the whole investigated energy range of 6–45 MeV and the results of measurements of 97.4% enriched ^{100}Mo are given in Fig. 1. The peak in the cross-section at about 17 MeV is due to the ^{100}Mo(p,2n)^{99m}Tc reaction. This was attributed earlier [6], wrongly, to the ^{98}Mo(p,γ)^{99m}Tc process. The second peak at about 42 MeV reported earlier [6] could not be observed in the work done in the INC laboratory at Jülich [7]. Presumably, it originated from some impurity. The INC study led to the conclusion that ^{99m}Tc can be produced at a cyclotron in small amounts for local use only. The amount of ^{99}Mo produced is small and its specific radioactivity rather low. There is thus no substitute for reactor produced ^{99}Mo/^{99m}Tc generators.

Despite the negative remarks mentioned regarding the possibility of ^{99m}Tc production at a small or medium sized cyclotron, it should be pointed out that the accelerator production of ^{99}Mo/^{99m}Tc is not completely uninteresting. If a high energy and high intensity proton accelerator were available that could be utilized to produce spallation type neutrons, the fission of ^{235}U could be used to produce ^{99}Mo in large quantities.

3. THE RADIONUCLIDES ^{64}Cu AND ^{67}Cu

The radionuclide ^{67}Cu ($T_{1/2}$ = 61.8 h), emitting low energy β^- particles (E_{β^-} = 0.6 MeV), is almost ideally suited for endoradiotherapy. The shorter lived ^{64}Cu ($T_{1/2}$ = 12.7 h) emits low energy β^- and β^+ particles and has a half-life suitable for slow metabolic studies using PET. Furthermore, ^{64}Cu radiopharmaceuticals are promising for cerebral and myocardial perfusion studies and for labelling antibodies.

The radionuclide ^{64}Cu was produced previously via the ^{63}Cu(n,γ)^{64}Cu reaction, utilizing thermal neutrons in a research reactor. Due to the rather low specific radioactivity of the product, some use has been made of the ^{64}Zn(n,p)^{64}Cu reaction with fission neutrons [8]. However, the yield is low and contamination from ^{67}Cu cannot be avoided. A better method is the ^{64}Ni(p,n)^{64}Cu reaction [9], which can be used at a small sized cyclotron. The yield and purity of the product are high, although the highly enriched target material is rather expensive [9]. Production via this route has been successfully developed, including an efficient recovery of the enriched target material [10]. This radionuclide is now produced in GBq quantities at a cyclotron and used extensively in radioimmune therapy, especially in the USA.

The radionuclide ^{67}Cu is also occasionally produced via the ^{67}Zn(n,p)^{67}Cu reaction, utilizing fission neutrons [11]. However, both the yield and quality of the product are rather low. It is now most commonly produced with protons of

TABLE 1. CALCULATED YIELDS OF NO-CARRIER-ADDED ^{64}Cu AND ^{67}Cu VIA REACTOR AND CYCLOTRON PRODUCTION ROUTES

Product	Production route	Optimum energy range	Yield at end of bombardment (EOB) (MBq)
^{64}Cu	^{64}Zn(n,p)	FS (10^{14} n·cm^{-2}·s^{-1})	88/g·h
	^{64}Ni(p,n)	12 → 9 MeV	236/μA·h
^{67}Cu	^{67}Zn(n,p)	FS (10^{14} n·cm^{-2}·s^{-1})	12.5/g·h
	^{68}Zn(p,2p)	70 → 50 MeV	16.9/μA·h
	^{70}Zn(p,α)	18 → 8 MeV	2.0/μA·h

energies above 50 MeV [11, 12]. A recently reported method of production makes use of the ^{70}Zn(p,α)^{67}Cu process over the energy range of 12 to 21 MeV [13]. However, the feasibility of the method to produce large quantities of ^{67}Cu has hitherto not been demonstrated.

A comparison of the production routes of ^{64}Cu and ^{67}Cu in no-carrier-added form, both at a nuclear reactor and a cyclotron, is given in Table 1. If a high current target (about 30 μA) is available at a cyclotron, the batch yields of both radionuclides are much higher than those in reactor production. It is evident that the production of these two emerging radionuclides at a cyclotron is advantageous.

4. THE RADIONUCLIDE ^{103}Pd

The radionuclide ^{103}Pd ($T_{1/2}$ = 17.0 d) decays almost exclusively by EC to ^{103m}Rh ($T_{1/2}$ = 56.1 min). The latter de-excites through a heavily converted internal transition. As a result of both processes (EC and IT), X rays and Auger electrons are emitted which are ideally suited for internal therapy. Therefore, the radionuclide is being increasingly used in the treatment of prostate cancer.

Originally ^{103}Pd was produced via the ^{102}Pd(n,γ)^{103}Pd reaction in a nuclear reactor. Due to the relatively low abundance of the target isotope ^{102}Pd (1.02% in natural Pd), the specific radioactivity achieved is rather low. Increasing the enrichment of ^{102}Pd leads to enhanced specific radioactivity of ^{103}Pd but still the product is used for internal therapy with caution due to two reasons:

— Toxicity of Pd;

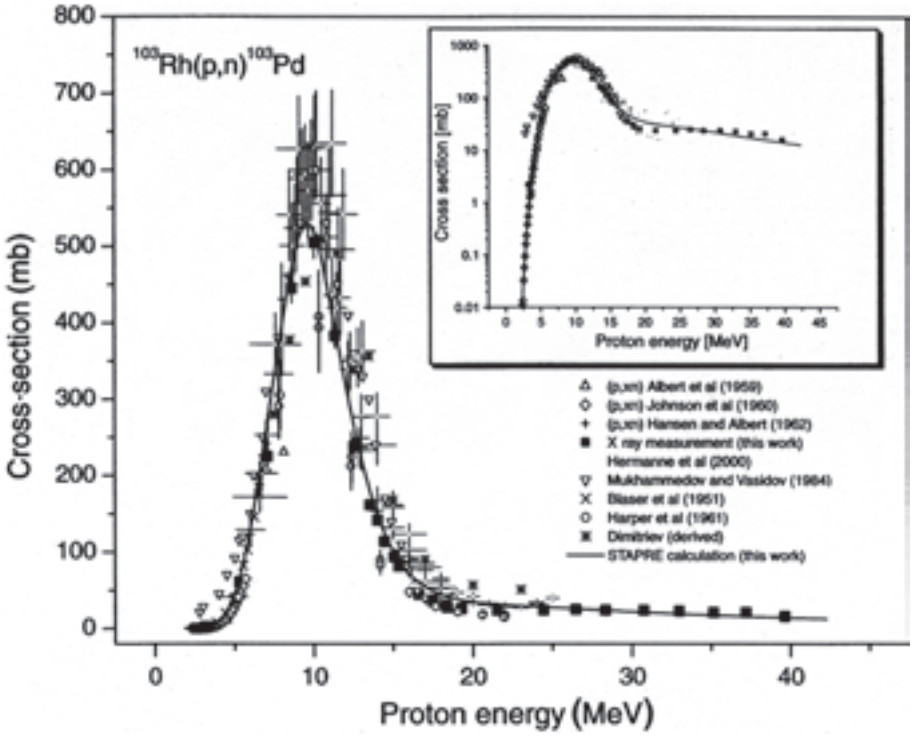

FIG. 2. Excitation function of the $^{103}Rh(p,n)^{103}Pd$ reaction with all the available data obtained via neutron counting and activation measurement. For comparison, the nuclear model calculation using the code STAPRE is also shown. In the inset, the data are shown on a logarithmic scale to depict the energy dependence in the threshold region (cf. Ref. [14]).

— Strong absorption of soft radiation (X rays and Auger electrons) in the Pd matrix.

In contrast to reactor production, the production of ^{103}Pd at a cyclotron is very effective. The most commonly used process is the $^{103}Rh(p,n)^{103}Pd$ reaction. A detailed study reported recently from the INC laboratory at Jülich [14] established the cross-section database for this reaction. The results are shown in Fig. 2. Evidently, the reaction cross-section is rather high so that a sufficiently high yield of ^{103}Pd can be obtained even at a small sized cyclotron. Furthermore, the product is of high specific radioactivity. A comparison of the reactor and cyclotron production of ^{103}Pd is given in Table 2.

The target technology and chemical processing methods have been well established to obtain this radionuclide in high purity and sufficient quantity. To

TABLE 2. REACTOR VERSUS CYCLOTRON PRODUCTION OF ^{103}Pd

Parameter	Reactor production	Cyclotron production
Nuclear reaction	^{102}Pd(n,γ)^{103}Pd	^{103}Rh(p,n)^{103}Pd
Cross-section	3.2 b with n_{th}	503 mb at 10 MeV
Yield of ^{103}Pd	30 MBq/g·h[a]	660 MBq/100μA·h[b]
Radionuclidic purity	Low[c]	Very high
Specific radioactivity	Low	High
Thin sample preparation[d]	Not possible	Possible through chemical separation

[a] Calculated for a flux of 10^{14} n·cm^{-2}·s^{-1}.
[b] Nominal current used in production runs.
[c] ^{109}Pd impurity one week after EOB: 34%.
[d] Necessary to suppress the self-absorption effect to a minimum.

date, about 20 cyclotrons have been installed in the USA with the exclusive aim to produce ^{103}Pd. It thus constitutes a relatively large enterprise.

5. CONCLUSION

As discussed above, the short lived β^+ emitters ^{11}C, ^{13}N, ^{15}O and ^{18}F, used extensively in PET studies, can be produced only at a cyclotron. The SPECT and therapeutic radioisotopes, on the other hand, are produced at both reactors and cyclotrons. The four specific cases considered in this work revealed that ^{99m}Tc, the most commonly used SPECT radionuclide, is obtained from a generator system loaded with ^{99}Mo which is produced via the fission of uranium in a nuclear reactor. The two emerging radionuclides ^{64}Cu and ^{67}Cu are best produced at a cyclotron. Similarly, ^{103}Pd, a radioisotope finding increased application in the treatment of prostate cancer, is produced more advantageously at a cyclotron than at a reactor. However, many other therapeutic radioisotopes are commonly produced at nuclear reactors. These examples demonstrate that reactors and cyclotrons are complementary facilities for medical radionuclide production. The cyclotron produced radionuclides are comparatively more costly. In the future, if high thermal neutron flux is available at research reactors, many more therapeutic radionuclides (e.g. ^{153}Sm and ^{186}Re) could be produced with higher specific radioactivity. Similarly, if a harder neutron spectrum is available to facilitate the (n,p) process, many more therapeutic radionuclides could be produced with good yields.

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Part 4

NEUTRON TRANSMUTATION DOPING

4.1. TRANSMUTATION DOPING: AN OVERVIEW

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1. BACKGROUND

Neutron transmutation doping (NTD) is the process of creating non-radioactive impurity isotopes, also known as dopants, from the host atoms of semiconductor materials by irradiation with thermal neutrons and subsequent radioactive decay. The purpose of adding dopants, either by NTD or alternative chemical means, is to alter the resistivity of the semiconductor material to produce the desired electrical characteristics in the fabrication of semiconductor components. The technique to dope silicon by neutron transmutation was discussed in the 1950s and 1960s by Lark-Horovitz [1] and then by Tanenbaum and Mills [2], and significant commercial use was launched in the 1970s. It is particularly useful for adding small amounts of dopants to semiconductor materials for achieving low resistivity with high spatial uniformity of dopant in the semiconductor material.

NTD is the preferred method over conventional, chemical means for introducing dopants in semiconductors, in the manufacture of small, discrete low power devices, and especially in the manufacture of power electronics that need to operate at high temperatures, voltages and currents. Nuclear research reactors are the ideal source for performing the doping process. The growing demand for such semiconductor materials, therefore, presents opportunities and challenges for existing research reactor facilities desiring to operate in a commercial, high throughput service for fee operation. For new facilities in the design or planning stages, the demand for such semiconductor materials presents an opportunity to design the reactor and experimental facilities from the ground up in order to achieve the high quality and throughput required.

To that end, it was decided to include NTD as a topic for publication in this Compendium, to allow the assimilation of available information about the operation and utilization of research reactor facilities for this purpose.

NTD is applicable to silicon (Si), gallium arsenide (GaA), gallium phosphide (GaP), germanium (Ge) and selenium (Se) semiconductor materials. For the purposes of this paper, discussion is limited to NTD of silicon, the most widely used semiconductor material.

2. THE PROCESS OF NEUTRON TRANSMUTATION DOPING

Silicon has three natural isotopes: ^{28}Si (92.2%), ^{29}Si (4.7%) and ^{30}Si (3.1%). The basic process of doping Si is by creating phosphorous (P) atoms by the absorption of thermal neutrons in ^{30}Si, to create the unstable radioisotope ^{31}Si. This decays by beta emission to the stable isotope ^{31}P.

The primary and secondary nuclear reactions are:

$^{30}Si(n,\gamma)^{31}Si$ —2.62 h→ ^{31}P ($\sigma_c = 0.19$ b)

$^{31}P(n,\gamma)^{32}P$ —14 d→ ^{32}S (induced radioactivity)

$^{28}Si(n,2n)^{27}Si$ —4.2 s→ ^{27}Al (negligible)

$^{28}Si(n,\alpha)^{25}Mg$

$^{29}Si(n,\alpha)^{26}Mg$

Therefore, the introduction of the dopant, phosphorus, is by thermal neutron capture in ^{30}Si, transmuting it to the unstable radioisotope ^{31}Si. This subsequently decays to the stable isotope ^{31}P by beta decay, with a half-life of 2.62 h. These phosphorus atoms in the silicon act as electron donors, creating an n-type semiconductor material. Less than 10 ppm of ^{31}P need to be created from the transmutation process in order to achieve the range of resistivities required for usable semiconductor components. For most applications, a final resistivity of 50 ohm-cm yields silicon that can be used in the manufacture of a majority of components.

3. FACILITIES FOR PERFORMING NTD IN RESEARCH REACTORS

In general, high throughput facilities processing in the order of tonnes of silicon per year, the sophisticated irradiation rigs and support facilities outlined in the following discussion, with a significant degree of automation, would probably be needed, whereas for smaller quantities that may be processed at smaller power research reactors, the degree of sophistication and automation can be commensurately reduced. The basic requirements for establishing an NTD production facility in a research reactor are as follows:

— A highly thermalized spectrum to reduce the radiation damage from fast neutrons in the semiconductor crystal lattice. A high thermal to fast

neutron ratio is desirable, such as those obtained with well moderated neutron spectra (D_2O, Be, graphite). A thermal to fast neutron ratio of at least 7:1 is desirable.

— To that end, the neutron spectrum must be well established, so that the ratio of thermal to fast flux in the irradiation location can be well characterized. In this regard, it should be noted that:
  - It is the reactor operator, as the irradiator, who must guarantee the neutron fluence received which is then correlated against the required resistivity;
  - The requirements for NTD, especially the need for a constant, steady flux over an extended period of time, may conflict with other utilization programmes.

— A location in the reactor where the above flux conditions can be met, such as beam ports, reflector channels, or other places in the reflector region where space is available for high throughput operations:
  - New reactors being planned or designed should undertake the inclusion of special, properly designed NTD irradiation facilities if it is planned that a high throughput programme for a semiconductor programme will be part of the utilization. Generally, for medium to high power research reactors (>5 MW), NTD can be a significant element of utilization;
  - For existing research reactors, the resources to plan and install new NTD irradiation facilities can be significant, so such plans should be carefully thought out, especially in terms of the ability to provide the product quality expected from the research reactor.

— The irradiation facility must be designed for ensuring high radial and axial homogeneity in the doping process, including the use of neutron flux flatteners.

— Due to the relatively long irradiation times to obtain the required resistivity, stable reactor operations over extended periods are required.

— Space will need to be provided — and resources will need to be expended — for related in-reactor as well as out-of-reactor facilities and capabilities to support an NTD programme. These include:
  - Ingot preparation, loading and unloading facilities;
  - Ingot heat treatment facilities to anneal out the effects of fast neutron radiation damage;
  - Radioactivity and resistivity measurement facilities. These include the capability to accurately characterize (map) the neutron flux, as well as on-line measurements of flux during irradiation, for example, by the use of self-powered neutron detectors (SPNDs);

- An established quality assurance plan and detailed quality control procedures to ensure that the customer is receiving the desired and contracted for product quality;
- Since customers of the doped semiconductors are likely to be commercial, for-profit entities — as opposed to research institutions — the capability to perform irradiation planning and scheduling is necessary to ensure that customers can expect the timely receipt of a quality product.

4. ADVANTAGES AND DISADVANTAGES OF NTD

Typically, the advantages of NTD for semiconductor materials over competing means for the addition of dopants are as follows:

— Improved axial and radial uniformity (homogeneity) of dopant in the crystal. This is due to the fact that isotopes under transformation (e.g. ^{30}Si) are distributed uniformly in the target, and to the long range of neutrons in the target materials;
— Precise control over the desired resistivity. Precision target doping of 3% or better can be achieved;
— Lack of a microresistivity structure in the crystal.

Good spatial uniformity is a distinct advantage for achieving high current carrying capability in semiconductors. These advantages of low resistivity semiconductors have led to substantial improvements, for example, in the fabrication of high voltage power rectifiers and thyristors.

NTD is not, however, without its disadvantages. The major ones are:

— Induced radioactivity in target materials which may require relatively long cooling times before the doped materials can be returned to the customer.
— Radiation damage in the crystal lattice by the fast neutrons in the spectrum. This creates localized damage clusters which can degrade the semiconductor device. For this reason, a high thermal to fast neutron ratio is considered essential, as noted previously. Since some fast neutrons will inevitably be present, their effect cannot be entirely eliminated from NTD, but is overcome by the thermal annealing treatment of the ingots prior to device fabrication, a process which will remove the fast neutron damage effects.

— The high cost of conducting an irradiation programme can be a disadvantage in being cost competitive with other doping mechanisms (NTD can be as much as 50% more expensive than chemical doping means). For most research reactors, therefore, an NTD programme will need to be balanced with other utilization programmes to avoid the NTD procedure being too costly to conduct.

5. THE NTD MARKET

The market for silicon doped by neutron transmutation was reinvigorated by new applications late in the 1990s and early in this decade, and has now stabilized at approximately 100 t/a. There are five major producers which share the world market: three in Japan and two in Europe. The persistent crisis in electronics (overproduction and sales at low prices) is limiting the development of silicon doped by neutron transmutation whose cost, as noted previously, is approximately 1.5 times that of the traditional doping means.

It is estimated, at the time of this publication, that wafers cut from ingots doped by neutron transmutation represent a turnover of approximately €100 million. The cost of NTD using research reactors represents approximately 10% of the sales price of wafers, or approximately €10 million.

6. SUMMARY

Included in this Compendium are four contributions from research reactors with experience in commercial NTD:

— In paper 4.2, Alberman and Blowfield provide additional information on technical issues related to NTD, along with brief descriptions of irradiation set-ups currently or previously utilized for NTD;
— In paper 4.3, Kuntoro and Hastowo describe the experience of attempts to establish NTD at RS-GAS, a 30 MW multipurpose research reactor in Indonesia;
— In paper 4.4, Strydom and Louw describe NTD utilization at SAFARI, a 20 MW multipurpose research reactor in South Africa;
— In paper 4.5, Nielsen and Hegaard describe the NTD experience at DR-3, a 10 W multipurpose research reactor in Denmark.

Not represented by the contributions in this Compendium are NTD programmes that are — or have been — carried out to various degrees at other low, medium and high power research reactors.

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4.2. A REVIEW OF SILICON NEUTRON TRANSMUTATION DOPING AND ITS PRACTICE AT FRENCH AND BELGIAN RESEARCH REACTORS

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1. INTRODUCTION

Silicon and its applications in electronics are part of our everyday life, for example, in cellphones, smart cards and computers. It is less known that silicon is also extensively used in industry for equipment operating at high temperatures, at high voltages and high currents. The latter application is also found in household equipment, such as white goods and digital programmable air conditioning systems.

Besides neutron transmutation doping, there are two other methods of doping silicon. The most common method is chemical doping, in which gas charged with doping agents, such as phosphorous or boron, diffuses into a silicon ingot during the manufacturing process inside a vacuum furnace. Whereas such chemical doping results in a relatively uniform distribution of the boron, variations of 30% or more are observed with phosphorus. The second alternative process is ionic doping, in which the doping agent is introduced directly into the silicon wafer.

If high quality components are required, it is necessary to resort to neutron transmutation doping, as this method makes it possible to create a homogeneous phosphorous dopant and thereby generate the desired electrical characteristics uniformly throughout the silicon.

The neutron transmutation process as applied to silicon was first reported by Lark-Horovitz in 1951 [1], but at the time its use was limited to a small number of research projects and to the production of very high resistivity silicon for nuclear particle detectors.

This paper reiterates most of the information contained in Refs [2, 3] and discusses the latest technical issues concerning the subject.

FIG. 1. Ingots of silicon.

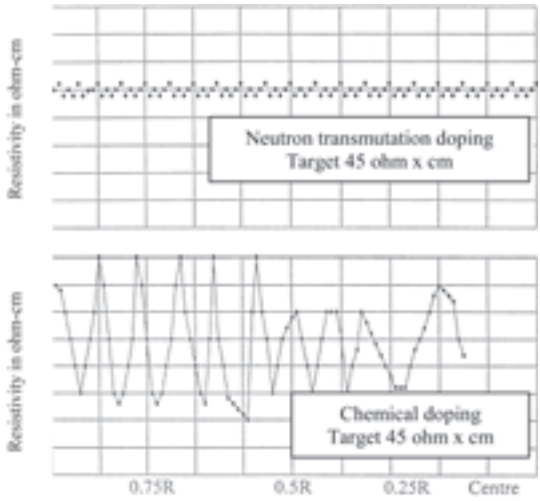

FIG. 2. Variation of radial resistivity after neutron transmutation and chemical doping, respectively.

1.1. The advantage of neutron transmutation doping

Figure 1 shows silicon ingots ready for neutron transmutation doping. The advantage of neutron transmutation doping (NTD) of silicon is that it produces a very uniform dopant of phosphorous throughout the crystal (Fig. 2). After annealing, the resistivity of the crystal, in both the radial and axial directions, is typically within an accepted tolerance of ±5%, whereas a silicon crystal which is doped by chemical means (the cheaper, conventional process, widely used in the silicon industry) has a substantially greater deviation. More regular distribution of the doping agent means that wafers of larger diameters can be used to fabricate high power electrical components that reliably perform to their design specification without breakdown due to the formation of 'hot spots'.

NTD of silicon is used when the manufacture of small discrete components requires many thousands of identical pieces of n-type silicon to be cut from the same wafer. This is extremely important in the mass production of electronic components that need to have exactly the same electronic properties. In larger sizes, the uniform distribution of the doping agent can meet the operational requirements for components transporting high electric power. The main existing applications of silicon doped by neutron transmutation are indicated in Table 1. The corresponding resistivity ranges are compiled in Table 2.

TABLE 1. APPLICATIONS OF SILICON DOPED BY NEUTRON TRANSMUTATION

Power component	Typical applications
MOSFET	Power supplies for computers, TVs, hi-fi equipment, car controls, etc.
IGBT	Inverters and controls for elevators (switch/inverter circuitry), industrial motors (reversing switch controls), air conditioning systems, etc.
Rectifiers and turn-off thyristors	Electrical drive controls and inverters (reversing switches), industrial motors (inverters), etc.; power distribution systems, etc.

TABLE 2. RESISTIVITY RANGES FOR ELECTRICAL COMPONENTS

Resistivity (Ωcm)	Component
15–40	Power transistors Automobile diodes Low voltage thyristors Diode stacks Charge coupled circuits Silicon controlled rectifiers
40–100	Rectifiers Avalanche diodes Insulated gate bipolar transistors
100–1000	High power thyristors Power diodes
5000–20 000	Nuclear detectors

1.2. The market

It is estimated that, in terms of value, three quarters of the market for NTD of silicon is for electrical power components and the remaining quarter is for low power semiconductors. The market for silicon doped by neutron transmutation grew strongly during the 1980s but slipped back in the early 1990s as a result of the introduction of an improved chemical doping technique referred to as ‘premium’, offering a resistivity variability of ±15% instead of ±30% that was obtained with the former conventional chemical doping method.

2. MONOCRYSTALLINE SILICON

Ultrapure polycrystalline silicon (procured from India and Brazil) is used, containing very few impurities after a chemical purification treatment (involving the production of trichlorosilane). For the neutron transmutation doping market, the polycrystalline form is commonly converted into monocrystalline silicon by the float zone (FZ) method, producing ingots with diameters of up to 150 mm. Using this technique, the polycrystalline silicon is moved longitudinally through a high frequency electromagnetic field (inside an argon atmosphere) which creates a local molten area, thus enabling the entrapment of impurities and the creation of the monocrystalline structure. FZ crystals, which are produced in lengths of up to 500 mm, are then irradiated in a nuclear research reactor to transmute some of the silicon atoms into phosphorous atoms. After annealing to repair damage to the crystal lattice caused by irradiation, the crystals are sliced, lapped, engraved, polished and delivered in the form of diffused or epitaxial wafers.

Diffused wafers are typically converted into electrical power components such as thyristors, and epitaxial wafers into charge coupled devices. FZ silicon is the material most commonly used for large discrete power components, some of which are made from silicon doped by neutron transmutation. Alternatively, single crystal silicon can be produced by magnetic Czochralski (MCZ) pulling, i.e. pulling a crystal from a bath of molten polycrystalline silicon. Checking for residual process impurities is generally carried out at a research reactor by activation analysis, which is capable of detecting one part per billion.

It should be noted that the monocrystalline silicon intended for neutron transmutation doping is chemically pre-doped to provide an initial uniform state with a resistivity in the 2000–5000 Ωcm range. This initial state can be of n-type or p-type, depending on whether the impurity is an electron donor or an acceptor.

3. NEUTRON IRRADIATION

3.1. Principles of NTD of silicon

Silicon has three natural isotopes: ^{28}Si (92.2%), ^{29}Si (4.7%) and ^{30}Si (3.1%). During neutron irradiation, ^{30}Si atoms are transmuted by neutron capture to the unstable radioisotope ^{31}Si. With a radioactive half-life of 2.6 h, it subsequently decays to the stable isotope ^{31}P by beta particle emission.

The phosphorous atoms act as electron donors, resulting in the creation of an n-type semiconducting material. Only a small number of the ^{30}Si atoms —

TABLE 3. MAIN REACTORS USED FOR SILICON DOPING (2002)

Moderator	State	Reactor	Thermal power (MW)	Diameter of ingot (in) (Y = yes; N = no)				
				3 (76.2 mm)	4 (101.6 mm)	5 (127 mm)	6 (152.4 mm)	>6
D_2O	Australia	HIFAR	10	Y	Y	Y	Y	Y
D_2O	France	ORPHEE	14	Y	Y	Y	Y	N
D_2O	Japan	JRR 3M	20	Y	Y	Y	N	N
D_2O	Norway	JEEP 2	2	Y	Y	Y	N	N
D_2O	Republic of Korea	HANARO	30	Y	Y	N	N	N
H_2O	Sweden	R2	50	Y	Y	Y	N	N
H_2O	Belgium	BR 2	60	Y	Y	Y	N	N
H_2O	France	OSIRIS	70	Y	Y	Y	N	N
H_2O	South Africa	SAFARI 1	20	Y	Y	Y	N	N
H_2O	USA	MURR	10	Y	Y	N	N	N

around 1–10 ppm — need to be transmuted to produce the range of resistivities shown in Table 2.

3.2. Nuclear reactors

Neutron transmutation doping was carried out on an industrial scale from the mid-1970s at Risø, Denmark, and at Harwell, United Kingdom. The facilities there were closed later but other research reactors obtained the necessary know-how and the production equipment needed to meet the ongoing world demand.

There are about ten centres in the world with research reactors that irradiate silicon for manufacturers of silicon wafers. The main centres are located in Australia, Belgium, France, Japan, Norway, South Africa, Sweden and the United States of America. Table 3 lists the main reactors used for silicon transmutation doping in a heavy water (D_2O) and a light water (H_2O) environment, respectively.

The research reactors mentioned have a thermal power in the range of 2–100 MW and can all (except JEEP 2) produce more than 5 t/a of irradiated material. The type of research reactor can be important, as only thermal

neutrons can introduce the desired dopant. A research reactor moderated/reflected with light water generates a harder spectrum than one with a heavy water environment. A high fast neutron flux gives rise to enhanced dislocations in the silicon crystal lattice compared with a research reactor moderated and reflected with heavy water. In most cases, however, the dislocation damage can be eliminated or considerably reduced by annealing the irradiated crystal under controlled conditions for temperature and time.

3.3. Neutron transmutation doping checks

In research reactors, the perturbed thermal neutron flux density is usually measured with a flux scan at the beginning of each reactor cycle. Small cobalt foils (dosimeters) are placed along the axial centre of a special set of reference silicon crystals. By measurement of the activated cobalt foils, the perturbed neutron flux density can be calculated. This technique is also used to 'on-line' calibrate neutron flux measurement devices such as self-powered neutron detectors (SPND), which are also known as collectrons.

During irradiation, the concentration of the doping agent (C_D) produced by the neutrons in the silicon is determined by the number of atoms of ^{30}Si that are transmuted into ^{31}P, which can be written as:

$$C_D = N\sigma\Phi_t \text{ [atoms/cm}^3\text{]} \quad (1)$$

where

N is the number of atoms of ^{30}Si in the initial silicon material;
σ is the effective cross-section of neutrons for ^{31}Si generation (b);
Φ_t is the neutron fluence (n/cm^2).

For silicon with a density of 2.33 g/cm^3, an atomic weight of 28.086 and an isotope ratio of 3.09 wt% for ^{30}Si, N results in:

$$N = 1.544 \text{ E21 atoms/cm}^3$$

The effective cross-section for neutrons has been set at 0.118 b. This value was measured during the NTD development phase at the Harwell laboratory in the United Kingdom.

For consideration of the concentration of the doping agent in the original material (C_S), the total concentration of the doping agent (C) can be written as:

$$C = C_D + C_S \quad (2)$$

Thus, for n-type silicon, the relationship between the concentration C of the doping agent ^{31}P and the resistivity ρ can be written as:

$$C = 1/(\rho\ \mu\ \varepsilon)\ [\Omega cm] \quad (3)$$

where μ is the displacement mobility of electrons in the crystal lattice which can be set at 1350 $cm^{-2} \times V \times s^{-1}$ for silicon and ε is the electron charge (1.6×10^{19} C).

By combining Eqs (1–3), the effective neutron fluence Φ_t for a target of specific resistivity ρ can be written as:

$$\Phi_t = 2.54\ E19\ (1/\rho - 1/\rho_o) \quad (4)$$

where ρ_o is the average resistivity measured in the original material.

After irradiation, the crystals are shipped back to the customer's factory for annealing and carrying out resistivity measurements at the seed (top) and tail (bottom) ends of each crystal. Using this information, the final radial and axial variations are determined, making it possible to calculate the average resistivity resulting from the entire process. The contracts entered into with customers generally specify that axial variations must not be greater than ±5% and that the average resistivity measured must be within ±10% of the specified target values. Resistivity results are regularly reported by each customer, making it possible to adjust the parameters of the irradiation process so that greater accuracy in meeting resistivity targets can be achieved in future.

4. IRRADIATION FACILITIES

4.1. Design

The silicon irradiation devices used for NTD consist of two principal types:

— Vertical/horizontal longitudinally static silicon rotating devices;
— Vertical/horizontal longitudinally dynamic silicon rotating devices.

Both concepts are aimed at exposing silicon crystals to a thermal neutron flux such that they receive a predefined, uniform dose to produce a homogeneous resistivity to a specified target value.

In the case of the longitudinally static type, spreading resistivity variations along the axis of silicon crystals are minimized using various types of neutron flux shaping devices, i.e. metallic neutron absorption screens or light water

jackets, whereas the latter device is designed to traverse the silicon through the neutron flux to achieve this important characteristic of dopant uniformity. However, both systems rotate the silicon to achieve the least possible radial resistivity gradients (RRGs).

High flux research reactors primarily exist for the purpose of performing scientific experiments in support of multifarious fields of 'materials' and (nuclear) 'beam' exploration. Accordingly, their geometric design and operation concept are based on the need to meet these particular criteria, which are often not compatible with the requirements of bulk production NTD silicon schemes.

Therefore, a silicon irradiation position that is technically and economically viable, and the space available to accommodate the necessary in-pile NTD process equipment in close proximity to the reactor core, are the main factors that influence the adoption of a longitudinally dynamic or static design concept.

Operators are motivated to produce NTD silicon because of economic pressures to generate a significant contribution to the operating and maintenance costs of their research reactors. However, attempts to exploit their full commercial potential can conflict with important technical considerations. Compromises are therefore sought between the density of the thermal neutron flux available to maximize production capacity versus the optimum neutron spectrum to achieve an acceptable cadmium ratio plus silicon crystal nuclear heating because irradiation temperatures above 100°C are considered to be undesirable.

4.1.1. Designs with rotation only

Rotating facilities are suitable for in-pool irradiation channels: the ingots are generally placed in a basket by a suction gripper and are introduced vertically into the set-up. This system can be used to introduce and remove baskets without difficulty while the reactor is in operation. Such facilities are used, for example, at the OSIRIS research reactor (light water moderated) and at the ORPHEE research reactor (heavy water reflected), both at Saclay, France (see Fig. 3). Each basket generally contains two ingots either of 250 mm length (OSIRIS) or 300 mm length (ORPHEE).

Neutron flux is at a maximum in the region corresponding to the central plane of the core and diminishes at the edges. It is, therefore, necessary to equip the device with neutron screens to make the flux more regular. At OSIRIS, nickel screens that are thinner at the edges are used.

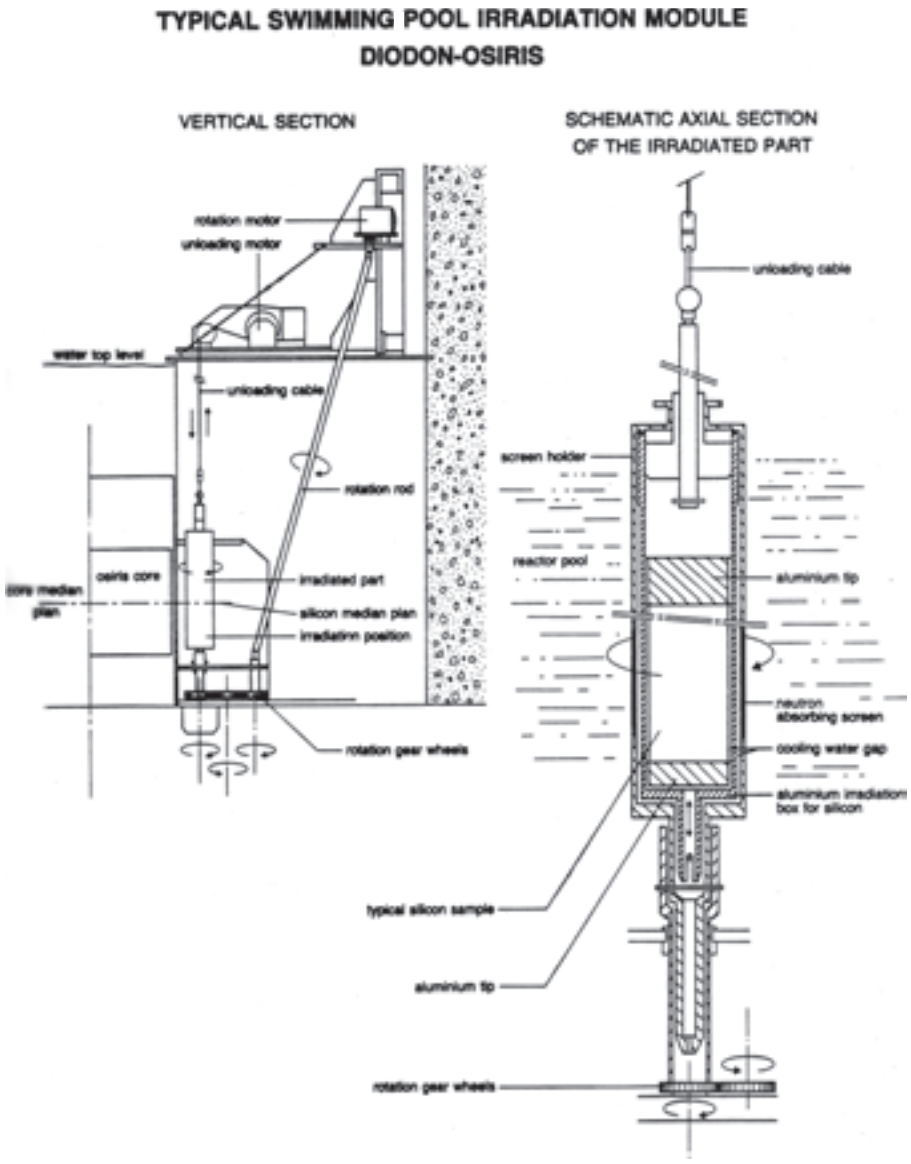

FIG. 3. Silicon irradiation set-up at OSIRIS (DIODON).

At ORPHEE, light water is used that varies in thickness inside the irradiation channels placed inside the heavy water reflector.

Before each irradiation sequence, the thermal neutron flux is systematically measured using a collectron probe to determine the corresponding actual

flux density at the irradiation position. This can vary by 10–20% during the reactor operating cycle because of fuel burnup and the subsequent repositioning of the control rods and the specific core loading with experiments, etc. This measurement is used to calibrate the process collectron instrumentation assigned to the set-up, which is used to monitor the effective duration of each irradiation. The data are fed into a computerized control system. This system is used to stop the irradiation as soon as the required fluence is reached, and also to interrupt the irradiation in the event of a control signal discrepancy. In all cases, raising the ingot out of the neutron flux is automatic.

At the 70 MW OSIRIS reactor, a resistivity of 60 Ωcm corresponding to a neutron fluence of 4×10^{17} n/cm^2 is obtained in around 4 h, whereas at the 14 MW ORPHEE reactor, 12 h are necessary to irradiate batches consisting of two crystals for the same dose (4×10^{17} n/cm^2). At ORPHEE, the neutron flux is much lower in the respective irradiation channels (for 4 or 5 in ingots). Also, due to the axial variation in flux, the positions of the ingots are swapped during irradiation such that the lower ingot becomes the top one halfway through the irradiation process. Adoption of this technique produces a more homogeneous thermal neutron dose.

4.1.2. Designs with longitudinal movement plus rotation

Devices with longitudinal movement plus rotation are designed to enable a capsule containing silicon ingots to pass right through the research reactor core or through the related reflector so as to ensure a highly uniform exposure of the silicon to the thermal neutrons predominantly. Thus, the need for any additional flux flattening screens becomes unnecessary. Such a system (SIDONIE) is in use at the 60 MW BR2 research reactor at Mol, Belgium.

On the other hand, such systems make it difficult to monitor the local flux in real time. It is, therefore, necessary to calibrate them at strategic times throughout the reactor operating cycle (Fig. 4).

Thus, by using Eq. (4) and the results of the flux scan, it is possible to predetermine the speed and the number of passes required for each irradiation to meet the customer's resistivity requirements. A downward pass from the start position of the SIDONIE system, and a return pass upwards at an average speed of 1.8 mm/s, produce an integrated neutron dose of approximately 4×10^{17} n/cm^2. Therefore, a typical thermal neutron dose of 4×10^{17} n/cm^2 to achieve a resistivity of around 60 Ωcm requires eight passes. Since one pass at this speed takes approximately 15 min, a typical irradiation sequence can be performed in about 2 h. As the reactor cycle progresses, silicon irradiation times are calculated, taking into account any changes in the neutron flux

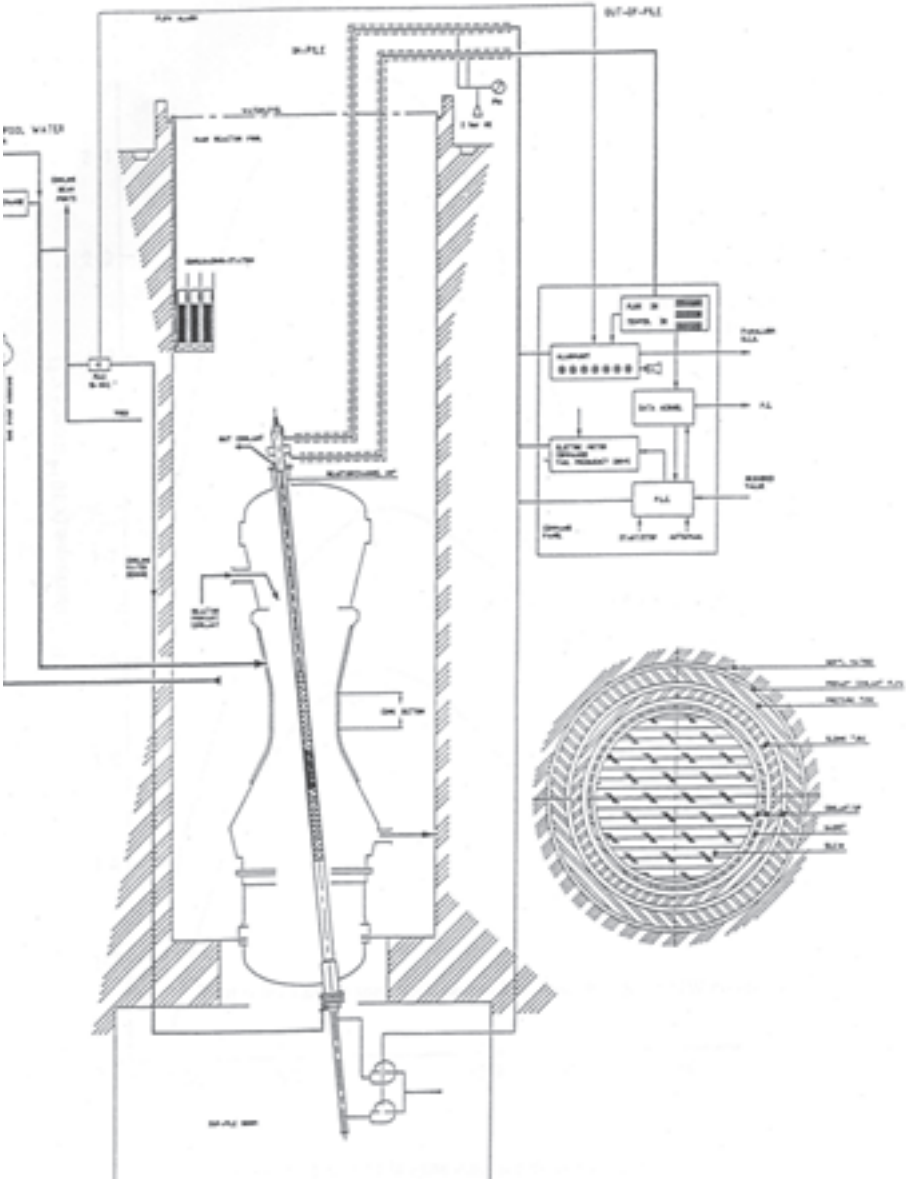

FIG. 4. Set-up for irradiation of silicon at BR2 (SIDONIE).

density (at the core position H2, SIDONIE) due to reactor fuel burnup and/or any changes in the position of the reactor control rods.

The operation of the SIDONIE system is computer controlled to regulate the speed and the number of passes at which the silicon is moved in and out of the thermal neutron flux. The speed of each pass is measured and the system is automatically adjusted so that the final average speed can be adapted to obtain the required thermal neutron dose (and hence the desired resistivity) with a high degree of accuracy.

4.2. Instrumentation

Collectrons, i.e. self-powered neutron detectors, are miniature neutron probes used to locally measure the thermal neutron flux averaged over around 100 mm. Collectrons are, therefore, particularly suitable for the thermal neutron flux density measurements required for accurate silicon doping. Naturally, these probes burn up as they are irradiated and thus it is necessary to recalibrate them regularly.

To do this, dosimetry using the classic Au/Co (foil) activation analysis technique is employed. Typically, Au or Co dosimeters of about 10 mm length are irradiated in the NTD silicon facility in positions corresponding to the location of the collectrons. The dosimeters are made of ultrapure materials and the capture of a thermal neutron produces the radioisotopes ^{198}Au and ^{60}Co, respectively. By evaluating their decay activity in relation to a 'known' standard radiation source, the specific activity for each dosimeter can be derived. When this is equated to their irradiation time, the thermal neutron flux density can be calculated and hence used to calibrate the collectrons. By encapsulating some of the dosimeters in cadmium (at a thickness of 1 mm), it is possible to determine the contribution to the activation of the dosimeters made by epithermal neutrons.

The ratio between the activity of an unshielded detector and one embedded into cadmium is referred to as the 'cadmium ratio' (RCd). The higher it is, the more the neutron spectrum is thermal. Typical values of that ratio used for silicon doped by neutron transmutation are:

— At OSIRIS: RCd = 10;
— At BR2: RCd = 25;
— At ORPHEE: RCd = 500.

4.3. Operating procedure

Each reactor centre uses its own know-how and method of operation for producing NTD of silicon. However, in every case, it is necessary to handle the ingots with great care as they are very fragile.

Furthermore, to meet the desired quality specifications, it is necessary to ensure that the set-up is properly calibrated and that the correct irradiation conditions exist at all times during the irradiation process (for example, flux stability after reactor startup), the latter being established by proper verification.

Immediately after irradiation, the silicon is highly radioactive due to the ^{31}Si atoms decaying to ^{31}P. However, ^{31}Si decays relatively rapidly (with a radioactive half-life of 2.6 h), generally over the subsequent 3 d, after which time it is well below the limits as specified in the international standards for 'exempt' radioactive materials. There is also a small radioactive contribution from a secondary reaction in which ^{31}P atoms capture a neutron and decay to ^{32}Si (radioactive half-life: 14.6 d). Three days after irradiation, the ^{32}P parent atoms account for most of the residual radioactivity as the ^{31}Si atoms decay more rapidly. However, after 3 d, the overall level of radiation generated from this chain is nearly always below the IAEA exemption limit and therefore it can be readily deemed as being 'exempt' material.

For quality control reasons, at any plant performing NTD, the conditions of irradiation are maintained to a predefined standard. Particular care is also paid to marking, manipulating, cleaning, sorting and transporting in accordance with stipulated quality assurance procedures. The feedback from customers is of importance, too, as the customers perform the annealing process and make in-depth resistivity measurements in order to check that the doped material meets the resistivity specifications and that the radial and axial tolerance limits are complied with. These results are of fundamental importance, as the conditions of irradiation can be changed to allow for variations in resistivity after the annealing process, which ensures that the resistivity targets continue to be accurately met. These procedures guarantee a final product being a high quality material that fully meets the customer's specification and is therefore completely suitable for its intended application.

5. CURRENT ISSUES

Up to the mid-1990s, the standard ingot had a diameter of 3 in (76.2 mm). Now three quarters of the demand are for 5 in ingots (127 mm) while 3 in ingots have almost completely disappeared from the market.

This trend towards larger diameters is perfectly normal with silicon of 300 mm diameter being commercially available at the present time. However, large diameters raise questions concerning both the quality and the future of silicon doped by neutron transmutation.

5.1. Quality of doping large diameter ingots

The advantage of silicon doped by neutron transmutation is as explained previously, namely, the homogeneity which makes it possible to maintain the resistivity within ±10% of the target specified. For large diameter crystals, this result is mostly obtained in research reactors that provide a silicon irradiation environment with a heavy water type neutron spectrum. In light water spectra and at resistivities greater than 100 Ωcm, 5 in ingots exhibit substantial detrimental 'edge effects'. This phenomenon, which is still not fully explained, could prohibit the use of light water moderated research reactors for this purpose. This could have a significantly adverse impact on the availability of world capacity because most of the reactors that are currently involved in the NTD of silicon business are of this type.

For 6 in (152 mm) diameter ingots that are irradiated in a light water moderated research reactor, the radial resistivity gradient (RRG) is too large even when they are rotated throughout the irradiation process. The 6 in ingots, which currently make up 10% of the market, are only irradiated in heavy water moderated research reactors or those that can provide a similar neutron spectrum for irradiating silicon. However, the volume of such irradiation devices can be quite disruptive to the operation of a research reactor and its experiments. For this reason, it is necessary to place any 6 in irradiation facility relatively far away from the core. This location results in significantly lower fluxes and a longer irradiation time. Consequently, there is a greater opportunity for the neutron flux density to drift from the measured value, thus necessitating more thorough checking of the neutron fluence throughout the process.

5.2. Transport

As mentioned previously, after the irradiation and the decay period, the residual radioactivity is well below the exemption limits as stipulated in IAEA regulations. However, standards change, and recently the IAEA Regulations for the Safe Transport of Radioactive Material (1985) [4] have been updated by the IAEA Regulations for the Safe Transport of Radioactive Material (1996) [5], which require a more stringent measurement of the radioactive contamination per unit area for obtaining an exemption certificate.

The ultrapure silicon ingot remains the only industrial product which can be returned to the manufacturer in its original packaging after its irradiation in a nuclear reactor. However, with standards becoming ever more onerous and with the increasing international demand for more rigorous checks on any

object comprising a radioactive risk, the long term availability of silicon doped by neutron transmutation may be brought into question.

6. CONCLUSION

The role of NTD of silicon in the semiconductor market for electrical power systems is a major incentive for the development of improved characteristics in terms of fabrication techniques and materials, in order to obtain, as economically as possible, components that are more compact, generate less waste and permit higher power ratings. The market demand for NTD silicon will be met as long as there are research reactors capable of offering a reliable irradiation service of adequate capacity and quality to the customer at a cost that is competitive with high quality chemically doped silicon.

Production on a 'just in time' basis is becoming less and less compatible with the operation of research reactors, which are subject to increasingly stringent safety checks, particularly as many of them are more than 30 years old and therefore may be subject to ever increasing down times and longer outages for refurbishment.

Consequently, cooperation between research reactors will become increasingly necessary to guarantee the continued availability of silicon doped by neutron transmutation for industrial customers.

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4.3. EXPERIENCE WITH NEUTRON TRANSMUTATION DOPING FACILITY OPERATION AT THE RSG GAS REACTOR

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1. INTRODUCTION: THE RSG GAS PLANT

The RSG GAS multipurpose research reactor is an open pool type, light water cooled and moderated plant, using MTR type fuel assemblies with LEU in the form of U_3O_8–Al dispersion; in brief, its features can be seen in Fig. 1,

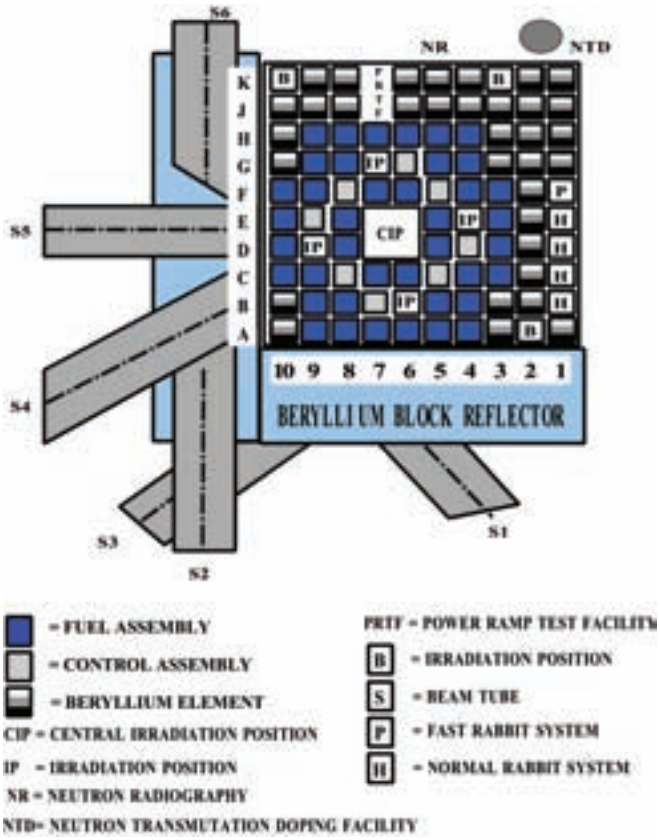

FIG. 1. Core configuration of the RSG GAS reactor.

TABLE 1. MAIN DESIGN PARAMETERS OF THE RSG GAS RESEARCH REACTOR

Reactor power (MW)	30
Number of fuel + control assemblies	40 + 8
Cycle length, MWD	750
Average burnup at BOC, % loss of U-235	23.3
Average burnup at EOC, % loss of U-235	31.3
Average discharge burnup at EOC, % loss of U-235	53.7
Excess reactivity at BOC, cold, without xenon, %	9.2
Reactivity worth of control rod system (8 control rods), %	–14.5
Shutdown margin at BOC, cold, without xenon, %	–2.2

and Tables 1 and 2, originating from Ref. [1]. From 1998 on, the oxide fuel was changed stepwise to silicide-one (U_3Si_2–Al) with the same uranium density of 2.96 g/cm³. The reactor core has 600 mm active height and a rectangular cross-section and is located in the reactor pool at 12.45 m below the pool surface. The reactor core is arranged in a 10 × 10 array at a grid plate and it consists of 40 standard fuel assemblies, eight control assemblies, four in-core irradiation positions and one central irradiation position (CIP), and five reflector irradiation positions equipped with rabbit systems and further irradiation holes in the reflector region. Beryllium elements and a beryllium block surround the core. At nominal power of 30 MW, the core provides thermal neutron fluxes of the order 2×10^{14} n·cm^{-2}·s^{-1}. The reactor also has an out-of-core irradiation facility capable of power ramp testing (Power Ramp Test Facility), a Neutron Radiography Facility and a Silicon Doping Facility was installed later. Finally, there is an array of beam tubes. The RSG GAS research reactor was designed to operate in a cycle mode with a cycle length of 25 d full power and 8 cycles/a.

The core reactivity is controlled by eight control rods with two AgInCd fork type control blades each to be inserted in the eight control assemblies. The control rods hang down from a cantilever placed above the reactor pool. One of the eight is chosen as a regulating rod for automatic control, the others act as a control rod bank. All control rods have the same speed of 0.5 mm/s.

2. DESIGN FEATURES OF THE NTD FACILITY

The Neutron Transmutation Doping (NTD) Facility is located near one of the reflector borders consisting of Be elements (see Fig. 1). The facility is illustrated in Fig. 2. The aluminium irradiation capsule (Fig. 3) was designed to

TABLE 2. IRRADIATION FACILITIES AT THE RSG GAS REACTOR

Name	Position	Application
IPL	CIP (D-6, D-7, E-6, E-7)	PWR/PHWR small fuel bundle test under thermohydraulic conditions of the nuclear power plant
Chouca	IP-2 (D-9)	Creep behaviour test of structural material samples
Cyrano	IP-4 (G-7)	Study of the characteristics of fuel pin and pellet cladding interaction
PRTF	Out-of-core	Power ramp test for short pin of PWR/BWR fuel
NR	Out-of-core	Neutron radiography of fuel
MTR loop	IP–3 (E-4)	Investigation of behaviour of minibundle MTR fuel
Irradiation holes	IP-1 (D-6)	Radioisotope production
	IR-1 (A-2)	General irradiation
	IR-2 (K-3)	General irradiation
	IR-3 (K-10)	General irradiation
Beam tubes	S-1	Iodine loop, I-125 radioisotope production
	S-2	Neutron radiography
	S-3	—
	S-4	Triple axis spectrometry (TAS)
	S-5	Neutron guide to FCD, SANS, HRSANS, HRPD
	S-6	Powder diffractometer (PD)
Rabbit system	B-1, C-1, D-1, E-1, F-1	NAA, irradiation
NTD	Out-of-core	Irradiation of silicon ingot Maximum ingot diameter: 7 in; length: 400 mm

accommodate silicon crystals in cylindrical form with maximum dimensions of 7 in diameter and 400 mm length. Any irradiation capsule is placed into a supporting tube. The average thermal neutron flux at the silicon crystal was calculated to be about 3.3×10^{12} n·cm^{-2}·s^{-1}. The reactivity feedback to the core was calculated to be very small, i.e. less than 0.002% [2]. Rotating the irradiation capsule with a speed of 2 rev./min was designed to eliminate the radial flux gradient at the silicon ingot. Moreover, a stainless steel absorber of specific form (see Fig. 4) was installed outside the supporting tube to shield the neutron flux at the position having higher flux levels, to achieve an axially constant flux. To detect the effective neutron flux, the facility is equipped with a self-powered neutron detector (SPND). The entire facility can be shifted

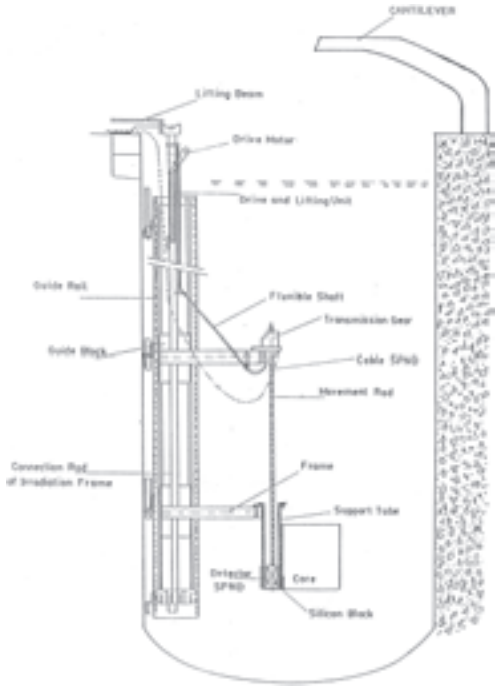

FIG. 2. The NTD Facility at the RSG GAS reactor.

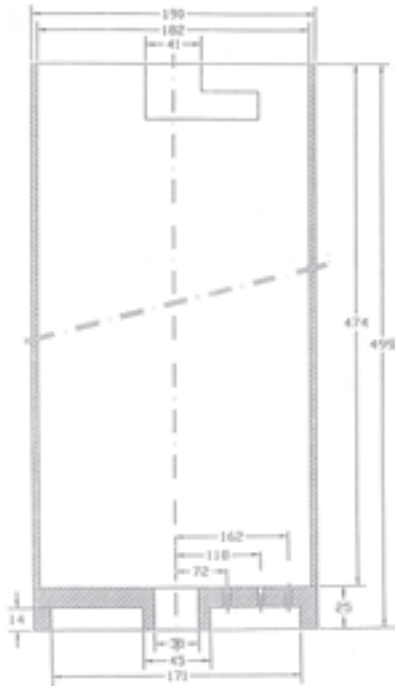

FIG. 3. Irradiation capsule of the NTD Facility.

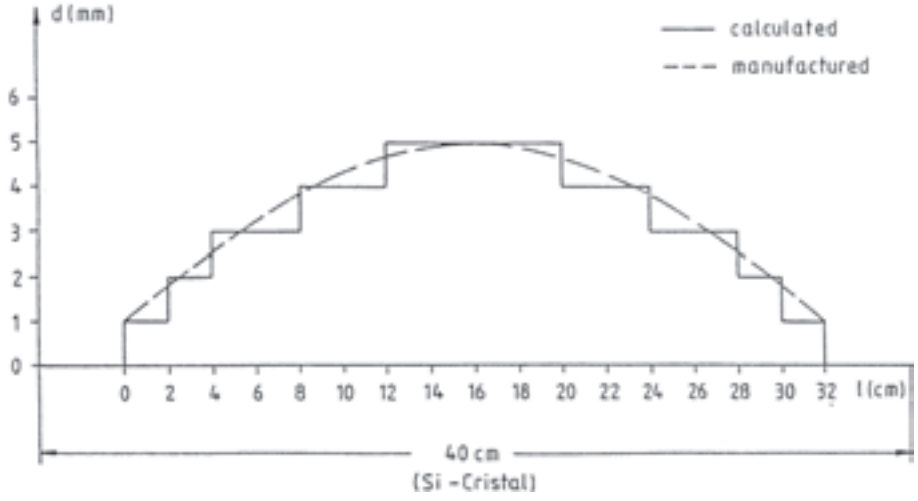

FIG. 4. Shielding effect of the SS absorber screen.

downward during the operation cycle, depending on the actual position of control rod bank. The loading and unloading of the silicon ingots was foreseen during reactor shutdowns.

3. PERFORMANCE OF THE NTD FACILITY

The performance of the NTD Facility has been tested by transmutation doping of a silicon ingot in cooperation with the Shin-Etsu Company, Japan. The irradiation was performed at a power level of 22 MW for 4 d. The reactivity effect of the inserted ingot with its capsule was confirmed to be very small (less than 1 cent). The measured flux at the SPND was about 1.35×10^{11} n/cm^{2}s. After irradiation, the crystals were sent to Japan for measuring the resistivity in terms of Ω cm and its homogeneity. To perform these measurements, the ingots were annealed and cut into wafers, which were later annealed again. The obtained results were as follows [3].

The resistivity was measured at five locations of each wafer at 3 mm from the edge (marked by '3 mm'), at the middle of the radius (marked by '2/R') and at the centre line (marked by 'C'), all as indicated in Fig. 5. The results are presented in Tables 3 and 4 and in Fig. 6 for both the ingots tested, i.e. T1-00001 and T1-00002, respectively.

The results of the resistivity measurement are shown in the tables as follows:

— Rho 1 refers to the resistivity readings of the different wafers sliced off from the annealed ingots;

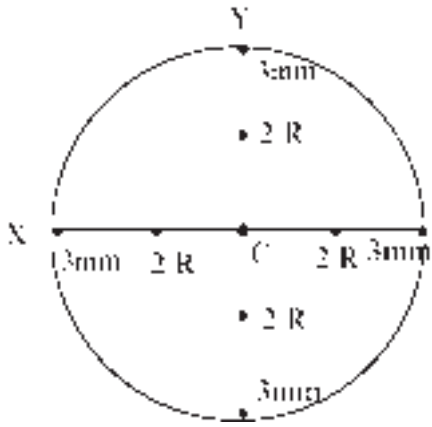

FIG. 5. Measuring points at any wafer of the Si ingot.

TABLE 3. RESISTIVITY OF WAFERS FROM INGOT T1-00001

No.				rho 1						rho 2						SHIFT
				3mm	R/2	C	R/2	3mm	RRG	3mm	R/2	C	R/2	3mm	RRG	(%)
T1-1-1	K	1	X	63.6	63.5	63.9	63.6	59.1	8.1	64.7	66.7	67.3	67.0	66.0	4.0	5.1
			Y	65.3	63.4	63.4	62.8	62.7	4.1	65.0	66.7	66.9	67.1	65.2	3.2	
		2	X	63.6	63.2	62.8	62.7	63.0	1.4	65.8	67.7	67.4	67.6	66.2	2.9	
			Y	59.3	62.8	62.8	62.6	63.2	6.6	67.0	68.2	67.2	67.4	66.2	3.0	
	P	1	X	65.3	63.6	63.1	63.6	65.6	4.0	67.6	68.3	67.9	68.1	67.5	1.2	7.1
			Y	60.0	63.9	63.0	63.5	66.5	10.8	68.9	67.8	67.8	68.8	68.2	1.6	
		2	X	65.3	63.8	62.5	62.5	64.0	4.5	67.2	68.0	67.6	67.8	68.3	1.6	
			Y	62.8	63.1	62.3	62.8	65.7	5.5	68.0	68.5	67.9	67.6	67.9	1.3	
T1-1-3	K	1	X	52.0	64.7	64.3	64.2	65.8	26.5	68.4	69.3	69.4	69.6	69.9	2.2	7.3
			Y	65.8	64.5	64.7	65.1	66.6	3.3	68.9	69.0	69.2	69.2	69.8	1.3	
		2	X	61.1	64.4	64.0	64.0	65.9	7.9	68.2	69.1	68.9	68.6	70.1	2.8	
			Y	66.3	64.1	64.0	64.7	64.5	3.6	69.2	68.4	68.7	69.0	68.2	1.5	
	C	1	X	63.6	60.5	59.9	60.4	51.3	24.0	66.1	65.9	65.9	66.7	66.4	1.2	9.1
			Y	61.1	60.6	60.0		63.3		66.0	65.9	65.5	66.9	66.4	2.1	
		2	X	45.8	60.8	59.9	59.8	61.8	34.9	66.0	66.3	65.3	66.1	67.4	3.2	
			Y	62.9	60.2	59.2	60.6	62.3	6.2	66.9	66.6	65.4	66.5	66.0	2.3	
	P	1	X	58.7	60.4	63.1	61.2	58.3	8.2	61.2	64.9	66.7	63.8	61.0	9.3	5.4
			Y	54.5	60.5	62.9	60.2	59.4	15.4	62.4	64.3	66.6	63.3	60.2	10.6	
		2	X	58.7	60.0	61.8	60.6	59.9	5.3	61.1	64.1	66.1	63.0	61.1	8.2	
			Y	58.5	59.5	62.1	60.2	51.7	20.1	61.9	64.3	66.1	64.2	60.6	9.1	

TABLE 4. RESISTIVITY OF WAFERS FROM INGOT T1-00002

No.				rho 1						rho 2						SHIFT
				3mm	R/2	C	R/2	3mm	RRG	3mm	R/2	C	R/2	3mm	RRG	(%)
T1-2-1	K	1	X	61.3	60.5	60.5	60.3	60.5	1.7	61.9	64.0	64.1	63.4	64.0	3.6	5.1
			Y	51.4	60.8	61.1	60.2	60.3	18.9	62.7	63.9	64.3	64.0	61.2	5.1	
		2	X	59.8	59.0	60.3	59.2	50.3	19.9	62.0	63.9	64.7	64.6	63.7	4.4	
			Y	59.6	59.9	59.6	59.9	60.6	1.7	62.6	63.5	64.4	64.3	62.8	2.9	
	P	1	X	64.2	60.5	60.9	61.5	50.8	26.4							
			Y	63.1	61.2	60.9	62.1	63.3	3.9							
		2	X	54.0	61.4	60.3	60.8	62.8	16.3	64.8	65.4	65.2	65.7	67.0	3.4	7.5
			Y	59.9	60.2	60.4	61.1	64.2	7.2	65.3	66.0	65.2	66.1	66.0	1.4	
T1-2-3	K	1	X	64.5	62.9	62.4	63.0	65.8	5.4	68.0	67.9	66.9	68.0	68.5	2.4	6.7
			Y	65.3	62.6	62.5	63.8	58.0	12.6	70.1	68.8	66.7	67.7	67.6	5.1	
		2	X	64.8	62.7	62.6	63.1	55.6	16.5	67.7	67.7	66.8	67.0	69.0	3.3	
			Y	64.5	62.3	62.7	62.9	65.8	5.6	68.0	67.6	67.2	67.3	66.3	2.6	
	C	1	X	62.8	59.1	59.3	60.5	60.2	6.3	64.9	65.0	64.5	66.2	66.2	2.6	8.1
			Y	61.9	59.0	59.5	59.8	62.4	5.8	65.0	65.2	64.7	65.7	65.6	1.5	
		2	X	45.3	60.1	59.5	59.1	61.3	35.3	65.2	65.4	64.8	64.6	65.7	1.7	
			Y	62.9	59.5	59.3	59.9	63,4	6.9	65.4	65.0	64.7	64.9	64.5	1.4	
	P	1	X	58.8	60.0	64.2	61.3	59.3	9.2	60.6	63.2	66.9	64.6	61.8	10.4	4.0
			Y	58.7	60.1	63.6	61.0	59.0	8.3	61.8	64.5	67.0	63.4	60.6	10.6	
		2	X	58.9	60.8	63.4	61.6	59.5	7.6	60.8	64.3	66.6	64.6	63.1	9.5	
			Y	57.7	61.7	63.2	60.3	59.7	9.5	61.5	64.2	66.6	64.9	61.0	9.2	

— Rho 2 refers to the resistivity readings of the wafers which were annealed as wafers once more;
— Shift indicates the difference in value between rho 1 and rho 2;
— Two different wafers indicated as 1 and 2 in Tables 3 and 4 were sampled per each axial length; axial lengths were characterized by T1-1-1/K, /P and by T1-1-3/K, /C, /P (see Fig. 6);
— At each wafer, resistivities were measured in two directions X and Y, as indicated in Fig. 5, to find any distinctly deviating resistivity values. The resulting resistivity gradient (RRG) is given in both Tables 3 and 4.

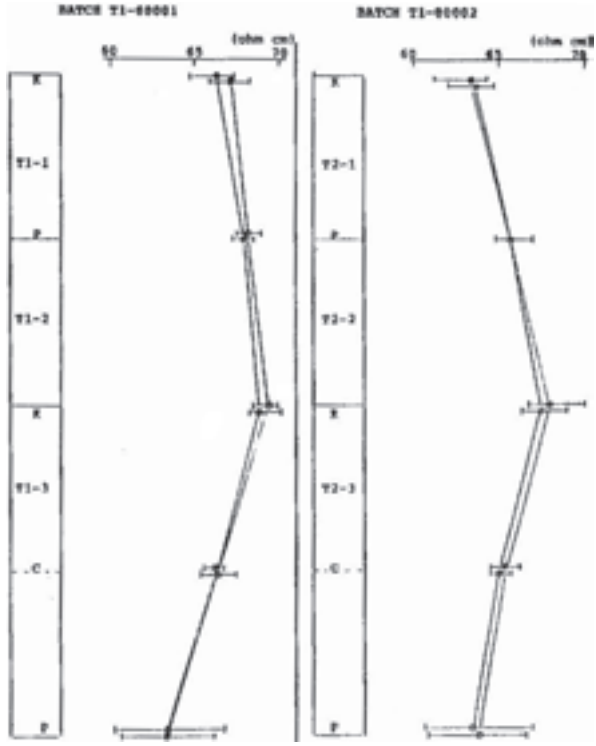

FIG. 6. Resistivity axial distribution of the silicon wafers of ingot T1-00001 and T1-00002.

4. CONCLUSIONS

— Since the NTD location is rather far from the core,[1] the thermal neutron flux has already decreased significantly. A neutron flux at the irradiation facility of the order 10^{11} n·cm^{-2}·s^{-1} is rather low. A flux level that is ten times higher would be preferable. An irradiation time of three or four days per ingot does not pay off economically. Moreover, the difficult handling of the irradiation capsules caused by a high water flow rate challenges the utilization of the facility. Ways out of the latter problems might be:

[1] This location is a consequence of the late design and installation of that facility which had to stay outside the 10 × 10 grid plate entirely because of its axial dimensioning.

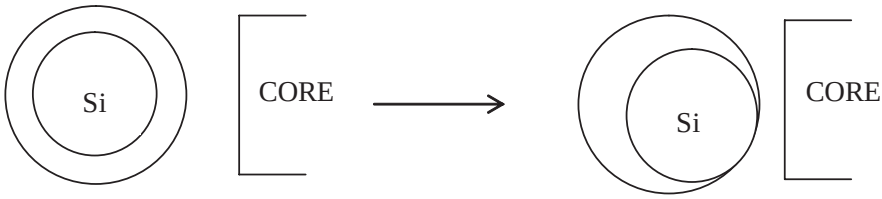

FIG. 7. Proposed modification of the supporting tube of the NTD Facility.

- Change the location of the NTD Facility to a position nearer to the core (not easy to realize due to the need to cut off part of the grid plate, see footnote);
- Remove the excess beryllium of the reflector (this measure is possible only with a shift of the position of the NTD);
- Change the way of centring the ingots with their Al adaptor in order to avoid the effect of too much light water around the ingots when irradiating pieces with diameters less than 7 in, as illustrated in Fig. 7;

— The flux level of non-thermal neutrons and thus, the low cadmium ratio at the position of the ingot will remain unfavourable. However, implanting a heavy water tank for the NTD Facility seems too much of an effort to envisage in terms of the potential business.

— Resistivities were less homogeneous than expected, obviously since around the middle of the axial length there was higher resistivity than near the ends of the ingots. This means that the steel absorber screen needs modification for its proper shielding function. This improvement was not performed by the time of writing of this paper.

— The radial resistivity gradients RRG at both ends of the ingot cylinder, especially at T1-1-3/P and T1-2-3/P, were very high (see the RRG values of rho 2 in Tables 3 and 4) and beyond the level acceptable to the client. The resistivity profiles there show higher resistivity at the wafer centre and lower at the wafer edges. Such an effect may stem from the refraction of neutrons in the dummy silicon or aluminium used around the ingots when irradiating.

— The resistivity values of rho 1 shifted to those of rho 2; when annealing, the wafers showed differences of more than 5%. This means that the damage to the silicon lattice caused by the fast neutrons, which is healed by the annealing, is rather large. This is a typical effect at irradiation facilities in light water moderated plants. Those results confirmed that beryllium reflectors are not as good as heavy water for NTD applications. Light water moderated and beryllium reflected reactors have rather low cadmium ratios (ratio of thermal to fast flux). The RSG GAS reactor supplies a ratio of about 20 measured at the location of the NTD Facility.

For heavy water moderated and reflected reactors, cadmium ratios of 120 can be achieved in the reflector region. That low ratio of the RSG GAS reactor causes crystal defects inside the Si ingots higher than they occur at other plants.

— The resulting resistivities deviated from the target resistivity by more than 30%. This type of problem could be solved by adjusting the applied neutron fluence.

As a conclusion, it can be said that the NTD Facility at the RSG GAS research reactor, as originally designed and installed, does not fully meet its foreseen function, i.e. to supply the quality of semiconductors as required by the market. The resulting cadmium ratio is too low, and a rather low thermal flux at the irradiation position adds to that disadvantage. However, the experience gained with the facility at the RSG GAS reactor can be used for similar facilities at other research reactors that are planned for a similar application.

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[2] SIEBERTZ, O., Neutron Physical Aspects of Design and Operation of the Silicon Doping Facility, Interatom-Notiz, Ident. No. 54.07698.8 (1988).

[3] SHIN-ETSU COMPANY, Report on Resistivity Measurement of Silicon Ingots Irradiated at RSG-GAS Reactor, private communication (1990).

4.4. NEUTRON TRANSMUTATION DOPING OF SILICON AT THE SAFARI-1 RESEARCH REACTOR

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Pretoria, South Africa

1. INTRODUCTION

Ultrapure silicon crystals are doped with dopants such as phosphorous or boron to produce n or p type semiconductor material, respectively. Conventional phosphorous doping of n type silicon is achieved by incorporation of the dopant during crystal formation from the melt. This method leads to microscopic resistivity variations as high as ±15% [1], which are unacceptably high for devices such as rectifiers for high voltage direct current transmission [2].

The neutron transmutation doping (NTD) process is based on the transmutation of ^{30}Si atoms by the capture of thermal neutrons into ^{31}Si, which decays to ^{31}P by the emission of beta particles with a half-life of 2.62 h. A uniform thermal neutron dose distribution over the ingot will therefore ensure a homogeneous distribution of phosphorous atoms throughout the crystal, resulting in uniformly doped n type silicon with accurately predetermined resistivities. A complete description of the NTD process, the techniques and the facilities at research reactors worldwide can be found in Ref. [3].

The SAFARI-1 research reactor is owned and operated by the South African Nuclear Energy Corporation (Necsa). It was commissioned in 1965 and has a design power of 20 MW. Commercial and material research programmes at the reactor are supported with extensive infrastructure, including a fuel fabrication plant, hot cell facilities for the production of medical isotopes, a disposal site for radioactive waste, and a theoretical reactor physics group.

The range of commercial irradiation services on offer at SAFARI-1 has been extended by the installation of the Silicon Irradiation Facility (SILIRAD) in the pool side region of the reactor. The neutronic characterization of the facility was originally carried out with a combination of theoretical neutron transport calculations and experimental flux and spectrum determinations, using multiple foil activation techniques. Commissioning of SILIRAD1 took place in 1992 with a series of trial irradiations, which were performed in close collaboration with an international silicon supplier. Through continued R&D,

it has been upgraded and improved to meet the growing international demand for NTD of silicon.

2. SAFARI-1 RESEARCH REACTOR

The reactor is located at Pelindaba, approximately 30 km west of Pretoria, South Africa. The reactor (tank in pool), which is similar in design to the Oak Ridge reactor (ORR), is light water cooled and moderated with an 8×9 core lattice which currently contains 26 fuel assemblies (active height 600 mm) and six control assemblies, as depicted in Fig. 1. The remaining lattice positions are either aluminium or beryllium reflector elements. The locally produced fuel assemblies consist of 19 flat plates each, composed from uranium–aluminium alloy meat and cladded with aluminium. The reactor vessel is cylindrical in shape, with one flattened side which forms the northern wall of the rectangular core box, thereby providing an easily accessible pool side facility, directly adjacent to fuel assemblies and therefore relatively high neutron fluxes. SILIRAD is positioned in this pool side region.

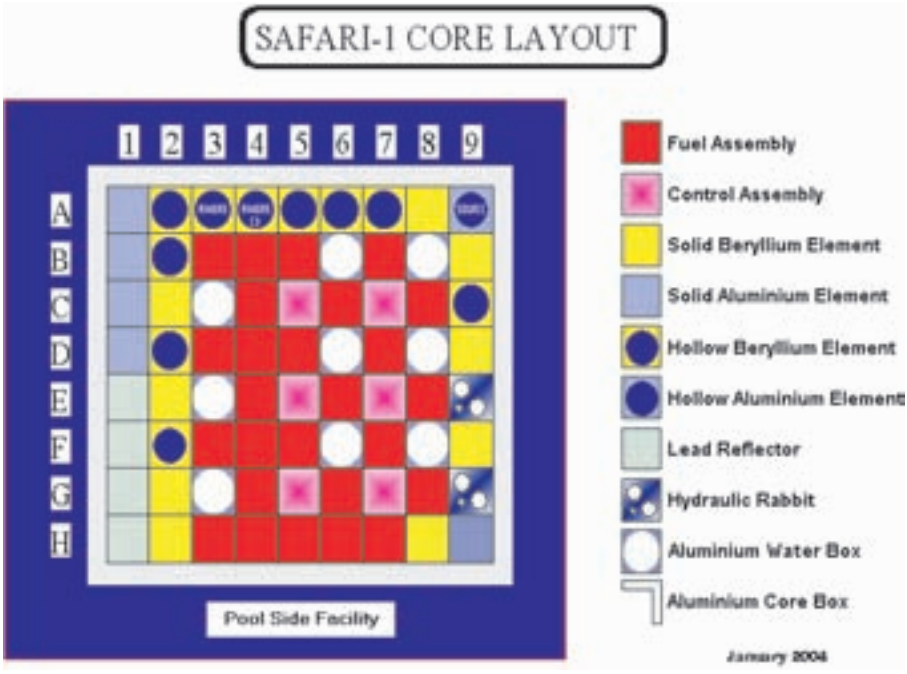

FIG. 1. The SAFARI-1 core layout.

3. CHARACTERIZATION OF THE IRRADIATION POSITION

A theoretical model for the determination of neutron fluxes as a function of energy in the pool side region was developed. This model is based on neutron transport theory methods, utilizing the one and two dimensional computer codes XSDRNPM [4] and TWOTRAN [5]. Appropriate neutron cross-section libraries were prepared with the relevant modules in SCALE 3.1 [6]. These theoretically determined neutron energy spectra are used for the determination of position dependent, one group cross-sections for cobalt activation foils and also ^{30}Si. Thermal to fast and cadmium ratios were also determined.

A series of experiments were performed to obtain a comprehensive set of experimental data, which would serve to neutronically characterize the irradiation position initially. These experiments included flux profile and energy spectrum measurements. The flux profiles were measured with the aid of a rig in which cobalt foils (Cd covered and uncovered) were positioned at appropriate intervals on the horizontal and vertical core centre lines, spanning the region 300 mm east and west of the vertical centre line (see Fig. 2), and 300 mm below and above the horizontal centre line (see Fig. 3). A flux profile perpendicular to the reactor face was also measured (Fig. 4). Thermal flux values reflected in Figs 2–4 were obtained from measurements in the water, at a reactor power of 20 MW.

The neutron energy spectrum was determined by using the SAND II computer code [7] in conjunction with multiple foil activation data. Spectral

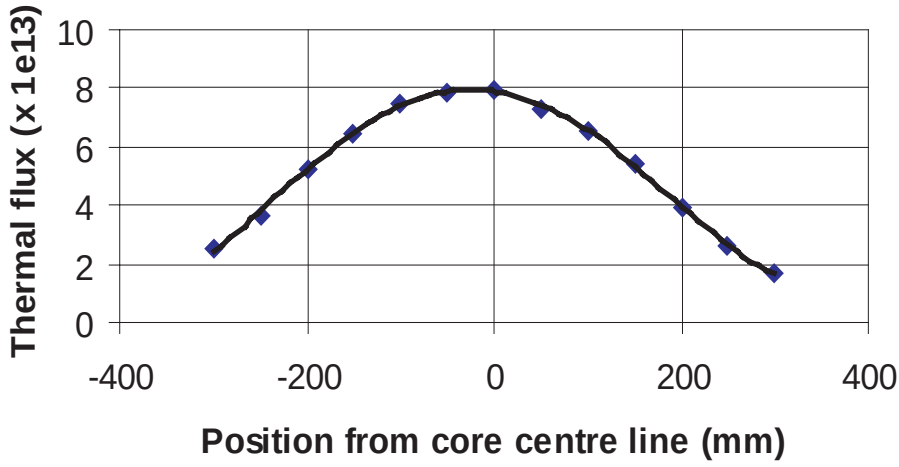

FIG. 2. Thermal flux ($n \cdot cm^{-2} \cdot s^{-1}$) profile as a function of position from east to west.

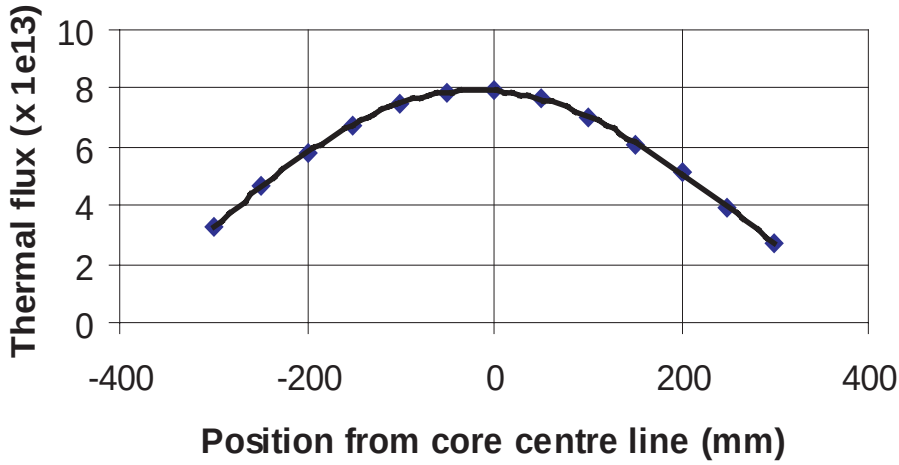

FIG. 3. Thermal flux ($n \cdot cm^{-2} \cdot s^{-1}$) profile as a function of position from bottom (–) to top (+).

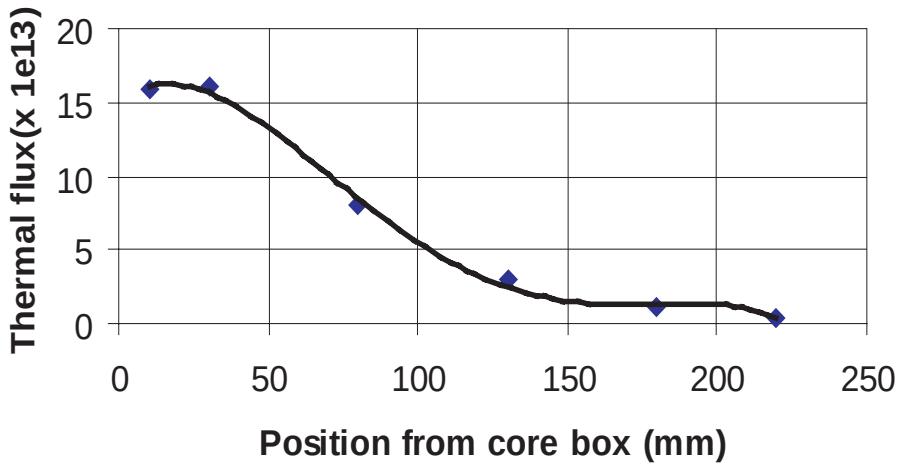

FIG. 4. Thermal flux ($n \cdot cm^{-2} \cdot s^{-1}$) profile as a function of perpendicular distance from core face.

parameters as determined by these theoretical and experimental analyses are summarized in Table 1.

Due to the neutronic characteristics of the silicon and the construction material of the facility, mainly aluminium, as well as the position of the facility in the pool side region, there are no adverse reactivity effects on the core.

TABLE 1. SPECTRAL PARAMETERS APPLICABLE TO THE POOL SIDE REGION

Distance from core face	^{59}Co cross-section (b)	Thermal/fast flux ratio	Cd ratio
10	19.5	5.8	12.5
30	23.64	10.3	22.1
80	26.35	13.9	38.3
130	26.59	13.2	44.9

4. THE SILIRAD FACILITY

On the basis of the theoretical and experimental analyses, an irradiation position was selected.

Provision was made, however, in the design for the easy variation of this position so as to optimize the irradiation conditions. The silicon ingots are loaded into aluminium canisters, which are driven along the horizontal core centre line with a helical motion. The canisters serve as protection against accidental knocks. Even though steep flux gradients are observed in this region (see Figs 2 and 3), the horizontal and rotational motion of the ingot through the flux field ensures resistivity homogeneity without the use of any flux flattening shields. The first facility (SILIRAD1) was designed to accommodate a single stage feedthrough. As the commercial demand grew, this facility was upgraded to a double stage feedthrough.

This double stage facility (SILIRAD2) features two feedthrough systems that can be operated totally independently. The two stages are vertically adjacent to each other with respect to the core face. Both stages are equipped with loading and unloading trays to receive and dispose of the silicon, packed in the aluminium canisters. Neutron flux levels are monitored with five self-powered neutron detectors to provide input for the determination of the required translation speed for each translation stage.

SILIRAD2 is currently designed to accommodate silicon ingots with a maximum diameter of 150 mm and a maximum length of 600 mm. The production capacity is a function of the reactor power level, starting resistivity and target resistivity of the silicon received for transmutation.

5. NTD OF SILICON CHARACTERIZATION

Following the installation of SILIRAD1, a number of trial irradiations were performed with 100 mm diameter ingots, supplied by an international silicon supplier. The irradiated ingots were returned to the supplier for detailed evaluation. According to Ref. [1], the evaluation, irrespective of the silicon producer, consisted of the following:

— Resistivity measurements at each end of the ingot with a four point probe after annealing at 800°C;
— Resistivity measurements at each end of the ingot with a four point probe after annealing at 1200°C. A shift in resistivity would indicate irradiation damage, i.e. too low a thermal to fast neutron flux ratio;
— An axial resistivity profile over the ingot length;
— A radial resistivity profile over the ingot with a four point probe;
— Carrier lifetime determination with photoconductive decay measurements.

The feedback on these parameters was used to establish the ideal irradiation position for the first SILIRAD and to refine the calculational method used for determining the irradiation time. The same evaluation process is used today every time that silicon with different from previous diameters, or silicon from new sources, is received for irradiation.

Some parameters measured during the initial trial irradiation are presented in Table 2.

According to Ref. [3], the temperature during irradiation may influence some of the mentioned parameters. Therefore measurements were conducted with a thermocouple embedded in an aluminium ingot having the typical dimensions of a silicon ingot, to determine the expected temperature during irradiation. A temperature of approximately 80°C was found.

TABLE 2. TYPICAL VALUES OBTAINED FROM SILICON IRRADIATED AT SAFARI-1

Parameter	Value
Resistivity shift between 800°C and 1200°C annealing	2.2%
Axial resistivity profile	3.8%
Carrier lifetime	2700 μs
Accuracy (% difference from target value)	4%

6. PRODUCTION PROCESS

The production process can be roughly divided into pre-irradiation handling, irradiation, post-irradiation handling, radioactivity monitoring and dispatch. All production procedures are controlled by quality assurance procedures, according to ISO-9000 and ISO-14000. Computer software has been developed for irradiation planning and scheduling, and all relevant irradiation history information is stored for future reference in a computerized database.

Preparation of the ingots for irradiation consists of cleaning, inspection for as-received damage and packing into allocated canisters. A dedicated room is allocated for these activities on the beam port floor.

The packed canisters are transferred to a customized storage rack in the spent fuel pool, adjacent to the reactor pool, for storage until irradiation commences. This storage rack can accommodate approximately 80 canisters. For the irradiation process, canisters are transferred under water to and from this storage rack.

Owing to the high purity of the silicon and the clean environment at SAFARI-1, post-irradiation activity is mainly due to ^{32}P, which requires a period of only a few days' decay in order to reach the recommended IAEA activity limits suitable for transport. Due to the buildup of residual activity on the aluminium canisters, a dedicated shielded transfer cask and hot cell are utilized to minimize exposure to personnel. The transfer cask is used to transfer canisters between the storage rack and the hot cell, where the packing and unpacking of silicon are performed.

Upon completion of the irradiation and unpacking from the canisters, the silicon is once more cleaned to remove any possible surface contamination. The cleaned silicon is then transferred to a dedicated counting room, remote from the beam port floor, with a very low activity background. Each silicon ingot is individually monitored for any residual activity and certified as "exempted radioactive material" prior to dispatch.

The complete NTD process was subjected to a probabilistic risk assessment, including a failure mode and effects analysis, with respect to radiological hazards. An independent body, the National Nuclear Regulator in South Africa, granted approval for operation.

Acceptable turnaround times are achieved due to an efficient production line and dedicated personnel, backed by easy access to an international airport, servicing most major destinations with direct flights and effective customs arrangements. A group of approximately six persons are allocated to the operation of the NTD process, with dedicated responsibilities ranging from

pre- and post-irradiation handling, irradiation planning, scheduling and quality assurance.

7. CONCLUSION

The production of NTD of silicon grew from a humble R&D project at SAFARI-1 in 1990 to a full scale commercial programme, as a joint venture between the groups SAFARI-1 Research Reactor and Nuclear Technology Products (NTP) at Necsa. SAFARI-1 acts as an irradiation service provider to NTP, which is responsible for the commercial management of the various irradiation products stemming from the reactor.

Detailed characterization and careful selection of the irradiation position lead to the production of acceptable NTD of silicon. The unique translational and rotational process ensures an almost homogeneous dopant concentration without the burden of flux flattening shields, which would result in unacceptable lower thermal flux levels.

The fact that acceptable NTD of silicon can be produced with a light water moderated reactor, contrary to earlier beliefs [3], may also be attributed to improved crystal growth processes, as well as post-irradiation annealing techniques [1].

Utilizing the pool side of SAFARI-1 for the production of NTD of silicon without impairing the reactor core, leaves the core available for the production of medical isotopes and material research, requiring higher thermal neutron fluxes. The commercial utilization at SAFARI-1 is complemented with state of the art neutron radiography, tomography and diffraction facilities available at the beam ports.

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4.5. THE NEW HORIZONTAL FACILITY FOR NEUTRON TRANSMUTATION DOPING OF SILICON AT DR 3

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1. INTRODUCTION

The DR 3 research reactor built at Risø, Denmark, was a 10 MW heavy water cooled and moderated research reactor of a design similar to the British PLUTO. It had been operating since 1960 and its conversion to LEU (U_3Si_2/Al) fuel started in 1988. Since December 1990, the DR 3 has been running on a full LEU core.

The DR 3 was originally built as a materials testing reactor, but for most of its lifetime it was used as a multipurpose research reactor. With a cold neutron source, six three-axis spectrometers and a small angle neutron scattering instrument, the DR 3 was appointed as a Large European Beam Facility and these neutron beam instruments were intensively used by researchers from Risø and from other European Community countries.

The main production activities at the DR 3 were neutron transmutation doping (NTD) of silicon, radioisotope generation and activation analysis. In its last configuration, the DR 3 had seven facilities for NTD, specified in Table 1.

Figure 1 shows a horizontal cross-section of the DR 3 reactor with the experimental through tubes as well as the vertical irradiation positions. The

TABLE 1. OVERVIEW OF THE SILICON IRRADIATION FACILITIES AT DR 3

Position in DR 3	Year of installation	Size (in)	Coolant	Orientation	Control	Average flux ($n \cdot cm^{-2} \cdot s^{-1}$)
4VGR3	1977–1978	3	Air	Vertical	Manual	3.0 E+12
4VGR5	1977–1978	3	Air	Vertical	Manual	2.5 E+12
7V1	1986	5	H_2O	Vertical	Manual	25 E+12
7V3	1981	4	D_2O	Vertical	Manual	20 E+12
7V4	1983	4	D_2O	Vertical	Manual	30 E+12
7T2	1997	5	H_2O	Horizontal	Computer	17 E+12
7T4	1990	5	H_2O	Horizontal	Computer	17 E+12

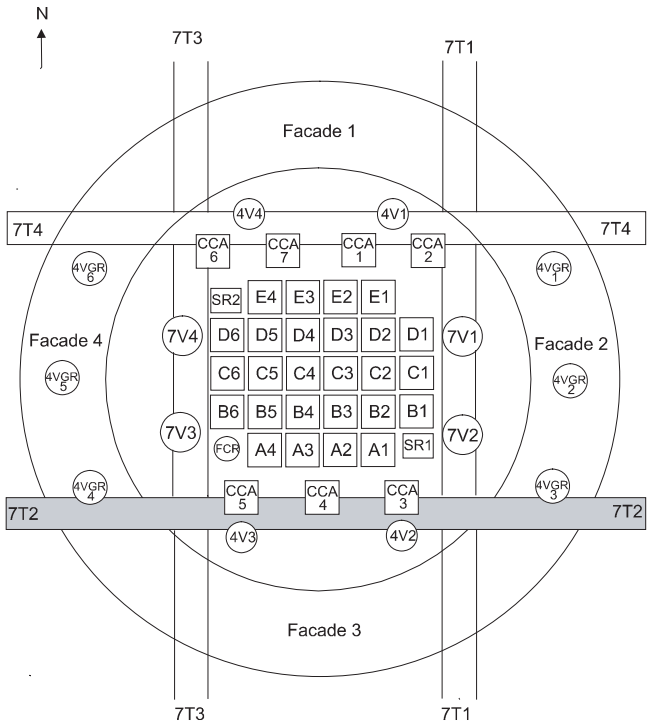

FIG. 1. Horizontal cross-section of the core and reflector area of the DR 3. The reactor block is cubic (not shown in the figure), comprising four facades. Facade 1 faces north. The facades are used as references relative to the reactor core.

shaded tube in Fig. 1 shows where the horizontal NTD facility for silicon doping described in this paper was installed.

2. DESCRIPTION OF THE MECHANICS OF THE NEW HORIZONTAL FACILITY AT 7T2

The new NTD facility [1] was installed in one of the 7 in through tubes and consists of the following components:

- A lift to load and unload aluminium cans containing silicon crystals, placed at Facade 2;
- A large drum in air containing the loading storage with 12 positions in its inner circle and the unloading storage with 24 positions in its outer circle;
- A small drum inside a water tank, which is the irradiation storage;
- An irradiation tube with a guide rod, which is placed at Facade 4.

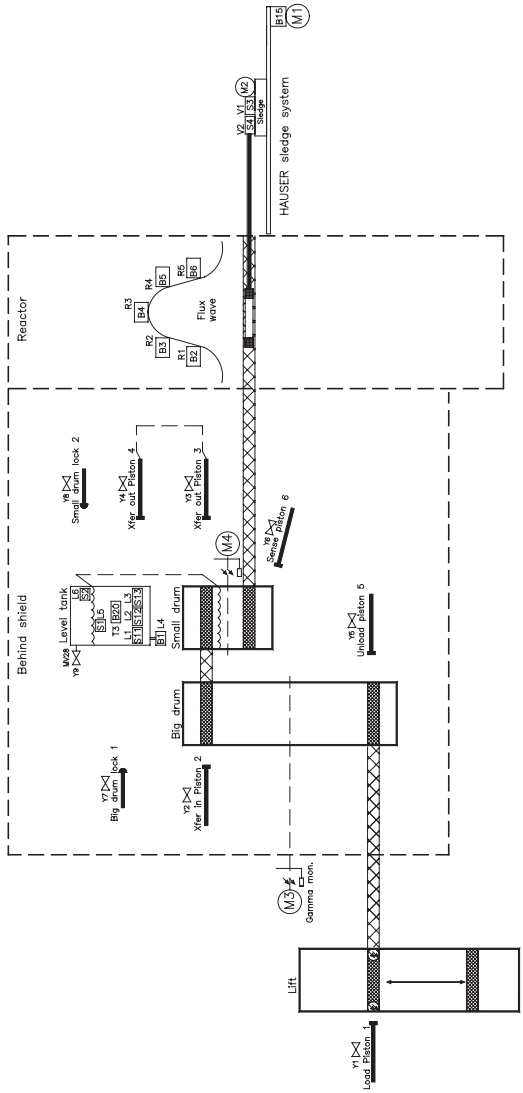

FIG. 2. The new horizontal silicon irradiation facility in 7T2.

A schematic drawing of the facility is shown in Fig. 2.

2.1. In-pile part

The in-pile part of the NTD facility consists of an irradiation tube filled with light water. Between this irradiation tube and the reactor liner there is a 13 mm gap filled with CO_2. In this gap, five beta emitters are placed which

control the irradiation process. Inside the irradiation tube there is a hollow guide rod which provides the rotation of the silicon ingot during irradiation as well as the longitudinal movement through the reactor tank.

2.2. Light water system

The light water system of the NTD facility consists of a level tank, circulation pumps, an ion exchanger placed at Facade 4, a heat exchanger placed in the return pipe, and a tank housing the small storage drum, all interconnected with pipes. There are two independent light water circuits: the main circuit and the indication circuit.

2.2.1. Main circuit

The main circuit acts as hydraulic transport for the can during irradiation and its flow pushes the can against the end of the guide rod. The flow of the main circuit also serves as coolant for the irradiation tube as well as for the can. It is possible to reverse the direction of the flow of the main circuit; such an operation is done:

— Automatically, before irradiation to prevent the can from entering the irradiation tube;
— Automatically, after irradiation to transport the can from the 'cooling position' to the small drum (see Fig. 3);
— Manually, when difficulties are encountered while attempting to get contact between the irradiation can and the guide rod. Reversing the flow for a moment may resolve this problem without extending the irradiation time;
— Manually, in the case of a mechanical failure — during which it is important to remove the can from the reactor core to avoid spoiling the silicon.

2.2.2. Indication circuit

The indication circuit provides water flow through a nozzle placed in a catch at the end of the guide rod. An observed increase in the generated pressure indicates engagement between the can and the guide rod. This circuit also includes a cleaning facility for the water in the system.

2.3. Automated storage

The storage facilities, which consist of two corresponding drums with horizontal axes, shown in Fig. 3, are placed behind a concrete shield.

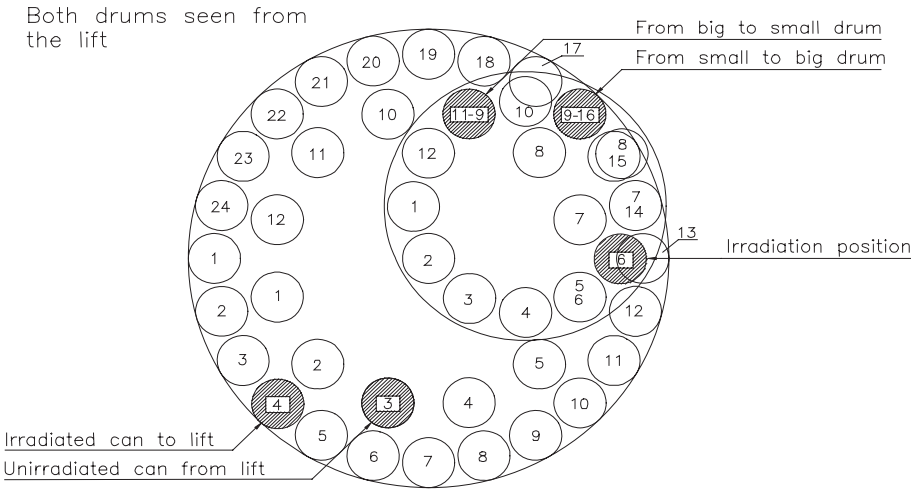

FIG. 3. Corresponding drums of the loading/unloading system (vertical section).

2.3.1. Storage of irradiated cans

After irradiation, the storage of cans takes place in the small drum inside the related water tank, which is shielded with lead. That small drum has 12 positions in which the irradiated cans are stored for two days prior to transfer to the large drum. At two different positions in each case, the storage of irradiated cans is connected to the irradiation tube on one side and to the large drum on the other side. The upper positions of the small drum are above the water level.

2.3.2. The large drum

The large drum comprises two rings for storage: the loading storage with 12 positions at an inner circle and the unloading storage with 24 positions placed at an outer circle. The large drum is dry and the irradiated cans are stored here for the subsequent four days.

2.3.3. Loading and transfer of cans

The transfer between the small and the large drum is done automatically during the process. The loading and unloading of cans from the large drum is performed by means of pneumatic pistons (see Fig. 2).

2.4. Irradiation cans

The diameter of the silicon crystals to be irradiated is 5 in (127 mm) and the maximum length is 500 mm. The aluminium cans are 600 mm long with a diameter of 132 mm. At each end of a can, a 50 mm thick graphite disc is placed to avoid flux depression at the ends of the silicon crystal. Outside the can at both ends, bearings from graphite are placed supporting the rotation of the can. Inside the can, small cobalt wires are installed as monitors to enable checking the dose received.

3. INSTRUMENTATION

The irradiation process is computer controlled and can be performed automatically by the instrumentation, based on preloaded data. During the irradiation, the instrumentation provides characteristic data for the process supervision and documentation.

The instrumentation consists of equipment for surveillance in terms of reactor safety and reactor operation, as well as personnel and system security. The system comprises sensors for pressure, flow, temperature, conductivity, position, rotation, etc, and electronic circuits for actuation of the reactor trip, as well as equipment for signal processing and for control of the facility.

The position of the guide rod is registered in the computer in two ways: by a signal from a Versa Module Europa (VME)[1] microprocessor (see Fig. 4), which controls the guide rod, and by an absolute encoder on the gear shaft of the rod driving motor (HAUSER). The positions of the drums are controlled by two resolvers, one for each drum, and the exact position of each drum relative to the irradiation tube and to the transfer tube is indicated by a disc with holes, which are scanned by a light sensor. Locking bolts lock the drums during transfer; the small drum is also locked during irradiation.

To finally load or unload the cans, a lift is used which corresponds to the large drum.

3.1. VME microprocessor

To link the computer to the mechanical components, the VME microprocessor is used. The function of the VME is to control the hardware, such as

[1] The VME is a flexible, open ended system which makes use of the Eurocard standard.

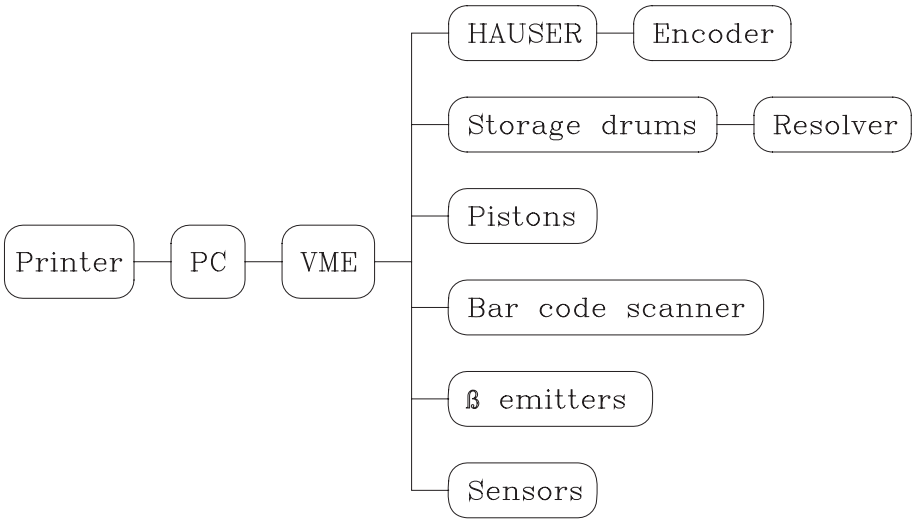

FIG. 4. Overview of the components of the control system.

the pistons, the drums and the position of the guide rod, according to the instructions loaded into the computer. The VME also takes care of the storage statistics; it is programmable and can be read by the computer.

3.2. Software

The computer program, which has been developed at the Risø National Laboratory, provides the interface with the operator. The program can work in two subsequent modes:

— In operation mode, it is possible to start loading and unloading cans and to start an irradiation cycle.
— In service mode, the system can be operated manually between irradiation phases or in the case of a failure. A key is required to enter into the service mode.

The most important parameters as well as the activities performed are continuously shown on the monitor of the computer. Alarms are also listed and shown. In addition, a protocol of the status of the three modes of storage is available.

4. THE IRRADIATION PROCESS

4.1. Start of the loading process

A reactor operator loads a can with silicon crystals into the lift and types the number of the related can into the computer. The lift goes up and passes a bar code scanner which checks whether the number on the can corresponds to the number typed by the operator. In the meantime, the large drum turns to an empty position. At the top of the lift, a piston pushes the can into the empty hole in the inner circle of the large drum (see Figs 2 and 3). To avoid unirradiated cans piling up inside the reactor, containment cans are usually delivered to the reactor four at a time, once a day.

4.2. Loading the cans

After loading the cans, the irradiation process is ready to commence. The reactor operator inserts a floppy disk containing the irradiation data into the computer. One floppy disk usually contains data for four cans. In the computer program, the operator selects the irradiation sequence of the loaded cans and starts the irradiation. The large drum turns until the first can is at the transfer position towards the small drum. At the same time, the small drum has turned such that an empty position is placed at the transfer position. A piston transfers the can from the large to the small drum and the small drum moves the can to the irradiation position. During that procedure, the main water flow in the irradiation tube is directed towards the drum (the direction is reversed) to prevent any can from being pushed into that tube. When the small drum is at the irradiation position and locked there, and the guide rod is in its 'starting position' (see Fig. 5), the water flow is reversed and directed away from the drum (forward direction). Now the can is pushed into the irradiation tube until it reaches the hollow guide rod and, thus, the indicating pressure is raised. Finally, the guide rod starts to rotate.

4.3. Irradiation

Starting from the 'starting position' (see Fig. 5), the guide rod together with the can passes the active core towards the opposite side with a maximum speed of 4.5 mm/s. Having arrived at the 'start of irradiation' position there, the guide rod reverses its direction of movement, now moving towards the large drum and herewith the actual irradiation starts. The indicating flow is still maintaining a pressure such as to ensure and to monitor the contact between the hollow guide rod and the can.

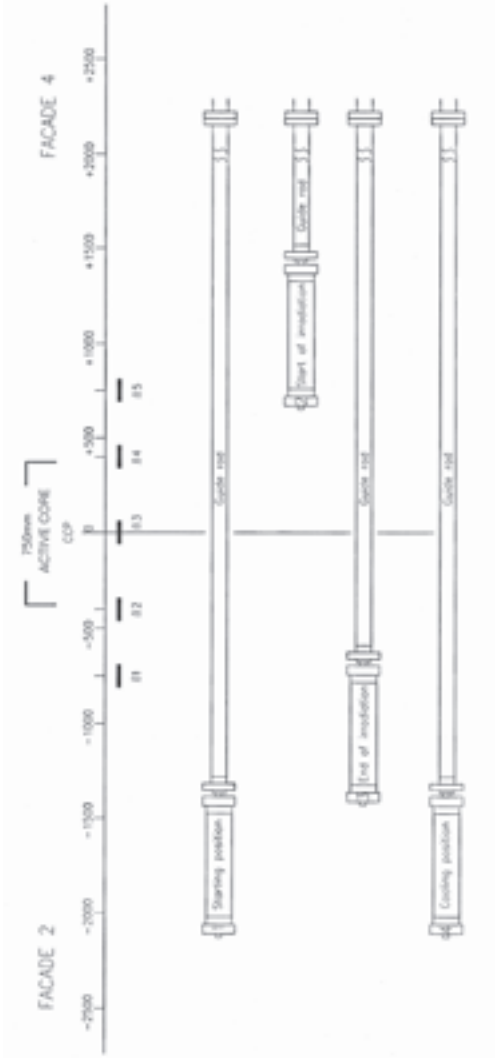

FIG. 5. The various positions of an irradiation can.

The speed of the guide rod is computer controlled, based on the signals from the beta emitters (see Section 5). After having passed the active core (as shown in Fig. 5) and having arrived at the 'end of irradiation' position, the speed is increased to its maximum of 4.5 mm/s until the can has reached the 'cooling position'. The can stays at this position for 30 min, whereupon it is transferred back to the small drum by reversal of the main flow.

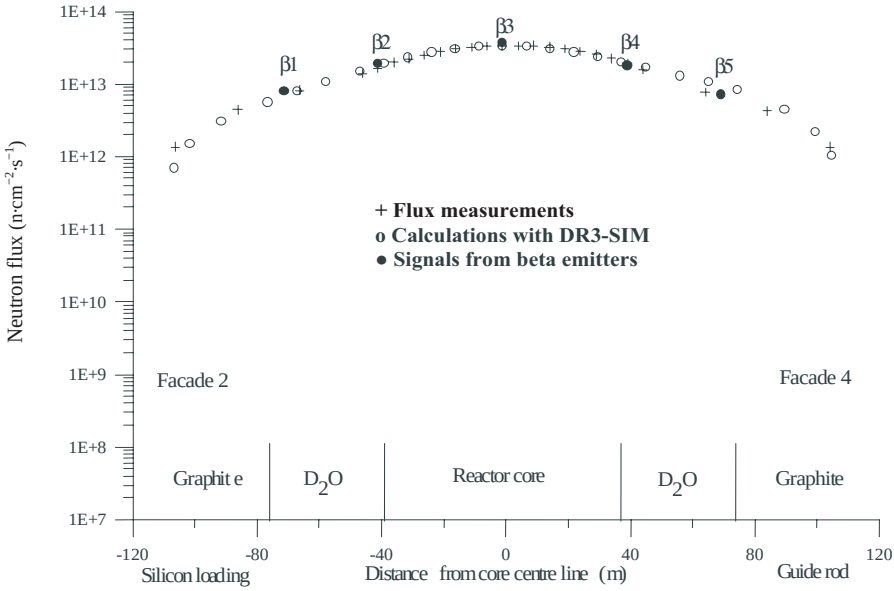

FIG. 6. Unperturbed thermal flux inside 7T2.

4.4. Unloading

When the can with its crystals inside has decayed while progressing through a storage sequence in both the small and the large drum for a total of six days, and the activity level is reasonably low, the operator is allowed to unload that can by selecting the unloading procedure of the computer program indicating the number of that can. Before the selected can is pushed into the lift, a gamma monitor between the large drum and the lift measures the residual radiation level. A health assistant checks the measured radiation level before allowing the can to be lowered. Finally, the operator removes the can.

5. FLUX CALCULATIONS

5.1. Flux scan

Before the NTD facility was fully installed, some flux measurements were carried out. Therefore, a thin tube (6 mm diameter) was placed inside the irradiation tube. A polythene tube with short pieces of embedded cobalt wire was inserted into that special flux scanning tube. During a flux scan, it is not

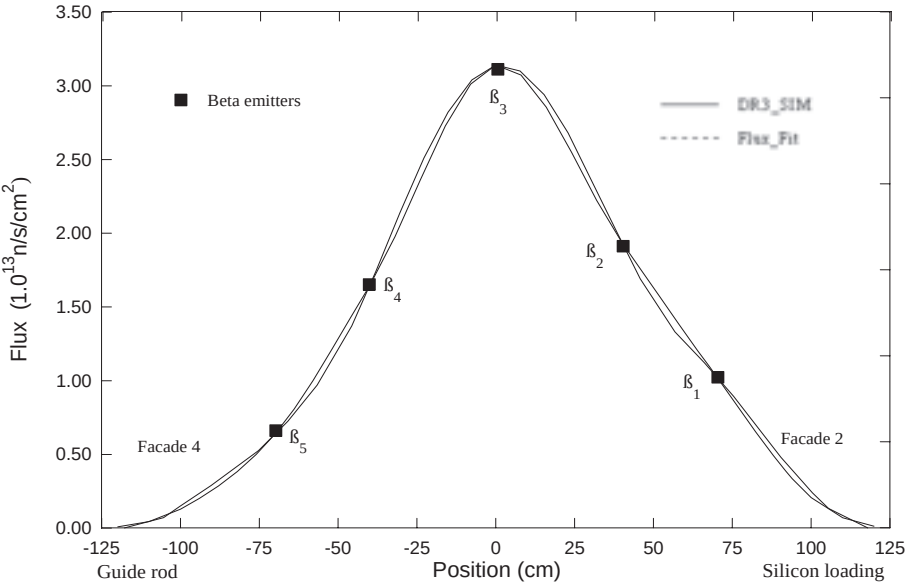

FIG. 7. The unperturbated thermal neutron flux at the centre line of the irradiation tube; the tube is filled with H_2O.

possible to perform an irradiation. Flux values corresponding to the signals from the beta emitters (see Section 2.1.) were read. With the operational data of the reactor valid at the time of the flux scan, a run of the computer program DR3 SIM (a program especially developed for the DR 3 research reactor) was performed.

The results of the flux scan, the readings of signals of the beta emitters and the calculations with the computer program are shown in Fig. 6.

5.2. Calculations prior to irradiation

Calculations of the unperturbed neutron flux in the irradiation tube filled with H_2O form a basis for performing the irradiation process [2]. The computer program DR3 SIM calculates the flux profile across the tube. Thereafter, the results from DR3 SIM are used in another computer code based on diffusion theory, to perform more detailed calculations.

The values obtained from the five beta emitters (see Fig. 7) are used as input to a spline interpolation for the actual flux, which is integrated to gain the average flux. These calculations are implemented in the computer code and determine the speed of the guide rod through the irradiation tube to achieve the specified irradiation dose.

5.3. Calculations during irradiation

During irradiation, signals from the beta emitters are used to adjust the speed in case of variations of the neutron flux.

As the can moves through the irradiation tube (see Figs 5 and 7), it will distort the signal from the beta emitters due to the absorption of neutrons. To obtain a usable signal, it is thus necessary to get the signals from those beta emitters which are located away from the actual position of the can, i.e. the signals from β1 and β2 when the can is moving from the 'start of irradiation' position to the centre of the active core. Those signals are used for the calculations. During the second part of its path, i.e. when the can is moving from the centre of the active core to the 'end of irradiation' position, the signal from β4 and β5 are used.

6. INTERACTIONS

6.1. Interaction with the vertical silicon irradiation facility 7V3

Calculations with the computer program DR3 SIM have not shown any significant influence from the changes of the state of the vertical silicon facility 7V3. Readings of signals from the beta emitters when a silicon crystal is lowered into or pulled out of the 7V3 rig have not shown any influence either.

6.2. Interaction with the fine control rod

The vertical position of the fine control rod (FCR) influences the flux profile. Figure 8 shows the variations in the flux profile inside 7T2 as a consequence of that influence at three vertical positions of the FCR. To minimize this variation, a limited interval (25–35 cm) for the FCR position was chosen. The controlled variation of the speed of the can during irradiation based on readings from the beta emitters, as explained in Section 4.3 and shown in Fig. 7, provides some compensation for the variations originating from the positioning of the FCR. If, for any reason, the FCR position causes problems, the irradiation sequence is stopped until the preferred interval for the FCR position is reached.

6.3. Absorber rigs

To maintain an almost constant angle of the coarse control arms (CCAs) of the reactor control, a movable absorber rig is placed inside every fuel

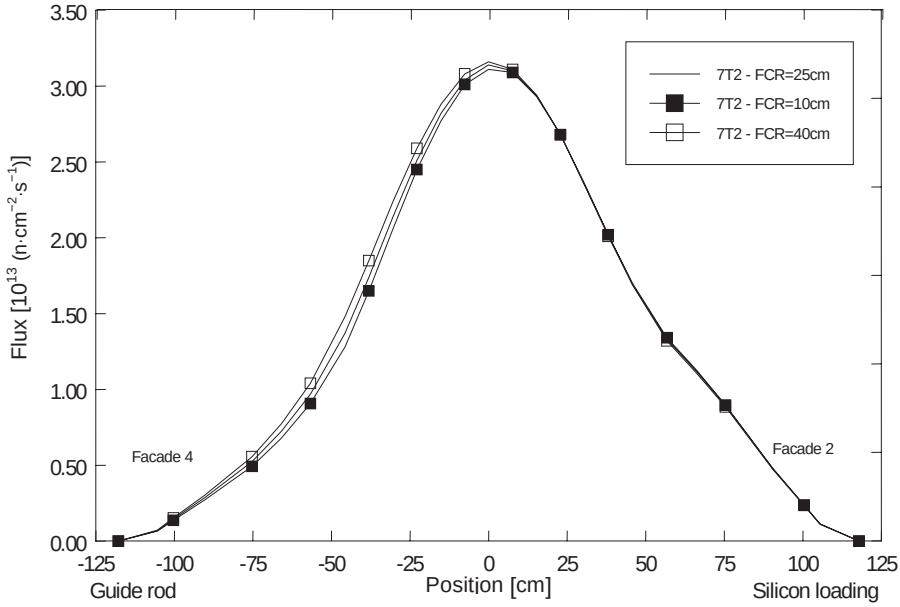

FIG. 8. The influence of the FCR on the neutron flux inside 7T2.

assembly. During the reactor cycle, those rigs are drawn out of the core zone one by one. Readings of the signals from the beta emitters when those absorber rigs are inserted into the A row of the core (see Fig. 1), i.e. nearest to the 7T2 tube, have shown no detectable change of the flux distribution inside that irradiation tube.

6.4. Reactor trip

In the case of a reactor trip during irradiation, the NTD system is able to compensate for the lack of neutron flux. The speed of the guide rod is lowered according to the flux level.

7. SAFETY ISSUES

7.1. Reactivity changes

An aluminium can filled with silicon and graphite placed right in front of the core results in a reactivity increase of 0.053% dk/k. If a can is inserted from the small drum into the irradiation tube due to a forward main flow while the

guide rod is in its 'start of irradiation' position (the main flow should be reversed in this situation), the can will be transported into the irradiation area with the speed of the water flow, which is 108 mm/s. That movement of the can results in a maximum reactivity increase of 0.0094% dk/k/s. Due to a certain slip between the water flow and the can during that transport, the actual speed of the can will be lower than the velocity of the water. Thus, these calculations are conservative. The limiting operating conditions for the DR 3 permit a maximum increase in reactivity of 0.5% dk/k/s for experiments with adjustable reactivity value.

7.2. Loss of water

A leak in the 7T2 system will drain the irradiation tube and there will be no cooling of the can or of the tube itself. The gamma heat will accumulate in the aluminium tube and cause a temperature increase of 0.3ºC/s, if the reactor is not shut down. Three level indicators placed in the level tank (see Fig. 2) will cause a reactor trip in a 'two out of three' logic, if the water level falls below its lower limit. There will be an alarm prior to the trip.

7.3. Loss of main flow

In the case of low flow or no flow at all, the cooling of the can and of the irradiation tube is reduced. Three pressure transducers in a 'two out of three' logic cause a reactor trip in the case of low flow. An alarm generates a preliminary warning prior to the trip. To be able to reverse the flow manually, a delay of 3 min between the alarm and the trip has been installed. As a typical irradiation time is around 5 h, a delay of 3 min will influence the result in the case of an extremely low dose only.

7.4. Health physics

The computer program does not allow the operators to remove an irradiated can unless it is decayed by storage over 6 d. The radiation level of the can is monitored automatically by a gamma monitor prior to any transfer from the large drum to the lift. In the case of a radiation level higher than acceptable, the removal sequence in the program cannot be activated and the can remains inside the big drum. Additionally, a health assistant checks the radiation level when the can arrives at the upper position of the lift.

8. IMPROVEMENTS OF 7T2 RELATIVE TO 7T4

From the experience gained from the first horizontal facility (7T4), some improvements had been made to the design of 7T2 prior to operation. The five most significant improvements were:

- Five beta emitters placed in five different positions have resulted in a better determination of the flux distribution;
- The software has been upgraded to a WINDOWS based program, which is much easier to use than the code applied at 7T4. Furthermore, an enhanced state of information regarding the irradiation is available on-screen, as well as indications of all the activities and the alarms;
- The speed of the loading and unloading procedures has been increased considerably;
- Improved flux calculations have made it possible to dope the silicon crystals more accurately. The standard deviation in resistivity is 1–2% only;
- A parameter of asymmetry to compensate for any imbalance of the core resulting from its fuelling has been implemented in the irradiation code. It ensures more homogeneous doping along the crystals by controlling the speed with respect to that said imbalance, as the two halves of the flux fit of Fig. 7 can be corrected individually.

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Part 5

MEDICAL THERAPY

A Note on the Papers of Part 5

In the foreword of IAEA-TECDOC-1223, crucial issues were addressed which reflect the IAEA's harmonized policy in the field of boron neutron capture therapy (BNCT):

> "One of the factors which need[s] to be addressed is the timing of the further development of NCT facilities. It should be emphasized that all current work is still at the research stage. Many of those now involved believe that there is little need for many more research facilities until such time as the treatment shows more promising results. For this and other reasons discussed in the report, very serious consideration should be given by research reactor owners and operators before spending large sums of money converting their facilities for NCT."

The contributions that follow in Part 5 reflect the opinions of the authors and their institutions, but do not represent the policy of the IAEA. This policy is stressed by a recent review (2005) of the current clinical status of BNCT[*] consistent with the statements expressed above:

> "The BNCT is a promising and possibly curative method of treating Glioblastoma multiforme, but at present this procedure is far from perfect. Because of the lack of selectivity of the boron carriers, it appears so far that radiation toxicity limits the radiation dose, so that tumour damage is modest. Current investigations and developments are aimed at targeting the boron carriers to the tumour, in order to limit the damage to the healthy, surrounding tissue."

[*] VAN RIJ, C.M., WILHELM, A.J., SAUERWEIN, W.A., VAN LOENEN, A.C., Boron neutron capture therapy for glioblastoma multiforme, Pharm. World Sci. **27** (2005) 92–95.

5.1. NEUTRON CAPTURE THERAPY: AN OVERVIEW

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1. BACKGROUND

Neutron capture therapy (NCT) is a technique that was designed to selectively target radiation which has a high linear energy transfer (LET) to cancerous tumours at the cellular level. It differs from conventional radiation therapy, one which involves the use of high energy X rays and electron beams which have a low LET, and where the energy deposited in the tumour as ionizations is spatially sparse and, therefore, the corresponding dose absorbed is generally lower. While standard radiotherapy treatments, chemotherapy and surgery have successfully cured many kinds of tumours, the potential efficacy of NCT for selectively targeting and destroying cancerous cells with little or no damage to normal tissue has attracted the attention of many scientists and medical professionals around the world.

NCT is a radiation therapy modality that brings together two components of a treatment that, taken separately, have minimal effect on cancerous cells. As such, the physical principle behind NCT as a two component or binary system is simple and elegant. It is based on the nuclear reaction that occurs when a naturally occurring or stable isotope, which has been concentrated into the tumour cells by attaching it to tumour seeking compounds, is irradiated with low energy or thermal neutrons. This isotope must have an unusually high affinity for absorbing thermal neutrons with a corresponding release of energetic particles which deposit their energy only in cells in close proximity to it, primarily the cancerous cells, leaving adjacent non-cancerous cells largely unaffected.

The concept of boron neutron capture therapy (BNCT)[1] using the naturally occurring isotope ^{10}B was proposed shortly after the discovery of the neutron by Chadwick at the University of Cambridge in 1932, and the discovery that the naturally occurring isotope ^{10}B has an unusually large

[1] In this overview, the terms NCT and BNCT are used interchangeably due to the fact that the vast majority of the work involving therapeutic treatment of cancers with this technique involves the use of boronated compounds. Tumour treatment without using boron compounds, which is not discussed in the overview, is discussed in paper 5.3.

thermal neutron capture cross-section. While there are other nuclides that have a high propensity for absorbing low energy neutrons, the properties of ^{10}B are attractive for a variety of reasons, including its well understood chemistry allowing it to be incorporated into an appropriate carrier compound. When ^{10}B is irradiated with low energy or thermal neutrons, it yields highly energetic helium-4 (He) nuclei, or alpha particles, after briefly becoming ^{11}B and recoiling ^{7}Li ions:

$$^{10}B(n,\alpha)^{7}Li$$

The alpha particles and lithium ions from this reaction are closely spaced ionizing events which have a combined range in tissue comparable to that of cellular dimensions (12–13 μm) and have a high LET. Hence, they deposit most of their energy within the cancerous cells where the concentration of ^{10}B is higher due to targeting by an appropriate chemical compound that has been 'tagged' with ^{10}B. Thus, a higher dose is delivered in the tumour cells relative to normal tissue during irradiation with neutrons.

In the last decade, there has been renewed interest in a number of countries aimed at further development of NCT, both in the development of better neutron beams and in the development of better compounds as boron carriers to cancerous tissues [1]. To that end, it was decided to include NCT as a topic for publication in this Compendium, to allow the assimilation of available information about the operation and utilization of research reactor facilities for this purpose.

Since this is a binary treatment, success of the technique is dependent not only on the availability of irradiation facilities such as those at research reactors, but on the development of improved compounds. The papers in this part do not attempt to address the development of improved compounds as carriers for boron nor the clinical considerations of NCT. The focus of Part 5 is on the design and utilization of research reactor facilities for NCT, both at existing research reactors being modified and utilized for NCT, and the design of new reactors which include dedicated NCT beams as part of the original design.

2. BRIEF HISTORICAL PERSPECTIVE

BNCT was first proposed as early as 1936 by Locher [2], who theorized that targeting the tumour with a higher radiation dose is best accomplished by selectively concentrating the boron in the tumour and then exposing it to thermal neutrons. This is the rationale for the clinical implementation of

neutron capture therapy. Following this and other initial suggestions that BNCT might be useful for the treatment of cancers, interest developed in its application for high grade, inoperable brain tumours (glioblastoma multiforme or GBM as well as melanoma), an idea proposed by Sweet in 1951 [3, 4]. The idea was to exploit the reduction in the blood–brain barrier in the vicinity of the tumour to concentrate the boron in the tumour relative to the normal brain tissue. The compound used initially as the boron carrier was sodium tetra borate (borax).

The initial clinical trials for BNCT were conducted at the Massachusetts General Hospital with irradiations being performed at the Brookhaven National Laboratory graphite reactor using thermal neutrons. These early trials by Sweet et al. were not successful, as they led to clinical problems, such as unacceptable scalp reactions and other diseases due primarily to insufficient beam penetration resulting in damage to healthy brain tissue. In an effort to reduce these problems, trials were initiated at the Massachusetts Institute of Technology Research Reactor in the early 1960s, but with maximum tumour debulking and an open cranium. While there was evidence of temporary benefit to the tumour itself in a few patients, all patients subsequently died after treatment. Therefore, these early trials failed to show evidence of therapeutic efficacy.

NCT became dormant at this time, except for the work of Hatanaka [5] and Mishima [6] in Japan. New boron compounds were developed and used in the Japanese trials on human patients in the 1960s and early 1970s. The two new compounds used were sulphydryl containing polyhedral borane (BSH) and borated phenylalanine (BPA), which showed significantly more favourable tumour to brain concentrations than the earlier borax compounds. Hatanaka's work with GBM patients and Mishima's work with melanoma patients have led to renewed and continuing international interest in NCT, including the development of more suitable neutron beams and improved boron carrier drugs.

The recent attention to NCT has focused on the use of more penetrating epithermal neutron beams with significantly reduced fast neutron and photon components, in an effort to reduce scalp reaction without the complications of using open craniums, as was done by Hatanaka. Such an epithermal neutron beam was first constructed at the Massachusetts Institute of Technology Research Reactor in 1994, with the trials limited to melanoma patients. At about the same time, the first BNCT trial for brain tumours was initiated using an epithermal neutron beam at the Brookhaven National Laboratory's Medical Research Reactor. More recently, a new type of epithermal neutron beam has been put into operation at the Massachusetts Institute of Technology Research Reactor. This beam, known as the fission converter beam (FCB), uses a fission

converter (fuel driven by the reactor core) as the source of neutrons for the therapeutic beam [7]. This is currently the highest intensity operating epithermal neutron beam with a higher beam quality (negligible beam contamination from fast neutron and gamma components) which allows increased tumour to tissue doses. The facility, which also has improved patient comfort features during BNCT irradiations, has been licensed to operate by the US Nuclear Regulatory Commission.

3. NCT NEUTRON BEAM CONSIDERATIONS

To perform BNCT, the basic beam requirement is that an adequate thermal neutron field must be created inside the boron loaded tumour cells, and within the target volume in the area affected. For deepseated tumours well below the surface, epithermal beams will generally be favoured, while for tumours near the surface, thermal beams will suffice. In both cases, beam quality, in part, will determine the success of the treatment. Beam quality relates to the types, energies and relative intensities of all the radiation components present.

Studies on phantoms show that an epithermal beam entering tissue creates a maximum thermal neutron flux field at a depth of 2–3 cm, and drops off exponentially thereafter. The penetration can be increased by increasing the energy and directionality of the epithermal beam. In contrast, a thermal beam impinging on the tissue surface falls off exponentially from the surface. As a result, current trends for GBM patients are to use epithermal beams, whereas thermal neutron beams are used for melanoma treatments or open cranium GBM treatments, as was practised during earlier work by Hatanaka.

Epithermal beams (0.5 eV–10 keV) are considered state of the art for NCT work. Current thinking is that an epithermal beam intensity of $\sim 10^9$ n·cm^{-2}·s^{-1} is desirable to achieve reasonable irradiation times. Beams less intense are usable, but will result in longer irradiation times. Beam intensities will also need to be balanced against the requirements of improved beam quality in such a way that the time the patient is in the beam is minimized while achieving the desired result (maximum tumour dose and minimal normal tissue damage). Eventually, this will have an impact on the clinical acceptability of NCT. Tumour boron concentrations will also affect patient irradiation times; the better the drug for concentrating the boron in the tumour, the shorter the irradiation time. To avoid unduly long irradiation times, fractionation is also a consideration.

Beam quality parameters for BNCT are as follows, with the aim of maximizing the epithermal flux component of the beam:

— For the fast neutron component, the target number is typically taken to be 2×10^{-13} Gy·cm^2 per epithermal neutron;
— For the gamma component, the target number is typically taken to be 2×10^{-13} Gy·cm^2 per epithermal neutron;
— The ratio of thermal to epithermal flux should be at least 0.05;
— For beam directionality, a target value for the ratio between total neutron current and flux is 0.7.

Finally, if a facility is also to be used for radiobiology research for NCT, in addition to therapeutic use for tumours, a beam should be designed which provides access to both thermal and epithermal beams with the required beam parameters. Research facilities which study, for example, the effect of boron carrier compounds using cell cultures or small animals, require a pure thermal neutron field at the irradiation point.

4. FACILITIES FOR PERFORMING NCT

In general, it can be argued that multipurpose research reactors may not be the most practical source of neutron beams for NCT. In most cases, multiple uses of a research reactor can put limitations on and result in conflicts for their optimum utilization for therapeutic applications. Additionally, most research reactors are separated from hospitals, thus their use for NCT can present practical difficulties. Practically speaking, however, the only facilities currently available for NCT clinical trials are research reactors. Therefore, an emphasis is being seen on modifying existing research reactors for NCT or, when designing new reactors, including dedicated NCT facilities in the multipurpose design. Indeed, new research reactor designs have been proposed as part of this renewed interest in NCT, which would be dedicated therapeutic NCT machines, and which would be located in, or in the vicinity of, hospital facilities.

NCT facility designs in lieu of the use of multipurpose research reactors that have been proposed are reactors specifically designed and dedicated for NCT, accelerator based neutron sources, a ^{252}Cf neutron source, as well as a combination of the above (e.g. combining a subcritical assembly with an accelerator or ^{252}Cf source). Further discussion on alternate sources can be found in Ref. [1].

Multipurpose research reactor facilities currently being utilized, or planned, for NCT need to use techniques, such as spectrum filtering and fission converters or a combination, to achieve the desired beam intensity and purity at the patient irradiation location. The availability of new filter materials has allowed research reactors to achieve the desired beam intensity and purity.

At multipurpose research reactors, there are some basic facility requirements that need to be satisfied in order to successfully carry out clinical NCT and deliver therapeutic doses to the cancer patient. These are:

— A large diameter beam port, thermal column or other void space with a direct view of the core;
— Fixtures and space to install the fission converter assembly, filter assembly and massive beam shutters;
— Space for a shielded treatment room with instrumentation to ensure patient safety (e.g. with reactor scram capability);
— Space for medical facilities, e.g. for patient preparation and laboratories;
— Capabilities for, or access to, sophisticated reactor analyses, as well treatment modality modelling;
— Access to hospital and medical staff.

5. SUMMARY

Included in Part 5 are five contributions from research reactors or reactor designers currently involved with NCT facility design, operation and use. While these papers provide a sample of ongoing work, since the resurgence of NCT about a decade earlier, there is a vast amount of information that is available on both aspects of this binary radiation therapy modality for the treatment of cancers, i.e. the design of appropriate neutron beams, and the continuing development of carrier drugs. To that end, the reader is also referred to the references cited in this overview for additional information on NCT history, advances and current status:

— Paper 5.2 (Whittemore, Razvi and Ellis) discusses the design of an NCT facility as part of a new multipurpose, 10 MW research reactor. The paper also includes a design and performance comparison of such a new facility with existing small and medium power research reactor facilities that have been modified and retrofitted to perform NCT;
— Paper 5.3 (Waschkowski, Lange and Böning) deviates from a discussion of a binary radiation therapy modality, and discusses the direct irradiation of near surface cancers with fast reactor neutrons;
— Paper 5.4 (Harling, Binns and Riley) discusses the current NCT programme at the Massachusetts Institute of Technology Research Reactor, which was specifically designed from the outset for BNCT research and treatment;

— Paper 5.5 (Auterinen) describes the conversion of a small research reactor to a facility primarily used for NCT clinical trials;
— Paper 5.6 (Blaumann, Calzetta Larrieu and Longhino) reports on the characteristics and current status of a dual energy beam design planned for installation at a low power research reactor.

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5.2. DESIGN OF A MEDICAL THERAPY FACILITY IN A MULTIPURPOSE RESEARCH REACTOR

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1. INTRODUCTION

Multipurpose research reactors typically operate at higher (= 5 MW) power levels to provide the higher neutron fluxes to meet the various needs, purposes and end-user requirements for which the reactor was planned. To incorporate medical therapy facilities in an existing, or planned, multipurpose reactor, careful compromise among facilities is usually necessary to ensure not only state of the art purity of the therapeutic neutron beam, but also adequately short irradiation times. Over the years, the techniques to perform therapeutic treatment of cancer patients using neutron beams from nuclear research reactors and neutron capture therapy (NCT) treatment drugs, have steadily evolved from thermal (= 0.5 eV) neutron beams to epithermal (0.5–10 000 eV) neutron beams.[1] This paper deals mainly with the epithermal neutron technique for NCT, and focuses on the details of the reactor structure and the biological shield. It does not consider further details of the numerous additional design requirements of the patient treatment room required to safely treat the patient in a research reactor facility.

A number of studies have demonstrated that a small (<500 kW), dedicated nuclear reactor can be an ideal neutron source for NCT with added benefits, such as lower overall costs and reduced design complexity, which in turn can contribute to wider use of the technique. Where that is not feasible, it would be desirable to incorporate NCT treatment facilities into the design of a multipurpose research reactor. To that end, the purpose of this paper is to discuss the compromises required to include an efficient NCT facility in a higher power multipurpose reactor. Inclusion in a multipurpose reactor of a fission converter assembly for NCT retains, in some respects, the advantages of a small (= 100 kW) and dedicated neutron source.

[1] It should be noted that the therapeutic treatment of certain tumours is also practised at some facilities using fast neutron beams.

TABLE 1. TYPICAL FACILITY REQUIREMENTS FOR MULTIPURPOSE REACTORS

Feature	Typical number	Possible location	Typical mandated performance
Radioisotope production	10–30	Reflector	10^{13}–10^{14} n·cm^{-2}·s^{-1} (thermal) (at least one fast neutron location)
NTD	2–4	Reflector with low fast flux	~20 t/a
Beam experiments:			
Tangential port	3–6	Reflector, low fast flux	10^{13}–10^{14} n·cm^{-2}·s^{-1} (thermal)
Radial port	1–2	Reflector, max. fast flux	10^{13}–10^{14} n·cm^{-2}·s^{-1} (fast)
Cold n-source	1–2	Reflector, max. thermal flux	10^{13}–10^{14} n·cm^{-2}·s^{-1} (thermal source)
Hot n-source	1	Reflector, max. thermal flux	10^{13}–10^{14} n·cm^{-2}·s^{-1} (thermal source)
Neutron activation analysis	2–3	Reflector	10^{11}–10^{13} n·cm^{-2}·s^{-1} (thermal, epithermal)
Sample irradiation sites	2–4	In-core, edge of core	10^{13}–10^{14} n·cm^{-2}·s^{-1} (thermal, fast)
Medical therapy (NCT) Large port, 1 m^2	1	View of large portion of fuelled section (fission converter or core)	5×10^9 n·cm^{-2}·s^{-1} (0.5–10 000 eV) (at patient location)

2. DESIRABLE FEATURES IN A MULTIPURPOSE RESEARCH REACTOR

The uses of a multipurpose reactor include a broad range of applications (Table 1). The problem encountered when including many of these applications in a single reactor is that more than one will necessarily vie for the same optimum location. An example would be the desire to position certain beam ports for the peak available thermal neutron fluxes, while simultaneously providing for several high flux thermal neutron irradiation sites, such as those required for neutron transmutation doping (NTD). To accommodate such a combination, it is necessary to provide a reactor design that spreads the peak thermal spectrum over a relatively broad expanse. Another compromise involves the requirements for extended higher power operation, on the one hand, while also providing for necessary target manoeuvres and short NCT irradiations. As examples, many irradiated samples, especially activation analysis samples, need to be removed before the end of long operation cycles;

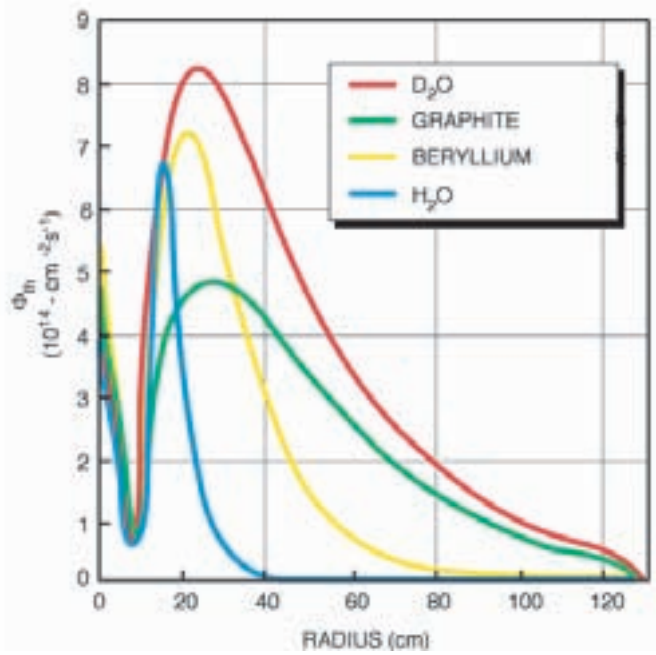

FIG. 1. Thermal flux characteristics of four different reflector materials, for the FRM-II research reactor configuration.

and the irradiation of NCT human patients must be completely terminated but usually without interfering with a full power long term operation.

In Fig. 1, the radial expanse of thermal flux in four typical reflectors is compared. D_2O provides the largest thermal peak flux that is spread over the broadest radial extent. Its performance in this regard is significantly better than

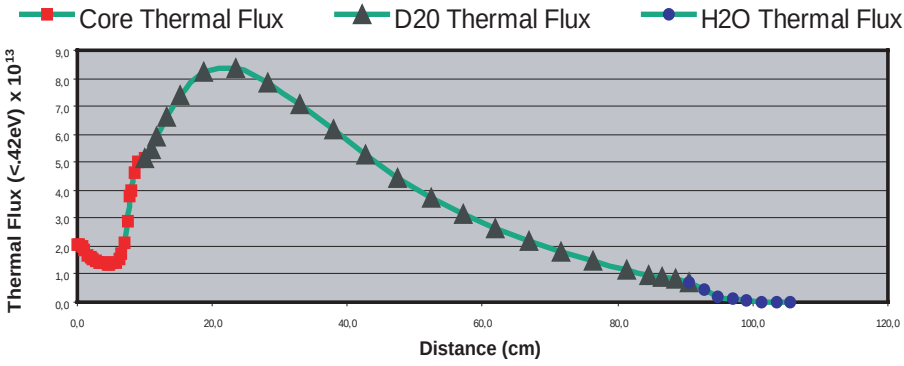

FIG. 2. Thermal neutron flux distribution in a D_2O reflector.

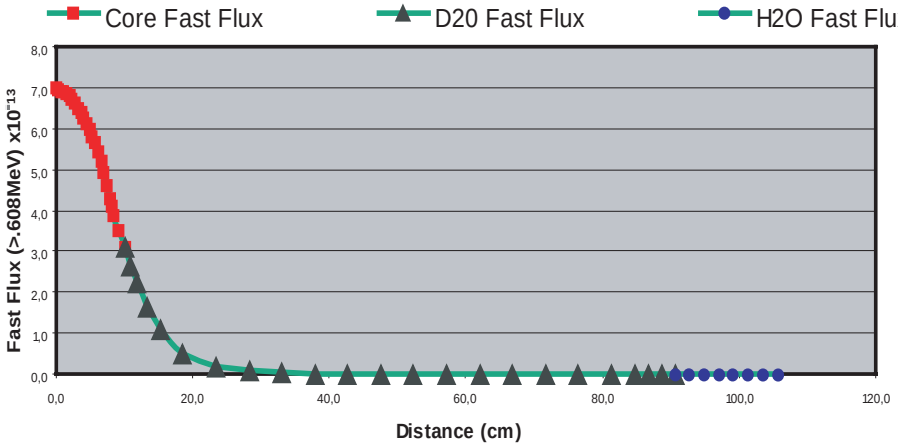

FIG. 3. Fast neutron flux distribution in a D_2O reflector.

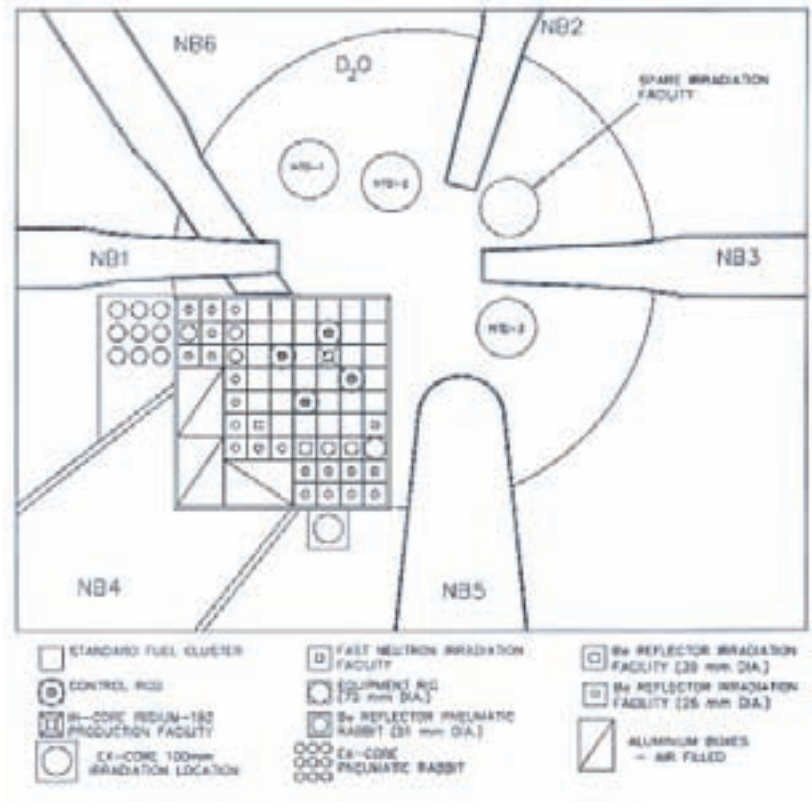

FIG. 4. Reactor core and reflector layout.

with Be. Graphite does provide a broader expanse at half-width but provides only about half of the neutron peak intensity. A light water reflector, while its peak intensity is high, has a radial expanse so small it cannot accommodate even a small beam port. In high performance multipurpose reactors, therefore, a reflector incorporating D_2O, Be or a combination of both is required. Figures 2 and 3 illustrate, for a D_2O reflected Triga research reactor, the broad extent of thermal flux and the near absence of fast neutrons in most of the reflector.

Figure 4 presents an example of a multipurpose Triga reactor design layout that incorporates an NCT facility together with several other applications. This design takes advantage of the special properties of both D_2O and Be as reflector materials. Except for a single row of Be blocks, the NCT facility views directly a portion of the core and also has provisions for use of a fission converter assembly, if desired.

3. DESIGN OF NEUTRON CAPTURE THERAPY FACILITIES

The object of the design discussed here is to make available to the end-user in a multipurpose research reactor, a near optimum NCT neutron beam and neutron filter, i.e. high intensity of epithermal (0.5–10 000 eV) neutrons with a minimum of fast neutron and gamma ray background in the therapeutic neutron beam. It must be noted at the outset that studies have amply demonstrated that a small (= 500 kW) reactor can be an optimum neutron source [1]. Some of the advantages of a small reactor for NCT are shown in Table 2. In the design of an NCT facility as part of a multipurpose reactor, the demands for near optimum performance of a myriad of facilities, including NCT, must include attention to design features, such as the four discussed in the following subsections, which affect the overall design of the reactor.

3.1. Design features important in an NCT facility

3.1.1. Biological shield thickness

The total thickness of the existing biological shield in the region of the NCT facility needs to be reduced in the interest of bringing the patient closer to the neutron source, while preserving general radiation safety. Optimum performance of NCT with a multipurpose research reactor requires considerations of means to reduce the overall distance of the patient from the effective neutron source. One method (Fig. 5) to achieve this result is to replace a portion of the concrete biological shield with high density concrete or embedded metal, such as iron plates that have a much higher density. An iron

TABLE 2. BENEFITS OF A LOW POWER REACTOR DEDICATED FOR NCT

(1)	Patient can be safely located closer to the neutron source.
(2)	Therapeutic dose rates that can be achieved are comparable to those in high power (multipurpose) reactors.
(3)	Reduced shielding size and associated costs.
(4)	Direct view of core is readily available.
(5)	Neutron beam is easy to shut off, thus reducing or eliminating the requirements for a massive shutter.
(6)	Lower costs when compared with a higher power multipurpose reactor, e.g. for operating staff and fuel cycle requirements.
(7)	Balance of plant and other auxiliary components are simplified or even eliminated compared with those for larger (5–10 MW) multipurpose reactors.

thickness of just under 0.5 m can reduce the overall biological shield thickness by about 0.8 m.

3.1.2. Fission converter assembly

A fission converter assembly outside the active region of the core can be installed to serve as an efficient NCT neutron source. Experience has shown

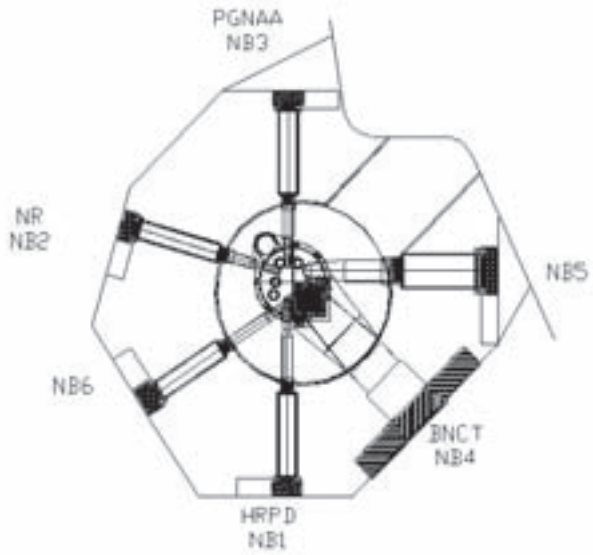

FIG. 5. NCT biological shield layout with a 0.48 m iron shield component.

that optimum NCT performance is provided by a fission converter operating at 90–100 kW [2, 3]. In a multipurpose research reactor, this method can be used to approach the small dedicated reactor performance, thus effectively bringing the patient closer to the effective neutron source [2, 3].

For the Triga type multipurpose research reactor described, safety considerations of the facility from the use of a fission converter assembly are no different from those for the fuel in the reactor core. The reason is that the fission converter fuel is located in the same region and cooled by the same coolant flow as the reactor core. Therefore, for the design described here, there are no additional or unique facility safety concerns that need to be considered.

3.1.3. Safety of the NCT patient

While this paper deals only with personnel safety issues related to the multipurpose research reactor itself, the design of the beam line delivering the patient dose must take into account background dose and dose to healthy tissue. To that end, the optimum, normal treatment protocol will carefully control the unwanted dose to healthy tissue surrounding the tumour and to all other regions of the patient's body. In addition, abnormal or accident conditions must be controlled so as to ensure that the NCT patient is not otherwise endangered by reactor operation. Patient safety is further discussed in Section 3.4.

3.1.4. Use of components

Careful attention must be given in the selection of NCT beam specific components to avoid those that activate heavily and with a long half-life, and that will complicate the later decommissioning of the facility. An example would be the use of lead as an albedo reflector of fast neutrons in place of nickel.

3.2. Examples of recent NCT facilities retrofitted in existing reactors

A variety of approaches have been employed recently to provide near optimum NCT performance. As illustrated, part of the improved NCT performance is due to the use of fluorine in the filter to produce the desired epithermal therapeutic neutron current while producing a minimum of fast neutron and gamma ray background in the beam. A major improvement in the design ensures a relatively small distance between the patient and the effective neutron source. One system (see Ref. [1]) arranges to have the fluorine containing filter with a view directly over a small (= 250 kW) research reactor

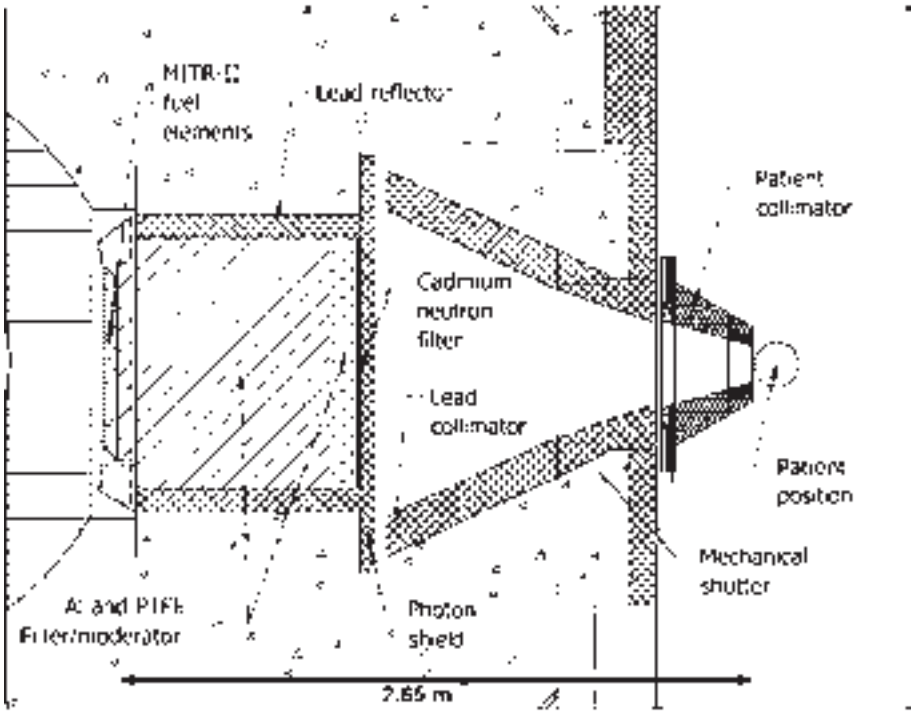

FIG. 6. Plan view of the Massachusetts Institute of Technology (MIT) fission converter based neutron beam for NCT.

core. Two multipurpose research reactor systems (see Refs [2, 3]) with power levels of 5 MW (Fig. 6) and 2 MW (Fig. 7), respectively, use different fission converter systems, each with a small distance (~2 m) between the low power effective neutron source (= 100 kW) and the patient.

3.3. NCT facility in a new multipurpose research reactor

A 10 MW multipurpose Triga reactor has been designed recently, including provisions for a high performance epithermal NCT beam. The core and reflectors (Fig. 4) are arranged for optimum performance of the several required utilization facilities. The portion of the reflector accommodating several high thermal flux facilities (beam ports, NTD) is D_2O. The portion of reflector which will support a variety of up to 20 isotope irradiation facilities is composed of individual blocks of Be, each with a central axial irradiation site. While the peak thermal flux level is slightly less than with a D_2O reflector, these irradiation sites will accommodate a variety of sample diameters that can

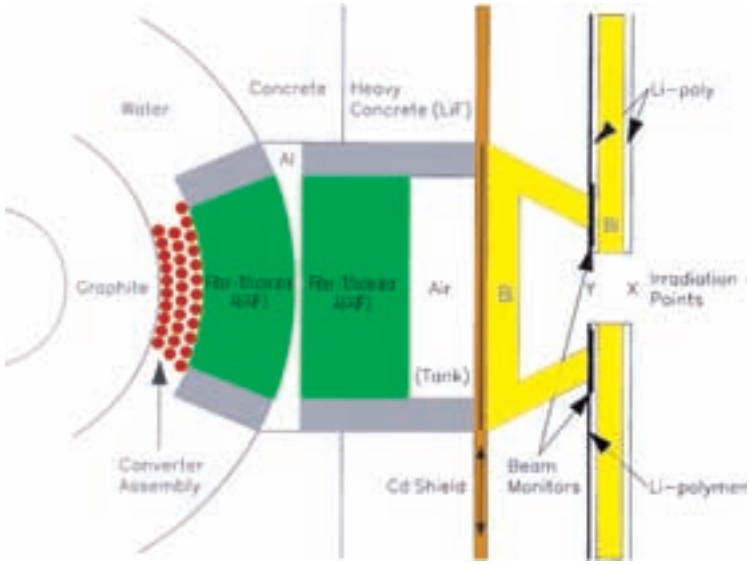

FIG. 7. Plan view of the University of California Davis neutron beam for NCT.

FIG. 8. Proposed NCT beam line and filter for a multipurpose reactor.

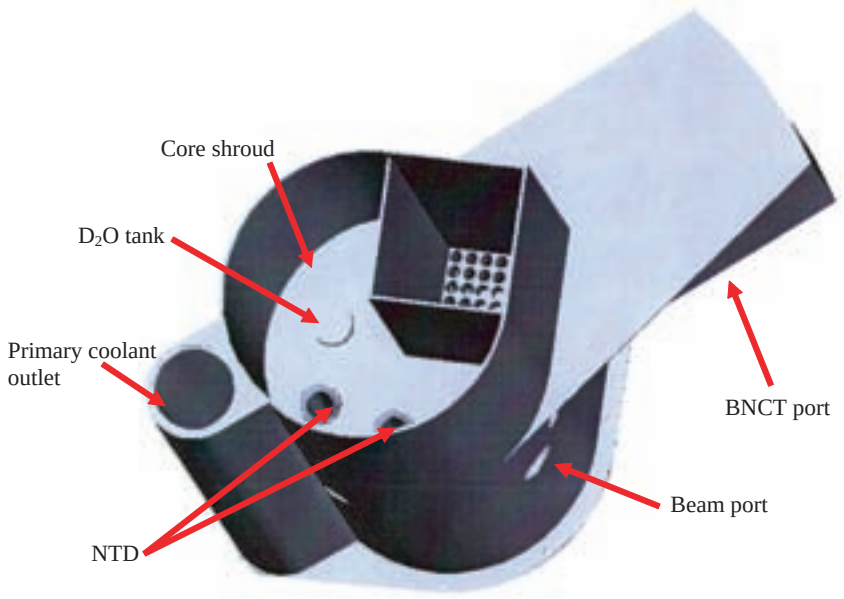

FIG. 9. Proposed core structure for a multipurpose research reactor.

be altered or relocated during the reactor lifetime as the user desires. Considerable flexibility is thus designed into this portion of the reflector.

The NCT port (NB4) has a direct view (Figs 4, 8) over a significant portion of the fuelled core perimeter. The single row of Be blocks in this area has relatively little effect on the fast neutron current for the NCT port; but this single row of Be blocks is extremely important in maintaining a reasonably symmetrical power distribution in the core. Core, reflector regions and NCT port are designed as a single, aluminium support structure (Fig. 9) mounted inside a stainless steel pool liner. Figure 10 gives a cutaway view of the biological shield structure showing the NCT beam port.

The NCT port inside the stainless steel pool liner (Fig. 8) is designed to be variously air filled or water filled. When water filled, this region (a) prevents significant activation of the affected region of the steel liner tank wall when no NCT operation takes place; and (b) acts as a massive shutter when needed for NCT operation.

3.4. Performance of a near optimum NCT beam filter

The designs of the new NCT beam line components described here are essentially the same as those reported for recently retrofitted facilities (see Refs [1, 3]). Two of these facilities have been constructed and detailed

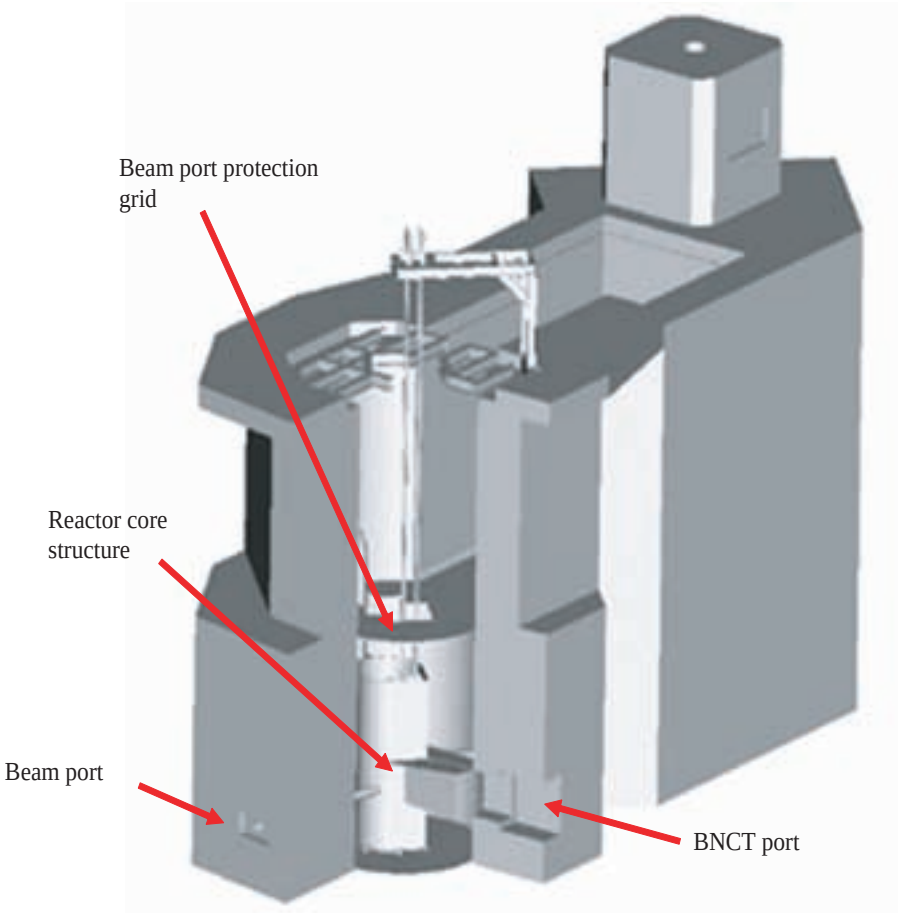

FIG. 10. Cutaway view of a biological shield showing reactor structure and NCT beam path.

performance measurements made, confirming their respective MCNP calculations for NCT. These NCT facilities include NCT filters containing fluorine and aluminium, and all have been designed with reduced distance between the patient and the effective NCT neutron source.

The facility design for epithermal NCT shown in Fig. 8 is based on filter material containing AlF_3 (Fluental) available from Finland (see paper 5.5). The section of the NCT port in the reactor's water filled tank is shown air filled, as would be the case during NCT operation. Air filled aluminium cans complete the neutron path to the row of Be reflector blocks at the core edge. The air in the NCT port outside the reactor shroud can be replaced with water as a shutter to control NCT patient exposure. When not operated for NCT, it would

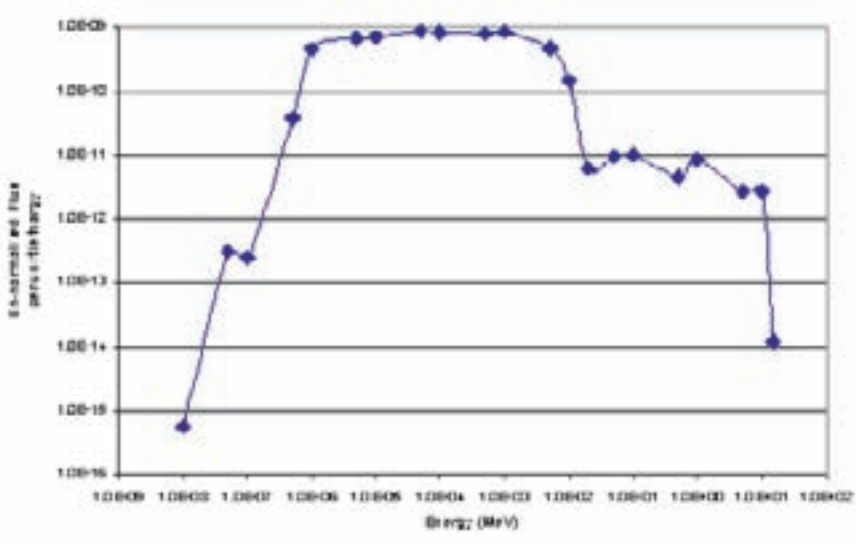

FIG. 11. NCT energy spectrum from a filtered beam.

normally be filled with water. As a neutron shutter, the water in this port can be moved in or out in about 30 s or less to control essentially all radiation to the NCT patient. The NCT therapeutic dose to a tumour is also controlled by a boron or lithium fast acting shutter.

In the region of the biological shield near the NCT port, the rather large distance between the NCT patient and the effective neutron source has been reduced as noted above, through use of an iron component with 0.48 m thickness (Fig. 5).

For the NCT beam line components shown in Fig. 8, MCNP calculations have been performed to assess performance and compare with other facilities. The resulting energy spectrum for the therapeutic epithermal neutron flux (0.5–10 000 eV) is shown in Fig. 11. It is readily seen that the fast neutron flux above about 10 000 eV falls rapidly by a factor of 100 or more. This fact ensures very low values for the NCT background in therapeutics due to fast neutrons. Similarly, judicious use of bismuth controls the gamma ray background also to very low values. Harling et al. [4] have prepared a comparison of NCT performances versus total specific doses for a number of current NCT facilities (Fig. 12). Total specific dose is defined as the RBE Gy dose per epithermal neutron fluence.

The MCNP results for the design described in this section for a 10 MW multipurpose research reactor give therapeutic epithermal neutron fluxes, with

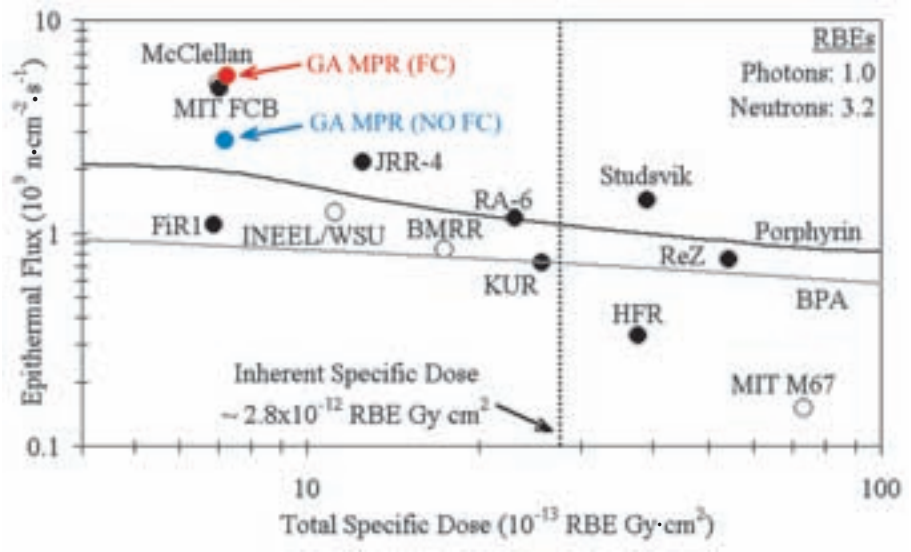

FIG. 12. Summary of NCT performance with various research reactors.

or without a fission converter assembly, of 5.7×10^9 n·cm^{-2}·s^{-1} and 2.7×10 n·cm^{-2}·s^{-1}, respectively, at a distance of 75 mm outside the patient collimator.

Table 3 summarizes the performance results, including background doses for this NCT facility compared with those for FiR 1 (the Finnish BNCT facility) and MIT (FCB) reactors. These values have been added to those originally presented in Fig. 12. The retrofitted facilities at the University of California Davis (UC Davis), MIT (FCB) and FiR 1, as well as the General Atomics (GA) facility, all have very small background doses (~7×10^{-13} RBE Gy·cm^2) compared with the inherent specific dose of 28×10^{-13} RBE Gy·cm^2 caused by the therapeutic epithermal neutrons in healthy tissue (essentially capture gamma rays in hydrogen and ^{14}N capture recoil protons).

If a fission converter assembly is included in the final design of the NCT facility described, it would consist of a row of standard UZrH fuel rod clusters placed immediately adjacent to the single row of Be reflector blocks.

3.4.1. Normal reactor operation

The optimum NCT filter assembly described ensures that the dose to the tumour is essentially therapeutic with a significantly reduced dose to healthy tissue from fast neutrons and photons in the beam, as demonstrated. The patient irradiation facility, not addressed in this paper, should provide space to orient the patient so the body is not in the direct therapeutical beam, i.e. most

TABLE 3. PERFORMANCE OF SELECTED NCT FACILITIES

Reactor	Location	Power (MW)	Epithermal flux ($n \cdot cm^{-2} \cdot s^{-1}$)	Neutron current/flux	Total specific dose fast neutron and photon dose/fluence (10^{13} RBE $Gy \cdot cm^2$)
FiR 1	Finland[a]	0.25	1.1×10^9	0.77	6.8
MIT (FCB)	USA[a]	5	5×10^9	≥0.84	7.0
GA MPR (FC)	Thailand[b]	10	5.7×10^9	0.6–0.8	7.0
GA MPR (No FC)	Thailand[b]	10	2.7×10^9	0.6–0.8	7.0

[a] Operating.
[b] Planned.

of the patient's body is usually oriented so as to remain out of the beam. This feature requires additional compromise for multipurpose research reactors because other user facilities are usually located nearby. Thus, an adequately large shielded NTC treatment room can encroach on space needed for other users.

The patient irradiation time is likely to be only a few minutes (5–10 min). The start and stop of the irradiation needs to be reasonably short, a small fraction of a minute at most. For a small NCT reactor (= 500 kW), it is usual to control the irradiation by shutting down the reactor without the use of a massive shutter [1]. For a much larger multipurpose research reactor that is intended for full power uninterrupted operation, the NCT patient exposure time must be controlled by other means, most often by a fast acting boron or lithium shutter (for the therapeutic beam) in addition to a massive shutter (for fast neutron and photon background). For those multipurpose research reactor facilities for NCT using a fission converter, the massive shutter can be replaced by a lighter shutter made from boron or cadmium to remove the thermal neutron source flux for the fission converter just as for the small (<500 kW) NCT reactor. The remaining fission product gamma rays in the converter are sufficiently attenuated by the NCT filter assembly so as not to be a radiological problem. Those multipurpose research reactors used for NCT without a fission converter must provide a massive shutter so that medical attendants can safely approach the NCT facility to bring and remove the patient without shutting down the reactor. This massive shutter can use water to fill a port (as shown above for the design discussed) or it can be a massive removable shutter usually fabricated from concrete.

3.4.2. Abnormal reactor operation

The bedridden NCT patient is particularly vulnerable to any of the abnormal, but rare, reactor operating events. The most serious consequences would be a reactivity excursion, either as a startup accident or insertion of unplanned reactivity while at power. For multipurpose research reactors operated at full power continuously during NCT operation, there is obviously less risk of a startup accident. The unplanned addition of reactivity during full power operation is exceedingly rare; nevertheless, additional vigilance to militate against such an accident is required during NCT operations. It may be noted that multipurpose research reactors having a strong prompt negative temperature coefficient of reactivity make it impossible to sustain accidental patient dosage or reactor damage from either a startup or reactivity insertion accident. Such a reactor contributes significantly to the overall safety of an NCT patient.

4. CONCLUSION

Inclusion of an NCT treatment facility in a higher power (e.g. 5–10 MW) multipurpose research reactor involves compromise among the several user facilities. It is demonstrated here that the resulting NCT facility can be included without serious effect on the performance of other facilities. Moreover, the resulting NCT facility can function at near optimum with low background (fast neutron and photon) and optimum therapeutic treatment with short (5–10 min) irradiation times, depending of course on the boron drug used.

For a multipurpose research reactor retrofitted for NCT, the optimum desired compromises are frequently hard to make without major reconstruction. In some instances, use of a fission converter as the NCT neutron source can reduce the conflict with other user facilities required for the research reactor.

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5.3. FAST NEUTRONS FOR TUMOUR TREATMENTS AND TECHNICAL APPLICATIONS AT THE FRM-II

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1. INTRODUCTION

In modern research reactors moderated with heavy water, the beam tubes are arranged tangentially to the core in order to obtain intense beams of thermal (or cold) neutrons with a low contamination of fast neutrons and gamma rays. However, at the FRM-II, a special facility was installed to produce additionally one beam of fast neutrons with a large area (300 × 200 mm) and a high intensity (1.5×10^9 n·cm^{-2}·s^{-1}). The neutron spectrum looks similar to a fission spectrum and is suitable as a standard spectrum. The facility can be used for the following applications and experiments.

1.1. Medical applications

— Therapy of near surface cancers with fast reactor neutrons to support other percutaneous therapy (no BNCT);
— Development of boron neutron capture therapy (BNCT);
— Research on biological dosimetry and on metabolic processes.

1.2. Technical applications

— Radiography and tomography on technical samples in non-destructive material testing;
— Irradiation of samples with fast reactor neutrons;
— Supply of a standard source of a quasi-fission spectrum;
— Source with quasi-mono-energetic neutron spectra at epithermal energies using filter techniques (Si for 143 keV; ^{56}Fe for 24 keV).

1.3. Nuclear physics

— Determination of strength functions;
— Experiments to examine the optical nucleus model;
— Experiments on p and d scattering.

2. CONVERTER FACILITY

The basic structure of the converter facility is shown in Fig. 1. An assembly of two converter plates with high enriched uranium (93% of ^{235}U) can be moved from its rest position outside the moderator tank into its working position in front of the nose of the beam tube SR10. The size of the luminous area is determined by the internal dimensions (150 × 150 mm) of the beam tube nose.

In the working position, the converter plate assembly is oriented at an oblique angle with respect to the beam tube axis (see Fig. 2 and also Fig. 3 of

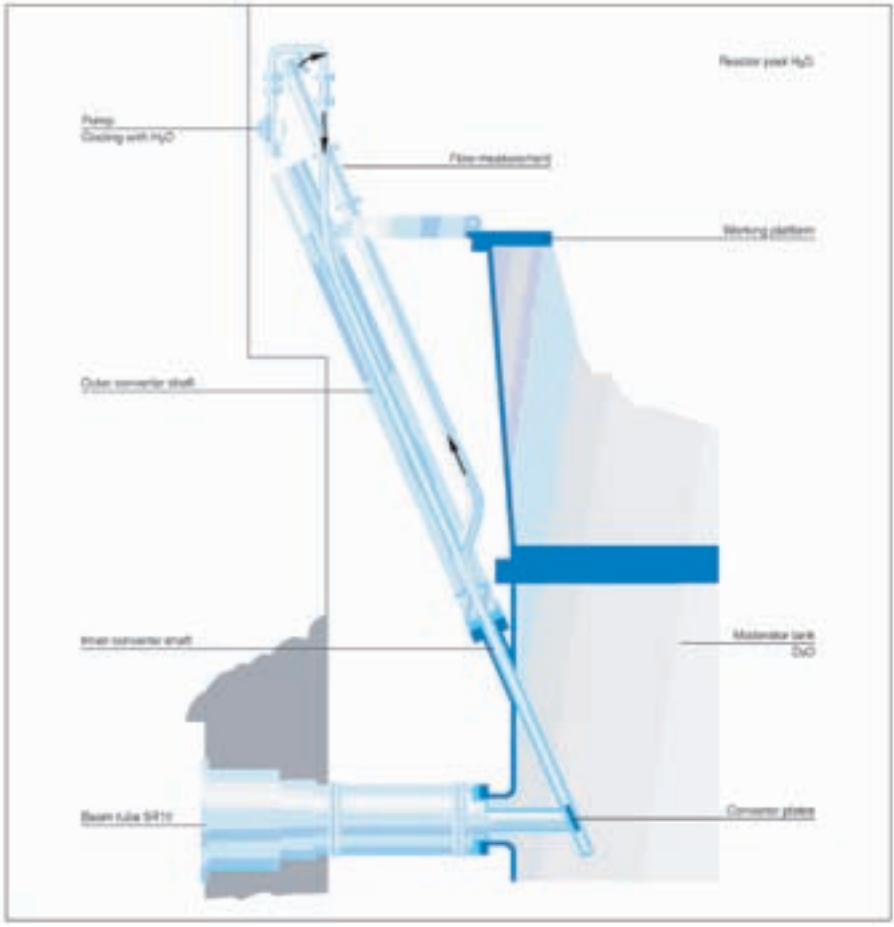

FIG. 1. Vertical section of the converter facility of the FRM-II. The position of the single cylindrical fuel assembly of the reactor core is at the axis of the cylindrical moderator tank, i.e. at the right side of the figure at the vertical level of the beam tubes.

paper 6.4 by Böning and Neuhaus). The thermal power produced is approximately 80 kW. Technical data are shown in Table 1. The heat produced is dissipated to the pool water by two redundant cooling circuits. The flow of coolant as well as the position of the converter plate assembly is monitored to ensure reactor safety. The pumps are safe with respect to an earthquake, and they are battery buffered to provide an emergency power supply free of interruption.

The fast neutron beam can be shut off by means of four drums accommodated in the beam tube port, operating in two sections as a redundant system. In the case where both shutter elements fail simultaneously, a reactor scram is initiated automatically. Additionally, a manual scram exists for special requirements. The safety relevant aspects of the converter facility have been settled together with the licensing procedure of the research reactor. Such components which are relevant for the medical treatment of human patients must fulfil the German 'medical product law' in agreement with the latest European regulations (as of 7 August 2002).

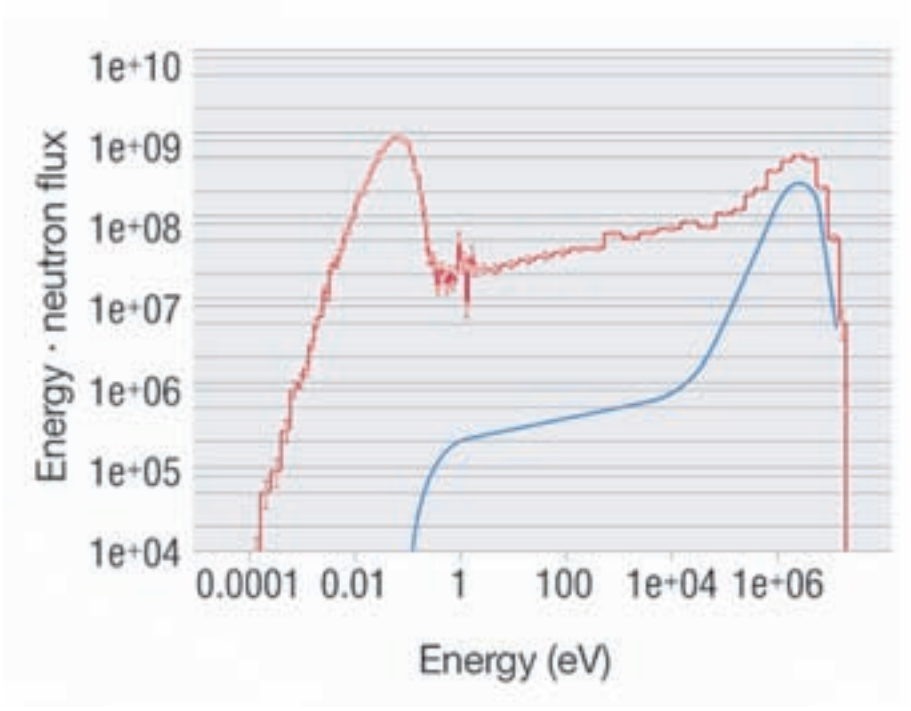

FIG. 2. Energy dependence of the neutron flux at the irradiation position outside the beam tube SR10. The product of neutron energy and neutron flux density is plotted in units of $cm^{-2} \cdot s^{-1}$.

TABLE 1. TECHNICAL DATA OF THE CONVERTER PLATES (THE CONVERTER PLATE ASSEMBLY CONTAINS TWO PLATES WHICH ARE ARRANGED IN PARALLEL, SEPARATED BY A GAP OF 4.5 mm FILLED WITH COOLANT WATER)

External dimensions of 1 converter plate	$250 \times 176 \times 3.26$ mm^3
Dimensions of the uranium zone	$225 \times 160 \times 2.50$ mm^3
Uranium mass of 1 plate (as contained in U_3Si_2-Al dispersion fuel)	270 g U
Enrichment of U-235	93%
Total thermal power of the converter plate assembly	approx. 80 kW
Thermal power of plate facing fuel element	46.9 kW
Thermal power of plate facing beam tube nose	32.7 kW
Throughput of coolant (pool water)	33 m^3/h
Coolant flow velocity along plates	2.9 m/s

The neutron spectrum in the irradiation position is shown in Fig. 2 (red curve). A good approximation of a standard neutron spectrum can be obtained by filtering the primary beam with 20 mm of polyethylene (PE) and 1 mm of cadmium (blue curve).

3. IRRADIATION ROOMS

As shown in Fig. 3, the irradiation rooms consist of a pool wall recess, the medical treatment room and a room for technical and physical experiments. The pool wall recess, which is accessible once the beam shutter has been closed, provides enough space for installations of accessories such as, for example, a set of filters, a light source for optical adjustment, several dosimeters and also a sample position for high dose rate irradiation. Using a multileaf collimator, it is possible to shape arbitrary beam cross-sections. The irradiation position of the patient is on the patient couch in the medical irradiation room at a distance of 5.5 m from the converter plates. A compact beam catcher has been built as a sandwich of polyethylene, lead and iron which is installed in the rear wall of the medical irradiation room, in order to keep the dose rate low, for instance, at the control desk outside of the irradiation rooms.

The adjusted technical installation may remain in place also after medical treatments, since the room for technical and physical experiments is located behind the medical room. For such experiments, the beam catcher as just mentioned can be opened, and interchangeable collimators and even

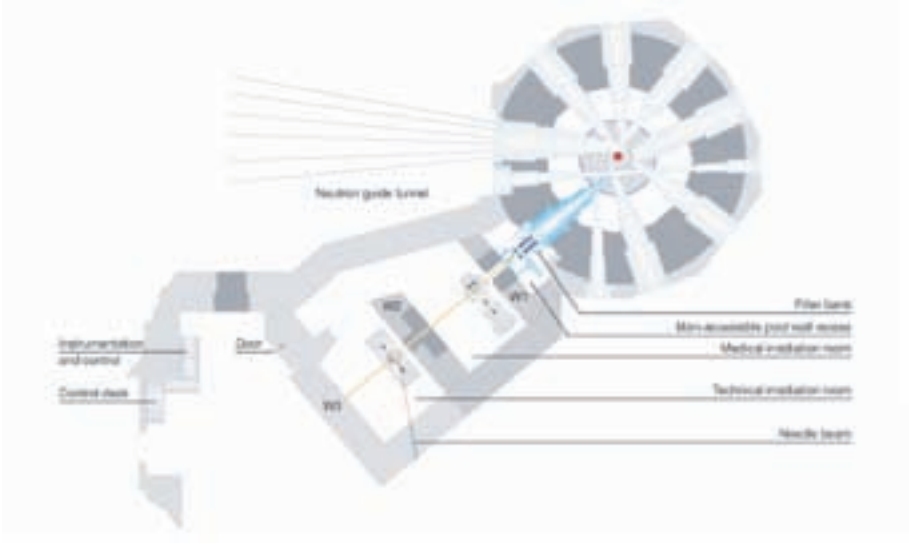

FIG. 3. Horizontal section of the converter facility with irradiation rooms.

multichannels can be installed. Using a scattering target, a narrow passage can be provided through the side wall at an angle of 60° to the beam axis (Fig. 3). In this way, a 'needle beam' of scattered neutrons with energies around 300 keV can be made available in the experimental hall.

4. DESIGN AND PERFORMANCE

The walls of the irradiation casemate are made of shaped blocks and may thus be dismantled. In the medical area, heavy concrete with a density of 4.5 g/cm^3 was used, while standard concrete with a density of 2.4 g/cm^3 was sufficient in the technical applications room. Colemanite, a mineral containing boron and also a high content of crystal water, has been added to both types of concrete thus the shielding could be optimized.

The door of the irradiation rooms (Fig. 3) is 1.05 m wide to allow the transport of service components. For maintenance and repair work, walls W1, W2 and W3 can be moved on air cushions from their normal position into a storage position near the wall of the experimental hall to provide access for the beam tube loading machine and to transport heavy or bulky items into the irradiation rooms.

The flux data in the irradiation positions as well as some technical data are compiled in Table 2. Alternatively to the use of fast neutrons, the converter

TABLE 2. FLUX DATA AND SOME TECHNICAL CHARACTERISTICS OF THE IRRADIATION POSITIONS (FAST FLUXES — CURRENT DENSITIES — ARE GIVEN FOR NEUTRON ENERGIES LARGER THAN 0.1 MeV)

	Pool wall recess	Medical irradiation position	Technical irradiation position
Distance to the converter plates	3.5 m	5.5 m	9.5 m
Maximum beam cross-section	280×190 mm^2	300×200 mm^2	150×150 mm^2
Fast neutron current (without filter)	3.4×10^9 n·cm^{-2}·s^{-1}	1.5×10^9 n·cm^{-2}·s^{-1}	5×10^8 n·cm^{-2}·s^{-1}
Fast neutron current (with filter 2 cm PE + 1 mm Cd)	2.5×10^9 n·cm^{-2}·s^{-1}	1.1×10^9 n·cm^{-2}·s^{-1}	4×10^8 n·cm^{-2}·s^{-1}
Thermal neutron current (without filter)	4.5×10^9 n·cm^{-2}·s^{-1}	2×10^9 n·cm^{-2}·s^{-1}	7×10^8 n·cm^{-2}·s^{-1}
n-dose rate	7.3 Gy/min	3.3 Gy/min	—
γ-dose rate	3.7 Gy/min	1.7 Gy/min	—

plates may be moved in the converter shaft (Fig. 1) back from the working position at the beam tube nose into a rest position outside the moderator tank. In this case, a homogeneous thermal neutron beam with a large cross-section and a current density of 1.5×10^9 n·cm^{-2}·s^{-1} can be made available as characterized by an extremely low fast neutron ($E_n > 0.1$ MeV) contamination of only 4×10^5 n·cm^{-2}·s^{-1}.

For the care and convenience of the patients, dedicated side rooms exist, as do a separate entrance to the reactor building and the experimental hall, a waiting room, a lavatory for handicapped persons, a recovery room and a change cubicle.

5. MEDICAL ASPECTS

Irradiation therapy represents a basic method of cancer treatment; about 60% of all cancer diseases can be successfully treated in oncology. Because of the special properties of fast neutrons, fast neutron therapy further extends the treatment possibilities. So, fast neutrons have a high biological effectiveness in destroying cancer cells. In the DNA helix (the gene strings), fast neutrons

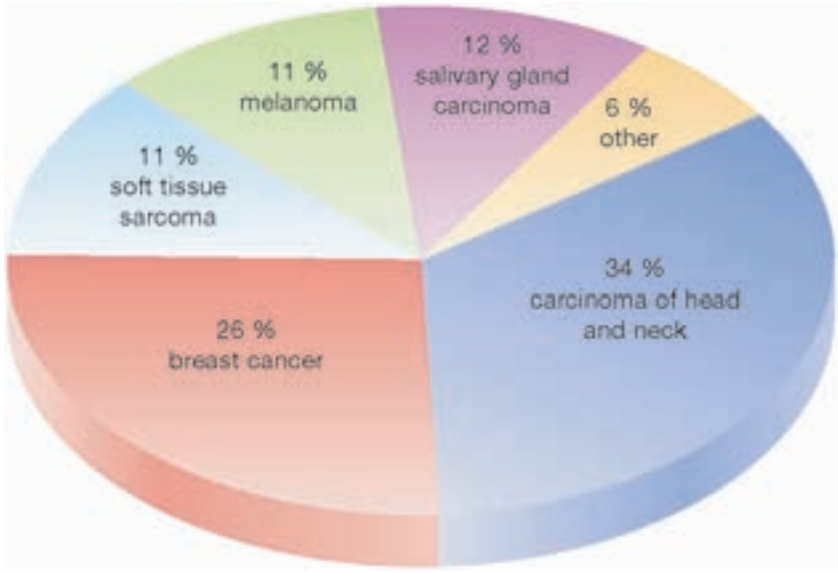

FIG. 4. The various types of cancers which have been treated at the old FRM.

produce about 20 times more double string breaks than X rays. Double string breaks stop the self-repairing mechanism in the cell. This damage of the DNA and — with the same importance — other essential biomolecules results in the death of the cell. Neutron therapy is very advantageous in cases where cancers are resistant to X rays. Underhypoxic tissue can be more efficiently treated with neutrons. As a rule, patients have to be irradiated only once instead of several times. Responsibility for the medical case lies with the clinical hospital, Klinikum rechts der Isar, which is also part of the Technische Universität München. The staff of the FRM-II is only responsible for providing the best possible working conditions.

Some 900 patients already have been treated in about 2500 single irradiations at the old FRM research reactor (‘Atom-Ei’). Some of these treatments were curative, although most of them were palliative. Various types of cancer treated thus far are presented in Fig. 4. As a rule, fast neutron irradiation treatment is being funded, to a certain degree, by health insurance.

At the new FRM-II, a much better facility will be available at which therapeutic irradiations can be realized under conditions which are significantly superior to those at the old FRM. For instance, it will no longer be necessary to subdivide a required larger irradiation area into several smaller areas (with uncertain overlap zones). Typically, the irradiation times will cover a 20–60 s range.

To carry out irradiation on human beings, the device has to be approved according to German ‘medical product law’ and a CE certificate must be obtained.

6. CONCLUSION

The converter facility at the FRM-II fills a gap for using fast reactor neutrons for medical treatments and technical applications in accordance with world best quality. The outstanding features of the neutron beam are its intensity, homogeneity, cross-section and spectral conditions.

Direct irradiations of near surface cancers with fast reactor neutrons are presently being performed at only a few places in the world. The beam intensity is optimal at FRM-II. The radiography and tomography of technical samples with fast neutrons originated from the old FRM as a by-product of the development of detectors and methods. Performing radiography and tomography with fast neutrons gives complimentary information to using thermal or cold neutrons and can be applied, for example, to thick samples and those with a high hydrogen content. Nuclear physics experiments with fast neutrons are not so popular. The possibility to install filters in the pool wall recess allows performing experiments with quasi-mono-energetic neutron beams in the keV energy range.

In this sense, the converter facility is a high performance device, which presents a variety of possibilities for medical, technical and other applications. Its outstanding features help to promote progress in several fields of research.

5.4. NEUTRON CAPTURE THERAPY AT THE MITR-II

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1. INTRODUCTION

Neutron capture therapy (NCT) for cancer is an experimental cellular tumour targeting therapy. This promising but complex radiation therapy requires administering tumour seeking compounds that usually contain ^{10}B and concentrate preferentially in tumour cells followed by irradiation with suitable neutron beams.

Although research in NCT with boron compounds (BNCT) has a long history of development following its origination in the United States of America in the 1950s, the last 15 years have seen a significant expansion of BNCT research in the USA and elsewhere. Major progress has been made on many fronts, including development and construction of high performance epithermal neutron beams, which are approaching the theoretical optimum. Capture compounds have been improved and shown to preferentially accumulate in certain tumours, but further improvements are needed to realize clinical success with BNCT. Various necessary supporting technologies have been established, such as boron analysis, and treatment planning and understanding of the radiobiology of BNCT has progressed significantly.

A wide range of research facilities and interdisciplinary capabilities are required for the development of BNCT as a viable cancer therapy. Some of these facilities, in particular, the neutron irradiation facilities needed for preclinical and clinical research require significant capital investments and are available at very few sites. Currently, the only site in the USA with high performance epithermal and thermal neutron irradiation facilities is located at the Massachusetts Institute of Technology Research Reactor (MITR-II).[1] The 5 MW research reactor operates with a high capacity factor, 24 hours per day for 250–300 days per year. These neutron facilities, along with a variety of

[1] The MIT NCT facilities are supported by the US Department of Energy through the Innovations in Nuclear Infrastructure and Education Program, Office of Nuclear Energy Science, and Technology, as well as the Office of Environmental and Biological Research. MIT provides the bulk of the support for operating the MITR-II.

supporting technologies and capabilities, were developed to support preclinical and clinical NCT research programmes for MIT and its medical collaborators. The recently constructed or upgraded facilities that include thermal and epithermal neutron beams with automated, fault tolerant control systems and macroscopic, as well as microscopic, boron analysis capabilities are described. A brief summary of the status of clinical trials at MIT is also provided.

2. THERMAL NEUTRON IRRADIATION FACILITY

Low energy thermal neutron beams are necessary for preclinical studies with small animals or cell cultures and for clinical irradiations of shallow malignancies. The recently upgraded and renovated thermal neutron irradiation facility located beneath the core of the MITR is shown on the lower left of Fig. 1.

Source neutrons for the thermal neutron beam (designated the M-011 beam) originate in the D_2O reflector/moderator below the core of the MITR. These neutrons are very well thermalized before they enter the vertical beam line, and have an energy spectrum close to that of a 304°C Maxwell–Boltzmann distribution that is optimal for NCT irradiations of relatively superficial targets within 30 mm of the surface. The thermal neutrons are collimated by reactor grade graphite walls of the vertical beam line that also elastically scatter many neutrons back into the main beam, which helps preserve beam intensity. Three separate shutters of water, Boral and lead plus borated polyethylene are used to control the beam before it reaches the opening into the irradiation room.

Irradiations are monitored by four similar fission counters located at the edge of the beam and spaced at 90° intervals. Their output is processed in nuclear instrumentation module (NIM) electronics and sent to industrial quality programmable logic controllers (PLCs). The four fission counters are used to monitor not only intensity but also symmetry of the beam during an irradiation. This provides an overall system check of beam stability. The beam monitoring, control and safety systems utilize redundant fission counters, signal processing, electronic circuits and PLCs that permit irradiations to safely continue should any individual component fail. The PLCs log essential data to a dedicated computer and issue commands that automatically open and close shutters to ensure that neutron fluences (doses) are delivered to within 1% of the prescribed target. Safety interlocks to protect the staff and patients are also continuously monitored by the PLCs. A computer with a large screen monitor is used for on-line display of the status of the irradiation and for record keeping but does not control the irradiations. This important function is exclusively assigned to the redundant PLCs. All important systems have backup power

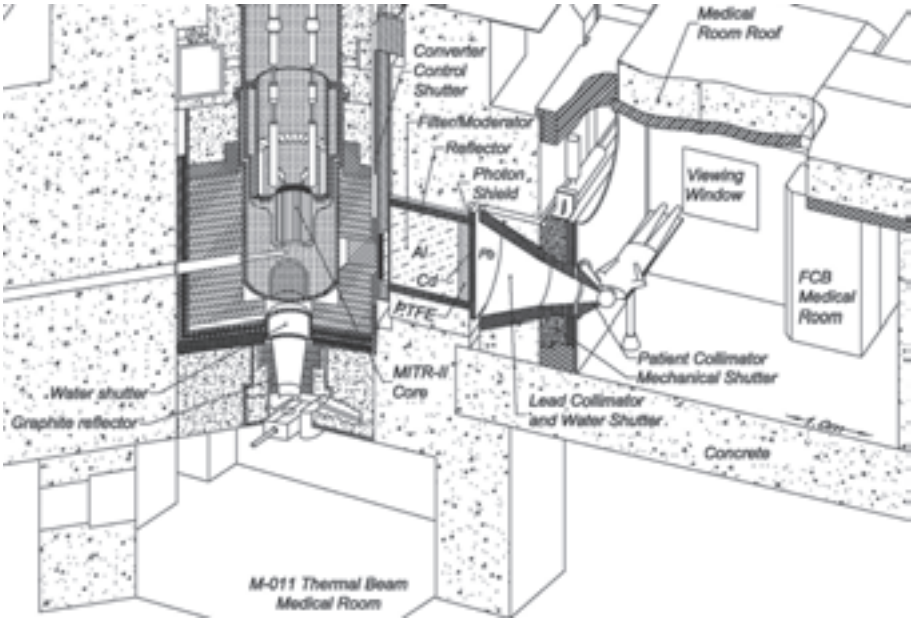

FIG. 1. An isometric of the MITR-II showing the thermal neutron irradiation facility, M-011 directly beneath the core of the 5 MW research reactor and the epithermal beam facility, FCB (right side of figure), located in the main reactor hall.

from an uninterruptible power supply. Manual controls, located on the M-011 operator's console, can be used to override the automated functions of the PLCs at any time by closing shutters or, if necessary, scramming the reactor.

Irradiations using the M-011 beam are performed in a well shielded irradiation room, shown in Fig. 1. Experiments or patients can be viewed during an irradiation through a shielded window and closed-circuit television monitors. Background dose rates, with the reactor at full power and all shutters closed, are low enough to allow staff to enter and work in the irradiation room.

Small animal or cell culture irradiations are usually carried out by positioning the animals or cell flasks in a lithiated polyethylene box. For in vivo irradiations, a 25 mm thick lithiated polyethylene lid shields the animals and directs a collimated beam to the part of the animal to be irradiated. Phantoms, ionization chambers and gold foils can also be positioned in the shielding box to experimentally determine dose rates from thermal neutrons, gamma rays and fast neutrons. Figure 2 shows a photograph of the shielding box with rats positioned for brain irradiations. The box is inserted into a recess in the lower beam shutter that accurately positions the animals in the beam line when the shutter is opened. Typical beam-only, no capture compound, irradiation times

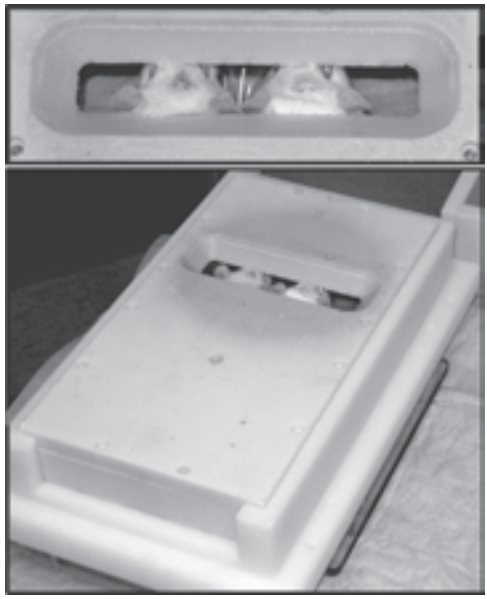

FIG. 2. Shielding box used for small animal experiments with tumour bearing rats positioned for brain irradiations in the thermal neutron beam.

for a single field to tissue tolerance, assumed to be 12 (RBE) Gy (using RBEs of 1 and 3.2, respectively, for photons and neutrons) are 21 min. With a boron compound that accumulates 18 μg of ^{10}B/g of normal tissue, a tolerance dose of 12 (RBE) Gy takes 4.4 min for a single field assuming the RBE for the $^{10}B(n,\alpha)^{7}Li$ reaction is 3.8.

When patients are irradiated in the thermal beam, an additional collimator is mounted below the lowest shutter and the patient is positioned on a couch which is raised hydraulically towards the ceiling and the aperture of the collimator. Figure 3 shows a patient collimator. Patient collimator aperture size and shape can be readily changed. Currently two collimators with circular apertures and diameters of 80 mm and 120 mm are available. Intensities for patient or large animal irradiations are lower than those for small animals irradiated in the shielded box. Nevertheless, with capture compounds, such as boronated phenylalanine (BPA), 12 (RBE) Gy tolerance doses can be delivered in about 6 min in a single field.

The M-011 beam has been fully characterized using the procedures routinely employed for mixed field dosimetry at MIT. Table 1 provides in-air measurements of the thermal neutron flux, and photon and fast neutron dose rates at the position typically used for small animal irradiations. The high thermal neutron flux of 5.9×10^{9} n·cm^{-2}·s^{-1} combined with low photon and

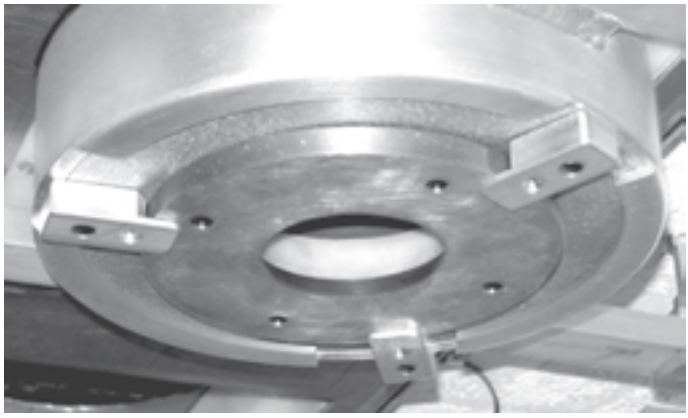

FIG. 3. A collimator used with the thermal neutron beam when irradiating patients or large animals.

negligible fast neutron contamination show that the M-011 beam can be expected to provide excellent performance for NCT irradiations.

In Fig. 4, the RBE weighted dose versus depth behaviour is shown along the central axis of an ellipsoidal water filled head phantom for all important dose components in the M-011 beam using BPA as a capture compound and the 120 mm beam aperture. The slow neutron and photon components are measured using paired ionization chambers, while the $^{10}B(n,\alpha)^{7}Li$ and $^{14}N(n,p)^{14}C$ doses are determined from the measured thermal flux using kerma coefficients of 8.66×10^{-8} and 7.88×10^{-12} Gy·cm^2, respectively. The boron concentration of 18 μg/g, which has been observed in preclinical and clinical research, is used together with a tissue nitrogen concentration of 3.5% in muscle. A boron concentration of 65 μg/g is applied for the tumour with an assumed RBE of 3.8. Normal tissue dose is dominated by the $^{10}B(n,\alpha)^{7}Li$ reaction due to the relatively high concentration of BPA. Photons and thermal neutron capture reactions in tissue nitrogen also produce some adventitious dose to normal tissue. However, the fast neutron dose component in this high purity beam is so low that it has negligible influence on the total normal tissue dose even at the expected maximum near the surface. The useful penetration or advantage depth (AD) and the maximum normal tissue dose rate (that is also the tumour dose rate at the advantage depth, ADDR) shown in Fig. 4 are, respectively, 35 mm and 2.4 (RBE) Gy/min. Normal tissue tolerance doses of 12 (RBE) Gy can be reached in 6 min and a therapeutic ratio greater than one would be expected at depths less than or equal to the AD.

A well benchmarked Monte Carlo model of the M-011 beam has been developed. This model allows the accurate calculation of flux and doses for any conceivable experimental study including clinical trials.

TABLE 1. IN-AIR MEASUREMENTS OF THERMAL NEUTRON FLUX, FAST NEUTRON AND PHOTON DOSE COMPONENTS IN THE M-01 THERMAL BEAM AT THE POSITION USED FOR SMALL ANIMAL IRRADIATIONS. THE ABSORBED DOSES PER NEUTRON PER cm^2 OR SPECIFIC DOSES ARE ALSO PROVIDED. REACTOR POWER IS 5 MW

Thermal flux (10^9 n·cm^{-2}·s^{-1})	Fast neutron dose rate (mGy/min)	Photon dose rate (mGy/min)	Specific fast neutron dose (10^{-13} Gy·cm^2)	Specific photon dose (10^{-13} Gy·cm^2)
5.9 ± 0.3	<20	188 ± 10	<0.5	5.3 ± 0.4

The MIT thermal neutron irradiation facility, with its high intensity, low background beam and automated control systems, is well suited for the majority of preclinical research studies in BNCT. Small animal and cell culture irradiations can easily be performed in the facility and clinical studies of shallow tumours can also be fully supported.

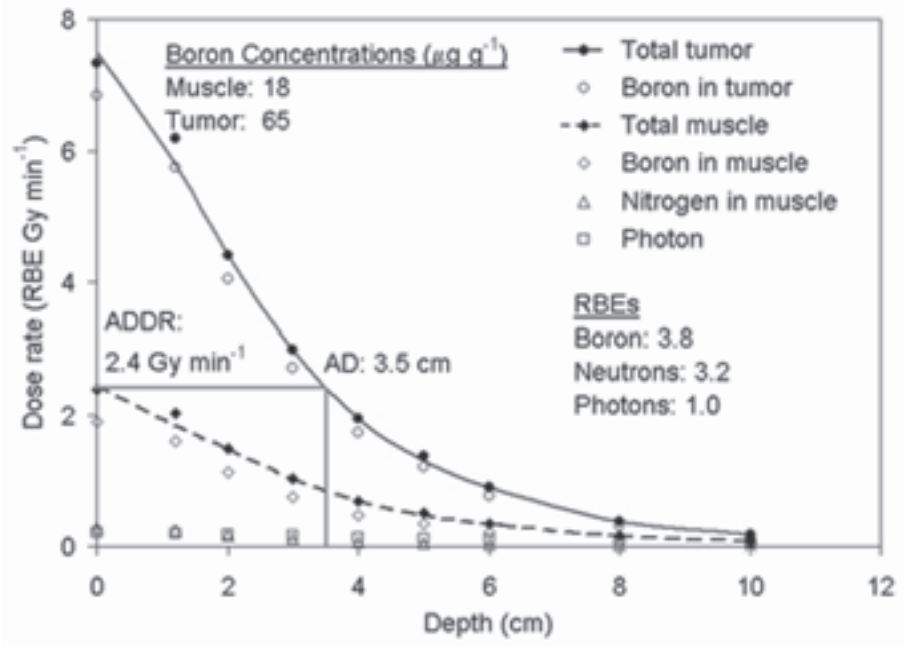

FIG. 4. The weighted depth–dose distribution for the M-011 beam measured along the central axis of a water filled ellipsoidal head phantom using the 120 mm diameter aperture.

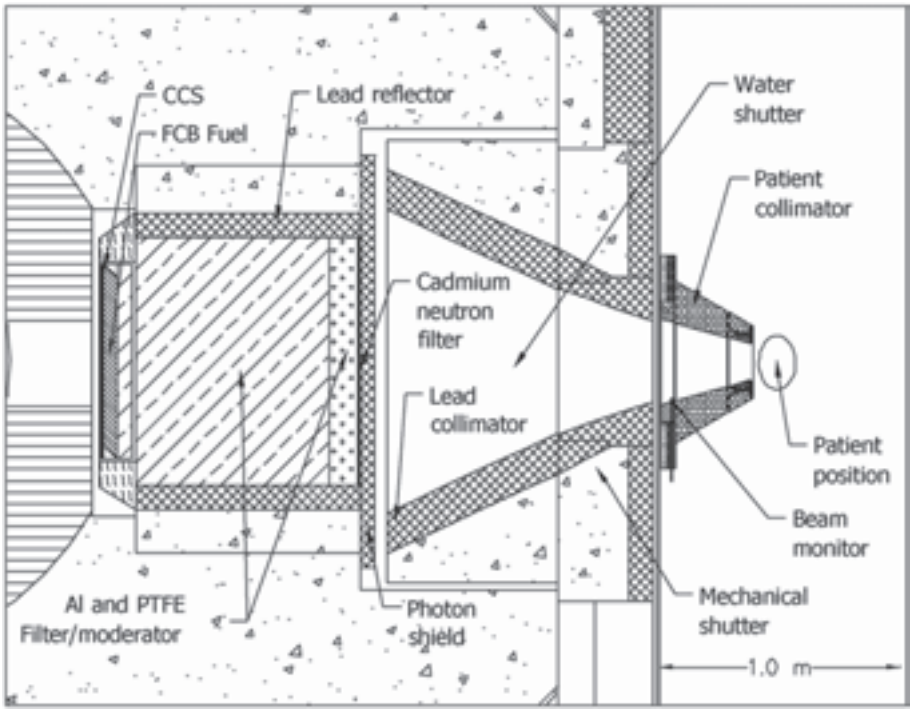

FIG. 5. Plan view of the epithermal beam line (FCB) from the converter fuel to the patient position.

3. EPITHERMAL NEUTRON IRRADIATION FACILITY

The MIT operates a recently commissioned state of the art high intensity, low background epithermal neutron (0.5 eV < E < 10 keV) irradiation facility known as the FCB. An isometric view of the FCB facility is included on the right side of Fig. 1, while Fig. 5 shows the horizontal beam line in greater detail.

The neutron source for the FCB is a highly subcritical fission converter which is cooled by heavy water and driven by thermal neutrons from the MITR. The converter is currently fuelled with 10 burned MITR fuel assemblies and produces 83 kW of power with the reactor at 5 MW. The licence limit of 250 kW allows for significant increases in converter power. The fission neutrons originating in the converter are moderated and filtered by aluminium, polytetrafluoroethylene (Teflon) and cadmium. Undesired photons are attenuated with a lead shield. The resulting large area epithermal neutron beam is directed towards the patient irradiation position by a lead lined

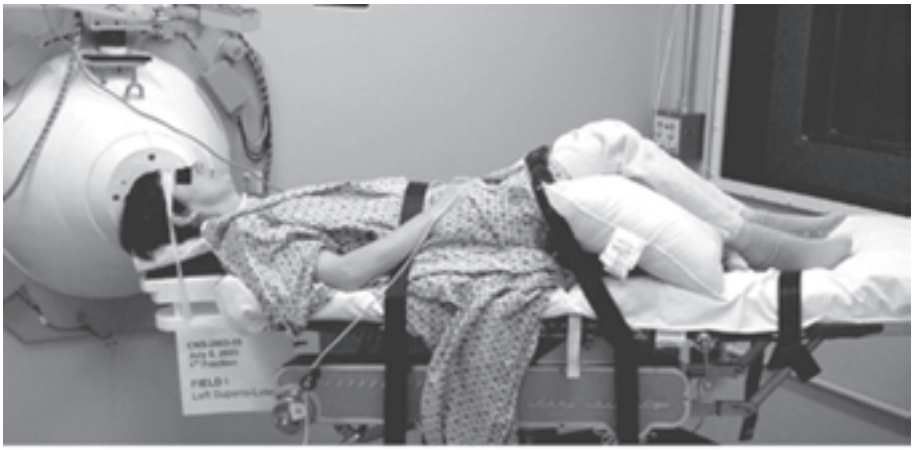

FIG. 6. Set-up for a brain irradiation in the FCB medical room.

collimator. A final or patient collimator constructed of lead and lithium or boron loaded epoxy protrudes into the well shielded medical irradiation room and features easily variable apertures ranging from 80 to 160 mm in diameter. Other sizes or shapes of apertures can be implemented if required. The variable aperture patient collimator and its extension into the medical room allow easy positioning of patients for radiation field placement anywhere on the body. The irradiation room has sufficient space for gurneys and positioning couches. Patient observation during irradiation is facilitated by a large shielded glass window and closed-circuit television monitors. Two-way audio communication is also available between the patient and clinical staff. Lasers, including a beam's eye view, are used for precise patient set-up.

Three shutters control the FCB. A beam monitoring and control system which is similar to that previously described for the M-011 thermal beam is used to automatically open and close the three beam shutters: a thermal neutron shutter on the reactor side of the converter can vary the converter's power over two orders of magnitude, a large water shutter in the lead collimator and a fast acting lead and boron loaded heavy concrete shutter near the patient collimator. Prescribed neutron fluences are routinely delivered to within 1% of the target using this system. Figure 6 shows a typical patient set-up for a brain irradiation in the FCB.

Table 2 provides the in-air epithermal neutron flux and dose rates from photon and fast neutron radiation in the beam. The epithermal flux of 3.2–4.6 $\times 10^9$ n·cm^{-2}·s^{-1}, depending upon final collimation, is currently the highest available of any epithermal neutron beams suited for BNCT and, if desired, higher intensity can be obtained by optimizing converter fuel loading and/or

TABLE 2. IN-AIR MEASUREMENTS OF THE EPITHERMAL FLUX, FAST NEUTRON AND PHOTON DOSE RATES IN THE FCB AT THE PATIENT POSITION WITH SEVERAL DIFFERENT COLLIMATORS. SPECIFIC DOSES ARE ALSO PROVIDED. REACTOR POWER IS 5 MW AND THE FCB POWER IS 83 KW

Aperture diameter (mm)	Epithermal flux (10^9 n·cm^{-2}·s^{-1})	Fast neutron dose rate (mGy/min)	Photon dose rate (mGy/min)	Specific fast neutron dose (10^{-13} Gy·cm^2)	Specific photon dose (10^{-13} Gy·cm^2)
160	4.62 ± 0.60	38.2 ± 4.0	96.7 ± 4.2	1.38 ± 0.22	3.49 ± 0.48
120	4.29 ± 0.56	36.3 ± 3.8	93.7 ± 4.1	1.41 ± 0.22	3.64 ± 0.50
100	3.41 ± 0.44	34.0 ± 3.5	90.1 ± 3.9	1.66 ± 0.26	4.40 ± 0.60
80	3.17 ± 0.40	34.1 ± 3.5	74.4 ± 3.2	1.79 ± 0.28	3.91 ± 0.54

increasing reactor power without affecting the excellent beam characteristics. Specific beam contamination from photons and neutrons is also provided in Table 2.

The contamination levels in the FCB are very low and have a minimal influence on beam performance, which is illustrated by the example shown in Fig. 7 where the dose versus depth distributions are shown along the central axis of a water filled ellipsoidal head phantom in the 120 mm aperture beam. Tumour and normal tissue (blood) concentrations and the applied RBEs are consistent with those observed for BPA and are the same as those used in Fig. 4 for the thermal beam, except that a nitrogen concentration of 2.2% is applied for the brain and a compound RBE of 1.3 is used to account for the microdistribution of boron that accumulates in normal brain. The useful penetration or advantage depth using the 120 mm collimator is 93 mm. This is adequate to treat the deepest locations in an average human brain. The peak dose rate to normal tissue is 1.25 (RBE) Gy/min, which implies that a peak tolerance dose of 12 (RBE) Gy can be delivered in 9.6 min with a single beam placement.

A ^{6}Li filter has been constructed to further increase the useful beam penetration by approximately 6 mm with the BPA compound. An increase in converter power by 30–50% is planned in the near future that will help compensate for the intensity loss when the lithium filter is used in an irradiation.

The high intensity, high purity FCB is very well suited for clinical trials or large animal irradiations where deep penetration to the cancer site is required.

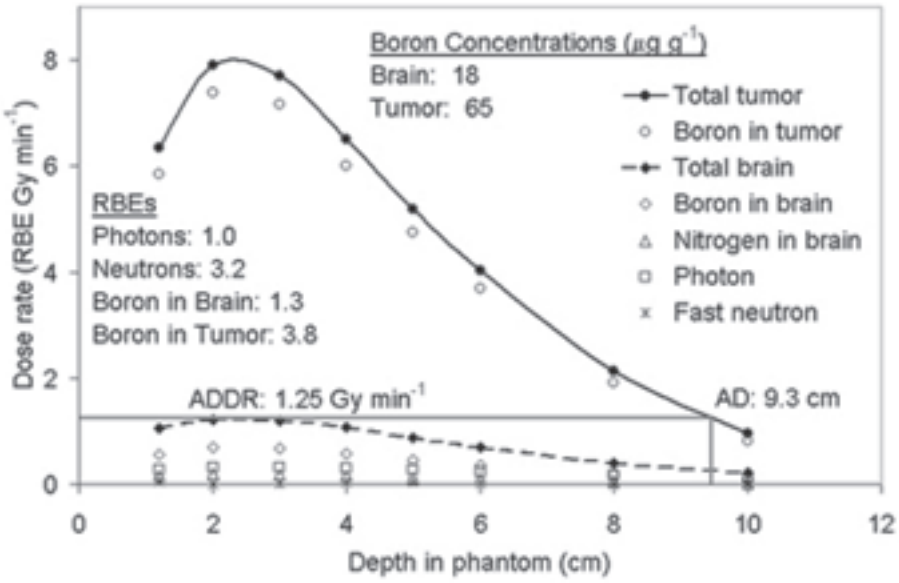

FIG. 7. Dose versus depth distribution of the fission converter beam in an ellipsoidal water head phantom using a 120 mm diameter collimator aperture.

4. PROMPT GAMMA BORON ANALYSIS

Non-destructive analysis of biological samples for the ^{10}B isotope is carried out using a prompt gamma neutron activation analysis (PGNAA) system at the MITR-II, shown schematically in Fig. 8.

This facility uses the unconventional approach of a diffracted beam rather than a direct beam with all thermal energy components. The intensity losses inherent in this approach are partially compensated by using a high mosaic spread graphite crystal with vertical focusing. Furthermore, by using a diffracted beam, the background of high energy neutrons and gammas is greatly reduced. The solid state, high purity germanium detector can then be placed near the sample to maximize geometric counting efficiency without overloading the detector or counting electronics. The low dose rate in the analytical beam line also reduces shielding requirements, allowing samples to be safely placed manually and eliminating the need for precise remote positioning devices. Typical samples consist of blood or tissue with volumes ranging from 0.1 to 10 mL. No special sample preparation is required and the high sensitivity of the MIT prompt gamma system, 18 counts $s^{-1} \cdot \mu g^{-1}$ of ^{10}B, allows rapid analysis, usually in several minutes, for 2 μg or more of ^{10}B. The

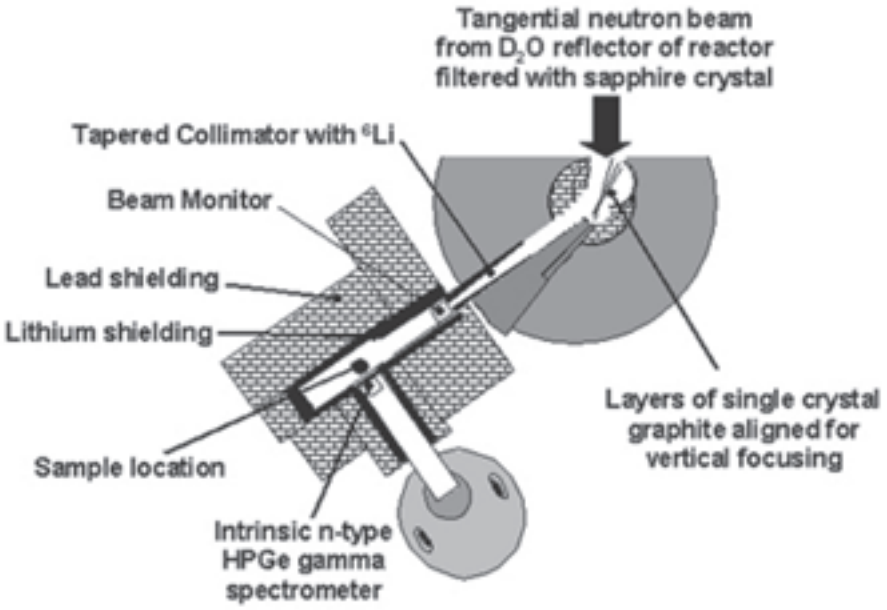

FIG. 8. Beam line of the prompt gamma activation analysis system.

PGNAA facility is located in the main experimental hall of the reactor, convenient to both the thermal and epithermal neutron beams so that boron concentrations can be measured concurrently with the use of these facilities. PGNAA results can be readily incorporated into animal experiments or patient irradiations to adjust beam delivery and precisely administer the desired (or prescribed) dose from capture in boron.

5. MICROSCOPIC IMAGING OF ^{10}B IN TISSUE

MIT in collaboration with scientists from the New England Medical Center and the Beth Israel Deaconess Medical Center developed a technique for imaging the boron microdistribution superimposed on the corresponding cell morphology in tissue samples. High resolution quantitative autoradiography (HRQAR) has a spatial resolution of about 2 μm and a sensitivity for ^{10}B of approximately 0.1 μg/g. HRQAR can be applied to frozen tissue sections (2–4 μm thick) that have been stained to reveal the desired cellular structures. Figure 9 is an image of rat skin with black superimposed tracks representing boron atoms that were obtained by HRQAR. Though labour intensive, this technique can be very useful for evaluating radiobiological effects arising from

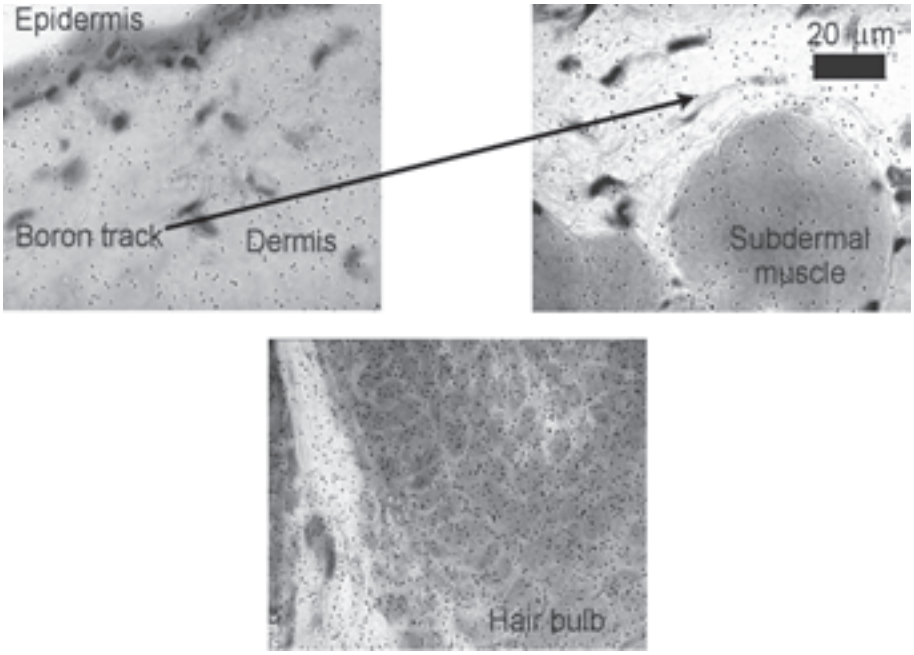

FIG. 9. HRQAR autoradiogram showing tracks from neutron reactions with boron contained in tissue. From the microscopic image analysis of this sectioned rat skin, the boron concentration in the epidermis, dermis and subdermal muscle was measured at 20 µg/g while the hair bulb measured 60 µg/g. (Photographs: W.S. Kiger III.)

different boron compounds or disease sites where the response is mediated by parenchyma cells.

6. PRECLINICAL AND TRANSLATIONAL RESEARCH

A wide variety of clinical and translational research is carried out using the neutron facilities at the MITR. The thermal neutron irradiation facility and the prompt gamma analysis systems are often used for studies to determine the efficacy of experimental capture compounds and new approaches to drug delivery using small animals with different types of implanted tumours. Other studies using both the thermal and epithermal neutron beams are designed to study the sensitivity of normal surrounding tissue to NCT irradiation or to help understand the mechanisms that mediate biological response to radiation exposure. Researchers outside MIT also have access to the NCT facilities at the MITR-II, as well as to the resident physics and engineering expertise that is necessary to design, perform and analyse results from their research.

7. CLINICAL STUDIES

The MIT NCT group in collaboration with Boston medical researchers has performed experimental clinical trials or studies of BNCT. Using a previously available epithermal neutron beam (M-67), a total of 20 patients were irradiated in a Phase I trial for highly refractory brain cancers glioblastoma multiforme (GBM) and intracranial metastatic melanoma. The M-67 beam was also used for five cases of peripheral metastatic melanoma. Using the newer epithermal neutron beam from the FCB, six GBM patients were irradiated in a Phase I/II clinical study, and one patient with peripheral metastatic melanoma in a Phase II clinical study. These studies have helped establish an average whole brain tolerance dose for BNCT using the capture compound BPA. With GBM, these studies also indicate that using only one or two irradiations, BNCT achieves survival comparable to conventional photon therapy that requires many weeks of treatment. The clinical results for both peripheral and intracranial metastatic melanoma using BPA mediated BNCT are encouraging, though not yet conclusive, due to the small total number of patients. Partial or complete local control has been observed in most of the melanoma patients that were evaluated.

8. SUMMARY

Research and clinical studies in NCT require a wide range of specialized neutron facilities. High intensity and high purity thermal and epithermal neutron irradiation facilities, as well as macroscopic and microscopic boron analysis capabilities, are crucial for this research and have been developed for dedicated use at the MITR-II. These facilities are presently being used to advance this complex but promising cellular targeting radiation therapy.

5.5. EXPERIENCE WITH A MODERN BNCT FACILITY AT THE 250 kW FiR 1 TRIGA RESEARCH REACTOR

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1. INTRODUCTION

When the future of the reactor was challenged, utilizing the Finnish Research Reactor 1 (FiR 1) — a 250 kW Triga research reactor located at the Technical Research Center of Finland (VTT) — as a neutron source for boron neutron capture therapy (BNCT), was screened as a viable option in 1990. The initiative for starting BNCT in Finland came from the medical radioisotope group at VTT, which had established close contacts with the boron carrier development work in the United States of America. Also on the basis of contacts with the medical radioisotope group, the Finnish BNCT project started incorporating medical and medical physics design constraints [1–3]. As the primary requirement, the suitability of the FiR 1 nuclear reactor for BNCT was evaluated. The basic design performed by VTT showed that an epithermal neutron beam suitable for BNCT of malignant brain tumours, such as glioma, could be constructed with world class performance characteristics [4]. After gaining enough support from both the medical community as well as private and state financing sources, a decision was made in 1994 to construct a BNCT facility at the FiR 1.

The FiR 1 is located within the Helsinki metropolitan area, with about one million inhabitants, at Otaniemi, Espoo, about 6 km from the largest hospital in Finland, the Helsinki University Central Hospital. The FiR 1 was put into operation in 1962. It functioned as a training and research reactor for neutron activation analysis, radioisotope production and neutron physics until the mid-1990s. In 1996, an epithermal neutron beam was constructed based on a new neutron moderator material, Fluental, developed at VTT [4, 5, 19]. After the successful demonstration of a high purity epithermal beam, the patient irradiation room was constructed by partly cutting into the original concrete shielding of the reactor (Fig. 1). The Fluental moderator section was shortened to create, at that time, the highest intensity and best purity epithermal neutron beam for BNCT. The entire reactor building was renovated, including the construction of irradiation simulation and monitoring rooms as well as a laboratory for boron analysis, thereby creating a dedicated clinical BNCT facility at the reactor site [2].

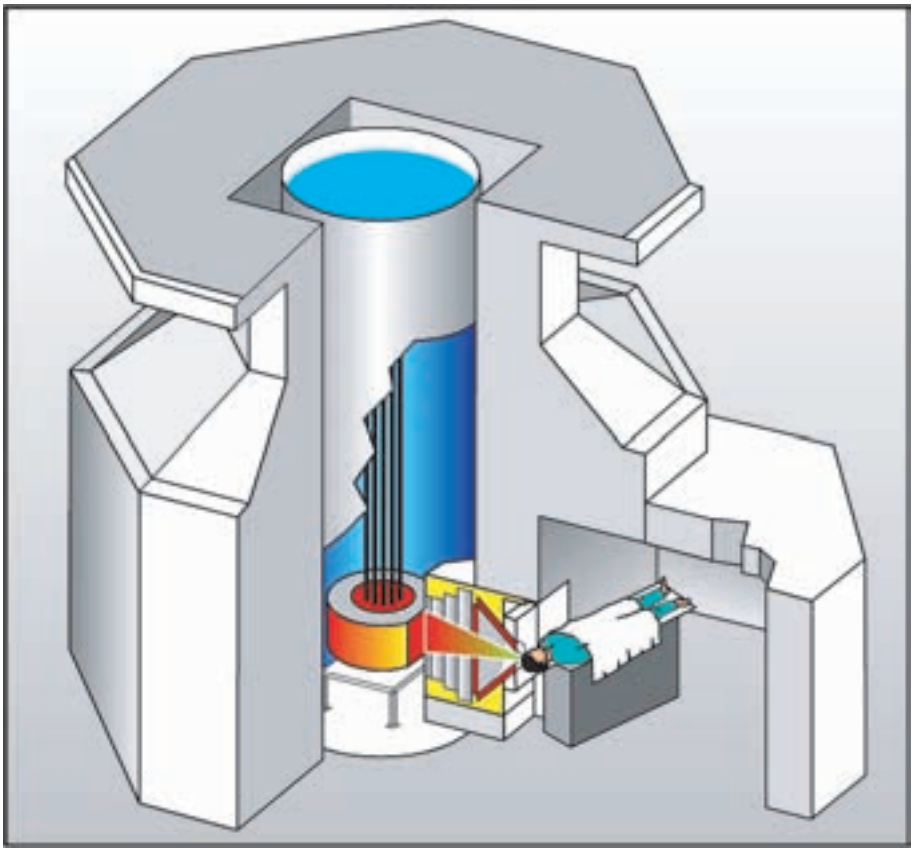

FIG. 1. The BNCT facility at the FiR 1.

Clinical trials for glioma and later for other tumours have been performed since May 1999 by the treatment organization led by the development company Boneca Corporation [6–8]. The BNCT facility has been licensed for clinical use and is being surveyed by the Radiation and Nuclear Safety Authority (STUK). The epithermal neutron beam at the FiR 1 is well characterized and particularly well suited for BNCT because of its low hydrogen recoil and low incident gamma doses, and its high intensity and penetrating neutron spectrum characteristics [9–13].

Currently, BNCT is the main justification for operating the FiR 1. The BNCT work dominates the current utilization of the reactor: three or four days per week are reserved for BNCT application and the rest for other purposes, such as radioisotope production and neutron activation analysis.

2. SUITABILITY OF A TRIGA FOR BNCT

2.1. FiR 1 design

The FiR 1 operated by the VTT is a 250 kW Triga Mark II open tank reactor with a graphite reflector and a core loading of 15 kg U containing 3 kg ^{235}U (19.9% enrichment) in the special Triga uranium zirconium hydride fuel (8–12 wt% U, 91% Zr, 1% H) (see Refs [14–16]). The advantages of the Triga design for BNCT include large flux per watt feature and inherent safety of the Triga fuel. Due to the strong negative fuel temperature coefficient of reactivity, a characteristic of Triga fuels, and the easy operation of this type of relatively low power research reactor, the FiR 1 is a safe neutron source for a clinical BNCT facility.

At the FiR 1, there is a good neutronics linkage from the core to the thermal–epithermal column system. This link was created by cutting a thermal column into the cylindrical radial reflector of the core, and due to the space between the core and that column being void to a large extent due to the tangential beam tube and the irradiation ring. The removal of the thermal column did not influence the reactivity level of the core. This was first checked by calculations and confirmed later. Control rod positions have been reorganized and an asymmetric core loading employed to maximize the source flux in the direction of the BNCT moderator. Using Ni activation in the LS[1] irradiation facility, it was verified that the flux towards the BNCT moderator improved by about 30% applying the core rearrangement (Figs 2 and 3).

The option to improve the neutron production with a fission converter plate in the thermal column turned out to give no benefits in beam intensity. A fission converter, which produces a harder neutron spectrum, would require a thicker moderator, leading to the same beam intensity as using the reactor itself as the neutron source applying a thin moderator [4].

The FiR 1, which was originally built in 1962, is kept in good repair. In 1981, the instrumentation was completely renewed and is still performing well. In 1997, the cooling systems as well as the whole reactor building were renovated.

[1] LS: Lazy Susan, the name given to a circular irradiation facility surrounding the core.

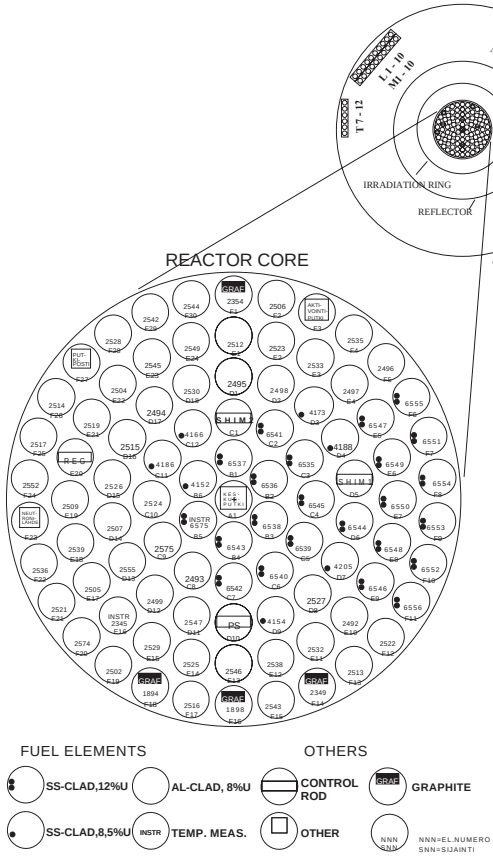

FIG. 2. The FiR 1 core loading. The fuel elements with the highest concentration (12 wt%) of uranium are located towards the BNCT facility (top right). The regulating control rod (REG) is located at the opposite side. The SHIM 1 control rod is fully pulled out when reactor power is on.

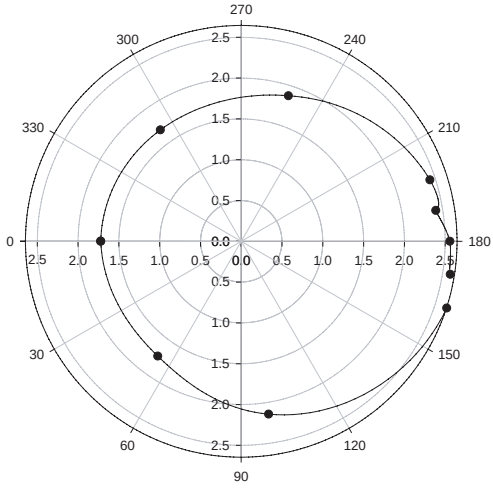

FIG. 3. Angular distribution of ^{58}Ni(n,p) reaction rates at the FiR 1 LZ in units of 10^{14}/s proving the inclined power distribution of the core. The BNCT facility is located on the right.

2.2. Easy operation

A BNCT irradiation is started by manual reactor startup and terminated by manual scram. During BNCT irradiation, the reactor is operated at full power (250 kW). During the irradiation, the reactor power is maintained at the specified level by the automated reactor control system, but a reactor operator performs the initial startup procedure manually. The power is raised at the request of the BNCT facility operator. The length of the irradiation is controlled by the measurands delivered. The BNCT facility operator initiates a manual scram when the specified amount of beam monitor units is reached.

Without applying a beam shutter, a rather rapid operation is achieved: startup after closing the irradiation room door takes 3 min. The patient can be removed 5 min following the reactor scram, avoiding unacceptable personal doses.

2.3. Reactor safety

Due to the inherent safety features of Triga fuel and the easy operation of this type of relatively low power research reactor, the FiR 1 is a safe neutron source for a clinical BNCT facility, as already mentioned. The reactor has a good safety and availability record spanning 42 years of operation.

Moreover, the medical team, together with the reactor personnel, have been trained for incidents and emergency situations at the reactor. Emphasis has been put on reactor staff and communication with personnel of the BNCT facility. For example, if the situation allows, the shift supervisor of the reactor will consult the persons responsible for patient irradiation — the radiation oncologist and the medical physicist — before making decisions about shutting down the reactor or evacuating the reactor building in the case of an emergency, such as a leaking fuel element.

3. BNCT FACILITY AT THE FiR 1

The old training and research reactor facility has gone through major changes in becoming a BNCT clinic. Firstly, an epithermal beam facility was constructed into the radiation shield of the reactor, and around that facility exit, an irradiation room with a patient positioning system was added. Rooms for the BNCT facility control, patient preparation, boron analysis and dosimetry work have been created. Offices and meeting rooms have been rebuilt. A new entrance for BNCT personnel and BNCT patients has been opened for easy access to the BNCT facility. The top of the reactor tank was

separated from the reactor hall in order to confine contamination in case of leakage from irradiation samples or fuel elements. The ventilation of the building, the emergency power supply system, the heat exchangers, and the secondary cooling loop of the reactor — including the cooling towers — were completely redesigned and rebuilt.

3.1. Design process

The design of the treatment facility at the FiR 1 was performed in three phases. During the first phase in 1996, a conceptual design was conducted. Based on that, in the second phase, the architectural and in-house technical planning for the renovation of the reactor building was accomplished. In the third phase, a more detailed design concept for the irradiation room was carried out, including patient positioning and fixation.

During the first phase, the patient treatment and irradiation process, together with the requirements of dosimetry research and radiobiological experiments, were analysed. This work was performed with an industrial designer as a project consultant. Using interviews, meetings and written enquiries, the consultant prepared a report containing the analysis and a conceptual design for the treatment facility. With this approach, all activities to be performed at the treatment facility were included in the work planning. The analysis and planning also included all the BNCT linked activities at the hospital, especially the imaging for treatment planning.

During the conceptual design, space and other resources were allocated to the different activities in the reactor building. All the dispatched personnel with descriptions of their duties were listed, as well as all equipment and all technical systems required for patient irradiations or research activities. The starting point was an analysis of the patient logistics during the whole visit of the patient at the FiR 1 and the supervision of the entire procedure. The technical systems considered included communication facilities (telephone system, data terminals and television monitoring), patient monitoring equipment, and systems for patient transport and fixation. The comfort of the patient and his or her relation to the treatment environment were considered. Visual design principles were established. In addition, hygienic aspects, cleaning and waste management were taken into account. Offices and other spaces for the BNCT clinic and for the research facility were planned. Systems and procedures for personnel and visitor access were created.

The final design phase of the irradiation room included detailed planning of the room dimensions, radiation shielding and equipment, especially the patient fixation and positioning system. A cardboard model of the beam surroundings was made to evaluate whether the planned cutting of the concrete

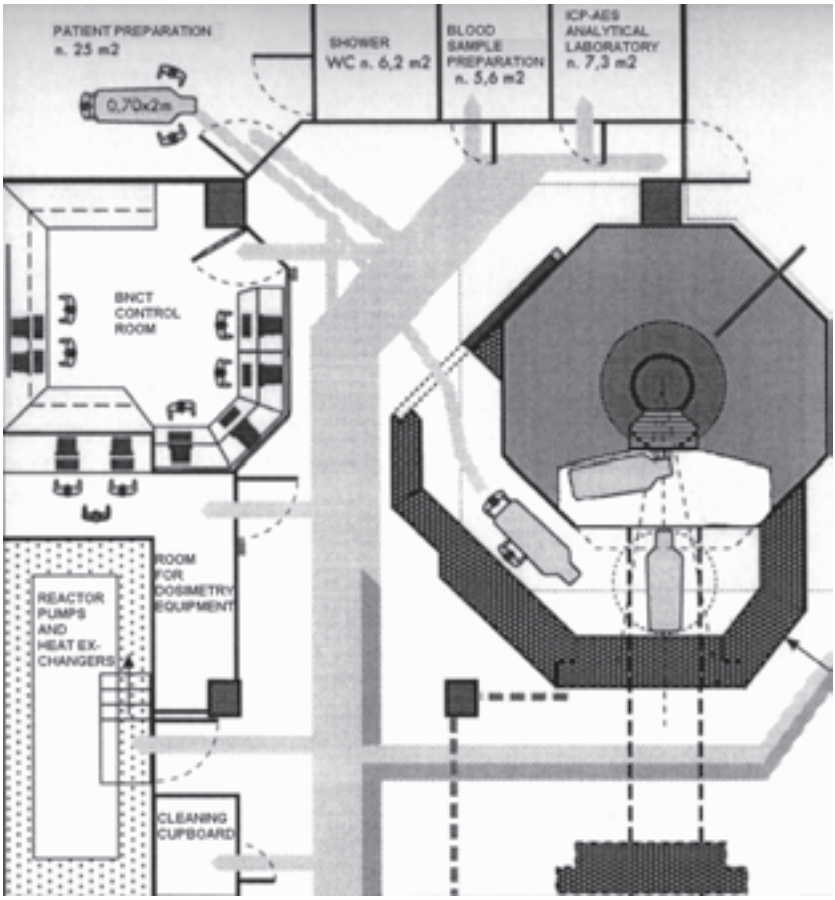

FIG. 4. Layout of the ground floor of the BNCT facility at the FiR 1.

would provide enough space for patient positioning and for personnel to work comfortably. The installation of monitoring cameras, positioning lasers and lighting was also evaluated with that model. This work was performed employing the same industrial designer as project consultant.

3.2. Renovation of the reactor building

The reactor building underwent a major renovation during 1996–1997. The ground floor of the reactor hall was provided with a new entrance, easily accessible by any patient vehicle, with a radiotherapy control room, and rooms for patient preparation and laboratories (Fig. 4). The top of the reactor tank was separated from the reactor hall in order to confine contamination in the case of leakage from irradiation samples or fuel elements (Fig. 5).

FIG. 5. The new containment room built over the open top of the reactor pool.

3.3. Epithermal neutron beam

The original thermal column space was a natural choice for constructing the epithermal neutron beam at the FiR 1. The column hole was 1200 mm in both height and width, i.e. wide enough to incorporate the beam components, and extending very close to the reactor core cutting partly into the graphite reflector. The graphite from the thermal column was removed and a support structure for a moderator container was installed in mid-1995.

Placing the patented Fluental neutron moderator (see Refs [17, 19]) into the thermal column cavity, starting from the very bottom (Fig. 6), produces the epithermal neutron field. Fluental is composed of 69% aluminium fluoride, 30% aluminium and 1% lithium fluoride. The thermal neutron irradiation load from the graphite reflector of the reactor is removed applying a boral plate. By placing the wide moderator close to the reactor core, more neutrons are sucked into it. These neutrons have a wide energy spectrum that is compacted to the epithermal range by collisions and capture reactions within the moderator. Neutrons pass through the bismuth shield, which efficiently attenuates the gammas originating from the reactor core and from neutron activated structures.

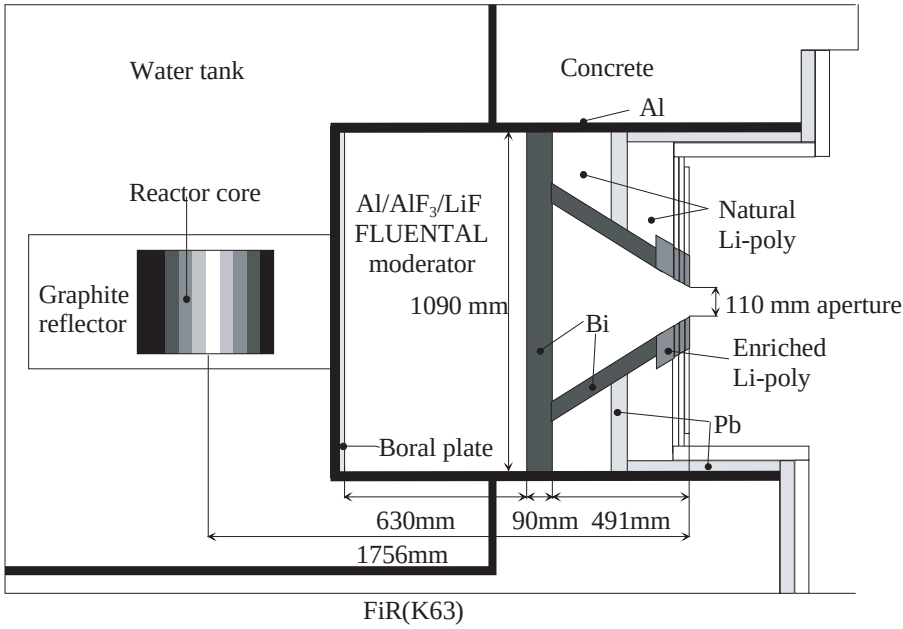

FIG. 6. The epithermal neutron irradiation facility at FiR 1 in its current configuration with a 630 mm thick Fluental moderator. For patient treatments, beams of 140 and 110 mm diameter are normally used.

The moderator was optimized for providing a high epithermal flux combined with a low fast neutron dose, both at the body surface, where it has the highest value, and deeper inside the tissue, at the peak of thermal neutron induced doses. A design goal of 2.6×10^{-13} Gy/epithermal n/cm^2 was set for the fast neutron dose per unit epithermal flux in the free beam, as proposed by Wheeler et al. [13]. An epithermal neutron flux of 10^9 n·cm^{-2}·s^{-1} was considered sufficient for convenient treatment times with current boron carrier compounds. The thermal (< 0.5 eV), epithermal (0.5–10 keV) and fast neutron (>10 keV) fluxes measured at the exit plane using a 140 mm diameter collimator at 250 kW power, are 8.1×10^7 n·cm^{-2}·s^{-1}, 1.1×10^9 n·cm^{-2}·s^{-1} and 3.4×10^7 n·cm^{-2}·s^{-1}, respectively. The corresponding neutron spectrum is shown in Fig. 7. The undesired fast neutron dose is 2×10^{-13} Gy/epithermal n/cm^2 and the corresponding gamma contamination 0.5×10^{-13} Gy/epithermal n/cm^2 [5, 9, 12].

A conical beam collimator enables the use of the epithermal field for the irradiation of humans. A bismuth inner cone with optimized thickness reflects neutrons back to the beam. Outside, lithium polyethylene limits the field by attenuating neutrons of all energies. Around the exit aperture, ^{6}Li enriched

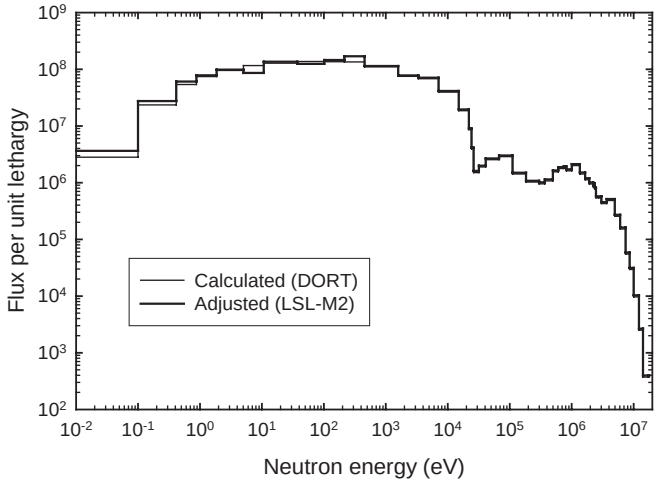

FIG. 7. Calculated and adjusted neutron spectra in the 140 mm aperture plane in the FiR 1 epithermal neutron beam [10].

polyethylene is used to minimize the creation of thermal neutrons and to capture gammas close to the target. As a compromise between flexible patient positioning and a favourable field distribution at the target, a cone with a 60° solid angle was found to be optimal [20]. The diameter of the beam exit aperture can be selected from 200 mm down to 80 mm in steps of 30 mm by the number of aperture collars in place.

The neutron flux decreases rapidly with distance from the beam port. Therefore, the target area, the patient's head, is required to be kept in contact with the beam port, causing an additional constraint to patient positioning. Several cubic metres of the concrete biological shield had to be cut away (Fig. 8) in order to allow positioning of the patient close enough to the reactor core with proper body orientation. The limiting factor in the concrete removal was the stability of the original biological shield. Structural strength analysis performed by O. Majamäki (IVO Power Engineering Ltd) showed that the thermal column could be widened only to the point that a suspending arch is still formed over the irradiation position (Fig. 9). Therefore, the cutting was limited to halfway along the side faces.

3.4. Epithermal neutron beam monitoring

An on-line neutron beam and reactor monitoring system has been developed and installed as an integral part of the facility, and it is used to control precisely the neutron delivery for the treatment [23]. At the bismuth gamma shield downstream from the moderator (Fig. 6), the epithermal neutron

FIG. 8. On both sides of the thermal column, a part of the biological shield was removed using diamond drilling and sawing.

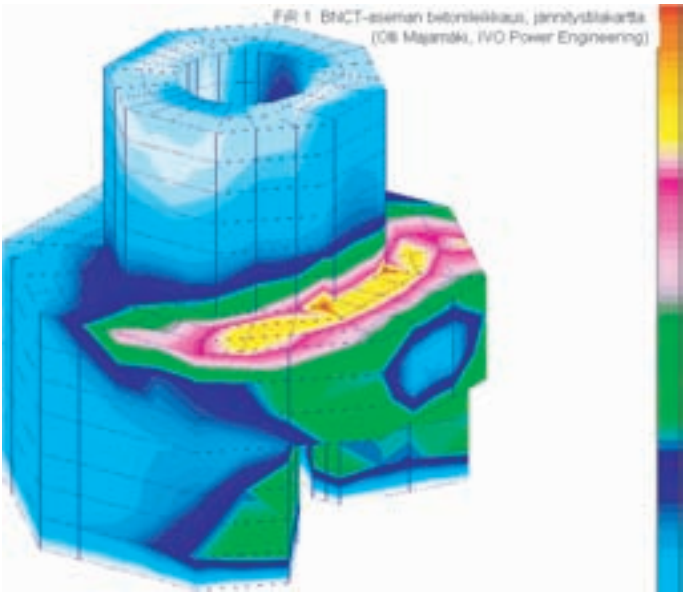

FIG. 9. Stress pattern of the biological shield after cutting for the BNCT facility.

beam is monitored. The detectors are three pulse mode ^{235}U fission chambers for the epithermal neutron flux and one current mode ionization chamber for the gamma dose rate. The detector signals are monitored on-line with a computer based instrumentation package (LabView, USA), which records and displays the actual counting rates and the total counts of the detectors in the beam. In addition, the reactor in-core power instrumentation and control rod positions are monitored via another LabView instrumentation monitoring system. The main purpose of the monitoring system is to provide a dosimetric link to the dose to the patient during treatment, as fission chamber counting rates have been calibrated to the induced Au and Mn reaction rates and to the absorbed dose rate at reference conditions in a tissue substitute phantom [20].

3.5. Shielding around the epithermal beam

The shielding around the beam aperture has two functions. Firstly, it limits the dose in the areas outside the target volume. Here the criterion is that the dose to the patient's body from the reactor core and the activated structures through the shielding should be clearly lower than the dose from the radiation induced at the target area. Secondly, the shielding limits the decay radiation dose level for personnel working inside the irradiation room after shutdown of the reactor; for example, when repositioning the patient for the second or third irradiation treatment.

To attenuate the gamma radiation streaming, especially through the concrete around the epithermal beam assembly, a 50 mm thick lead layer covers the concrete on all sides around the column hole. This lead layer continues also into the beam assembly up to the bismuth cone (Fig. 6). For neutron shielding, both commercially available RX215 (Reactor Experiments, USA) lithiated polyethylene and a composite made from lithium carbonate, polyethylene powder and paraffin wax originating from the facility's own production have been used. In that in-house produced shield, 30% lithium carbonate concentration by volume was used for fast neutron attenuation and 50–60% for the in-room scattered field. Lithium was selected as the neutron absorbing isotope throughout the beam collimator and the irradiation room to enable background-free boron prompt gamma monitoring in the future. The complications created by tritium production from the lithium neutron capture have been manageable, and no tritium contamination of personnel has been detected.

To reach more efficient patient positioning, shoulder recesses were later constructed around the aperture so that the patient's shoulders are fitted to the recesses when positioning is performed to a lateral field and the patient is in a supine position [21]. The recesses were designed so that the sideways bend in

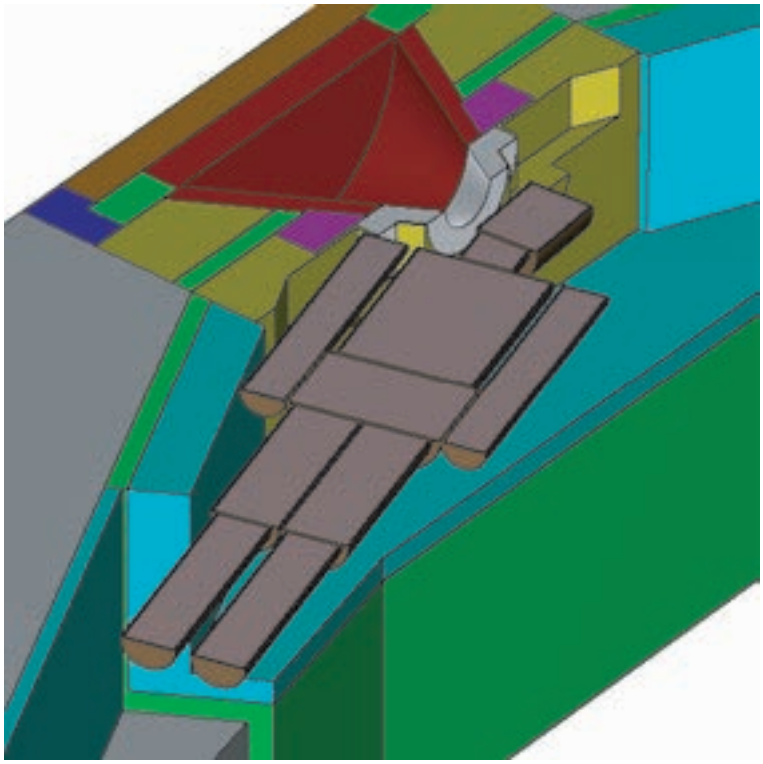

FIG. 10. The MCNP model of the anthropomorphic BOMAB phantom and the FiR 1 epithermal neutron beam collimator with shoulder recesses on both sides of the beam aperture.

the neck would not exceed 15°, which was considered to be the limiting angle for static fixation. In the original design of the BNCT facility at the FiR 1, the whole body dose of the patient was estimated with a DORT[2] model to be 0.6 Sv/h [22]. With the MCNP[3] model, the dose in a common 55° angle position was estimated to be (without the recesses) about 0.4 Sv/h, and in the worst case (with the arm inside the shoulder recess), the effective dose increased to 0.7 Sv/h (Fig. 10). According to the performed measurements, the MCNP model underestimates the physical neutron dose in air by about 40%, but the

[2] DORT: Two-dimensional discrete ordinates transport code, Oak Ridge National Laboratory, USA.

[3] MCNP: A general Monte Carlo n-particle transport code, Los Alamos National Laboratory, USA.

same model has given good results in dosimetric phantoms. These effective dose estimates are, therefore, indicative only.

If a person were to enter the irradiation room immediately after shutdown of the reactor and was exposed to the direct beam for 1 min, the subsequent dose would remain below 5 mSv. In practice, the person would never be exposed to the direct beam at the aperture but at the sides and away from the beam port, so the dose would be much lower. It has been measured that the patient could even be removed from the beam when the reactor is running without exceeding a dose of 5 mSv.

Normal practice is to wait until the radiation level in the main parts of the irradiation room is around 30 μSv/h before entering the room. This takes only a couple of minutes. The dose level close to the beam aperture and in the beam remains higher but these places can be avoided when removing the patient from the irradiation room.

3.6. Irradiation room

A heavy concrete shielded therapy room was built around the irradiation position (Fig. 1) [22]. Criteria for the construction were an easy and clean assembly at the reactor, and the possibility of removing part of the ceiling for maintenance work at the beam, e.g. when changing the moderator thickness or the bismuth collimator. No load was allowed to be put on the biological shield of the reactor so large beam lengths were required. In addition, optimized shielding for both gammas and neutrons was sought. Therefore, heavy concrete was selected as the basic material, and by casting the concrete into steel tubes, good structural strength was combined with high density (4.35 g/cm^3). The achieved high density ensured that with the 800 mm wall and roof thickness, the dose rate levels outside are well within the design objectives and actually lower than outside the original biological shield. The neutron field in the irradiation room is depressed by a factor of 200 by lining the inner walls and the ceiling with 30 mm thick lithium–plastic elements, which were manufactured in-house. In the direct beam shine area, there are three layers of the 30% elements; elsewhere, a single layer of the 50–60% elements is used. The floor is made of steel plate box elements filled with the same lithium–plastic mixture.

The irradiation room has two doors. The main entrance is at the end of the corridor. The hanging sliding door has 100 mm of lithiated plastic on the inner surface, followed by 10 mm of steel, 100 mm of lead and 10 mm of steel. A service entrance was formed from the original thermal column's 1 m thick heavy concrete door which moves on the original tracks. The main entrance is the weakest point of the irradiation room in regard to neutron shielding;

FIG. 11. A patient in the irradiation position supported by vacuum cushions on the irradiation bed. Shoulder recesses here are only partly employed and partly filled with lithiated plastic to reduce the dose to the patient's body.

however, the shielding is sufficient due to the depression of the neutron field inside the room by the lithium–plastic lining.

The irradiation room has both incoming and outgoing ventilation ducts. The ventilation is balanced so that the room is maintained at slightly lower pressure than the reactor hall. Due to the high fire load from the lithiated plastic shielding, fire protection is provided by an automatic HI-FOG (Marioff, Finland) water mist system. This system is highly efficient and safe to the patient and personnel.

General lighting is by fluorescent lamps. Close to the beam aperture, where the ceiling is less than 500 mm away from the patient, light guides are used to create an even illumination without taking up any space and to bring additional light through four adjustable light guides to any place around the beam aperture, see Fig. 11. The illumination level can be selected with a preprogrammed controller.

The design for the patient support system (irradiation bed, Fig. 11) was altered so that the whole system is used both in the simulation and in the actual

irradiation, and so that it is moved on wheels. The original design involved the use of a trolley on rails in the irradiation room. There were several reasons for this. One was the wish to minimize the radiation doses to personnel by minimizing the time personnel had to stay inside the irradiation room for positioning, and by creating a remote controlled system for emergency evacuation of the patient from the irradiation. In addition, it was considered to be important not to bring a possibly activated irradiation bed into the preparation room. Later, after experience in the research phase, it was realized that activation of the patient bed was not a problem and that doses to personnel manoeuvering the patient bed in the irradiation room would be tolerable in all circumstances. One consequence of this rather late change in the design concept was that the floor in the irradiation room had to be reconstructed to be absolutely horizontal.

3.7. Supervision of the BNCT facility

The BNCT facility is supervised from a dedicated control room (see Fig. 4). There is enough space for the patient monitoring equipment and personnel, as well as for the beam monitoring equipment and the BNCT operator using it. The BNCT operator has a permanent telephone line to the reactor operator, to initiate reactor startup. Also, the reactor scram is initiated from the BNCT control room. This is a new feature in the reactor operation scheme. Previously, the reactor could only be scrammed by the reactor operator or with one of the reactor safety switches.

The irradiation control team, including the radiation oncologist and the physicists responsible for treatment planning and boron pharmacokinetics modelling, meets in the control room.

The door to the irradiation room with its indication lights can be observed through the wide windows of the BNCT control room. At the door there are indication lights from the door microswitches, from movement sensors and radiation level monitors in the irradiation room, as well as from the reactor power status. An acoustic alarm and warning lights will indicate if someone is walking inside or into the irradiation room, or when the radiation level exceeds a set level. The same safety features as required at traditional high energy linac treatment facilities are applied here. In discussions with the regulatory authorities, it was decided that none of the alarms will scram the reactor. Such an arrangement was not considered necessary for the safety of personnel but would have increased the risk of unnecessary scrams causing harmful interruptions in the irradiations.

3.8. Patient preparation room and boron analysis laboratory

Adjacent to the BNCT control room, there is the patient preparation room with a separate toilet and shower. In the preparation room, the boron compound is infused. The patient is fixed in the planned orientation on the irradiation bed using laser alignment and a beam aperture simulator. The blood samples taken from the patient are prepared and analysed in the boron analysis laboratory with one room for the sample preparation and the other for the ICP-AES[4] machine. Currently, a direct doorway connects the two rooms.

4. LICENSING

In May 1999, the STUK licensed the BNCT facility at the FiR 1 to perform BNCT radiotherapy complying with experimental protocols accepted by the ethical committees. The licence also required an inspection and approval by the municipal health care authorities, as well as an approval by the regional governmental medical authority. The facility met all the requirements as designed, without any need for improvements.

The operating licence of the FiR 1 was renewed at the beginning of 2000. The reactor is now explicitly licensed to be used as part of a BNCT treatment facility. The reactor personnel now have the right and responsibility to weigh the benefits of completing a patient irradiation, for example, by continuing to run the reactor even in an emergency situation, such as a fuel element leakage, against the risks resulting from radiation and for the safety of the reactor.

The operating licence is edited by the VTT, an independent government research organization under the Ministry for Trade and Industry. The management organization of the reactor can be seen in Fig. 12. A special management position, the BNCT manager, was set up for managing the BNCT facility. In addition to the reactor manager and the BNCT manager, currently there are two additional licensed supervisors and five licensed operators. They all have these positions supplementary to their main duties at VTT.

The radiotherapy licence is held by the Boneca Corporation [8], a firm owned by the Clinical Research Institute at the Helsinki University Central Hospital, VTT and Sitra, the Finnish National Fund for Research and Development. The management organization for the radiotherapy licence is shown in Fig. 12. As can be seen, the two organizations overlap and they are partly included in each other's licensing documents. VTT is responsible for the

[4] ICP-AES: Inductively coupled plasma–atomic emission spectrometry.

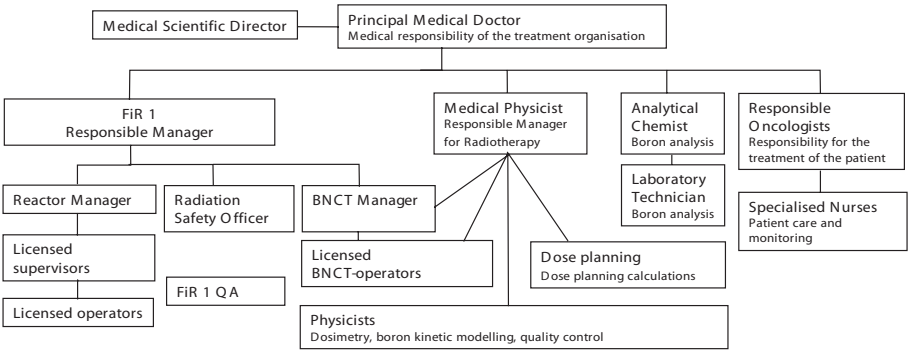

FIG. 12. The management organization for the radiotherapy licence.

maintenance, operation and safety of the reactor and for radiation safety of the BNCT irradiation facility. The Boneca Corporation is responsible for patient irradiations, especially for the dose that the patients receive, i.e. dosimetry, treatment planning, boron determination and patient positioning in the neutron field.

5. CLINICAL TRIALS

Three clinical trials are currently running at the FiR 1 BNCT facility [7, 8]. Since May 1999, over 20 patients with glioblastoma, until now an incurable brain tumour, have been treated with BNCT using boronophenylalanine (BPA), see Ref. [24], as the boron carrier compound within the context of a prospective clinical trial (protocol P-01). BPA fructose is given at the BNCT facility as a two hour infusion prior to neutron beam irradiation. Blood samples are analysed for blood boron concentration using ICP-AES in a dedicated analytical laboratory at the reactor [25]. The irradiation is given in one or two fractions from two fields. The irradiation procedure typically lasts for about one hour. In another trial (protocol P-03), some patients with recurring or progressive glioblastoma following surgery and conventional cranial radiotherapy have been treated with BPA based BNCT. Also recently, a trial has been started for adult patients with histologically confirmed recurrent inoperable head and neck carcinoma after standard external beam radiotherapy (HN-BPA-01-2003 trial).

The conclusion is that BPA based BNCT has been relatively well tolerated both in previously irradiated and unirradiated glioblastoma patients. Efficacy comparisons with conventional photon radiation are difficult to perform due to patient selection and confounding factors, such as other

treatments given, but the general results support the continuation of clinical research on BPA based BNCT [7].

6. COST OF OPERATION AND AVAILABILITY OF THE FiR 1 FOR BNCT

The expenditure of designing and accomplishing the construction work of the BNCT facility at the FiR 1 was about €4 million. Half of that budget was applied to the renovation of the reactor hall and the other half has been spent on research, development and construction of the epithermal beam. The epithermal irradiation facility was constructed by VTT under contract with Radtek Inc., Espoo, Finland. Radtek was a company formed to combine private capital and State technology development funding (TEKES — Technology Development Center Finland, Sitra) for this purpose. Later, the ownership of the beam facility was transferred to VTT.

The basic cost for maintenance and operation of the reactor is about €400 000 per year, including licensing and administration. The operating costs of the reactor are moderate, as one operation shift includes only the reactor operator and the shift supervisor. As they are not fully occupied by the reactor operation, they are allowed to perform other duties during the reactor shift. With an increasing number of patients, more reactor operators have to be involved, causing a stepwise increase of operation costs. The radiation protection has one duty officer. If there is the demand, the reactor can be made available for BNCT treatments from early morning until late evening, allowing the irradiation of several patients a day. For the moment, Mondays and Fridays are reserved for activation analysis and isotope production; from Tuesday to Thursday the reactor is reserved for BNCT.

At VTT, the reactor is considered to be a self-supportive service unit without financial support for operation from VTT basic funding or other government sources. The funding is based completely on sales and contracts with customers.

7. CONCLUSIONS

Close to 30 patients have been treated at the FiR 1 since May 1999, when the licence for patient treatment was granted to the responsible BNCT treatment organization, Boneca Corporation. VTT as the reactor operator has a long term contract with the Boneca Corporation to provide the facility and irradiation services for patient treatments. The BNCT facility has been licensed

for clinical use and is being surveyed by several national public health authorities, including the STUK. The treatments are given in collaboration with the Helsinki University Central Hospital, located only about 6 km from the reactor.

The FiR 1 also has an important national role in providing local enterprises and research institutions in the fields of industrial measurements, pharmaceuticals, electronics, etc., with isotope production and activation analysis services.

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5.6. DUAL BNCT BEAM DEVELOPMENT AT THE RA-6 REACTOR FACILITY

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1. INTRODUCTION

Boron neutron capture therapy (BNCT) was first proposed almost six decades ago and first performed four decades ago [1], and it has been exhaustively discussed elsewhere [2, 3]. In recent years, it has been the subject of renewed worldwide research activity.

BNCT is a binary technique that involves the concurrent presence of a flux of neutrons and a ^{10}B capture agent. Their interaction generates heavy particles (^{4}He and ^{7}Li) whose path lengths are smaller than the typical diameter of a tumour cell (10 μm). As a few alpha particles suffice to destroy a tumour cell, the destructive effect of the capture process would mainly occur in those cancer cells that have selectively accumulated boron. Nowadays, BNCT facilities have spread over the world, and clinical trials are being performed at several places. Not only are epithermal beams being used for deep-seated tumours, thermal beams were also applied for the treatment of superficial tumours, such as skin melanomas [4, 5].

In an effort to evaluate the applicability of using a nuclear research reactor for cancer treatment, a BNCT facility has been developed at the RA-6 reactor. The BNCT facility was built a few years ago, and it was first developed as an epithermal beam [6, 7]. Recently, the original epithermal beam has been thermalized, resulting in a mixed beam named 'hypothermal' or 'mixed beam' of mostly lower energy neutrons, but with an energy distribution that is harder than a pure thermal one, in such a way that the previous epithermal beam configuration could be easily recovered. To this extent, the final result is a dual beam: a switchable 'mixed/epithermal' beam. The referenced mixed beam has already been used for in vivo experiments with small animals [8, 9]. The mixed beam characteristics could potentially represent a benefit for skin melanoma treatment (extending up to a few centimetres depth in the tissue) due to better penetration [10]. Its performance for this application has been exhaustively evaluated [11] and the first human clinical trial has been performed recently [12]. The subsequent presentation includes the beam description, as well as its characterization and modelling features.

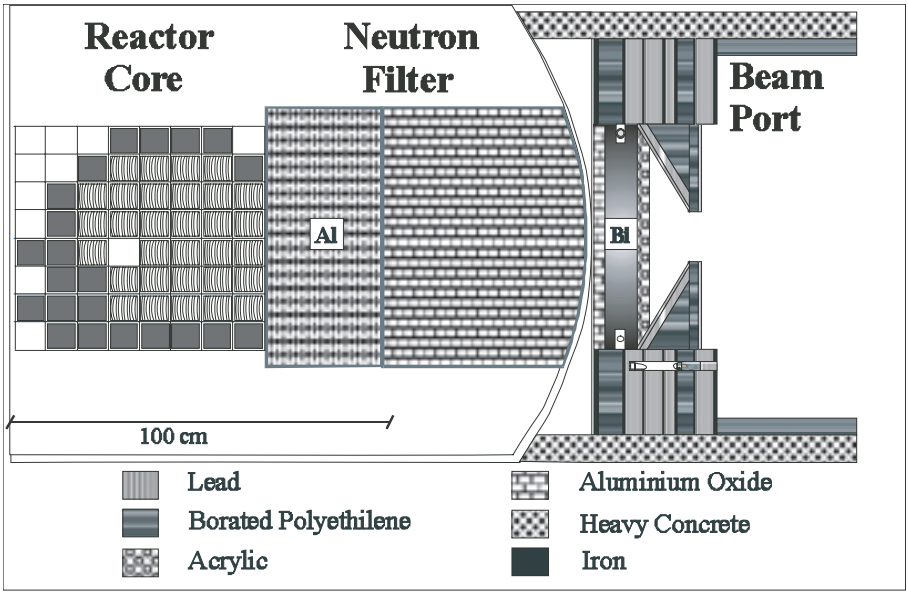

FIG. 1. Scaled, schematic representation of the BNCT facility showing the main components of the beam port.

2. MATERIALS AND METHODS

RA-6 is a pool type reactor, with a nominal power of 500 kW using high enriched fuel (90% ^{235}U). By removing its internal and external thermal column, an epithermal beam was developed a few years ago [6, 7]. In 2001, the original epithermal beam was modified to achieve a neutron beam with more appropriate characteristics for the treatment of surface tumours.

The following were considered as boundary requirements at this stage:

— Use of the same internal filter as the previously developed epithermal beam;
— A thermal neutron flux peak with about 1.0×10^9 n·cm^{-2}·s^{-1} at 10 mm phantom depth;
— Easily removable modifications to restore the epithermal beam.

By simulating the beam configuration and its performance in a water phantom, it was found that a hyperthermal beam could have the best results. Several materials, such as graphite, heavy water and acrylic, with different thicknesses were considered for thermalization of the original epithermal

beam, and to finally obtain the required neutron spectrum and beam intensity. A selection was made for an acrylic cylindrical block of 620 mm in diameter and 30 mm thickness, arranged in several pieces — as in a puzzle — so that it would be easy to remove [10].

The beam port configuration of the BNCT facility is shown in Fig. 1. The array of the selected materials provides the desired beam tuning, the collimation, as well as the shielding outside the beam. By rejecting the acrylic plate, the epithermal configuration is recovered. In order to determine the characteristics of the neutron beam either for the reference physical dosimetry or for treatment planning applications, a series of coupled computations and measurements were performed, as part of the beam set-up process, using the Monte Carlo MCNP4C radiation transport code [13] and standard phantoms and measurement methods.

2.1. Beam modelling and assessment

A detailed neutron and photon source at the core level was obtained through a coupled neutron gamma KCODE[1] calculation with the MCNP4C code.

As a result of this calculation, a surface source at the beam port was obtained representing, in a simple and efficient manner, the RA-6 NCT beam so that it can be used in the treatment planning code. This was done in a sequence of several stages:

— By using the MCNP4C transport code, along with a complete and detailed representation of the NCT facility, a track by track source was obtained at the external surface of the port;
— A simple set of software was developed to differentiate energy, direction and space in order to attain average values for neutron and photon intensity in the selected intervals;
— Two spatial, angular and energy coupled distributions were defined to represent the neutrons and the photons originating from the beam.

A source normalization process (two independent normalization factors for the fast neutron source and for the gamma source) was performed by comparing calculations and measurements in a cubic water phantom. The defined distribution sources were modified in order to achieve a better agreement between measured and calculated flux and dose distributions.

[1] KCODE: the fission source option in MCNP calculations used for reactor core modelling.

2.2. Beam physical dosimetry

Dose and neutron flux measurements were performed in 12 positions along the central beam axis, each separated by 5 mm, in a water filled acrylic shell of 10 mm wall thickness, 400 mm height, 400 mm width and 200 mm depth as the phantom. The phantom thickness had been reduced to 5 mm in a centred circle of 300 mm diameter on its front wall, i.e. the wall in contact with the beam port surface.

Neutron flux measurements were performed by foil activation, using the well known cadmium difference method, and the paired ionization chambers technique. Bare and Cd covered gold wires were activated in an acrylic holder along the central axis of the water phantom. Their activities were determined by using an HPGe gamma spectrometer. The thermal (0–0.5 eV) neutron cross-section of Au and the cadmium factor[2] were obtained through detailed neutron transport calculations by the MCNP4C transport code.

The doses were determined from ionization chamber responses, by the paired ionization chambers technique following the approach given by Rogus et al. [14].

The ionization chambers used were thimble shaped, with 10 mm^3 active volume, manufactured by Far West Technology. The chamber with the highest neutron sensitivity was a tissue equivalent chamber, the IC-18 model, flushed with methane based tissue equivalent gas. The other chamber was a graphite type, the IC-18G model, flushed with carbon dioxide. Both chambers were fed with a 250 VDC battery array. Details about the determination of the sensitivity parameters of both ionization chambers have already been reported in Ref. [15].

The chambers were placed in a waterproof housing and positioned in such a way that the chamber axis was aligned with the central beam axis. The chamber response was measured with a Keithley electrometer, Model 6517A. Readings, corrected for pressure and temperature, were recorded and processed.

Beam dosimetry results have been confirmed through a worldwide project of dosimetry intercomparison performed throughout the different stages of beam development [16–19].

[2] The cadmium factor is a correction factor used in the cadmium difference method in order to determine the foil thermal activation from the difference between the bare foil activity and the cadmium covered foil activity.

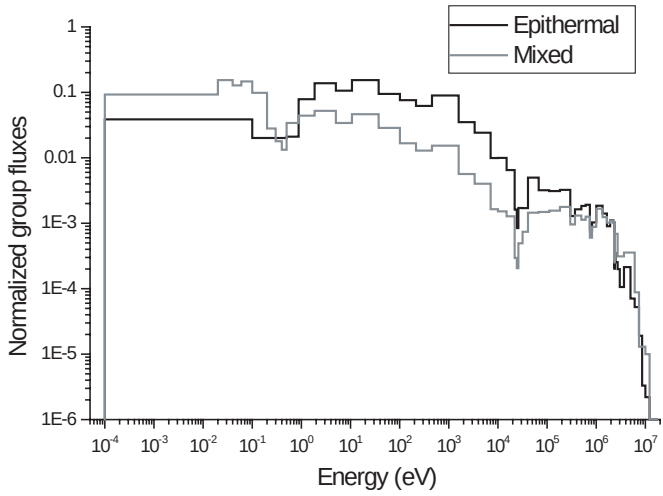

FIG. 2. Source neutron spectrum at the beam centre for epithermal and mixed beams.

3. RESULTS

The neutron spectra at the beam centre are shown in Fig. 2 for the epithermal and the mixed beams. The group fluxes are normalized with the total flux for each spectrum. The thermal neutron flux and the dose rate profiles, measured along the central beam axis, are shown in Figs 3 and 4 for the epithermal beam and in Figs 5 and 6 for the mixed beam.

Uncertainties in the thermal flux measured values were nearly constant at 7%, mainly due to detector cross-section and other statistical contributions. The present data show a shallow thermal flux maximum, present at the first centimetre following the phantom external wall.

This result is consistent with the hypothermal spectral energy characteristics of the beam.

As the paired ionization chambers technique is used in dose rate profile measurements, the intrinsic geometry of the ionization chambers enables the possibility for more superficial positions. Furthermore, the loss of the charged particle equilibrium condition near to the phantom surface would render the dose rate profile measurements (fast neutrons and photons) useless.

The relative uncertainties reported for the gamma dose rate are about 5%. The measured profile results from the incoming gamma field (beam contamination) superimposed to an inner source, associated to thermal neutron capture in hydrogenous media.

The uncertainties derived for the fast neutron dose rate (Figs 3 and 5) are about 25% in the first points. This is fully compatible with similar

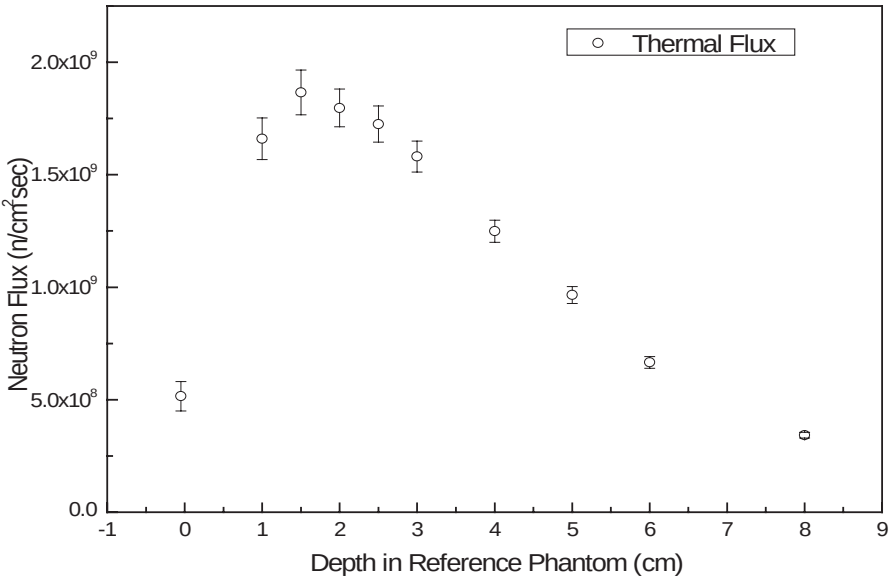

FIG. 3. Physical dosimetry along the central axis of the RA-6 epithermal beam: thermal neutron flux versus depth in phantom.

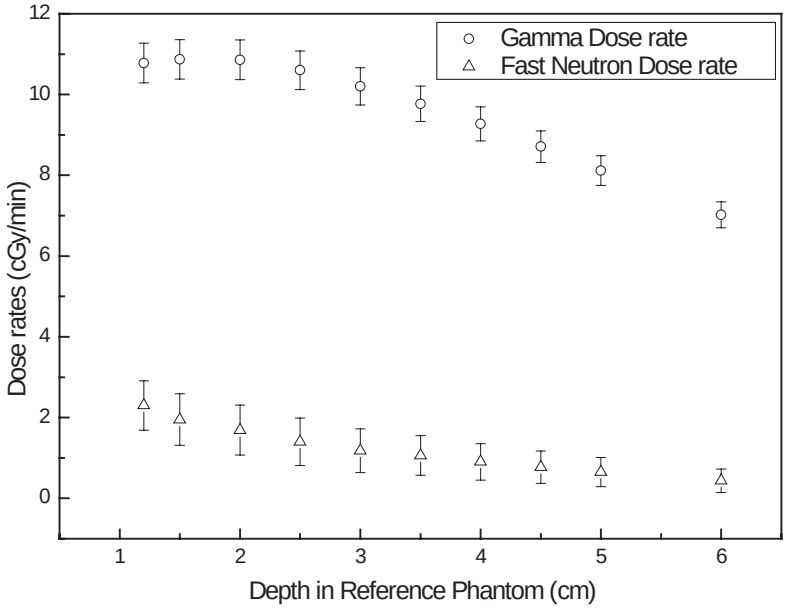

FIG. 4. Physical dosimetry along the central axis of the RA-6 epithermal beam: fast neutron and total gamma absorbed dose rate profiles specified for muscle tissue versus depth in the phantom.

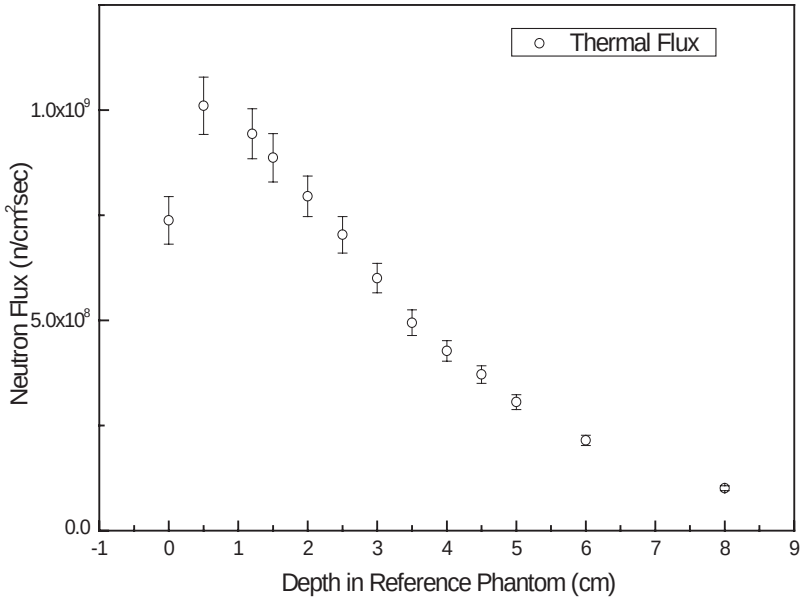

FIG. 5. Physical dosimetry along the central beam axis of the RA-6 mixed beam: thermal neutron flux versus depth in the phantom.

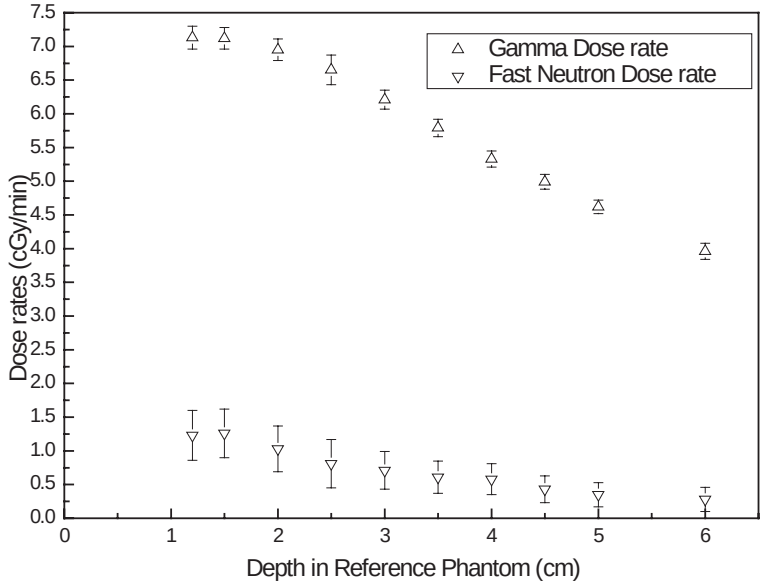

FIG. 6. Physical dosimetry along the central axis of the RA-6 mixed beam: fast neutron and total gamma absorbed dose rate profiles specified for muscle tissue versus depth in the phantom.

measurements reported in Ref. [19]. The fast neutron dose rate profile shows the usual exponential decay behaviour in relation to the moderation of the fast neutrons in the media.

A beam spectra analysis (see Fig. 2) shows the following differences between both beams' energy distributions: the fractions of the thermal flux ($E_n < 0.5$ eV), the epithermal flux (0.5 eV < E_n < 10 keV) and the fast flux ($E_n > 10$ keV) are 0.06, 0.89 and 0.05 for the epithermal beam, while being 0.68, 0.30 and 0.02 for the mixed beam.

At the phantom location, measurements show the following differences between the performances of both beams: in the case of the mixed beam, the thermal flux peak appears at the outer position, due to the softer energy spectrum of that beam compared with the epithermal beam. The fast neutron dose rate is higher in the epithermal beam. The photon dose rate is higher in the epithermal beam, due to the higher intensity of the thermal peak, and the peak appears at an inner position due to its correlation with the location of the thermal peak.

Experimental dosimetry, together with the surface source obtained from beam modelling, was integrated into the treatment planning system by the determination of normalization factors. These factors are linear adjustments to the computed profiles that provide a better absolute agreement with physical measurements.

4. CONCLUSIONS

A dual epithermal/mixed beam is available for BNCT applications at the RA-6 reactor facility. This switchable beam allows the selection of the best configuration in view of the tumour depth, optimizing the performance of the treatment.

The mixed beam performance for skin melanomas has been exhaustively evaluated and the first human clinical trial has already been performed.

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Part 6

BEAM TUBES

6.1. ISSUES TO BE CONSIDERED WHEN DESIGNING BEAM TUBES

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Germany

1. INTRODUCTION TO BEAM TUBES

At the end of the 1940s, research reactors were already being designed with beam tubes. Most of the research reactors constructed in the 1950s, especially during the late 1950s, had beam tubes. Early examples of such reactors were the low intensity test reactor (LITR) at ORNL, Oak Ridge, and the MTR at Idaho Falls, both in the USA, which stands for a class of similar reactors that were subsequently built.

Research reactors without such tubes were also built in the 1950s, the most well known of which is the Triga Mark I series (from 1957) which, however, was followed by the Mark II series with beam tubes. Thus, beam tubes and their features are well known to the research reactor community.

As beam tubes are the interface between the reactor core, as the neutron source, and the scientific instruments, they are still frequently discussed when research reactors are upgraded or newly designed. The reasons for this are discussed in the following sections, where several issues are touched on.

2. ORIENTATION

In the early designs, nearly all of the beam tubes pointed directly to the core; they were — under the nomenclature of today — radial beam tubes. Later, 'through' tubes (passing through the core) and tangential beam tubes (touching the core border but avoiding any direct view to the core zone) were added. In today's designs, tangential beam tubes dominate over radial ones, since suppressing the background radiation originating directly from the core volume is important for the quality of the neutron beams.

Although the wide majority of beam tubes providing neutrons to scientific instruments are horizontal tubes, there are several inclined and — more rarely — even vertical arrangements (e.g. at RHF, Grenoble, France, and WWR-M, Gatchina, Russian Federation). The vertical tubes use deceleration by gravity to produce very cold neutrons. Inclined beams (e.g. at PIK at Gatchina, Russian Federation) allow for instruments in an experimental hall at elevated levels.

3. STARTING POINT

Early research reactor designs had beam tubes starting somewhere near the core. There was no way to properly calculate the neutron flux exiting the tube independent of the position of the beam tube entrance. In addition, the core sizes were less fixed. With a changing number of fuel assemblies in the core, such optimization would have made no sense.

The research reactor designer now has to demonstrate that the entire system consisting of core, reflector and beam tubes is optimized regarding both selected materials and geometry in order to achieve the best possible performance. This task was also of special interest for the numerous upgrading projects of research reactors in the last two decades, in which an existing fixed arrangement of beam tubes had to be linked to a renewed core plus reflector system. Such optimizations require skill in calculation but also a clear view of the reactor operator and the related scientific community with respect to the preferences of some beam tubes against others out of the original arrangement.

An easily overlooked issue is the positioning of the beam tube in view of the active core height. Since most research reactors are operated with an axial flux distribution slightly peaked below the axial core centre (due to inserted control/regulation rod(s)), the optimum position of a beam tube also tends to be somewhat below the axial centre of the active core.

4. SIZE

There is a trend towards wider beam tubes. Operators of research reactors that have small diameter tubes from the original configuration complain of their disadvantages. This situation is of special importance when the insertion of a cold neutron source (CNS) (see Part 7) is envisaged and several neutron guides point to its moderator volume. A typical example of that disadvantage exists at ORNL's HFIR for its CNS. A means to overcome that issue was demonstrated at the BER-II at Berlin, Germany, where the existing thermal column was replaced by a wide and conical beam tube with a CNS.

Large sized beam tubes, such as those that exist at the RHF of the ILL, Grenoble, France, and the FRM II of the Technische Universität München, Munich, Germany, are, however, linked to substantially higher costs, as they need huge shutter systems of high precision at their exits, as well as a sophisticated arrangement of radiation shielding. However, they can be equipped with bundles of neutron guides, bundles that widen towards the experimental area, thus giving space for numerous instruments along their paths.

5. REFLECTOR MATERIAL

Although not a part of the beam tube itself, the reflector, which the beam tubes are embedded in, must be mentioned here. Early research reactors with beam tubes had light water surrounding the core and sometimes had reflector elements of graphite or beryllium or beryllium oxide between the active core border and the beam tube front end. Often the systems, comprising the core, the reflector and the beam tube, were not optimized.

Beam tubes running inside a graphite reflector (for example, at the Triga Mark II) were a distinct improvement. Today, designers realize the clear advantages of beryllium and heavy water as reflector materials and also understand the problems of positioning beam tubes in a solid beryllium reflector in the optimum way. This is true mainly for tangential tubes of larger diameters because inside beryllium, the thermal flux peak is situated rather near the active core border. This thermal flux peak moves even nearer the outer core border when water gaps are located in the path of the neutrons between the core border and the beam tube. This effect occurs because water slows neutrons down in lengths that are shorter than beryllium. These gaps cannot be totally avoided due to the need to cool the core outer surface and the beryllium. If, at the design stage, these effects are not considered carefully, especially for a tangential tube of large diameter, the centre line of the tube may be positioned outside the location of the thermal flux peak. The latter will lead to a less than optimum performance of the tubes. This effect is also true for the beryllium layers in front of radial tubes.

Examples of beryllium embedded beam tubes are found at ORNL's HFIR, at PNPI's WWR-M, and at the upgraded plants BER-II and FRG1 in Germany. Designers prefer heavy water as the reflector and its rather wide — and more distant to the active core — peak of the thermal neutrons enables an easier optimization of the positioning of the tubes around the core. Since a heavy water reflector is more costly than a reflector of beryllium, such a choice is not easily made, given financial considerations, only where there is a need for the highest possible performance. Consequently, high performance sources, such as CEA's ORPHEE, ILL's RHF, the new Munich Neutron Source FRM II and the Chinese CARR, use heavy water reflectors.

6. PERTURBATIONS BY OTHER USER FACILITIES

Around the core of research reactors, there is, in modern designs, competition for the preferred locations where most of the neutrons are available, not only from beam tubes but also from other installations, for

example, Si doping or radioisotope production or activation analysis. The latter is true even at plants where beam tubes, due to their importance, are located at optimum positions and clearly dominate the other installations near the core. In this case, the rather narrow high performance cores, the demand for a large number of typically large diameter tubes and the limited space around those cores, lead to beam tubes which interfere with other tubes located nearby and compete with them for the neutrons leaving the core volume.

All such perturbations have to be taken into account when optimizing the system consisting of the core, the reflector and the installations for utilization. The interaction of installations can lead to quite a different optimum arrangement than any single installation would suggest.

7. SAFETY ISSUES

As beam tubes are located at rather low levels inside the reactor pool, penetrating the pool liner and the biological shield of the pool, they create a vulnerable spot at the barrier provided by the pool wall. Any heavy load falling into the pool may damage them. Radiation induced corrosion of the Al alloys from which most beam tubes are manufactured may weaken their stability over the long term. Since neutrons will travel in a way that is as non-attenuated as possible from the entrance to the exit of each such tube, walls inside the tube are not readily accepted by experimentalists. The issue becomes more complex when neutron guides are extended into the tubes to start as close as possible to the neutron source.

There are many different barrier designs necessary and/or requested by the licensing authorities in different countries, in number as well as in design.

Arrangements of fixed separation walls/membranes, made mostly from Al alloys, and movable shutters (inserts from special concrete, plugs rotating around the tube axis or perpendicular to the tube axis, guillotine shutters, horizontal shielding plugs, etc.) inside and at the end of the tubes provide safety against neutron leakage, as well as necessary shielding. Sections of neutron guides embedded into the rotating or linear moving shutters complicate the adjustment of the heavy pieces of shielding even further. The solutions in this field are so numerous that it is worth calling upon existing experience when designing something new.

The ideal beam tube is empty, which means that it is evacuated, so as to prevent scattering of neutrons inside the beam tube. However, reactor designers dislike big vacuum volumes at low levels inside the pool, as there is always a non-zero probability of a sudden break of a barrier (for example, by a heavy load falling into the pool, or sudden pressure pulses on a structure

already brittle from radiation). There is the linked risk of a 'water hammer': a sudden intrusion of water of high velocity into the vacuum. By sheer momentum it could destroy the metal barrier in the beam tube at the outer end. That would lead to a significant leak in the reactor pool wall at a level below the upper end of the active core and, thus, to the possibility of a partially dry core which may subsequently melt. As that sequence is a worst safety case scenario, for a swimming pool type research reactor, the designer will seek to prevent a water hammer by filling the beam tube with helium at or above atmospheric pressure. Designing for a water pressure slightly above the inner gas pressure also reduces the problems of a helium tight tube system to a watertight system.

Although it is not principally a safety issue, it should be clear that, in any case, water inside a beam tube has to be avoided. Evaporated water will provide substantial neutron absorption and scattering and, thus, will reduce the beam intensity. In addition, water and radicals generated by radiation induced dissociation may contribute to corrosion effects of the tube.

8. NEUTRON GUIDES

Neutron guides have seen the fastest progress in the last two decades, leading them to be dealt with here comprehensively. Their importance at neutron sources with n-beams is significant. They fill and extend at least half of the neutron beams at a modern beam tube research reactor, and guide thermal and cold neutrons to the instruments in the experimental hall or in separately built neutron guide halls which enable the long distances that are necessary for special experimental set-ups and for utilization of multiple instruments at one beam.

It was discovered by chance at the FRM at Garching, Germany, that neutrons are reflected by metal walls. This has led to sophisticated arrangements of bent multilayer guides supplemented by devices focusing neutrons, which has resulted in significant improvements in upgrading the performance of the neutron beams at research reactors. Other measures, such as core compacting and optimizing of the reflector and positioning of the core, involve problems in enlarging the neutron flux at the instrument in the same manner that neutron guides can improve the performance of specific instruments, especially in combination with cold neutrons.

9. NEUTRON RADIOGRAPHY FACILITIES

Neutron radiography uses neutrons to X ray samples and enables — contrary to the transmission by gammas — the visualization, especially of light atoms, preferably hydrogen. At research reactors, such a non-destructive analysis method is applied in two different ways. One is underwater radiography, which allows the investigation of irradiated fuel pins and, thus, enables the follow-up of changes of fuel pellets at different burnups when interrupting an ongoing irradiation for intermediate radiography.

The more frequent application of n-radiography takes place outside the reactor pool and the biological shield, that is, in air. A suitably shaped and long (mostly conical) collimator provides the neutron beam at an area at the position of the object under investigation with a rather small angle deviation. The object is placed at the outer end of that collimator and the picture is taken right behind the object (by n-sensitive films or an array of position detectors).

Numerous early research reactors had an n-radiography facility installed, mostly horizontally — at Triga plants often vertically — for probe positions at the top of the pool. Plants with intensive use of n-radiography facilities in air are the MURR and the USD/MNRC in the USA, as well as at ORPHEE in France, and at the new FRM II in Germany, the latter enabling the imaging of movable parts in a probe due to its high performance. Frequently used at research reactors is n-radiography: it shows impressive results in various fields, such as quality control, ageing of devices and dynamics of lubrication.

Important parameters for a high quality n-radiography device are the length of the collimator tube and the short distance of the imaging device from the object. Both factors influence the resolution of the image. Placement of the starting point of the device near to the reflector peak of the neutron flux allows it to receive an intense beam permitting rapid imaging and the observation of dynamic processes.

10. MATERIALS

In order to select suitable materials, one has to distinguish between the inner beam tube part (the thimble) and outer parts, for example, those which penetrate the biological shield or are welded to the pool liner.

The thimble materials are determined by low neutron absorption as well as adequate experience with their long term behaviour under neutron and gamma bombardment (loss of ductility, embrittlement, etc.).

Nearly all existing beam tubes are manufactured from various Al alloys. Such alloys are specified according to the material grades (by international

standards) of 5052-0 (AlMg 2.5), 5754-0 (AlMg3) and 6061-T6 (AlMg1SiCu) or similar alloys, such as the French Ag3Net. Some designs try to avoid welding at the innermost part, others do not. It is obvious that beam tubes at high performance reactors, such as RHF at Grenoble, France, or HFIR at Oak Ridge, USA, need more careful consideration of radiation induced damages than do low power facilities, such as Triga Mark II or comparable plants in the kilowatt power region.

Whereas the low performance reactors can assume beam tubes which need no replacement due to radiation damage during the reactor lifetime, the high performance plants cannot assume this, and have to design for replacing the thimbles at the end of a predicted operation period which is based on irradiation experience with the material used. Therefore, flanging instead of welding of the thimble to the outer part is often applied in those cases.

Whereas the thimbles mostly rely on Al alloys, the downstream parts are often made from stainless steel. This means that somewhere downstream along the tube, an Al alloy has to be connected to stainless steel. Such transitions of materials require proper design as well as adequate experience.

6.2. RECENT BEAM UPGRADING DEVELOPMENTS AT THE HOR REACTOR

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1. INTRODUCTION

The HOR is a pool type research reactor at the Interfaculty Reactor Institute (IRI) of the Delft University of Technology, Netherlands, using MTR type fuel assemblies. It has been in operation since 1963, maintaining good performance by upgrading instruments and beam tube facilities, as well as keeping good maintenance records over the years for the reactor equipment and ancillary facilities. The reactor core set-up itself has been modified several times, the latest major change being the core conversion from HEU to LEU fuel with core compaction from 32 to 20 fuel assemblies. At the moment, the HOR is operated on a regular basis at a thermal power level of 2 MW, attaining a modest thermal flux of about 2×10^{13} n·cm^{-2}·s^{-1}. The reactor comprises a variety of irradiation facilities used, among others, for radioisotope production and neutron activation analysis. It is equipped with six horizontal radial beam tubes, originally used for neutron scattering experiments. Over the years, the research activities have grown steadily, both in the development of new techniques and in applying those techniques in new research areas [1]. In the last decade of the 20th century, utilization of the beam tubes was extended to include neutron activation analysis [2] and positron beams [3, 4]. In the framework of neutron beam research applications, one of the larger beam tubes was equipped with a new type of neutron guide system, to produce 'clean' neutron beams, i.e. with optimal thermal neutron intensity, combined with very low fast neutron and gamma contamination [5, 7]. Although the HOR is equipped with a number of advanced in-core irradiation facilities, the primary utilization now focuses on neutron beam experiments plus positron beam research applications.

2. GENERAL LAYOUT AND PROJECT BACKGROUND

Figure 1 shows the general arrangement of the core and beam tubes in the pool. The original design is of the conventional type, i.e. an integrated system

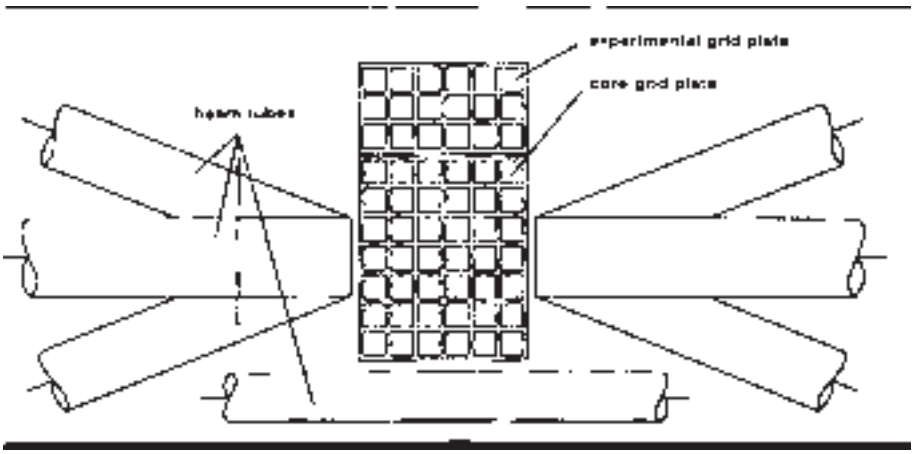

FIG. 1. Arrangement of a grid plate and beam tubes.

approach for maximum performance was not state of the art in the early days. The design is characterized by beam tubes facing the water reflected reactor core, which was originally movable and of variable size. There are six radial tubes and one tangential tube, all of horizontal type. The radial tube front ends are facing the core, three of them at the left hand side and three of them at the right hand side, in a straight line to the core. The radial beam tube front ends are approximately at the location of the unperturbed thermal flux peak. The horizontal beam tubes are axially positioned at the core centre line, not at the optimum axial flux peak slightly below the centre line. The two central radial beam tubes, with their central lines crossing the core centre at midplane, have rather wide outer diameters of 245 mm. The other radial tubes, each facing the upper or the lower half of the core at the left hand and the right hand side, have a smaller diameter of 195 mm. This conventional beam arrangement is typical for many research reactors of older design. Currently, the core is of the compact type, beryllium reflected on three sides. In the upgrading project, the existing arrangement of beam tubes had to be linked to the renewed compact core and reflector configuration. The beryllium block reflector option for embedding the beam tubes was not yet installed in this project.

The central beam tube R2, which is used to accommodate the neutron guide system presented here, faces the right hand side of the core with its axis perpendicular to the core surface. It is directly facing the three fuel assemblies in front of it (see Figs 1 and 9), with a small water layer of about 20 mm in between. The smaller beam tubes R1 and R3 are also facing that side of the core. They are positioned nearly symmetrically with respect to beam tube R2, looking at the core upper and lower half, just above and below tube R2,

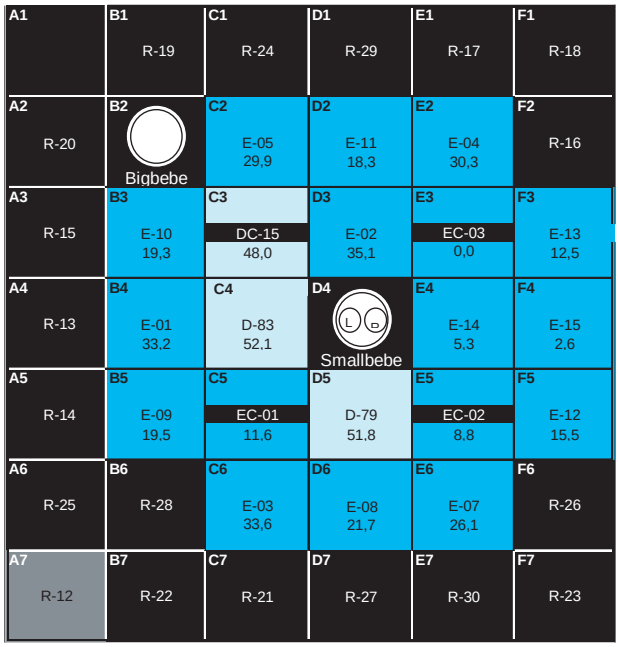

FIG. 2. Typical configuration of the HOR core.

respectively. The angle between the axes of R1 and R2, and between R2 and R3, is about 26°.

In the present project, the beam tubes are surrounded by pool water. The option of Be block embedding has already been incorporated into the upgrading licence. The system of core plus reflector plus beam needs to be reoptimized for the future Be block application, together with the CNS insertion in the beam.

The design of the beam tubes uses a two barrier approach for the in-pool sections with an aluminium outer tube envelope (AlMg3) and an inner tube insert for containing the experimental set-up. In the original design, the space between the outer and inner tube could be filled with water for cooling and shielding purposes. For the new design of beam R2, the space between the outer and inner tube normally is filled with helium at low overpressure when the tube is in use. The n-guides are enclosed in an aluminium plug, which is placed in a vacuum container. In the case of a supposed major break of the in-pile part of the beam tube (thimble), i.e. a postulated simultaneous (guillotine) break of in-pool sections of the outer and inner beam tube, there are several barriers in the design for preventing loss of pool water or mitigating the consequences. At the interface of the in-pile and out-of-pile part just outside the biological shield, the in-pile jacket is mounted to a barrier flange, which forms

a part of the secondary barrier against loss of water. The out-of-pile jacket is also mounted to this barrier flange. At the barrier flange interface, the guides are closed using 2 mm thick aluminium windows (AlMgSi1), which can fully withstand the dynamic water hammer pressure in the case of a major break. At the front and at the end of the jacket containing the out-of-pile curved guides, the guides are also closed with aluminium windows. So, there is a consecutive line of defence against loss of pool water. Should there be a failure, nevertheless, to cope with such an event, the rate at which water is lost will be limited considerably by the reduced flow area of the guides and beam inserts. As a further instrumental measure, the pressure in the vacuum system is continuously monitored and displayed in the control room with alarm and protective actions at predefined trip levels. To reduce the risk of a major break, there are tight operating restrictions to be maintained for the handling of heavy loads in the pool beam tube area.

The beams are guided into the beam hall, using dedicated isolation facilities for penetrating the reactor containment wall, for subsequent utilization purposes at the research instruments. The curved sections of the n-guides end[1] just before the containment[2] wall. Four beam shutters of $^{6}Li_2CO_3$ are placed at this position. On demand, or in the case of an emergency situation, the shutters are closed automatically in order to effectively prevent neutrons from entering the experimental area. Opening them has to be performed locally and is key controlled, whereas movement and status of the shutters is visually and acoustically displayed in the area. At the position of the guides, the containment wall of the reactor has 1 mm thick aluminium windows, sufficient to withstand the containment design overpressure of 0.1 bar.

Figure 2 shows a typical HOR core configuration (core 03-01). The core is beryllium reflected on three sides only, partly due to the constraints of the nuclear design. It has 20 fuel assemblies in total, four of which are control assemblies. Conversion from HEU to LEU is proceeding, and there are still three HEU assemblies remaining. The conversion is due for completion in 2004. The beam tubes situated at the right hand side of the core are facing the fuel directly, with a small water gap in front.

[1] At the end of the curved sections of the neutron guides, the neutrons pass through an Al window and enter a special section in air. Just before the reactor containment wall, there are four $^{6}Li_2CO_3$ beam shutters (see Fig. 4) that can be moved into the beam in order to effectively absorb the neutrons. After that section, the neutrons pass through the Al windows of the safety containment wall and enter the straight n-guide sections going into the beam hall.

[2] A safety pressure reactor containment (0.1 bar overpressure maximum).

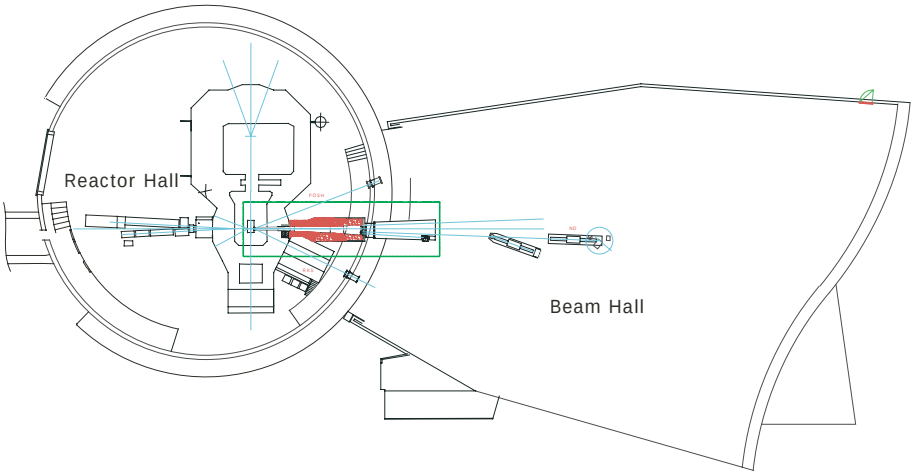

FIG. 3. Layout of reactor hall (diameter 25 m) and beam hall (the neutron guide system discussed here is indicated by a green rectangle).

In the beginning of the 1990s, elaborate preparations were ongoing within the framework of a licence application for fuel conversion from HEU to LEU. In addition, there were initiatives for reinforcing the research infrastructure and facilities at the reactor, and additional funds for that purpose were obtained. In 1995, it was decided to include into the 3 MW licence applications for the fuel conversion an extension of the plant by constructing a beam hall adjacent to the reactor hall. The licence was granted by the end of 1996 and a project was started in 1997 for establishing suitable neutron and positron beams at the existing beam tube facilities. The beams were to be guided into the beam hall, using dedicated isolation facilities to penetrate the reactor confinement wall, for subsequent utilization purposes at the research instruments. The licence approval also included the optional embedding of the beam tube front ends in the pool within a beryllium block reflector piece adjacent to the core. This option has not yet been realized during this project. Moreover, the design includes provisions for the optional installation of a cold neutron source at a later stage. Figure 3 shows the general layout of the reactor hall and adjacent experiment hall with the neutron guide system, as built.

Two beam tubes are connected to instruments in the beam hall. One is used for positron beam applications (Fig. 3). The other one is used for a variety of neutron beam applications (n-depolarization, spin echo SANS, n-depth profiling, diffractometer (planned)). The design objective for the neutron guide system was to produce four thermal neutron beams with high signal to noise ratio, using the existing neutron beam tube R2 with its large diameter. The n-beam part of the €1 million project was completed with the successful

commissioning of the system in 2000. The full scope positron beam part of the project was fully commissioned in 1999. Details about the design and development of that intense reactor based positron beam have been given in Refs [3, 4], and are not discussed here. In the framework of a continued development perspective, a feasibility study for a major system upgrade was completed recently [8]. The bulk of the material presented in the following discussion has been compiled from earlier publications [9, 11].

3. SYSTEM FEATURES

3.1. Design considerations

In choosing the principal design characteristics of the neutron guide system, the following considerations were taken into account. The system should fit into the existing outer beam tube arrangement R2, and the neutron beams should pass through proper penetrations of the steel reactor hall containment into the adjacent experimental hall. So the general geometry and many of the boundary conditions were already fixed. The interface between the reactor hall and the experimental hall with their dimensions had a great influence on design choices. The distance from the starting point of the first part of the guides to the containment interface is about 7 m. For the coating of the glass sections, the supermirror type with m = 2 was chosen with about 60% better performance than pure Ni. The m = 3 type supermirror is more expensive, whereas the life expectancy is probably lower. In general, with increasing bending radius, more microchannels are needed in order to keep the curved channels closed for neutrons travelling in a straight line over a certain distance. Of course, the total cost increases with the length and number of stacked microchannels. From a cost point of view, the curved sections were to be applied in the first part only, to be followed by straight sections in the experiment hall. In this stage of the project (the CNS still being optional), the design was optimized for the transmission of neutrons of thermal energy. From optimization studies, it followed that a curvature with a radius of 400 m should give the best performance (signal to background ratio). In our case, for the 7 m curved section with a bending radius of 400 m, a number of three stacked microchannels was found to be necessary.

3.2. Design and implementation

The four neutron guides consist of two sections: a 7 m long curved section, followed by a straight section of 4.2 m or 5.2 m (for the dimensions, see Table 1

TABLE 1. DIMENSIONS OF THE FOUR NEUTRON GUIDES A–D. THE BENT MULTICHANNEL SECTIONS, PLACED IN THE REACTOR HALL (SYSTEM I REFERS TO IN-PILE SECTION, SYSTEM II TO OUT-OF-PILE SECTION), ARE FOLLOWED BY STRAIGHT SECTIONS (SYSTEM III), GUIDING THE THERMAL NEUTRONS INTO THE BEAM HALL (SEE ALSO FIGS 3, 4 AND 6)

System number Neutron guide		I, II A	I, II B, C, D	III A	III B, C, D
Microchannel width	(mm)	5.5	5.5	5.5	21.5
Microchannel height	(mm)	40	80	40	80
Number of microchannels		1	3	1	1
Glass wall thickness	(mm)	2.5	2.5	2.5	2.5
Guide length	(m)	7	7	4.2	5.2
Bending radius	(m)	400	400	∞	∞
Direct line of sight	(m)	4.195	4.195	–	–

and Figs 4 and 6). The curved sections of three guides (denoted B, C and D) consist of three stacked microchannels. The glass walls separating the microchannels attenuate the fast neutrons and gamma rays, making the neutron guide system act as an efficient filter [5, 7]. The fourth guide (A) consists of only one microchannel. The straight sections of guides B, C and D are single channels, yielding a more homogeneous intensity distribution. The curved sections consist of two parts. System I constitutes the in-pile part of the guides, starting just within the biological shield at a distance of 1 m from the nose of the beam tube. The guides are kept in an aluminium plug (Fig. 5), placed in a vacuum container. This container is designed in such a way as to facilitate, at a later stage, the incorporation of a cold neutron source. System II is the out-of-pile part of the curved guides, starting at the end of the biological shield of the reactor and ending at the wall of the reactor hall. At this position, four beam shutters of $^{6}Li_2CO_3$ are placed, marked as 'shutters' in Fig. 4. At the position of the guides, the confinement wall of the reactor has 1 mm thick aluminium windows. The straight sections, system III, guide the neutrons into the beam hall. System I had to fit into the inner beam tube, leading to the optimum packing of the guides at the core side, as shown in Fig. 5. At the end of the straight guides, the separation lengths of the guides A, B, C and D are 250, 420 and 520 mm, respectively.

The curved part of the applied guides uses optical sections (float glass type with mirror coating) glued together into complete guide sections. The curved part of the large guide uses standard sections of 400 mm length. In the

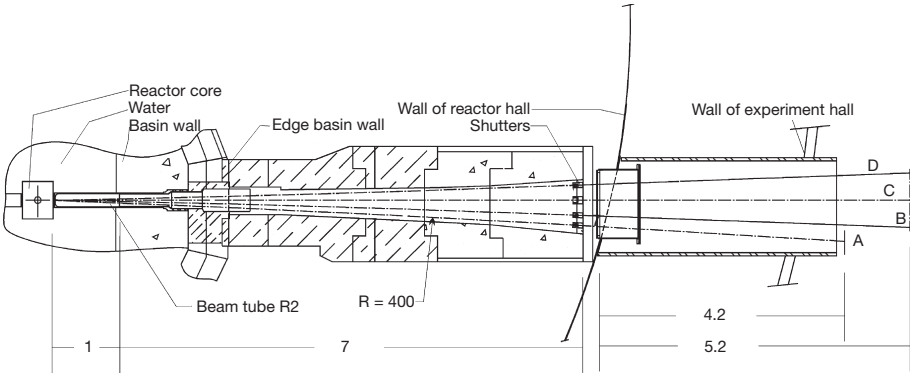

FIG. 4. Schematic top view of the installed guides in beam tube R2 (dimensions in m).

curved part, 18 bent sections have been used. In the straight part, 17 sections were applied. Special attention was given to the glass substrate (2.5 mm) and spacer thickness. The maximum value for the step between neighbouring sections due to differences in the thickness of mirrors and spacers is about 0.020 mm. Both the horizontal and vertical internal surfaces of the guides are coated with so-called m = 2 supermirror coating (NiMo/Ti) [10], i.e. the critical angle for reflection for neutrons is twice that of natural nickel. A typical reflection curve is shown in Fig. 7.

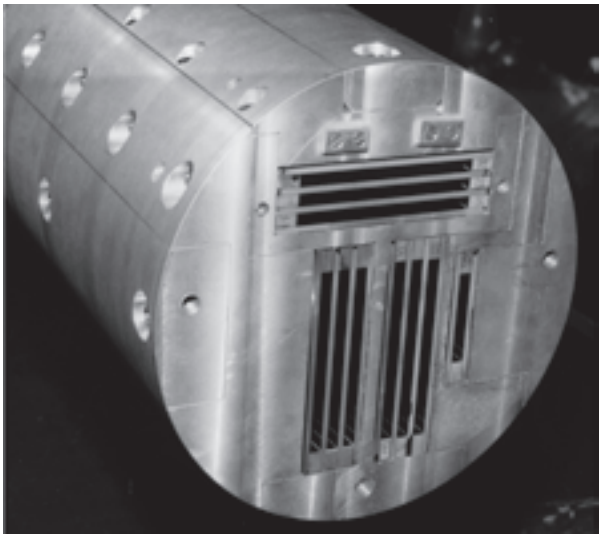

FIG. 5. Front view of the neutron guide system at 1 m from the core. Top: guide C; bottom (left to right): guides D, B and A. Behind the plates on top, two thermocouples are attached.

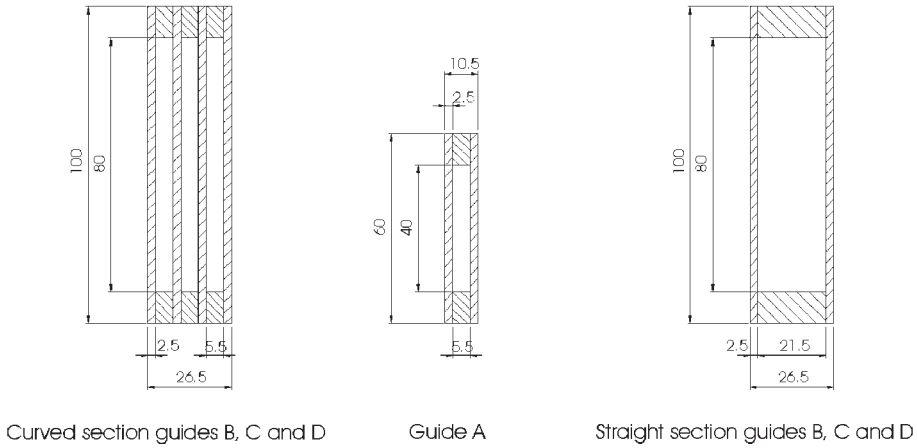

FIG. 6. Cross-sections of the curved (7 m) and straight (4.2 or 5.2 m) sections of the installed guides (dimensions in mm).

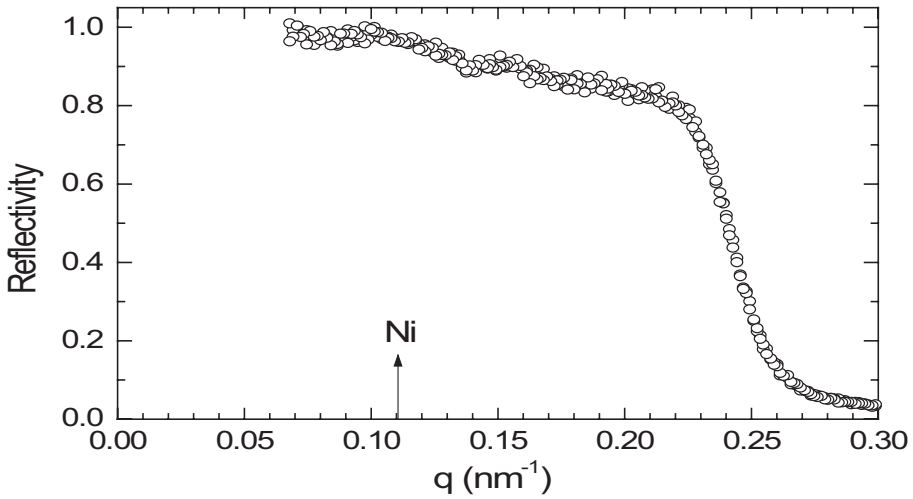

FIG. 7. Typical reflectivity curve of the supermirror coating used [9], as a function of $q = 2\pi\theta/\lambda$, with θ the glancing angle and λ the neutron wavelength. The arrow indicates the critical q-value for natural nickel.

Since the bent neutron guides act as filters, they can be considered a secondary radiation source. The strength of this source was calculated, resulting in a line source strength of approximately 10^9 fast neutrons $m^{-1}s^{-1}$ and 10^{10} gammas $m^{-1}s^{-1}$ from 1 to 4 m from the start of the guides [9]. At 4.2 m (being the direct line of sight), the line source strength drops several orders of

magnitude. From this point onwards, the main contribution to the source strength originates from the imperfect reflection of thermal neutrons, producing capture gammas. Using this calculated source strength, the shielding was designed. The main part consists of a layered shielding of 60% Fe, 35% polyethylene and 5% boron plastic, with a radial thickness decreasing from 800 mm at 1.5 m from the start of the guides to 400 mm at 5 m.

The neutron guides (including coating), the aluminium plug, the vacuum containers and the main part of the shielding were supplied by the Petersburg Nuclear Physics Institute, Gatchina, Russian Federation. Fabrication and installation were carried out and controlled within a well defined quality assurance regime.

4. PERFORMANCE

4.1. Shielding

Measurements at 2 MW power demonstrated that the shielding around the neutron guide system was adequate. The dose equivalent rates, measured at the surface of the shielding blocks, are smaller than 10 μSv/h. The four beam shutters of $^{6}Li_2CO_3$ absorb neutrons very efficiently. Due to the rather low gamma contamination resulting from the shutters when used, no additional shielding is necessary.

4.2. Signal to background conditions

The signal to background conditions at the exit of the beam R2 are much more favourable than those at the output of beam L2. The latter uses curved neutron guides of 3 m length (r = 200 m), delivering a comparable neutron beam intensity at the guide exit of about 3×10^7 n·cm^{-2}·s^{-1}. In practice, the gamma contamination of the R2 beam at 1 m distance from the aluminium exit window in the beam line is about 20 μSv/h, mainly due to neutrons captured in that window. This is about a factor of 20 less than L2. The fast neutron contamination is also reduced drastically.

4.3. Temperature

At 2 MW reactor power, the temperature measured at the nose of system I (Fig. 5), reaches an asymptotic value of 329 K after approximately two days of operation.

4.4. Transmitted thermal neutrons

A quantitative assessment was performed of the quality and performance of the system, i.e. comparing the measured and the calculated values following advanced methods. The developed methods give a reliable prediction of the neutron guide performance (with unavoidable imperfections in practice). In the ideal situation, we have perfect geometry of the guide and perfect reflectivity of the coating. In practice, there are limitations. There are three main causes of loss in performance: non-perfect reflectivity, waviness of the mirrors and non-perfect junctions (steps) between optical sections [8, 12]. The waviness of the channels can cause considerable flux losses. Additionally, air gaps and attenuation by aluminium windows reduce the performance/output. The losses are dependent on the neutron wavelength. Short wavelength neutrons are more sensitive to all imperfections. Total losses are expected to be about 50%. About half of these losses are due to non-perfect supermirror reflectivity. The other losses are caused by the waviness and imperfect junctions of the optical sections. The cost of the optical elements strongly depends on their manufacturing accuracy.

The transmission of thermal neutrons was simulated by means of the Monte Carlo technique [9]. For ideal geometry, and using the real reflection properties of the coating (Fig. 7), the integrated thermal neutron fluxes are given in Table 2. As the source, a Maxwellian spectrum for a moderator temperature of 329 K was used, together with an isotropic neutron flux of 1.4×10^{13} n·cm^{-2}·s^{-1} at the entrance of the beam tube. The calculated spectrum, displayed in Fig. 8, has a maximum at $\lambda = 0.16$ nm. Neutron fluxes were measured by means of Au foil and Cu foil activation. In the 20 mm wide water

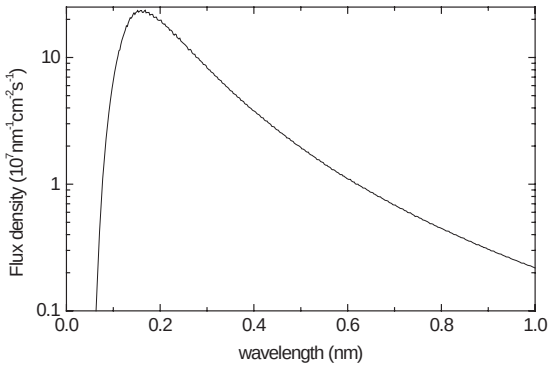

FIG. 8. The thermal neutron spectrum at the exit of the neutron guide, calculated by means of the Monte Carlo technique for an ideal geometry. In the simulation, realistic reflection curves were used.

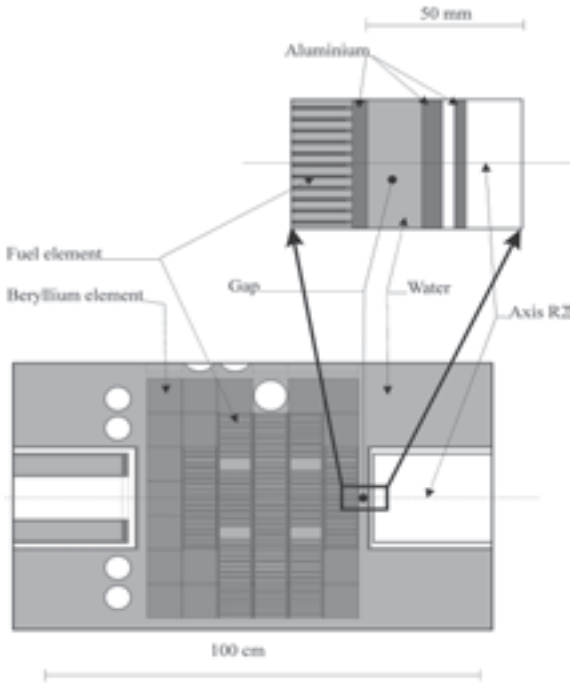

FIG. 9. Actual geometry of core and beam tube R2.

gap between the core and the nose of the beam tube, the measured flux decreases from 1.15 to 0.9×10^{13} n·cm^{-2}·s^{-1}.

By applying sophisticated methods, the influence of imperfections on the transmission of the system has been considered in detail [11, 12]. To assess the performance of the installed neutron guides with respect to their specifications, a new measurement method was introduced [11]. It is based on a comparison of the measured neutron intensity at the exit of the guides with the calculated intensity for an ideal geometry. In this way, a so-called quality factor is derived, taking into account the effect of imperfections of geometry, waviness, misalignments of the glass sections, etc. The method uses a combination of foil activation measurements in front of the beam tube in the water gap and neutron direction dependent Monte Carlo calculations for determining the equivalent isotropic thermal neutron flux distribution at the entrance of the neutron guides. The measured flux values were corrected for the anisotropic flux direction distribution obtained from the Monte Carlo calculations. An adapted version of the standard three-dimensional Monte Carlo code (KENO Va) for core validation purposes was used in determining the transmission for the actual geometry of the core and beam tube arrangement. Figure 9 shows the actual detailed model geometry (horizontal cross-section).

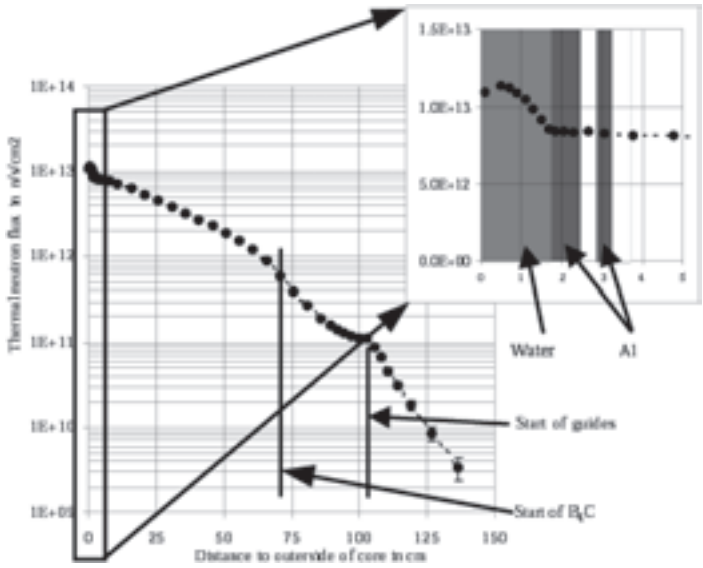

FIG. 10. Calculated thermal neutron flux as a function of distance to the core boundary.

The model was used to obtain accurate values for the neutron flux at the water gap between the core and the front end of beam tube R2, as well as in the front end section of the beam tube, under the prevailing conditions during the foil measurements. The results are displayed in Fig. 10. The calculated flux was scaled to match the actual (isotropic) flux values measured by foil activation. The reflector peak in the water gap can be clearly seen. When entering the beam tube, the flux diminishes only slightly in the first part, but decreases further with increased distance from the core. At about 600 mm, the flux decreases more rapidly due to shielding of neutrons from the surrounding water. The beam tube design incorporates a boron carbide shielding layer, which surrounds the tube in order to prevent neutron activation of steel construction parts, located inside the concrete biological reactor shield. The start of this B_4C shielding layer along the neutron flight path is indicated. As a consequence of this, an extra flux reduction is experienced downstream from the start of the boron carbide section. At approximately 1 m, there is a stronger flux decrease due to the start of the guides. However, since only those neutrons travelling in the right forward direction are of interest (neutrons heading for the B_4C shielding are almost completely lost anyway), this flux decrease is not significant in terms of the guide output.

For determining the quality factor, only the forward flux contribution, i.e. neutrons travelling from the core towards the neutron guides, is meaningful.

TABLE 2. CALCULATED VALUES OF THE TRANSMITTED INTENSITY AND THE AVERAGE WAVELENGTH OF THE SPECTRUM AFTER SYSTEMS II AND III FOR IDEAL AND EXPECTED GEOMETRY, ASSUMING AN INCOMING NEUTRON FLUX $\Phi_O = 1.4 \times 10^{13}$ n·cm^{-2}·s^{-1} AND NEUTRON TEMPERATURE OF 329 K [11]

Guide	After system	Ideal transmitted intensity (I_i) 107 n·cm^{-2}·s^{-1}	Ideal average wavelength (nm)	Expected transmitted intensity (I_e) 107 n·cm^{-2}·s^{-1}	Expected average wavelength (nm)
A	II	7.97	0.291	5.21	0.283
	III	7.04	0.288	4.06	0.277
B, C and D	II	7.16	0.279	4.84	0.273
	III	5.99	0.285	3.94	0.282

The transmitted neutron intensity at the exit of the guides was also measured by foil activation.

The foils placed at the exit of systems II and III covered the cross-section of the guides. The reactor power during the activation was 2 MW. The average wavelength of the neutrons in the spectrum at the exit of the neutron guides is approximated by the calculated average neutron wavelength using the expected transmission of the guides, T_e (λ,Ω), that was determined numerically. The waviness of the mirrors and the non-perfect junctions between optical sections were taken into account. For every guide, the average wavelength of the neutrons is determined assuming the 'ideal' transmission and 'expected' transmission. The results are shown in Table 2. The average wavelength for the ideal and expected transmissions deviate by less than 5%, indicating that the error introduced by taking the calculated average wavelength instead of the real one will be even smaller.

For every channel, the total neutron intensity was measured at the end of system II: $I_{m,II}$ and at the end of system III: $I_{m,III}$. The measured intensities are shown in Table 3. The neutrons travelling through the neutron guide pass through several aluminium windows with a total thickness of 4–6 mm and through an air gap of 240 mm. This is taken into account in calculations of the expected intensity I_e.

Table 3 displays the overall results of the comparison, including the measured and expected quality factors for the guides. The measured and expected quality factors were calculated by dividing I_m and I_e by I_i. The measured quality factors correspond very well to the expected quality factors,

TABLE 3. INCOMING NEUTRON FLUX (Φ_o), IDEAL TRANSMITTED INTENSITY (I_i), MEASURED TRANSMITTED INTENSITY (I_m) AND MEASURED AND EXPECTED QUALITY FACTOR (QF_m AND QF_e) FOR GUIDES A–D AFTER SYSTEMS II AND III [11]

Guide	After system	Φ_o 10^{13} n·cm^{-2}·s^{-1}	I_i 10^7 n·cm^{-2}·s^{-1}	I_m 10^7 n·cm^{-2}·s^{-1}	I_e 10^7 n·cm^{-2}·s^{-1}	QF_m	QF_e
A	II	1.35	7.67	5.09	4.99	0.66	0.65
	III		6.96	3.32	4.04	0.49	0.58
B	II	1.36	6.96	5.08	4.73	0.73	0.68
	III		5.83	3.65	3.85	0.63	0.66
C	II	1.32	6.83	4.35	4.64	0.65	0.68
	III		5.64	3.10	3.72	0.55	0.66
D	II	1.36	6.96	4.93	4.73	0.71	0.68
	III		5.83	3.56	3.85	0.61	0.66

which were determined using a realistic waviness of the mirrors and the real reflectivity of the coating.

4.5. Reactivity influences

The R2 beam set-up is evacuated and filled with helium at low overpressure when in use. The rather wide dimension of the beam tube with an outer diameter of 245 mm at the front end facing the core extends over the full range of the three grid positions F3 to F5, filled with standard fuel assemblies (see Fig. 2). In this way, it constitutes a big hole in the water reflector, imposing a significant negative reactivity effect, which must be accounted for from a safety and operational point of view. The reactivity effect was determined from calculations to be of the order –0.5% at maximum. The best estimate derived from the critical rod positions before and after the beam tube modifications resulted in a value of about –0.4%.

5. CONCLUSIONS

Within the framework of an ambitious, innovative research facility project, a new neutron guide system with high signal to background ratio was designed, built, installed and successfully commissioned. Starting in one of the main beam tubes, four neutron guides feed thermal neutrons into the recently

built beam hall for beam instrument utilization purposes. Provisions are made to include, at a later stage, a cold neutron source in the nose section of the beam tube. Also, to further enhance the available neutron flux, the option of beam tube embedding in a beryllium reflector block was included in the licence procedure. At the exit of the 12 m long guides, the thermal neutron flux is $3–4 \times 10^7$ n·cm^{-2}·s^{-1}, which is an excellent result for a 2 MW reactor. A quantitative assessment was completed of the quality and performance of the system, i.e. comparing the measured and the calculated values following advanced methods. The developed methods give a reliable prediction of the performance of the neutron guides with their unavoidable imperfections in practice. The potential to further enhance the output of the guides by additional optimization of the core–beam tube interface and extension with provisions, such as beam tube embedding and a cold source application, is certainly promising for another major boost in research output capabilities.

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6.3. HIGH UTILIZATION OF THE NEUTRON BEAM SYSTEM AT THE FRG-1 AT GEESTHACHT

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1. INTRODUCTION

The GKSS Research Center Geesthacht GmbH has operated the MTR type swimming pool research reactor FRG-1 (5 MW) for more than 40 years. The FRG-1 was originally designed and constructed in 1957–1958 (first criticality 23 October 1958) to serve general scientific research needs in both fundamental and applied research. It is obvious that during the lifetime of the reactor, the areas of research have changed more than once. The result of these changes is, on one hand, new experimental facilities at the outer end of the beam tubes and, on the other, numerous design changes at the reactor. At present, 13 neutron scattering instruments are continuously operated at the beam tubes, two thirds of which use cold neutrons.

2. BEAM TUBES

The FRG-1 is primarily used as a neutron source for neutron beam experiments. A supplementary application is the irradiation of samples for universities, industry and research centres (see paper 9.3 by Knop et al. of this Compendium).

In order to make available a high neutron flux for neutron scattering experiments, the FRG-1 has a set of eight beam tubes, consisting of seven radial tubes (see Fig. 1) and one tangential through tube. The tangential beam tube has a very low background, as it does not have a direct view to the core. For the reduction of measurable background from gamma radiation and fast neutrons at the experimental positions, three of the radial beam tubes have installed a system of curved neutron guides.

The beam tubes which are made from AlMg3 are approximately 3100 mm long with a diameter of either 150 mm or 200 mm. One tube has been extended to 350 mm diameter in parts to accommodate the neutron guide system. At its inner end, this tube is facing the cold neutron source (CNS). The principal structure of the beam tubes is represented in Figs 2 and 3. All beam tubes are double walled (two barriers against loss of pool water) and filled with demin-

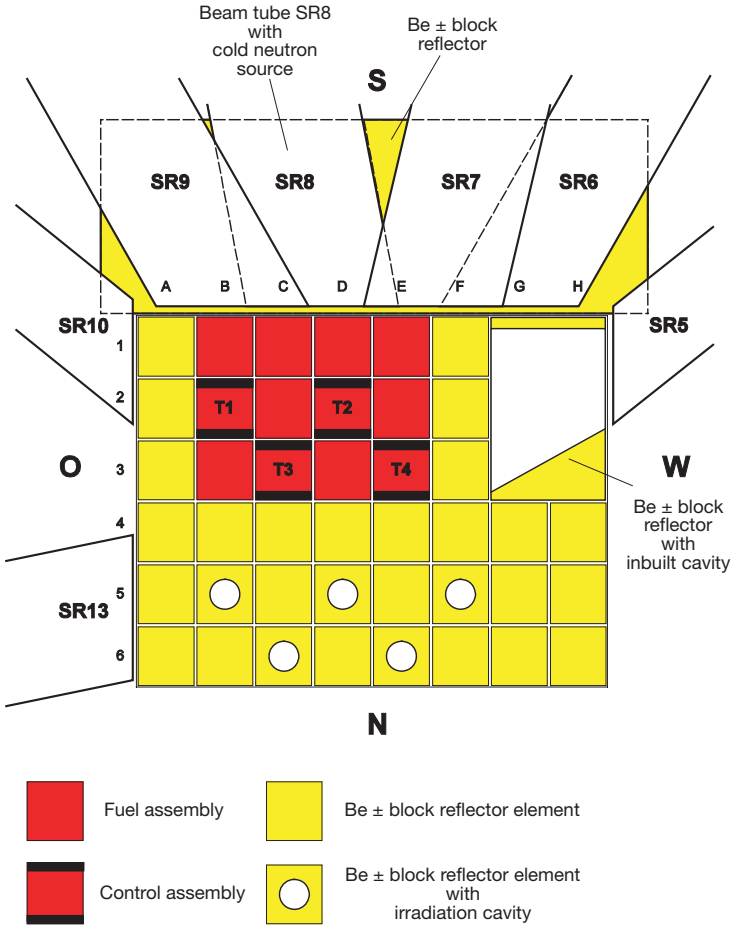

FIG. 1. Model of the 3 × 4 core with a beryllium reflector and beam tubes.

eralized water when not in use, namely, the gap and the inner beam tube. The water gap between the inner and the outer beam tube is approximately 1 mm. In order to open the neutron beam (see Fig. 2) for an experiment, the internal tube, i.e. collimator, is emptied of water and filled with helium. The neutron beam collimation is determined by the length and the diameter of the internal beam tube.

3. INCREASE OF NEUTRON FLUX

High neutron flux and signal to background ratio are crucial for the beam quality. Mainly for the first reason, GKSS accomplished two core compactions.

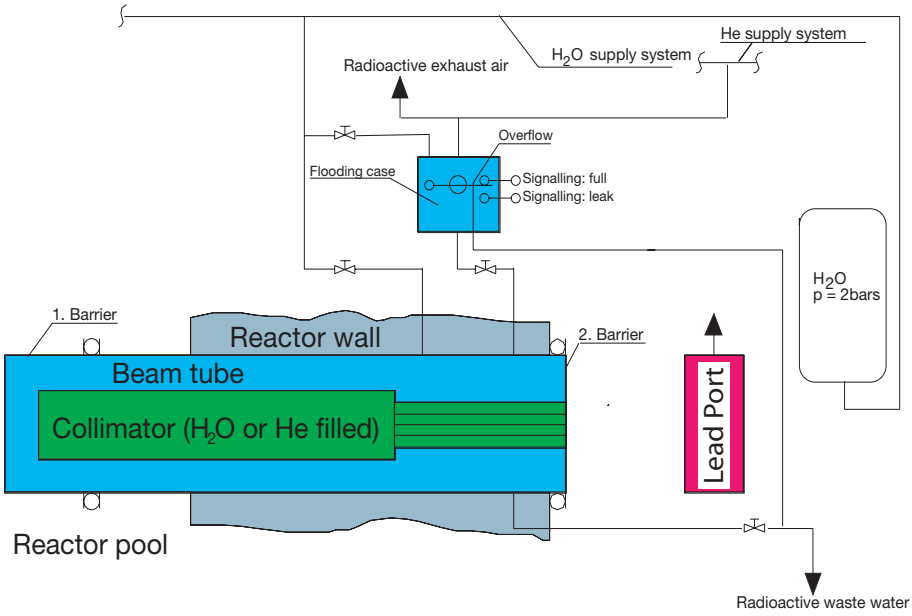

FIG. 2. Beam tube operational monitoring system.

The first step towards increasing neutron flux was carried out in February 1991 when the FRG-1 was converted from high enriched uranium (HEU 93%) to low enriched uranium (LEU 20%) in one step. At the same time, the reactor core was reduced from 49 to 26 fuel assemblies (see Fig. 3) [1]. As a result of this compaction, the neutron flux at the beam tubes was increased by more than a factor of two.

At the beginning of the year 2000, a further reduction of the core size was made in order to again increase the neutron flux at the beam tubes, in particular at SR8, which faces the cold neutron source (CNS). With this compaction, the reactor core was reduced from 26 to 12 fuel assemblies. Figure 1 shows the resulting 3×4 core of the FRG-1 as it has been operated since 2000.

To achieve this reduction, the fuel density in the fuel plates had to be increased from 3.7 g U/cm^3 to 4.8 g U/cm^3 [2]. The second core compaction gave rise to a further flux enhancement by another factor of two.

The extreme core compacting is distinctly illustrated in Fig. 4, where all the cores of the process are shown. Following the last core compaction, the unperturbed neutron flux at the position of CNS is now 1.4×10^{14} n·cm^{-2}·s^{-1}.

The core compaction was accompanied by measures to improve the reflector region of the FRG-1. In 1988, before the first core compaction, at the south side of the core a beryllium block reflector was installed in which the core

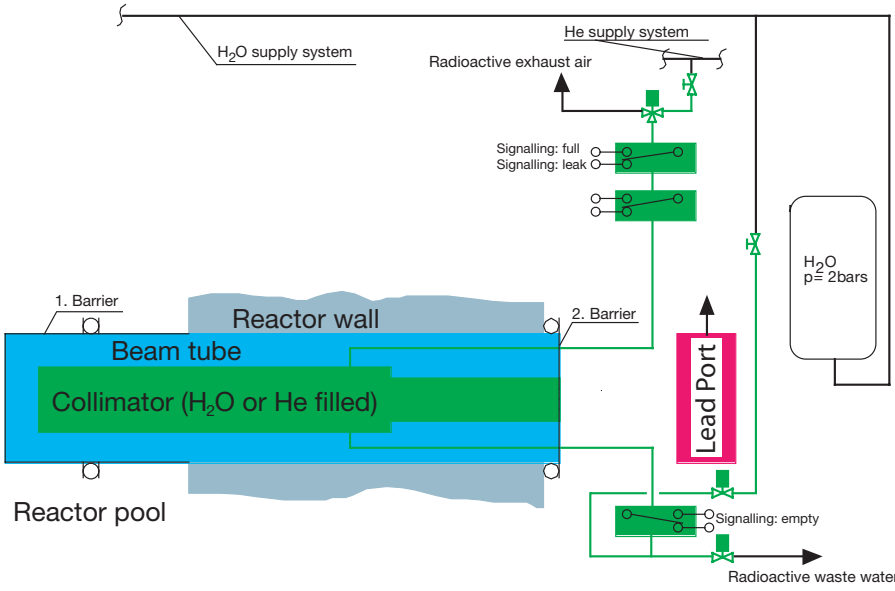

FIG. 3. Collimator operational monitoring system.

directed ends of most beam tubes are embedded. At the same time, in beam tube SR8, inside the beryllium block reflector, the CNS, a supercritical gaseous H_2 moderated cold neutron source, was installed. At that time, the position of the moderator chamber of the CNS was optimized by two-dimensional diffusion calculations.

In 2000, during the second core compaction, new beryllium reflector elements were added at the new grid plate. The old grid plate, with a matrix of 8 × 10 positions for fuel assemblies, had no shroud to guide and enclose the coolant streaming through the core area. To reduce the coolant bypass flow resulting from the lack of a shroud, the new grid plate (6 × 8 matrix) was equipped with an aluminium shroud. Consequently, the new 3 × 4 compact core is completely encased with beryllium, partially (at three core sides) by filling the remaining positions at the grid with Be reflector elements. A three-dimensional Monte Carlo computation was performed for each beam tube. These extensive calculations have resulted in the following optimization measures: positioning of the new 3 × 4 core on the grid plate directly in front of beam tube SR8 with its CNS. Since the reduction of the core size led to a distance of approximately 250 mm between the core and the start of beam tube SR5, while the maximum of the thermal neutron flux arose at approximately 90 mm from the core, there was the need for a better adaptation of SR5. From the results of the three-dimensional Monte Carlo calculations, a beryllium

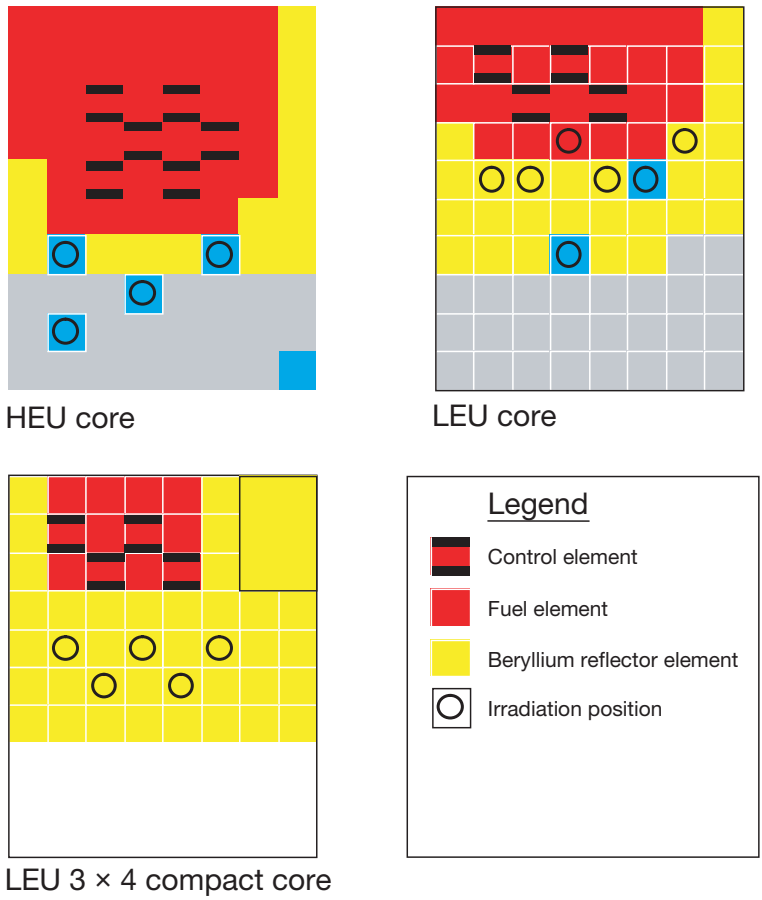

FIG. 4. The different cores of the FRG-1.

block reflector was designed that included a special cavity box filled with helium to provide optimal beam tube coupling with the thermal neutron spatial distribution. Additionally, the water gap between the grid plate outer border and beam tube SR5 was minimized. A special mobile aluminium spacer displaces most of the water there. This Al spacer can be withdrawn to enable the movement of the FRG-1 core from its standard position in the pool, according to the FRG-1 licence. The measures above were accompanied by an optimization of the five irradiation positions at the north side of the core. The irradiation facilities provided are beryllium elements with bore holes drilled at their centres, with a diameter of 50 mm each. Due to their positioning, the reactor cycle length was not affected and the reactivity feedback by interaction from sample handling was minimized. These changes have resulted in a

convincing upgrade of the FRG-1 to a modern middle flux neutron source and have, at the same time, contributed to addressing the effects of ageing.

4. BEAM TUBES RELEASE AND SAFETY ASPECTS

The beam tube operation (see Figs 2 and 3) is controlled by a programmable logical controller (PLC). The release switch to open the beam tubes can be operated from the reactor control room only. A request to do this cannot be made by persons other than a radiation protection technician or the personnel responsible for the related instrument. By this procedure, it is ensured that the instrument apparatus and especially the shielding is in adequate condition. After the release, the collimator can be emptied and automatically rinsed with helium. The use of helium instead of a vacuum in the collimator avoids the water hammer if the sudden break of one barrier occurs. The helium and the water from the beam tube system are connected to the exhaust air system and to the radiation waste water system (see Figs 2 and 3). An uncontrolled activity release is excluded by this measure. The status of the beam tubes (empty, full, leaking) is indicated at several places, such as on panels at the experiment hall, in the reactor control room and within a monitor system.

In addition to the open/close condition of the lead port, the beam tube and the collimator, a leakage monitor is integrated in the system. Should a leakage occur, optical and acoustical alarms are actuated so that appropriate measures can be initiated immediately. In addition, every four hours the beam tube system is controlled by shift workers. Furthermore, a loss of coolant incident is prevented by two barriers at both the beam tube ends (pool, experimental hall). The reactor core can also be moved sideways while hanging on the movable pool bridge.

5. CONCLUSION

The two core compactions of the FRG-1 during the last 12 years are very good examples of upgrading projects at research reactors in which existing fixed arrangements of beam tubes have to be linked to a renewed core plus reflector system. Also, three-dimensional Monte Carlo calculations have been demonstrated as an excellent tool for optimization of material selection and geometry (core, reflector, spacer, cavity, water gap) in order to gain the best performance.

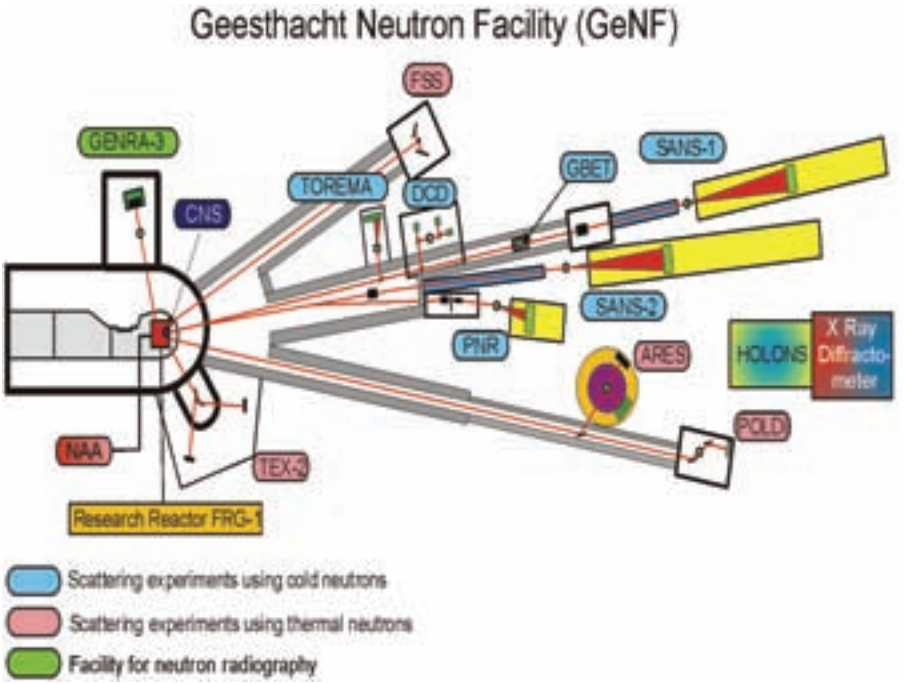

FIG. 5. Instruments at the FRG-1.

With those prerequisites, modern instruments (see Fig. 5) at unpolarized and polarized neutron beams (cold, thermal) offer a broad range of experiments and applications for research. Along with safe reactor operation with a reliable operating schedule, a high availability of about 250 full power days per year is provided. Such a level of performance is of great importance. The FRG-1 core compactions, the installation of the CNS and numerous other measures for improvement help to ensure facility operation over the next ten years, during which the FRG-1 should be serving as a national neutron source for scattering and irradiation experiments.

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6.4. NEUTRON BEAMS OF FRM-II AT THE TECHNISCHE UNIVERSITÄT MÜNCHEN

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1. THE NEW RESEARCH REACTOR FRM-II

The FRM-II is the new high performance research reactor which has been built by the Technische Universität München (TUM) in Garching near Munich, Germany [1, 2]. The reactor concept provides for a single cylindrical fuel assembly with only 240 mm outer diameter containing 8.1 kg of highly enriched uranium in a silicide fuel with relatively high density. This 'compact core' is cooled by light water H_2O and placed in the centre of a large tank of heavy water D_2O (having both 2500 mm diameter and height).

At only 20 MW maximum reactor power, unperturbed thermal neutron flux density of about 8×10^{14} n·cm^{-2}·s^{-1} will be obtained outside the core, with a high spectral purity and in a large usable volume because of the D_2O moderator. So this reactor has been clearly optimized with respect to beam

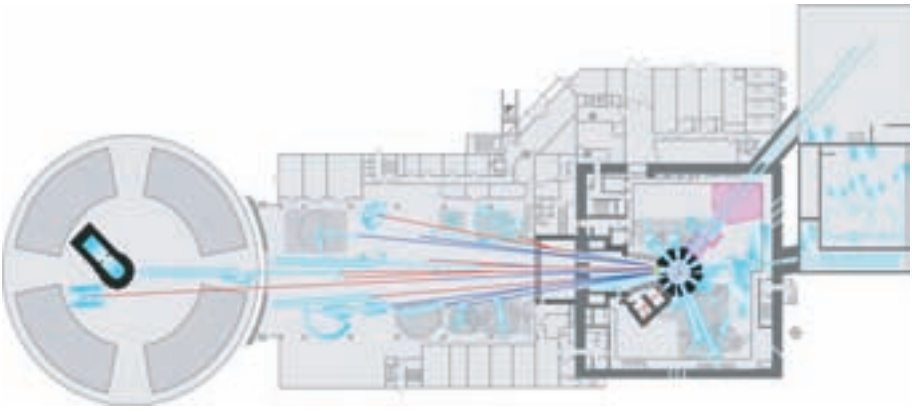

FIG. 1. Layout of the FRM-II facility. The FRM-II reactor building (centre right) with its experiment hall on the ground floor level has a square cross-section with about 40 m inner dimension and thick walls for safety, even in the event of an airplane accident. The neutron guide hall (centre left) can be extended at a later date into the hall of the old FRM (far left) which has been shut down. The building at the far right represents an option related to a fission fragment ion accelerator and additional instruments using cold neutrons.

tube applications, but will also be very attractive for sample irradiations and other fields of utilization. The FRM-II not only produces thermal neutrons but also provides for a number of secondary sources (cold, ultracold, hot and high energetic neutrons, as well as positrons and fission fragment ions) with a total of ten horizontal and two inclined beam tubes and five irradiation facilities, including an option for a vertical beam tube directly from the cold neutron source. An extended system of neutron guides gives access to a wide space for the instruments — not only in the experiment hall of the reactor building but also in a supplementary neutron guide hall (Fig. 1). By the end of the nuclear startup procedure, some 19 instruments will be ready. It will be possible to perform 30–35 experiments simultaneously.

2. BEAM TUBES OF THE FRM-II

2.1. General arrangement of the beam tubes

Figure 2 shows a horizontal cut at fuel assembly level through the reactor pool. The reactor is controlled by a single control rod (consisting of a hafnium absorber with a beryllium follower underneath) which also serves as a shutdown rod and which, during the reactor cycle, moves upwards in the inner tube of the fuel assembly. Additionally, there are five shutdown rods in the moderator tank next to the core, which are shown in their shut down positions in Fig. 2 but which are totally withdrawn during reactor operation. The ten horizontal beam tubes (labelled 'SR' from the German 'Strahlrohr' or beam tube) are numbered clockwise. SR1, SR2 and SR4 point to the cold neutron source, SR9 to the hot source and SR10 to the fission neutron converter. SR4 will incorporate a source of ultracold neutrons and the through tube SR6 a fission target for the fission fragment ion accelerator. The two inclined beam tubes, SR11 and SR12, and the optional vertical tube SR13 cannot be seen in Fig. 2. The large beam tube on the far left (SR1) contains six neutron guides (which further 'downstream' split into even more, see Section 3), supplying the whole neutron guide hall with cold neutrons and demonstrating the importance of the cold neutron source.

The following general features of the arrangements of the beam tubes include:

— All beam tubes are 'tangential' to the core, which means that they do not directly 'see' the fuel assembly. This is very important since all (unwanted) direct radiation from the core — mainly fast neutrons and gammas — cannot directly reach the outer end of the beam tube, which

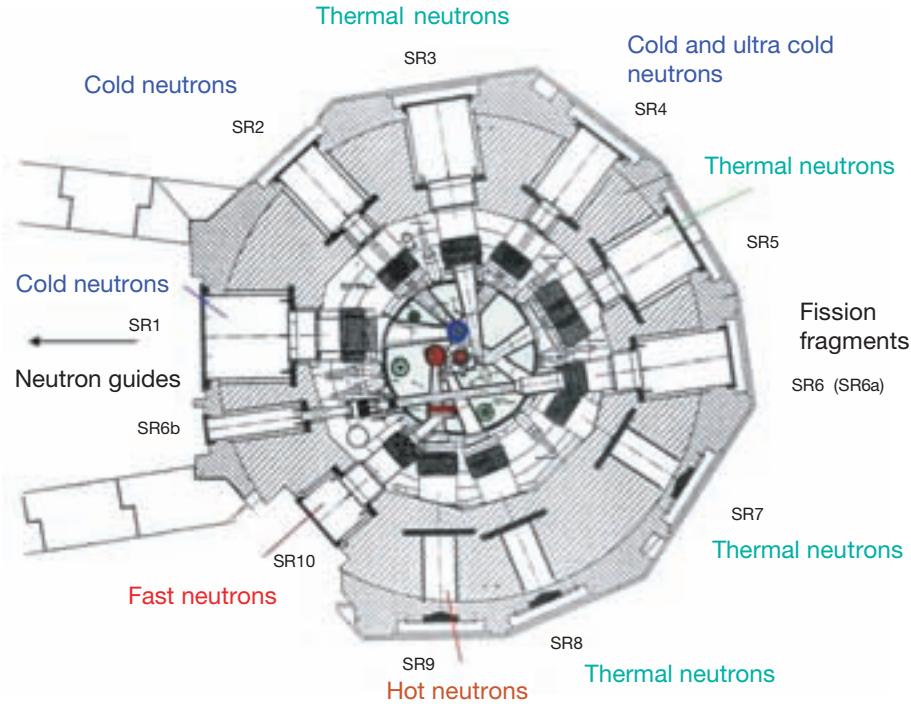

FIG. 2. Horizontal cut through the reactor pool at the level of the fuel assembly. From the outside, one can see the concrete pool structure with the beam tube penetrations, the light water of the pool, the heavy water in the moderator tank (which has a diameter of 2500 mm), some tank installations and the reactor core in the centre.

significantly reduces the background at the position of the experimental instruments. This is the best solution, however, it is only possible at a reactor with a heavy water moderator (D_2O), since only in this case the ('reflector') thermal neutron flux peak is (radially) broad enough and located sufficiently away (some 120 mm) from the core surface. At a research reactor using a light water moderator (H_2O), the thermal flux peak would be narrow and located only about 20 mm away from the core surface. So, only 'radial' beam tubes (i.e. pointing to the core surface) would make sense at such a facility.

— To arrange all the beam tubes horizontally is the best choice concerning their coupling to the field of thermal neutrons which has cylindrical symmetry around a vertical axis according to the orientation of the cylindrical fuel assembly. It is also the best choice concerning their experimental utilization which occurs in large experimental areas with dedicated neutron scattering instruments. This is even more so if the

beam tubes are to feed neutron guides which may be up to 100 m long. Nevertheless, at the FRM-II, two inclined beam tubes (SR11 and SR12) are also installed, connected to horizontal experimental platforms: one will use a set of single or double monochromators to provide an outgoing horizontal beam with a variety of fixed neutron energies for a neutron diffractometer, and the other incorporates an intense source of positrons which can easily be directed by their charge and thus provide a horizontal positron beam downstream.

— A regular pattern of beam tube noses in the moderator tank — say, a beam tube nose at every 30–40° — would generally be a good choice as far as neutron flux depression and mutual interaction effects are concerned. Practically, however, this does not seem to be feasible because of the other important installations in the moderator tank. For example, the cold neutron source and the hot neutron source are both relatively large and must be placed in the peak thermal flux region, i.e. close to the core. The cold source, as the most important installation in the moderator tank, is linked to three beam tubes simultaneously (SR1, SR2 and SR4). There is also a through tube, SR6, where a fission target of about 1 g ^{235}U can be inserted from one side (SR6b) to allow an intense beam of fission fragment ions to be extracted on the other side (SR6a) (Fig. 2). For all these reasons, the horizontal beam tubes had to be positioned at different vertical levels — which, in principle, also allows fitting them to the neutron scattering instruments that require the neutron beam at different heights.

— The cross-section of a typical beam tube is shown in Fig. 3 (in a semi-schematic diagram as was directly used for Monte Carlo calculations [3]). The beam tube nose is made of aluminium alloy (AlMg3) and mounted directly to the appropriate flange of the D_2O tank (using a double seal with a controlled space in between). The outer part of this flange is connected to the steel liner structure in the pool wall by a ‘compensator tube’ which has a double bellow (the space between its walls also being controlled) to compensate for small fluctuations in geometry. Most of the inner space of the beam tube is consumed by a shielding plug, filled with heavy concrete and connected to the outer flange of the liner structure in the pool wall. The shielding plug consists of a fixed inner part and has a layer of neutron absorbing boron carbide at its front face, and an outer part (‘shutter’) which can be rotated around the beam tube axis. Two neutron beam channels with cross-sections of typically 80 mm width and 120 mm height are provided in the shielding plug. Both channels are prepared to host neutron guides on their full length. Since these two

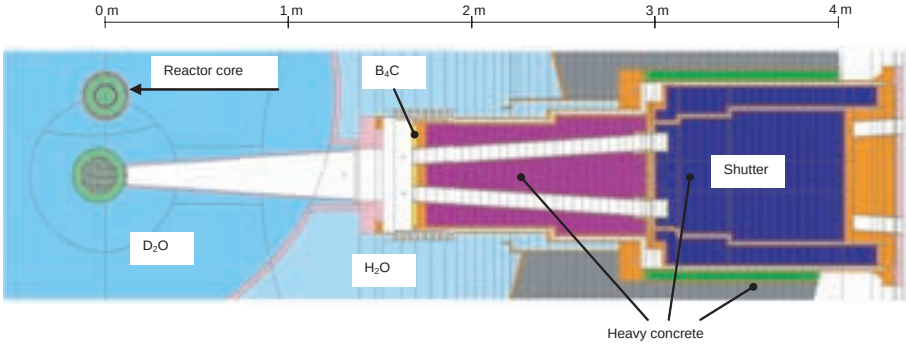

FIG. 3. A horizontal cut through a typical beam tube of the FRM-II (in this case, SR9 leading to the hot neutron source). Shown is the Monte Carlo model used to calculate the influence of the unavoidable gaps on the radiation background at the position of the scientific instruments. The shutter can be rotated around the beam tube axis and is shown here in its 'fully closed' position [3].

channels are separated by an angle of about 8°, it is generally possible to place two neutron experiments at the end of most beam tubes, as can also be seen from Fig. 1. The rotating shutter, which is shown in Fig. 3 in its 'fully closed' position, has three penetrations and four angular positions so that either one or both of the neutron beam channels can be 'opened' or closed. It is for safety reasons that the outer cover of the beam tube has two aluminium windows (with a total thickness of about 5 mm) and also that the inner volume is filled with helium gas of 1.2 bar absolute pressure (to avoid a 'water hammer' in the event of a sudden large leak).

2.2. Neutron flux depression at the beam tube noses

The beam tube noses have been made of the alloy AlMg3 (i.e. aluminium with 3 wt% of magnesium), which has a low cross-section for neutron absorption. Typically, the wall thickness at the front faces is 5 mm but the side walls are thicker and have a conical shape. Some of the thermal tubes have a circular cross-section of about 80–100 mm inner diameter, but SR7 is rectangular with 140 mm width and 90 mm height to provide an ideal geometry for a double focusing monochromator using a slit diaphragm inside the beam shutter as a virtual source. This clearly demonstrates that the beam tubes have been designed from the beginning towards their later utilization.

With respect to the 'unperturbed' case, i.e. the moderator tank being filled with D_2O only and nothing else, the installation of beam tubes (and of other facilities) always leads to a loss of reactivity and to a depression of the

thermal neutron flux at the position of the beam tube. There are two reasons for this effect: firstly, the neutron absorption of the various structural materials of the installations is always stronger than that of D_2O; and secondly, there is neutron leakage out of the peak flux into the D_2O moderator tank through the beam tubes (or other empty spaces). Unfortunately, most of these leaking neutrons disappear in the side walls or the shielding material. Only a very small fraction arrives at the outer end of the beam tubes and can be used for the experiments. From these arguments, the front faces of the beam tubes should be small to keep the flux perturbations small but, on the other hand, these front faces should be large to provide a large source area for the neutron beam. It is clear, therefore, that a compromise is required.

For the FRM-II, the position and the geometry of the beam tubes (and of the cold and hot neutron sources) have been optimized by three-dimensional Monte Carlo calculations [4]. The axes of the tangential beam tubes are about 120–150 mm from the core channel tube surface, which corresponds to the position of the peak thermal flux, but the beam tube noses have been withdrawn from this 'point nearest to the core' (as defined by a line perpendicular to both the beam tube axis and the core axis) by about 50–200 mm. This withdrawal has only little effect on the neutron fluxes at the outer end of the beam tube but strongly reduces the perturbation effects. For example, the (average) thermal neutron flux in a 5 mm thick layer of D_2O at the position of the nose of the beam tube SR5 is 7.5×10^{14} n·cm^{-2}·s^{-1} in the unperturbed case and 33% less with the beam tube installed. For SR7, which has a particularly large rectangular cross-section (140×90 mm^2, as mentioned) and other beam tubes very close, the flux depression comes out to be 38%. At the typical position of a monochromator, i.e. 1 or 2 m outside the outer end of a typical thermal beam tube, thermal neutron flux densities of about 10^{10} n·cm^{-2}·s^{-1} are expected.

2.3. Reduction of reactivity as caused by the beam tubes

The same reasons that result in the neutron flux depressions at the positions of the beam tube noses, see Section 2.2, also lead to a loss of the excess reactivity of the reactor. At the FRM-II, the total loss of reactivity as caused by all experimental installations in the moderator tank (i.e. including the cold and hot neutron sources, etc.) varies slightly during the cycle, since it depends somewhat on the position of the central control rod. At the beginning of cycle (BOC), the position of the hafnium/beryllium border of the control rod is slightly below the core midplane where the installations have a larger influence on reactivity. At the end of cycle (EOC), the control rod is fully withdrawn and the influence of all installations on reactivity becomes more

uniform. The operational period between BOC and EOC, the cycle, which consumes one fuel assembly, is about 50 full power days (non-stop).

The three-dimensional Monte Carlo calculations provided the following results for the reactivity reduction $\Delta\rho$ as caused by all experimental installations (Fig. 2) in the D_2O moderator tank [4]:

— All installations, at BOC: $\Delta\rho = -(5.02 \pm 0.05)\%$
— All installations, at EOC: $\Delta\rho = -(4.83 \pm 0.06)\%$

The loss of reactivity at EOC is practically most relevant, since under the conditions close to the end of the reactor cycle, an excess reactivity $\Delta\rho$ of 1% is consumed in the course of about one week. All these values of reactivity have to be compared with the excess reactivity $\rho = 12.3\%$ which is effectively available at the FRM-II at BOC.

The losses of reactivity $\Delta\rho$ originating from single beam tubes have also been determined by the Monte Carlo calculations mentioned [4]. Naturally, the individual values of $\Delta\rho$ strongly depend on the sizes of the beam tubes and on how close to the core they are positioned. This positioning does not only mean their horizontal distance from the core surface, but also their vertical distance from the core midplane. With respect to interaction effects, it is also important whether there are other big installations nearby or not. At EOC, typical values of the reactivity losses $\Delta\rho$ due to the installation of certain beam tubes are (with the remaining reactivity losses $\Delta\rho$ for the case that the inner volumes of the beam tubes were flooded with D_2O, given in brackets):

— SR1: $\Delta\rho = -(0.33 \pm 0.02)\%$ [$-(0.15 \pm 0.01)\%$ when flooded]
— SR3: $\Delta\rho = -(0.21 \pm 0.02)\%$ [$-(0.13 \pm 0.02)\%$ when flooded]
— SR5: $\Delta\rho = -(0.25 \pm 0.02)\%$ [$-(0.14 \pm 0.02)\%$ when flooded]
— SR7: $\Delta\rho = -(0.27 \pm 0.02)\%$ [$-(0.16 \pm 0.02)\%$ when flooded]
— SR1/2/3/4: $\Delta\rho = -(0.77 \pm 0.03)\%$ [$-(0.38 \pm 0.02)\%$ when flooded]

Since the values of $\Delta\rho$ in the first column consider the combined effects of both the enhanced neutron absorption in the structural materials and the neutron streaming effects in the ‘vacuum’ of the beam tubes, and since the values in the second column only reflect the enhanced neutron absorption, it follows that each of these two effects contributes about half to the total reduction of reactivity. The flooding of a given beam tube with D_2O leading to an increase of reactivity is to be used as an input parameter for the discussion of accident scenarios. Due to mutual interaction effects, the increase of reactivity $\Delta\rho$ resulting from the simultaneous flooding of neighbouring beam tubes with D_2O is always slightly smaller than the sum of the individual contributions.

3. NEUTRON GUIDES AT THE FRM-II

As a consequence of its neutral character, the flight direction of a neutron beam can only be influenced by total reflection or scattering. To provide an optimized neutron flux for the instruments, only the first method is of practical importance, due to the limited spectral range of Bragg scattering. The optimized use of beam tubes at the FRM-II implies spatial restrictions for the construction of the instruments. In the experiment hall of the reactor building (Fig. 1), a compromise between required beam divergence and distance from the source has been found, taking into account the ratio of neutron flux to background at the instrument. In the neutron guide hall, high priority is given to maintaining a low measuring background.

For both situations, neutron guides provide an essential performance gain for the scientific instruments. These guides consist of glass tubes with rectangular cross-sections (typically 60×100 mm^2) that are coated inside. By using a coating that maximizes reflection, neutrons can be guided over long distances without significant losses. These coatings require a very flat and smooth surface of the carrier structure. Therefore, glass (polished or float glass) is used as a carrier material. Additionally, the glass can be used as a neutron absorber for neutrons that are beyond the critical angle of total reflection using boron containing material. To prevent scattering in the air, these guides are evacuated. The gain in performance of the instruments is achieved by optimizing different parameters, namely, a cross-section of the guide, coating (critical angle of reflectivity), curvature and, eventually, inner blades. Numerous Monte Carlo simulations were performed to optimize these conditions for each instrument.

For the case where the natural divergence from the source, as defined by the beam tube nose to the instrument, does not match the special requirements of the instrument, neutron guides are installed in the beam tubes starting at about 2 m from the source (Fig. 3). The minimum distance is given by the heat load from the radiation level inside the beam tubes and the heat dissipation from the guides, respectively. In the experiment hall of the FRM-II, six instruments out of nine will use neutron guides, whereas the neutron guide hall is entirely served by neutron guides, i.e. all instruments use guides there.

The angle of total reflection is limited by the refraction index of the coating material. For neutrons, such angles are in the range of 0.1–1°, depending on the linearity of the neutron wavelength. Nickel and especially the isotope ^{58}Ni have the highest refraction index for neutrons. To overcome this limit, a special coating of alternating layers of Ni and Ti extends the range of total reflection using Bragg scattering from these layers with variable spacing. At the FRM-II, most of the guides use these so-called supermirrors with m = 2,

where m = 1 corresponds to the maximum angle of total reflection of natural Ni. To focus the beam on the sample position, even higher coatings on small sections up to 3.6 m have been used. The choice of the coating depends on the required divergence at a given wavelength for the instrument. Therefore, different parameters have to be taken into account. To optimize the neutron flux for the instrument, the reduced reflectivity at higher values of m has to be considered. Whereas for low m (<1.2), nearly 100% reflectivity can be achieved, a maximum of about 90% is still feasible for m = 2, but it drops down to 60% at m = 3.6. This is due to the increasing number of layers for super-mirrors with high critical angle of reflectivity. An actual review of mirror systems for cold and thermal neutrons was given in Ref. [5].

The wavelength dependence of the total reflection of the guides can be used for selecting a range of neutron wavelengths with neutron guides. By using curved guides with a curvature radius in the range of 1000 m, short wavelength neutrons are prevented from entering the neutron guide hall. At the entrance of the neutron guide hall (Fig. 1), each neutron has to be reflected at least once, i.e. the curvature of the guides prevents a direct sight to the cold source. The range of typical cut-off values is 1.8–3.7 Å. The transmission curve of a guide system can be further optimized by using an S shaped guide to obtain a steeper cut-off at short wavelengths. An additional advantage of the curvature stems from the reduction of the gamma dose in the guide hall, as the gamma rays are not reflected inside the guide but absorbed in their outer shielding. This decreases the experimental background as well.

By replacing the Ni–Ti coating with magnetic layers, the neutron guides can serve to produce a polarized neutron beam. In this polarizing section, the neutrons have to be reflected at least once. In case the neutron guide is too short for the given curvature, additional blades inside the guide can ensure this condition. An external guide field from permanent magnets hinders the depolarization of the beam.

The neutron guide hall at the FRM-II is served by the beam tube SR1 containing six neutron guides with cross-sections from 60×120 mm^2 to 50×170 mm^2 (Fig. 4). Further downstream, these guides are split horizontally and vertically to provide optimized beam cross-sections for the different instruments. For one instrument, the reflectometer with horizontal sample alignment (optimized for liquid surfaces), a cross-section of 170×14 mm^2 (width by height), is requested. Normal widening and focusing would have led to a significant loss in intensity. Therefore, a toroidal guide was constructed which turns a 14×170 mm^2 section around 90° at a length of 36 m. At the end of the guide, a horizontal focusing section further increases the neutron flux at the sample position.

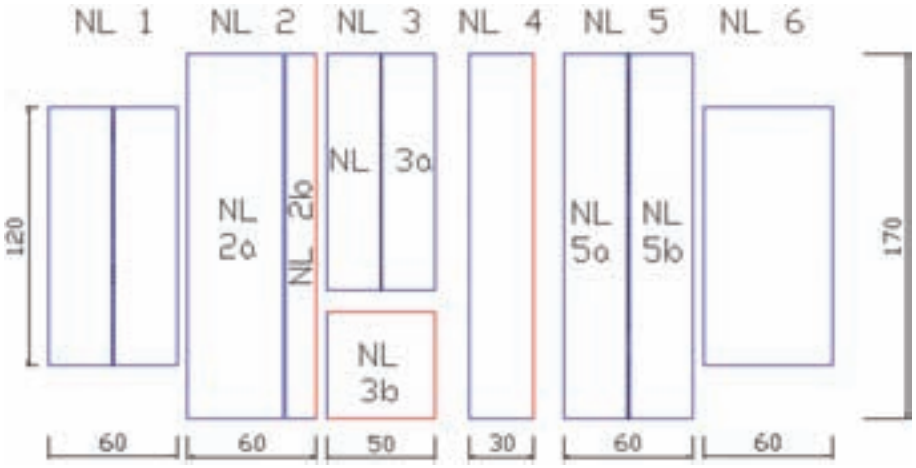

FIG. 4. Cross-sections of the neutron guides for the neutron guide hall, as seen in the neutron flight direction from beam tube SR1 to the guide hall. The initial six neutron guides are named 'NL' from the German 'Neutronenleiter'. The dimensions are given in mm.

In total, 650 m of neutron guides are installed at the FRM-II using 220 m^2 of mirror surface, out of which 146 m^2 are covered with a supermirror coating.

4. CONCLUSION

At the new FRM-II, an optimized system of beam tubes and neutron guides has been designed to meet the different requirements of the scientific users. Some of the beam tubes supply thermal neutrons, some are linked to cold or hot sources to provide cold (even ultracold) or hot neutrons, respectively, one tube supplies fast fission neutrons, and two tubes are designed to provide beams of positrons or fission product ions with unprecedented intensity. A system of 650 m of neutron guides multiplies the experimental area accessible with thermal and cold neutrons, reduces the background originating from fast neutron and gamma radiation, and can even deliver polarized neutrons.

In conclusion, the advanced design concept of the new reactor, in combination with the numerous beam tubes and neutron guides which have all been optimized a priori for their different fields of application, render the FRM-II an outstanding and most attractive source for all kinds of scientific investigations with neutrons. In addition to these devices dedicated to neutron beam applications, six facilities (not described in this paper) have been installed to irradiate samples in a high flux of thermal neutrons.

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6.5. NEUTRON BEAM SYSTEM AT THE CHINA ADVANCED RESEARCH REACTOR

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1. INTRODUCTION

The China Advanced Research Reactor (CARR) is designed to be a safe, reliable multipurpose research reactor with high performance. It will play an important role in nuclear science, research and development in China in the 21st century. To achieve that aim, the CARR will provide a powerful capability for conducting a great deal of fundamental and engineering research applications, comprising material science, life science, environmental science, research in the fields of nuclear physics and chemistry, and other important areas. After obtaining the construction licence, work on the CARR project's civil construction began in August 2002 and is ongoing. The reactor is scheduled to reach its first criticality in October 2006.

The CARR project consists of a reactor, its associated auxiliary systems and experimental facilities for utilization. The complex CARR structure comprises seven major components: a flow guide tank, a heavy water tank, a core vessel, a decay tank, pipes of the primary loop, nine horizontal beam tubes and 21 vertical holes. It is a tank-in-pool and inverse neutron trap type with a thermal power of 60 MW. The fuel assemblies are of MTR plate type applying U_3Si_2–Al fuel (U_3Si_2 dispersed in Al) with 19.75 wt% ^{235}U. The maximum unperturbed thermal neutron flux is expected to be 1×10^{15} n·cm^{-2}·s^{-1} in the central region of the core and 8×10^{14} n·cm^{-2}·s^{-1} in the heavy water reflector. The core has an active height of 850 mm and an equivalent diameter of 400 mm. The core is surrounded by a heavy water tank of 2.5 m height and 2.2 m outer diameter. This large D_2O reflector around the core serves to prevent fast and epithermal neutrons escaping, and thus provides sufficient irradiation space. Slightly pressurized H_2O serves as the coolant and core internal moderator. A large quantity of undermoderated neutrons leaks from the core to be further moderated in the heavy water reflector. The reactor pool, which houses the core and the reflector, has a diameter of 5.5 m and a depth of 15 m. It is lined with 16 mm thick stainless steel sheets and will be filled with demineralized water. Figure 1 shows the CARR structure schematically.

One of the major fields of utilization to be covered by the CARR is neutron scattering. Moreover, radioisotope production, neutron activation

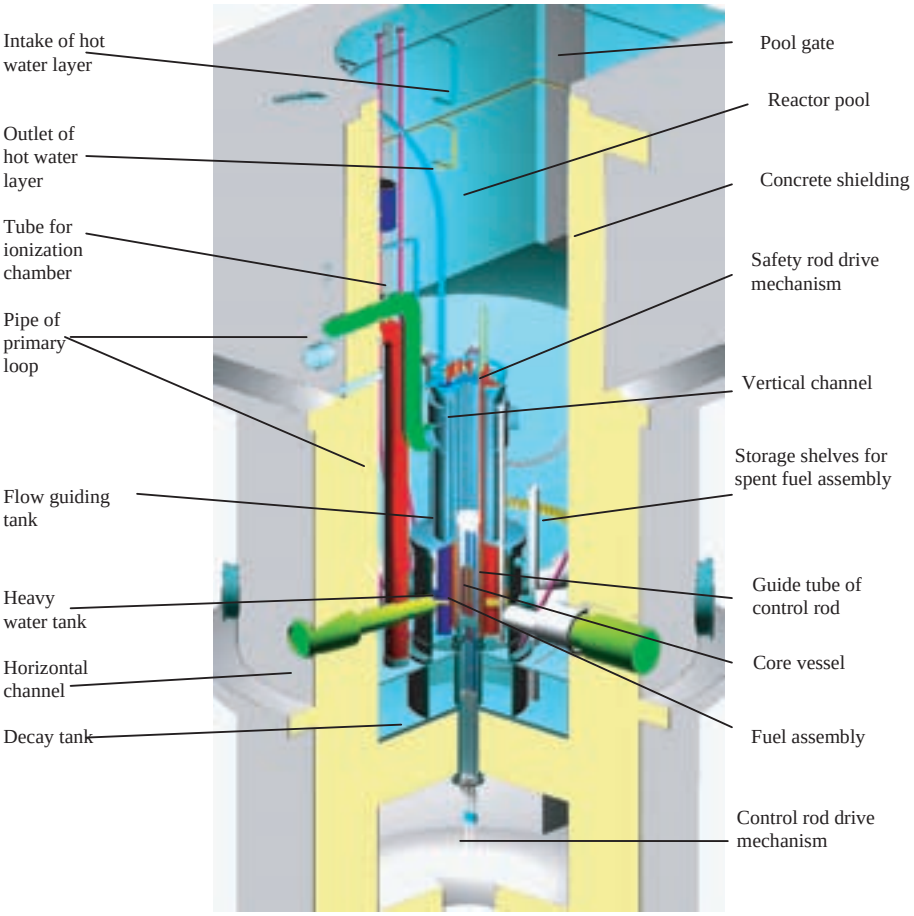

FIG. 1. Schematic representation of the structure of the CARR.

analysis (NAA), neutron transmutation doping (NTD) of silicon, nuclear fuel and material testing, neutron radiography and technical training are other tasks of the CARR plant.

2. HORIZONTAL BEAM TUBE CHARACTERISTICS

The nine horizontal beam tubes of the CARR are all of the tangential type; they are named 'HT' and numbered from 1 to 9 counterclockwise. The n-beams they will deliver are widely originating from the thermal neutron peak in the heavy water. However, there are four secondary sources in the reflector tank of the CARR. HT1 points to a cold neutron source, HT7 to a hot neutron

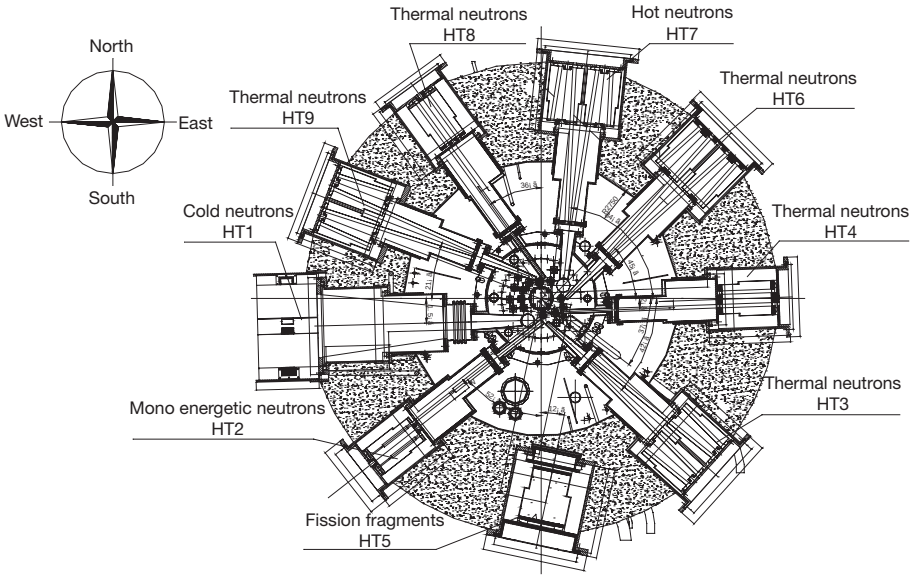

FIG. 2. Layout of the CARR's horizontal beam tubes.

source, HT2 to a mono-energetic neutron source and HT5 to a fission fragment source. The rest, consisting of HT3, HT4, HT6, HT8 and HT9, use the thermal neutron peak. Figure 2 shows the layout of the CARR's horizontal beam tubes and Fig. 3 shows the cross-sectional view of a typical beam tube realized as HT3/HT6/HT7/HT9. Each beam tube has its own characteristics for its special experimental purpose. The position and the geometry of the beam tubes have been optimized by MCNP4A calculations. These calculations show the total loss of reactivity caused by all experimental installations in the heavy water tank to be 2.99% $\Delta k/k$ at the beginning of service life. The horizontal beam tubes with some of their characteristics are listed in Table 1. All noses of beam tubes are made of 6061–Al. The wall thickness at the front face and the side walls is 5 mm and the tubes have a conical shape. All beam tubes are filled with helium. The vertical hole for the cold neutron source (CNS) has an inner diameter of 260 mm and is located at the position of maximum thermal neutron flux of about 5.5×10^{14} n·cm^{-2}·s^{-1} at the level of the nose of beam tubes.

3. NEUTRON SCATTERING EXPERIMENTAL FACILITIES

Figure 4 shows the arrangement of the ground level of the reactor building of the CARR and of the neutron guide hall. The reactor building is about 34 m high and has a square cross-section of 34 m × 34 m inner dimension.

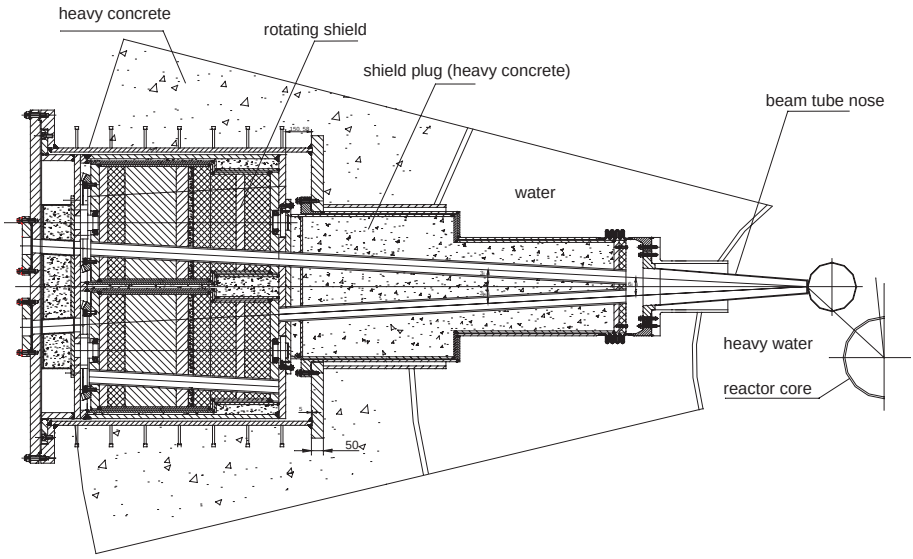

FIG. 3. Cross-section of a typical beam tube (for HT3, HT6, HT7 and HT9).

The confinement of the reactor building with a leakage rate of 2.5 vol.%/d is the ultimate barrier against an uncontrolled release of radioactive fission products to the environment. One side of the reactor building is connected to a neutron guide hall, which is 60 m × 36 m and 12.5 m high.

The vertical holes and tubes and the horizontal beam tubes with associated utilization facilities will be installed in the core and the heavy water

TABLE 1. HORIZONTAL BEAM TUBE CHARACTERISTICS

Beam tube		Nose dimension (mm)	Number of neutron beams	Elevation from core centre (mm)	Shutter type
HT1	Cold neutron	200 × 200	4	100	Lifting
HT2	Multifiltration neutron	100 × 100	1	100	Rotating
HT3	Thermal neutron	70 × 140	2	100	Rotating
HT4	Thermal neutron	70 × 140	1	100	Rotating
HT5	Long tangential	∅ 170	1	+400	Rotating
HT6	Thermal neutron	70 × 140	2	100	Rotating
HT7	Hot neutron	70 × 140	2	100	Rotating
HT8	Thermal neutron	70 × 140	1	100	Rotating
HT9	Thermal neutron	70 × 140	2	100	Rotating

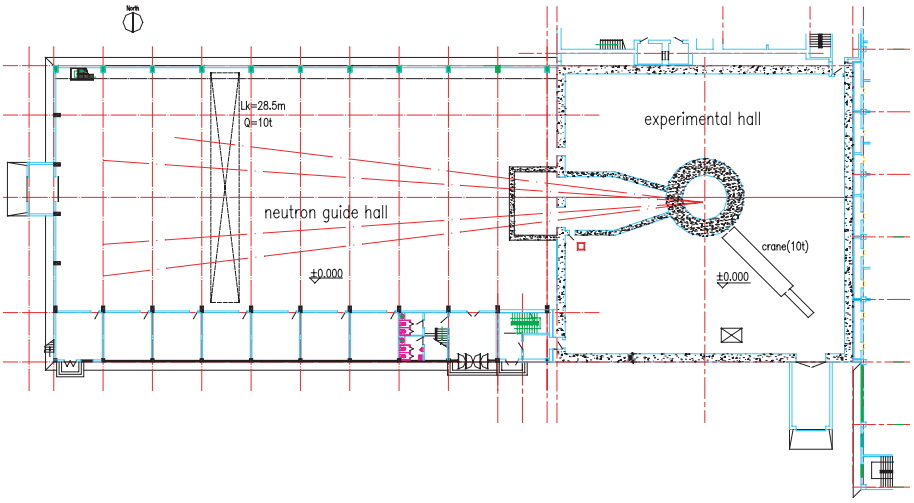

FIG. 4. Ground level of the reactor building and neutron guide hall of the CARR.

reflector of the CARR. The CNS and the hot neutron source are essential facilities in this concept. Concerning the instruments for neutron scattering, many of those installed at the Heavy Water Research Reactor (HWRR) of the CIAE will be upgraded and moved to the CARR for their future utilization. Thus, the instruments planned for the CARR comprise one triple axis spectrometer, one four circle spectrometer, one time of flight spectrometer, one rebuilt texture neutron powder diffractometer, one new multipurpose hot neutron spectrometer and one new high resolution powder diffractometer, all of which will be placed inside the reactor experiment hall. One horizontal type polarized neutron reflectometer using cold neutrons from a ^{58}Ni (or super-mirror) guide will also be constructed and placed in the neutron guide hall of the CARR.

4. COLD AND HOT NEUTRON SOURCES AND NEUTRON GUIDES

The CNS is under design and will be installed in the CNS hole and tube in the heavy water reflector. The temperature of the liquid hydrogen in the CNS will be around 20 K. The gain factor of the cold neutrons should be higher than ten in the range of 4–20 Å. The hydrogen will cool down the thermal neutrons and supply cold neutrons which will be guided outwards along the cold neutron beam tube (HT1) equipped with neutron guides, then reaching the instruments located in the neutron guide hall. Four cold neutron guides, each having a beam

cross-section of 30 mm × 150 mm, are installed for transporting cold neutrons to the neutron guide hall. The natural Ni and ^{58}Ni (or supermirror) mirrored guide tubes will be constructed soon after the CARR becomes critical. Two additional neutron guides are planned to be installed later.

The hot neutron source is not under design and not scheduled for the time being. Installation of it is planned in the HNS hole of the reflector tank. The temperature of the graphite blocks of the hot neutron source will be about 2000 K. The gain factor of hot neutrons (0.4–0.8 Å) is foreseen to be about 15 at 0.5 Å. Two hot neutron beams will be provided.

6.6. NEUTRON BEAM DESIGN AT THE UNIVERSITY OF CALIFORNIA, DAVIS, McCLELLAN NUCLEAR RADIATION CENTER

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1. INTRODUCTION

The McClellan Nuclear Radiation Center (MNRC) is centred on a 2 MW Triga research reactor, built by General Atomics (GA) for the United States Air Force (USAF), and originally dedicated to state of the art neutron radiography of aircraft structures. The MNRC was operated by the USAF from January 1990 to February 2000, performing this function as well as a variety of research projects for the US Department of Defense, industry and universities. The USAF closed the military base in January 2001, however, the MNRC was maintained and transferred to the University of California at Davis (UCD) in February 2000. The UCD/MNRC has become a world-class leader in neutron imaging; the heart of this state of the art neutron imaging is the design of the four beam tubes described in this paper.

2. NEUTRON BEAM TUBES

2.1. Beam tube description

The neutron beam system (Fig. 1) consists of a beam tube, aperture, shutter and beam stop for each of the four radiography bays. This system features an optimized 'source end' design of the graphite reflector, together with a beam extraction port to ensure maximum thermal flux and maximum uniformity across the source plane. The resulting beams have an L/D ratio ranging from 60 to 400, and the neutron fluxes range from 1.5×10^6 to 1.7×10^7 $n \cdot cm^{-2} \cdot s^{-1}$.

All four beams are capable of simultaneous operation. The apertures, located at the edge of the reactor graphite reflector, are replaceable in all beam tubes. Aperture change out is accomplished by remotely removing a section of the reactor reflector, replacing the in-tank tube section with one having the desired aperture and replacing the reflector section. The in-tank beam tube

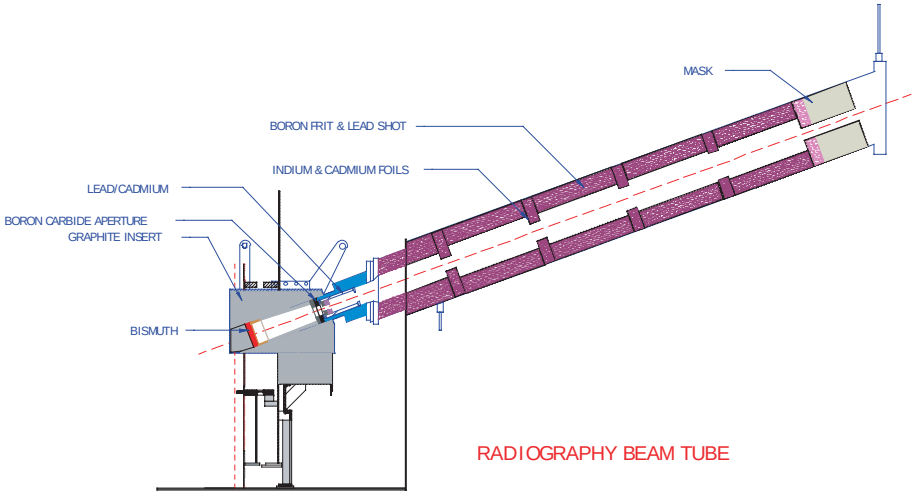

FIG. 1. Radiography beam tubes at the UCD/MNRC. The major components of the beam tube are shown; the graphite insert section, the tank wall section and the bulk shielding section.

section also provides a means of effecting future beam upgrades (e.g. cryogenically cooled filters and neutron scattering material in the beam line) by providing space along the beam tube for such upgrades.

Two shutters are installed at the output end of each beam tube. The first is a massive 'biological' shutter with an 'open' position incorporating beam masks and a 'closed' position to allow safe personnel access to the exposure bay. Due to the weight of these shutters, they move very slowly and therefore are not used for timing radiographic exposures. A second, fast operating, lightweight shutter, attached to the biological shutter, allows the thermal neutron beam to be controlled for accurately limiting film exposures. Beam stops are provided for each bay to minimize the thermal neutron albedo and to control neutron background outside the exposure room.

The four beam tubes spaced at 90° intervals around the base of the reactor tank penetrate the reactor graphite reflector and provide a direct path for neutrons to each of the radiography bays (Fig. 2). The beam tubes are positioned tangentially with respect to the reactor core and are inclined (20° or 30°) with respect to the horizontal plane (Figs 2, 3 and 4).

2.2. Beam tube construction

Each of the four beam tubes is made up of three major sections: the in-tank section, the tank wall section and the reactor bulk shielding section.

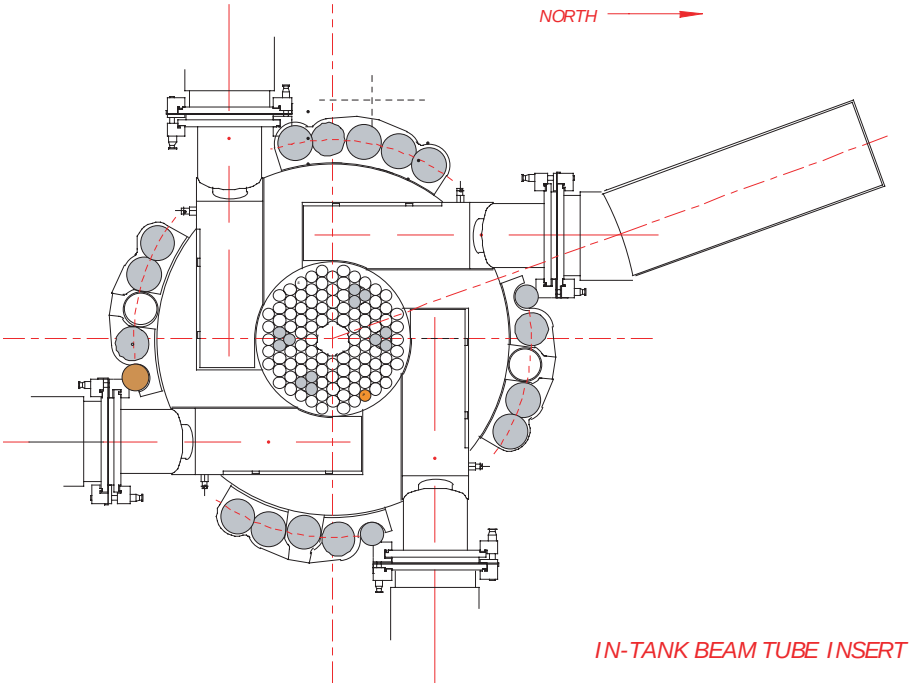

FIG. 2. Arrangement of the four tangential beam tubes and the beam tube inserts at the MNRC. The beam tube ends in the graphite reflector are seen surrounding the Triga core. The tank wall section and the connections between insert and tank wall sections are also shown.

The in-tank section of the beam tube contains a replaceable aperture, consisting of a neutron absorbing material such as boron carbide. This arrangement, housed in a watertight aluminium container, is shown in Fig. 4. This section is the most important part of the beam tube since it is part of the reactor core reflector, and also functions to define a source of neutrons, as well as purifying and shaping the neutron beam. The in-tank section consists of a large graphite block with a 152.4 mm (6 in) diameter hole bored along the beam centre line. The key elements within the bore are a lower graphite section (which serves as a source of neutrons), a bismuth crystal which attenuates gamma rays, and a boron carbide aperture, which limits the source of and shapes the beam. A graphite tapered sleeve below the aperture is also located in the bore. All of these components are contained in an aluminium housing that transforms into a 317.5 mm (12.5 in) diameter circular section with bellows and flange along with a bolt-on faceplate. A lead coated metal O ring forms the seal between the flange and the faceplate. The faceplate and the in-tank assembly have two tube fittings that connect to the helium supply or a vacuum

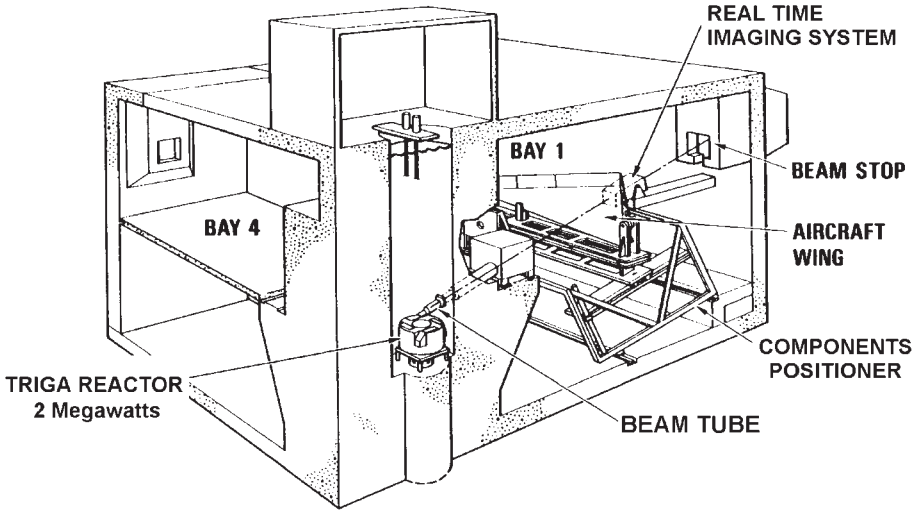

FIG. 3. The arrangement of the MNRC showing the Triga reactor, the massive concrete monolith and the beam tube. The facility was purpose built for the radiography of aircraft parts and so large radiography bays capable of handling such pieces surround the reactor. Note also that the beam tube is 20º from horizontal. This design enabled the reactor core to be built below ground level, while the radiography bays themselves are above grade.

system. The entire unit is watertight and can be removed remotely and replaced from the tank top. The assembly mates with the tank wall section of the beam tube to provide a water free path within the reactor tank for the neutron beam.

The tank wall section of the beam tube consists of a 317.5 mm (12.5 in) diameter pipe welded to the tank wall and a special flange welded to the core end, as shown in Fig. 4. An annular aluminium container filled with 50 vol% boron frit and 50 vol% lead shot is located within the pipe section. The internal surface of the aluminium container is coated with gadolinium paint. The tank wall section does not penetrate the tank wall and serves as a watertight container when assembled, as well as both a neutron and gamma shield. The gadolinium helps prevent scattered neutrons from re-entering the beam. This section is also helium filled.

A two piece bolted clamp holds the in-tank and tank wall section flanges together. The clamp bolts can be remotely removed and replaced from the tank top.

The bulk shielding section of the beam tube extends from the outside of the tank wall to the radiography bay (see Fig. 1). The housing for this section is a 508 mm diameter steel pipe and bellows assembly embedded into the

Materials list

1. Graphite end plug
2. Bismuth crystal
3. Aluminium spacer
4. Graphite sleeve
5. Boron-carbide aperture
6. Lead-cadmium shield
7. Lead-cadmium sleeve
9. Reflector insert body
9. Bellows
10. O-rings
11. Aluminium sleeve
12. Boron carbide aperture (2 MW)
13. Lead aperture (2 MW)

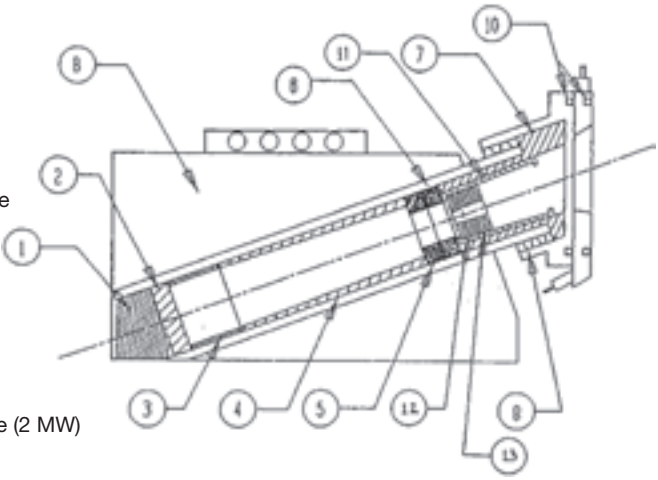

FIG. 4. The in-tank section of the neutron beam tubes at the MNRC showing the reflector insert and its various internal components. The entire section is housed in an aluminium container.

concrete. The bellows assembly provides flexibility for expansion and contraction. The pipe is in close proximity, but is not physically attached to the tank wall. Within the housing are a number of annular shaped aluminium containers filled with 50 vol% boron frit and 50 vol% lead shot. Their primary function is to provide neutron and gamma ray shielding, help shape the beam, and prevent scattered neutrons from re-entering the beam, thus maintaining beam purity. The annular section next to the tank wall is permanently installed. The remaining annular boron–lead filled sections can be removed and replaced with units of different internal diameters if the beam cross-section (i.e. size) needs to be changed. The two annular containers at the exit of the beam tube into the radiography bay contain 100% boron frit and a commercial shielding material (MHL-277), respectively. These elements are the final beam shapers and both are excellent neutron shields.

Both assemblies can be replaced from the radiography bay. The inner surfaces of all containers in this section are also coated with gadolinium paint.

The ends of the beam tubes are closed with aluminium plates. All three sections of the beam tube are equipped with gas lines that are attached to the helium supply or the evacuation system and are used to prevent degradation of the neutron beam. As a consequence, there is very little ^{41}Ar formed in these beam tubes because of the near complete absence of air.

3. NEUTRON RADIOGRAPHY SYSTEMS

Capabilities of the neutron radiography systems at the MNRC include real time, film, and filmless and computer tomography.

The real time system consists of large positioning tables that will handle objects up to 3600 mm tall and 10 200 mm long, with a mass of up to 2272 kg. This system uses a Thompson tube coated with gadolinium oxysulphide and a silicon intensified tube (SIT) camera. The frame rate is 30 frames/s. The nominal resolution of this system is 50 μm (0.0020 in) at an L/D ratio of 125.

Film radiography is done in the direct mode using gadolinium converter screens and high resolution X ray film. The nominal resolution of this system is 5 μm (0.0002 in) at an L/D of 275. The film system L/D ratio can be changed from an L/D of 60 to one of 400.

The filmless radiography system uses scintillator plates to produce the image, and can produce a comparable radiograph in approximately 50% of the time it takes for conventional film. Another advantage is the lack of processing needed for the filmless system. The film reader reads the scintillator plate and converts the light image to a digital radiograph. The nominal resolution of this system is also 5 μm (0.0002 in) at an L/D ratio of 275.

The neutron computer tomography (CT) system consists of a charge coupled device (CCD) camera with 23 μm pixels × 23 μm pixels, a scintillator screen and an object turntable. The CT system is capable of imaging objects up to 200 mm (8 in) in height and, depending on the material, 50–75 mm (2–3 in) thickness. The nominal resolution of this system is 250 μm with an L/D of 125.

Further details concerning imaging system performance and the component positioning systems can be found in Refs [1, 2].

4. LIMITATIONS AND CONSTRAINTS IN THE DESIGN OF BEAM TUBES

There are two limitations present in the design that place limitations on the efficient operation of the facility. The original USAF specification required that the reactor tank be only 2.3 m in diameter. While this size may be adequate for most low to medium power research reactors which do not require a large number of in-core equipment manipulations, it resulted in a reactor tank at the MNRC facility that is too small to work efficiently on the multitude of activities that are now conducted using this reactor. For example, in-tank fixtures tend to be hard to position, as space is limited with the variety of experiments present in the core. This also results in making the removal and replacement of fixtures a complex task.

The second design restriction evolves from the below ground placement of the reactor core structure in the reactor pool, done so for safety reasons due to the siting of the facility adjacent to an active military base runaway. This resulted in having the beam tubes at an angle of 20–30° from the horizontal. Even though this does not have an impact on the large majority of radiography work to be done, it does make other experiments, such as fluid flow studies, very difficult.

5. CONCLUSION

The UCD/MNRC has developed into a world-class neutron imaging system facility during its 13 years of operation. The resulting neutron beam has produced neutron images of the highest quality [1–3]. The beam tubes have performed their function for the last 13 years without any major operational or maintenance problems.

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Part 7

SECONDARY NEUTRON SOURCES

7.1. HISTORICAL OVERVIEW

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1. INTRODUCTION TO SECONDARY SOURCES

For more than four decades, experimentalists and research reactor designers together have planned and built secondary neutron sources at research reactors to generate neutrons with energies different from the thermal spectrum of the reactor core. The methods applied are either to:

— Prevent the moderation in the moderator, i.e. to use or to extract fast (fission) or epithermal neutrons prior to their slowing down;
— Moderate the thermal neutrons in the reflector region with moderators of different temperature, such as hot graphite — hot neutron source (HNS) — or cold hydrogen rich materials, such as liquid H_2 and D_2, methane, etc. — cool neutron sources (CoNSs) — and cold neutron sources (CNSs) or even cold helium or D_2 ice, the latter especially for ultracold neutron sources (UCNs).

Aside from those two methods, nuclear reactions are applied to shift neutron spectra or to generate neutrons of a certain energy, such as secondary fission sources (fission converters at MITR or FRM-II) or nuclear reactions, such as ^{6}Li to get 14 MeV neutrons (e.g. at the Triga in Vienna).

2. NON-NEUTRON SOURCES

Supplementary to the secondary neutron sources, the free neutrons from the fission processes of the core are used to generate other particles and to establish beams of such particles. Examples for those supplementary secondary sources are:

— Positron sources;
— Fission fragment sources.

The resulting beams of electrically loaded particles can be bent by magnets and as such can be easily guided into any direction or to any target.

3. COLD NEUTRON SOURCES

The main secondary sources these days are the HNS and the CNS, the latter having by far the highest importance and most widespread application. Cold neutrons became very popular in neutron research due to the extended field of research in addition to the steadily improved ability to guide and even focus them by neutron guides and lenses.

Consequently, Part 6 of the Compendium predominantly focuses on CNS and their various realizations; some examples of HNS are also mentioned.

Examples of CNS based on hydrogen are compiled in Table 1,[1] linking the CNS to the research reactor it was or is used at. The oldest CNS was installed in 1957 at the BEPO research reactor at Harwell, United Kingdom.

The liquid CNSs as listed in Table 1 mostly apply the thermosiphon principle, i.e. the liquid hydrogen in the cold moderator chamber boils due to heat from the bombardment by neutrons (mostly slowed down in the hydrogen) and by gamma radiation (mostly absorbed in the structural material of the moderator chamber). The hydrogen gas from boiling rises to reach a condenser where a surface, cooled by cold helium, cools the hydrogen so that it returns to the liquid phase and drops or flows again into the moderator chamber. The amount of bubbles from the boiling reduces somewhat the amount of hydrogen in the cold moderator chamber. To avoid such bubbling, a one phase (i.e. only liquid hydrogen) thermosiphon was used at the FRM.

When the supply of cold helium to the heat exchanger at the condenser is interrupted, the moderator chamber is emptied from liquid hydrogen rather fast and its integrity is threatened by heating from gamma radiation. This status mostly requires a fast shutdown of the research reactor and a break of the isolating vacuum to enable cooling of the moderator chamber. However, modern cryogenerators, which provide the cold helium at about 16–20 K, are rather reliable in delivering good operational records for CNSs.

The alternative principle for applied CNSs with hydrogen at around 20 K is to use gaseous supercritical hydrogen, which is kept at a pressure of 14 bar to avoid liquefaction. This principle provides the same density as liquid hydrogen and delivers an easy circulation of the cold media by fans to remove the heat generated in the cold hydrogen plus the cold moderator chamber in which it is contained. This circulation provides an advantage on the other hand, i.e. one

[1] All the tables shown in this paper are composed from data of openly published literature and may contain errors in the figures which result from publishing a different status of the secondary sources in design and even in realization, in terms of later modifications (the author would be interested in corrections where appropriate).

TABLE 1. EXAMPLES OF HYDROGEN BASED CNS

RR name	Location	Country	Power MW (th)	Moderator	Insertion; cooling	Environment	Operating since	Status/ remarks	Constructor	Cryopower W[a]
BEPO	Harwell	United Kingdom		lq. + s H_2 + others	Horizontal; in-core	Graphite	1956	End of operation 1961	AERE	
BR 1	Mol	Belgium	4	lq. H_2 + others	Vertical; in-core	Graphite	1960	Source removed	SCK CEN	
DIDO (1)	Harwell	United Kingdom	15	lq. H_2/D_2	Horizontal; direct cooling	D_2O	1960	End of operation 1965	AERE	42/–
EL 3	Saclay	France	17	lq. H_2/D_2	Horizontal	Graphite	1962	Reactor shutdown 1979	CEA	45/80
SILOETTE	Grenoble	France	0.1	lq. D_2/H_2		D_2O	1964	Test set-ups only; tests for RHF	ILL	
FiR 1 (1)	Espoo	Finland	0.25	H_2/CH_4 (78K)	Horizontal	Graphite	1964	End of operation 1968		Low
HERALD (1, 2)	Aldermaston	United Kingdom	6	lq. $H_2 + D_2$	Vertical; direct cooling	D_2O	1966	Reactor shutdown 1988	AERE	200/500

TABLE 1. EXAMPLES OF HYDROGEN BASED CNS (cont.)

RR name	Location	Country	Power MW (th)	Moderator	Insertion; cooling	Environment	Operating since	Status/ remarks	Constructor	Cryopower W[a]
DIDO (2)	Harwell	United Kingdom	27	lq. H_2	Horizontal; direct cooling	D_2O	1967	Reactor shutdown	AERE	
FR 2	Karlsruhe	Germany	44	lq. H_2	Horizontal; direct He cooling	D_2O	1969	Reactor shutdown	KFK	50/–
FRJ 2 (1)	Jülich	Germany	15–25	lq. H_2	Horizontal; thermo-siphon + cell cooling	D_2O	1969	Replaced by FRJ 2 (2)	KFA	1000/1500
FiR 1 (2)	Espoo	Finland	0.25	lq. H_2		Graphite	1972			
JEEP 2	Kjeller	Norway	2	lq. H_2	Thermo-siphon	D_2O	1972	In operation	Institute for Energy Technology (IFE)	60
RHF (1)	Grenoble	France	57	lq. D_2	Vertical; thermo-siphon	D_2O	1972 replaced 1977	In operation	ILL	6500
DR 3	Ris	Denmark	10	sc. H_2	Horizontal; forced convection	D_2O	1975	Reactor shutdown	RNL	620

TABLE 1. EXAMPLES OF HYDROGEN BASED CNS (cont.)

RR name	Location	Country	Power MW (th)	Moderator	Insertion; cooling	Environment	Operating since	Status/ remarks	Constructor	Cryopower W[a]
HFBR	Brookhaven	USA	60	lq. H_2	Horizontal; direct He cooling	D_2O	1980	Reactor shutdown	BNL	700/3500
WWR-M	Gatchina	Russian Federation	18	lq. H_2	Vertical; direct cooling	Beryllium	1980	Out of operation since 1985	LNPI/	300/600
ORPHEE (1, 2)	Saclay	France	14	lq. H_2 2 sources	Vertical; thermo-siphon	D_2O	1980	In operation	CEA + TA	2 × 700/ 2 × 500
FRJ 2 (2)	Jülich	Germany	23	lq. H_2	Horizontal; thermo-siphon + cell cooling	D_2O	1985	In operation	KFA/FZJ	
WWR-M	Gatchina	Russian Federation	18	lq. D_2 (60%) + H_2 (40%)	Vertical; liquid phase thermo-siphon	Beryllium	1986	Out of operation since 2000	LNPI/ PNPI	1800/4000
KUR	Kyoto	Japan	5	lq. D_2	Inclined thermo-siphon	Graphite	1987	In operation	Kyoto University	100/240

TABLE 1. EXAMPLES OF HYDROGEN BASED CNS (cont.)

RR name	Location	Country	Power MW (th)	Moderator	Insertion; cooling	Environment	Operating since	Status/ remarks	Constructor	Cryopower W[a]
RHF (2)	Grenoble	France	58	lq. D_2 + 10% H_2	Horizontal; forced convection	D_2O	1987	In operation		3000
FRG 1	Geesthacht	Germany	5	sc. H_2	Vertical; forced convection	Beryllium	1988	In operation	INTER ATOM	1700/2300 after core compact
FRM	Garching	Germany	4	lq. H_2	Vertical; one phase thermo-siphon	Be/H_2O	1988	Reactor shutdown	TUM/ Linde	700
HWRR	Beijing	China	10/15	lq. H_2	Horizontal; thermo-siphon		1988	Out of operation since 1990	IAE/CEA	
JRR-3M	Tokai Mura	Japan	20	lq. H_2	Vertical; thermo-siphon	D_2O	1990	In operation	NKK + TA	350
BER-II	Berlin	Germany	10	sc.H_2	Horizontal; forced convection	Beryllium	1992	In operation	INTER-ATOM	1900
MURR	Columbia, MO	USA	30	lq. H_2	Horizontal	Beryllium	1994		University of Missouri	

TABLE 1. EXAMPLES OF HYDROGEN BASED CNS (cont.)

RR name	Location	Country	Power MW (th)	Moderator	Insertion; cooling	Environment	Operating since	Status/ remarks	Constructor	Cryopower W[a]
NBSR (2)	Gaithersburg	USA	20	lq. H_2	Horizontal; thermosiphon	D_2O	1995	Replaced by NBSR (3)	NIST	800 / 2200
WWR-M	Gatchina	Russian Federation	18	Solid D_2 (SDS)	Horizontal; direct cooling	Graphite + H_2O	1999	Temporarily out of operation since 2001	LNPI/ PNPI	60/150 at 5K
WWR-SM	Budapest	Hungary	10(20)	lq. H_2	Horizontal; direct cooling	Beryllium	2000	In operation	KFKI/ PNPI	200/300
NBSR (3)	Gaithersburg	USA	20	H_2	Horizontal; thermosiphon	D_2O	2002	In operation	NIST	1200/ 2200
FRM-II	Garching	Germany	20	lq. D_2 (+ 5% H_2)	Inclined vertical thermosiphon	D_2O	2003	Under licensing	TUM/ Linde	5000
HFIR	Oak Ridge	USA	100 (85)	sc. H_2	Horizontal; forced convection	Beryllium	2004	In assembly	ORNL	3000/ 3800

TABLE 1. EXAMPLES OF HYDROGEN BASED CNS (cont.)

RR name	Location	Country	Power MW (th)	Moderator	Insertion; cooling	Environment	Operating since	Status/ remarks	Constructor	Cryopower W[a]
RRRP	Lucas Heights	Australia	20	lq. D_2 sub-cooled	Vertical; forced convection	D_2O	2005	Under design	PNPI (INVAP)	4300/ 5000
HANARO	Daejeon	Republic of Korea	30	lq. H_2/ D_2	(Vertical, thermo-siphon)	D_2O	2005	Under design	KAERI	1000/–
PIK	Gatchina	Russian Federation	100	lq. H_2 or D_2	Horizontal; single phase thermo-siphon	D_2O	2006	Under design	LNPI/ PNPI	5600/6000
PIK	Gatchina	Russian Federation	100	lq. D_2	Two phase thermo-siphon	D_2O	2006	Under testing	LNPI/ PNPI	5600/ 11 000
CMRR	Beijing/ Mianyang	China	20	lq. H_2 (sub-cooled)	Vertical; single phase thermo-siphon		2005	Under design	INPC/ PNPI	1500
CARR	Beijing	China	60	lq. H_2	Vertical	D_2O	2007	Under design	CIAE	3500

[a] The figures in this column give either the heat to be removed from the cold moderator (left figure) or the cryopower installed at the source (right figure). In many cases, the literature supplies only one of two, and frequently it is not clear which of them is meant.

does not need to shut down the research reactor in case the cryoplant fails to supply cold helium for the primary heat exchanger. The ongoing forced convection transports the heat to an emergency cooling by, for example, liquid nitrogen. This circulation, moreover, makes the source easily independent from the way of installation: horizontal, vertical or inclined. The principle is, however, limited to not too large moderator chambers due to the pressure and the thickness of the walls of the moderator chambers. This restriction will not allow using the principle for D_2 CNS.

Every publication on CNS mentions the risk originating from hydrogen–oxygen mixtures and their possible ignition and explosive character. So, enclosures by containments made from layers or gaps of vacuum and helium all around the hydrogen containing vessels and pipes are mandatory for the hydrogen sources.

4. COOL NEUTRON SOURCES

To avoid the safety issues linked to hydrogen, and to the cost related to such an application, several attempts have been made with other materials, especially those organic compounds which contain many hydrogen atoms, such as methane, mesithylene, propane, ethylene, etc. Water in the form of ice has also been applied. Examples of such sources, which are sometimes called CoNSs, due to their enhanced temperature level (80–100 K), are compiled in Table 2. Their number is distinctly less then those applying hydrogen at around 20–25 K. A major problem is the chemical disintegration of organic compounds under radiation. Thus, CoNSs are mostly applied at low power research reactors.

Such CoNSs frequently use liquid N_2 for delivering the cold. Even cooled metal bars have been applied for cold transfer.

Moreover, normal or heavy water ice (e.g. at NIST, United States of America), or methane ice as a block or as tiny spheres are applied. The latter technology is especially popular at spallation neutron sources.

5. CoNSs AND CNSs AT SPALLATION SOURCES

Since cold neutrons are of importance as probes, the new neutron sources which no longer rely on the fission process, i.e. the spallation neutron sources, also install CoNSs and CNSs around the spallation target, embedded in the moderator surrounding that target. These CoNSs and CNSs partially use the same technology as at the research reactors, but they also modify that

TABLE 2. EXAMPLES OF COOL NEUTRON SOURCES (CoNSs)

RR name	Location	Country	Power MW_{th}	Moderator	Insertion; cooling	Environment	Operating since	Status/ remarks	Constructor	Cryopower watt
ZOE (EL-1)	Fontenay	France		lq. neon	Vertical	Graphite			CEA	
AVOGADRO	Saluggia	Italy	1	lq. propane	Vertical	H_2O	1969			
IRT-M	Moscow	Russian Federation	5	lq. prop. (90 K)	Horizontal	Beryllium	1970	CoNS stopped 1975	Kurchatov Institute	700
Triga Mk II	Ljubljana	Slovenia	.25	CH_4 (100 K)	Horizontal	Water	1971–1973		University of Ljubljana	40
GALILEO GALILEI	Pisa	Italy	5	lq. propane			1974			
DHRUVA	Bombay	India	100	lq. CH_4	Horizontal	D_2O	1986		BARC	
NBSR (1)	Gaithersburg	USA	20	D_2O + 8% H_2O (40–50 K)	Horizontal direct He cooling	D_2O	1987	Replaced by NBSR (2)	NIST	1000
Triga Mk II	Ithaca, New York	USA	0.5	Mesithylene	Horizontal	Graphite	1988–1990		Cornell University	30
HOR	Delft	Netherlands	3	CH_4		H_2O	1990		IRI	
NETL (Triga Mk II)	Austin, Texas	USA	1	Mesithylene (40 K)	Horizontal	Graphite	1991		University of Texas	12/17

TABLE 2. EXAMPLES OF COOL NEUTRON SOURCES (CoNSs) (cont.)

RR name	Location	Country	Power MW_{th}	Moderator	Insertion; cooling	Environment	Operating since	Status/ remarks	Constructor	Cryopower watt
IBR-2M	Dubna	Russian Federation	2/1500 pulsed	Solid CH_4 (20–60 K)		H_2O	1994–1999	Damaged 1994, replaced 1999	FLNP-JINR	104/200

TABLE 3. EXAMPLES OF CNSs AND CoNSs AT SPALLATION NEUTRON SOURCES

SS name	Location	Country	Moderator	Insertion; cooling	Environment	Operating since	Status/ remarks	Constructor	Cryopower watt
KENS-1	Tsukuba	Japan	CH_4 ice (25 K)		Water (polyethylene)	1980			7
ISIS (1)	Chilton	United Kingdom	sc. H_2 (25 K)	Horizontal/ forced convection	Beryllium	1985			300/580
ISIS (2)	Chilton	United Kingdom	lq. CH_4 (100 K)	Forced convection	Beryllium	1985			300
IPNS (1)	Argonne	USA	S + lq. CH_4 (110 K)			1986		ANL	
IPNS (2)	Argonne	USA	lq. CH_4 (108 K)					ANL	

TABLE 3. EXAMPLES OF CNSs AND CoNSs AT SPALLATION NEUTRON SOURCES (cont.)

SS name	Location	Country	Moderator	Insertion; cooling	Environment	Operating since	Status/ remarks	Constructor	Cryopower watt
LANSCE	Los Alamos	USA	lq. H_2 (20 K)	(2 moderators) forced convection	Beryllium	1986		ANL	160
IPNS (3)	Argonne	USA	lq. H_2 (20 K)					ANL	50/300
SINQ	Wrenlingen	Switzerland	lq. D_2 (25 K)	Horizontal	D_2O	1996	In operation	PSI	1505/2200
JSNS	Tokai Mura	Japan					(3 CNSs) under design	JAERI/ KEK	
SNS	Oak Ridge	USA					Under design	ORNL	
ESS	Undecided	Europe	CH_4 balls				Endangered	FZJ	
SINQ	Wrenlingen	Switzerland	Solid D_2	Vertical; direct cool	D_2O	2005–2006	Under design (UCNS)	PNPI/PSI	100/ 300 (at 5 K)

technology, mostly since they are not bombarded and heated up continuously, but by neutron pulses. Table 3 composes some of the CoNSs and CNSs at spallation neutron sources.

6. VERY COLD AND ULTRACOLD NEUTRON SOURCES

The fourth group of sources generating neutrons at low energies is called either very cold neutron sources (VCNSs) or ultracold neutron sources (UCNSs), depending on the temperature of the moderator. Table 4 composes a historical overview on those VCNS/UCNS installations at research reactors. The rather low intensity of the neutron population at those energies has limited their application, but new designs at high performance research reactors may soon change that limitation.

Already in 1982, at the RHF at Grenoble, France, liquid helium was used to slow down neutrons to very low energies (ultracold neutrons). The FRM-II at Garching, Germany will intensify the attempts using subcooled D_2 ice to get large numbers of ultracold neutrons to use them as a UCN beam. Parallel to the RHF at the WWR-M at Gatchina, Russian Federation, two ways to get UCN were tested: a small liquid H_2 source inside a beryllium reflector directly cooled by liquid helium, and a large He cooled moderator pot using D_2 ice as the slowing down medium.

7. HOT NEUTRON SOURCES

The slowing down of thermal neutrons escaping from the core and existing within the reflector of a research reactor is only one way to achieve a secondary neutron source. Another avenue is also possible, through accelerating the thermal neutrons to energies higher than about 40°C of the classical reflector. All arrangements of materials, which establish a moderator at temperatures distinctly higher than the named 40°C, are called hot neutron sources (HNSs). Figure 1 illustrates the example of FRM-II, where the spectral shifts are generated at and by a CNS (D_2 at 25 K) and an HNS (graphite at about 2500 K).

From Table 5, which lists existing HNSs, it is clear that the importance of neutrons with higher energy than thermal ones cannot compete with the importance of cold neutrons.

TABLE 4. EXAMPLES OF VCNSs AND UCNSs AT RESEARCH REACTORS

RR name	Location	Country	Power MW_{th}	Moderator	Insertion; cooling	Environment	Operating since	Status/ remarks	Constructor	Cryopower watt
WWR-M UCN-1	Gatchina	Russian Federation	16	Be at 20 K + gravity	Vertical; direct He cooling	Beryllium	1968		LNPI	
RHF UCN/ VNC-1	Grenoble	France	57	H_2O gravity		D_2O	1975	Operated until 1985	ILL	
WWR-M UCN-2	Gatchina	Russian Federation	18	H_2, H_2/D_2 + gravity	Vertical; direct He cooling	Beryllium	1980	Out of operation since 1985	LNPI/ PNPI	300/600
WWR-M UCN-3	Gatchina	Russian Federation	18	H_2, D_2/H_2 (10 K) + gravity	Vertical; single phase thermo-siphon	In-core	1986	Temporarily out of operation since 2001	LNPI/ PNPI	1800/4000
IR-8 UCN	Moscow	Russian Federation	8	H_2					RRCKI	
NBSR UCN	Gaithersburg	USA	20	^{4}He superfl. (1 K)		D_2O		Magnetic confinement	NIST	
RHF UCN-2	Grenoble	France	57	D_2 (CNS)	Horizontal	D_2O		Steyerl-Turbine	ILL	
KUR UCN	Kyoto	Japan	5	Superfl. ^{4}He	Horizontal			Steyerl-Turbine		

TABLE 4. EXAMPLES OF VCNSs AND UCNSs AT RESEARCH REACTORS (cont.)

RR name	Location	Country	Power MW_{th}	Moderator	Insertion; cooling	Environment	Operating since	Status/ remarks	Constructor	Cryopower watt
FRM-IIUCN	Garching	Germany	20	D_2 ice	Direct He cooling	D_2O	2005	Test facility at Triga Mainz	TUM	
PIK UCN	Gatchina	Russian Federation	100	Subcooled H_2 (1.5 K)	Horizontal	D_2O	2006	Test at WWR-M	PNPI	5300

TABLE 5. EXAMPLES OF HNSs AT RESEARCH REACTORS

RR name	Location	Country	Power MW_{th}	Moderator	Insertion	Environment	Operating since	Status/ remarks	Constructor
RHF HNS	Grenoble	France	58	Graphite	Vertical	D_2O		In operation	ILL
DHRUVA HNS	Bombay	India	100	Graphite (1600 K)	Horizontal				BARC
ORPHEE HNS	Saclay	France	14	Graphite (1400 K)	Vertical	D_2O		In operation	CEA
FRM-II HNS	Garching	Germany	20	Graphite (2500 K)	Vertical	D_2O	2004	Under licensing	TUM
PIK HNS	Gatchina	Russian Federation	100	He		D_2O	2006	Under design	PNPI

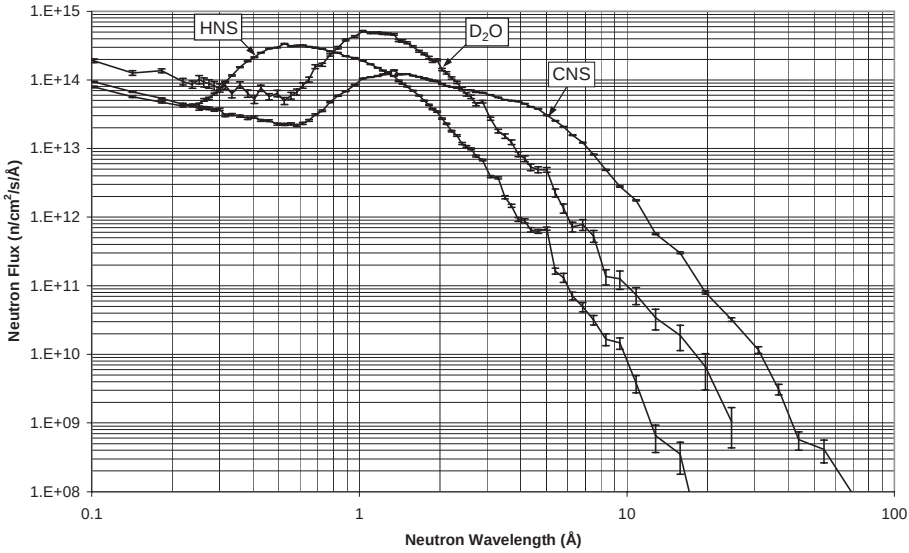

FIG. 1. MC calculated neutron spectra at the heavy water reflector and its built-in secondary neutron sources at the FRM-II at Garching, Germany (courtesy of the Technische Universität München). CNS: liquid D_2 at 25 K; D_2O: thermal moderator at 40C; HNS: graphite at about 2500 K.

7.2. SECONDARY SOURCES

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1. INTRODUCTION

An experimental neutron beam reactor essentially is an intense source of thermal neutrons for carrying out experiments in the fields of:

— Solid state physics (including chemical and biological applications);
— Nuclear physics and possibly neutron physics.

The majority of these experiments are carried out at thermal neutron beam tubes with a minimized background of fast and epithermal neutrons and of gamma radiation.

Moreover, facilities integrated in the reactor exist, which are designed to produce (and then to permit the extraction of) neutrons in well defined energy ranges by means of specific moderators. Essentially, such facilities are:

— Cold neutron sources (CNSs) realized by the insertion of a low temperature moderator operated below 25 K;
— Hot neutron sources (HNSs) realized by the insertion of a high temperature moderator operated at about 2000 K.

This paper illustrates some of the features of those facilities and will use the French secondary sources as relevant examples.

2. GENERAL CONSIDERATIONS ON CNSs

2.1. General design

The purpose of a CNS is to multiply low energy flux (below 5 meV) as existing at a research reactor by a factor of 10–50. This is accomplished by installing a container of liquid deuterium or hydrogen, mostly inside the reactor reflector.

The complete CNS comprises the following elements:

— Cryostat comprising its essential component, the moderator cell that contains the cold gas during operation. This cell is surrounded by the envelope providing both heat insulation under vacuum and the border against the surrounding reactor environment (support of the moderator cell, embedding in the external liquid which is frequently heavy water);
— Condenser where the deuterium or the hydrogen are liquefied by heat transfer with a cold helium circuit;
— Helium refrigerator, consisting of the compressor, the turbine and storage units for the gas in different states between the situations in the cold and hot state;
— Accessories and the devices for pre-operational and operational conditioning of the installation (evacuation and maintenance of the cryogenic vacuum, connection tubing between the various storage units, etc.);
— Facility control and the associated safety equipment.

2.2. Choice of moderator

The moderator fluids are most frequently liquid hydrogen or deuterium and, at higher temperatures, some organic substances.

The dimensions of the moderator cell depend mainly on two parameters:

— Type of moderator used, as in the case of liquid deuterium, the neutrons are thermalized after passing through a cold moderator with a thickness in the order of 350 mm. The use of liquid hydrogen only requires a moderator thickness in the order of 50 mm;
— Dimensions of the neutron guides to be illuminated.

One of the direct consequences of the combination of these parameters is that the liquid deuterium sources are particularly suitable for the use of cells whose general shape is circular, and for the possibility of a simple installation of several neutron beam tubes pointing at that same moderator cell. Such designs require the use of a considerable quantity of deuterium (in the order of 5–20 L in the liquid state, i.e. at the level of the moderator cell).

Liquid hydrogen sources, on the other hand, require a smaller quantity of gas at the liquid state (a few L) and often apply to cells which are flattened or of annular shape. This means that those cells have a more complex structure and are more difficult to manufacture.

These features altogether generally lead to higher total volumes of gas and installed refrigerating power when applying deuterium instead of hydrogen.

2.3. Operating conditions of liquid moderators

Apart from the effect of neutron thermalization, the boiling of liquefied gas at the cell level has two roles:

— Natural circulation of the gas by the thermosiphon phenomenon resulting in return of the gas out of the moderator cell in the gaseous phase;
— Heat transfer medium to dissipate the heat generated in the structure material of the moderator cell and nearby, in-core supply tubes.

Before starting the research reactor, most of the CNSs must be put into a stable operation mode, especially those applying the thermosiphon principle. In general, a minor disturbance to the neutron source operation results in a research reactor power manoeuvre to regain the normal operating values of the CNS. When a major disturbance occurs (operational instability or loss of refrigeration), a shutdown of the research reactor is necessary and often automatically initiated.

In the design of certain liquid hydrogen CNSs,[1] the possibility of decoupling the operation of the research reactor from that of the cold source is provided for. This was the case for the first CNS at the ORPHEE Reactor at Saclay in France. At ORPHEE, the moderating cell was made of refractory stainless steel and thus permitted operation of the research reactor without the CNS in cold operation. A high portion of the heat generated at and around the moderator cell was then dissipated by radiation to the adjacent structures (which were blackened to increase their emissivity). The CNS lower end structures themselves were cooled by the surrounding heavy water of the reflector.

This operating mode was not adopted for the CNSs with annular cells as currently installed at the ORPHEE Research Reactor (see Fig. 1).

2.4. Choice of architectural features

Two main architectures of a CNS may be used.

[1] CNSs applying supercritical hydrogen provide this decoupling by their design (see, for example, papers 7.1., 7.3. and 7.5).

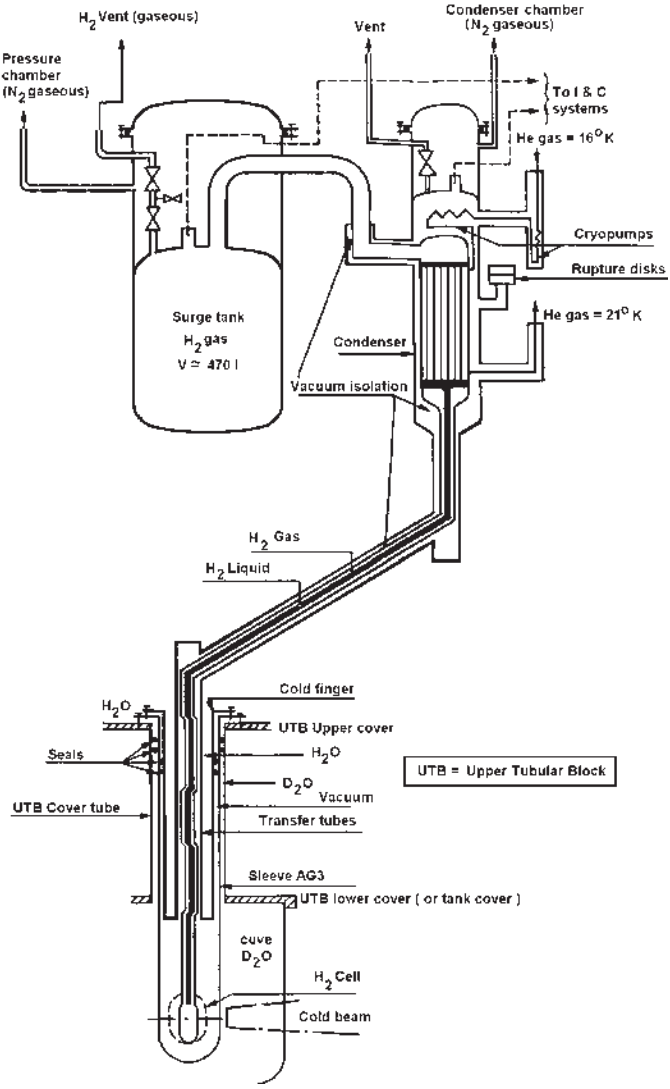

FIG. 1. ORPHEE's vertical CNS.

2.4.1. Vertical type CNS

Under the vertical type architecture, the condenser is placed inside the reactor swimming pool and supplies the moderator cell with liquefied gas by gravity. Boiling occurs in the cell and the gaseous phase returns to the condenser. The installation operates as a natural thermosiphon.

The horizontal beam tubes are arranged radially around the moderator cell. They are completely independent from the CNS facility and distribute the cold neutrons to the experiment zones after passing through the swimming pool wall structures. This independence and the constraints for the dissipation of the heat generated in the structures by the radiation from the core imply, from the viewpoint of design, the need for a layer of water between the envelope confining the cryogenic vacuum surrounding the CNS and the core directed front windows of the neutron beam tubes. This layer of water, reduced to a minimum thickness, constitutes an unavoidable element of disturbance in the vertical type architecture.

2.4.2. Horizontal type CNS

The horizontal type architecture has the advantage that neutrons are leaving the swimming pool undisturbed from its moderator filling. The cold moderator cell is placed directly at the end of the horizontal beam providing the outlet for the cold neutrons towards the experimental zones.

This type of architecture makes it possible to live without a layer of water in the path of the cold neutrons and thus permits the flux to remain undisturbed — avoiding any layers of warmer material.

On the other hand, a significant portion of the tubes for the liquefied gas supply and the return of the gaseous phase are horizontal and traverse long distances inside major flux areas. Thus, that configuration is less favourable for setting up and maintaining stable operation under the thermosiphon principle. This type of operation is, nevertheless, still possible with a number of additional constraints to be taken into account at the dimensioning and design stage.

2.5. Fundamental safety concepts

In general, significant design provisions should be implemented to avoid any risk of reaction between the air and the D_2 or H_2 filling. Apart from the transfer phases (initial filling and conditioning, down cooling and reheating, and the maintenance operations), which make use of procedures, the safety of which is ensured by automatic sequences; most CNS installations utilize the general principle of double confinement[2] between the units containing hydrogen (or deuterium) and the air. The following technical features contribute to the safety of the installations:

[2] Some CNSs use triple containment throughout, see papers 7.3 and 7.5.

— Systems of tanks immersed in the water of the swimming pool research reactor;
— Double shells with space between the shells filled with inert gas (or evacuated for the cold components);
— Permanent monitoring of leak tightness by a reasonable choice of the pressure differences between the various parts and systems, also applying double seals and integrated leak detection.

3. EXAMPLES OF CNSs IN FRANCE

3.1. CNSs at the RHF of the Institut Laue-Langevin, Grenoble, France

TABLE 1. ESSENTIAL DATA OF THE HIGH FLUX REACTOR (RHF) AT GRENOBLE

Thermal power	58.3 MW
Max. unperturbed thermal flux at the reactor	1.5×10^{15} n·cm^{-2}·s^{-1}
Max. perturbed thermal flux at the beam tubes	1.2×10^{15} n·cm^{-2}·s^{-1}
Coolant	Heavy water
Coolant flow through the reactor assembly	2400 m^3/h
Coolant velocity	17 m/s
Coolant pressure (outlet)	4 bar
Coolant temperature (outlet)	50°C
Reactor cycle length	50 FPD

3.2. RHF's vertical CNS

The vertical CNS assembly contains 20 L of liquid deuterium at 25 K and comprises the following main elements (see Fig. 2):

— Deuterium circuit, including the in-pile part;
— Refrigerator;
— Control panel.

3.2.1. Deuterium circuit of RHF's vertical CNS

The deuterium circuit of RHF's vertical CNS comprises three parts interconnected by tubes. The moderator cell as part one is a sphere of non-alloyed

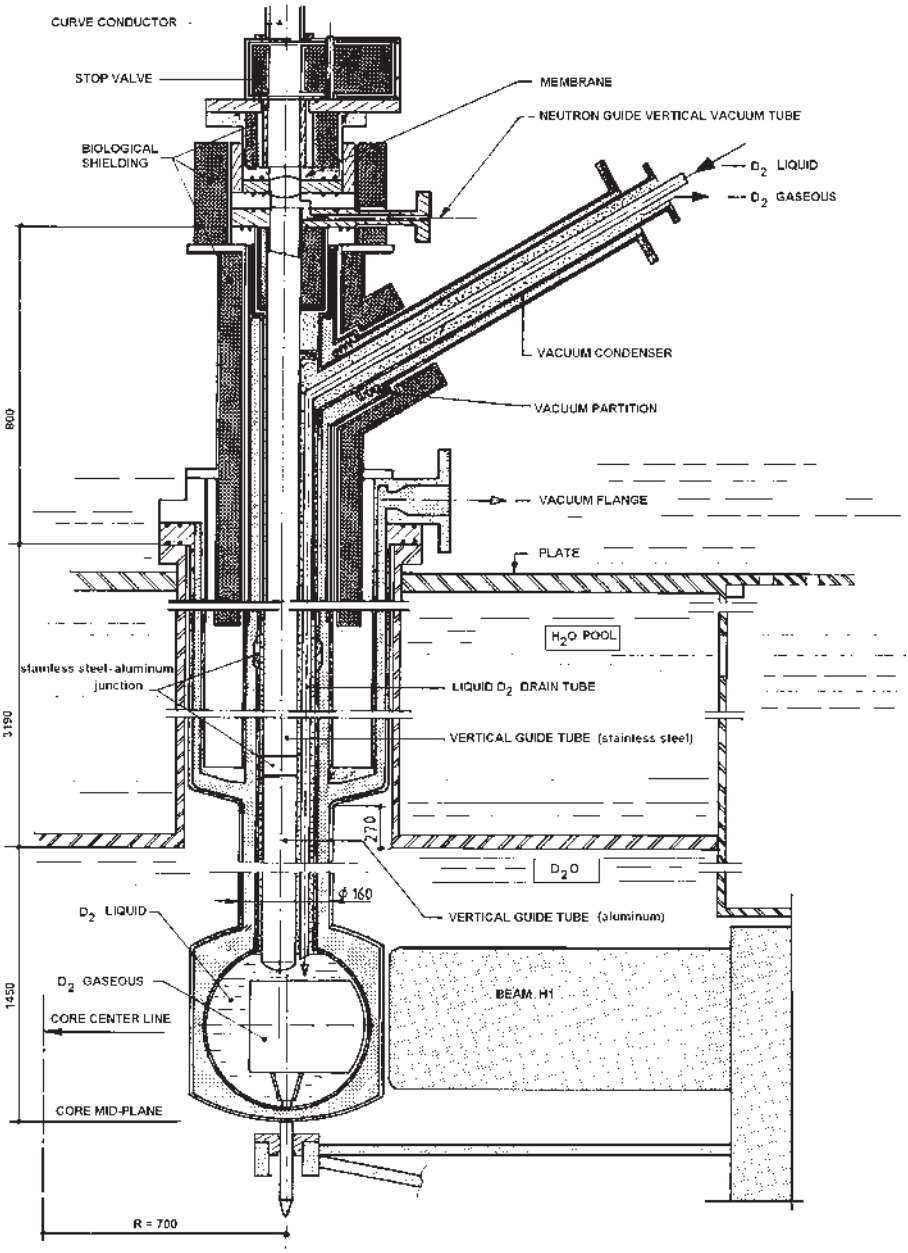

FIG. 2. RHF's vertical CNS.

aluminium manufactured by hydraulic forming and shaping. Its wall thickness, which should be as small and as constant as possible, is 1.55 mm of hard aluminium, except at locations/perpendicular to the welds, where the thickness is increased to 2 mm. The moderator cell is enclosed in a Zircaloy vacuum shell of 6 mm thickness, which follows the shape of the moderator cell and its outlet tube. The shell then widens towards the shaft to provide space for a protecting plug filled with water.

The space between cell and shell is kept under vacuum with no connection to the environment as long as the cryostat is cold.

3.2.2. In-pile part

The in-pile part of RHF's CNS penetrates vertically into the heavy water via a shaft from above the reflector tank.

3.2.3. Condenser

The condenser is immersed into the RHF swimming pool and comprises, around a central passage allowing free circulation of the deuterium, a bundle of 500 vertical tubes in which the deuterium can be liquefied. These tubes are cooled by cold gaseous helium provided by the refrigerator. The link with the in-pile part of the CNS is provided by a thermosiphon permitting the continuous circulation of the deuterium in two concentric tubes: descent by gravity of liquid in the internal tube and gas rising in the outer tube.

The condenser and the thermosiphon are protected and thermally insulated by vacuum. The pump unit which maintains this vacuum is placed on the edge of the swimming pool, enclosed in a box containing nitrogen under pressure to avoid any risk of air coming into contact with the cold parts of the circuit.

3.2.4. Buffer tank

All the deuterium necessary to support CNS operation is contained in an 18 m^3 double shell tank under 3 bar pressure. That tank is arranged at the edge of the swimming pool. The space between the two shells is kept under a nitrogen atmosphere at a pressure of about 1.3 bar.

The buffer tank at the edge of the swimming pool is connected to the condenser by double shell tubing, with the gap between the shells being protected by the same pressurized nitrogen. This arrangement and coupling enables the deuterium to 'breathe' between the condenser and the buffer tank.

Tubes connect the buffer tank to the environment outside the research reactor building to enable the controlled disposal of deuterium, if necessary.

3.2.5. Refrigerator of RHF's vertical CNS

The refrigerator of the two CNSs of the RHF with its nominal power of 10 kW at 25 K uses helium and operates on a Brayton cycle. A flow of 6600 Nm^3/h helium is compressed to 14 bar by the compressors. This helium is cooled down in an exchanger column from 330 K to 28 K, thereafter decompressed in two gas bearing turbines down to 2.8 bar at 19 K.

This cold gas is then sent to the condenser for heat transfer with the deuterium and thus, warms up to 25 K, and then returns to the exchanger column, where it finishes transferring heat with the high pressure gas circulating in the opposite direction. Finally, the helium returns to the compressor, completing the cycle. The adjustment between high and low pressure is controlled by servo valves. A 25 m^3 buffer tank absorbs the fluctuations of the total quantity of helium in circulation and enables storing all the cycle gas when the system is not operating.

3.2.6. Control panel of RHF's vertical CNS

The refrigerator control panel of RHF's vertical CNS permits automatic control and continuous monitoring of the vertical CNS cooling.

3.2.7. Environment of RHF's vertical CNS

The vertical CNS supplies a total of six neutron guides (30 mm × 200 mm section) 110–130 m long and serves more than 12 instruments (see Fig. 3).

It enhances the neutron flux at wavelengths above 3 Å (4.5×10^{14} $n \cdot cm^{-2} \cdot s^{-1}$).

The vertical service tube pointing to the moderator cell is sufficiently large to incorporate a vertical neutron guide of 70 mm diameter constructed of aluminium and lined with a thin (0.15 mm) nickel layer. This section is followed by a curved square shaped guide (70 mm × 70 mm) with a curvature of a radius of 13 m and a length of 13 m. The guide supplies very cold neutrons (VCNs) to an interferometer and to the 'neutron turbine'. Finally, the turbine provides ultracold neutrons (UCNs) to the experiments.

3.3. RHF's horizontal CNS

RHF's horizontal CNS contains 6 L of liquid deuterium at 25 K and enhances neutron flux at wavelengths above 3 Å (8.0×10^{14} $n \cdot cm^{-2} \cdot s^{-1}$) (see Fig. 4). Its moderator cell has the shape of an orthocylinder of 210 mm

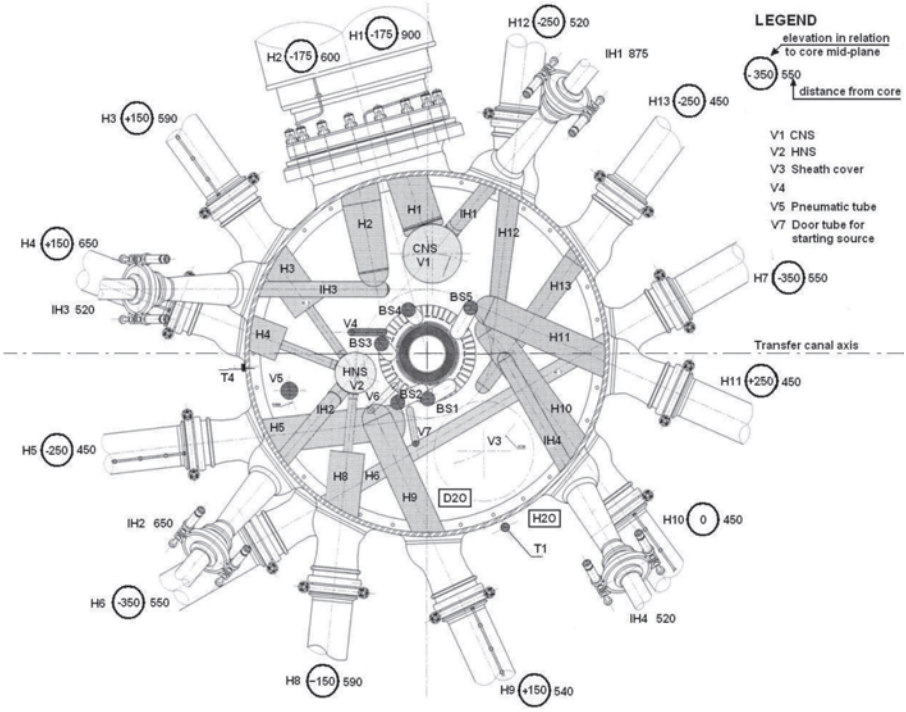

FIG. 3. General arrangement of the HNS, the CNSs and the neutron beam channels at the RHF at Grenoble, France.

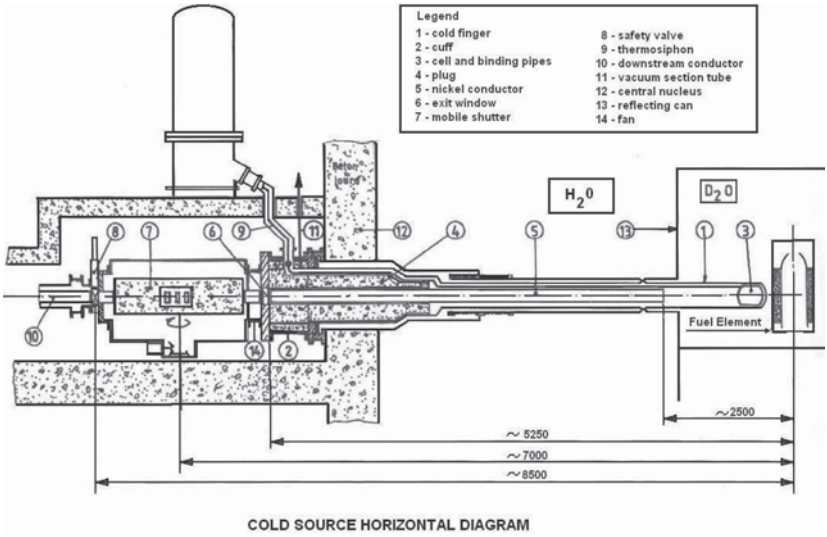

FIG. 4. RHF's horizontal CNS.

diameter filled with liquid deuterium, which is housed in a 230 mm diameter horizontal tube. This CNS feeds two main neutron guides.

All the downstream parts of the thermosiphon operate horizontally. The condenser is placed at the roof of the 'casemate', which provides biological shielding at the exit of the cold neutrons from the research reactor.

Nickel guides are arranged in the sleeve inside the swimming pool. To contribute to the good performance of the installation, they are as close as possible to the moderator cell.

3.4. RHF's HNS

The aim of the HNS is to increase the energy of the neutron flux at a location inside the reflector tank. Via thimbles, the neutrons of increased energy are sent to the experimental areas. Four thimbles are used at the HNS of RHF at Grenoble, France.

For RHF's HNS, the following features apply (see also Fig. 5):

- The moderator is a graphite block; its height is about 300 mm and its diameter 200 mm;
- The distance between the vertical core axis and the HNS axis is 520 mm;
- The temperature of this graphite block is about 2400 K;
- The graphite block is heated by the radiation from the reactor core. The graphite block is surrounded and isolated by several thin layers of graphite wool;
- All devices of the HNS are located inside a double vessel made from two Zircaloy shells;
- There are two distinct helium capacities: the so-called hot helium circuit encloses the graphite block itself and the isolating graphite wool. This circuit is totally closed, but it is connected to an expansion vessel. The second circuit, the so-called cold helium circuit, totally encloses the first one. In normal operation, the second circuit is also totally closed and it ensures the insulation between the hot He circuit and the heavy water of the reflector tank of the RHF;
- At the axial height of the graphite block inside the HNS there is a very small gap between the two shells of the enclosing Zircaloy structure. In this gap, there is a thin layer of cold helium. The accuracy of the thickness of this layer is a fundamental parameter for the insulation of the HNS, since this layer governs the heat balance and adjusts the graphite temperature. It has to establish the right compromise between the temperature of the graphite block and the heat dissipation to the heavy water.

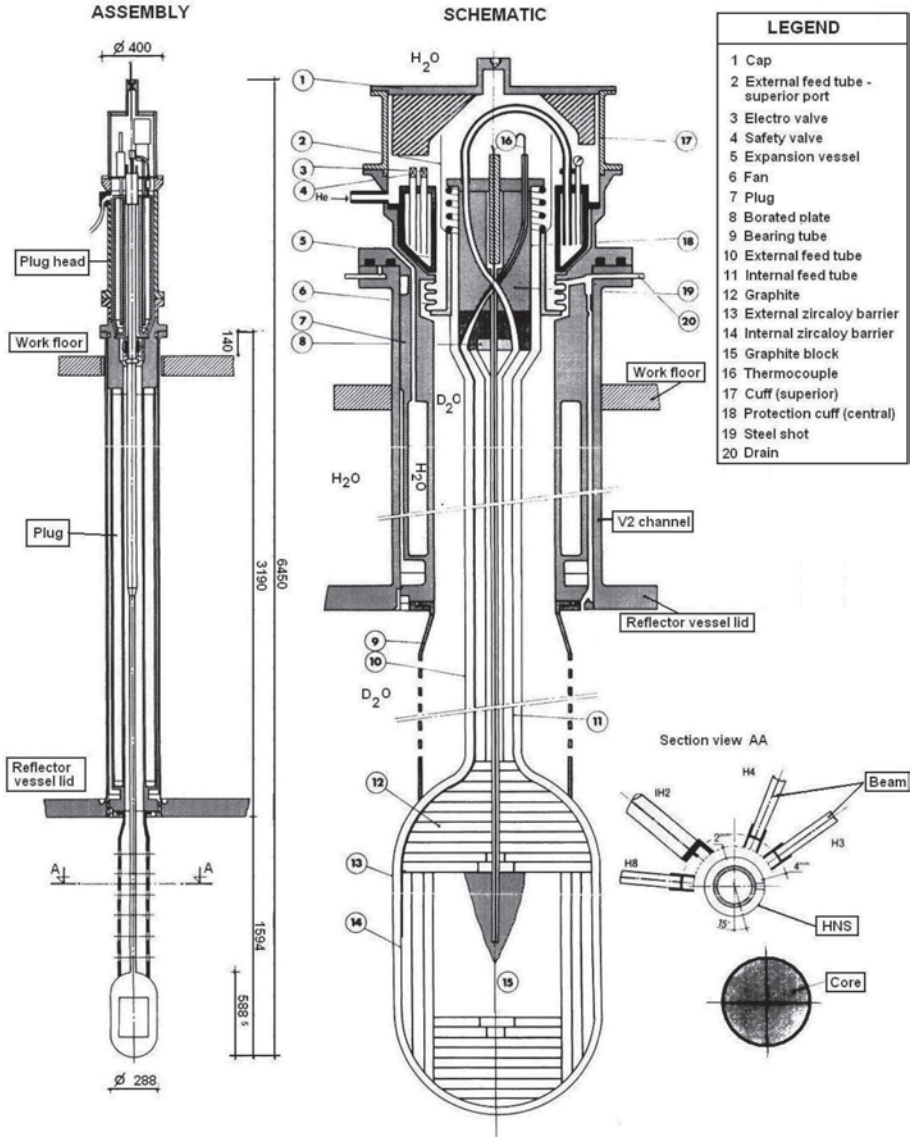

FIG. 5. RHF's HNS.

— The main basic data of RHF's HNS are:

- Hot helium circuit:
  Nominal pressure at cold conditions: 6 bar;
  Nominal pressure at hot operating conditions: 8 bar;

- Cold helium circuit:
  Nominal pressure at cold conditions: 5.4 bar;
  Nominal pressure at hot operating conditions: 5.8 bar;
- Temperature of graphite block at normal operation: 2400 K;
- Temperature of the inner Zr vessel: about 250°C;
- Temperature of the outer Zr vessel: <120°C;
- Volumes of the circuits:
  Internal volume of the inner vessel: 29.3 dm^3
  Graphite volume: 8.7 dm^3
  Graphite wool volume: 2 dm^3
  Actual inner gas volume: 18.6 dm^3
  Internal volume of the outer vessel: 30.6 dm^3
  External volume of the inner vessel: 30.2 dm^3
  Actual gas volume of the outer vessel: 0.4 dm^3
  Expansion vessel of the hot helium circuit; internal volume: 45 dm^3
  Expansion vessel of the hot helium circuit; external vessel: 8306 dm^3

Further characteristic data of RHF's HNS include a width of the gap (thickness of the He layer) between the two Zr shells at cold conditions about 0.1 mm and at hot operation about 0.07 mm.

The instrumentation of RHF's CNS comprises:

— The underflux part of the HNS is fitted with thermocouples in order to supervise the temperature of the inner Zr vessel. Any temperature higher than 300°C generates an alarm.
— The board panel located at the top of the reactor pool is fitted with all the needed pressure sensors in order to ensure a safe process monitoring of the HNS.
— Any accidental filling of any part of the HNS with water during operation of the research reactor is prevented by an automatic reactor scram upon sensing reduced cold helium pressure.

7.3. DEMANDS ON AND REQUIREMENTS FOR THE COLD NEUTRON SOURCE AT HMI'S BER-II AT BERLIN

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1. INTRODUCTION

In the discussion about purpose designed research reactors, one has to distinguish between different fields of research. If experiments are justified through the need of fission energy technology, a secondary source such as a cold neutron source (CNS) might be avoidable. If investigations by means of neutron scattering are desired, however, a CNS is absolutely necessary.

From 1959, the Hahn-Meitner-Institut Berlin (HMI), currently part of the Helmholtz Gesellschaft, has been located in the south-west of Berlin, Germany. In parallel with upgrading from 5 to 10 MW thermal power HMI's pool type research reactor BER-II, which took place between 1985 and 1991, the CNS was installed. It is important to describe the combined demands and requirements in order to understand technical aspects of the design, which is presented in the following discussion.

It is somewhat delicate to argue pro fission and pro a research reactor in Germany these days. The argument in terms of research for nuclear power has been lost, whereas technologies, such as spallation as a neutron source and synchrotron radiation as an alternative probe, which are partially complementary, partially competing, are gaining momentum. However, the subsequent examples will demonstrate the demand of scientists and the necessity to have a middle intense neutron source such as the BER-II equipped with a CNS in the western European scientific environment.

2. DEMANDS ON THE BER-II

At the BER-II of the HMI, research is done in the fields of fundamental physics, chemistry, biology, material and engineering science. Up to 70% of the overall time for scientific measurements is given to external users. Figure 1 shows the distribution of users across Europe; the number of users from other countries, such as the United States of America, Australia and Japan, is still increasing.

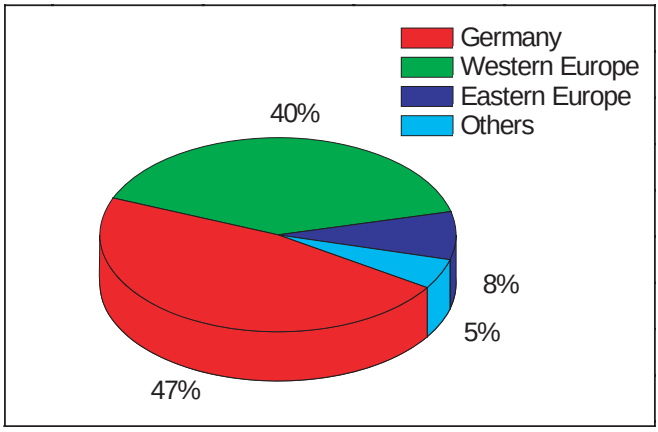

FIG. 1. Distribution of BER-II's users.

A measure for the success and progress in science is the number of scientific publications coming out of BER-II's usage. Figure 2 shows a comparison of papers presented by different neutron scattering sites. The numbers are taken out of the proceedings of the International Conference on Neutron Scattering ICNS 2001.

There are 654 publications in total. The BER-II of HMI with its thermal power of 10 MW plays a major role compared with the ILL (50 MW) and the FZJ (20 MW). Instead of having high flux, the HMI competes with its excellent sample environment: high and lowest temperatures, highest magnetic field and a diamond pressure cell for investigations with thermal and cold neutrons. Of particular interest, the instruments supplied with cold neutrons are overbooked by a factor of three.

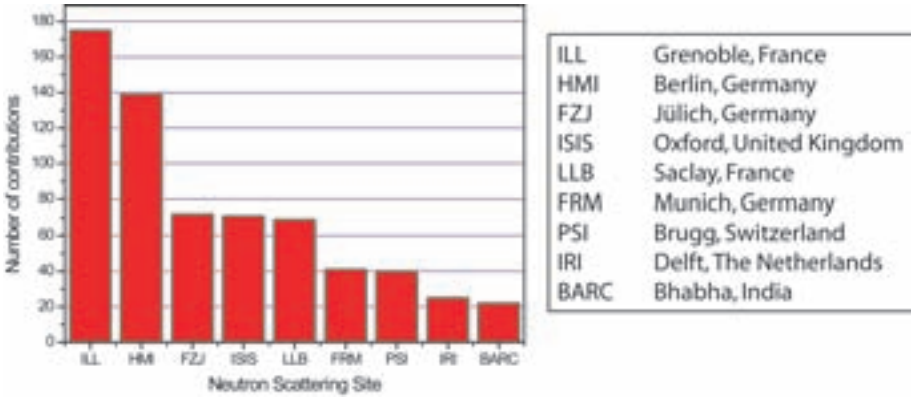

FIG. 2. Publications out of ICNS 2001 proceedings per research institute.

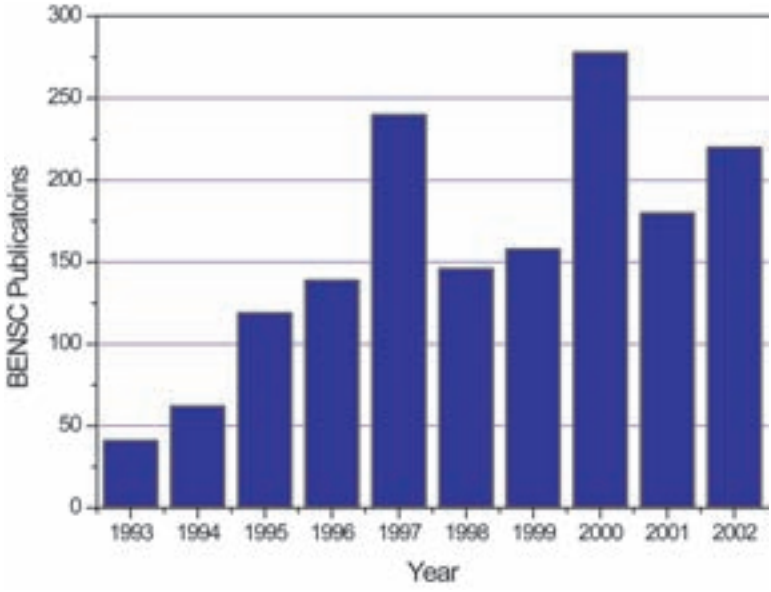

FIG. 3. Evolution of BENSC publications.

Additionally, Fig. 3 shows the evolution of publications prepared at BER-II. International conferences contributed strongly in 1997 and 2000. There is a stable tendency of an increasing number of publications per year coming out of the Berlin Neutron Scattering Center (BENSC).

Figures 1–3 demonstrate the need of fission neutrons for the scientific community. They also show the need of a CNS, since 70% of publications out of Fig. 3 are based on research conducted using cold neutrons, and this percentage is increasing. The applied method is non-destructive and also comprises the ability to investigate massive metallic samples. The main advantage of cold neutrons is the opportunity to study physical, chemical and biological phenomena on a scale from nanometer up to microns by means of neutron scattering with neutrons of long wavelengths from 4 to 20 Å. This is a scale beyond the atomic view — possible with thermal neutrons, and the optical view — done with ordinary microscopy. Using cold neutrons moderated in cryogenic hydrogen leads to a gain factor of 10–50 compared with thermal neutron spectra.

There are more than one thousand researchers in Europe searching for facilities supplying thermal and especially cold neutrons. This will not change within the next decade because of non-compatible experiments at synchrotron radiation facilities and due to a lack of sufficient spallation sources. There are no operating or planned nuclear scattering facilities without a CNS.

3. REQUIREMENTS FOR THE CNS

The main requirements for the CNS at the BER-II, which had to be met, are:

— BER-II was already in operation, when it was refurbished, upgraded and equipped with a CNS at the end of the 1980s;
— CNS had to be installed at the position of the thermal column, which was an advantage at the same time due to the rather extended space provided by that column;
— All necessary cooling machines, compressor, helium and hydrogen buffer, vacuum systems, gas handling, separate freon cooling machines and heat exchanger for the moderator cell circuit had to be arranged apart from the existing BER-II building;
— A maximum flux of cold neutrons out of the source plus a maximum number of 'cold' experiments placed downstream the cold neutron beams;
— Fulfilment of all the complex and stringent regulations for and arrangements with the safety authorities of a nuclear facility in Germany.

The experiments as planned and as installed now at the BER-II are shown in Fig. 4. Thermal neutrons are used on the right hand side around the pool. But the major part of the actual and future utilization is taking place and is planned to proceed with cold neutrons. So the CNS was fitted horizontally into a conical beam tube from where neutrons are guided. The neutron guides are directed to the hall with its 'cold' instruments. The tight arrangement forced HMI to install the CNS components in a separate building and even outside of the buildings, the latter because of the use of explosive hydrogen.

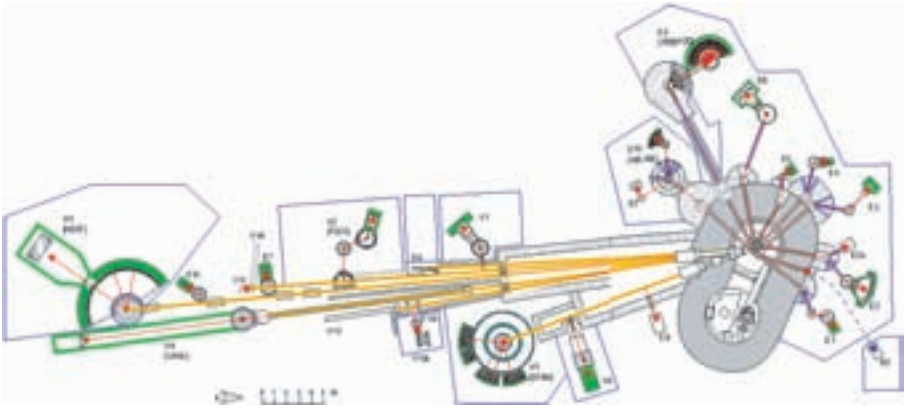

FIG. 4. Using thermal and cold neutrons on E and V experiment places at the BER-II.

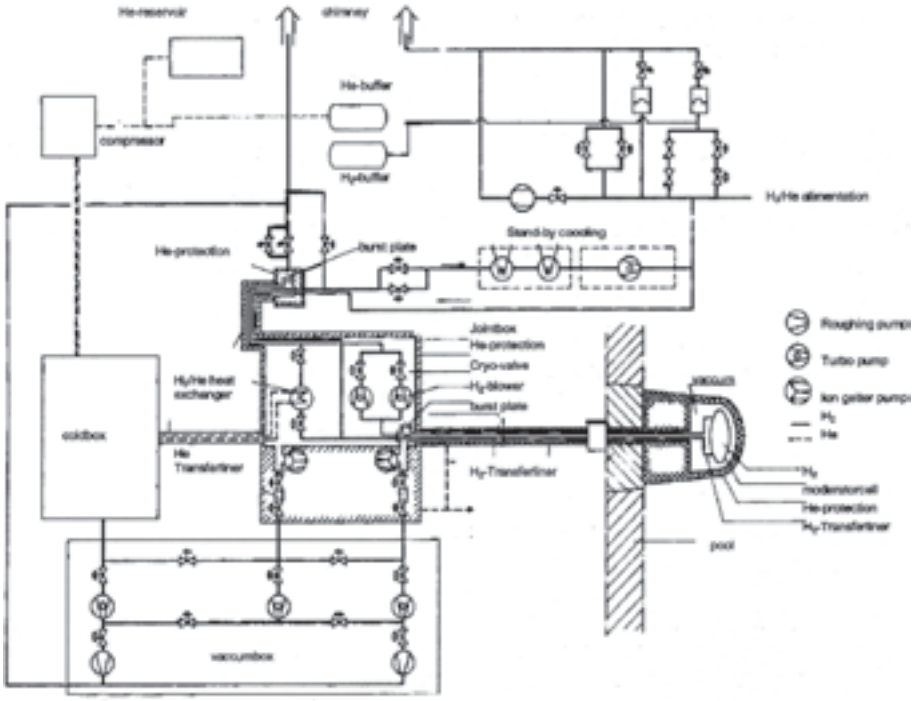

FIG. 5. Complete process diagram of the CNS.

The fact that the in-pile part of the CNS had to replace the thermal column made a horizontal CNS insert necessary, which did not favour a thermosiphon design and, therefore, a design applying supercritical hydrogen with forced convection was chosen. This decision leads to the advantage of single phase hydrogen with easy implementation of emergency cooling. Figure 5 shows the complete process diagram of the CNS system. The moderator cell with a volume of about 800 ml is surrounded by the isolation vacuum. A helium filled protective layer encloses the vacuum space. The resulting three barrier system, which encases all hydrogen carrying components inside the buildings, secures the systems against any hydrogen hazard. The hydrogen transfer liner connects the moderator cell with the joint box, which contains the hydrogen blowers, the helium/hydrogen heat exchanger and some cryogenic valves. To compensate for the density change of hydrogen due to cooling and to install a second independent non-cryogenic cooling system for the moderator cell in order to avoid its melting in case the main cryogenerator is lost, the joint box is connected to an outside hydrogen buffer and to supplementary cooling machines via a transfer liner.

TABLE 1. CHARACTERISTIC DATA OF THE CNS AT THE BER-II

Hydrogen system	Pressure	>13 bar
	Temperature	>14.2 K
	Maximum pressure	20 bar
Moderator cell	Volume	831 cm^3
	Maximum pressure cycles	400
	Maximum thermal neutron fluence	10^{22} n/cm^2
Hydrogen circuit		
Normal operation	Pressure	13.9–17.3 bar
	Moderator cell temperature	25–35 K
	Hydrogen flow	1.1 L/s
	Thermal heat input	about 2000 W
Standby operation	Pressure	13.9–17.3 bar
	Moderator cell temperature	about 0°C
Hydrogen masses	H_2 circuit	970 g
	Transfer liner	40 g
	H_2 buffer	8300 g
Hydrogen buffer	Volume	6.3 m^3

To complete the CNS system, additional components and systems include:

— 'Air liquide helium refrigerator' (Helial), with the cold box, the compressor, the helium buffer and the helium handling system;
— Hydrogen gas handling system;
— Vacuum box.

The principal design of the CNS was developed at the Risø National Laboratory (RNL) in Denmark. The technical design and the construction were completed in collaboration with Interatom/Siemens and RNL.

4. DESIGN AND SAFETY FEATURES

Some features concerning operation and safety against incidents or accidents have to be compiled to understand the CNS in its function and its interface with the research reactor:

— It was already mentioned that the CNS uses supercritical hydrogen as moderator. Under the described conditions, the moderator density can be

enhanced by a factor from 4 to 5, depending on pressure and temperature differences with respect to ambient conditions. In its homogeneous gas phase, the gas can be transported with simple blowers.
— If the helium refrigerator or any other component of the main refrigeration system fails, there is a standby cooling system, which allows continued reactor operation at 10 MW, without overheating the moderator cell.
— In case of transfer liner or moderator cell fracture, a depressurization is foreseen directed towards the exterior via the isolation vacuum. Although the conical beam tube, which contains the moderator cell, is nearest to the core, it is designed to be stable up to an internal pressure of 30 bar.

The independence of the research reactor from the status of the moderator cell, in any case, is absolutely important due to their close proximity. In fact, there is high grate 'space craft fuel' 100 mm from the fission zone of the core. Therefore, a safety analysis was performed and an SAR written, which described investigations of all credible scenarios. From this SAR, the points concerning operational safety and safety against failures are listed below:

— Reactivity coupling to the core;
— Fracture of the moderator cell;
— Complete breakdown of moderator cell cooling;
— Fracture of the conical beam tube combined with destruction of the vacuum chamber and hydrogen transfer liner;
— Oxyhydrogen detonation outside the three barrier system;
— Oxyhydrogen detonation inside the three barrier system;
— Failures during operation:
  • Failure of the helium refrigerator;
  • Failure of the hydrogen blower;
  • Failure of the standby cooling system;
  • Behaviour of the helium refrigerator after reactor shutdown or scram;
  • Failure of power supply;
  • Insufficient cooling of vacuum chamber;
  • Loss of hydrogen;
  • Leak of hydrogen into the isolation vacuum;
  • Leak of hydrogen outside the isolation vacuum, outside the building;
  • Leakage in the helium hydrogen heat exchanger;
  • Leakage in the standby cooling system;
  • Failure of pressure control of the hydrogen system;
  • Loss of cryovacuum;
  • Loss of helium protection gas out of the three barrier system.

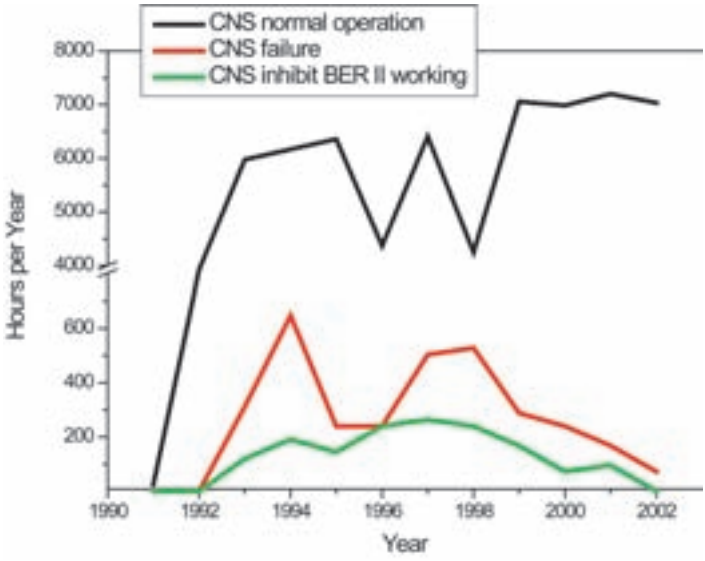

FIG. 6. Performance of CNS and BER-II.

At the beginning of the CNS operation, there was frequent trouble with the hydrogen blowers and the flanges of the hydrogen circuit. Due to improvements at both components, however, the performance of the CNS during the last years is very good. Figure 6 summarizes the reactor and the CNS operation since the BER-II upgrading.

Except longer periods of maintenance due to ventilation refurbishment in 1996 and the installation of a special experiment in 1998, the CNS shows a high availability and increasing reliability. One scientist and two technicians are responsible for the proper operation of the CNS system. There are three further reactor operators, which may support those three if there is need. Maintenance contracts are used for the helium refrigerator, the compressor and the additional cooling aggregates.

5. SUMMARY

Running a fission reactor for research is necessary for the development of nuclear technology in the fields of nuclear power generation, for the generation of radioisotopes, and for basic research by neutron scattering, etc. Since the refurbishment, the BER-II has focused on basic research. Using a research reactor as a neutron source for neutron scattering has yielded rapid progress in material science, and extended the view of fundamental processes at the atomic

scale over the last five decades. Today, the range of research is further extended from the atomic scale to the biological and macroscopic scale by the means of so-called secondary neutron sources. As one of these sources, a CNS is in high demand within the scientific community. Most of the operational requirements of a CNS result from the combination of nuclear fuel with highly explosive hydrogen next to it. This combination is the best means for progress in neutron scattering at the moment. Competing non-fission neutron sources, which avoid nuclear fuel and the problematic nature of its disposal, may emerge in a sufficient amount in the future. Even then, the CNS as a secondary neutron source will be an absolute must. Finally, the BER-II was well advised to install a CNS early, i.e. 20 years ago. This investment has paid back substantial returns for the scientific community at HMI.

7.4. PNPI EXPERIENCE WITH THE DEVELOPMENT OF COLD AND ULTRACOLD NEUTRON SOURCES AT RESEARCH REACTORS

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1. COLD AND ULTRACOLD NEUTRON SOURCES AT THE WWR-M REACTOR AT THE PNPI

1.1. Introduction

The Petersburg Nuclear Physics Institute (PNPI) is located in Gatchina, the Russian Federation, and operates an 18 MW WWR-M research reactor. The WWR-M was commissioned in December 1959 and formed one of the ten best swimming pool reactors of the time. The successful design of the fuel assemblies and reflector was used in other research reactors (in the Russian Federation at Kiev, Obninsk, Alma-Ata; and also in Germany, Hungary, Poland and Vietnam).

By the start of the 1970s, considerable experience had been accumulated in operating the WWR-M1 and WWR-M2 fuel assemblies. Comprehensive research had been done on their heat engineering and corrosion features, and the sealing level had been determined. The result of investigations has enabled the design of the next generation WWR-M5 fuel assemblies, which had half the wall thickness (1.25 mm) and double the fuel concentration: 125 g ^{235}U/L [1]. The surface per unit volume in the core attained the record value of 6.6 cm^2/cm^3, which provided a maximum specific power of 0.9 ± 0.1 MW/L in the pool reactor [2]. The reactor was completely converted to WWR-M5 fuel assemblies in 1980. The power was raised to 18 MW. The assemblies contained more ^{235}U (90% enrichment), which resulted in freeing more than half of the space in the core for experimental equipment. In particular, a source for cold neutrons and ultracold neutrons was located at the centre of the core, whose output was greater than that of analogous sources in the most powerful ILL research reactor at Grenoble, France [3].

Since the middle of the 1970s, cold neutron and ultracold neutron generation has been in progress there. As intended, this programme has increased substantially the experimental capabilities of the research reactor as an instrument for fundamental physics research. It has allowed progress to be made in the investigations which require extremely high neutron beam

intensity, such as the search for neutron electric dipole moment, neutron lifetime and correlation constant measurements, the search for P-odd effects in interactions with cold neutrons and so on.

The layout of the WWR-M research reactor with cold neutron sources (CNSs) and ultracold neutron sources (UCNSs) located at different positions is shown in Fig. 1.

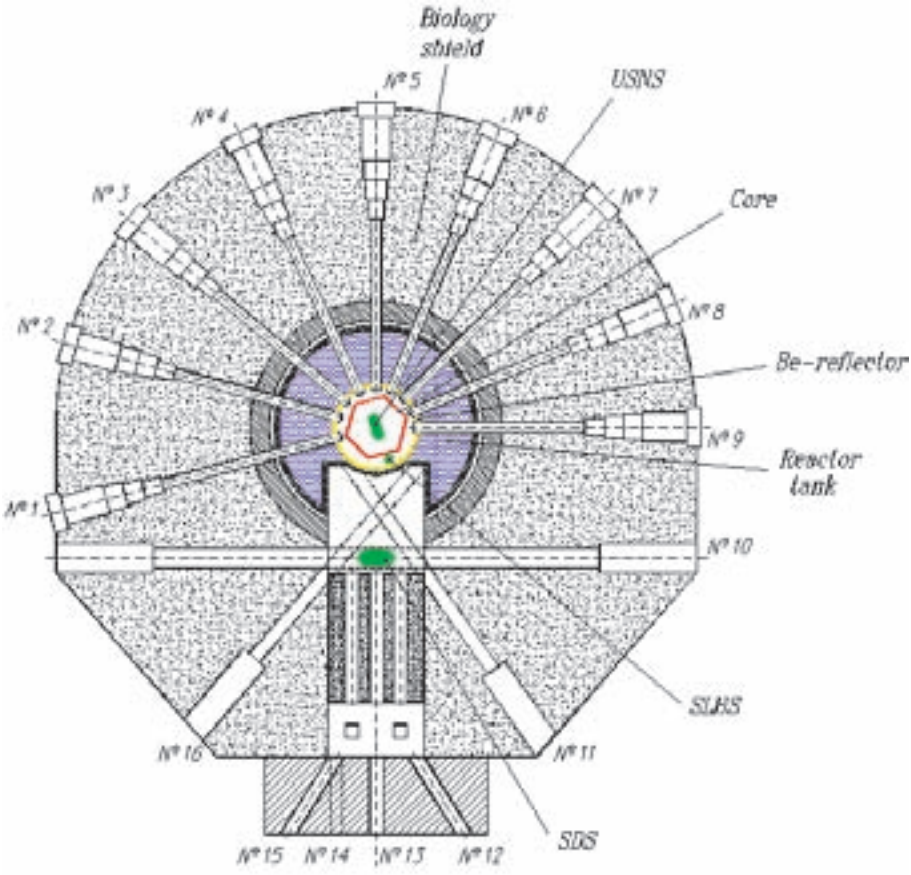

FIG. 1. Overhead view of WWR-M reactor. Small liquid hydrogen source (SLHS); universal supercooled neutron source (USNS); solid deuterium source (SDS); maximum power level: 18 MW; minimum fuel cycle length: 6–12 d; active fuel high: 600 mm; active fuel region volume: 100–150 L; U-235 loading: 8.5 kg; horizontal/vertical experimental channel: 17/3.

1.2. Small liquid hydrogen source

Provided that the moderator volume is not large and the heating is not high, the simplest technique is a direct cooling of the moderator chamber by means of a cold helium circulation. This principle was used in the first hydrogen UCNS at the WWR-M reactor [4]. The source was placed at the Be reflector where the perturbed neutron flux was 6×10^{13} n·cm^{-2}·s^{-1} for thermal neutrons and 8×10^{12} n·cm^{-2}·s^{-1} for neutrons with energy E > 1 MeV. The specific nuclear heating was 8 W/g for hydrogen and 0.3 W/g for the chamber material. The chamber contained 150 cm^3 of liquid hydrogen and the total nuclear heating was 300 W. The chamber was made like a heat exchanger that allowed us to keep liquid hydrogen subcooled up to a reactor power of 12 MW. However, at 18 MW just 15% of vapour was in the chamber. This UCNS gave a gain factor of about 30 and 4×10^4 n/s of ultracold neutron flux integrated up to the velocity 7 m/s. It was operating from 1980 to 1985 producing, at the time, the highest ultracold neutron intensity in the world.

The assembly of the small liquid hydrogen source is shown in Fig. 2. Probably the maximum capability of the method of direct cooling was demon-

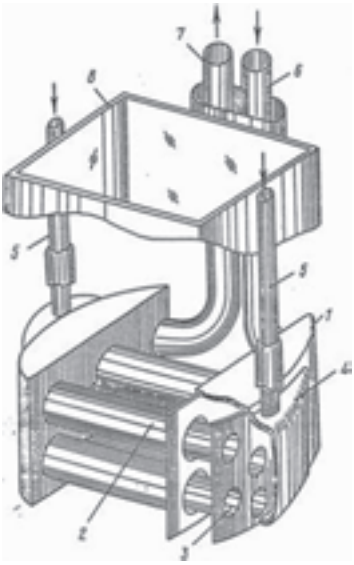

FIG. 2. Small liquid hydrogen source: 1. helium collector; 2. helium jacket; 3. tubes with moderator; 4. hydrogen collector; 5. hydrogen input tubes; 6. helium input tube; 7. helium output tube; 8. neutron guide.

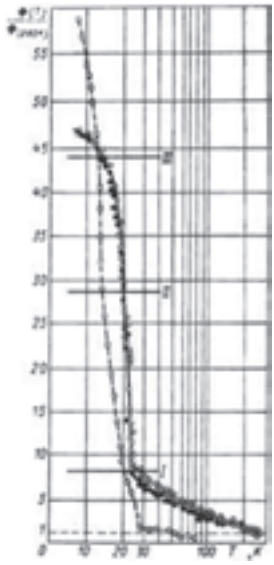

FIG. 3. Temperature dependence of the yield of UCNs for different states of hydrogen.

strated in that case. It certainly could be recommended for application as the simplest and the most efficient means of heat removal, when the radiation heating is not very high. Temperature dependence of yield of UCNs is presented in Fig. 3. The white points in Fig. 3 are obtained during the process of cooling of the hydrogen and the black points during the process of heating of the hydrogen. Area I corresponds to the gaseous state of hydrogen; area I–II is related to the process of liquefaction of hydrogen; area II–III to the cooling of liquid hydrogen and the area of III for solid hydrogen.

1.3. Universal supercooled neutron source

To obtain the maximum cold neutron intensity at the WWR-M reactor, a large hydrogen source was placed into the centre of the reactor core. In this case, the extremely high specific heating required a high power method of heat removal. It was found that the best way would be a natural circulation of supercooled liquid hydrogen between the chamber and the external heat exchanger. This method is known as a thermosiphon, but it has not been used regularly with supercooled liquids before. It can maintain the moderator in the chamber a few degrees below boiling point at the highest reactor power.

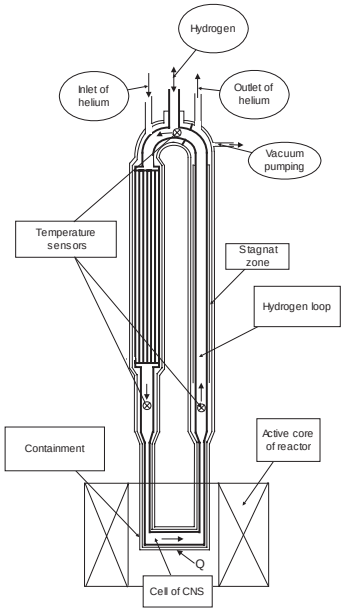

FIG. 4. Supercooled liquid hydrogen thermosiphon.

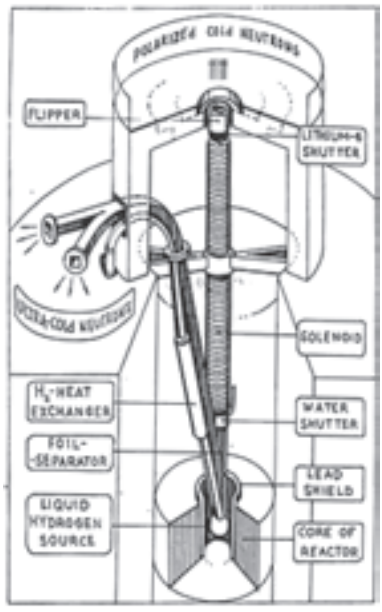

FIG. 5. Schematic diagram of the universal liquid hydrogen source.

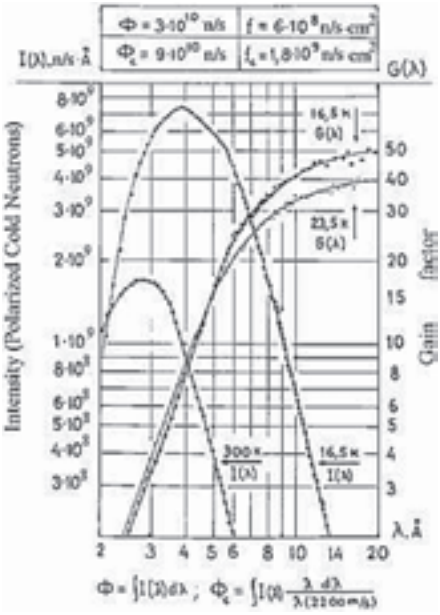

FIG. 6. Spectra of the neutron fluxes and the gain factor for polarized beam of cold neutrons.

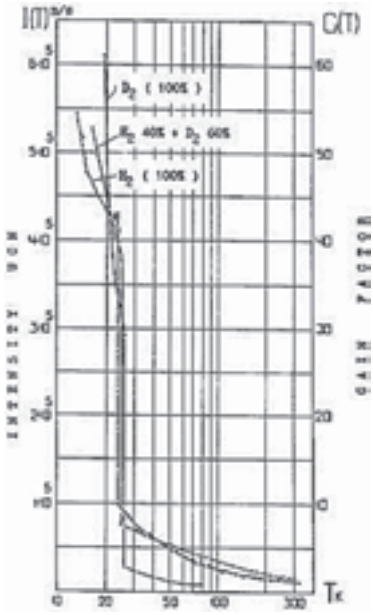

FIG. 7. Temperature dependence of the yield of UCNs for different moderators.

This new neutron source has been operating at the WWR-M research reactor since 1986 [5]. It is a universal source since it produces both ultracold and polarized cold neutrons. The chamber with the moderator is placed inside the flux trap in the centre of the reactor core where the flux is (1.5–2) × 10^{14} n·cm^{-2}·s^{-1} for thermal neutrons and 2 × 10^{13} n·cm^{-2}·s^{-1} for neutrons with energy E > 1 MeV. The chamber made from Zircaloy has a volume of 1 L. The specific nuclear heating was 18–20 W/g for hydrogen and 0.7 W/g for Zircaloy. The total nuclear heating with 100% hydrogen was 2.8 kW. The liquid mixture of 40% of hydrogen and 60% of deuterium is used as the moderator. In this case, the total nuclear heating is 1.8 kW.

A layout of the source at the research reactor and the neutron data obtained are presented in Figs 5–8. The neutron characteristics for cold neutrons (see Fig. 6) are practically the same for both moderators: 100% hydrogen and hydrogen 40% + deuterium 60%, respectively. The detailed studies of hydrogen/deuterium mixtures have been carried out in a special experiment.

An attractive feature of the source is that the thermosiphon circuit is placed entirely in the cold helium flow, the hydrogen being surrounded by two walls with helium between them plus a vacuum containment. Such an approach

to the source design improves the heat transfer, allows independent operation of the research reactor and of the source, and substantially increases the reliability of the construction and hydrogen safety.

1.4. Solid deuterium source

The experimental results of ultracold neutron (UCN) production by means of a solid deuterium source at WWR-M reactor are considered in Ref. [6]. A gain factor of ultracold neutron yield from solid deuterium at 13–14 K to ultracold neutron yield from gaseous deuterium at 300 K is 1230. At solid deuterium temperature 18.7 K (triple point), the gain amounts to 550.

A layout of the source is shown in Fig. 9. The source chamber (diameter 150 mm, length 300 mm with two elliptical heads) is made from zirconium alloy. It has a volume of 6 L (Fig. 9). The chamber has double walls (2 × 0.5 mm) with cold helium flowing between them from the cryogenic refrigerator (capacity 150 W at 4.5 K). The deuterium from a tank (volume 6 m^3) supplies the chamber. By He cooling, the deuterium is condensed and transforms to the solid state inside the chamber.

The relative gain factors for neutrons of different wavelengths are shown in Figs 10 and 11.

A special technical solution is necessary to decrease the deuterium temperature to values lower than 10–12 K. The problem is the loss of the thermal contact between the cold chamber wall and the solid deuterium

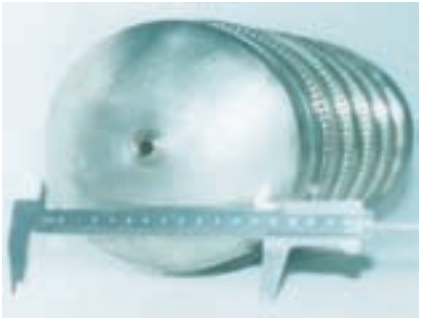

FIG. 8. The cell of the solid deuterium source made from zirconium alloy.

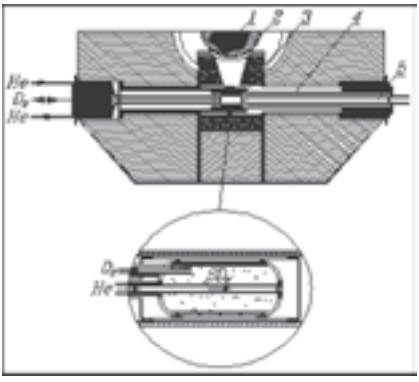

FIG. 9. Arrangement of the solid deuterium source in the research reactor: 1. chamber with solid deuterium; 2. reactor core; 3. beryllium reflector; 4. vacuum container; 5. ultracold neutron guide.

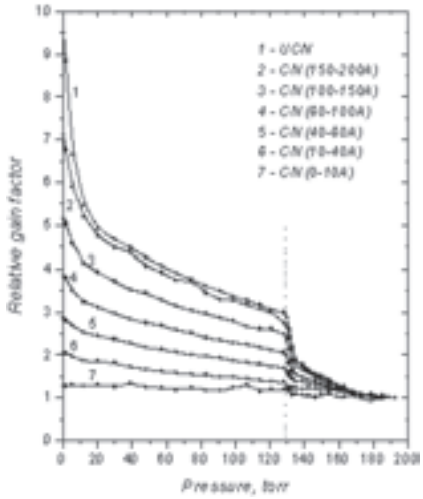

FIG. 10. The relative gain factor for neutrons with different wavelengths as a function of pressure in the system.

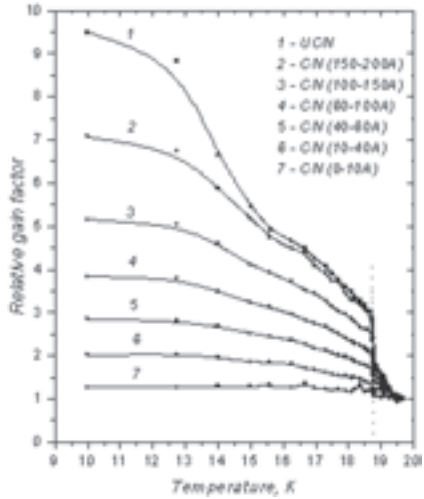

FIG. 11. The relative gain factor for neutrons with different wavelengths as a function of temperature.

because of a volume decrease of the deuterium during cooling. The cooling remains possible at some torr of vapour pressure. For example, at 12 K the saturated vapour pressure is 0.75 torr, but at 10 K only 5×10^{-2} torr.

A possible technical solution is to place a spiral tube at the inner chamber wall to achieve a good thermal contact. As a first step, the experiments were performed with a simple design. The second step will be with the same design, but with deuterium containing some helium quantity for heat exchange. Only the third step will include the more complicated chamber design if it is necessary.

The next step of investigation is a decreasing of the source temperature down to 6–7 K with the aim of studying the capability of increasing the ultracold neutron yield. It is necessary to note that at 6–7 K, the heat conductivity of the solid orthodeuterium increases by an order of magnitude. It improves the source capacity for increased heat load. The result of the investigation can be used for new projects with solid deuterium sources at high flux reactors with heavy water reflectors. Heavy water is a good shield from high energy neutrons and gamma radiation. That allows having a low level of heat load at high thermal neutron flux (research reactor PIK, and research reactor RHF, Grenoble, France). Another possibility is using solid deuterium sources at neutron spallation sources, where the relation of heat load to neutron flux is better than at fission based research reactors. For example, installation of a

TABLE 1. COMPILATION OF PARAMETERS OF THE SECONDARY SOURCES AT WWR-M

Parameter	Value		
	SLHS	USNS	SDS
Moderator substance	Hydrogen	Mixture 40% H_2, 60% D_2	Deuterium
Thermal neutron flux at 18 MW ($n \cdot cm^{-2} \cdot s^{-1}$)	6×10^{13}	2×10^{14}	1.7×10^{12}
Cold neutron		6×10^{8}	Gain 1230
Ultracold neutron ($n \cdot cm^{-2} \cdot s^{-1}$)	1×10^{3}	6×10^{3}	
Moderator chamber volume, L	0.15	1(6)	6
Heat load on construction material, W	100	800	40
Heat load in moderator, W	84	1000	40
Total heat load with losses, W	300	2000	90
Available capacity of cryogenic refrigerator, W	500	4000	150
Temperature level, K	20	20	5
Hydrogen tank volume, m^3	0.16	5	6
Pressure, warmed up, MPa	0.25	0.25	~0.1
Pressure, cooled down, MPa	0.15	0.15	High vacuum
Helium flow rate, g/s	16	100	10
Oxygen impurity, volume %		3.4×10^{-4}	
Nitrogen impurity, volume %		2.0×10^{-2}	

solid deuterium source for ultracold production is planned at the 1 MW spallation source at Los Alamos (United States of America).

The main parameters of the secondary sources at the WWR-M are presented in Table 1.

2. CNS AND UCNS IN THE VERTICAL CHANNEL OF PIK AT PNPI

The high flux beam type research reactor PIK of 100 MW power and with a thermal neutron flux in the heavy water reflector of 10^{15} $n \cdot cm^{-2} \cdot s^{-1}$ is being built in PNPI and is designed for studies to be conducted in various fields of fundamental science, as well as for the solution of a broad sphere of applied problems.

Neutron radiation is a universal tool for scientific research in physics, chemistry, biology, geology, material science, medicine, for semiconductor

production technology, for industry, etc. Both ranges of application, the studying of fundamental properties of the neutron itself, as well as its interaction of neutrons with nuclei and matter, up to neutron therapy, i.e. from purely scientific research up to modern process monitoring methods, are being widened.

The experimental capabilities of the PIK research reactor are determined not only by a high neutron beam intensity, which is approximately an order of magnitude higher than currently functioning medium power reactors; but also by the availability of hot, cold and ultracold neutrons from the respective secondary sources. So, in comparison with research reactors created during 1950–1960, the PIK research reactor will provide unique capabilities both for deepening studies of what is currently conducted in the Russian Federation applying neutrons and for the conduction of new investigations that are inaccessible now at domestic research reactors.

The programme of research at the PIK research reactor shall provide the possibility for foreign researchers to participate in its projects and experiments. The neutron beam parameters and experimental capabilities of the PIK research reactor are unique and in the next 10–15 years such research reactors will not be built anywhere in the world.

PIK's reactor core with a volume of about 50 L is placed in a heavy water reflector and is a compact intensive fission neutron source of 100 MW power (see Table 2). The heavy water reflector provides the best ratio of thermal neutron flux to reactor power at a low gamma radiation background. The thermal neutron flux in the heavy water reflector exceeds 10^{15} n·cm^{-2}·s^{-1}, and in the light water trap (central experimental channel) equals 5×10^{15} n·cm^{-2}·s^{-1}. The replaceable vessel allows varying the reactor core parameters within a wide range.

Deuterium CNSs located in heavy water reflector are planned for the PIK research reactor. Such sources allow getting the largest gain in cold and ultracold fluxes in comparison with other moderator types. At present, a model of the source is installed in the reactor tank in order to perform full scale experiments. The vertical cold neutron source (VCNS) of PIK includes:

— Deuterium loop;
— Cryogenic plant (CP);
— Deuterium supply system;
— Nitrogen supply system;
— Vacuum supply system;
— Control and data acquisition systems.

TABLE 2. PARAMETERS OF THE RESEARCH REACTOR PIK

Power	100 MW
Maximal specific power	6 MW/L
Reactor core volume	51 L
Reactor core inner diameter	390 mm
Reactor core height	500 mm
PIK type fuel elements:	
Enrichment:	90% U-235
Fuel in a copper-beryllium matrix:	UO_2
Uranium density in the matrix:	2.2 g/cm^3
Stainless steel cladding thickness:	0.15 mm
U-235 fuel concentration:	600 g/l
D_2O reflector:	
Diameter/height:	2.5/2.0 m
Cooling circuit:	
Coolant:	H_2O
Pressure:	50 atm
Flow rate:	2400 m^3/h
Inlet/outlet temperature:	50/90°C

The deuterium loop of the system is divided into in-pile and out-of-pile parts. The in-pile part of deuterium loop (Fig. 12 [7], Figs 13 and 14) is installed in a vertical channel (PIK 00020) of PIK's heavy water reflector and includes:

— CNS chamber, see Fig. 12 (1);
— Deuterium condenser, see Fig. 12 (7);
— Cryogenic pipelines, see Fig. 12 (5);
— Light water casing, see Fig. 12 (6).

These units are assembled with industrial pipes enabling the dripping of light water, vacuum system links, air taps, etc.

The deuterium condenser, see Fig. 12 (7), is manufactured from stainless steel 12Cr18Ni10T. It is a heat exchange apparatus connected with the out-of-pile part of the deuterium loop and the CP. The in-pile part of the loop has vacuum insulation. For the residual heat removal from the moderator cell wall after reactor shutdown, the vacuum isolating gap of the cell is filled with helium with a pressure equal to the pressure in the deuterium loop, provided through the pumpdown piping.

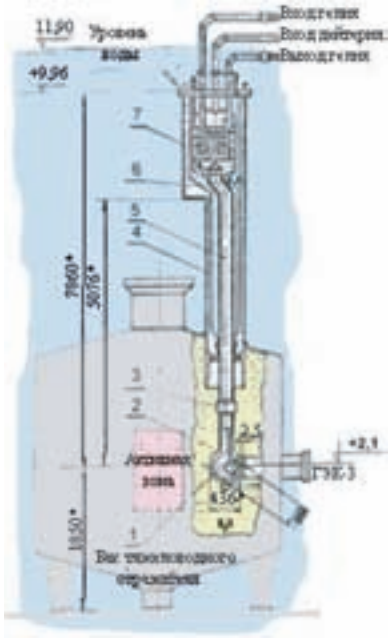

FIG. 12. Arrangement of the CNS at PIK: (1) CNS chamber; (2) zirconium casing; (3) adapter; (4) vertical channel PIK 00020; (5) cryogenic pipelines; (6) light water casing; (7) deuterium condenser.

The results of model calculations [8] of the CNS's neutron spectra are presented in Table 3 and in Fig. 15. The ultracold neutron fluxes $\Phi_{ucn}(E_{ucn})$ by layers, integral neutron fluxes

$$\int_0^{E_{ucn}} \Phi(E)dE$$

gain factors

$$G = \frac{\Phi(E)}{\Phi_{\max well}(E)_{T=23K}}$$

and just as relative neutron fluxes under T = 23 K and

$$T_{eff.} = T_0 \left(\frac{\bar{v}}{\bar{v}_0} \right)^2$$

are shown in Table 2. Here, E_{ucn} boundary spectrum energy of the UCN, equal to 4×10^{-7} eV, $T_0 = 293$ K, $\bar{v}_0$ — average neutron speed at T_0 and $\bar{v}$ — average neutron speed of the concerned spectrum. The total neutron flux is normalized to unity.

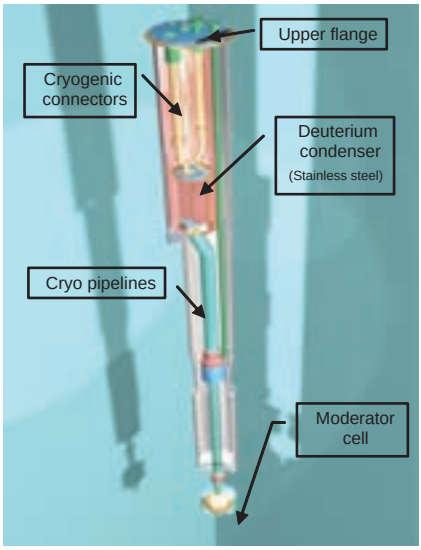

FIG. 13. 4' view of the CNS's vertical channel.

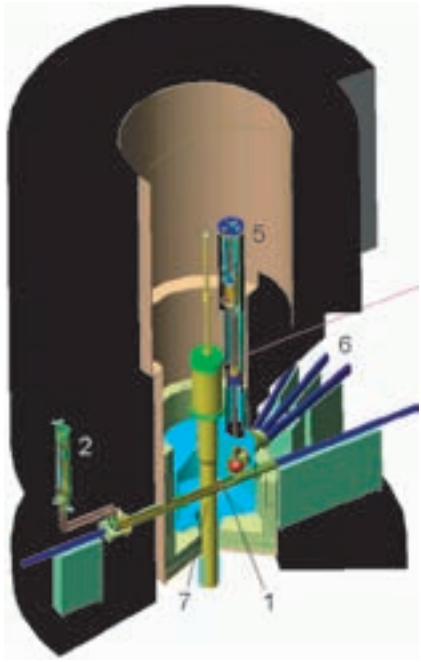

FIG. 14. Three-dimensional view of the reactor PIK with CNSs in the vertical channel and in HEC 4-4'.

The distance between the centres of the CNS and the research reactor core is 780 mm. The total neutron and gamma heating in the source is 6 kW, 2.7 kW thereof in liquid deuterium and 3.3 kW in the moderator cell construction materials. The cell of the source is manufactured from aluminium alloy AD0 (1050-O), and the vacuum containment from zirconium alloy. At the place of CNS arrangement, a flux of thermal neutron of $\Phi_t = (2–4) \times 10^{15}$ n·cm^{-2}·s^{-1} is expected, but the fast neutron flux is less then 10^{11} n·cm^{-2}·s^{-1}, both at the PIK research reactor rated power of 100 MW. The deuterium volume in the cell is 25 L. The working pressure in the deuterium loop is 0.15 MPa, but in the 'warm' state (at 300 K) 0.3 MPa. The CNS chamber has a leakproof funnel with dimensions of 100 mm × 190 mm × 200 mm, precisely centred about the centre of the channel HEC-3. The CNS moderator cell is presented in Fig. 16.

The development of modern high performance CNSs at the research reactor PIK will allow a wide spectrum of fundamental research to be carried out in the field of elementary particle physics and condensed matter.

To extend the research, setting up a universal source of cold and ultracold neutrons in the GEK-4 horizontal channel is planned. That source should

TABLE 3. CNS ULTRACOLD NEUTRON FLUX (TOTAL NEUTRON FLUX NORMALIZED TO 1 n·cm^{-2}·s^{-1} EVERYWHERE)

Distance from CNS centre (mm)	$\Phi_{ucn}(E_{ucn})$ (n·cm^{-2}·s^{-1} eV)	$\int_0^{E_{ucn}} \Phi(E)dE$ (n·cm^{-2}·s^{-1})	Gain factor G	$\frac{\Phi_{ucn}(E_{ucn})}{\Phi_{\max well}(E_{ucn})}$ T = 23 K	$\frac{\Phi_{ucn}(E_{ucn})}{\Phi_{\max well}(E_{ucn})}$ T = $T_{eff.}$	$T_{eff.}$ (K)
30	0.200	1.13×10^{-7}	108	0.66	1.7	36.6
50	0.184	1.08×10^{-7}	100	0.62	1.8	37.9
70	0.176	1.04×10^{-7}	97	0.59	1.9	40.4
80	0.170	1.00×10^{-7}	93	0.57	2.0	43.4
90	0.160	0.95×10^{-7}	87	0.54	2.2	46.3
100	0.149	0.87×10^{-7}	81	0.50	2.4	50.3
110	0.132	0.78×10^{-7}	72	0.44	2.6	56.0
120	0.111	0.66×10^{-7}	61	0.37	2.95	64.7
127	0.181	0.49×10^{-7}	49	0.27	3.1	77.5

greatly improve the experimental facilities compared with the analogous ones at the WWR-M. A very important experiment on the neutron electric dipole moment is planned, as well as experiments involving precision measurements on the beta decay of the neutron determining lifetime and correlation constants. The purpose of the precision measurements on the neutron beta decay is to

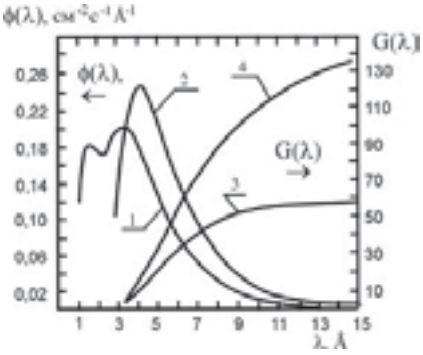

FIG. 15. Neutron spectra at the end of channel and gain factors. Spectrum at the end of channel; Maxwell spectrum under T=23 K; gain factor for channel's spectrum; limiting gain factor.

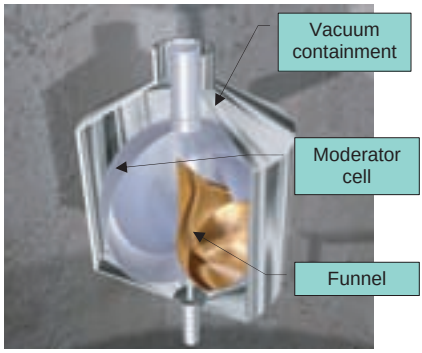

FIG. 16. CNS moderator cell with funnel inside.

check the standard model at a new level of accuracy and thus, detect possible deviations. Much space will also be given to research on weak nucleon–nucleon interactions by use of an intense beam of polarized cold neutrons. It is planned to perform experiments to detect the violation of T-parity in the neutron beta decay, and also for polarized and oriented nuclides. Preparations are also being made for neutron–optical and neutron–interferometric methods.

3. ADVANCED LIQUID HYDROGEN CNS AT THE BRR

3.1. Introduction

The specific importance of the implementation of an advanced CNS at the Budapest Neutron Center (BNC) is emphasized both in the Report of the Large-Scale Facilities Panel — 1995 and by the Joint Meeting of UNESCO Physics Action Council and CEI Scientific Council (Vienna, 1995). (CEI refers to the Central European Initiative.) In the Joint Meeting, it was stated:

> "Budapest Neutron Center: ... In reconfirming CEI's evaluation on the quality of the Budapest Neutron Center, the joint CEI-UNESCO Council indicates as crucial the support of the acquisition, to the Center, of the planned cold neutron source, which would allow a very definite improvement in its capabilities to become an international focal point of interaction in multidisciplinary research activities."

The advanced liquid hydrogen CNS for cold neutron beam facilities is used for the production of neutrons with wavelength of more than 4 Å (<5 meV) at the 10 MW Budapest Research Reactor (BRR) of the BNC. The main BRR reactor parameters are flux at a core-trap: 2.3×10^{14} n·cm^{-2}·s^{-1}; at a thermal beam exit: ~ 3×10^{9} n·cm^{-2}·s^{-1}; core: 220 fuel assemblies (36% ^{235}U); Be reflector (200 mm); operation: ~3500 h/a (typically, 15 × 10 d cycle).

The CNS concept, project development, design, manufacturing, installation, as well as commissioning have been executed jointly by the CNS group from PNPI and specialists from the BNC and Hungarian companies. The construction of the CNS was supported by Hungarian national funds, the IAEA and the European Community. It was the first time that specialists from different countries have designed and manufactured a CNS in an international collaboration under IAEA patronage.

The technical documentation and the Safety Analysis Report of the CNS were completed and approved by the national regulatory body in 1997. The

fabrication, installation and out-of-pile tests were then performed during 1998–1999. In 2000, installation into the BRR as well as final commissioning were provided and the operation licence was obtained in January 2001. The installation of the CNS has been followed by the replacement of the obsolete neutron guides with a new supermirror guide system both for the in-pile and the out-of-pile part. Five instruments (also partly modernized) have been (re)installed at the cold neutron beams and since early 2001, the 'CNS facility' is routinely operated.

3.2. Cold source construction

The main components of the CNS system are:

— In-pile part (LH_2 moderator cell, vacuum containment, n-guides);
— Refrigerator system (He cold box, compressors, pipelines, buffer tank for helium);
— H_2 system with the buffer volume;
— Auxiliary equipment (vacuum systems, etc.);
— Control and data acquisition system;
— Beam shutters and neutron guides.

The following is a detailed description of those components.

The in-pile part (Fig. 17) contains the main functional element of the CNS system, which is the moderator chamber filled with about 0.5 L liquid hydrogen. The Al alloy cell is placed at the end of a horizontal tangential beam tube in a hole of the Be reflector around the core, positioned close to the maximum of the thermal neutron flux distribution. The design concept [9] realized for the CNS is the direct cooling of the condensed hydrogen in the moderator cell. In most cases, the nuclear heat load of LH_2 or LD_2 moderators

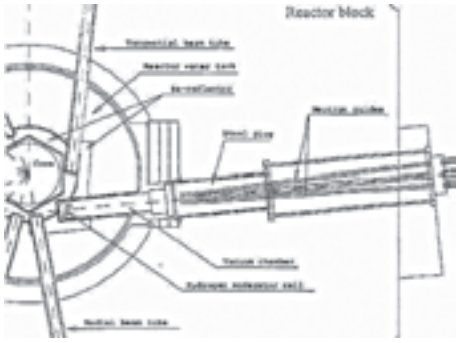

FIG. 17. Layout of the CNS plug in the BRR's horizontal channel.

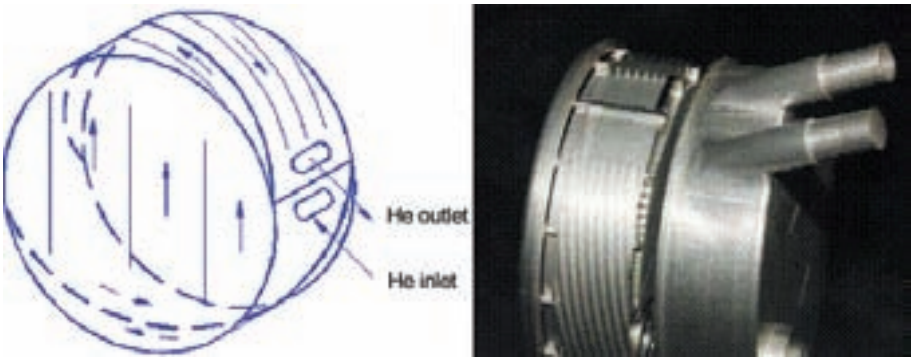

FIG. 18. Direct cooling of the moderator cell at the BRR.

is usually removed in heat exchangers cooled by cold He gas. In the BRR case of a small moderator cell, the relatively low estimated heat release (100–150 W) makes the implementation of a heat exchanger built together with the LH_2 moderator feasible, i.e. the direct cooling of the condensed hydrogen in a double walled moderator cell by the through flow of cold He gas. This novel technical solution provides excellent performance in cold neutron output as well as improved safety characteristics due to the extra inert gas barrier around the hydrogen containing moderator chamber, preventing hydrogen/air mixture formation. The scheme of direct cooling of the chamber is presented in Fig. 18. This also enables the minimization of the necessary amount of hydrogen and makes the out-of-pile hydrogen system far less complex. The liquid H_2 moderator cell is surrounded by an explosion proof vacuum containment, made of Al alloy and cooled by ambient temperature He gas flow. The in-pile plug also has a steel shielding part; this includes the bunch of three supermirror coated neutron guides, as well as the cryogenic pipelines connecting the in-pile part to the He refrigerator system plus all vacuum, H_2 and He volumes. In Fig. 19, the photograph shows the CNS plug during the out-of-pile tests with a front view of the Al alloy explosion proof vacuum containment and with the moderator cell inside. In addition, the steel part of the plug is seen, containing the in-pile guide sections and pipelines. In Fig. 20, the photograph shows the moderator chamber with all hydrogen and helium pipelines, installed in the vacuum containment. These tubes penetrate through the stainless steel flange with an aluminium membrane in the central part, which allows increasing the extraction of the cold neutrons from the cell.

The cryogenic system of the CNS serves to dissipate the heat released in the liquid hydrogen as well as in the construction materials of the moderator cell (walls and pipelines) due to nuclear radiation, and to keep the hydrogen in the condensed phase. The basic part of the cryogenic equipment (Linde TCF 10

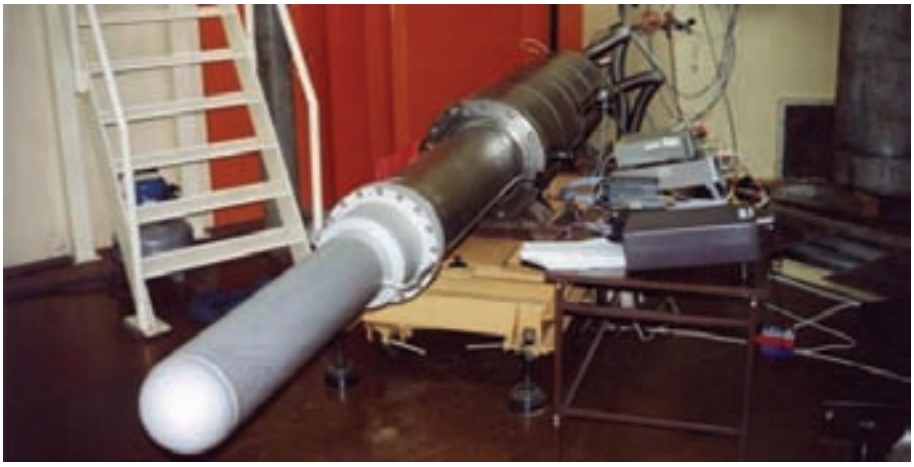

FIG. 19. Cold test of the CNS in the reactor hall before installation into the BRR.

FIG. 20. The CNS moderator cell with supply helium and hydrogen tubes.

model) is a helium gas refrigerator, which consists of a turbo-expander type cold box (fed by two screw compressors, 18 and 37 kW). A part of this helium system is the circuit for the ambient temperature He gas cooling of the blanket of the vacuum casing surrounding the LH_2 moderator as well as the gas purifier unit and the gas analyser (mass spectrometer).

The LH_2 moderator of the hydrogen system is connected to the 500 L buffer volume placed outside the reactor hall by a single hydrogen transfer line, which is normally open both in warmed up and cooled down state. For safety reasons, the whole hydrogen inventory is included in the supervised inert gas (He) containment.

An annex building (120 m^2) of the research reactor hall provides housing for the He refrigerator and the H_2 system. The annex also houses the auxiliary equipment, including the various vacuum systems, the safety He inert gas blanket surrounding the H_2 containing parts of the CNS assembly, and the H_2 system with the 0.5 m^3 double walled buffer volume (also lined with a He

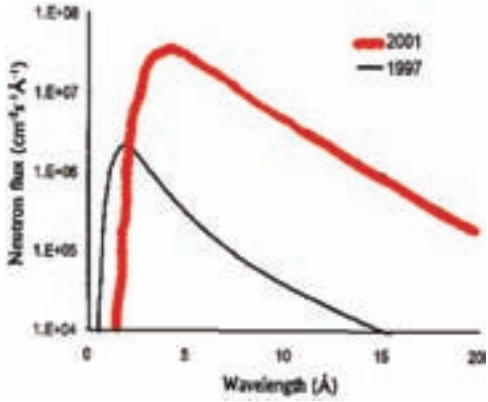

FIG. 21. Neutron spectrum measured by gold foil activation and time of flight spectrometer.

blanket). For safety reasons, the pneumatic valves of the vacuum and of the H_2 systems are also located inside the He gas filled boxes in the hydrogen room.

The control system, based on industrial standard programmable PLC units, ensures the full automatic control of the CNS equipment. The operational parameters and the status of all systems of the CNS assembly can be supervised on the control panels located in the research reactor hall and the reactor control room. Those parameters are also continuously monitored by a dedicated PC system.

The new supermirror neutron guide system starts at a distance of 1.15 m from the hydrogen cell. The first 6.5 m part of the guides, being the nearest to the core face, is aligned in vacuum jackets. Three independent beam shutters are placed in this section. Their novel construction enables individually opening and closing each of the three neutron guides under continuous vacuum by inserting into the beam path either a 0.5 m guide section or a neutron/gamma absorbing sandwich structure.

The operating parameters of the CNS are summarized in Table 3, including the design values defined at the beginning of the project. The major deviations between design and operating parameters can be explained by the fact that the moderator cell volume was considerably increased during the project implementation in order to optimize the illumination of the guides [10].

Figure 21 shows the spectral distribution and the calculated gain factor with reference to the old (thermal) guide system. Following the expectations, the Maxwell peak has moved to ~4 Å and the neutron flux gain factor (compared with the former situation) amounts to 30–60, depending on location and wavelength. This gain is composed as follows: a factor of 7–10 due to the CNS, a factor of 2 due to the elimination of losses replacing the old shutter

TABLE 3. OPERATING PARAMETERS OF THE BCNS[a]

Parameter	Design value	Operation value
Hydrogen cell volume (L)	0.35	0.49
Total heat load in the cell (W)	142	210
Pressure in the vacuum containment (MPa)	10^{-5}	10^{-5}
H_2 pressure at warm state (MPa)	0.22	0.36
H_2 pressure at cold state (MPa)	0.15	0.29
Cold helium mass flow (g/s)	10	10.5
Cold helium inlet temperature (K)	14	19
Cold helium outlet temperature (K)	15	23
Pressure drop across the helium loop (kPa)	14	45

[a] BCNS: Budapest Cold Neutron Source.

(obsolete construction) and another factor of 2–3 due to the supermirror guides.

The CNS at the BRR significantly increased the experimental capability of the 10 MW research reactor and placed it at the same level as high flux reactors, in terms of being an instrument for solid state physics research.

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7.5. SUPERCRITICAL HYDROGEN COLD NEUTRON SOURCE OF THE HIGH FLUX ISOTOPE REACTOR

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1. INTRODUCTION

In February 1995, the Deputy Director of the Oak Ridge National Laboratory (ORNL), United States of America, formed a group to examine the need for upgrades to the High Flux Isotope Reactor (HFIR) system in light of the cancellation of the Advanced Neutron Source Project. One of the major findings of this study was that there was a need for the installation of a cold neutron source (CNS) facility in the HFIR research reactor. In mid-1995, a team was formed to examine the feasibility of retrofitting a liquid hydrogen cold source facility into the HB-4 beam tube. These preconceptual studies were completed and documented in December 1995 [1].

Funding was received in 1996 to initiate the development of a reference design concept. This led to the documentation of a reference concept in 1998 [2]. Between 1996 and 1998, major changes in the design ground rules had an impact on the design approach that led to significant modifications in the concept:

— It was determined that it would be possible to replace the existing HB-4 beam tube with one that was significantly larger. This allowed for a larger moderator vessel and also created more space in the beam tube for hydrogen transfer lines.
— Later it was determined that a supercritical hydrogen system was a more stable system under transient conditions and that it could be implemented with little impact on the design concept. Thus, the reference concept was changed from a liquid hydrogen system to a supercritical hydrogen system operating between 14 and 15 bar pressure.

Since 1998, the project has proceeded with detailed design and procurement of key components. Present expectations are that over the next 18 months, the system will be installed and tested on the HFIR site, but not in the reactor. Following the successful completion of the out-of-pile testing, the system will be reconfigured with the CNS moderator vessel and new HB-4

beam tube installed in the reactor. A startup testing programme will then be implemented.

2. FUNCTIONAL REQUIREMENTS

There are a number of key requirements and goals for the design of the HFIR CNS that are documented in Ref. [2]. However, there are two major requirements that drive most of the design considerations:

- The purpose of the HFIR CNS is to increase the available neutron flux delivered to instruments at wavelengths from 4 to 12 Å. Optimization is to be based on the neutron brightness (/s/cm^2/steradian/). The gain factor on brightness, as measured on HB-4, for these wavelengths should be comparable to existing hydrogen CNSs.
- The HFIR CNS facility will be designed such that there is a low probability (less than 1×10^{-6}/a best estimate frequency) that neither the reactor nor the public will be endangered by accidents that occur within the CNS or as a result of the CNS facility interacting with the reactor or its safety systems. In addition, the design and operation of the HFIR CNS will follow National Aeronautics and Space Administration (NASA) guidelines and the US Department of Labor Occupational Safety & Health Administration (OSHA) standard 29 CFR 1910.103 for the use of hydrogen in either a gas or liquid state.

3. REFERENCE CONCEPT DESCRIPTION

In simple terms, the HFIR CNS system consists of a variable speed magnetic bearing hydrogen circulator that forces hydrogen at 15 bar pressure to a moderator vessel in the nose of the HB-4 beam tube returning flow to a helium heat exchanger and back to the circulator. A total of approximately 3 kW of heat is deposited into the hydrogen and removed by the helium heat exchanger. The helium is in turn cooled by a refrigeration system that can provide up to 3.8 kW of cooling at 20 K. Additional information on the main components is provided in the following discussion.

3.1. Pump module

The pump module is the heart of the system and consists of three variable speed magnetic bearing circulators (only one of which is in operation at any

given time), temperature and pressure monitors, and piping and isolation valves that allow the isolation of any given circulator. The pump module is vacuum insulated and also includes a helium cover gas barrier around all hydrogen systems. At this time, the variable speed circulators have been fabricated and tested, and the fully functioning pump module is expected to be delivered to ORNL by the end of 2003.

3.2. Hydrogen transfer lines

Transfer lines are required to transport cryogenic hydrogen between discrete components of the system. All cryogenic transfer lines are vacuum insulated and include a helium cover gas region. A semiflexible steel tubing system designed for hydrogen flow is used wherever possible. The first of three main transfer lines is expected to be delivered in October 2003. The remaining two sections of transfer line will be ordered as soon as the first section passes acceptance testing.

3.3. Moderator vessel

The moderator vessel will be located in the nose of the new HB-4 beam tube. The material used in the fabrication of the vessel is aluminium 6061-T6. The moderator vessel (shown in Fig. 1) is made by machining two pieces out of a solid block of aluminium and joining them with a single electron beam weld. The electron beam weld has been tested to failure which occurred at about ten times the normal operating pressure. Extensive flow modelling within the moderator vessel has been performed to ensure that adequate cooling is provided for all surfaces of the moderator vessel.

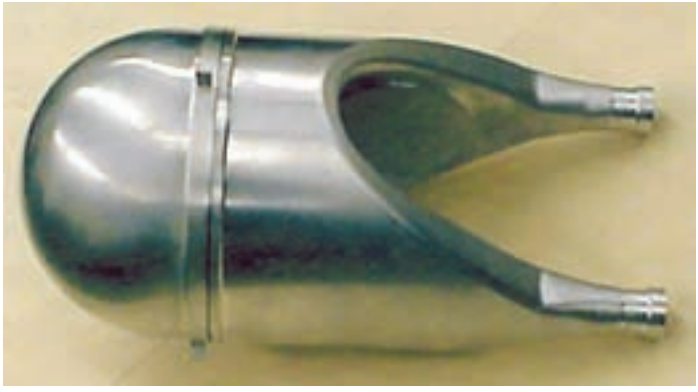

FIG. 1. HFIR CNS moderator vessel.

3.4. Heat exchanger

The heat exchanger provides a vacuum insulated interface between the hydrogen loop and the helium refrigerator system. Temperature sensors within the module provide temperature control during cool down to prevent possible freezing of the hydrogen. The heat exchanger is of the aluminium core type used in the construction of cryogenic refrigerators. The heat exchanger is surrounded by a double walled containment vessel. The inner wall provides vacuum insulation and the outer wall provides space for an inert blanket. The heat exchanger has been fabricated and has been delivered to ORNL.

3.5. Refrigerator system

The refrigerator provides the cooling power needed to maintain the hydrogen in the moderator vessel at approximately 20 K. The refrigerator system is composed of five major equipment items: a bank of five helium screw compressors, a vacuum insulated cold box that contains all interstage heat exchangers, a second cold box containing four two-cylinder piston expanders, a motor control centre, and an inventory control system. Refrigeration power depends on the number of compressors and expanders used, but fine operational control is provided by in-line heaters in the helium refrigerant circuit. The complete refrigerator system is located in a dedicated separate structure outside of the research reactor building. The refrigerator system has been installed and has been satisfactorily tested beyond the design requirements.

3.6. Support systems

Although the CNS system has been described in simple terms, there are a number of complex systems that are required to make the CNS functional. These systems include the vacuum systems, gas handling systems, inert blanket systems, instrumentation and control systems, and vent systems. These systems are in various stages of fabrication or design, but are expected to be completed during the first half of 2004.

4. CNS PERFORMANCE AND OPTIMIZATION STUDIES

A number of analyses and optimization studies have been performed to address the physics performance of the HFIR CNS.

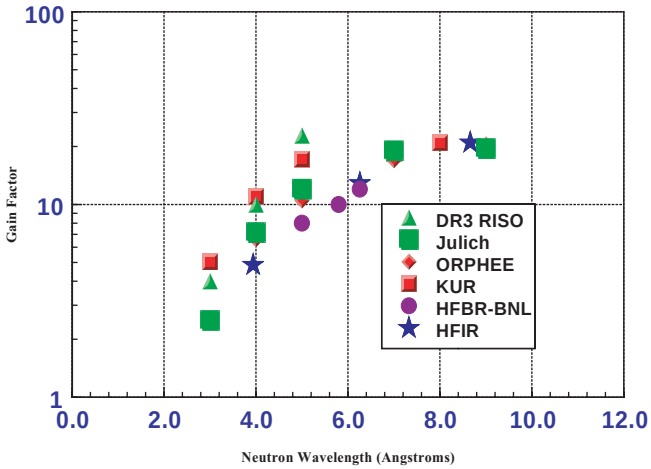

FIG. 2. Comparison of estimated HFIR CNS gain factors with existing hydrogen CNS.

4.1. Predicted CNS performance

As stated in Section 2, the purpose of the CNS is to provide gains in the number of 4–12 Å neutrons comparable to that provided by other hydrogen CNSs. Figure 2 provides a plot of the calculated gain factors for the HFIR CNS and compares them with gain factors determined for other existing hydrogen CNSs. As seen from this figure, the estimated gain factors are comparable with those obtained by several existing hydrogen CNSs at other reactor facilities around the world. The estimated brightness of the cold neutron beam is provided in Fig. 3.

4.2. Re-entry cavity in CNS geometry

It has been known for some time, based on measurements at the Institut Laue-Langevin (ILL), that reentry cavities in a deuterium CNS could lead to significant gains in the cold neutron current down the beam tube. In addition, the experience at the National Institute of Standards and Technology (NIST) research reactor implies that such gains can also be achieved with re-entry cavities in hydrogen CNSs. Therefore, early in the development of the HFIR moderator vessel geometry, considerations were made to design the transition from hydrogen flow lines to moderator vessel in a manner that would create a geometry resembling a reentry cavity. The analysis performed for this arrangement showed an approximate 30% increase in the cold neutron current down the beam tube when compared with other geometries.

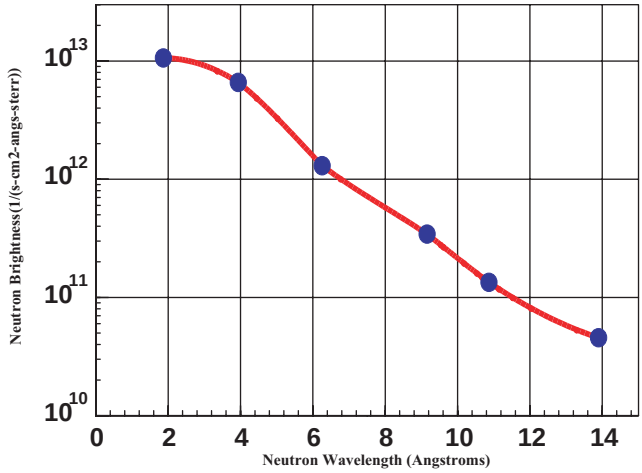

FIG. 3. Predicted neutron brightness for the HFIR cold source.

4.3. Impact of ortho/para hydrogen ratio

Early infinite slab parametric studies [1] implied that the orthohydrogen to parahydrogen ratio would have a significant impact on the performance of the HFIR CNS for the moderator thickness being considered. This was supported by evaluations performed at NIST. However, when various ratios were used in the model of the HFIR CNS, the results documented in Ref. [2] indicated that the ortho/para hydrogen ratio had very little impact on the performance of the HFIR CNS for the neutron wavelengths of interest. This was a concern to the designers because the results were contradictory to the previous studies. As a result, a Monte Carlo study was performed where the HFIR CNS geometry was changed to the NIST CNS geometry in a series of steps. It was found and documented in Ref. [2] that the actual geometry of a CNS had a significant impact on the effects of the ortho/para hydrogen ratio. It was determined that the impact was significantly different for the smaller HFIR CNS located in a beryllium reflector when compared with the larger NIST CNS. The ORNL model of the NIST CNS, in fact, exhibited ortho/para ratio effects very similar to those advertised by the NIST organization. This provided confidence that the HFIR CNS was relatively unaffected by the ortho/para hydrogen ratio and thus, the uncertainty in this ratio was no longer considered an issue.

5. HFIR CNS OPERATING MODES

The HFIR CNS system will have two approved operating modes. The first mode of operation or primary operating state represents the normal operating mode with hydrogen temperature between 18 and 21 K. The second operating mode has been defined as a standby operating state and will be used when the normal refrigerating system is off-line and cooling is provided by once through flow of liquid nitrogen from a 42 m^3 tank located on a hill above the CNS refrigerator equipment. This mode of operation provides hydrogen flow at reduced pressure with a temperature between 100 and 120 K, which is sufficient to cool the moderator vessel with the research reactor operating at normal power levels.

6. SAFETY PHILOSOPHY AND PRINCIPAL SAFETY FEATURES OF THE HFIR CNS

The primary purpose of the CNS safety evaluations is to ensure that the CNS and any of its systems have no impact on the reactor safety systems or the overall safety level of the research reactor. A secondary purpose of the CNS safety evaluations is to ensure that on-site personnel and equipment also are not adversely impacted by the CNS or any of its systems. The safety goal for effects on the research reactor is to have a low probability ($< 1 \times 10^{-6}$/a best estimate frequency) that fuel will be damaged or a nuclear safety function lost because of the CNS or any of its equipment. The safety goal for the CNS facility is that a major release of hydrogen into the research reactor building should be a low probability limiting event, with no loss of the research reactor confinement safety function.

The CNS safety work is proceeding in parallel with the development of the CNS design. At each stage of the design and testing, the safety evaluations have provided input to the decision making. The final safety evaluations of the CNS facility will be used as an authorization basis document to support the safety review by ORNL safety committees and by the USDOE. This analysis is expected to be an attachment to an Unreviewed Safety Question Determination (USQD) document that will be submitted to add the CNS facility to the HFIR authorization basis. The appropriate amount of summary descriptive and safety information is expected to be inserted into Chapter 10 of the HFIR SAR following initial operation of the CNS.

The safety of the CNS is being ensured by multiple approaches. Adequate conservatism is being given to the design of the hydrogen boundaries to ensure a high integrity boundary. A quality assurance programme is in place to control

design, analysis, procurement, fabrication, installation and operation of the facility. Development testing is being carried out as part of the design, and final testing will be performed as part of the startup of the facility. An accident analysis is being performed to examine the bounding aspects of the design, as well as the probability and consequences of various event sequences.

The HFIR CNS safety design philosophy is a defence in depth approach that provides several means to avoid accidental contact between hydrogen and air, and also provides the means to mitigate a hydrogen release, given that a release is assumed to occur. The principles of conservatism, simplicity, redundancy, fail safe design, and passive safety features are included in the design as much as possible.

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Part 8

NUCLEAR SAFETY TESTS AND DEDICATED FACILITIES

8.1. INTRODUCTION TO NUCLEAR SAFETY TESTS AND DEDICATED FACILITIES

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1. JUSTIFICATION OF THE TOPIC

This Compendium is intended to cover the variable field of research reactors and their tasks and experimental capabilities as comprehensively as possible. To approach this aim, it was necessary to include research reactor plants which solely or predominantly serve to investigate safety issues in terms of the fuels and the cores of nuclear power plants, including related accident scenarios. As long as those plants are multipurpose research reactors, these safety related tests and the facilities needed to perform them — even when they completely occupy a research reactor plant for a certain period — fit seamlessly into the spectrum of tasks of the Compendium.

There are research reactor plants, however, which are designed and used entirely for one task or one series of safety related experiments. Furthermore, some research reactors are built only as a test facility for a power reactor technology, which could contribute substantially to the future nuclear power generation as seen from the point in time of their construction. Nuclear technology has relied on such test plants in the past, which are included in Part 9 as dedicated facilities. Consistently, such plants are necessary in order to develop a new power reactor technology.

2. EXAMPLES OF PLANTS FROM THE HISTORY OF RESEARCH REACTORS

It would be an endless discussion to try to compile detailed histories of all research reactor plants which have or may have fallen into this category. It might be helpful, nevertheless, to mention a few and to characterize them and their purpose by one or two sentences. Six examples only are selected for this illustration:

— MZFR, a plant at Karlsruhe, Germany, which served to test a concept with heavy water moderation and light water cooling and was shut down when further concept development was stopped in Germany;

— HTTR, a temperature test reactor at Tokai Mura, Japan, designed and built to test the graphite moderated gas cooled reactor concept for Japan's future nuclear power generation;
— MASURCA, an air cooled critical facility at Cadarache, France, built to investigate and measure data for the qualification of codes in terms of fast reactors;
— GETR, a test reactor at Pleasanton, United States of America, testing boiling water technology prior to its implementation as one of the two dominating technical concepts for nuclear power generation;
— ARBUS, a test reactor at Dimitrovgrad, the Russian Federation, proving the technology of an organic cooled and movable power reactor;
— DRAGON, a test reactor at Winfrith, United Kingdon, testing systems and components under He cooling at very high temperatures.

These examples show that such dedicated facilities played a major role in the national programmes towards nuclear power in all major countries having developed that technology.

The contributions on MYRRHA serve as an example that dedicated facilities are still a step along the path when approaching a new line of nuclear plants, with or without power generation.

8.2. THE MOL-7C IN-PILE LOCAL BLOCKAGE EXPERIMENTS IN THE BR2 REACTOR AT SCK·CEN

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1. BR2 RESEARCH REACTOR

The BR2 research reactor, operated by the Belgian Nuclear Research Center (SCK·CEN), is one of the major material testing reactors (MTR) in the world. It is a tank-in-pool type reactor, cooled by forced circulation of light water under pressure. The light water also serves as a moderator.

The BR2 core is composed of a beryllium 'matrix', i.e. an assembly of hexagonal beryllium prisms with cylindrical bores (these form the reactor channels), which together form a cylindrical structure. This matrix occupies the central part of the reactor vessel and offers 79 cylindrical channels where fuel assemblies, safety and control rods, experimental rigs, irradiation devices or reflector plugs can be loaded.

Due to this specific design, all BR2 fuel assemblies consist of concentric cylindrical tubes. They have an active fuel length of 762 mm, derived from the length of the beryllium matrix. The diameters of the two types of available bores in the beryllium matrix determine the outer diameters of the fuel assemblies, i.e. either 84 or 200 mm. The standard BR2 fuel assembly contains 400 g ^{235}U in the form of UAl_x with burnable absorber material homogeneously mixed into the fuel 'meat'. The uranium is highly enriched with a density of about 1.30 g U/cm^3 'meat'.

The main objective of the BR2 is the irradiation of fuels and materials under high neutron flux (maximum thermal neutron flux for irradiation is of the order of 10^{15} n·cm^{-2}·s^{-1}). These fuels and materials are irradiated in experimental rigs, the complexity of which depends on the nature of the irradiation and the objectives of the irradiation programme.

The reactor core configuration depends on the experimental requirements, as well as on the criteria and rules which have to be fulfilled to guarantee the safety of the plant during all the operational states. This means that the reactor core loading (type, position and burnup of the fuel assemblies, position of the control rods, position of the experimental rigs, adjunction of absorbing screens, etc.) will be defined subject to the specifications of the experiments and is readjusted for each cycle. Therefore, the operating power

can vary from cycle to cycle. The total heat dissipation capacity of the BR2 cooling system is 125 MW.

2. EXPERIMENTAL PROGRAMME MOL-7C

In order to understand the facilities and the equipment to be applied for the tests, the experimental programme and its aims have to be briefly described.

In the safety analysis of liquid metal reactors (LMRs), the propagation of local cooling disturbances in individual fuel assemblies is considered to be a possible cause of severe accidents. The major cause of such disturbances is a local blockage of a fuel subassembly in its fissile zone, which may be caused by the swelling of defective fuel pins or the deposition of inert or fuel particles at the narrow passages of a pin bundle. During the initial phase, such blockages are expected to have a significant porosity. The resulting residual coolant flow may continue to provide for efficient cooling of the blockage and the related local increase in temperature may remain small. However, if the blockage is not detected before the sodium boiling temperature is reached, the evaporation of sodium may result in a substantial decrease of the residual sodium flow and subsequently cause a fast temperature increase. This would lead to rapid clad and fuel melting with the possibility of fault propagation, for example, by thermal interaction between the molten material and the coolant, by material relocation and by the formation of secondary blockages or by melt through of the wrapper tubes that form the boundary to the adjacent subassemblies. This type of accident is often addressed as local subassembly fault (LSAF) propagation.

A general strategy to prevent severe accidents in nuclear reactors is to provide lines of defence (LOD). In the case of LSAFs in LMRs, the main LODs are the prevention of faults by adequate design and quality assurance measures, as well as by the detection of the fault and the subsequent reactor shutdown if this LOD might fail. The most important related system is the delayed neutron (DN) detection system. It becomes effective as soon as fuel pin failures in the fissile zone appear. The melting of the cladding is accompanied by a continuous and significant increase in the DN detector signal. Protective actions become efficient only with delays, depending on the particular response and signal processing time of the respective system. For example, the increase of the DN signal can be perceived only after a delay determined by the coolant transport time from the core to the DN monitor, which in LMRs usually amounts to 10–30 s. It must be demonstrated that during this time, the fault sequence will not propagate to such an extent that the

coolability of the fuel subassembly as such is endangered. Beyond any detection, inherent safety (i.e. self-limitation of the fault without any safety directed action) is of high interest, but its suitability as an effective LOD must be evaluated.

The aforementioned problems were investigated in the MOL-7C experimental programme, which was conducted in the in-pile sodium loop (IPSL) at the BR2 under test conditions that were as close as possible to typical LMR fuel assembly operating conditions. Each test section for the IPSL contained a small oxide fuel pin bundle with a porous local blockage simulating the local cooling disturbance. The blockage size represented an upper limit of potential situations at LMRs. The blockage porosity was designed such that under representative LMR power conditions, local boiling was generated as the initial state of the fault. This means that the formation of a blockage and the possibility of its early detection were not the subject of the investigations. Accordingly, the general objectives of the MOL-7C experiments have been defined as follows:

— Investigation of the development of local subassembly faults in oxide LMR fuel assemblies starting with local boiling inside a porous blockage, of which important issues are:
  • Development of the fault inside the blockage;
  • Conditions and mechanisms for the propagation of the fault from the fuel pin bundle with the blockage to the entire subassembly;
  • Timescale of such a fault and propagation processes;
  • Dependency of these processes on parameters, such as bundle geometry, blockage location in the bundle, blockage material, fuel pin gas inventory and burnup of the fuel;
— Investigation of the response signals of different detection systems with the main emphasis on the DN detection system;
— Investigation of the long term behaviour of a fuel pin bundle with severe local damage at nominal power to explore the capability in terms of inherent safety.

An additional objective of the last experiments was the investigation of the fission product release into the sodium from the failed fuel pins (referred to as source term measurement (STM)). These experiments were performed as a joint programme of the Forschungszentrum Karlsruhe (FZK), Germany, and the Belgian Nuclear Research Center (SCK•CEN), Belgium, with the partial support of the Joint Research Centre, Ispra, Italy.

SCK•CEN was mainly responsible for:

— Detailed design of the test equipment;
— Construction of components and assembly of the loop;
— Assembly of the bundles of pre-irradiated fuel pins (MOL-7C/4-5-6);
— Loop instrumentation;
— Performance of experiments at the BR2;
— Realization and evaluation of the STM.

The entire programme, including the post-irradiation examination of the test bundles, covered a period of almost 20 years; the final evaluation of all tests was completed in 1995.

3. DESCRIPTION OF THE TEST FACILITY

The IPSL is designed for testing fuel pin bundles at the BR2 under conditions similar to those in typical LMRs. That experimental rig is composed of three major parts:

— In-pile section with a complete primary sodium circuit;
— Out-of-pile equipment with a secondary He circuit, a tertiary water circuit and various auxiliary circuits;
— Instrumentation system including the data acquisition system.

The following discussion gives a global description of the in-pile section with the fuel pin bundle, the means for the blockage and the associated instrumentation.

3.1. In-pile sections

Two different versions of the in-pile section were used for the experiments. The general configurations are schematically shown in Fig. 1. The single unit concept (left section in Fig. 1) was used in the MOL-7C/1-2 and the MOL-7C/3 tests with fresh fuel pins. To allow the remote loading of irradiated fuel pins, the in-pile sections used in the subsequent tests were divided into two parts: the (inner) test train and the (outer) in-pile section (right section in Fig. 1). The test train was assembled separately, including loading of the pre-irradiated fuel pins, and was subsequently inserted into the central bore of the in-pile section. In both versions, the in-pile section had a length of 8.5 m and a diameter varying from 122 mm (reactor core region) to 450 mm (at the top of the section). The

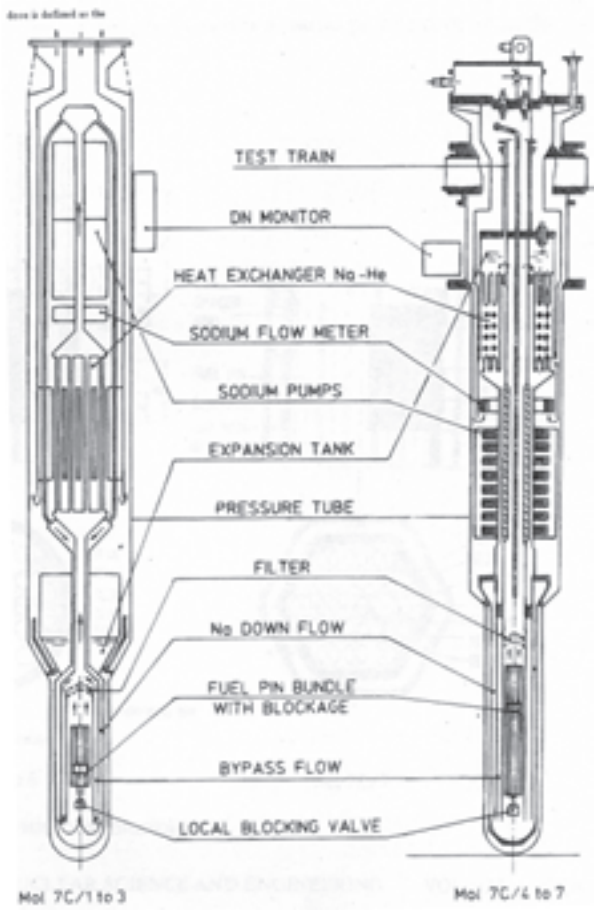

FIG. 1. General arrangement of the in-pile sections.

in-pile section contained the complete primary sodium circuit mainly consisting of the sodium expansion tank, the sodium–helium heat exchanger, two electromagnetic sodium pumps, and the test section with the fuel pin bundle. The whole system was arranged concentrically inside a pressure tube, which was surrounded by a cadmium screen in the core region, to provide an epithermal neutron flux in the thermal flux environment of the BR2.

The general flow scheme in the lower part of the loop was essentially the same in all tests. Sodium flowed downward in an outer annular gap to the lower end of the loop and after a U-turn, it streamed upward through the test section as well as through the parallel annular bypass. A filter consisting of niobium wool enclosed between two stainless steel wire meshes was located at the test

section outlet to retain particles, particularly from the fuel, at the lower part of the loop.

3.2. Test section

The test sections containing the fuel pin bundle and the blockage means were arranged at the lower central part of the loop. The general arrangement of the blockage region is shown in Fig. 2; the most important data of the test section are summarized in Table 1. In the sequence of the tests MOL-7C/1 to MOL-7C/6 with central blockages, the bundles consisted of 30 fuel pins, 6 corner pins and 1 central dummy pin, arranged in two concentric wrapper tubes which were separated by a gap filled with stagnant inert gas. A 1 mm ZrO_2 layer covered the outer surface of the inner wrapper tube to provide protection against melt through. The fuel pin diameter was 6 mm, and the pitch to diameter (pitch: diameter) ratio was 1.31. In MOL-7C/7, a bundle of 19 fuel pins of 7.6 mm diameter each, in combination with a pitch to diameter ratio of 1.16, was inserted together with the means to effectuate a corner blockage, all wrapped by two concentric hexagonal tubes. In the gap between these two wrapper tubes, sodium circulated with a velocity of about 0.1 m/s. The size of the gap, the thickness of the wrapper tubes, and the sodium velocity were selected according to typical LMR design values.

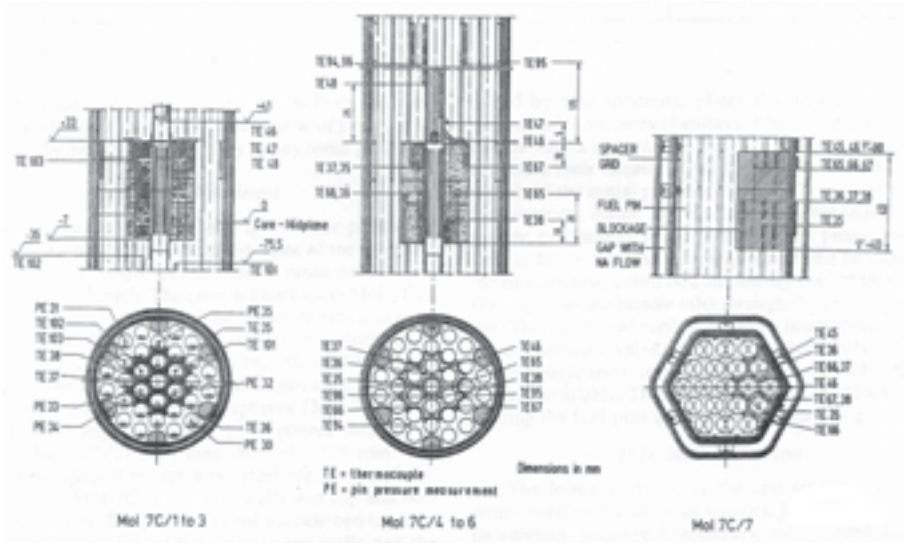

FIG. 2. Various test sections with local blockages.

TABLE 1. DATA COMPILATION FOR THE EXPERIMENTAL PROGRAMME

Test number	1	2	3	4	5	6	7
Bundle size (number of pins)	37 (30 fuel pins)			19			
Pin diameter (mm)	6			7.6			
Pitch/diameter ratio	1.31			1.16			
Pin length (mm)	800		1575	1562			
Fissile length (mm)	400		600				
Type of fuel	UO_2		UO_2-PuO_2				
Plutonium content (at.%)	—		30				
Clad material [a]	1.4970		1.4981	1.4970			
Burnup (at.%)	0.3		5	1.7	10	0.1	
Kind of blockage	Porous, 30% of bundle cross-section, 40 mm length						
Position of blockage	Central			Corner			
Blockage material	Inert (stainless steel)		Active (highly enriched UO_2)				
Year	1977	1978	1980	1983	1984	1988	1989

[a] According to the German standard.

The fresh UO_2 fuel pins used in the MOL-7C/1 to MOL-7C/3 experiments were manufactured especially for those experiments. The ^{235}U enrichment in the three rows of the bundle was adjusted such that the linear rating of all pins was about the same. High pressure filling gas in the pins simulated the free fission gas of irradiated fuel pins. The mixed oxide (MOX) fuel pins for the subsequent tests with 30 wt% plutonium were taken from irradiated or spare subassemblies from the KNK-II reactor at FZK. The constant enrichment of fissile material in all the pins caused a radial power profile in the bundle with deviations between –20% and +17% from the average value (highest values in the outer row of pins). In all the tests, the pins were configured using honeycomb grids.

3.3. Blockages

The porous blockages (see Fig. 2) were positioned near the axial midplane of the fissile zone of the fuel pins. They occupied about 30% of the bundle cross-section and were 40 mm in length. The central blockages (MOL-7C/1 through MOL-7C/6) included the six inner pins and parts of the pins of

the second row. The corner blockage in MOL-7C/7 was in contact with two flats of the inner hexagonal wrapper tube. The blockages consisted of a pervious cage filled with steel spheres (MOL-7C/1 and MOL-7C/2) or highly enriched UO_2 spheres (MOL-7C/3 through MOL-7C/7) with diameters of about 0.5 mm. The cages were formed by stainless steel meshes or perforated plates (MOL-7C/7) as sidewalls and top and bottom cover plates. The porosity of the particle bed, together with the permeability of the lateral cage walls and the gaps between the cover plates and the pins, allowed for a significant residual flow inside the blockage parallel to the main flow in the bundle. To ensure that the sodium boiling temperature in the blockage would be reached under nominal power conditions, the residual flow was reduced by two additional plates dividing the blockage horizontally into three chambers. The UO_2 particles were coated with a chromium layer of 7–10 μm thickness to ensure their chemical stability and to avoid high DN signals in the initial phase of the tests. To avoid too high temperatures inside the blockage and subsequent premature pin failures before the main test phase, particularly in the tests with high pin pressure, the blockages in the first six tests could be additionally cooled by sodium flowing from the bundle inlet through the central dummy pin. This additional cooling could be interrupted by closing a pneumatic valve, the local blockage valve (LBV), at the bundle inlet. In the MOL-7C/7 test, such a device was not available. The fuel inventory of the blockage plus the fuel pins amounted to 80–130 g.

3.4. Instrumentation

The loops, particularly the test sections, were equipped with numerous sensors, such as thermocouples, flow meters, pressure transducers, gamma radiation detectors and DN detectors. The instrumentation that was available inside and around the blockages is included in Fig. 2; a general view from the top cover of the BR research reactor is presented in Fig. 3. In this context, it is helpful to note the following particularities:

- Up to seven thermocouples were available to measure the temperatures at the blockage area.
- During the tests with fresh fuel pins, the internal pressure and the central temperature of some fuel pins were measured.
- To determine the temperature in the recirculation zone behind the blockage, several thermocouples were mounted to the central dummy pin downstream of the blockage (MOL-7C/1 through MOL-7C/6). This arrangement was not possible in MOL-7C/7. For this test, two thermocouples were attached to the blockage top plate. Additionally, three

FIG. 3. Top cover of the BR research reactor with a MOL-7C loop installed in the central 200 mm channel and the connections of the instrumentation of the loop.

thermocouples were mounted to the spacer grid immediately downstream (MOL-7C/4 through MOL-7C/7).
— To detect a possible melt through, several thermocouples were provided in the bypass (all tests) and in the interwrapper gap (only MOL-7C/7). They were attached to the outer and inner wrapper tubes, respectively.
— The DN system was located beside the expansion tank of the loop and consisted of neutron counters, neutron ionization chambers and gamma chambers. For each detector type, a linear and a logarithmic device were used.
— For acoustic noise measurement, various types of microphones were installed at different distances from the test bundle outlet.
— An underwater device for gamma spectrometry was installed in the BR2 reactor pool to measure the activity concentration of fission products in the sodium (for the MOL-7C/6 and MOL-7C/7 experiments only).

3.5. Source term measurement device

3.5.1. Description of the system

In safety studies of a severe core accident, one generally uses transfer coefficients (the fraction of the fission products in the damaged zone released into the sodium) to estimate fission product release. With gamma spectroscopy,

it was possible in the MOL-7C experiments to quantify the amount of some fission products released into the sodium, to deduce the release fraction (fraction of the total fission product inventory released into the sodium), to calculate transfer coefficients, and to compare the latter with transfer coefficients recommended by the OECD Nuclear Energy Agency (OECD/NEA).

For both the MOL-7C/6 and MOL-7C/7 experiments, a gamma spectrometry measurement system was used to measure the fission product release into the sodium at the level of the expansion tank. The related measurement system consisted of a gamma sensitive detector (high purity germanium detector, pre-amplifier and liquid nitrogen Dewar vessel) inside a stainless steel watertight cover, equipped for underwater operation at 10 m depth. The sensitive part of the germanium detector was positioned in the centre of an 8 t weight lead container to protect it from unwanted gamma fields. An air filled collimator tube allowed gamma rays emanating from the sodium expansion tank to reach the sensitive part of the detector. A calibration facility between the in-pile section and the collimator tube enabled the positioning of calibration sources.

Because safety experiments such as MOL-7C are not repeatable, a data acquisition system was implemented using two completely independent pulse analysing and data recording chains, each equipped with a high voltage supply, an amplifier, an analogue to digital converter (ADC), and a hybrid multichannel analyser (MCA) computer. Additional electronics measured the total dead time of the amplifier–ADC combination.

3.5.2. Calibration sources and methods

The measured fission products activities extracted from the spectra are proportional to the fission product density at the visible volume of the expansion tank. To determine this 'proportionality' factor (also referred to as 'efficiency'), calibration sources of a known decay rate are measured in the same measurement geometry. Because the efficiency of the measurement set-up is energy dependent, calibration sources must be used that emit gamma rays in the energy range of importance (0.1–2.0 MeV).

For the MOL-7C/6 experiment, the absolute efficiency of the measurement set-up at 1.25 MeV was determined in hot conditions with a calibrated ^{60}Co source. However, using ^{60}Co calibration sources has the following disadvantages:

— As both sources (the ^{60}Co and the visible volume of the expansion tank) emitted gamma rays practically at the same energy, it was impossible to calculate the efficiency within the extended energy range from 0.1 to

1.4 MeV. Additional (off-line) measurements are needed to determine the energy dependence.
— These sources have a non-negligible self-shielding.

The additional measurements to determine the energy dependence of the efficiency were performed at the BR1 research reactor using ^{152}Eu sources. During the MOL-7C/7 experiment, the absolute efficiency curve could be determined in the relevant energy range (0.2–1.4 MeV). These europium sources were calibrated in the BR2 hot cells.

3.5.3. Data processing

Gamma detection is a discrete statistical process. During a given time interval, a gamma energy probability distribution is measured. The precision of determining the activity of a certain gamma line depends solely on the number of counted gammas. To obtain as accurate as possible estimates of the fission product activities, the number of counted gammas for each fission product should be large. This can be accomplished by increasing the measured activity (generally by changing the geometry) or increasing the total measuring time. For the MOL-7C/6 and MOL-7C/7 experiments, the measurement geometry and gamma emitting source were invariances of the experiment, and only the measurement time could be modified.

Choosing long measurement times contradicts the original goal of the experiment, which was to gain information on possible transients. To obtain the maximum amount of information on fission products released to the sodium during the transient phase, a short cycle time of 60 s with an effective measurement time of 54 s was adopted. The remaining 6 s were used to store the spectrum on tape and to print out essential information.

To reduce the possibility of missing activity transients at the level of the sodium expansion vessel, the two independent measurement chains connected to the detector performed measurements with overlapping measurement cycles. Before and after the transient phase, additional measurements were performed with longer measuring times of 10, 30 and 60 min.

4. CONCLUSIONS

The BR2 is well suited to execute such highly sophisticated experimental programmes involving the disintegration of a number of fuel pins, which may even be pre-irradiated.

Certainly the capability to pursue irradiation after the damage of the fuel pins is a very special feature. Although the BR2 is not a research reactor dedicated to safety experiments, it was clearly demonstrated that complex and demanding safety experiments can be executed to the full satisfaction of the customer. The very high neutron flux densities in a compact MTR core, the availability of large (200 mm diameter) experimental channels, the flexibility of the core configuration and the capability to offer representative irradiation conditions are major assets.

Authorization to execute this type of irradiation could be obtained because of the existence of a system of multiple barriers (confined integrated test loop, aluminium pressure vessel, containment building) and because a maximum amount of sodium allowed in an experimental device was already fixed in the course of the overall safety analysis of the BR2.

8.3. NSRR EXPERIMENTS ON LWR FUEL BEHAVIOUR UNDER REACTIVITY INITIATED ACCIDENT CONDITIONS

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1. INTRODUCTION

The Japan Atomic Energy Research Institute (JAERI) has conducted experimental and analytical studies of light water reactor (LWR) fuel behaviour under accidental conditions. One of the most important accidents to assess is the reactivity initiated accident (RIA) which could cause fuel failure and consequential damages to the reactor vessel and to the core internal structure. In 1975, JAERI started an experimental programme of RIA simulating pulse irradiation on LWR fuels using the Nuclear Safety Research Reactor (NSRR).

The first phase of that programme at the NSRR was conducted with fresh, i.e. unirradiated, LWR fuels. Over 1200 pulse irradiation experiments were parametrically performed under standardized test conditions. Fuel behaviour during the pulse irradiation, including cladding temperature, rod pressure and so on, was measured and the results were presented mainly as a function of peak fuel enthalpy. The Nuclear Safety Commission (NSC) of Japan utilized the experimental results to establish safety evaluation guidelines for reactivity insertion events at LWRs [1]. For fresh fuel experiments, test capsules for specific purposes have been developed, such as a visual observation capsule with a periscope and a high speed camera [2], and a capsule connected to a high temperature and high pressure water loop to generate BWR/PWR coolant conditions [3]. Research reactor fuels, including uranium-silicide type [4] and Triga [5] fuels, were tested as well. Unirradiated MOX fuel tests were also carried out [6] in cooperation with the Power Reactor and Nuclear Fuel Development Corporation (presently, the Japan Nuclear Cycle Development Institute, JNC).

The second phase of the NSRR programme started in 1989 and was directed to pre-irradiated fuel experiments. Prior to the experiments, the NSRR facilities were modified to enable handling and testing of high burnup fuels. Experiments were carried out on LWR fuel rods, which were irradiated in a commercial BWR/PWR or in the Japan Materials Testing Reactor (JMTR). Regarding irradiated MOX fuels, tests of prototype advanced thermal reactor (ATR) Fugen fuel rods have been performed in cooperation with the JNC since 1996 [7]. Results of

pre-irradiated fuel experiments have shown that high burnup fuel rods have a different failure mode from fresh rods and that the failure threshold in terms of peak fuel enthalpy decreases with burnup, mainly due to degradation of the cladding mechanical strength resulting from hydride precipitation [8, 9]. These experimental results were reflected in safety evaluation guidelines for the RIA of high burnup fuels, which was established by the NSC in 1998 [1]. The highest burnup levels of tested fuel rods so far are 61, 60 and 30 GWd/t for BWR, PWR and ATR/MOX fuels, respectively. The burnup of test fuel rods is being raised in response to the worldwide trend towards increased burnup.

This paper provides information on the NSRR facilities and experimental technique, and main results of fresh and irradiated fuel experiments.

2. FACILITIES AND EXPERIMENTAL TECHNIQUE

2.1. NSRR

Figure 1 shows vertical and horizontal cross-sections of the NSRR. The NSRR is a modified Triga annular core pulse reactor (ACPR). The Triga type

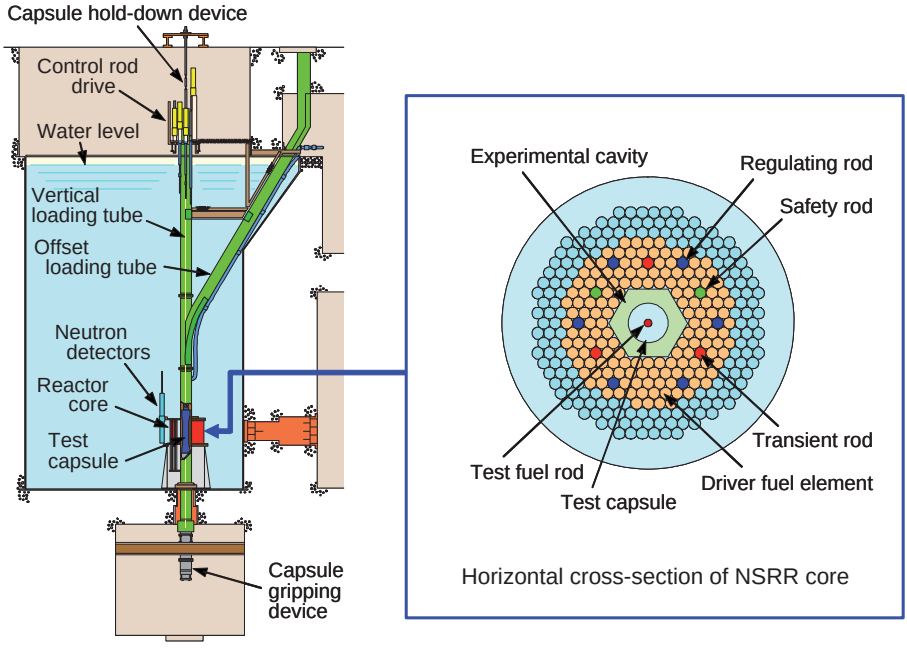

FIG. 1. Vertical and horizontal cross-section of the NSRR.

reactor is known for the unique nature of its strong negative feedback to reactivity insertion. The NSRR, in particular, is a modified type with enhanced negative feedback. The stainless steel cladding of the driver core fuel rod is dimpled to make circular protuberances on the inner surface of the cladding, which in turn keeps the fuel clad gap open and reduces the thermal conductance. Therefore, the peak fuel temperature is preserved, resulting in the enhanced temperature feedback. Such techniques enable the NSRR to produce a high power pulse reaching ~23 GW safely. Figure 2 shows the

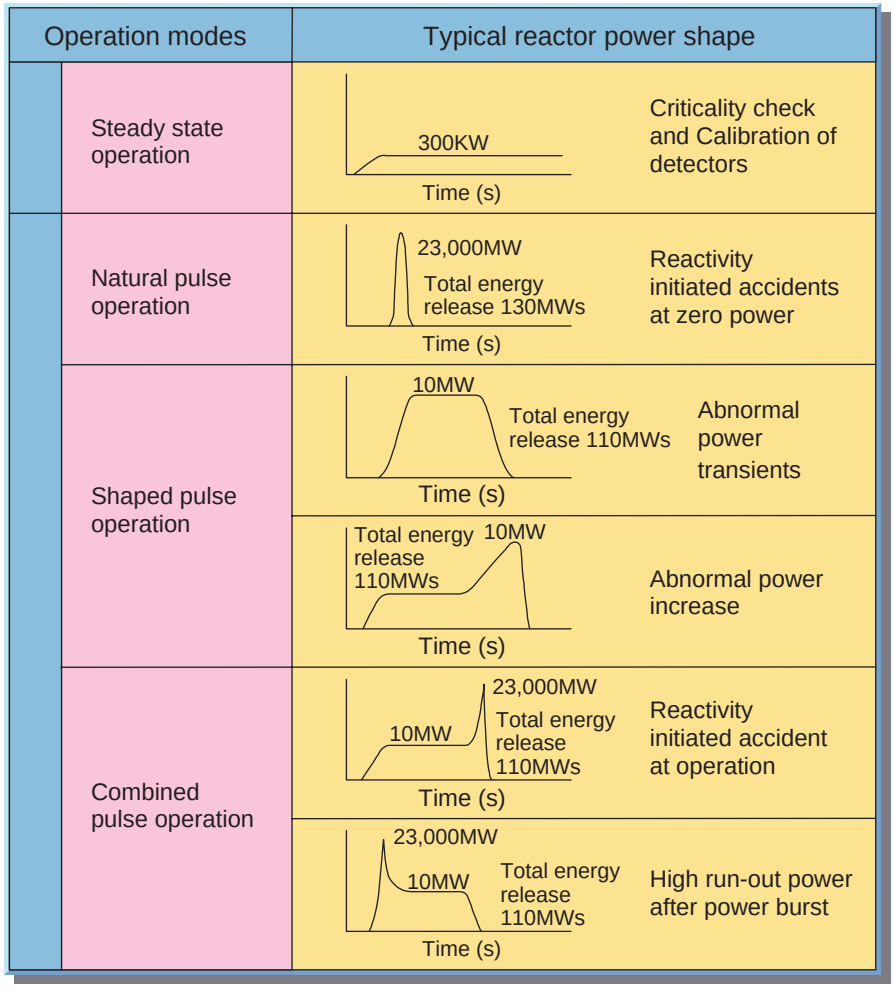

FIG. 2. NSRR operation modes.

available operation modes of the NSRR. Single pulse and more complicated power histories can be generated by using the transient rods and the regulating rods, which are control rods driven by pneumatic systems and electric motors, respectively. Most of the NSRR experiments have been performed with the natural pulse operation mode to simulate power excursion in the RIA. The pulse shape, which is specified with height and width, depends on the total energy release (i.e. time integrated power). The full width at half maximum and the power peak are approximately 4 ms and 23 GW, respectively, providing a total energy release of 130 MJ. Using the shaped pulse mode, power oscillation tests have been performed. The power history is shown in Fig. 3. These tests were intended to investigate fuel behaviour during an anticipated transient without scram (ATWS) [10].

In addition to the operational capability, the NSRR has a large experimental cavity at the core centre as shown in Fig. 1, which enables insertion of a test capsule with a diameter up to 200 mm into the reactor core and permits flexible design of the test capsule. JAERI has taken advantage of the NSRR features to carry out accident simulating experiments under various conditions.

2.2. Test capsules

JAERI has developed several types of capsules to test various fuels under different coolant conditions. Test fuels are presently categorized into UO_2, MOX, U silicide and Triga fuels. The pulse irradiation tests are performed

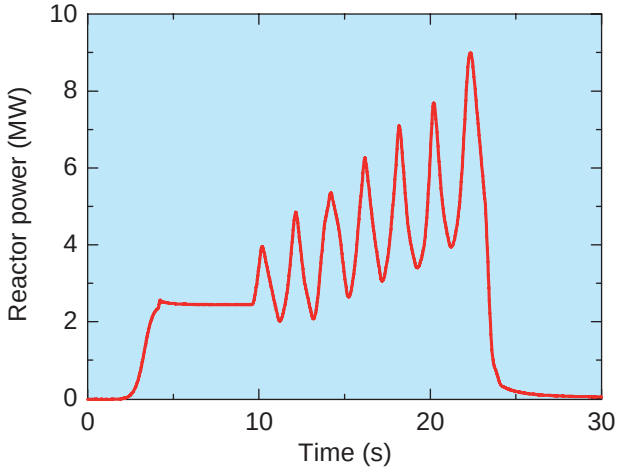

FIG. 3. Power history in power oscillation test.

under roughly two sets of coolant conditions; stagnant water at room temperature (~20ºC) and atmospheric pressure (~0.1 MPa), and stagnant/flowing water at BWR/PWR operation temperature and pressure.

2.2.1. Test capsules for fresh UO_2 fuels

Figure 4 shows the atmospheric test capsule for fresh UO_2 fuel rods, which is made of stainless steel and has an inner diameter of 120 mm. Most of the fresh fuel tests were performed with this capsule containing stagnant water at room temperature and atmospheric pressure. Various pieces of equipment can be installed in the capsule according to need, including an electric heater to raise water temperature, an electric pump for forced convection, a flow channel tube to adjust hydraulic diameter, etc. Bundled rods can also be tested with this capsule to compare the fuel failure threshold between the single rod and bundled rods systems.

Figure 5 shows the schematics of the visual observation capsule. The image of the test fuel is conducted by means of the periscope to the high speed camera located above the reactor core. The view field of the system is ~30 mm in diameter at the test rod surface. Fuel behaviour under RIA conditions was successfully visualized for the first time in the world with this capsule [2]. Samples of the obtained pictures are given in Fig. 6.

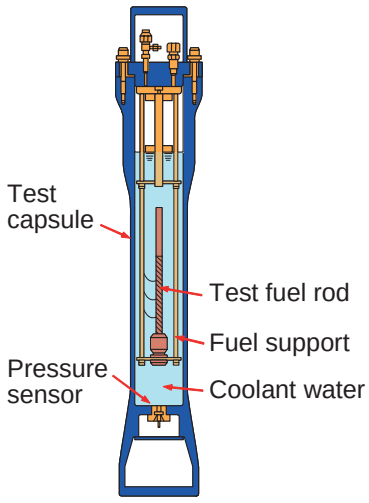

FIG. 4. Test capsule for fresh UO_2 fuel.

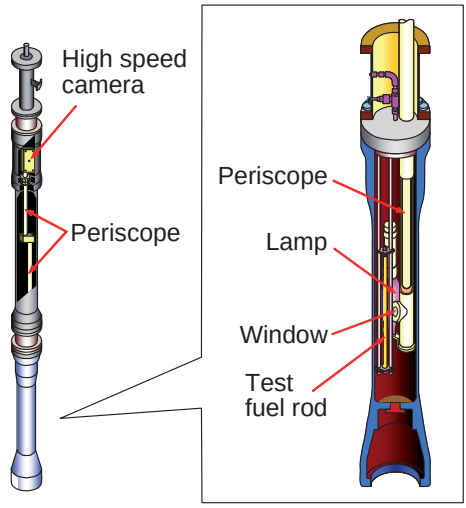

FIG. 5. Visual observation capsule.

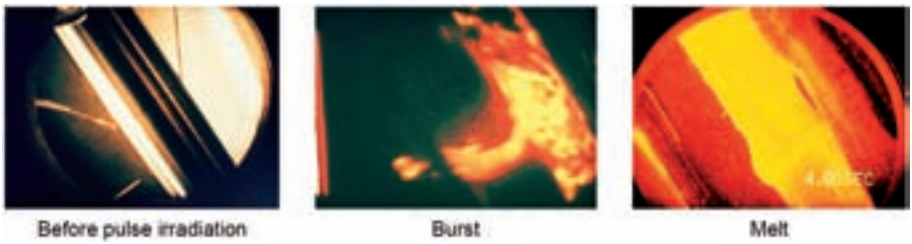

FIG. 6. Fuel rod appearance during pulse irradiation tests.

JAERI has also developed a capsule with a high temperature and high pressure water loop system, which can produce BWR/PWR coolant conditions [3]. Experiments with this capsule showed that fuel failure thresholds at high temperature are similar to that at room temperature. All the tasks with this capsule have been performed and the water loop system was disassembled.

2.2.2. Test capsules for other fresh fuels

Test capsules for fresh MOX, U silicide and Triga fuels have been individually designed, but the bases are common with that of the fresh UO_2 test capsules.

2.3. Test capsules for irradiated fuels

The test capsule for irradiated fuels has a double container structure to ensure pressure resistance and airtightness for radioactive materials. A schematic of the irradiated fuel test capsule is shown in Fig. 7. The test section is smaller than that of fresh fuel test capsules, but the same kind of equipment and sensors can be installed. Coolant conditions for this capsule are normally room temperature and atmospheric pressure. The coolant temperature can be raised up to ~90°C using an electric heater. This type of capsule is available for irradiated UO_2 and MOX fuels.

2.4. Test fuel rods

Most of the fresh UO_2 tests have been carried out with the 'NSRR standard test fuel rod', which is a short rod segment of a 14 × 14 PWR fuel assembly including highly enriched UO_2 pellets, up to 20%. The rod design is shown in Fig. 8. The rod internal gas is normally helium of 0.1 MPa at room temperature. The gas pressure and components can be changed according to

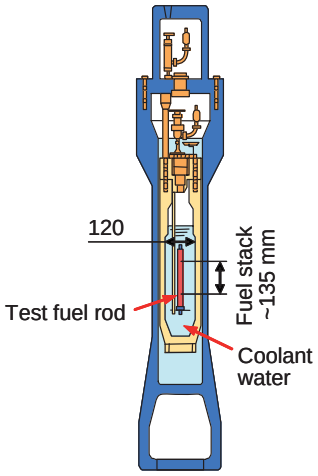

FIG. 7. Irradiated fuel test capsule.

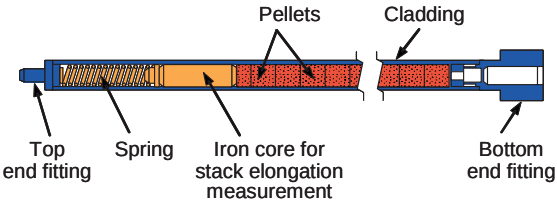

FIG. 8. NSR standard test fuel rod.

test objectives. The free space inside the rod can be filled with water as well, to simulate a waterlogged fuel rod [11]. Fresh fuel rods with other designs have also been tested.

Irradiated fuel rods for the NSRR tests are categorized into two types: NSRR standard test rods irradiated in the JMTR and shortened and refabricated segments of the fuel rods used in a commercial BWR/PWR or in the ATR. Basic designs of the rod end plugs and other parts are common to those for the NSRR standard test fuel rod.

2.5. Instrumentation

The following sensors can be installed at the test rods or test capsule:

— Cladding surface thermocouple;
— Coolant water thermocouple;
— Capsule pressure sensor;
— Rod internal pressure sensor;
— Cladding elongation sensor;
— Pellet stack elongation sensor;
— Water column velocimeter;
— Cladding surface strain gauge;
— Fuel centre thermocouple.

Regarding the cladding surface thermocouple, Pt and Pt/Rh wires are directly welded to the cladding surface. In the case where the rod has a thick oxide layer at the surface, the oxide is partially removed using a grinder prior to the welding. The water column velocimeter is to measure the water level movement, which gives information on mechanical energy generation at fuel failure. The velocimeter consists of two parts: a polyethylene float including a ring-shape magnet and a centre shaft including a coil, the winding direction of which changes every 6 or 3 mm. The velocimeter produces a sine wave signal for the float movement and the signal frequency is proportional to the velocity.

Signals from all sensors are pre-amplified and sent to the NSRR data acquisition system. Data had been recorded to a magnetic tape using a high speed analogue recorder until 1999. The present system has a real time AD converter, which digitizes signals with a 16 bit resolution at sampling frequencies up to 200 kHz.

3. RESULTS OF FUEL EXPERIMENTS

Results from the various fuel tests performed at NSRR are summarized in the following discussion.

3.1. Results of fresh fuel experiments

Pulse irradiation experiments with the NSRR standard test fuel rods have been performed to clarify fuel enthalpy thresholds for fuel state changes, including oxidation, failure and fragmentation. Figure 9 shows post-pulse rod

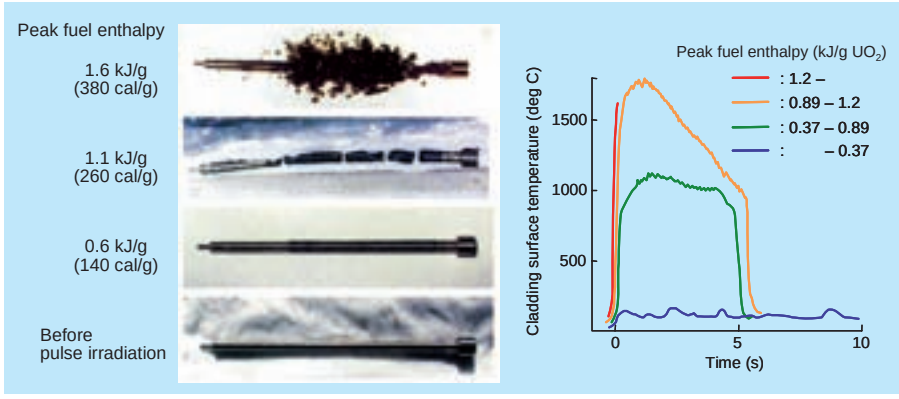

FIG. 9. Post-pulse appearances of fresh fuel rods.

appearances as a function of the peak fuel enthalpy. At a peak fuel enthalpy of 0.6 kJ/g, the cladding surface became dark due to oxidation and the occurrence of cladding deformation. At 1.1 kJ/g, the oxidized cladding was fractured by a thermal shock at quenching. Furthermore, at 1.6 kJ/g, the fuel melted and had thermal interaction with coolant water, resulting in fuel fragmentation and mechanical energy generation. Cladding surface temperature histories for several peak enthalpy conditions are also shown in Fig. 9, where the signal for the highest enthalpy is stopped at an early time because the thermocouple detached due to fuel melting.

The thresholds for fuel state changes were also obtained. Cladding oxidation and deformation start at 0.37 kJ/g (88 cal/g). Rod oxidation and deformation occur during film boiling and, thus, this threshold corresponds to that for departure from nucleate boiling (DNB) at the rod surface. The threshold for fuel failure is 0.89 kJ/g (212 cal/g), where fuel failure is defined as any kind of cladding penetration caused by cracking, burst and so on. The threshold for fuel melting is 1.2 kJ/g (285 cal/g), which is regarded as a threshold for mechanical energy generation in safety evaluation. These thresholds were obtained from single rod experiments, while bundled rods experiments showed that each threshold decreases by ~15% in a fuel assembly. Considering the bundle effect of -15% and data scatter of 0.04 kJ/g (10 cal/g), the NSC determined the safety evaluation criteria for fuel failure and mechanical energy generation as 0.71 kJ/g (170 cal/g) and 0.96 kJ/g (230 cal/g), respectively [1]. That is, fuel enthalpy of Japanese LWRs is not allowed to exceed 0.71 kJ/g in an abnormal transient and 0.96 kJ/g even in an accident. It should be noted again that the NSRR standard tests were performed with coolant water at room temperature and atmospheric pressure. These conditions are close to those at the BWR cold startup, which is the most conservative situation to test fuel integrity. The NSRR test results, therefore, have been accepted and reflected in safety evaluation guidelines in other countries.

One of the irradiation effects on fuel states is the rod internal pressure increase due to fission gas accumulation. This effect was simulated by prepressurization of the NSRR standard test rod [12]. Figure 10 shows the lower limit of the peak fuel enthalpy for fuel failure and the upper limit for non-failure, as functions of rod overpressure (internal to external pressure difference). The fuel failure threshold decreases with the pressure in the range above 0.6 MPa (6 kg/cm^2), because high rod internal pressure enhances cladding ballooning and a consequent burst. The appearance of the burst rod is shown in Fig. 11. Rod ballooning occurs after the cladding temperature rise at DNB. Therefore, the ballooning does not occur below the DNB threshold (0.37 kJ/g), even at high pressures. Taking into account the bundle effect and the data variance, the NSC determined the acceptable fuel design limit [1].

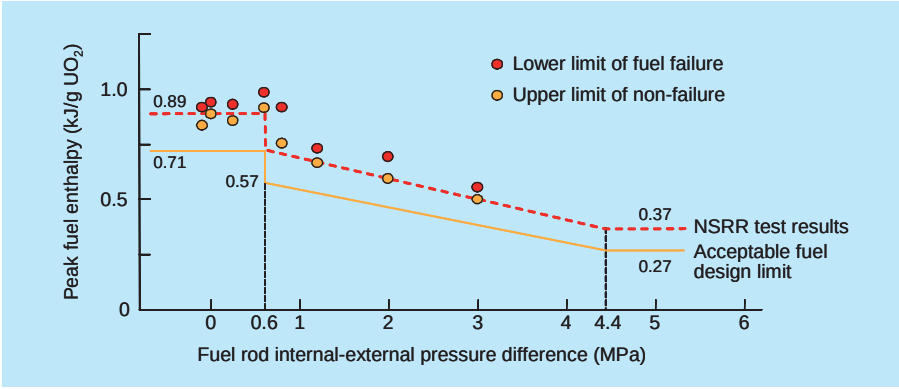

FIG. 10. Results of tests with prepressurized rods.

Fuel failure can cause mechanical energy generation in the pressure vessel, such as pressure pulses and water hammers. As described above, the fuel enthalpy of Japanese LWRs must be lower than 0.96 kJ/g (230 cal/g) even in an accident, to avoid mechanical energy generation due to thermal interaction between melted fuel and coolant water. A fuel rod, however, can burst at lower enthalpy if the rod has penetrating defects and is waterlogged. Figure 12 shows an appearance of the waterlogged fuel rod after the pulse irradiation. Rapid steam generation in the pellet/cladding gap raised the rod internal pressure so quickly that the cladding burst occurred when the cladding temperature was still low. This failure caused high pressure pulses and water hammer in the test capsule. A series of waterlogged fuel tests showed that the threshold enthalpy for the cold burst of a waterlogged rod is 0.38 kJ/g (90 cal/g), and provided data for thermal to mechanical energy conversion ratio during fuel failure [11]. The failure criterion of waterlogged rods was determined to be 0.27 kJ/g (65 cal/g) by the NSC in the same manner as for the other criteria [1]. In addition, Japanese safety evaluation guidelines prescribe that 1% of the total rods in a reactor must be assumed to be

FIG. 11. Appearance of a burst rod.

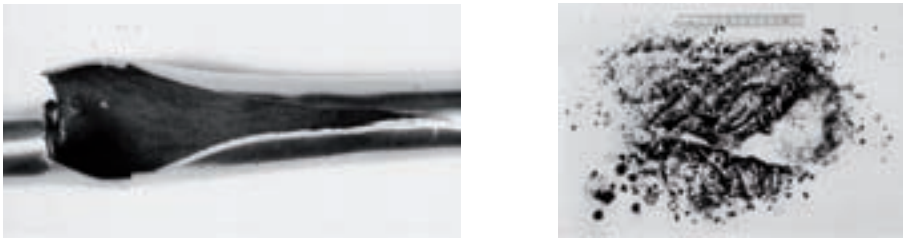

FIG. 12. Post-pulse appearance of a waterlogged fuel rod.

waterlogged. Using this information and these assumptions, the total number of burst rods and mechanical energy generation are evaluated in the safety evaluation process for Japanese LWRs.

3.2. Results of irradiated fuel experiments

Long irradiation causes changes in the fuel, including pellet swelling due to fission product accumulation, cladding creep-down under coolant pressure, degradation of cladding mechanical integrity because of surface oxidation and hydrogen pick-up, etc. It has been anticipated that these changes could lower the fuel failure limit in terms of fuel enthalpy. NSRR experiments with irradiated fuel rods were started in 1989 to answer these questions and to investigate post-failure events. More than 70 experiments have been carried out so far.

The NSRR experiments have indicated that high burnup fuel rods could fail at low fuel enthalpies in a different failure mode from those of fresh fuel rods [8, 9]. The dominant failure mode of fresh fuel rods was melting or brittle fracture at high temperature, while high burnup fuel rods failed due to pellet cladding mechanical interaction (PCMI) at low temperature. Figure 13 shows the appearance of a PCMI failed PWR rod. A large axial crack was generated and some portion of fuel was dispersed into the coolant water. As indicated by the cladding cross-section in Fig. 14, the cladding inner region shows ductile fracture, though the outer region suggests brittle fracture due to hydride

FIG. 13. PCMI failure of an irradiated PWR fuel rod.

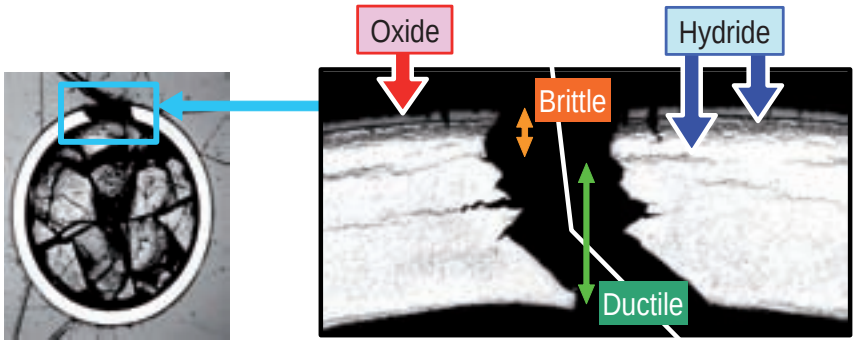

FIG. 14. Cross-section of a PCMI failed PWR rod.

precipitation. Transient measurements indicated that the cladding surface temperature was below 100°C at the failure. The experimental results are summarized in a failure map together with data from SPERT/PBF (United States of America) [13] and CABRI (France) [14] tests in Fig. 15. The horizontal axis is burnup of the test rods and the vertical axis is fuel enthalpy increase, which is defined as 'peak fuel enthalpy' — 'initial fuel enthalpy' for non-failed rods and as 'fuel enthalpy at failure' — 'initial fuel enthalpy' for failed rods. The marks for fuel failure show that the failure limit decreases with fuel burnup. The NSC established the PCMI failure criterion as a step function of fuel burnup, as drawn in Fig. 15 [1].

PCMI failures of high burnup rods occur at low enthalpy. Thus, the fuel temperature does not exceed the fuel melting point at failure and a thermal interaction between the melted fuel and the coolant water is impossible. The NSRR experiments, however, showed that high burnup fuel failure could cause mechanical energy generation due to thermal interaction between solid fuel fragments and coolant water. In high burnup fuels, fission products (FPs) are accumulated at fuel grain boundaries. Thermal expansion of the accumulated FPs at the pulse irradiation separates the grains as illustrated in Fig. 16, which can be a driving force for the fuel fragmentation at the PCMI failure. The fuel temperature is below the melting point at failure, but the total surface area of the fuel is quite large after the fragmentation. Therefore, a contact of the fuel fragments with coolant water can lead to energetic steam generation, as illustrated in Fig. 17 [15]. The generated mechanical energy is measured in the NSRR experiments with a capsule pressure sensor and a water column velocimeter.

As described above, high burnup fuel tests have shown the hydride assisted PCMI failures at low fuel enthalpies. However, some cladding might

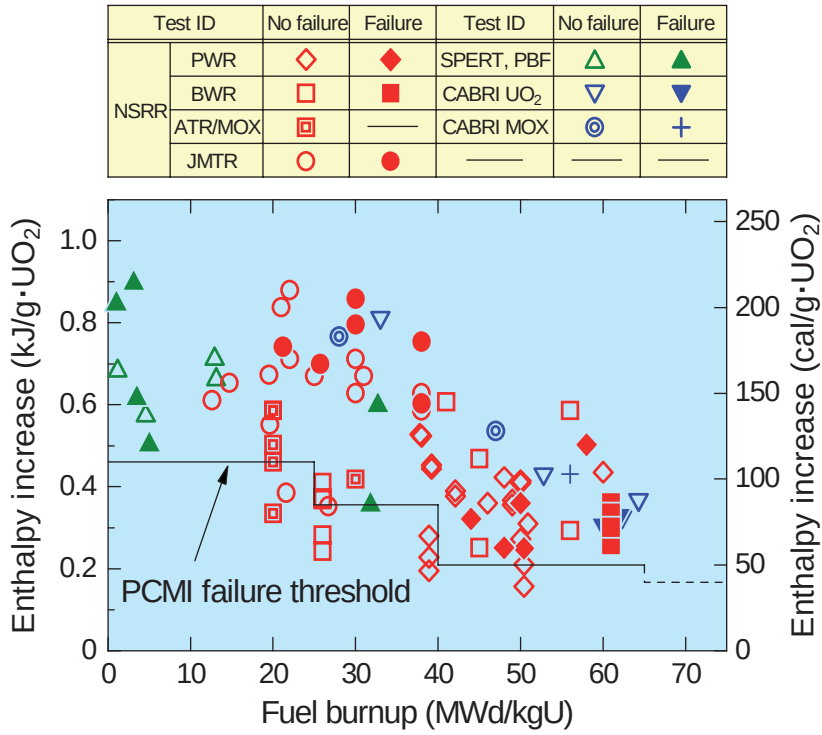

FIG. 15. PCMI failure map obtained from NSRR, SPERT/PBF and CABRI tests.

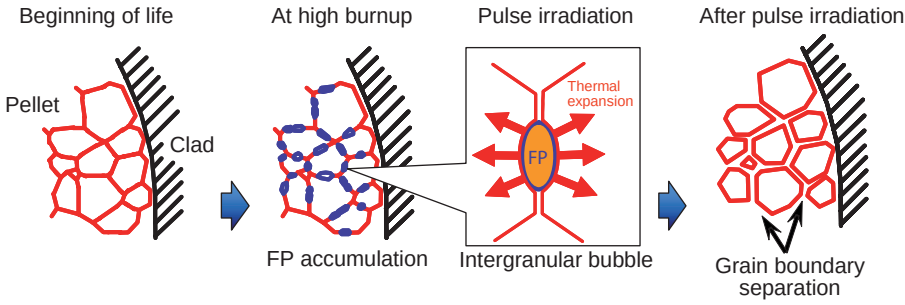

FIG. 16. Grain boundary separation at pulse irradiation.

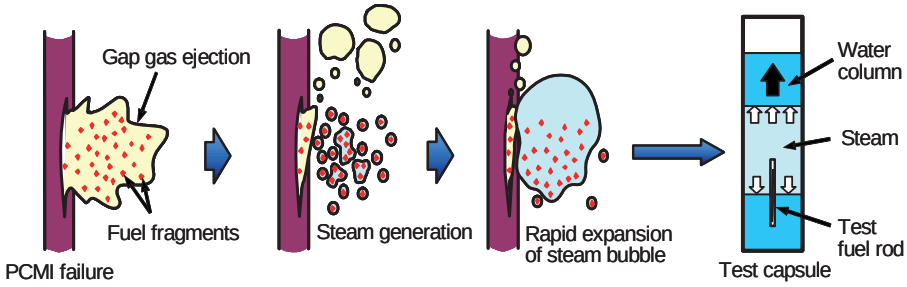

FIG. 17. Thermal interaction between fuel fragments and coolant water at PCMI failure.

survive if the tests had been performed at high temperatures, because the hydrogen solubility limit in the cladding increases with temperature. To clarify the high temperature effect on the PCMI failure limit, a new capsule is being developed for irradiated fuel experiments. The schematic of the high temperature/high pressure test capsule is given in Fig. 18. This capsule is equipped with an electric heater to produce BWR operating temperatures and

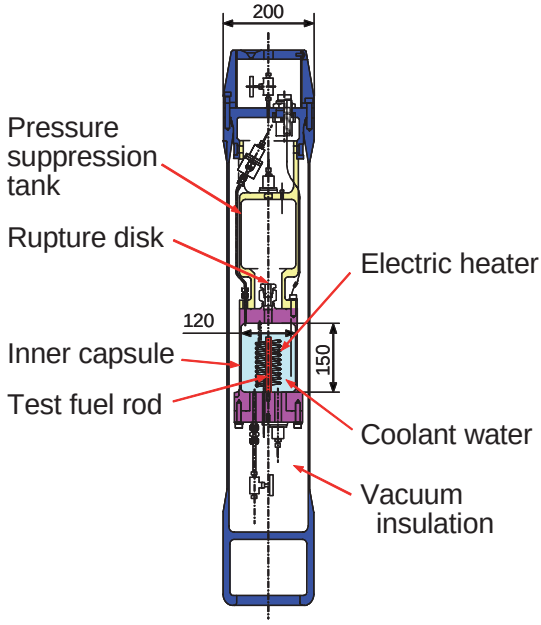

FIG. 18. High temperature/high pressure test capsule.

pressure. The main objective is to test the temperature effect above ~200°C. Therefore, high pressure is not the primary interest and the PWR operation conditions are not targeted.

4. CONCLUSIONS

JAERI has utilized the NSRR to perform accident simulating experiments of light water reactor fuels since 1975. Over 1200 experiments have successfully been carried out with fresh fuel rods to clarify fuel behaviour under RIA conditions and to obtain fuel failure limit. The results have been reflected in safety evaluation guidelines for the RIA in some other countries as well as in Japan.

The NSRR facilities were modified to test high burnup fuels, and more than 70 experiments have been performed with irradiated fuel rods since 1989. These experiments have provided necessary information for the safety assurance of high burnup LWR fuels. Further safety research using the NSRR is expected to support fuel burnup extension and MOX fuel introduction. In order to meet the requirements, the capability of the NSRR facilities is being extended.

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8.4. THE PHÉBUS REACTOR AND ASSOCIATED EXPERIMENTAL CAPABILITIES

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1. INTRODUCTION

The PHÉBUS FP facility located at Cadarache, France, offers a rather unique opportunity to check computer codes used to assess the safety of light (pressurized) water reactor (LWR) plants and the efficiency of accident management procedures. The facility can accommodate a series of integral in-pile experiments to evaluate whether all important phenomena governing both core degradation and subsequent fission products released to the containment are correctly understood and properly modelled. The experiments utilize actual irradiated core material from nuclear power plants (NPPs) at a global scale of 1/5000 with respect to a 900 MWe NPP. The knowledge of the amount and of the nature of radioactive products present at any time in the atmosphere of the reactor containment is particularly important in order to evaluate the potential source term which could be released to the environment in case of a leakage of the NPP containment.

The range of phenomena investigated encompasses the entire sequence starting with the early phase of cladding oxidation, hydrogen production and material interactions, and ending with the late phase consisting of core material melt propagation and corium pool formation. In parallel, the whole chain of fission product behaviour is investigated: the release of fission products and structure materials from the fuel as degradation proceeds, the transport and deposition in the circuits of the plant as vapours or aerosols depending on the volatility under the various physical–chemical conditions simulated, and the deposition at the different surfaces existing in the containment, including possible iodine revolatilization as a consequence of long term radiochemical effects.

The two core melting experiments at PHÉBUS, FPT-0 and FPT-1, performed in December 1993 and July 1996, respectively, have reached very advanced states of degradation, comparable to what was observed at the partial core melting at the US-NPP TMI-2, and generated a wealth of results on core degradation and fission product behaviour.

The resulting database is used to develop and to validate the computer codes used to assess the safety of currently operating plants, to check the

efficiency of accident management procedures and also support the design of future plants.

2. PHÉBUS REACTOR

The PHÉBUS reactor is a 40 MW research and test reactor designed and built by the Commissariat à l'énergie atomique (CEA) of France in the early 1970s, for LOCA programme experiments. It comprises a low enriched uranium dioxide core (1816 fuel rods of 0.8 m length in 36 fuel assemblies, 2.78% enriched, with Zircaloy cladding) and closed loop light water cooling (maximum cooling capability 20 MW), all embedded in a water pool. The core is controlled by 162 Hf control rods and a lateral neutron shield. The irradiation position at the centre of the core contains a double walled cell which is inserted in an in-pile test section. An independent high pressure water loop cools the outside of the in-pile section. Auxiliary pressurized systems are available for the injection of fluids into the in-pile section.

The existing reactor building (Figs 1 and 2) underwent considerable modifications for seismic protection and general safety backfitting prior to FPT-0. Moreover, a new building was constructed adjacent to the existing plant, containing a part of the circuits potentially charged with fission products, (remote) handling devices and two small hot cells. A 350 m^3 steel confinement encloses all circuits and a 10 m^3 vessel simulating a reactor containment.

FIG. 1. PHÉBUS plant.

FIG. 2. The PHÉBUS reactor.

The Institut de radioprotection et sûreté nucléaire (IRSN) runs the PHÉBUS FP programme, which includes in situ pre-irradiation (9 d) of the test fuel prior to the high temperature transient, which required two major modifications of the PHÉBUS reactor:

— Construction of closed primary/secondary core cooling loops with pumps, primary water delay tank, heat exchanger and cooling tower;
— Increase of the reactor core reactivity margin by supplementary modular graphite reflector elements.

The PHÉBUS FP programme is co-financed by the European Commission, the United States Nuclear Regulatory Commission, the CANDU Owners Group (COG) from Canada, Japan's Nuclear Power Engineering Corporation (NUPEC) and the Japan Atomic Energy Research Institute (JAERI), the Korea Atomic Energy Research Institute (KAERI) of the Republic of Korea and, from Switzerland, the Paul Scherrer Institut (PSI) and the Swiss Federal Nuclear Safety Inspectorate (HSK).

3. IN-PILE SECTION

The term 'in-pile section' refers to the cylindrical device (6 m long, 121–250 mm outer diameter, 250 kg total mass) loaded into the safety tube. Its lower part contains the test fuel and the instrumentation, the upper part includes service lines, instrument feed-through and shielding. Vertical and

horizontal sections through the test fuel area are shown in Figs 3 and 4. The design of the in-pile section, which is specific for each PHÉBUS FP test includes, except for FPT-4, a remotely operated foot valve connecting the fuel bundle to the high pressure water loop during the in situ pre-irradiation phase. Together with the Zircaloy shroud and the high temperature instrumentation (thermocouples and ultrasonic thermometers), the foot valve was among the items demanding major development.

The test bundle, including two Zircaloy spacers (as for PWR) and four Zircaloy stiffeners, contains 20 fuel rods in a 12.6 mm pitch matrix. They used 4.5% enriched UO_2 (≈10.5 kg) in Zircaloy-4 cladding.

The geometry of the fuel rods (9.5 mm outer diameter, 1 m active length) as used in FPT-1 corresponds to the fuel of the reactor BR3 at Mol, Belgium. The fuel rod internal pressure is 2.8 MPa as in the BR3 rods (the internal fuel rod pressure is lower than at a PWR, but the system pressure in PHÉBUS is lower compared with the small break loss of coolant accident (SBLOCA) with the consequence that the pressure difference is equivalent). The central control rod (pressurized at 0.1 MPa as in a PWR core) contains 80 wt% silver, 15 wt% indium and 5 wt% cadmium as absorber material, enclosed in a 304 L stainless steel cladding. That rod is guided in a Zircaloy tube.

A thermal shield of two zirconia (ceramic of zirconium oxide) sleeves surrounded the fuel bundle for FPT-0. The inner zirconia ring was replaced by a thorium one (thorium oxide) for the FPT-1 test (see Fig. 3) and a pressure tube of Inconel (6 mm wall thickness), the latter coated on its internal surface by zirconia (1 mm layer).

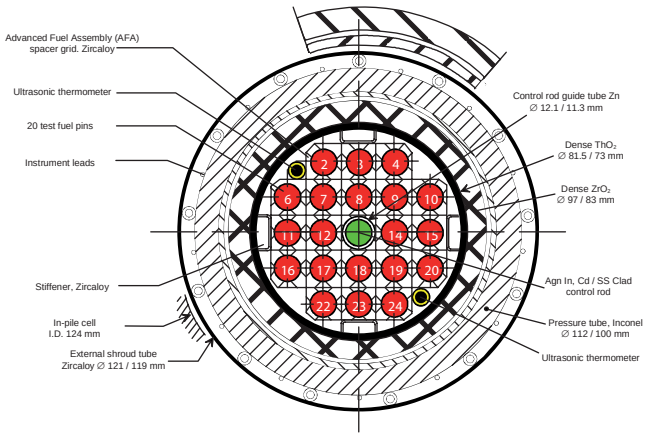

FIG. 3. FPT-0 in-pile device, horizontal section.

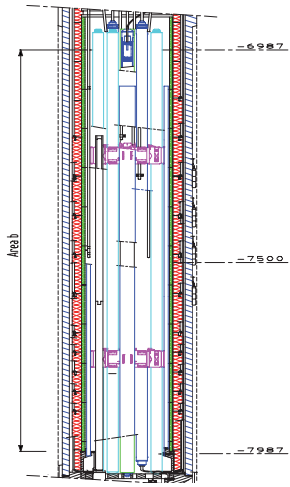

FIG. 4. PHÉBUS FP in-pile test device, vertical section.

The test fuel was heated by nuclear fission driven by the neutrons emitted from the core of the PHÉBUS reactor.

4. EXPERIMENTAL CIRCUIT

The description of the experimental circuit is shown in Fig. 5.

The pipe work between the upper end of the fuel bundle and the entrance into the confinement consists of the following components:

— Upper part (vertically, 3422 mm), along which the gas temperature (steam and fission product mixture) drops from over 2270 K to 970 K;
— Isothermal (973 K) pipe from Ni coated Inconel 600 (horizontally, 5130 mm long; 30 mm internal diameter);
— Main fission product pipe from Inconel 600 (see point C in Fig. 5) with sampling devices in a 973 K furnace (3797 mm long, 30 mm internal diameter);
— Vertical steam generator tube from Inconel 600, non-condensing under PHÉBUS's thermal hydraulic conditions (9663 mm long inverted U-tube, 20 mm internal diameter) with tube walls maintained at 423 K;
— Another isothermal (423 K) pipe from stainless steel (AISI 304 L) (see point G in Fig. 5) with sampling devices inside a 423 K furnace (horizontally, 3824 mm long; 30 mm internal diameter), comprising the connection

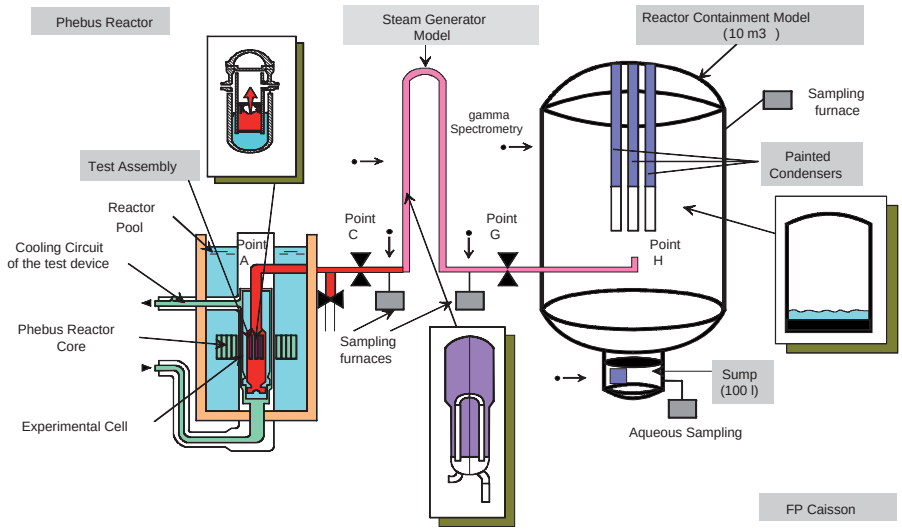

FIG. 5. Components of the experimental circuit.

to the containment vessel (surfaces of the pipes are not degreased or autoclaved prior to being mounted as is at PWR pipes).

The geometry of the test train and experimental circuit was designed to get representative concentrations of fission products in the main fission product pipe.

5. CONTAINMENT VESSEL

The volumetric scaling factor of the 10 m³ containment vessel (5 m high, 1.8 m inner diameter) in terms of a 900 MW PWR containment corresponds approximately to the core mass ratio between the PHÉBUS FP test fuel bundle and the core of that PWR, i.e. 1:5000. More precisely, the core mass ratio between a 900 MW PWR core and the PHÉBUS FP test fuel bundle is 8000:1. In this way, the fission product concentrations are representative of real reactor conditions, both in the circuits as well as in the containment vessel.

Heat transfer and steam condensation on the different surfaces constituting the reactor containment are simulated by three vertical condensers (2500 mm long) suspended at the top end of the vessel. Each of these condensers

comprises a condensing and a non-condensing part, respectively (1718 mm and 782 mm long).

The temperature of the wet part (condensing the steam injected in the containment) is controlled by a liquid organic loop and its temperature can decrease down to 60°C. The temperature of the non-condensing (or dry) part is controlled by electrical heaters and can be increased up to 150°C.

The cooled surfaces of the condensers (about 0.81 m^2 per condenser) are covered with epoxy paint as a source for organic iodine formation and the area scaling ratio (referring to the condensing reactor containment walls) is about 3000:1. The painted condensing surface to volume ratio of the containment model was then also comparable to an actual 900 MW PWR containment.

The non-condensing structures attached to the three condensers were also painted and accounted for about 0.37 m^2 per condenser (at PHÉBUS the ratio of non-condensing surfaces versus containment volume is about three times higher compared with an actual French 900 MW PWR). The outer walls of the containment vessel are slightly superheated in order to prevent any steam condensation and to minimize aerosol deposition.

The lower part of the containment vessel is closed by a curved bottom structure including a 0.1 m^3 sump. The sump has a diameter of only 0.584 m in order to produce a correctly scaled atmosphere–water exchange surface. It contains a painted structure simulating immersed painted surfaces of a reactor sump. The sump water is recirculated by an auxiliary loop. Another circuit injects water onto the vessel bottom structure, washing aerosols settled there into the sump.

6. AUXILIARY EQUIPMENT

Handling and examination of a large number of radioactive items required the design, construction and testing of many new devices, for example:

— Examination and control station (PEC) in the reactor building applied for pre- and post-irradiation radiography, gamma scanning, gamma transmission and emission tomographs. Gamma transmission and emission tomography techniques were developed within the PHÉBUS project in order to give qualitative and quantitative results about the fission product distribution in the fuel test section after the test. Gamma emission tomography was not performed for the FPT-0 but for the FPT-1 test;
— Test train handling and transport flask (12 t);
— Polar crane and power manipulator in the fission product confinement;

— Two fully equipped hot cells for fission product specimen measurement, handling and storage, and for transfers by medium and small size shielded transport containers.

A low pressure, unheated 100 m^3 tank occupies a separate concrete cubicle inside the PHÉBUS building. That tank serves as a gaseous effluent storage tank. The off-gas line between the containment and atmospheric vessels is fitted with a condenser, aerosol filters and active charcoal traps.

7. INSTRUMENTATION

This supplementary Part 7 covers the so-called scientific measurements only, as compiled in the FPT-0 Instrumentation Plan.

Process and safety instrumentation and control of the reactor, the high pressure water loop, and ventilation systems are only mentioned if required for an understanding of experimental sequences. The measurements taken fall into two major categories, as follows:

— Measuring temperature, pressure, flow rate, hydrogen concentration and humidity in the different circuit areas (about 500 transducers, sensors and detectors were recorded);
— Fission products (FP) measurements (here FP refers to fission and activation products, including measurements of structural, fuel and control rod materials) by two types of on-line instruments, i.e. gamma spectrometers (more than 30 000 gamma spectra were recorded) and optical aerosol monitors.

In order to widen and vary the time dependent measurements of FP as received by the on-line instrumentation described, sampling and off-line post-test analyses (PTA) were introduced and provided valuable time interval information. Sampling devices, such as grab samplers (capsules), filters, aerosol impactors (for the aerosol size determination) and iodine absorbers were designed and tested to PHÉBUS FP requirements. Special attention was focused on the instrumentation devoted to iodine measurements. Five selective iodine filters (‘Maypacks’) were mounted, each of them comprising three to four different and successive filtering media: a quartz filter for trapping the aerosol particles, a silver knit mesh filter for trapping gaseous molecular iodine and a silver zeolite filter for trapping all the gaseous iodine species not trapped in the upstream silver knit mesh filter and, in particular, for organic iodide. An

additional charcoal filter (potassium iodide doped) was included in some of the Maypacks.

In total, 46 samples were triggered during the overall FPT-0 test. All these samples were analysed by accurate gamma spectrometry measurements performed in a specific hot cell (called CECILE).

Difficulties were encountered in the application of the instrumentation during the tests because some instruments operated outside their proven range when exposed to the test environment comprising hot steam with variable amounts of hydrogen. As an example, the selective iodine filters (Maypacks) experienced some problems (aerosol pollution of the silver knit mesh filters and lower silver knit mesh efficiency than measured during the qualification tests performed in the laboratories), which made the iodine composition measurements difficult. Nevertheless, those measurements were still valuable and served for the quantification of the total gaseous iodine in the containment.

The necessary post-test analysis (PTA) of the sampling instruments was a largely unknown area. Therefore, the development of a working plan for FPT-0 started with a literature survey. These PTA, as performed in a number of European laboratories (Saclay, France; ITU Karlsruhe, Germany; PSI Villigen, Switzerland; AEA-T Dorchester, United Kingdom), provided useful and qualified data concerning the overall fission product mass balance, the aerosol composition and the morphology.

8. CONCLUSION

In the test matrix of Table 1, the entire PHÉBUS FP programme is described. Four tests of the matrix have presently been performed. FPT3 is in its final preparation stage, the pre-irradiation and test phase is currently planned.

An additional test (STLOC1) is concerned with the study of an air ingress phase during the fuel degradation period. The result of this additional phase should be a significantly changed degradation scenario and release mechanisms, but also circuit phenomena and containment chemistry could both be notably influenced. Once all financing partners have settled around this outstanding scientific objective, this test could reasonably be performed in 2008. The results of each test need three to four years to be released as a final data report, while the quantitative interpretation using computer codes and models could take another five years before completion and comparison with other tests of the matrix.

TABLE 1. PHÉBUS FP PROGRAMME TEST MATRIX

Test no.	Test fuel bundle	Research objectives of fuel bundle degradation and FP release	Primary circuit objectives	Research in the containment vessel	Test date
FPT-0	20 fresh fuel rods, 1 Ag-In-Cd control rod, 9 d in situ re-irradiation	Degradation up to fuel melting, FP release in low pressure steam rich conditions	FP retention in pipes, bends and a non-condensing steam generator, deposit analysis for granulome-try composition and chemical species	Aerosol behaviour under moderate relative humidity, iodine radiochemistry, sump water buffered at pH = 5	1993-12-02
FPT-1	BR 3 fuel at ~23GWd/t, 1 Ag-In-Cd control rod, 6 d in situ re-irradiation	As for FPT-0	As for FPT-0	As for FPT-0	1996-07-26
FPT-2	BR 3 fuel at ~32GWd/t, 1 Ag-In-Cd control rod, 9 d in situ re-irradiation	As for FPT-0, but in a steam poor atmosphere	As for FPT-0, with effect of boric acid on aerosols	As for FPT-0, but with evaporating sump at pH = 9	2000-10-12
FPT-3	As for FPT-2, but with one B_4C control rod, BR 3 fuel at ~24 GWd/t	As for FPT-2	As for FPT-2	As for FPT-2, H_2 recombiner coupons	Planned for 2004-10
FPT-4	Rubble bed of UO_2-ZrO_2 fragments, EDF fuel at ~38 GWd/t, no control rod, no re-irradiation	Melt progression in the debris bed, release of low volatile FP and of actinides	Aerosol retention by filters mounted above the in-pile section, post-test studies on samples		1999-07-22
STLOC1	As for FPT-1	Fuel degradation and FP release under air ingress	Deposition and chemistry of FP under air ingress conditions	As for FPT-1 or for FPT-2 (to be decided)	Planned for 2008

8.5. MYRRHA — A MULTIPURPOSE ACCELERATOR DRIVEN SYSTEM FOR R&D
State of the project at the end of 2003

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1. INTRODUCTION

The European Technical Working Group (ETWG) on Accelerator Driven Systems (ADSs) concluded in April 2001 in its report "A European Roadmap for Developing Accelerator Driven Systems (ADS) for Waste Incineration" [1] that partitioning and transmutation in association with ADSs and in combination with geological disposal can lead to an acceptable solution, from the point of view of public acceptance, for nuclear waste management problems. It also concluded, therefore, that strong support in this field from the European Community and the national programmes is needed to develop and build an experimental ADS demo facility in Europe.

It is obvious that the burden of nuclear waste is mainly due to transuranic elements and their associated long life and high toxicity. Therefore, when talking about waste transmutation, one is mainly concerned with the treatment of Pu and the minor actinides (Np, Am and, to a certain extent, Cm). If Pu can be transmuted efficiently in fast critical reactors and even in thermal reactors with a limitation on the number of recycling (degradation of the isotopic vector) in the form of MOX fuel, the situation becomes more demanding when it comes to the treatment of the minor actinides (MA). Indeed, the MAs have a very small value (0.03–0.1%) of delayed neutrons compared with U or Pu (0.4–1.4%, depending on the considered isotope). With such a reduced fraction of delayed neutron, the reactivity control in a critical reactor is becoming very challenging if one is willing to load a large fraction of the reactor core (>60%) with MA fuels and even more when targeting the use of U free or Th free MA fuels [1]. For all these reasons, the choice of subcritical systems for waste burning and, in particular, of MA in dedicated machines is recommended.

SCK•CEN, in partnership with IBA S.A. and many European research laboratories, has been working since 1998 to design a multipurpose ADS for R&D applications — MYRRHA [2] — and is conducting an associated R&D

support programme. MYRRHA is an ADS under development aiming to serve as a basis for the European experimental ADS to provide protons and neutrons for various R&D applications. It consists of a proton accelerator delivering a 350 MeV × 5 mA proton beam to a liquid Pb–Bi spallation target that in turn couples to a Pb–Bi cooled, subcritical fast core.

One of the core competencies of SCK•CEN is and has at all times been the conception, design and realization of large nuclear research facilities. One of the main SCK•CEN research facilities, BR2 (a 100 MW material testing reactor), has been in service for nearly 40 years, similar to the other major MTRs in the world. The MYRRHA facility has been conceived as potentially replacing BR2 and to be a fast spectrum facility complementary within European demands to the thermal spectrum Réacteur Jules Horowitz (RJH) facility, in planning in France. This situation would give Europe a full research capability in terms of irradiation capabilities for nuclear R&D.

Furthermore, the disposal of radioactive wastes still requires a fully satisfactory solution, especially in terms of environmental and public acceptability. Scientists are looking for ways to drastically reduce the radiotoxicity of high level waste (HLW) to be stored in a deep geological repository so as to reduce the time needed to reach the radiotoxicity level of the fuel ore originally used to produce energy. This can be achieved through the development of partitioning and transmutation (P&T) and burning MAs and, to a lesser extent, long lived fission products (LLFP) in dedicated waste burners, such as ADSs or in critical reactors in combination with energy production. The MYRRHA project contribution will be to demonstrate the ADS concept at a reasonable power level, to demonstrate the technological feasibility of MA and LLFP transmutation under realistic conditions, and the economical assessment of this technology as a credible waste management option.

2. PRINCIPAL FEATURES OF THE DESIGN OF THE MYRRHA FACILITY

The MYRRHA project is based on the coupling of a proton accelerator with a liquid Pb–Bi windowless spallation target, surrounded by a Pb–Bi cooled subcritical neutron multiplying medium in a pool type configuration with a standing vessel (Fig. 1) [2, 3]. The spallation target circuit is fully immersed in the reactor pool and interlinked with the core, but its liquid metal content is separated from the core coolant. This is a consequence of the windowless design presently favoured in order to use low energy protons on a very compact target at high beam power density in order to avoid sacrificing core performance.

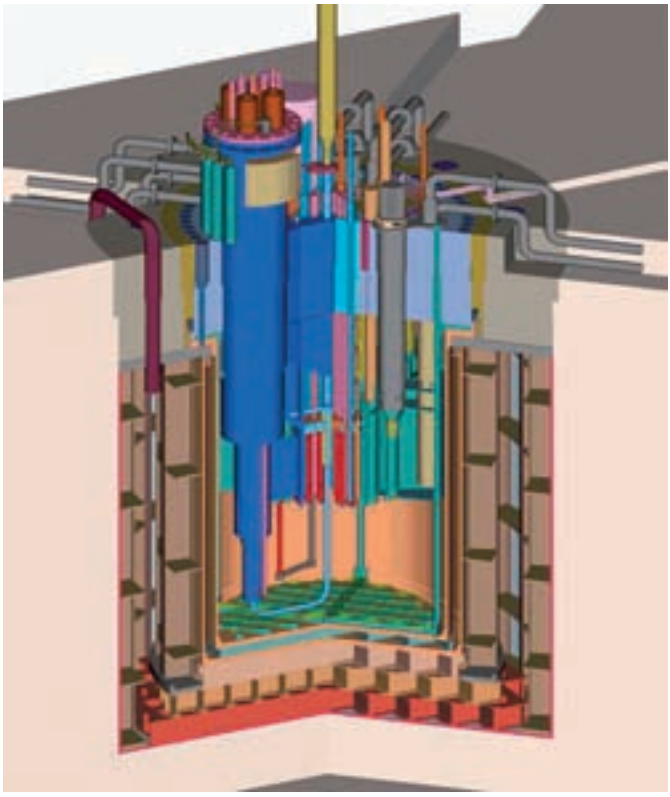

FIG. 1. MYRRHA three-dimensional vertical view.

The core pool contains a fast spectrum subcritical core cooled with Pb–Bi eutectic (LBE) and several islands housing thermal spectrum regions located in in-pile sections (IPS) in the fast core. The core is fuelled with typical fast reactor fuel pins with an active length of 600 mm arranged in hexagonal assemblies. The three central hexagons are left free for housing the spallation module. The core is made of hexagonal fuel assemblies of 85 mm flat-to-flat, composed of MOX typical fast reactor fuel ('Superphénix'-like fuel rods) with total Pu contents of 30% and 20%.

The core structure will be mounted on a central support column coming from the lid and being stabilized by the diaphragm, the separating septum between the cold and hot LBE coolant, which is fixed ultimately to the rim of the double wall vessel. Since access from the top is very restricted and components introduced into the pool will be buoyant due to the high density of the LBE, the loading and unloading of fuel assemblies are foreseen to be

carried out by force feedback controlled robots in remote handling from underneath. The pool will also contain the liquid metal primary pumps, the heat exchangers — currently considered to be using water as a secondary fluid — and the two fuel handling robots based on the well known rotating plug of fast reactors.

The spallation circuit connects directly to the beam line and ultimately to the accelerator vacuum. It contains a mechanical impeller pump and an LM/LM heat exchanger to the pool coolant (cold end). For regulation of the position of the free surface on which the proton beam impinges (whereby this defines the vacuum boundary of the spallation target), it comprises an auxiliary MHD pump. Further on, it contains services for the establishment of proper vacuum and corrosion limiting conditions.

The device is shown in Fig. 1 with the double wall pool containment vessel (inner diameter of about 4 m; height close to 7 m), is surrounded by a biological shield to limit the activation of the surrounding soil as the MYRRHA subcritical reactor will be installed in an underground pit. This shield will be closed above the vessel lid by forming an alpha compatible hot cell and handling area for all services to the machine.

3. TASK PROFILE

Along the above design features, the MYRRHA project team is developing the MYRRHA project as a multipurpose irradiation facility for R&D applications on the basis of an ADS. The project is intended to fit into the European strategy towards an ADS demo facility for nuclear waste transmutation, as described in the Preliminary Design Studies of eXperimental Driven Systems (PDS-XADS) FP5 Project [4]. As such, it should serve the following task catalogue:

- ADS concept demonstration: to test the coupling of the three components at a rather reasonable power level (a few tens of MW_{th}) to allow operation with representative thermal feedback and reactivity effects mitigation;
- Safety studies for ADS: to allow beam trip mitigation, subcriticality monitoring and control, optimization of restart procedures after short or long stops, feedback to reactivity injection;
- MA transmutation studies: to make use of high fast flux level ($\Phi_{>0.75MeV} = 10^{15}$ n·cm^{-2}·s^{-1});
- LLFP transmutation studies: to make use of high thermal flux level ($\Phi_{th} = 1$ to 2×10^{15} n·cm^{-2}·s^{-1});

— Medical radioisotopes: to make use of high thermal flux level ($\Phi_{th} = \sim 2 \times 10^{15}$ $n \cdot cm^{-2} \cdot s^{-1}$) as well;
— Material research: to make use of large irradiation volumes with high constant fast flux level ($\Phi_{>1\ MeV} = 1 \sim 5 \times 10^{14}$ $n \cdot cm^{-2} \cdot s^{-1}$);
— Fuel research: to make use of irradiation rigs with adaptable flux spectrum and level ($\Phi_{tot} = 10^{14}$ to 10^{15} $n \cdot cm^{-2} \cdot s^{-1}$);
— Initiation of medical and new technological applications: to test and study applications, such as proton therapy and proton material irradiation.

The present MYRRHA concept is driven by the flexibility and the versatility needed to serve the above applications. Some choices are also conditioned by objectives to make MYRRHA as demonstrative as possible for an industrial ADS prototype. The MYRRHA project team has favoured, as much as possible, mature or less demanding technologies in terms of R&D. Nevertheless, not all the components of MYRRHA exist. Therefore, a thorough five year R&D support programme has been established, to help manage related project risk areas and is subjected to annual revision according to the design option and the R&D progress.

4. DESIGN FEATURES AND PARAMETERS AND THEIR JUSTIFICATION

4.1. MYRRHA: Critical reactor versus ADS

Regarding the applications listed above, one could ask why not to go for a critical reactor? Indeed, nowadays material and fuel research is conducted in critical MTRs, radioisotopes are produced in these machines, transmutation studies could be conducted in critical reactors, but choosing the ADS route will trigger the possibility of demonstrating the ADS concept and will make available higher flux levels (thermal and fast), as these are driven by the spallation source. The R&D of an innovative ADS project will be an asset for attracting the new generation of scientists and engineers towards the nuclear sector. For all these reasons and particularly complementarily to a future European thermal spectrum MTR, SCK•CEN considers the ADS orientation as the most relevant option for a new fast spectrum R&D facility.

4.2. Main design parameters of MYRRHA and the MYRRHA spallation target

The performances of ADS in terms of flux and power levels is dictated by the spallation source strength, which is proportional to the proton beam flux at a particular energy and the subcriticality level of the core. The subcriticality level of 0.95 has been considered as an appropriate level for a first of its kind medium scale ADS. Indeed, this is the criticality level accepted by the safety authorities for fuel storage. Besides this aspect, various incidental situations that can lead to reactivity variation were considered. The majority of those effects would result in a negative reactivity injection or a limited positive reactivity injection not leading to criticality when starting at a k_{eff} of 0.95.

Fixing the subcriticality level — determining the nuclear gain — and the desired neutron flux in the position of the irradiation location for MA transmutation determines the required strength of the neutron spallation source. In order to achieve the performance mentioned at the modest total power level anticipated, the central hole diameter must be limited to a maximum diameter of 120 mm. As a consequence of this constraint and, in addition, needing a minimum lateral Pb–Bi target volume for allowing an effective spallation process, the proton beam external diameter is limited to 72 mm; whereby time averaging of a scanned pencil beam will shape the beam profile. The required spallation source intensity to produce the desired neutron flux at this location is close to 2×10^{17} n/s. At the chosen proton energy of 350 MeV, this requires 5 mA of proton beam intensity, and this in turn would lead to a proton current density on an eventual beam window of the order of 150 μA/cm^2. This exceeds the current density of other attempted window design for spallation sources, which have high uncertainties with regard to material properties suffering from swelling and radiation embrittlement by a factor of 3. As a result, the windowless spallation target design in MYRRHA was favoured [5–8].

4.3. Required accelerator

The proton beam characteristics of 350 MeV × 5 mA enable a fast neutron flux of 1×10^{15} n·cm^{-2}·s^{-1} ($E_{>0.75\ MeV}$) at the MA irradiation position under the geometrical and spatial restrictions of the subcritical core and the spallation source. These performances are regarded as being within the reach of the extrapolated cyclotron technology of IBA. Compared with the largest continuous wave (CW) neutron source — the Swiss spallation neutron source at the Paul Scherrer Institute (SINQ at PSI) with its cyclotron generated proton beam of 590 MeV and 1.8 mA — it is a feasible extrapolation.

The MYRRHA normal conducting cyclotron would consist of four magnet segments of about 45° (Fig. 2) with two acceleration cavities at approximately 20 MHz RF frequency. The diameter of the active field is in the order of 10 m, the diameter of the physical magnets is in the order of 16 m with a total weight exceeding 5000 t. Due to these very large dimensions, the superconducting magnets cyclotron option has been evaluated by the IBA and has led to a reduction of the magnet diameter by a factor of 2.

Nevertheless, taking into account the conclusions of the PDS-XADS work package 3 [9, 10] experts' group related to the accelerator reliability to be achieved for the ADS application, the linear accelerator (LINAC) option is now the favoured solution for the MYRRHA accelerator.

4.4. Subcritical core configuration

As mentioned previously, due to the objective of obtaining a fast spectrum core and the criterion that no revolutionary options were to be considered, SCK•CEN started the neutronic design of the subcritical core based on MOX classical fast reactor fuel technology. The fuel assembly design had to be adapted to the Pb–Bi coolant characteristics, especially for its higher density as compared with Na.

A first core configuration with the typical 'Superphénix' hexagonal fuel assembly (122 mm flat-to-flat with 127 fuel pins per assembly) with a modified cell pitch to answer the requested performances has been conceived. Nevertheless, this configuration is subject to the large radial burnup and mechanical deformation stress gradients that will make the reshuffling of the fuel assemblies difficult or even impossible. Therefore, the design moved towards a smaller fuel assembly, 85 mm flat-to-flat, with 61 fuel pins per assembly

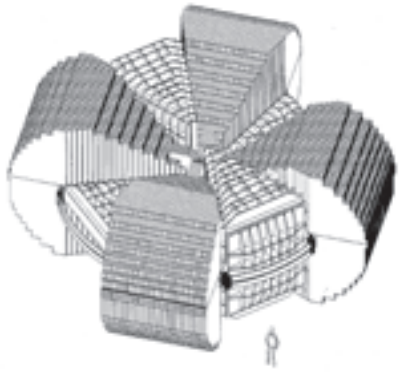

FIG. 2. MYRRHA high power proton accelerator cyclotron.

allowing a larger flexibility in the core configuration design. Indeed, the reactivity worth in the MYRRHA core of such a fuel assembly ranges between ~ 450 and 1600 pcm.

The active core height is kept to 600 mm and the maximum core radius is 1000 mm with 99 hexagonal positions. Not all the positions are filled with fuel assemblies but could contain moderating material (to create thermal neutron flux trap with $\Phi_{th} = \sim 2 \times 10^{15}$ n·cm^{-2}·s^{-1}) or a fast spectrum irradiation device. A typical MYRRHA configuration with k_{eff} of 0.95 can be achieved by using 45–50 fuel assemblies. There are 19 core positions accessible through the reactor lid capable of housing experimental devices equipped with their own operating controls supplied by services above the reactor lid. All other positions can house either fuel assemblies or non-on-line serviced experimental rigs.

The expected performance [11], in terms of fast and thermal flux, linear power in the core and total power in MYRRHA, is summarized in Table 1.

Two interim fuel storage options are foreseen inside the vessel on the side of the core fixed to the diaphragm. They are dimensioned for housing the equivalent of two full core loadings, ensuring in this way that no time consuming operations must take place in the out-of-vessel transfer of fuel assemblies or waiting for an approximate 100 d cooldown.

The MYRRHA operation fuel cycle will be determined by the k_{eff} drop as a function of the irradiation time or the core burnup. The targeted operating regime is three months of operations and one month for core reshuffling, experiment loading and maintenance. This will lead to a drop in k_{eff} of about 1000 pcm at maximum (16% drop in multiplication factor), which has only a minor effect at the locations for MA transmutation (18% flux reduction). Core reshuffling and partial core reloading with fresh fuel would allow compensating this loss of k_{eff}. After three operation cycles, it is foreseen to have a longer shutdown period, namely, three months instead of one month, for more in-depth maintenance and inspection, such as, for example, the replacement of the highest irradiated parts of the spallation target circuit.

4.5. MYRRHA subcritical reactor configuration

Due to the main objective of the MYRRHA facility of obtaining very high fast flux levels, it was obvious that the design should go towards a fast reactor core [12, 13]. As the aim was to realize the objectives within a limited time span and due to the high linear power to be achieved, it is obvious that a gas cooled fast reactor option was the best option. Indeed, at normal operation conditions, the thermal hydraulic problems related to the use of helium (or carbon dioxide) coolant in the MYRRHA subcritical core could be resolved

TABLE 1. MYRRHA FACILITY PERFORMANCES

Neutronics parameters		Units	Reference core	Core with H_2O thermal island
Spallation source	Proton beam			
	Energy E_p	MeV	350	
	Current I_p	mA	5	
	Source intensity	10^{16} p/s	3121	
	Neutron source			
	n/p yield (Y)		6.14	6.01
	Intensity	10^{17} n/s	1.92	1.88
Subcritical core	k_{eff}		0.9541	0.9359
	k_s		0.9590	0.9414
	Importance factor		1.13	1.10
	$MF = 1/(1 - k_s)$		24.40	17.07
	Fission rates/p		48.8	32.8
	Thermal power [a]	MW	51.2	34.4
	Ave. power density	W/cm^3	289.5	165.1
	Max. linear power			
	Around the target	W/cm	509.3	
	Around the thermal island	W/cm		705.4
	Max. flux > 1 MeV	10^{15} n·cm^{-2}·s^{-1}	0.76	0.57
	Max. flux > 0.75 MeV		1.00	0.74
	Max. flux < 0.5 eV (thermal island)			1.17
	Number of fuel pins		2745	3233

[a] E_f = 210 MeV/fission.

only by using high pressures (100–150 bar). However, even at such high pressure, the power necessary to circulate the gas loop is very high (~2–4 MW for CO_2 or He as compared with 0.2 MW for Pb–Bi). Beside that, a gas cooled ADS is less robust under accidental conditions than an ADS cooled by liquid metal, for example, for a depressurization accident. Therefore, the gas option was discarded.

When considering the liquid metal option, two designs were possible: the loop and the pool options. The loop option has been discarded due to the very

high vessel exposure, the risk of loss of coolant accident (LOCA) and loss of flow accident (LOFA), and the difficulty of interlinking the spallation target loop with the primary reactor cooling loop. Finally, one should mention the desired flexibility in loading and unloading experimental devices that can be more easily achieved in a pool design.

The pool design has been favoured because it avoids the penetration from beneath of the spallation target circuit into the main vessel and thus enhances the safety of the design. It allows also having an internal interim storage easing the fuel handling. The natural circulation (free convection) for the extraction of the residual heat in case of loss of flow (LOF) and loss of heat sink (LOHS) is demonstrated to be feasible, particularly with the large thermal inertia that is also an argument in favour of this design. With an emergency cooling system based on natural circulation both on primary and secondary sides, the core coolability can be ensured practically infinitely, even in situations of a complete loss of pumping power.

4.6. Safety considerations

Even if designers strive to achieve an inherently safe ADS system, one should not underestimate other significant safety considerations required to support the licensing of such an innovative system. As stated above, a number of reactivity perturbation initiating events have been studied in terms of the MYRRHA system. They lead either to negative reactivity effects or to a reactivity increase. The latter cases were addressed in the design to avoid their occurrence.

From the point of view of safety, the aim is to reduce the probability of the events and their associated off-site consequences in order to avoid the need for extensive countermeasures and to offer the licensing authorities the possibility of simplifying, or declaring unnecessary, the off-site emergency planning. This is the well known defence in depth safety approach that is followed in the MYRRHA design.

A common approach for safety analysis of the PDS-XADS projects has been established [14] and will also be applied for the MYRRHA project, for assessing its behaviour in design based conditions (DBCs) and design extended condition (DEC) situations. The list of protected (meaning with the proton beam shutdown) and unprotected (proton beam kept on) transients relevant to MYRRHA has been established and a reactor safety code for transient analysis (RELAP5.3)[1] model for MYRRHA has been developed to simulate these

[1] A special version for Pb–Bi coolant developed by Ansaldo Nucleare has been used.

transients [15]. One of the main accidents to be considered is the total loss of flow plus loss of heat sink accident resulting from a station blackout with the failure of both the primary and secondary systems (the proton beam is also supposed to be cut off). In such a case, natural convection will take over and the question arises immediately with respect to whether the natural circulation is sufficient to remove the decay heat released by the core after reactor shutdown. A first parametric approach to study the emergency cooling has shown that the natural convection mode in principle is fully capable of keeping the system in safe operational status, if some care is taken for possible clad temperature increase.

Recently these results were confirmed by using the RELAP5.3 code adapted for lead–bismuth. The RELAP calculations show that all the safety requirements are met in the design basis accident situations, and even the clad temperature is not significantly increased for protected accidents. In addition, some unprotected accidents are currently being analysed and the first results indicate that a sufficient delay (more than 1000 s) exists for taking the appropriate actions with regard to the primary system.

4.7. Remote handling system

Due to the high activation dose on the top of the reactor and the high potential of alpha contamination, it was decided from the very beginning to design MYRRHA to be operated remotely using robots. The proposed MYRRHA project at SCK•CEN will be operated using remote handling for all maintenance operations on the machine primary systems and associated equipment. Experience from similar projects [16, 17] has shown the importance of considering the implications of remote handling on the design of the plant from the earliest stage. Oxford Technologies Ltd (OTL) has been granted a contract for studying the implications of remote maintenance on the design of the MYRRHA machine and the overall project management. The study was conducted and reported herein following the first four steps of the whole life cycle approach that has previously been used successfully by OTL for the implementation of the remote maintenance system for the JET-Tokamak [18].

The study includes an analysis of the remote handling requirements of MYRRHA, defines an approach to be used for ensuring the implementation of a plant suitable for remote handling and concludes with a concept proposal for a system suitable for the fully remote maintenance of MYRRHA over its entire working life.

A remote handling system based on the ‘man in the loop’ principle implemented with two bilateral forces reflecting servomanipulators working under master–slave mode has been recommended. The slave servomanipu-

lators will be commanded by remote operators using cinematically identical master manipulators supported with camera controlled television feedback. The manipulators will have additional robotic capabilities to maximize operational capabilities; they will be positioned close to the task environment by means of remotely controlled transporters with sufficient reach and degrees of freedom to position the slaves at all relevant locations around the MYRRHA machine. The concept relies on the ability of the servomanipulators and the video feedback systems to create a sense of presence for the operators at the task location. In practice, all of the MYRRHA maintenance tasks will be performed directly by personnel using the arms, a range of cameras and cranes in much the same way as if they were next to the MYRRHA machine themselves. The remote manipulators and transporters will have computer controlled features, which will enhance and simplify the operations.

In-service inspection and repair (ISI&R) is also addressed by means of robotics based on an in-vessel inspection manipulator, periscopes or articulated arms equipped with ultrasonic cameras to be deployed when needed inside the MYRRHA vessel. The development of ultrasonic sensors operating under LBE at temperatures up to 500°C and a radioactive aggressive environment (gamma and neutrons) continues, in collaboration with the Ultrasound Institute at the Kaunas University of Technology, Kaunas, Lithuania [19–21].

5. COMPLEMENTARY R&D PROGRAMME

Despite the fact that it is intended to build this facility with a high degree of conventional technology, there are a number of features which do not comply with this approach. Therefore, since 1997, SCK•CEN has started an ambitious support research programme and is developing it according to the requests being received while making progress with the design. The present support R&D programme covers the following areas of highest uncertainties:

— In terms of the windowless spallation target design, the confluent flow pattern of the target formation co-axial with the proton beam on the one hand, and the compatibility of the LM flow towards the accelerator vacuum on the other hand, are investigated. For the first part, a number of design activities have been and are being performed to study the flow behaviour and to obtain an adequate design. Successful experiments have been performed using water and mercury as simulating fluids. Optimization experiments with water are currently going on. The results of these experiments will be carried over to experiments with the real fluid Pb–Bi. Computational fluid dynamics calculations are performed in parallel and

are indispensable, as it is impossible to experimentally simulate the heat deposition by the proton beam without actually having a beam. In summary, the results of these activities, although not yet totally conclusive, look very encouraging to yield the desired target configuration [5]. For the second part, SCK•CEN is presently carrying out the Vacuum Interface Compatibility Experiment (VICE) [7]. In a large UHV vessel (approximately 6 m high) of spallation loop dimensions, it is attempted to quantify the emanation of approximately 130 kg of Pb–Bi LM at 500°C in the vacuum pumping geometry relevant for MYRRHA and also to assess the resulting vacuum conditions, albeit without being able to provide the proton beam in this experiment.

— The LM corrosion aspects of the coolant are of high concern because MYRRHA would be the first facility in western countries to use the technology other than for experimental evaluation. By keeping close to present knowledge, mainly developed in Russian nuclear programmes, and making use of the knowledge now being acquired by European laboratories, which are collaborating with SCK•CEN, the MYRRHA design uses moderate temperatures and controlled oxygen contents of the LM (the key to the corrosion issue). Nevertheless, for MYRRHA the proposed choices have to be reaffirmed by experimental evidence. A programme has been conceived and experimental results are under way.
— The third aspect concerns the handling operations under LM, i.e. the force feedback mechanical aspects as well as the sensors and the fact that the medium is opaque and monitoring under light visibility is not an option. Development has commenced of ultrasonic sensors with the required properties to work under LM though not in direct contact with it. The concentrated effort is directed to ensure, in the first place, the safe and controlled loading and unloading of the subcritical core but will eventually be widened to all operations under LM. A test pool programme is in development in which key operations will be studied under LM in model form.
— As the remote handling approach is presently favoured for operation and maintenance, it is clear that an R&D support programme should be launched in this area of robotics under liquid metal reduced visibility and hot conditions. This programme is under preparation in collaboration with OTL Ltd and the Ultrasound Institute of the Kaunas University of Technology in Lithuania.
— The R&D programme has been updated to include a fuel programme. Indeed, the choice of conventional MOX fuel does not lead immediately to a straightforward solution, since the cladding needs to be compatible with the coolant LBE, so a minimum of fuel pin and assembly qualifi-

cation is still required under these circumstances. Moreover, some backup of fuel choice must still be considered at this stage. It is considered mandatory that for the licensing authority, the irradiation of the fuel pin(s) to relevant levels of the order of 100 dpa for cladding damage and up to 100 GW·d/t for fuel burnup needs to be successfully demonstrated before fuel manufacturing will be licensed.

— It is mandatory to generate an engineering design database for the preselected structural materials (the 9%Cr + 1%Mo martensitic steel T91 and the classical low carbon stainless steel A316L) for the different uses (fuel cladding, core structure and spallation target: T91; and reactor vessel: A316L) though different forms; thin tubes, plates, welded in MYRRHA irradiation and operation conditions. Therefore, more attention is paid to the selection, fabricability and weldability of the candidate materials.

6. CONCLUSION

By mid-2002, the MYRRHA predesign file had been submitted to an International Technical Guidance Committee (ITGC) to review the predesign phase achieved for the MYRRHA project. This international panel consisted of experts from research reactor designers, reactor safety authorities and spallation target specialists. The conclusions and recommendations of this panel were as follows:

— No show stoppers identified in the project;
— More attention given to safety case studies and iteration to the predesign before entering the detailed engineering phase;
— Some R&D topics to be addressed that can lead to timing bottlenecks very soon, such as fuel pin and assembly development and qualification;
— A decision made on the accelerator option (cyclotron versus LINAC) and, eventually, beam parameters to be revisited.

The MYRRHA team responded to the worries expressed above by the ITGC and worked further on the development of the project.

MYRRHA is responding to the objectives of the XADS facility in terms of demonstration and performance, and responding by design to some key issues related to the LBE ADS, such as:

— The LBE corrosion by leaving the majority of the system at ‘cold’ conditions and limiting the LBE velocity below 2.5 m/s;

— Criticality control during core loading by leaving the spallation target in position and loading from underneath;
— Avoiding spallation target window break by choosing the windowless design;
— Addressing the ISI&R and the O&M from the conceptual design by means of robotics and ultrasonic visualization.

MYRRHA is a challenging facility from many points of view and it will trigger a renewal of R&D activities within the fission community. Its development will attract young talented researchers and engineers looking for challenges. It will be a new irradiation facility for research and development in Europe for future innovative energy systems.

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8.6. PROTEUS RESEARCH REACTOR

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1. INTRODUCTION

The PROTEUS zero power reactor at the Paul Scherrer Institute (PSI) in Switzerland achieved first criticality in 1968 and since then has been operated as an experimental tool for reactor physics research on test lattices representative of a wide range of reactor concepts. Reactor design codes and their associated data libraries are validated on the basis of the experimental results obtained.

PROTEUS is normally configured as a driven system, in which a subcritical test zone is made critical by the surrounding driver zones. The advantages of driven systems can be summarized as follows:

- Smaller amount of test fuel is required;
- Large range of test zone conditions (including $k_\infty < 1$ states) can be investigated by changes in the driver loading alone, thus avoiding undesirable perturbations to the test zone which would influence the measurement conditions and thus affect the interpretability of the results;
- Necessary reactor control and instrumentation equipment (usually perturbing from the experimental viewpoint) can be located in the outer driver regions, thereby avoiding disturbance of the test lattice.

2. PROTEUS FACILITY

2.1. Zonal structure of PROTEUS

The PROTEUS research reactor is usually configured as a central test zone, driven critical by annular thermal driver regions (Fig. 1). The reactor contains the zones (from the centre moving outwards) outlined in the following discussion.

2.1.1. Test zone

The test zone, in the centre of the reactor, contains the configuration to be examined. This zone with a diameter of typically 500 mm is moderated with

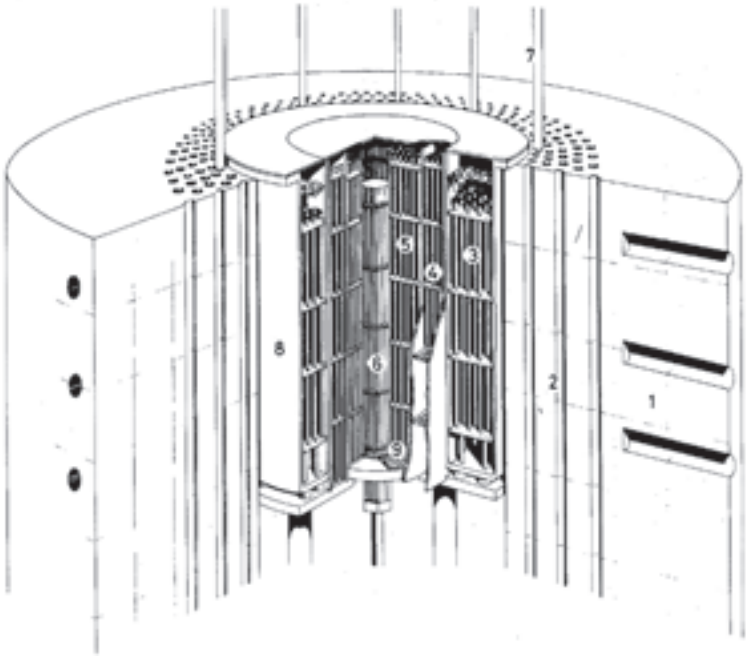

FIG. 1. Cutaway view of the GCFR PROTEUS configuration. (1) Graphite reflector; (2) graphite driver; (3) D_2O driver; (4) buffer zone; (5) fast zone; (6) test column; (7) safety and shutdown rods; (8) D_2O tank; (9) removable grid plate.

H_2O, mixtures of H_2O/D_2O or is without moderator (dry). Thus, lattice configurations for thermal, intermediate and fast reactor systems can be investigated.

2.1.2. Natural uranium buffer

Surrounding the test zone, the buffer region serves to minimize the effect of the thermal driver regions on the epithermal or fast test zone spectrum. The buffer region consists of tightly packed natural uranium metal rods with an active fuel length of 1.5 m in air.

2.1.3. D_2O zone

The D_2O zone serves both as a driver region and as a neutron bridge, acting to maintain sufficient thermal flux in the graphite driver, which in turn guarantees sufficient worth for the safety/shutdown rods. It is fuelled with 5% enriched UO_2 fuel rods, 1 m in length, and moderated with D_2O.

2.1.4. Graphite driver/reflector zone

The graphite driver/reflector zone contains driver fuel, safety, shutdown and control rods, and the nuclear instrumentation. The fuel is identical to the fuel in the D_2O zone. The reflector is a hollow graphite cylinder, 3.3 m in height with an outer diameter of 3.2 m.

2.2. Further features of the PROTEUS

All experiments are performed at room temperature. In the test lattice, mixtures of H_2O/D_2O can be used to simulate the reduced hydrogen density at operating temperature in power reactors.

Reactivity control in PROTEUS is achieved via independent groups of safety, shutdown and control rods, all of which are located either in the graphite driver zone or in the adjacent reflector zone, i.e. near to where the maximum thermal neutron flux occurs. The safety and shutdown rod groups are identical in construction, each consisting of four separately driven boron steel rods. The control rods, also four in number but driven from the bottom of the reactor, are of the 'zebra' type, i.e. the change in neutron absorption is being effected by the relative movement of co-axial cadmium segments. A special fine control rod located in the graphite zone consists of a wedge shaped copper blade with almost linear characteristics. This is mainly used for experimental purposes (reactivity worth measurements) and also enables servocontrol of the reactor while compensating for temperature drifts, etc. After 1998, the PROTEUS safety/shutdown rods were increased from 8 to 16 boron steel rods without changing the number of the rod drives, so that each drive now moves a pair of rods.

For a given type of test zone, the graphite driver loading is adjusted such that criticality is achieved with both groups of safety and shutdown rods fully withdrawn and the four control rods in an intermediate position. The zebra type of the latter, as well as the small worth of the fine control servorod, helps minimize the distortion of axial neutron flux distribution during reactor operation. The maximal thermal neutron flux of 5×10^9 n·cm^{-2}·s^{-1} corresponds to a reactor power of 1 kW and allows performing a wide range of integral reactor physics experiments.

3. CORE CONFIGURATIONS AND EXPERIMENTAL PROGRAMME

The main configurations of PROTEUS have been:

— 1972–1979: 16 configurations simulating the gas cooled fast reactor (GCFR);
— 1981–1990: 20 configurations simulating the light water high converter reactor;
— 1992–1996: 10 single zone configurations simulating the high temperature reactor;
— 1998–2004: 16 configurations simulating the LWR.

In the 1970s, the GCFR was investigated to check the performance of data sets and calculational methods used in fast breeder design. Experiments were carried out using 16 different core configurations. There were four principal materials applied for the test zones in the various configurations, viz. PuO_2/UO_2, depleted UO_2, ThO_2 and Th metal. A central fast test zone was driven critical by annular thermal driver zones. A reduction by a factor ~10 was thereby achieved for the Pu inventory of the fast test lattice relative to that which would have been necessary for a critical single zone reactor of the same composition. The central test zone had a diameter of 500 mm and contained over 2000 fuel rods of 1.40 m nominal length, arranged in a lattice of 10 mm hexagonal pitch. An optimal zone layout enabled the central neutron spectrum in the test zone to closely approximate that of a GCFR, an important criterion for carrying out integral measurements relevant to this type of reactor. A central test column could be moved in and out of the test zone (Fig. 1). Apart from being used for introducing measurement probes, such as foils and neutron spectrometers, the column also provided some flexibility in the experimental programme by being an easily replaceable central region of the test zone. The neutron balance, in test lattices consisting of PuO_2/UO_2 (12% fissile Pu) fuel rods, was investigated by integral reaction rate measurements. Direct neutron spectrum measurements in various environments provided complementary data, as did central reactivity worth measurements. These experiments helped resolve certain important discrepancies between fast reactor design predictions from Europe and the United States of America that were current at that time.

During the early 1980s, there was considerable international interest in the LWHCR concept, viz. the use of tight PuO_2/UO_2 fuelled lattices in pressurized water reactors for achieving high conversion ratios and thus improving uranium utilization. A ‘first of its kind’ series of integral experiments relevant for light water high conversion reactors (LWHCR) was carried out in PROTEUS during 1981–1982 (Fig. 2). Fuel material from the GCFR programme (12% PuO_2/UO_2 and depleted UO_2) was used to construct test lattices simulating fissile plutonium enrichments of 6% and 8%, respectively. Each of the test lattices was constructed using two different fuel rod types: 12% fissile plutonium mixed with depleted UO_2 as mixed oxide fuel

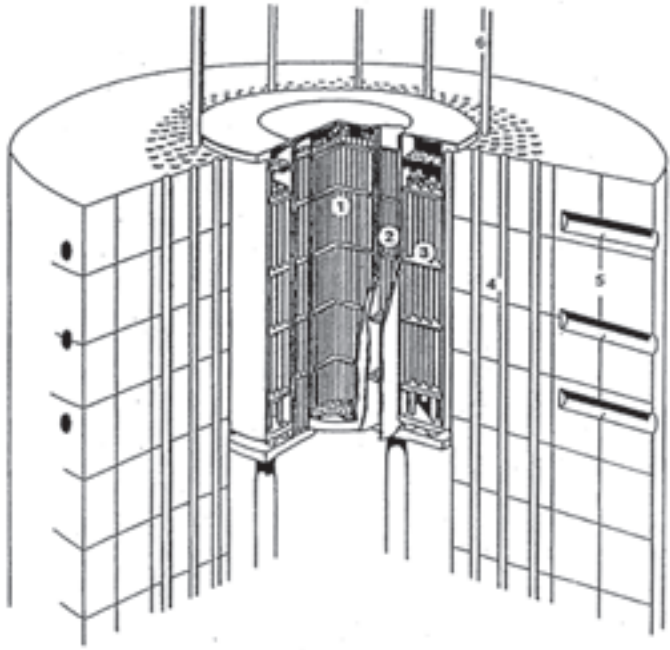

FIG. 2. Cutaway view of the LWHCR PROTEUS configuration. (1) Test zone; (2) buffer zone; (3) D_2O zone; (4) graphite driver; (5) graphite reflector; (6) safety and shutdown rods.

(MOX), the plutonium in the MOX corresponding to that from Magnox reactors. Three different moderators were used, viz. H_2O, Dowtherm (an organic liquid) and air (dry lattice), simulating 0, 42.5 and 100% void.

In 1985, phase II of the LWHCR programme was started with the same three moderators but in a single rod lattice of 7.5% fissile plutonium enrichment with the isotopic composition of the plutonium corresponding to that discharged from LWRs. The moderator to fuel volume ratio (V_m/V_f) was 0.5. The results from the tight pitch PROTEUS experiments were instrumental in shifting the emphasis towards core designs with 'wider' lattices ($V_m/V_f \sim 1.0$) than those originally considered desirable. In those lattices, safety related aspects were measured, including voidage effects, control rod worths, heterogeneity effects and the influence of B_4C poisoning. Comparisons of calculational and experimental results for k and reaction rate ratios in the tight and wider lattices revealed significant trends with moderation. To further broaden the database, a 'very wide' LWHCR lattice ($V_m/V_f \sim 2.0$) was investigated towards the end of the programme.

A programme of safety related reactor physics studies of the high temperature reactor with low enriched uranium fuel (LEU HTR) was launched in 1992. For these experiments, the reactor was configured as a single zone system, viz. a pebble bed core surrounded by graphite reflectors (Fig. 3). The main objective of the programme was to provide integral data related to criticality, control rod worth, and the effects of accidental water ingress and neutron streaming on the neutron balance in small and medium sized LEU HTR systems. Various combinations of pebble bed fuel elements, containing 16.7% enriched UO_2 particles, with bare moderator pebbles were used to study a range of moderation ratios and packing geometries. This experimental programme was completed in 1996.

Another experimental programme for code validation of modern LWR fuel assemblies started in 1998. The goal of the LWR PROTEUS project, a cooperative effort between Swiss Nuclear Utilities and the PSI, has been the validation of calculational codes in estimating reactor physics parameters, such as power distributions, reactivity and burnup effects in actual, full size LWR assemblies, which were loaded in the test zone of the reactor. Since these assemblies were 4.5 m in length and the active height of the PROTEUS driver regions is about 1 m, the test tank could be moved axially to enable stepwise investigation along the whole length of the test assemblies (Fig. 4). This type of system did, however, have some limitations for a large test zone of LWR fresh fuel, for instance, on the reactivity of the test zone. A highly reactive test zone can reduce the importance of the driver regions to such an extent that the control and shutdown systems have too little worth, which was a problem in the

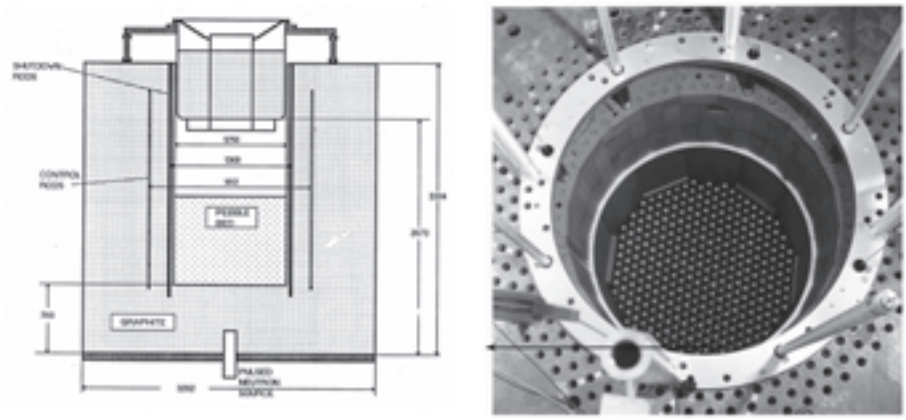

FIG. 3. Vertical cross-section of the HTR PROTEUS configuration (dimensions in mm, left) and top view of the pebble bed (right).

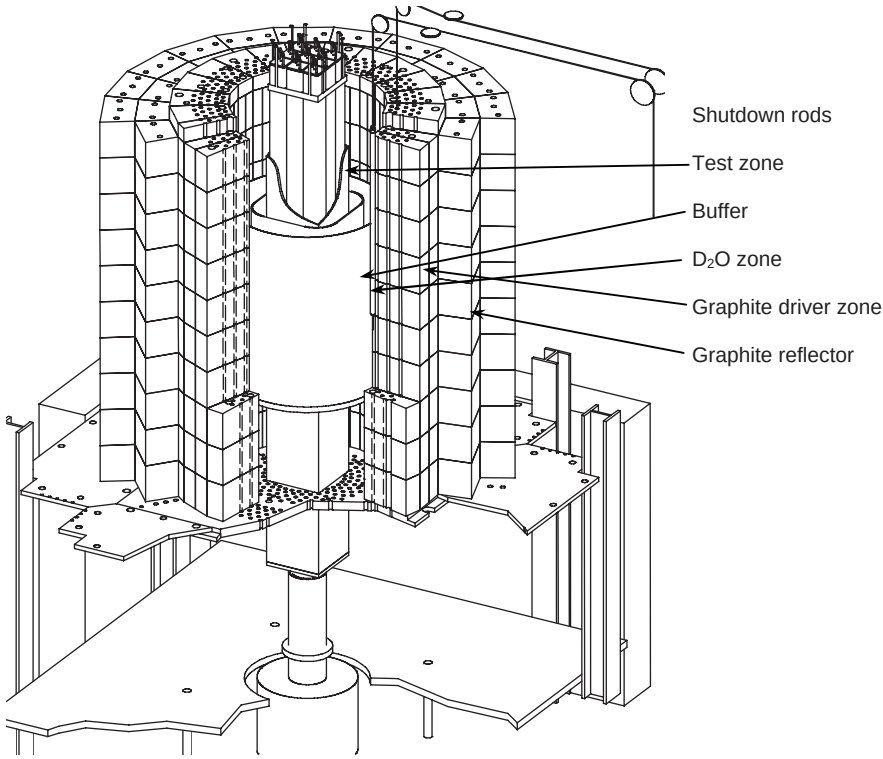

FIG. 4. Cutaway view of the LWR PROTEUS facility, showing the central test zone (vertically drivable).

early design phase of the LWR PROTEUS project. The problem was overcome by changes in the D_2O driver loading and by an increase in the number of shutdown and safety rods from 8 to 16. During phase I of that project, the test zone has been configured with 9 SVEA-96+[1] fuel assemblies (Fig. 5).

Each (SVEA-96+) assembly comprised 96 fuel pins arranged on a square pitch around a central water canal. The ^{235}U enrichment varied both axially and radially in the range of 2% to 5% and there are up to 16 Gd containing pins in each assembly. During the LWR PROTEUS phase II, the test zone was configured with a central PWR mock-up (mock-up with 11 × 11 four fresh PWR rods, 4.3 w% enriched), which were inserted into a central tank, the PWR tank, thus enabling the possibility to moderate this central PWR mock-up

[1] SVEA-96+ refers to a Westinghouse type BWR fuel assembly.

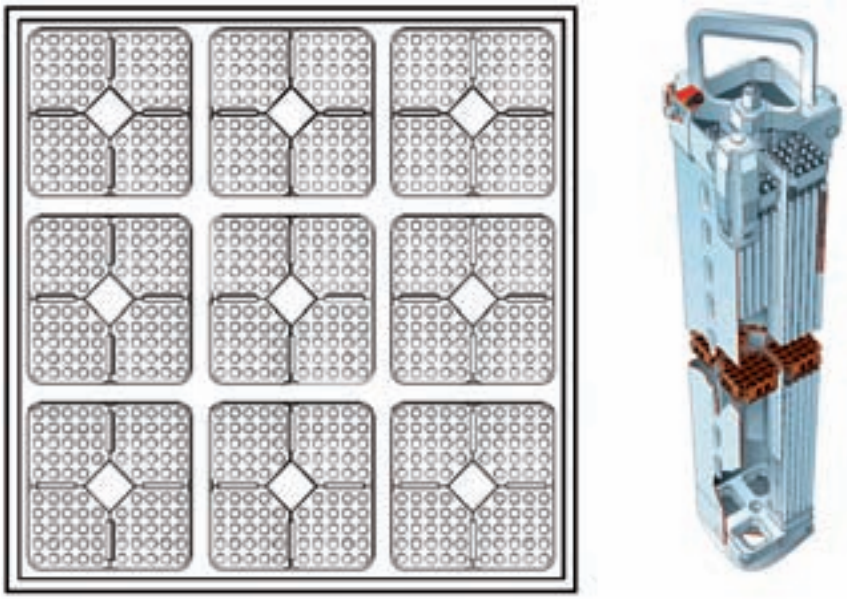

FIG. 5. Typical test zone configuration in LWR PROTEUS phase I (left). Westinghouse Atom (SVEA-96+) BWR fuel assembly (right).

independently from the moderation in the remaining test zone. The central pin position in the PWR mock-up is replaced by a guiding tube, which allows the insertion/removal of fresh and spent fuel assembly segments. The PWR mock-up and its PWR tank are surrounded by BWR fuel assemblies (SVEA-96+ before March 2002, OPTIMA-2[2] afterwards). For the reactivity experiments with highly active spent fuel segments, a specially designed and constructed container/sample changer was used (Fig. 6).

The reported activities and experiments performed at and with the PROTEUS facility clearly demonstrate the outstanding flexibility of the basic set-up in terms of the requirements originating from the different core types of nuclear power reactors under investigation. It is that feature of the PROTEUS that makes the plant valuable for the tasks associated with the approach to new power reactor systems, for example, within Generation IV and, thus, will ensure its further operation. However, updates of the safety features and the careful follow-up of ageing have to be addressed.

[2] OPTIMA-2 refers to a SVEA-96+ assembly with part-length rods.

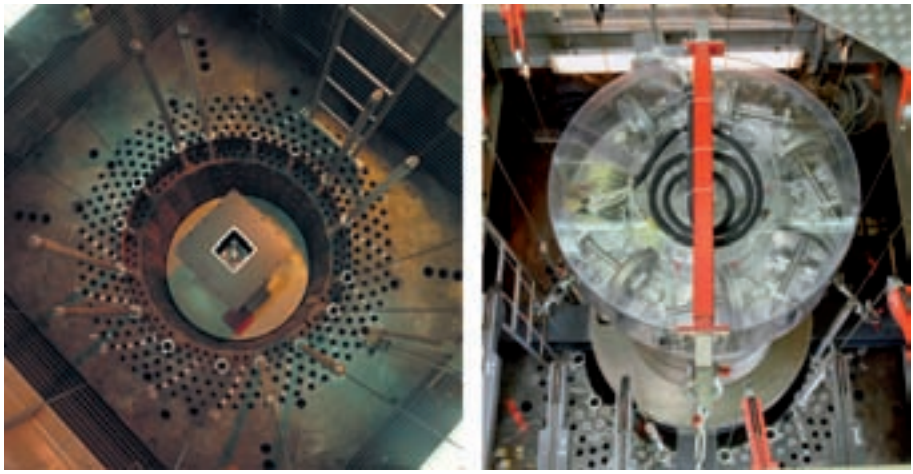

FIG. 6. View to the top of the reactor (left) and photograph showing the container/sample changer (right), sitting on top of the test zone, containing up to four high burnup fuel segments.

4. FUTURE ACTIVITIES

After more than 35 years of reactor operation, the Swiss Federal Nuclear Safety Inspectorate required a refurbishment of the PROTEUS facility with respect to the nuclear instrumentation and the control system. A new experimental reactor physics programme, starting in 2007, is envisaged for the detailed characterization of a full size high burnup LWR fuel assembly in the presence of fresh fuel assemblies. The general goals of that new programme are:

— Non-invasive measurements of a large array of burnt fuel rods surrounded by fresh fuel rods, and the validation of core loading follow-up calculational data;
— Experimental validation of the power distribution mismatch between very different fuel assemblies (fresh and burnt) under clean conditions, with and without the presence of control absorbers;
— Experimental acquisition of data related to instrumentation response, e.g. in the assembly gap of two very different assemblies (presence of a strong gradient) and the sensitivity of this response to a typical departure from nominal conditions (tube vibrations, bending, etc.); the purpose here is

the validation of detector models and detector constants in assembly codes;
— Influence of high burnup on safety characteristics, such as the void coefficient and control absorber effectiveness;
— Studies involved in the investigation of the power mismatch and in-core instrumentation effects for fresh and burnt assembly interfaces, for a range of simulated moderation and control conditions.

This programme with large amounts of spent fuel requires an upgrade of the safeguard and security installations of the facility, and special installations and shieldings to handle this type of fuel. The analytical work will be mainly devoted to the validation of assembly codes, as well as the capabilities of nodal methods to predict neutron flux transitions across interfaces between very dissimilar fuel assemblies.

Part 9

NEUTRON ACTIVATION ANALYSIS

9.1. INTRODUCTION TO NEUTRON ACTIVATION ANALYSIS

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1. PRINCIPLES OF NAA

Neutron activation analysis (NAA) is one of several analytical methods that may be used for the identification and quantitative measurement of trace elements. The technique is applicable to about 50 elements, including the rare earths, and offers sensitivities in the ppb range. NAA is closely associated with nuclear research reactors because these facilities, even ones of relatively moderate power levels such as 250 kW, offer sufficiently high neutron fluxes to achieve good sensitivities. However, other neutron sources, such as accelerators and isotopic generators, may be used.

The basic principle of NAA is to subject a sample (the unknown) to neutron bombardment with the objective of causing neutron absorption by the atoms of the elements that are to be identified. Thus, sensitivity is dependent on both the neutron flux of the available source and the absorption cross-section of the unknown. The absorption process results in an excited compound nucleus that de-excites by the emission of one or more gamma rays. Those are referred to as 'prompt' because they appear almost immediately. The de-excited nucleus is usually radioactive and, being neutron rich, will undergo beta decay (neutron converted to a proton with emission of an electron and a neutrino) with the subsequent appearance of one or more gamma rays that are referred to as 'delayed'. The delay is the time required for the beta decay to occur. Both the prompt and delayed gamma rays have unique energies that are characteristic of the nuclide that produced them. Hence, if one has a tabulation of possible decay schemes and if one can accurately measure the gamma ray energies, then it is possible to identify the unknown nuclides. Also, concentrations can be obtained if a standard that contains known amounts of the relevant trace elements is irradiated and counted under the same conditions as the unknown.

NAA has evolved into two distinct approaches depending on whether prompt or delayed gammas are used. If the former are to be measured, then the technique is known as PGNAA (prompt gamma neutron activation analysis). Conversely, if the latter are utilized, then the abbreviation is DGNAA (delayed

gamma neutron activation analysis). However, because DGNAA is the dominant mode, the acronym NAA is often used to mean DGNAA.

2. PROMPT AND DELAYED GAMMA NEUTRON ACTIVATION ANALYSIS

PGNAA is usually performed by placing the sample in a reactor beam port and by positioning the needed counting equipment adjacent to the port. PGNAA is done in real time with data analysis being performed concurrent with the irradiation itself. In contrast, DGNAA is done by placing the sample in a neutron flux field such as exists in a reactor core (or in an experiment tube that is adjacent to the core). The sample is irradiated with the duration of the irradiation being longer if one desires to identify elements at very low concentrations or with low absorption cross-sections, or if the resulting radioactive isotopes have long half-lives. The sample is then removed from the neutron flux, allowed to decay until safe to handle, and then counted. Beam port flux is orders of magnitude less than that available in-core. Thus, the set of possible applications for PGNAA and DGNAA differ, with PGNAA being limited to nuclides with large absorption cross-sections and DGNAA being the method of choice for everything else.

DGNAA may be further subdivided according to whether or not post-irradiation handling of the sample is required. If the irradiated sample can be counted without any chemical separations, the technique is called instrumental neutron activation analysis (INAA). Many (perhaps half) the elements that can be analysed using NAA can be detected at good sensitivities in this manner. If it is necessary to do post-irradiation chemical processing, such as separations, or to increase radioisotopic concentration, then the technique is called radiochemical neutron activation analysis (RNAA).

3. BENEFITS OF NAA

NAA offers a number of advantages relative to competing methods for trace element identification. For example:

— Samples require minimal handling prior to irradiation;
— There is no sensitivity to contamination after irradiation;
— It is non-destructive to the sample matrix;
— The field is technically mature;
— It is applicable to a wide range of elements.

Certain precautions need to be observed in its application. Several of the more important are:

— Research reactors operate at a steady power level but this does not mean that the neutron flux at the NAA irradiation location is constant. Power level is a function of the integral of the flux shape, and the shape can and will vary as control devices are moved to offset the effects of fuel depletion and fission product poisons such as xenon. Hence, standard reference materials should be used in order to offset the effect of flux variations.
— Strict adherence to cleanliness standards is essential during sample acquisition, preparation and packaging so as to avoid contamination with possible trace levels of impurities.
— Allowance must be made for the possibility of one or more gamma rays masking others. For example, situations sometimes arise where a short lived isotope that produces a high intensity gamma ray is present and this masks the signals from some of the trace materials. One solution is chemical separation. Another might be to do both short and long irradiations of the same sample. The former would activate only short lived nuclides. The latter would activate all nuclides, however, by allowing the sample to decay before counting, only the long lived ones would be seen.

There are many possible applications of NAA and many of these are described in the papers that follow. Several possible applications are noted here in order to provide an appreciation for the breadth of the field. The first is a prompt gamma application; the others are all delayed gamma:

— PGNAA is used to analyse boron in the blood of patients who will undergo boron neutron capture therapy (BNCT). This is an ideal application because boron has a large neutron absorption cross-section and knowledge of the boron concentration is needed as the BNCT proceeds. There is no time for a chemical analysis (see Part 5 for a discussion of this application);
— Ratios of trace elements in artefacts from archaeological sites are determined and then compared with those obtained from possible points of origin of the raw materials that were used to make the artefacts. In this way, trade patterns among ancient civilizations can be discerned;
— Heavy metal concentrations in hair can be used as an indicator of possible health conditions. For example, the hair of autistic children will show a lack of mercury (compared with controls), indicating retention in the body;

— Ratios of trace elements in coal used in a particular power plant can be compared with the same ratios associated with air particulates collected at sites remote from the power plant and thereby determine if the plant in question is or is not a pollution source for that site;
— The quality of groundwater can be monitored by tracing heavy metal concentrations and identifying trends;
— Trace element abundances in volcanic rocks can be determined, specifically, abundances of trace elements in volcanic rocks can be used to constrain:
  • Mineralogy and composition of the mantle source that was partially melted to create the magmas;
  • Extent of partial melting and how it changed with time;
  • Processes that affected the magmas during their ascent to the Earth's surface and their subsequent interaction with the atmosphere–ocean system.

4. AN OUTLOOK

NAA is, despite its strong advantages, facing an uncertain future. The number of research reactors is declining and so is the number of students who are developing expertise in the field. In addition, the competing technology of inductively coupled plasma (ICP) is growing in popularity. ICP involves exposing samples to plasma. This causes the atoms in the sample to emit light at wavelengths characteristic of the elements that are present. When coupled with a mass spectrometer, ICP can be used to identify individual isotopes. ICP's advantage is that it can detect every element except argon. Its drawback is that samples must be in liquid form or have been subject to acid digestion. Many laboratories that have traditionally offered NAA now also offer ICP-MS. The use of these two very different technologies in parallel provides a greater level of confidence in the resulting analysis than does either approach alone.

The papers to follow in Part 9 demonstrate the power and versatility of NAA.

9.2. DESIGN FEATURES AND EARLY OPERATIONAL EXPERIENCE WITH DHRUVA PNEUMATIC CARRIER FACILITY FOR ACTIVATION ANALYSIS

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1. INTRODUCTION

The Pneumatic Carrier Facility (PCF) installed in the 100 MW Dhruva Research Reactor enables researchers to carry out irradiation of very small quantities of materials that generate radioactive nuclides with half-lives of the order of a few seconds. Since commissioning, this facility is being regularly used for a variety of studies, such as neutron activation analysis, nuclear fission and delayed neutron measurements. The facility consists of three major sections: (a) the in-core assembly along with its piping and the sample carrier; (b) a sample sending and receiving station including the control console; and (c) an adjoining radiochemistry laboratory with all measuring instruments, etc. A polypropylene capsule (rabbit) of dimensions 25 mm diameter × 40 mm long is used as a sample carrier. It is pneumatically conveyed, using the sending and receiving station, into the in-core pneumatic carrier rod. Here the sample undergoes irradiation for a specified period of time after which the sample is automatically ejected out of the reactor core and received in the receiving station. In the receiving station, the sample is retrieved from the 'rabbit' and taken to the adjoining laboratory to carry out the required measurements. The transit time for the capsules varies from 2 to 5 s each way. The irradiation time for a capsule can be preset between 5 s and 999 s. The salient design features of this facility and the commissioning experiences are presented in this paper.

2. DHRUVA RESEARCH REACTOR

Dhruva is a 100 MW tank type multipurpose research reactor located at the Bhabha Atomic Research Centre (BARC), Mumbai, India. The research reactor is fuelled with natural uranium and is cooled, moderated and reflected by heavy water. Heat from the coolant is removed in tubular heat exchangers by demineralized water circulating in a closed loop. The secondary coolant, in turn, is cooled in a separate set of heat exchangers by seawater, which acts as the ultimate heat sink. The reactor core is vertical and coolant heavy water flows

from bottom to top through fuel assemblies located inside coolant channels in the reactor vessel. Maximum thermal neutron flux obtained in the reactor core area is 1.8×10^{14} n·cm^{-2}·s^{-1}. The reactor has 146 lattice positions that accommodate fuel assemblies, shut off rods and irradiation assemblies. Out of these, one position (C-25) has been used for the pneumatic carrier facility.

3. SALIENT DESIGN FEATURES OF THE PNEUMATIC CARRIER FACILITY

3.1. Pneumatic carrier rod

The pneumatic carrier rod (PCR) (Fig. 1) consists of three aluminium tubes (capsule tube, capsule return and cooling air tube and rod cooling air

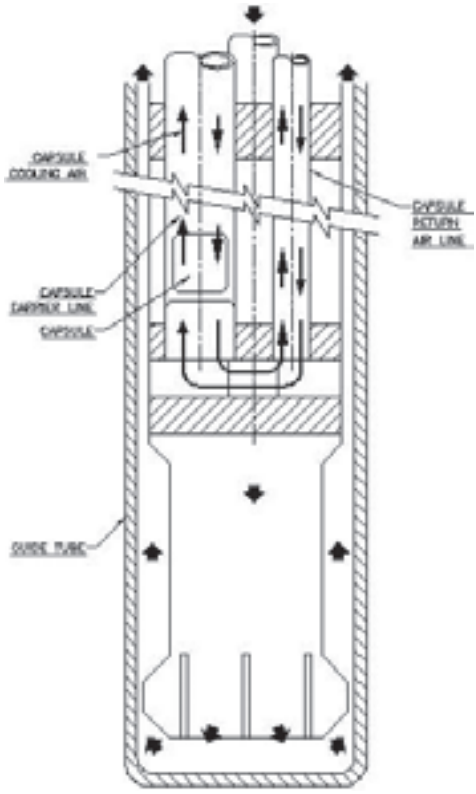

FIG. 1. Schematic of PCR in-core region.

tube) inside a main (outer) aluminium tube. The inner tubes and the main tube are joined by suitable connectors. The main tube ends at the capsule seat in the PCR. The capsule return line serves for driving airflow for returning the capsule as well as for providing a return path for the cooling airflow to the capsule at the time of irradiation. It ends in a junction box. The junction box consists of a capsule seat with several small nozzle like holes for airflow and an air plenum. The PCR cooling line is for separate continuous cooling of the PCR. The research reactor will be tripped on low as well as high cooling airflow. This cooling airflow dissipates the nuclear heat from the PCR.

In order to take care of the neutron and gamma rays streaming through annuli and cylindrical ducts, a line of sight principle has been applied for the shielding design (Fig. 2). The three internal tubes in the PCR are formed into a helix to avoid direct streaming. The mean diameter of the helix is kept more than the internal diameter of the pipe. The maximum outer diameter of the helix is also limited by the available space and hence capsule line, return line and rod coolant line are intertwined to form a composite helix having mean diameters of 30 mm, 38 mm and 38 mm, respectively, with the pitch being 2700 mm (Fig. 2). A garter spring is provided on the PCR to maintain a uniform gap with the annulus duct and also for middle support. The PCR is designed to be handled by the fuelling machine. The top support block of the PCR has been provided with tapped holes for handling. This part rests on coolant channel guide tube assembly top and the rod is locked in position by bolting the top hat onto the stump tube extension assembly of the coolant channel.

3.2. Capsule sending and receiving station

The capsule sending and receiving station (Fig. 3), as the name suggests, is a dual purpose cubical box made of stainless steel. It has a hinged top lid with a gasket for sealing. The box is fitted with hinged C clamps for closing the lid tightly. The box is fixed inside a fume hood. It has four openings other than the top lid. They are:

— Right side is connected to the stainless steel capsule carrier line;
— Left side is connected to the capsule shooting air line;
— Rear side is connected to the capsule return air exhaust line;
— Bottom is connected to a ball valve for dropping the irradiated capsule into a shielded lead flask.

The inside of the box is fitted with a half round capsule seat which is hinged to a lever. The capsule seat can be turned upside down by actuation of

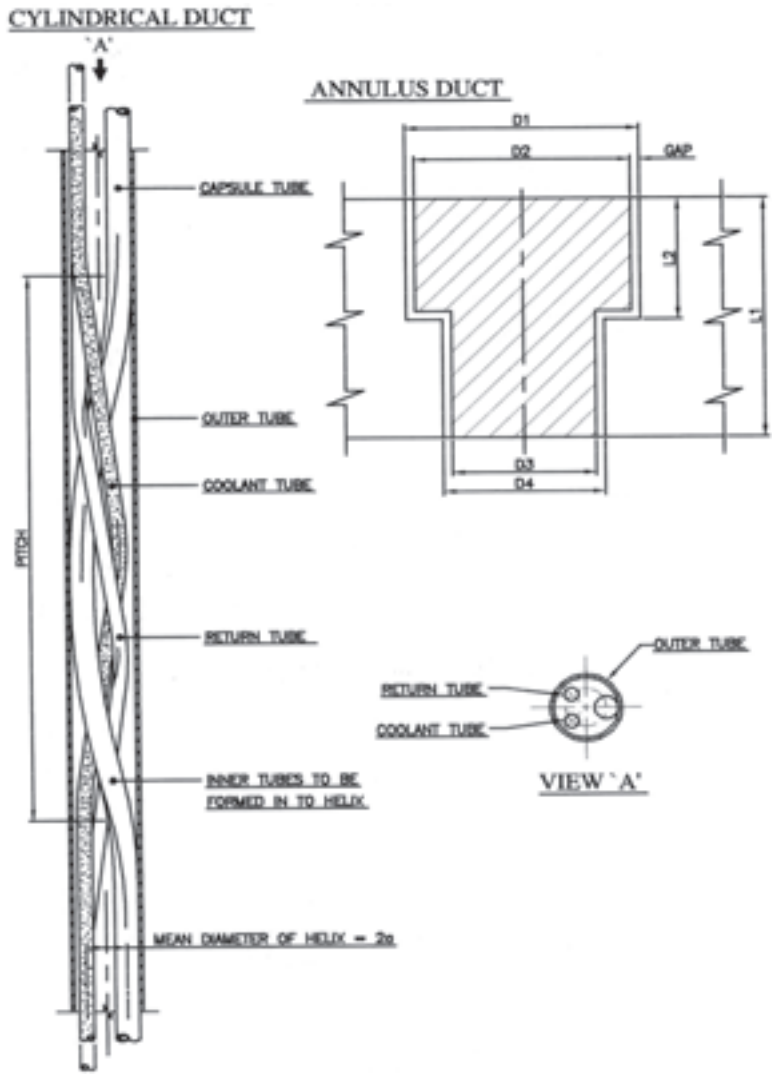

FIG. 2. PCR shielding design.

the lever to enable dropping of the capsule into a collection flask. The schematic diagram of the station is shown in Fig. 3.

The steering of the sample carrying capsule from the shooting station to the reactor and back is controlled by an electronic control system housed in a console located adjacent to the capsule shooting station. This console permits shooting capsules in both auto as well as in full manual mode. The control

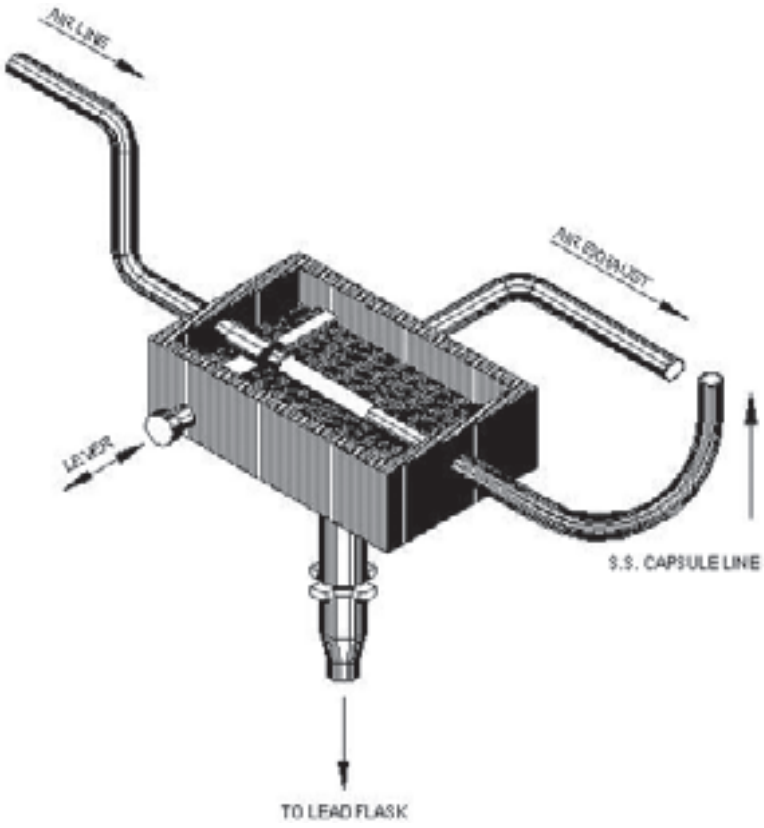

FIG. 3. Schematic diagram of the capsule sending/receiving station.

system operates on the basis of the signals generated by the infrared based capsule detection system. These detectors are located on the capsule lines for monitoring the movement of capsules to and from the reactor. The schematic diagram of the detector housing is shown in Fig. 4.

The control console consists of a box like structure made of aluminium sheets and supported on an angle structure. All the control switches and indicators are located on the front door of the panel. All the control timers, relays and other associated components and circuits are suitably located inside the console. The PCF control is on hard-wired relay logic. The relays in turn operate contactors. The contacts of the contactors are used to operate the individual solenoid valves. For all relays, contactors, timers (except for the irradiation timer) and solenoid valves, a 48 V DC power supply unit is used. For

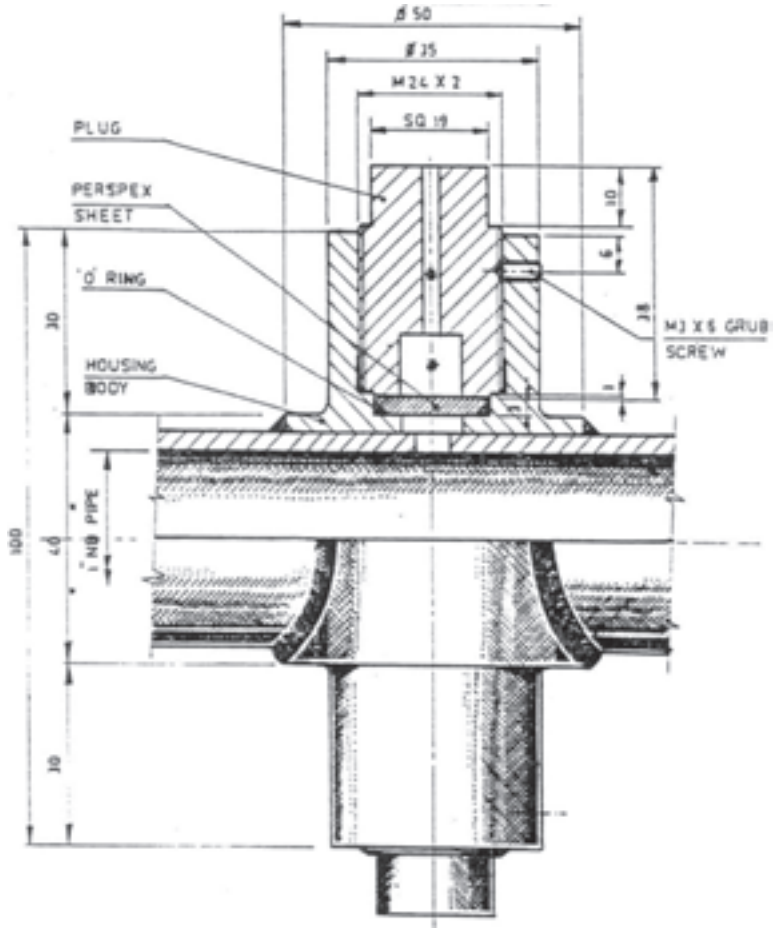

FIG. 4. Schematic diagram of the infrared detector housing.

infrared detectors and its associated circuitry, a separate 12 V DC power supply is used. The irradiation timer is supplied with 230 V AC supply. The control console incorporates the following features:

— Maximum time of irradiation that can be set is 999.9 s;
— Irradiation time measurement starts only after the capsule has actually entered the reactor pile;
— Only digital timers have been used for accurate time measurements;

— Proper delays in operation of solenoid valves have been incorporated to ensure the fail-safe return of a capsule in all circumstances;
— Sample cooling airflow indication (digital) is provided in the panel;
— Provision for positive on-line testing of detectors provided;
— Provision made for full manual operation as well as full auto operation;
— Detector signals also used for visual indication on the control console.

3.3. System piping layout

The system piping and its layout have been designed on the basis of functional requirements, available space and the flow diagram (see Fig. 5). For the capsule line from the PCF room to the PCR, the number of joints has been kept to a minimum to reduce potential leak points and capsule obstruction points. The bending (where required) radius of the capsule carrier line is kept more than the minimum specified radius of 500 mm. The total length of the capsule carrier line is about 40 m. Two detector housings have been installed at both ends of the capsule carrier line. Each of these housings accommodates an infrared light source and an infrared detector.

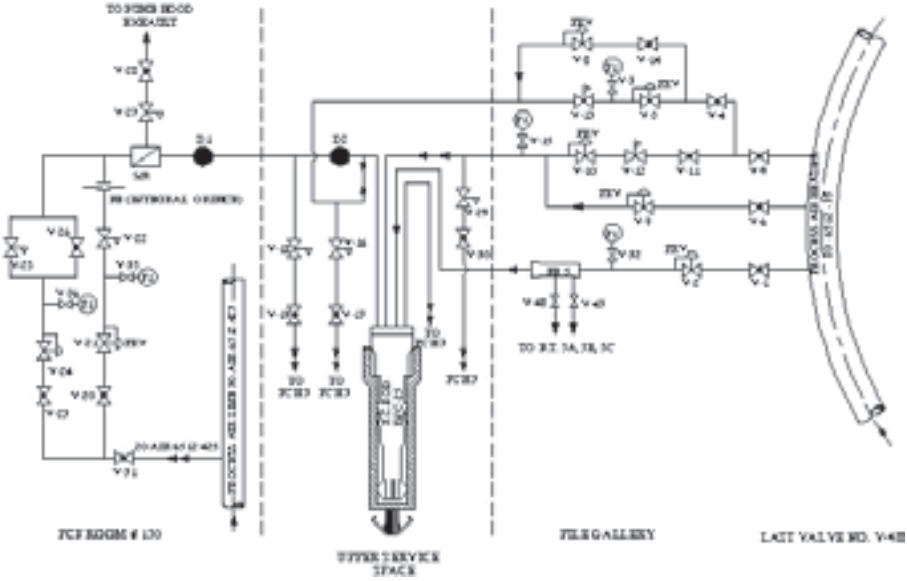

FIG. 5. Dhruva PCF flow diagram.

The system has been divided into three zones as per their locations; they are as follows:

— Zone I: PCF room;
— Zone II: pile gallery, reactor hall and first floor corridor of reactor annex;
— Zone III: PCR.

These zones are physically identified with the help of two infrared detectors, one located in the PCF room (D1) and one located inside the upper service space (D2). Thus, zone I is the capsule line lying between detector D1 and the send/receive station; zone II is defined as the capsule line between D1 and D2; and zone III is from D2 to the PCR assembly. These zones are defined to aid in the location of the capsule in case of any problems encountered during shooting.

3.4. Capsule (rabbit) design

The sample carrier capsules used in Dhruva PCF for sample irradiation are shown in Fig. 6. The material of construction of the capsules is polypropylene. This material can withstand neutron fluence of the order of 10^{18} n/cm^2. On this basis, the upper limit on total irradiation time on a single capsule has been set at 10^3 s inside the reactor core at full operating power of the reactor.

In view of that total irradiation time, the control system has been designed so that in auto mode, that time cannot be set more than 999.9 s at a time. The capsules are checked for proper size by passing through a set of go/no go gauges to ensure its safe transmission into and from the reactor pile. The material and the quantity of a sample for irradiation to be loaded into the capsule is chosen such that the temperature of the capsule does not exceed 125ºC or the heat produced by the irradiation sample is less than 500 mW. Compliance with these conditions is met by well set procedures already being followed for any pile irradiation request (PIR) received at Dhruva.

3.5. System operation

A well set procedure has been evolved for the operation of the facility. Accordingly, an irradiation request form, known as a PIR form, is filled in by the researcher and submitted to the reactor physicist along with the sample. The reactor physicist analyses the data given in the form and specifies restrictions, if any, to be applied. After obtaining clearance from the reactor physicist, the sample is put inside the capsule which is then loaded in the S/R station. Before shooting the capsule into the core, the reactor power is noted. The timer

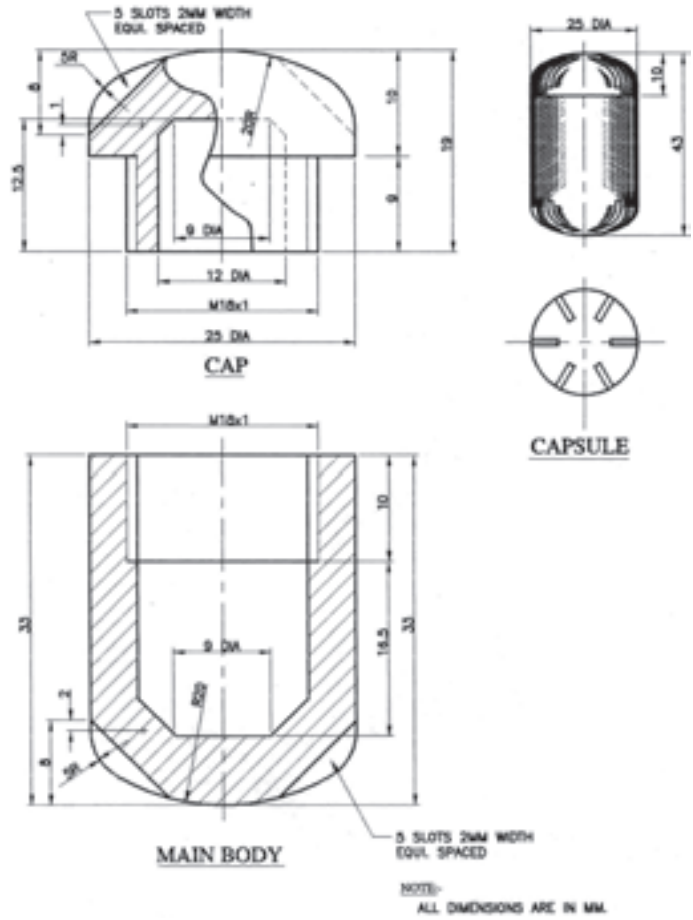

FIG. 6. PCF, capsule assembly and details.

on the console is set to the required value and the capsule is shot into the core. The capsule returns to the same loading station automatically after the completion of the irradiation. The sample is then removed from the capsule and taken to the adjoining counting laboratory for analysis.

4. SPECIAL DESIGN FEATURES OF DHRUVA PCF

The Dhruva PCF has some unique features that do not exist in other research reactors located at BARC:

— A dual air circuit concept for the return of capsules has been introduced. The capsule is ejected out-of-pile using a localized air circuit. This air, after coming out-of-pile, is exhausted through the pile ventilation exhaust system. As soon as the capsule comes out-of-pile, a separate fresh air circuit automatically conveys the capsule up to the receiving station. This concept has eliminated the possibility of contaminated air from the pile reaching the laboratory area.
— Infrared detectors have been used to detect the movement of the capsules. This has greatly improved the reliability and accuracy of detection of the capsule movement during shooting and return of capsules.
— Digital timers have been used to measure various time settings. This has resulted in improving the accuracy of the time data.
— The capsule size (sample carrier) in Dhruva is much larger (about four times in volume) compared with the capsules used in other BARC reactors.

5. COOLING REQUIREMENTS

The steering of the capsule is achieved with the help of compressed air. Two separate paths have been provided for the cooling air.

5.1. Cooling air for the PCR

The PCR cooling airflow requirement has been determined on the basis of estimated total heat generation by neutron and gamma rays due to their interactions with the structural materials of the PCR. On the basis of this and the total volume of the PCR inside the reactor core, the normal cooling flow has been specified as 1500 standard litres per minute (sL/min) with a low flow alarm set at 1200 sL/min and high and low flow reactor trips set at 1800 sL/min and 900 sL/min, respectively.

5.2. Capsule cooling airflow

The normal capsule cooling airflow of 300 sL/min (80% of full scale) has been arrived at on the basis of heat generation of 1 W by the irradiation sample. The capsule carrying that sample is cooled by air cooling flow. This cooling flow is provided through the same capsule line and in the same direction as the capsule shooting airflow, except that this flow is started 5 s after the capsule starts its descent into the PCR. The capsule cooling flow is crucial to the safety

of the capsule and the sample; hence a high and low flow setting has been set beyond which auto ejection of the capsule out of the pile is initiated.

6. COMMISSIONING EXPERIENCE

The functional requirements of the facility were verified through various commissioning tests. These tests were also used to optimize the system operating parameters. During commissioning, a few inadequacies were observed and were corrected, as explained below:

- As per design, cooling airflow for the capsule was from bottom to top through the PCR. This gave rise to considerable errors in measured irradiation data. Accordingly, the direction of capsule cooling airflow was reversed. This change has yielded very good results.
- The detection system for the capsule based on an optical detector system was found to be highly unreliable. This was changed to an infrared based detection system with consequent changes in the associated electronic circuits.
- Based on the experience gained during commissioning, changes in the control system were introduced to accommodate auto ejection of the capsule on certain predetermined parameters, such as high/low capsule cooling flow, low compressed air pressure, etc.

In order to make the facility available to researchers, it is necessary to qualify the facility. For this, two sets of qualification tests were carried out.

6.1. Measurement of neutron flux

For carrying out the flux measurements, accurately weighed aluminium foils of about 1 mg each were prepared and sealed in alkathene. A known quantity of gold foil was dissolved in aqua regia and diluted to a definite volume with dilute nitric acid such that an aliquot of the solution weighing about 10 mg contained about 1 g of gold. Accurately weighed quantities of the solution were evaporated to dryness on cellophane paper and sealed alkathene. Both samples were loaded in a rabbit. A number of such samples were prepared. A set of these samples was then irradiated using PCF at known reactor power for preset duration.

After irradiation, the rabbit was received in the same S/R station. The samples were retrieved from the rabbits and mounted on perspex plates and counted in an efficiency calibrated position using high purity germanium

(HPGe) detector coupled to a multichannel analyser. Similar procedures were carried out at 10, 20, 30, 40 and 50 MW reactor power. For each power level, about ten sets of samples were irradiated. Measurements of the flux values at all of the above power levels were found to be within 1% of each other.

6.2. Determination of facility sensitivity for various elements

For determining the sensitivity of the facility, concentrations of major, minor and trace elements in two reference materials were obtained. This experiment was also used to determine the limits of detection (LOD) for these reference materials. This experiment was carried out at 20 MW reactor power.

About 20–50 mg of the reference material samples were sealed in polypropylene tubes along with a known quantity of gold sample sealed separately in alkathene. These samples were then placed in the rabbits and sent to the core and irradiated for 60 s. The reactor power was maintained at 20 MW. After completion of irradiation, the samples were retrieved from the rabbits and mounted on perspex plates. These samples were then counted in an efficiency calibrated HPGe detector coupled to a multichannel analyser. Here gold was used as the standard for neutron flux determination. The elemental concentration of the reference materials was evaluated. It was found that the reported values were well within the confidence levels reported on these materials earlier by other agencies.

From the data generated during the above experiment, elemental sensitivities corresponding to a few short lived, moderately long lived and long lived isotopes were determined. These were compared with similar experiments carried out in the APSARA research reactor at BARC. The sensitivities obtained in Dhruva were found to be 10–100 times higher than those obtained in the APSARA. Even the limits of detection were found to be several orders better than those obtained in the APSARA plant.

7. GENERAL REMARKS

The Pneumatic Carrier Facility at the Dhruva Research Reactor has provided researchers in BARC with a very useful tool for carrying out neutron activation analysis on materials. The usefulness of this facility can be gauged from the fact that a sample which needs to be irradiated in the APSARA research reactor for 10 min or more for a meaningful result, needed only about 1 min irradiation in Dhruva to get the same or more accurate results. It may also be noted that being an independent, stand-alone facility operable from an exclusive laboratory, its use does not interfere in any way with other research

facilities of the research reactor. It also does not come in the way of reactor operation except that as a matter of precaution, the reactor power is lowered by 10% during capsule irradiations.

9.3. SAMPLE IRRADIATION AT THE MIDDLE FLUX REACTOR FRG-1

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1. INTRODUCTION

The GKSS Research Center Geesthacht GmbH has operated the 5 MW MTR type swimming pool research reactor FRG-1 for over 45 years. The FRG·1 has been upgraded and refurbished many times to follow the increasing demands of safe operation and the actual need for the highest neutron flux for scientific research in a competitive environment. High neutron flux with maximum availability is the permanent demand placed upon the operation of the fission neutron source by the scientists.

Thus, a first step for the increase of the neutron flux at the FRG-1 was carried out approximately 12 years ago. At that time, the FRG-1 was converted from high enriched uranium (HEU 93%) to low enriched uranium (LEU 20%) in one step. At the same time, the reactor core size was reduced from 49 to 26 fuel assemblies. Therefore, neutron flux density at the entrances of the beam tubes could be increased by more than a factor of two.

To further establish the FRG-1 as a national neutron source, another neutron flux increase was achieved by the second core compacting at the beginning of 2000, especially at the beam tubes and particularly at the place of the cold neutron source, improving its gain by a factor of two. This core compaction reduced the reactor core from 26 to 12 fuel assemblies. In this new compact core there are five irradiation positions embedded in the beryllium elements.

2. IRRADIATION FACILITIES

2.1. Out-of-core irradiation

The FRG-1's 3 × 4 compact core, which is completely encased by beryllium, possesses five out-of-core irradiation positions. Each irradiation position consists of a beryllium element with a central hole of 50 mm diameter. The arrangement of these irradiation positions near the compact core is shown in Fig. 1.

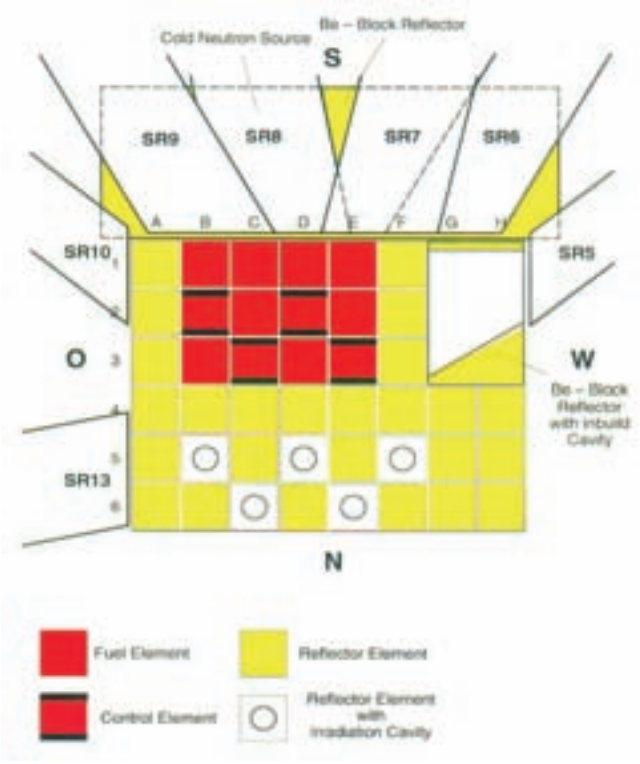

FIG. 1. FRG-1's 3 × 4 core with beryllium reflector and beam tubes.

For inserting the samples into the positions, aluminium capsules with a diameter of 15 mm, 20 mm and 45 mm, and a length up to 100 mm are available. For samples that require fast fluxes for activation, the best solution is to line the aluminium capsules with cadmium at the inside.

The thermal, epithermal and fast neutron fluxes are determined in regular intervals by means of iron and gold monitors. Typical neutron flux values at the five positions are specified in Table 1. The soft spectrum at the irradiation positions is clearly caused by their beryllium embedding.

Apart from the neutron flux, knowing the axial flux distribution and the position of the axial flux peak are of great importance for the activation. Therefore, the axial neutron flux distribution is determined in regular intervals at all irradiation positions by means of a minichamber (Fig. 2). Such axial distributions have plateaus of approximately 80 mm. At these plateaus, the samples are positioned such that the flux gradient is minimized over the sample length.

TABLE 1. NEUTRON FLUXES AT THE IRRADIATION POSITIONS IN THE REFLECTOR

Position	Neutron flux		
	Thermal	Epithermal	Fast
B5	6.8 E + 13	2.8 E + 12	4.1 E + 12
D5	8.3 E + 13	3.5 E + 12	5.4 E + 12
F5	5.4 E + 13	1.8 E + 12	2.2 E + 12
C6	4.3 E + 13	1.0 E + 12	1.3 E + 12
E6	3.8 E + 13	0.9 E + 12	0.9 E + 12

Unfortunately, the flux maximum (plateau) moves with the height of the control rods. Figure 2 shows the axial dependence of the flux peak versus the height of the control rod position for one irradiation position. Consequently, samples will be positioned in the flux peak for each operating condition.

For loading, the aluminium capsules are fastened to approximately 8 m long rods, which consist of three segments. The samples are slowly inserted into the irradiation position by hand and connected to a rotation motor. With this manual manipulation of well known samples, the reactivity influence is limited to 3 cents. For unknown samples, the reactivity influence is measured prior to

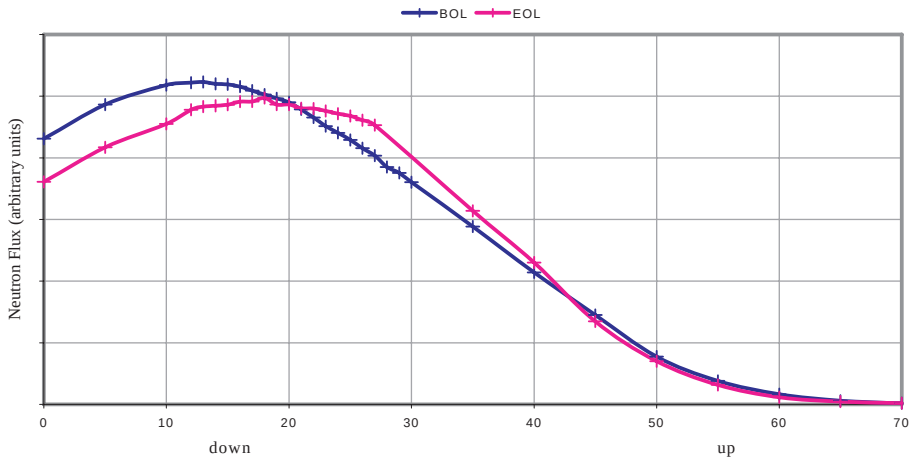

FIG. 2. Axial shaping of the neutron flux at the irradiation position B5.

the beginning of the reactor cycle. Under the conditions mentioned, the loading speed is negligible since the control rods easily compensate for such reactivity changes.

2.2. Pneumatic dispatch

The radial beam tubes of the FRG-1 supply neutron scattering experiments with high thermal and cold neutron flux. Inside the beryllium reflector and underneath the radial beam tubes is the tangential through tube SR4/12. The irradiation position of a new 'rabbit' system was installed into this beam tube. Operation of the pneumatic dispatch and the sample handling takes place from the experimental hall. For the new irradiation apparatus, the following conditions were defined:

— No reactivity influence on the reactor during irradiation execution, as the maximum reactivity step by the sample is limited to 3 cents;
— No influence of the samples on the reactor instrumentation;
— Two barriers principle against loss of pool water;
— No delivery of activated transportation gas during normal operation;
— Compact design of the facility and its shielding of the activated components;
— Fault alarms for operational failures;
— Automatic control of the facility and the data processing;
— Small activation of the components.

The basic part of the facility is the coaxial tube module made of AlMg3. Its inside transportation tube has a diameter of 36 mm. It is embedded in an outside tube of 60 mm diameter, in which the transportation gas for driving and cooling the sample container flows.

The irradiation position is at the end of the transportation tube beside the FRG-1 core. The transport of the polyethylene sample container to the irradiation position and back happens by means of a pressure difference of 70 Pa produced by a fan. For driving the container in or out, the direction of the gas flow is changed by opening and closing appropriate valves, which are actuated by a programmable logic controller (PLC) (Fig. 3). Nitrogen is used as the transport gas, whereby considerable activation is eliminated compared with that resulting from the use of air. Sensors for temperatures, humidity, pressure and activity are available within a closed line system acting as operating and fault detectors.

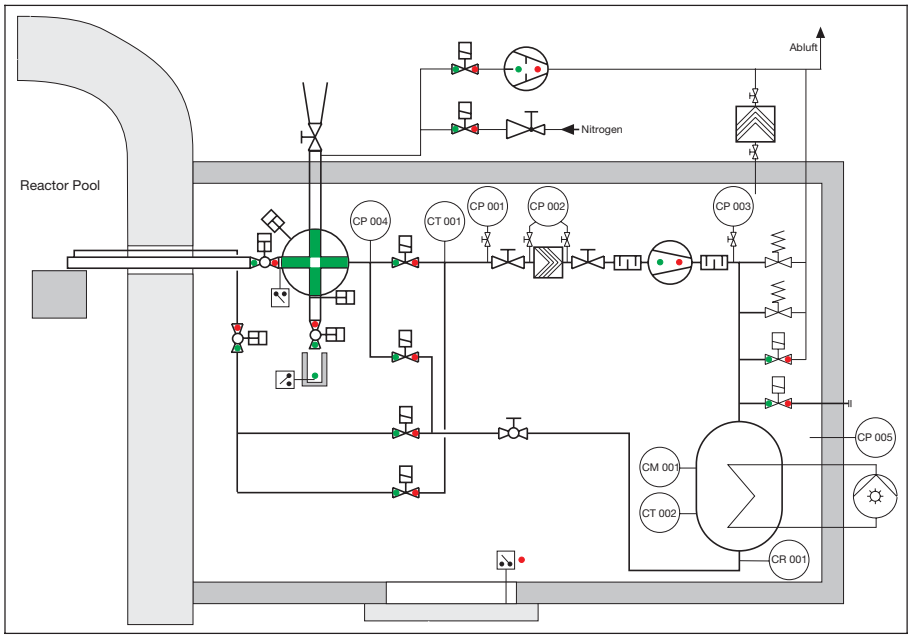

FIG. 3. Pneumatic dispatch at the FRG-1.

3. SAMPLE CONDITION AND CONDITION FOR IRRADIATION

In the reflector positions and in the pneumatic dispatch, approximately 2000 samples are irradiated per year with a total irradiation duration of 900 d for the entire set of samples in the five reflector positions. The irradiation times per sample vary from minutes to months. The samples originate from approximately 20 institutions, such as universities, industry, research centres and GKSS internal institutes. Due to the different users, very different samples are irradiated, such as ash, biological samples, cadmium, calcium, cellulose, cement caesium, cramics, chrome, cobalt, copper, diamond, germanium, gold, hafnium, holmium, indium, iodine, iron, lanthanum, krill, magnesium, manganese, minerals, molybdenum, car tyres, neodymium, nickel, niobium, phosphorus, piston rings, polyethylene, polymers, potassium, river and coastal sediment, rocks, ruthenium, sediments, silver, silicon, sodium, soil, steel, suspended matter, tantalum, teeth, Teflon, topaz, vanadium, zinc and zirconium. This long list is provided to offer some incentives for other operators to look for customers.

Such samples are examined by the GKSS irradiation service for the following characteristics:

— Humidity;
— Release of gases by thermal or radiolytic decomposition;
— Chemical attack of the sample on the capsule material;
— Reactivity feedback during insertion and irradiation.

Apart from these sample characteristics, the formal conditions important for safety are also examined and are logged in a special 'form for irradiation'. These are:

— GKSS form comprising designation, sample quantities, packaging, nuclides, half-life, activities, etc., filled out by the user;
— Examination of valid permission for handling by the user for the special nuclides;
— Examination of valid permission for transportation of the samples.

All these characteristics, as specified on the order form for irradiation, are examined and the form is signed by the user, the irradiation service department, the reactor operator and the radiation protection department for correctness. Afterwards the sample is ready to be irradiated.

4. SAFETY ISSUES

Apart from the examination described in Section 3, the following conditions generally apply for the safe execution of the irradiation:

— No irradiation of samples with considerable weight proportion of mercury or bromine.
— No irradiation of liquids and gases.
— Double packaging of the samples.
— No interaction with the reactor. The reactivity feedback must be smaller than 3 cents.
— No high pressure buildup in the capsule originating from the irradiated material.

The double packaging capsules consist of quartz (inside) and aluminium (outside). Activation as well as corrosion effects are minimized by the exclusive use of those materials.

Complementary measurements of the reactivity changes were accomplished for all irradiation positions, especially when applying samples with high interaction, such as cobalt or cadmium. Thus it is ensured that only samples with reactivity influence smaller than 3 cents are inserted.

5. SUMMARY

The GKSS has over 45 years of experience in sample irradiation. That experience and the consistent follow-up led to many technical and administrative improvements. The measures derived serve to avoid disturbances of the reactor operation. Further, the entire handling of the samples and their irradiation do not have any influence on the neutron flux at the beam tubes. This is of great importance, since the main interest at the FRG-1 is in fact neutron scattering experiments.

9.4. INSTRUMENTAL NEUTRON ACTIVATION ANALYSIS AT THE US GEOLOGICAL SURVEY

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1. INTRODUCTION

The United States Geological Survey (USGS) Triga Reactor (GSTR) has been in operation since 1969 and is maintained by the USGS to perform nuclear analyses of geological and biological samples. Controlled irradiation of samples by neutrons permit multi-element analyses by instrumental neutron activation analysis (INAA), Ar–Ar age dating, and fission track radiography and dating. More than 420 000 sample irradiations have been performed since the facility began operation. The analyses provide valuable information about both the composition and age of the samples. The design of the GSTR is well suited for INAA, fission track radiography and age dating, in that the slow rotation of samples around the reactor core results in samples being subjected to a homogeneous irradiation field, allowing the use of a single fluence monitor for an entire batch of samples being irradiated. The central irradiation facility of the GSTR is used for Ar–Ar age dating because of the high neutron flux available there.

Between 1970 and 1995, INAA at the USGS was primarily applied to the analysis of geological samples, mostly in the form of rocks and soils. Major, minor and trace element data obtained by INAA were used to determine the petrogenetic origin, economic significance or geochemical variability of the samples. Between the two laboratories located in Denver, Colorado and Reston, Virginia, in the United States of America, about 7500 samples per year were analysed by INAA during this period. INAA was the primary source of this type of data to USGS scientists until design improvements and moderating costs of inductively coupled plasma mass spectrometry (ICP-MS) instrumentation during the late 1990s, coupled with advances in decomposition techniques made ICP-MS an attractive alternative to INAA.

Since 1995, the USGS mission has increasingly focused its efforts on two areas:

— Assessing the quantity, quality and availability of natural resources;
— Understanding the physical, chemical and biological processes at work in natural systems. These applications, however, are not new to INAA and

the USGS laboratory adapted quickly to the analysis of environmental samples. As discussed in more detail below, INAA continues to be an indispensable tool in the study of the environment, despite competition from other analytical techniques.

There are several inherent advantages of INAA over other analytical techniques [1]. In addition to being a multi-element technique, a major benefit of INAA is that it is non-destructive. Irradiation of the original, as received material eliminates problems with incomplete dissolution that are commonly associated with other techniques. This is particularly crucial for some matrices where complete dissolution is difficult to achieve. Materials analysed by INAA remain available for treatment by other analytical methods or can be archived for posterity, which is important for precious and hard won samples, such as lunar rocks, meteorites and mineral separates. Another advantage of the technique is the linear relationship between count rate and concentration that typically holds from detection limit abundances to extremely high concentrations. Finally, with very few exceptions, large particle sizes in the sample do not affect the accuracy of INAA determinations.

2. DESCRIPTION OF REACTOR FACILITIES AND OPERATIONS

The GSTR is a low enriched uranium fuelled open pool type research reactor. The reactor operates in steady state mode at power levels up to 1 MW and it can be pulsed to about 1100 MW peak power. The traditional irradiation location for INAA samples is the rotary specimen rack (RSR), a 40 position, air filled, circular rack that is embedded in the graphite reflector that surrounds the outer edge of the core (Fig. 1). The RSR has a pneumatic capability that allows irradiated samples to be unloaded quickly and remotely, which reduces the radiation dose to facility staff (Figs 2 and 3 for system details). The RSR receives a thermal neutron flux of ~2×10^{12} n·cm^{-2}·s^{-1} at 1 MW and the rack is normally rotating during sample irradiation. The GSTR also has the capability to incorporate a 4.4 in diameter central irradiation cavity and two off-centre cavities of 2.4 in diameter.[1] Both 6 in and 8 in diameter external beam tubes are also available for experiments involving larger targets. Numerous 1.5 in diameter experiment positions are available, both in-core and out-of-core. The peak thermal neutron flux at full power is ~3×10^{13} n·cm^{-2}·s^{-1} in the centre of the core while the flux at the outer wall of the reflector is ~7.5×10^{11} n·cm^{-2}·s^{-1}.

[1] 1 inch (in) = 25.4 mm

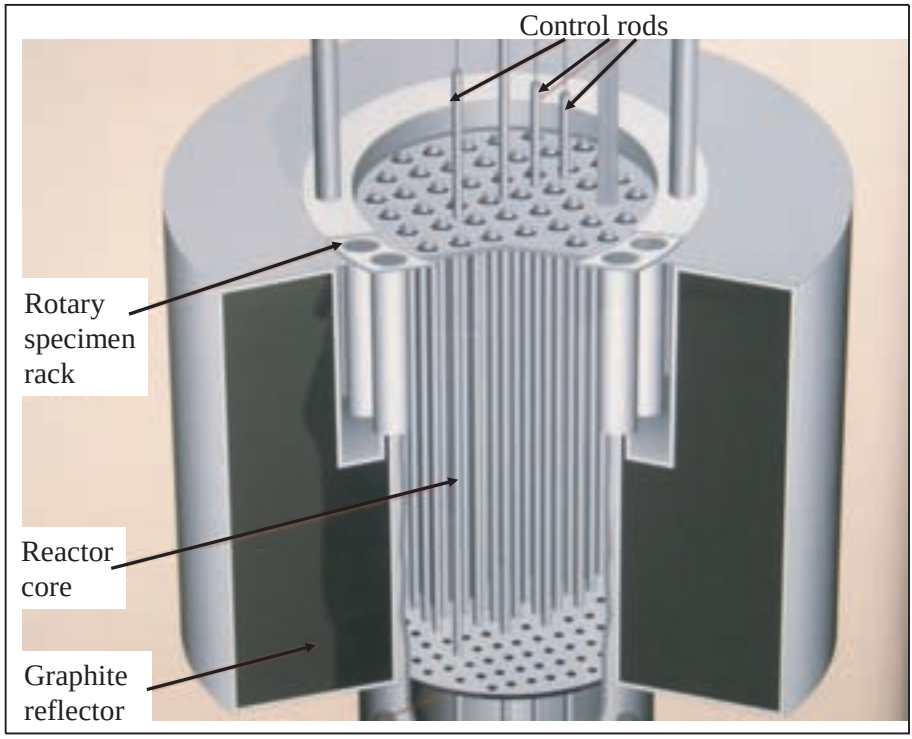

FIG. 1. GSTR core assembly.

3. CONSTRAINTS AND RESTRICTIONS ON REACTOR OPERATION

Pneumatic systems at the GSTR use atmospheric air, as the moving fluid and trace quantities of argon gas in air will readily activate to produce significant quantities of Ar-41 (half-life = 1.82 h). The GSTR staff restricts operation of the pneumatic systems to limit Ar-41 production and release to levels well below the regulatory limits.

The flexibility of the GSTR (Triga) design allows for the simultaneous irradiation of INAA samples and irradiation of specimens for isotope production, fission track studies, etc. Since the GSTR operates on a one shift per day basis, the overnight shutdown is used to allow the irradiated samples to decay in place prior to unloading from the reactor the next morning. The overnight shutdown also allows the Ar-41 generated in air filled facilities to decay. A drawback to the one shift per day schedule, however, is that 40 h

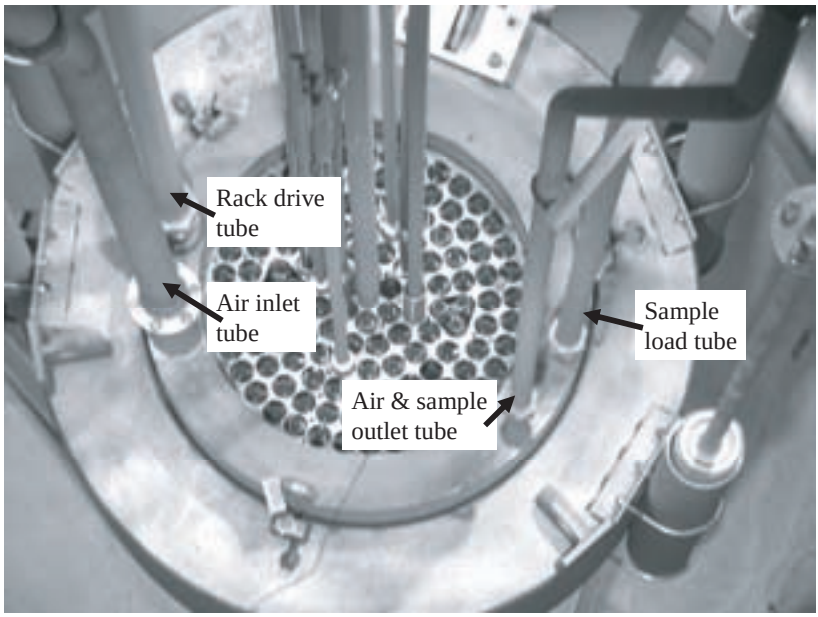

FIG. 2. GSTR rotary specimen rack details.

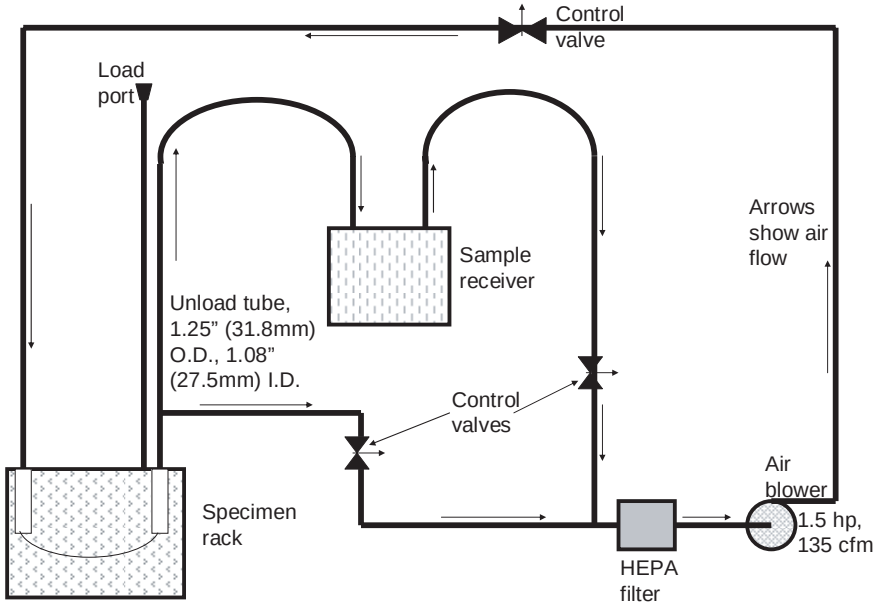

FIG. 3. GSTR pneumatic sample unloading system.

irradiations required for some Ar–Ar dating in the central thimble take several days to complete.

4. SOME RECENT INAA APPLICATIONS AND RESULTS

Despite the current focus of the USGS on environmental issues, INAA continues to provide important data for the understanding of geological processes. For example, in a 2002 collaborative study between the USGS and the Colorado Geological Survey [2], over 200 rock samples were collected from discontinuously exposed basaltic flows in an area of tectonic collapse in central Colorado, USA. These samples were analysed by INAA and other analytical techniques, including Ar–Ar dating which also utilized the USGS Triga Reactor. The determined geochemical compositions and ages were used to map the present extent of the individual lava flows, and thus determine the timing and extent of each collapse event. In particular, high precision determinations of Hf and Ta abundances obtained by INAA were extremely beneficial in correlating the lava flows. High precision INAA data also showed distinct differences in the concentrations of several elements (e.g. La, Yb, Th and U) within basalt suites that had very similar major element compositions and ages.

INAA is also being used to analyse a wider variety of environmental samples. Dust samples taken from various rural locations in southern California and Nevada near Owens (dry) Lake, California, contain elevated levels of the potentially toxic elements As, Sb, Th and U compared with soil and sediment samples [3]. Based on a comparison of dust compositions and potential local sources, four primary sources of the dusts have been identified: alluvial sediments, natural carbonate rich playas, the human induced Owens (dry) Lake playa and anthropogenic and/or volcanic emissions. Concentrations of As and Sb in dust samples decrease with increasing distance from the lake (Figs 4 and 5). Hair samples collected from individuals living or working in the vicinity of the Owens (dry) Lake dust are also being analysed by INAA. These data may provide a record of long term exposure to airborne dust from Owens (dry) Lake.

INAA is particularly suited for the measurement of parts per billion concentrations of iridium in geological materials. Elevated levels of Ir in 65 Ma[2] sediment samples are commonly accepted to record the fallout from a meteorite impact, which defines the Cretaceous Tertiary (K/T) boundary. This

[2] Ma refers to Mega annum (million years).

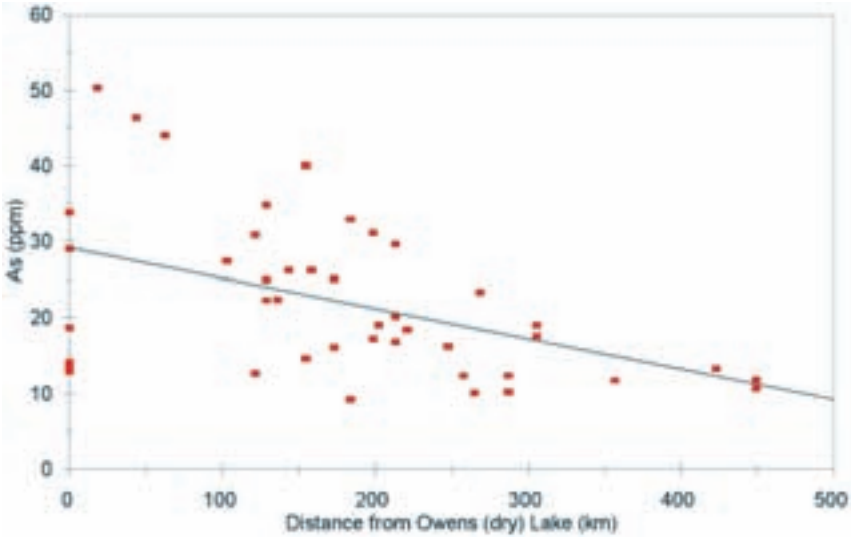

FIG. 4. As concentrations in dust samples relative to the distance from Owens (dry) Lake, California (the line shown has a regression coefficient of 0.19, but increases to 0.53 without 0 km samples).

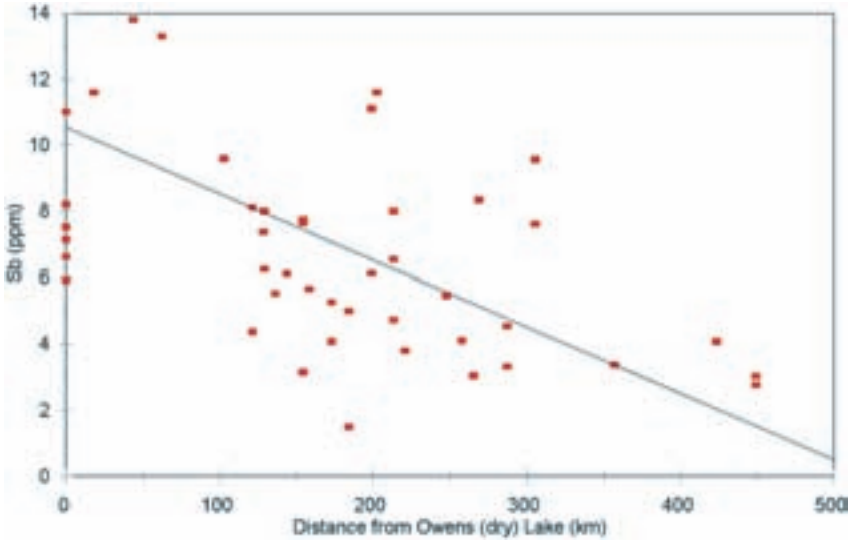

FIG. 5. Sb concentrations in dust samples relative to the distance from Owens (dry) Lake, California (the line shown has a regression coefficient of 0.26, but increases to 0.41 without 0 km samples).

impact is thought to be responsible for the extinction of thousands of species of plants and animals including the dinosaurs. Ir data, for some samples deposited at or near the K/T boundary, are listed in Table 1. The two samples with Ir contents greater than 1 ppb are presumed to be from this layer whereas the samples with Ir contents less than 1 ppb are adjacent to the layer.

There are few techniques other than INAA that can produce high precision data for Br in geological and biological materials. The US Environmental Protection Agency (EPA) requested the USGS to determine the concentration of Br in soils that may have been contaminated by road salt and automobile exhaust emissions. Br concentrations appear to be correlated with Cd (Fig. 6). The abundances of As and Sb were also determined by INAA. The plot of Fig. 7 may indicate variable contamination with a source rich in As and Sb.

This same source seems to be rich in Cd as well (Fig. 6). The high Br concentrations are likely to be from road salt applied to roads with high vehicle traffic and related to high exhaust emissions. The significance of these results remains under investigation by the EPA.

In the past few years, there has been an increased interest in INAA of biological samples related to ecological studies within the USGS. USGS scientists working in Yellowstone National Park in collaboration with Montana State University and the National Park Service found that elk in an area of abundant hydrothermal activity, the Madison-Firehole (MF) area, had a higher mortality rate than elk in the Northern Range (NR) area. Teeth extracted from elk that died by starvation (MF herd) and from those killed by predators (NR herd) were analysed by INAA to determine if any elemental concentration could be correlated with suspected dietary deficiencies and environmental stresses affecting the MF elk. The elk teeth from the MF subset contain lower Sr but higher As and Br abundances than the NR elk teeth (Figs 8 and 9). Although Sr^{+2}, AsO_4^{-3} and Br^{-} are known to substitute in the fluorapatite structure of teeth ($Ca_{10}(PO_4)_6F_2$), these preliminary results suggest elevated As and Br contents may indicate an environment that is detrimental to elk health.

TABLE 1. Ir ABUNDANCES IN SAMPLES COLLECTED AT OR NEAR THE CRETACEOUS TERTIARY BOUNDARY LAYER

Sample	Suspected impact layer	Ir (ppb)
Dawson-1	Yes	1.9
Dawson-2	No (3 cm from layer)	0.95
Sugarite-1	No	0.6
Sugarite-2	Yes	6.9
Sugarite-3	No	0.7

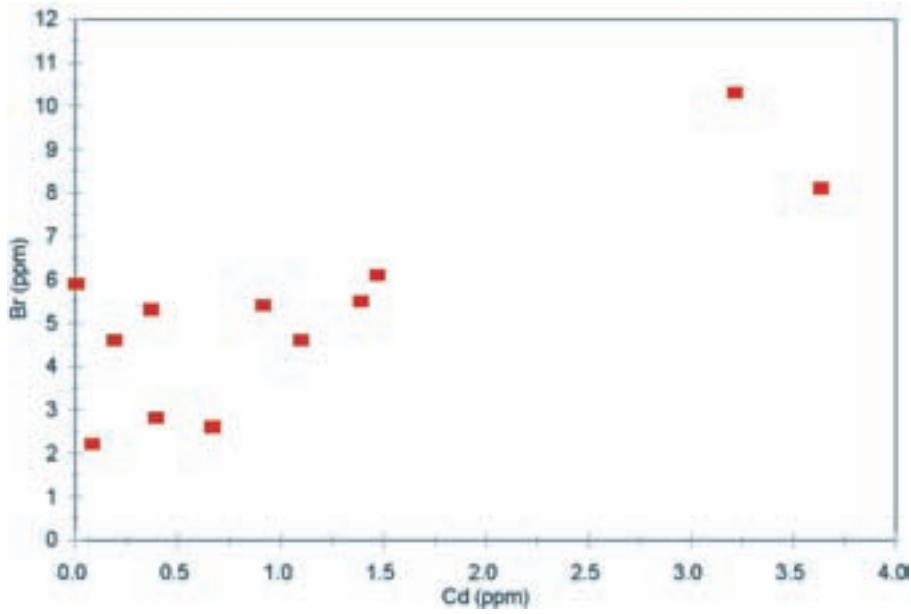

FIG. 6. Br versus Cd concentrations in contaminated soils.

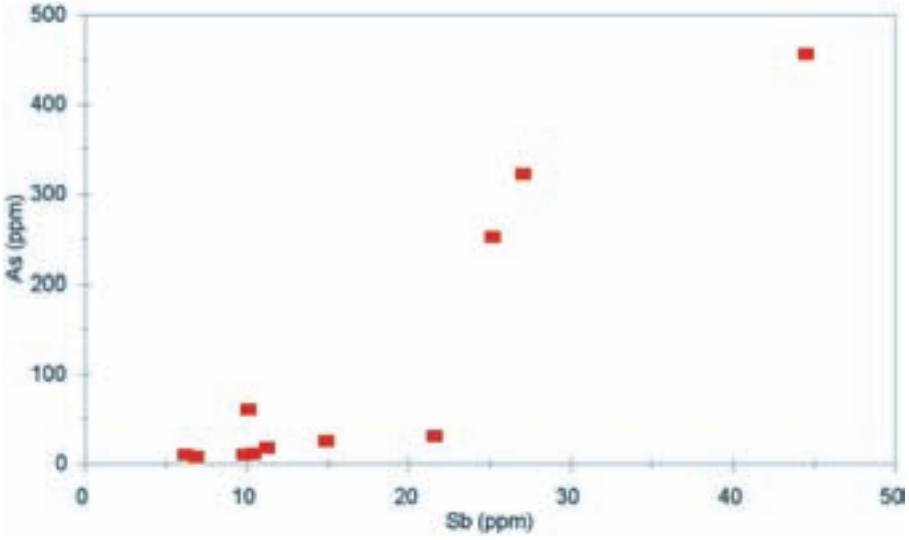

FIG. 7. As versus Sb concentrations in contaminated soils.

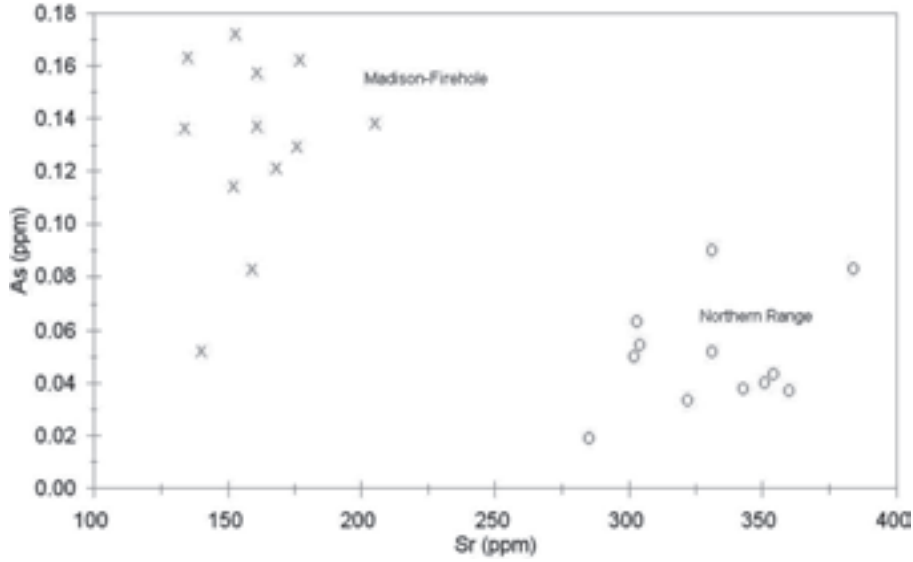

FIG. 8. As versus Sr concentration in elk teeth from Yellowstone Park (samples from Madison-Firehole denoted with an ×; samples from Northern Range with an ○).

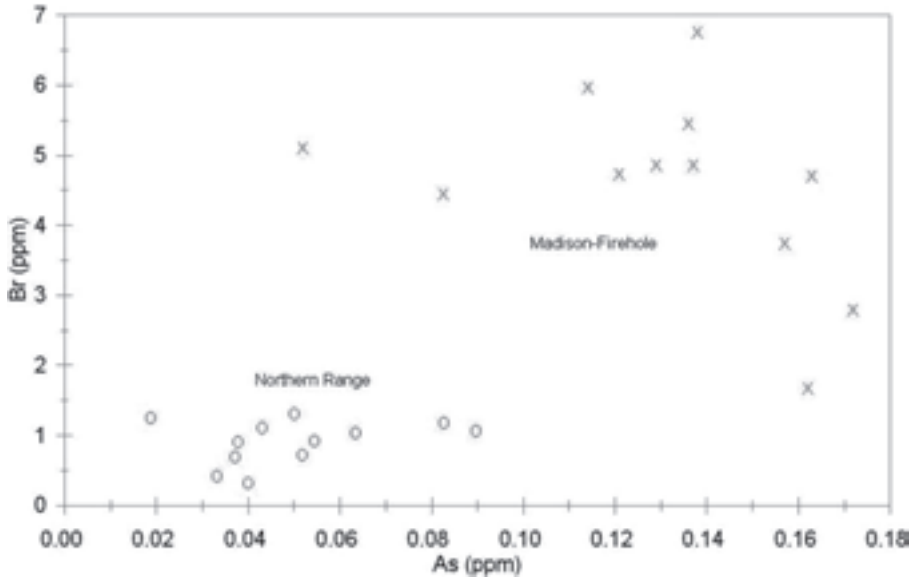

FIG. 9. Br versus As concentration in elk teeth from Yellowstone Park. Samples from Madison-Firehole denoted with an ×; samples from Northern Range with an ○.

5. CONTINUED IMPORTANCE OF THE GSTR

Over the past ten years, there have been several contributing factors leading to the decline in nuclear reactor usage at the USGS. For example, the popularity of competing techniques, such as ICP-MS, the concept that operating nuclear reactors is overly expensive and imposition of user fees that discourage full utilization of the reactor have significantly reduced operations. Nevertheless, the GSTR remains an important facility for irradiating samples for INAA, Ar–Ar dating and fission track radiography and dating. A small but important group of USGS scientists still recognize the benefit of the GSTR and their continued strong support will probably keep the reactor operational for the foreseeable future.

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9.5. NEUTRON ACTIVATION ANALYSIS AT THE McMASTER RESEARCH REACTOR

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1. INTRODUCTION

Since its first criticality in 1959, the McMaster Nuclear Reactor (MNR) has been the most powerful and versatile university reactor in Canada. The MNR facility has been designed for a wide variety of applied and basic nuclear research utilization. The moderately high neutron fluxes available in and near the reactor core make MNR invaluable for very sensitive neutron activation analysis (NAA) by academic, commercial and industrial users in a wide variety of fields.

Following a brief background on MNR, this paper discusses both organizational and hardware facilities available to users, with insights into their design and evolution. Considerations for both safe and successful use are discussed, always with the user in mind.

2. BACKGROUND

2.1. MNR

MNR was the first university based research reactor in the Commonwealth. As a pool type reactor, its design included a flexibility that has repeatedly proved itself as facilities have been upgraded and changed over the years. While originally designed for 5 MW, it operated mainly at 2 MW until the mid-1970s when additional cooling capacity was installed to allow continuous 5 MW operation. After several years at 5 MW, financial considerations forced a return to 2 MW operation on a cost recovery demand basis. The 5 MW capability has been carefully maintained in facility development. Hours of operation have been varied from one shift a day to full 24 h, 365 d/a operation. Currently, MNR operates weekdays from 8 am to 12 midnight at a thermal power of 2 MW.

The convenient location of MNR, in southern Ontario, Canada, near major transportation routes, provides ready access or delivery to Canadian and international users.

2.2. Evolution of NAA at MNR

From its earliest days, MNR supported the use of both instrumental and radiochemical NAA, with the availability of radiochemists and laboratory facilities in the neighbouring nuclear research building (NRB). The two original rabbit systems included delivery locations within the reactor building and in several NRB laboratories.

In 1978, MNR established a commercial partnership to develop NAA on a larger scale. It was a boom time for geochemical prospecting and NAA established itself as a method of choice for what might be termed biogeochemical prospecting for gold. The presence of gold in vegetation was positively correlated with low grade ore deposits. Delayed neutron counting (DNC) was also effectively used to analyse large volumes of samples for U content, to aid uranium prospecting. Similarly, the need to monitor workers in the uranium industry led to widespread use of DNC for uranium bioassays until more convenient and economical methods were developed. This expansion of activity led to the design and installation of new irradiation and counting facilities, including the current rabbit systems. Although the basic hardware has not changed, the software and instrumentation associated with these systems have been upgraded several times as better technology became available.

During the 1980s, it was recognized that professional expertise was essential for NAA to be successfully utilized, to assist the users and to maintain the NAA laboratories as state of the art facilities. This resulted in the formation of the Centre for Neutron Activation Analysis (CNAA), a unit within the MNR organization.

2.3. McMaster Institute of Applied Radiation Sciences

Until 2000, MNR operated as an autonomous unit within McMaster University, available on an equal basis to users, both within and outside the University. Users have included many other universities, government departments and industries in and outside of Canada. The high flux available at MNR for short term irradiations has attracted much of this interest. In 2000, the McMaster Institute of Applied Radiation Sciences (McIARS) was founded as an umbrella organization overseeing all radiation research facilities at the university, including MNR, the accelerators and the nuclear research building. Its mission was to use the unique combination of knowledge, skills, equipment, facilities and people available at McMaster to build and enhance the research excellence, capabilities and core competencies that exist both within the Institute, across Canada and around the world. This was a major step forward and represented a significant commitment by McMaster.

3. CENTRE FOR NEUTRON ACTIVATION ANALYSIS

The CNAA is a unique facility among Canadian universities, providing on-site, fully equipped laboratories and counting facilities, and the necessary professional assistance to utilize them for NAA. These facilities are made available on a rental basis for the use of those researchers who wish to employ NAA but do not have access to the necessary equipment. In many cases, NAA is peripheral to the main research topic and the cost of a full set of analytical equipment cannot be justified. Counting facilities include various gamma counting systems (detectors, associated electronics and multichannel analysers), computer assisted data reduction with varying degrees of sophistication, and technical support ranging from project assessment to experimental design and execution. In addition, there is ready access to the MNR irradiation facilities described below.

Teaching and training are a primary focus of the Centre. Whether an expert or a novice, each user is provided with on-site orientation in the laboratory, including recommended analytical practice, familiarization with counting/data handling equipment, and health and safety instruction. Thereafter, they are supervised and assisted as necessary.

CNAA is user friendly, with easy to use automatic irradiation and counting systems. The benefits of this policy are a more rapid orientation of users and a higher level of user productivity and independence. The latter is of particular importance since the Centre wishes its clients to view the facility as their own specialized laboratory, which happens to be located near the MNR.

The CNAA serves a wide variety of clients in industrial, academic and government sectors. While the Centre mainly operates as a user facility, with interested parties providing their own personnel to perform the analytical work, it can also be utilized on a contract or long term development project basis, with full technical support provided by CNAA personnel.

Al, Ca, Cl, Dy, I, In, K, Mg, Mn, Na, Rh, Se, Ti and V are among short lived elements commonly assayed. Prompt gamma analysis of rare earths is used in the study of geological materials. In vivo cadmium measurements have been used to characterize the extent of long term exposure to this element at smelters. In a number of studies, lead measurements have demonstrated that bone lead reflects cumulative exposure and, as stored lead is released, it becomes a source of endogenous exposure.

4. IRRADIATION FACILITIES

4.1. General

As mentioned already, the MNR is a pool type research reactor. An important advantage of this reactor type is the ease with which irradiation and other facilities can be added or modified. Table 1 summarizes the current available facilities, all of which have been planned with large volume use in mind. The location of these facilities is illustrated in Fig. 1.

A variety of sample sizes can be accommodated in the available facilities. In some cases, users irradiate batches of samples in this way and then assess the activity either in the CNAA or at their own facilities, remote from the reactor site. Alternatively, the larger irradiation sites can be used for semibulk samples. For analysis of short lived activity there are several rabbit systems available. A prompt gamma facility is available which allows for the analysis of elements not detectable by conventional NAA techniques.

All irradiation facilities have thermal and epithermal capability with the exception of the dry tube which is epithermal only. In addition to the four rabbit irradiation positions adjacent to the core shown in Fig. 1, there are two in-core high flux rabbit positions which may be used.

TABLE 1. DESCRIPTION OF THE IRRADIATION FACILITIES AT MNR

Site	Size (mm)		No. of sites	Flux[a] ($n \cdot cm^{-2} \cdot s^{-1}$)
	L	D		
Capsules	40	18	8	Up to 4.5×10^{13}
RIFLS[b]	750	60	2	5×10^{12}
LVR[c]	750	125	1	3×10^{11}
Dry tube	750	60	1	1×10^{12}
Rabbits	50	13	4	3×10^{12}

[a] Fluxes at 2 MW operation.
[b] Reactor irradiation facility for large samples.
[c] Large volume RIFLS.

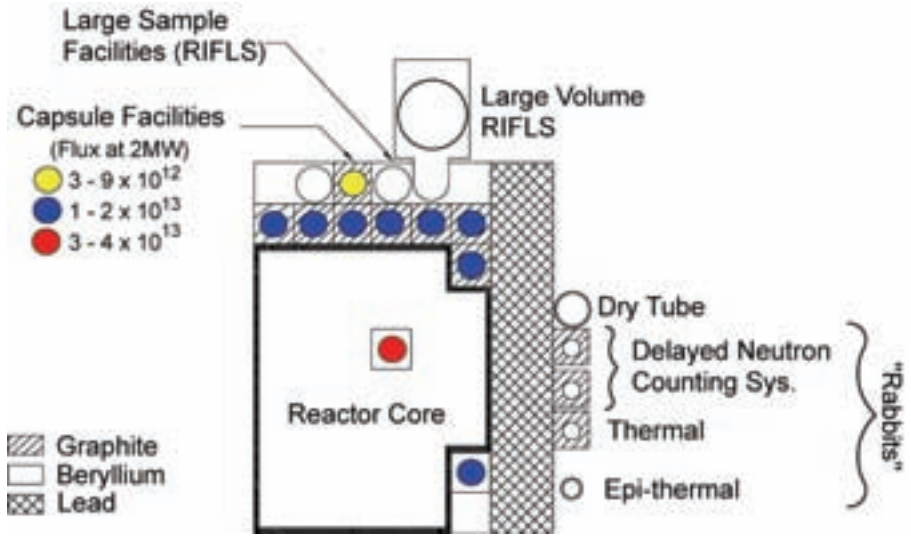

FIG. 1. MNR irradiation facilities.

4.2. In-core capsules

Single or small samples can be placed in cold weld sealed aluminium capsules which are irradiated in a graphite or beryllium reflector site or in the in-core flux trap, as shown in Fig. 1. The capsules accept samples up to 80 mm long and 18 mm in diameter. Up to five capsules can be simultaneously irradiated in a single site with good flux uniformity (±10%). A cadmium liner can be installed for epithermal irradiations. Storage and handling equipment is provided to allow short lived activities to decay before delivery to the user for analysis. Shielding containers are available to permit shipping in type A containers.

4.3. Reactor irradiation facility for large samples (RIFLS)

Larger samples can be handled in a RIFLS tube. These tubes are sealed aluminium chambers that sit in guide tubes beside the core graphite reflector, as shown in Fig. 1. Depending on the sample container, up to 100 samples can be irradiated at once. Objects or assemblies of samples up to 55 mm in diameter and up to 750 mm in length can be accommodated. RIFLS tubes are available for thermal and epithermal neutron irradiations. They are rotated during irradiation to minimize radial flux variations across the sample.

Following irradiation, these tubes are normally stored to allow short lived activity to decay before unloading. The unloaded samples may be placed in in-pool storage facilities where they decay until they can be counted.

4.4. Large volume RIFLS (LVR)

The large volume RIFLS are similar to RIFLS tubes but are larger and can accommodate samples up to 120 mm in diameter. The normal irradiation position for these is further away from the core, as shown in Fig. 1. Similarly to RIFLS, in-pool storage facilities are available for sample decay before counting.

4.5. Dry tube facility

The dry tube is an aluminium pipe which extends from the pool surface to the west side of the core. All sample handling is in air; samples are lowered into a cadmium jacketed irradiation end. Irradiation samples are then removed to the attached storage box. The facility will accept samples up to 55 mm in diameter and up to 300 mm in length. The dry tube is shielded from core gammas by a 120 mm lead block, which significantly reduces sample heating.

4.6. Short lived neutron activation analysis (SLNAA)

The greatest use of the CNAA has been for short lived analyses. The high flux available at MNR for short term irradiations has prompted much of this interest, particularly for the analysis of Al, Ca, Cl, Dy, I, In, K, Mg, Mn, Na, Rh, Se, Ti and V.

Each irradiation terminal used for SLNAA is accessed via a computer controlled pneumatic transport system which was designed and built in-house. As shown in Fig. 2, the operating station contains the automated sample insertion hopper and computer interface. The system allows the user to choose between thermal and epithermal irradiation terminals. The rabbit system has been integrated with the gamma counting facilities to include automatic sample positioning, interfaced to a PC based, multichannel analyser for data collection, as shown in Fig. 3. Since several irradiation terminals along with associated gamma counting equipment are available for use, a number of users can be accommodated simultaneously. Currently, the transit time from the irradiation site to the counters in the NRB CNAA laboratory is about 6 s.

This system may also be used for the irradiation of samples for longer lived activity whereby the samples are simply routed to a storage container following irradiation for later counting. It may also be used as an automated counting system.

FIG. 2. SLNAA operating station.

4.7. Delayed neutron counting (DNC)

DNC is a sensitive, selective and rapid technique for determining uranium at ppm levels or lower. The McMaster DNC system (Fig. 4) is a fully automated, computer controlled pneumatic transfer irradiation, analysis and data logging system. The system was designed for efficient throughput of large numbers of samples. Analysis of a batch of samples proceeds unattended with the automatic storage of results.

This system uses the same sample carriers as the SLNAA system and was custom designed for optimum counting efficiency. Because of the need for short transit times from irradiation terminal to counter, and also because of the high radiation fields from irradiated DNC samples, the system was installed in the reactor building in an area where the operating console is shielded from the counter. Following the count, the sample is routed to remote shielded containers where the samples can decay. One container is for saving samples for retrieval and another for samples destined for disposal. There are two terminals that can be manually switched into the system, one thermal and one epithermal.

FIG. 3. SLNAA counting terminal and detector.

4.8. Prompt gamma neutron activation analysis (PGNAA)

Prompt gamma neutron activation analysis (PGNAA), as shown in Fig. 5, is available at one of the reactor beam ports. As shown in Fig. 5, a rabbit system delivers samples to an irradiation terminal which allows a well shielded, high efficiency detector to view capture gammas emitted by samples upon neutron absorption. The technique is of particular use in the determination of boron, cadmium, europium, gadolinium and samarium. Boron is readily analysed in concentrations from a few per cent to the ppm level. As the technique is completely instrumental, losses of volatile boron compounds are avoided.

For valuable samples, the non-destructive nature of PGNAA can be fully exploited. The neutron flux during a typical PGNAA determination is several orders of magnitude lower than for conventional NAA. Consequently, both residual activity and potential radiation damage to delicate samples is dramatically reduced.

The rabbit system used for PGNAA is smaller than those used for standard NAA (12.7 mm versus 19 mm diameter). There are two irradiation

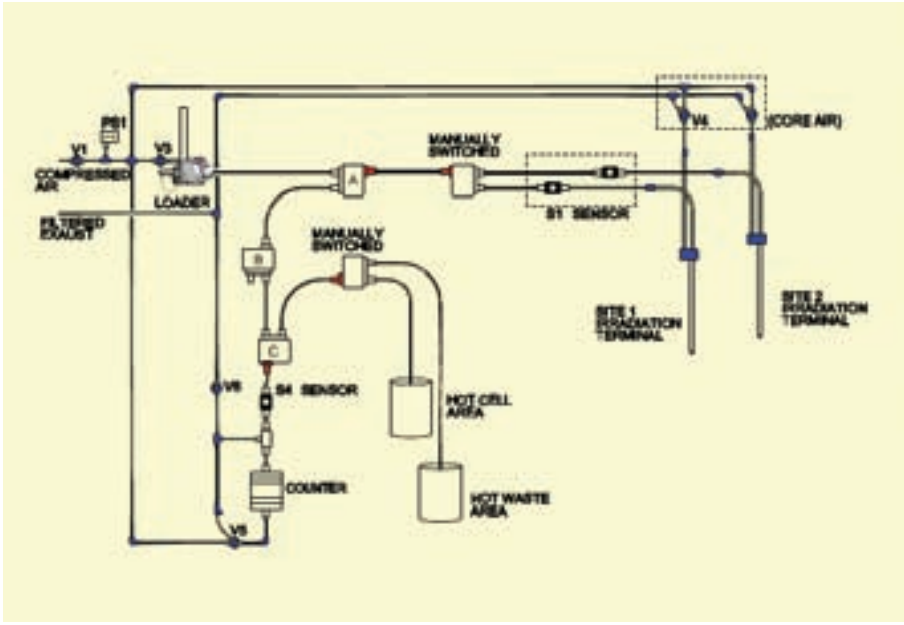

FIG. 4. The DNC system.

terminals available for use. One is designed to have a very low boron background for low level boron determinations. The other is more robust and used where low boron detection levels are not needed.

4.9. Facility design considerations

4.9.1. Rabbit systems

The 19 mm rabbit systems are fully automated to both avoid exposures (for DNC) and to maintain cost efficiency. Each system provides a choice of either thermal or epithermal irradiation terminals. The design is fail-safe in the case of loss of power, control system or pressurized air. The majority of the transfer lines of the systems are standard heavy duty polyethylene tubing, connected by Swagelock fittings that have been machined out to the proper diameter. The in-pool portion of the irradiation terminals are aluminium tubing. The polyethylene lines connecting to the terminals are shrouded in flexible metal conduits for protection.

Previous methods for epithermal irradiations involved placing the sample in a cadmium lined rabbit. Not only did the cadmium take up room in the

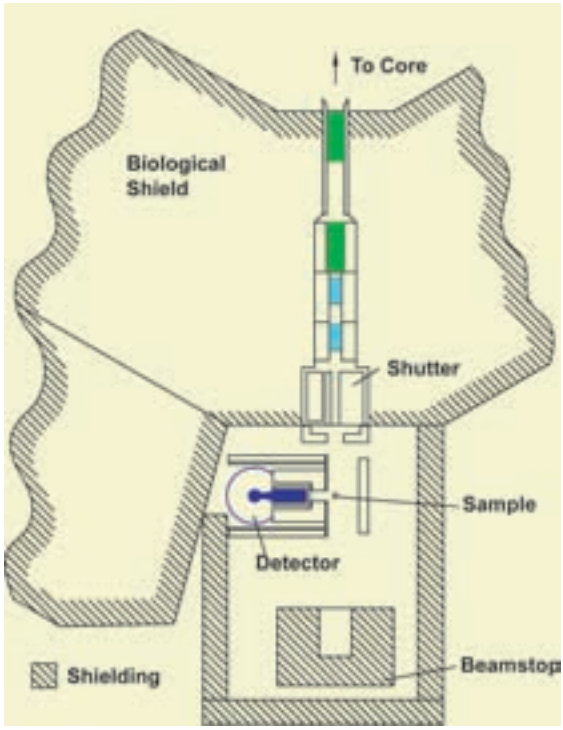

FIG. 5. PGNAA system at MNR.

rabbit but, when delivered, it was extremely active, requiring shielded remote handling. To avoid this, terminals were made that include cadmium shielding in their design. The only area not shielded is the entry point for the capsule. Since the terminals extend above the top of the core, and the samples are located well down in the terminal, this does not present any drawback.

Earlier experience with polyethylene capsules in the original MNR rabbit systems indicated that rabbits in terminals right near the core suffered heavily from gamma radiation damage, embrittlement and gamma heating. For the new rabbit terminals, a lower gamma background location was desired. Fortunately, such a location is available on the west side of the core where the former thermal column had been located. In order to protect rubber seals at the pool wall, the 120 mm thick lead slab which was used to reduce the gamma background in the thermal column was left in place adjacent to the core. It had been designed for 5 MW operation and provided an ideal location for the rabbit terminals, since not only is there a low gamma field (relative to the

available neutron flux), there is also a physical separation from the core which eliminates any reactivity concerns.

The frame which supports the lead shield also provides a suitable place for mounting the rabbit terminals and their graphite reflector blocks. The frame allows for easy removal and replacement of a terminal should it be necessary to do so.

The systems are designed to operate from the University's pressurized air system, which provides clean, oil free air at ~100 psi. This pressure is reduced for both rabbit transfer and system switching, and can be adjusted to control sample transfer rates. In the event of power or air pressure failure, the system goes to a fail-safe state that allows samples in the core to be ejected by an emergency air reservoir resident in each system.

Rabbit air exhaust is via HEPA filters in the reactor building exhaust system. Initially, there was some concern that ^{41}Ar created would present a problem but this turned out not to be significant. During normal rabbit operation, the air is not resident in the terminal for sufficient time to build up any significant amount of activity.

4.9.2. Rabbit vial design

When the first 19 mm rabbit systems were put into use, commercially available polyethylene vials with connected tops were used. The tops were popped into place and the connecting thread cut off before heat sealing the vials. Problems quickly became apparent with this arrangement, as the vial seal was at the point of maximum wear (the outside corner) as the rabbit travelled through the system. This led to rabbits in the DNC system (with the fastest transfer rate) sometimes opening and contaminating the system. A custom designed vial with a top that placed the seal away from the outside edge was developed and fabricated. One advantage of this arrangement was that the polyethylene used for the vials was first sent to MNR where it was tested for impurities. As a result, only polyethylene with low trace element content went into the capsules. These capsules were also supplied to other users of 19 mm systems. A smaller version of these vials was also manufactured for the PGNAA system. These vials, both opened and closed, are shown in Fig. 6.

4.9.3. RIFLS and LVR tubes

The need for RIFLS came from the desire to irradiate large batches of samples at one time. The RIFLS and LVR tubes were designed to be made from readily available tube stock, and to be accommodated in standard grid sites of the core. They are designed for ease of handling and to maintain the

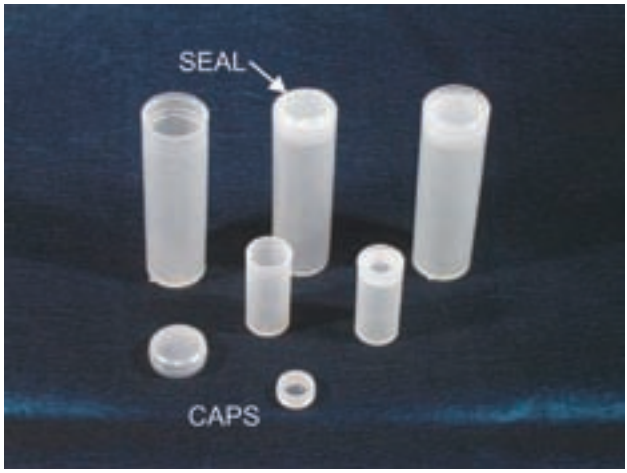

FIG. 6. Custom vials showing caps, bodies and closed vials, both sealed and unsealed.

integrity of their seals in the radiation environment. Rotating suspension systems are used to minimize radial flux variations. In-pool sample storage following irradiation is also available. Flux monitors included with samples allow for flux variation corrections during analysis.

4.10. Future developments

An additional short lived activation system is contemplated which will expand capacity and increase availability. Upgrades to the prompt gamma facility are planned to improve detection limits by reducing the background and increasing the sensitivity by increasing sample size.

5. SUMMARY

MNR has provided dedicated facilities for academic and commercial NAA since 1978. The key to its effectiveness has been a user oriented philosophy with many facilities growing out of ideas sparked by users. The key to the success of the CNAA, however, is the availability of a dedicated expert to assist the users. This is the most important feature of NAA usage at MNR.

MNR operations staff provides expertise to design, build and install equipment; and the CNAA staff provides the technical assistance to follow through from selection of technique to final report. This gives MNR the

advantages of flexibility and fast turnaround, which are passed on to customers as low cost, high quality services.

NAA services at MNR have been a valuable resource for McMaster and other universities, as well as for the Canadian nuclear research community. MNR's research and commercial facilities have continually been upgraded over the years to improve existing capabilities and expand the range of services which can be provided. Special facilities are continually being developed to accommodate new research requirements.

9.6. NEUTRON ACTIVATION ANALYSIS AT THE TRIGA MARK II RESEARCH REACTOR OF THE UNIVERSITY OF MAINZ

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1. INTRODUCTION

Research reactors of the Triga type [1] are light water cooled reactors using fuel moderator elements composed of an alloy of uranium-zirconium-hydride (UZrH) with 20% enrichment in ^{235}U [2, 3]. In the steady state mode, the Triga Mainz research reactor can be operated at power levels ranging from about 100 mW(th) up to 100 kW(th), depending on the requirements of the different experiments. Pulse mode operation is also possible, corresponding to a maximum pulse peak power of up to 250 MW(th) [4, 5], a neutron flux in the order of 10^{15} n/cm^2 per pulse and a pulse width (full width at half maximum (FWHM)) of about 30 ms. Here, the large prompt negative temperature coefficient of the Triga reactor — an inherent characteristic of the fuel moderator elements — reduces the power of the reactor within a few thousandths of a second, faster than any engineered device can operate. Pulse mode operation is especially advantageous for the production of very short lived nuclides with half-lives below 1 min. The reactor is used for neutron activation analysis, as well as for basic research in nuclear chemistry and physics, applied science and educational purposes. The average operation time of the Triga Mainz is about 200 d/a. About 90% of this time is used for reactor operation at the nominal power of 100 kW(th), and the rest for pulses and steady state operation below 100 kW(th) [6].

2. IRRADIATION FACILITIES AND PULSE MODE OPERATION

2.1. Irradiation facilities

Figure 1 shows a vertical cross-section view of the Triga Mainz indicating the different irradiation positions for neutron activation analysis (NAA). Mostly the rotary specimen rack with 80 irradiation positions and the central thimble are used. Here, the samples are transferred manually. Furthermore, for the production of nuclides with short half-lives (up to a few minutes), three

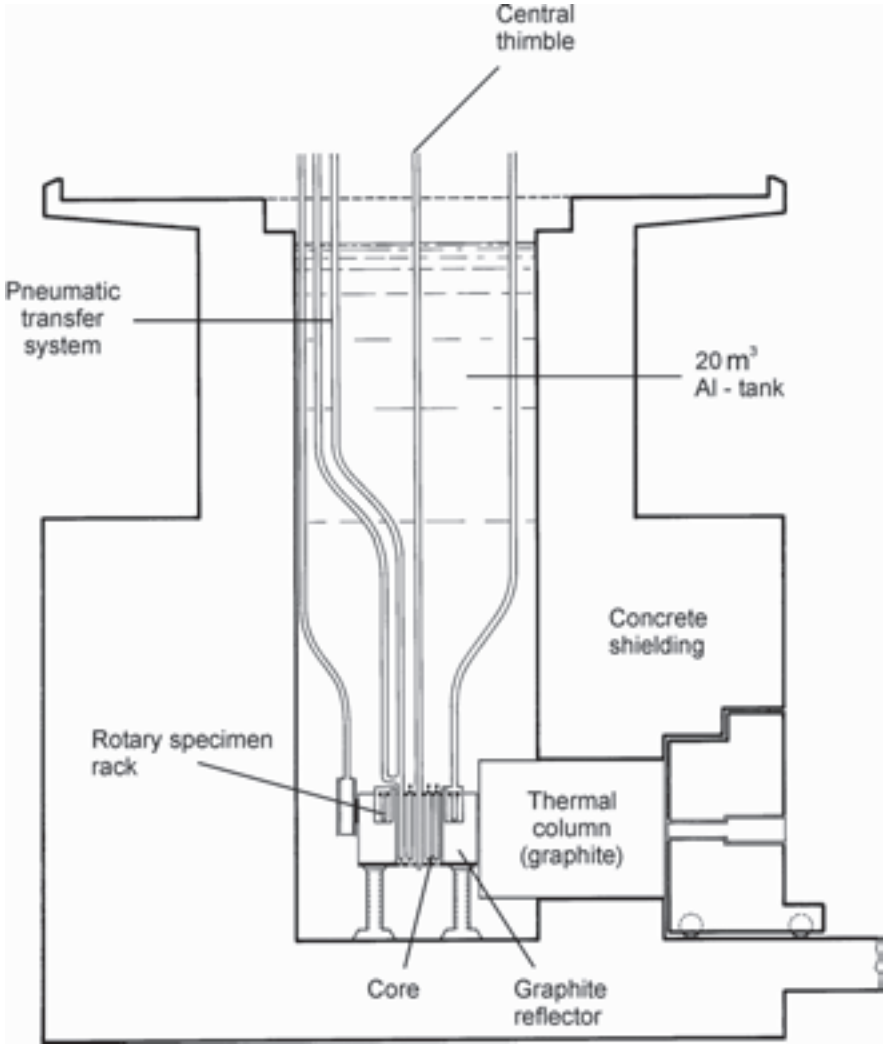

FIG. 1. Vertical cross-section view of the Triga Mainz indicating the position of the reactor core surrounded by a graphite reflector. The central thimble, the rotary specimen rack and the pneumatic transfer systems are used mostly for neutron activation analysis. The thermal column is a large experimental unit that provides well thermalized neutrons.

transfer systems (rabbit systems) are available. From a terminal located in the reactor hall or in a radiochemical laboratory, the samples are transferred pneumatically to the irradiation position and back. With these systems, transport times of 1–5 s can be achieved. In addition to that, the Triga Mainz is

equipped with four beam tubes where transport times of about 0.5 s are possible by means of special rabbit systems. The thermal column is a further irradiation unit that provides well thermalized neutrons. It consists of a boral lined aluminium container filled with blocks of graphite (see Fig. 1). For irradiations in the thermal column up to five graphite blocks (102 mm × 102 mm × 1270 mm) can be removed to introduce the sample. Table 1 summarizes the neutron fluxes at the various irradiation positions [6].

2.2. Pulse mode operation

For pulse mode operation, the reactor is brought to criticality at a low steady state power, normally 50 W, and then a control rod is shot out of the reactor core with compressed air. Due to this sudden insertion of excess reactivity, the power rises sharply with a reactor period of only a few milliseconds. The pulses have a shape that can be approximated by a Gaussian function with an FWHM in the millisecond range. The ratio of the pulse generated activity A_P to the saturation activity A_S [5], which is rapidly reached for short lived nuclides under steady state conditions, is given by Eq. (1):

$$A_P/A_S = (0.737 \cdot t_P \cdot R_P)/T_{1/2} \quad (1)$$

TABLE 1. THERMAL AND EPITHERMAL NEUTRON FLUXES AT THE DIFFERENT IRRADIATION POSITIONS OF THE TRIGA MAINZ AT A POWER OF 100 kW(th)

Irradiation position	Thermal flux[a] ($n \cdot cm^{-2} \cdot s^{-1}$)	Epithermal flux[b] ($n \cdot cm^{-2} \cdot s^{-1}$)
Rotary specimen rack	7×10^{11}	4×10^{10}
Transfer systems	$1.6 – 1.8 \times 10^{12}$	$4.6 – 5.6 \times 10^{10}$
Beam tubes	$1.0 – 5.4 \times 10^{11}$	$7.6 \times 10^{8} – 1.6 \times 10^{10}$
Central thimble	4.2×10^{12}	1.4×10^{11}
Thermal column (hot end)[c]	3.1×10^{10}	2.1×10^{8}
Thermal column (cold end)[c]	2.6×10^{7}	6.8×10^{2}

[a] $E_n \leq 0.4$ eV.
[b] $E_n \geq 0.4$ eV.
[c] Central irradiation channel.

Here, T_P is the pulse width at half maximum and R_P is the ratio of the pulse peak power to the maximum steady state power output (100 kW(th)). For a pulse peak power of 250 MW(th), T_P is ~30 ms leading to an activity ratio $A_P/A_S = 55\ s/T_{1/2}$. Thus, for a 55 s nuclide the activity produced with a 250 MW(th) pulse is equal to the saturation activity obtained by steady state irradiation. With Eq. (1), one can estimate for which short lived nuclides activation by pulsed irradiation is advantageous.

3. NAA

NAA is a versatile method for various analytical problems due to its simplicity, multi-element capacity and sensitivity [7, 8]. Instrumental neutron activation analysis (INAA) is performed without any chemical separation steps, whereas the radiochemical neutron activation analysis (RNAA) applies chemical procedures either prior to or after the neutron irradiations. Figure 2 shows the detection limits that can be achieved by INAA for a number of

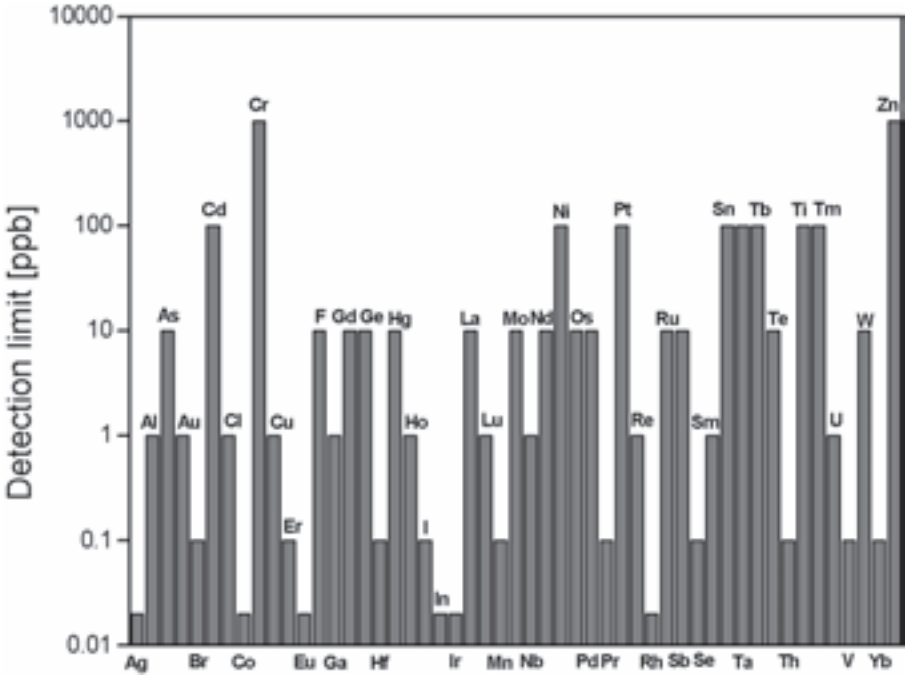

FIG. 2. Estimated detection limits achieved by INAA for a number of elements for a neutron flux of 10^{12} $n \cdot cm^{-2} \cdot s^{-1}$ at the Triga Mainz [6].

elements with a neutron flux of 10^{12} n·cm^{-2}·s^{-1} as provided by the Triga Mainz at a power of 100 kW(th) NAA, in combination with high resolution gamma spectrometry, can simultaneously determine 30–40 elements, down to a range of 0.01 ppb (10 pg/g) for some elements. The selectivity and the detection limit can further be improved by applying coincidence or anticoincidence systems reducing the Compton background [9–11]. Delayed neutron activation analysis (DNAA) is a special version and uses the counting of beta delayed neutrons emitted from very neutron rich fission products, as obtained by the irradiation of fissile material.

Neutron activation analysis can be performed in two different ways, either as an indirect method relative to standard samples (relative method), or as an absolute technique using tabulated reactor specific k_0 values [12]. Here, spatial and temporary fluctuations of the neutron flux are monitored by means of ^{197}Au.

3.1. INAA

At the Triga Mainz, INAA is used in various research fields, such as archaeology, environmental science and food monitoring, cosmochemistry and geochemistry, as well as in material science and industrial applications. INAA is also an important analytical method employed for the certification of materials [7, 8]. In cosmochemistry and geochemistry, NAA in combination with spark mass spectrometry was applied for trace analysis of extraterrestrial material. Among others, samples from the moon as obtained from the Apollo 11 mission, the Mars meteorite ALH 84001 and a Mars meteorite found in 1998 in the Sahara Desert, Libyan Arab Jamahiriya,were analysed by INAA to determine their elemental composition [13–17]. INAA in combination with anti-Compton spectrometry [9–11] has been used for monitoring atmospheric pollution. Here, the high detection efficiency and reliability of the detection system allowed the simultaneous determination of 36 elements in atmospheric dust collected in air filters [18].

3.2. RNAA

For the investigation of highly siderophile elements (noble metals, Re and Au) in geological samples, the nickel-sulphide fusion technique, in combination with NAA, is well suited. From the element pattern and ratios, one can conclude constraints on the accretion of the Earth [19]. They also yield important information about extraterrestrial components in melts from meteorite impacts [20]. Examples for the application of RNAA in environmental monitoring are — among others — the determination of platinum

group elements (PGE) in soil [21], and the determination of organic halogen compounds in water samples; absorbable organic halogens (AOX where X = F, Cl, Br, I) [22]. For PGE studies, the nickel-sulphide technique has been combined with NAA to determine PGE exhausted from catalytic converters in cars. Two irradiation cycles were applied. In a first cycle (5 min irradiation time), Rh and Pd were determined via their activation products 4.4 min ^{104}Rh and 4.7 min ^{109m}Rh. Subsequently, a 12 h irradiation was performed for the determination of Pt via the 3.1 d ^{199}Au from the reaction ^{198}Pt (n,γ) ^{199}Pt (β^-) ^{199}Au [21]. For the determination of the AOF value, a charcoal sample loaded with organic fluorine is irradiated for 30 s using a rabbit system. Within 10 s the sample was transferred to the detector site and the activation product 11 s ^{20}F was measured [22]. The same sample was then irradiated for 5 min in a rabbit system and after a cooling time of 2 min, the remaining AOX parameters were determined simultaneously via the decay of 37 min ^{38}Cl, 18 min ^{80}Br and 25 min ^{128}I, respectively [22].

3.3. DNAA

Delayed neutron activation analysis is used for the fast determination of fissile nuclides, such as ^{233}U, ^{235}U and ^{239}Pu by measuring the delayed neutron emission of some fission products [23]. From the nuclides produced in nuclear fission, about 110 are known precursors of beta delayed neutrons emission with half-lives ranging from milliseconds to minutes [24]. Delayed neutrons emitted by the long lived isotopes ^{87}Br, ^{88}Br and ^{137}I are generally measured. Furthermore, this technique was also applied for the determination of thorium in samples containing uranium. ^{232}Th fissions only with neutrons of an energy of 1 MeV and above [23] and, therefore, a cadmium cover for the absorption of thermal neutrons is used for eliminating delayed neutrons from thermal neutron induced uranium fission. At the Triga Mainz, the fast neutron flux is much lower than the thermal flux and thus the sensitivity for thorium is decreased compared with uranium and plutonium.

Under the conditions of the Triga Mainz, the detection limits are 10^{-11} g for ^{235}U and ^{239}Pu, and 10^{-6} g for ^{232}Th [6]. Typically, the samples are irradiated for 2 min using one of the rabbit systems for fast sample transfer and after a delay time of 15–20 s, they are counted for 1 min with ^{3}He proportional tubes in a circular arrangement. The delayed neutrons thermalized by paraffin and polyethylene are detected with an efficiency of about 30%. Figure 3 shows a schematic view of the neutron detector used for DNAA. Knowing the isotopic composition of uranium, a screening of a large number of persons can be carried out with the precision demanded by the authorities. Thus, DNAA can routinely be applied for the incorporation inspection of workers in the nuclear industry.

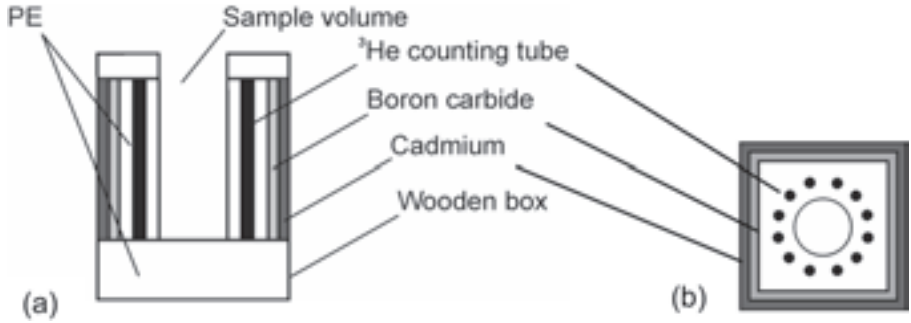

FIG. 3. Cross-section view (a) vertical and (b) horizontal of the neutron detector arrangement used for DNAA at the Triga Mainz.

4. SUMMARY

NAA was and still is an important field of research at the Triga Mainz. INAA and RNAA are applied to determine the elemental composition of various materials, such as extraterrestrial samples, rocks and sediments, and environmental samples. Using fast pneumatic transfer systems, activation products with half-lives in the minute range and below are accessible. In addition, INAA is applied for the certification of materials for science and industry. DNAA is routinely used for the determination of fissionable material in environmental samples and for the incorporation of inspections of workers in the nuclear industry.

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Part 10

TRAINING AND EDUCATION

10.1. TRAINING AT RESEARCH REACTORS
Requirements, features, constraints[*]

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1. INTRODUCTION

Since research reactors have considerable variations in their features, it is not easy to provide generally applicable guidelines for establishing training programmes within them, or to cite the general features that make a particular research reactor an excellent training reactor.

However, as low power research reactors are predominantly used for training purposes, they have characteristics that make them more suitable facilities for training. Moreover, special research reactors have been designed for training and education in the past.

Training and education has many facets, depending upon the objectives of the training programme, on the needs of the individuals to be trained — including background and the intended job or work subsequent to the training. This introduction to training at research reactors highlights some of the aspects which should be considered when applying a research reactor for training in the nuclear field.

2. OBJECTIVES

When considering the design of training courses or standard experiments for college and university students, the operator of the research reactor must determine the objectives for the different training sequences. This may include various subjects, such as general nuclear technology or nuclear physics, gaining experience in research or power reactor operators, training researchers at a scientifically used research reactor, educating people working with radiation at facilities other than reactors, or even training individuals who may be future experts on the licensing of nuclear facilities of all kinds.

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* This paper was prepared by A. Lee, AECL, on the basis of material from a presentation on the same subject delivered at the IAEA Technical Meeting on Purpose Designed Research Reactor Features (30 June–2 July 2003) at Vienna. The paper has also been reviewed by J. Razvi, General Atomics.

For university students, a broad programme is recommended. The range should comprise the demonstration of the reactor physics principles, as well as the generation of radioactive sources for their further investigation in a radiation physics laboratory.

For operators, the safe operation of the reactor should be the dominant objective. To a large extent, this will mean demonstrating to trainees the operating characteristics of the plant, such as core loading and unloading procedures, measurement of the effectiveness of all control means, the approach to critical status, reactor power manoeuvres, xenon transients and their control, slow and emergency shutdowns, etc. Unfortunately, some of those operational procedures require the interruption of the operation of the research reactor for a longer period of time and, thus, can not be part of a standard training course unless the reactor is dedicated for training exercises.

For the training of researchers, the operation of the research reactor itself is of less importance. Their interest will be directed to the experimental installations and instruments at the research reactor, using them to activate material samples and to investigate them after, or to use and to measure the radiation originating from the reactor core directly. This implies that basic laboratory equipment must be available at the reactor station.

One training subject, however, should be common to all three groups of trainees: radiation safety. Such training shall comprise methods of radiation protection, as well as measuring and shielding different radiation at a research reactor. As such, the trainees shall be made aware of all the safety measures, procedures and devices utilized at research reactors.

3. SUITABLE FEATURES

Research reactors are suitable in their own unique and different way for training and education. The questions that need to be asked are which features make a research reactor especially useful for training technicians, students and experts, and whether such features are the same for every variety of trainee mentioned.

A reactor for providing university student teaching facilities should offer an easily accessible core, a rather low reactor power, the ability to obtain flux maps and simple devices to activate material probes used to generate radioactive sources.

A training facility for reactor operators, on the other hand, should enable the demonstration of parameters, such as the increase of neutron multiplication from subcritical (source) level to various power levels. The variation of the power level should encompass several decades of power, and upwards as well

as downwards changes in power levels. In addition to the full scope of power variation, the reactivity variations shall be incorporated, especially those originating from the xenon transients but also, as far as possible, from inserting experiments and/or probes.

A research reactor most suitable for training researchers must provide the ability to install, to vary and to investigate prototypes of experimental devices, to test the devices themselves, but also their optimal positioning inside or around the reactor core.

4. CONSTRAINTS

Training and education is not an activity at a research reactor that can be started and stopped at any time. Many of the desirable training features generate, when applied, constraints to the operation of the reactor and its parallel use for other purposes. For example, flux mapping requires the installation of appropriate detectors (activation foils or wires) inside the core. Introducing as well as removing them has an impact on the core behaviour. Any introduction of a capsule with a probe inside generates a local (flux shaping) plus a global (reactivity feedback) disturbance of the core status. Frequent startups and shutdowns prevent any other use that may require a constant flux for a certain time interval. Thus, no research reactor operator should assume that good training courses and comprehensive education in nuclear matters can be performed without interference with other uses of the plant, such as scientific experiments or production processes.

Finally, the operator may be required to deal with side issues, such as the acceptance at the plant of personnel who are not familiar with a nuclear facility and have no prior experience of working at a reactor facility. This creates additional burdens on the operator, in terms of additional training and radiological safety supervision, as well as other administrative details (including implications for physical security measures).

5. EXAMPLES

The optimal research reactors for training and education purposes are low power reactors, and probably those without an ambitious research programme or involvement in large scale industrial production and services. There have been some research reactors which have been designed for just that purpose, such as the ARGONAUT and SUR type plants in the 1950s, the Canadian SLOWPOKE, and the Chinese MNSR, as well as the most

widespread and most numerous research reactor of any type, the Triga in its low power configuration. Of these, the Triga and the MNSR are still marketed. Some of these 'training reactors' offer established courses and contribute substantially to the acceptance of nuclear technology by conveying nuclear technology knowledge to university students and other trainees. Some research reactors operate installations that are especially suitable for training purposes, such as the second control room at the RA-6 reactor, or the critical core set-ups at many Japanese and French research reactors.

However, when training and retraining reactor operators, the cooperation at a more powerful plant seems to be the better solution, as such involvement in the standard procedures of a research reactor reflects the reality of the (later) job better than a low power reactor without complex features. The more the features of the 'training reactor' are similar to the plant, the better the operator can be prepared. For this, the best choice is a sequence starting with a small, simple, low power reactor, followed by a more powerful and complex research reactor with many operational constraints.

6. COOPERATION

It is worth evaluating the ideas and the implementation outlined in papers 10.3 and 10.4, which describe cooperation between several universities and research reactor facilities to offer a comprehensive education (and research) programme in nuclear technology at different research reactor plants with different user facilities, as well as nuclear and reactor physics lectures and courses at related universities. These examples can initiate similar cooperation between nuclear facilities and institutions in other regions for the benefit of the entire nuclear community.

10.2. THE AKR TRAINING REACTOR OF THE UNIVERSITY OF TECHNOLOGY DRESDEN AND ITS EXPERIMENTAL PROGRAMME FOR EDUCATION

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1. INTRODUCTION

The training and research reactor AKR (from the German Ausbildungskernreaktor) of the University of Technology Dresden was put into operation in the late 1970s, i.e. at a time when experts in Western and Eastern European countries expected an extensive development of nuclear energy generation. In the eastern part of Germany, i.e. the former German Democratic Republic, a 70 MW(el) WWER nuclear power plant (NPP) was in operation at Rheinsberg since 1966. A further five NPPs of the Russian WWER type (440 MW(e)) were put into operation at Greifswald on the Baltic Sea coast between 1973 and 1989. Three more plants were under construction at that site. At the end of the 1980s, a new NPP was being planned at Stendal, a 1000 MW(e) WWER plant.

For construction, licensing, operation and maintenance of these NPPs, a well educated and trained staff was required. Authorities and technical surveillance organizations should have qualified experts at their disposal, too. Consequently, in 1968 the University of Technology Dresden introduced appropriate courses for the education of nuclear engineers in order to contribute to fulfilling the demands of industry, science and administration. Students were taught in lectures, but theoretical knowledge had to be combined with practical experience based on an extensive programme of fundamental experiments in the fields of reactor physics, neutron physics, nuclear technology, radiation measurement techniques, radiation protection, radiation dosimetry and others. The full scale of this experimental programme can preferably be made available by small training reactors which can be operated with great diversity in terms of experimental intentions and without commercial restrictions.

In the early years, such practical exercises were carried out at zero power research reactors at the Central Institute of Nuclear Research at Rossendorf, located about 20 km from the university campus, outside of Dresden. However, the disadvantages quickly became evident:

— Due to the distance between the university and the research centre, it was a considerable effort both for students and for teachers. The rest of such a day was lost for other lectures.
— As for other research reactors, the facilities at the research centre were integrated in fixed schedules, e.g. as an experimental device for various research programmes or for radioisotope production. Flexible conditions for the reactor operation according to the requirements of students' experiments were usually not possible. Admittance conditions to the facilities in the nuclear research centre were very restrictive for the public (and students are more or less members of the public). Overcoming these restrictions required administrative efforts, advanced planning and quick changes in the composition of the student groups.

For those reasons, a university research reactor on campus was recommended. That plant should be an attractive experimental device for students, and its design and parameters should be adapted to the special requirements of education as per Table 1.

Between 1962 and 1979, corresponding training reactors were built and put into operation at 14 universities and colleges in Germany (East and West) (see Table 2 and Fig. 1), preferably 12 facilities of the SUR100 type [1]. Eight of these training reactors are still operational. However, it can be expected that the number of licensed training reactors in Germany will decrease dramatically within the next few years because the Federal Government in Germany is going to offer the universities financial support for the disposal of the nuclear

TABLE 1. CRITERIA AND THEIR REALIZATION AT A UNIVERSITY REACTOR

Criteria	Demands for realization
Siting on campus inside the area of the university	High safety and reliability requirements
Simple recalculation of the most important reactor parameters	Simple construction
Reactor operation by students permitted (supervised by the reactor staff)	Simple operational procedures
Miscellaneous possibilities for multiple use of the facility	Extension of the experimental features by accessory devices
Low effort for operation and maintenance	Simple construction and restriction to a minimum of necessary components

TABLE 2. TRAINING REACTORS IN GERMANY

Site	Institution	Type	Start of operation	Status
Aachen	College	SUR100	09/65	Operational
Berlin	University	SUR100	07/63	Operational
Bremen	College	SUR100	10/67	Decommissioned
Darmstadt	University	SUR100	09/63	Decommissioned
Dresden	University	AKR	07/78	Operational
Furtwangen	College	SUR100	03/73	Operational
Hamburg	College	SUR100	01/65	Decommissioned
Hanover	University	SUR100	12/71	Decommissioned
Karlsruhe	Research centre	SUR100	03/66	Decommissioned
Kiel	College	SUR100	04/66	Operational
Munich	University	SUR100	02/62	Decommissioned
Stuttgart	University	SUR100	04/64	Operational
Ulm	College	SUR100	12/65	Operational
Zittau	College	ZLFR	05/79	Operational

FIG. 1. German training reactor sites (operational and decommissioned).

fuel of their reactors if they decommission their plants. The remaining facilities must contribute to maintaining nuclear know-how and competence under the given political conditions — characterized by the current intention to phase out nuclear energy generation in Germany.

Nevertheless, even today and under those circumstances, there is considerable demand for young engineers and scientists in the nuclear sector to replace retired staff members of NPPs and of industrial suppliers, for working in fields of basic nuclear physics research, nuclear engineering, radiation protection, waste disposal and nuclear medicine, as well as in administration and technical surveillance organizations. Similar demands were also recently published in the United States of America [2], there additionally with more real prospects for the construction of new NPPs.

Concerning the AKR training reactor of the University of Technology Dresden, it is a common intention of the University and the Saxonian State Ministry of Science and Culture to continue student education at this reactor in the future. Therefore, a new licensing procedure for the AKR was started in 1998, including the modernization of the facility with considerable financial effort.

In recent years, about 1000 visitors each year took advantage of the possibilities of the AKR.

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 during practical exercises for:

- Students of nuclear engineering (duration of courses: two semesters);
- All students of physics (selected full day experiments);
- Lectureship students of physics and mathematics (selected full day experiments);
- Interested students of any faculty of the university (duration of course: one semester).

As by far not all universities and colleges own corresponding facilities to combine their lectures with practical exercises, students from other universities

of the region and from all over the country are welcome participants for practical courses at the training reactor. The duration of these courses and selection of the exercises are tailored individually to the special requirements.

For any exercise, scripts are available and can mostly be downloaded from the AKR web site describing the theoretical basis of the experiment and providing practical procedures, values to be determined, evaluation procedures, as well as how to discuss results, etc. In advance of any practical work, the preparation of the students to the special exercise is tested by means of a computer based checklist. Results of these tests are used by the lecturer to recognize deficiencies in knowledge or in preparation and to correspondingly concentrate efforts in colloquia.

The AKR's standard education programme (described here, for example, as exercises in basic neutron, reactor, nuclear and radiation physics, as well as in radiation protection at training reactors) are discussed in the following subsections.

2.1.1. Reactor startup procedures

Part 1 of the programme is the reactor startup procedure, including safety checks and an examination of reactor behaviour in various states. It is the basic exercise for all programmes at the research reactor and, in most cases, the first contact by the students with the plant. The exercise comprises startup of the reactor, adjustment of the critical state at various power levels, changes of the power level (increasing, decreasing), and various inspections and safety checks of the instrumentation and control units.

Included in that part is the study of basic reactor physics effects, such as:

— Correlation between reactivity, reactor period (or doubling time) and reactor power;
— Subcritical multiplication of the neutron startup source;
— Prompt reactivity steps;
— Stable positive reactor period with exponential power increase in super-critical reactor state;
— Correlation between prompt and delayed neutrons;
— Control of stable reactor power;
— Independence of the control rod position in the critical reactor state on absolute power level (if effects of temperature and fuel burnup are negligible as at a zero power reactor);
— Influence of the neutron startup source on the critical reactor;
— Dependence of neutron and gamma dose rate on the reactor power and on the distance from the reactor.

Usually this part is also the first close contact by the students with the special rules of radiation protection measures (changing clothes, wearing overshoes and dosimeters, rules of conduct according to the ALARA principle, check of contamination when leaving the radiation protection area, etc.).

2.1.2. Control rod calibration

For safe operation of a nuclear reactor, it is of high importance to know:

— Reactivity values of all control rods depending on their position;
— Excess reactivity;
— Shutdown reactivity brought about by the control rods.

During the exercises, position dependent reactivity values of all control rods are determined by measurements of the stable reactor period in combination with the reactivity compensation method. The physical background of the evaluation method is the INHOUR equation.

Results of the exercise are the integral reactivity curves (total reactivity), as well as the differential reactivity curves (characterization of the efficiency of a control rod at a given position). Furthermore, the excess reactivity is calculated as the maximum available reactivity to be released by withdrawal of all control rods, starting from the critical core. In contrast, the shutdown reactivity of the control rods is calculated as the negative reactivity value when all control rods are in their shutdown positions.

2.1.3. Critical experiment

The critical experiment is a check of proper fuel loading and core configuration of a nuclear reactor. At the AKR with its well sealed core, no exchange of fuel elements is feasible in a student's exercise. Instead, a stepwise variation of the distance of both core sections (see Section 3 of this paper) is carried out, leading to a corresponding variation of the subcritical multiplication of the neutron source. For any step, the multiplication factor k of the reactor and the corresponding reactivity ρ of the system are iteratively calculated. The result of this kind of critical experiment is the critical distance between both core sections.

2.1.4. Adjoint flux function

The intention of the exercise is to investigate influences on reactivity by introducing samples of different materials (e.g. PVC as a neutron absorber, PE

or graphite as typical neutron scattering materials) into the central irradiation channel of the reactor. The samples are shifted stepwise inside the channel and the reactor response is measured. The reactivity $\rho(x)$ is evaluated with respect to the distance of the sample from the core centre x and compared with calculations.

2.1.5. Pile oscillation experiments

While the exercise 'control rod calibration' and 'adjoint flux function' are based on the measurement of the stable reactor period (i.e. in the equilibrium state between prompt and delayed neutrons), pile oscillation experiments require a more detailed way of looking at the theoretical basics of reactor kinetics, including the solution of point kinetic equations for time dependent reactivities $\rho(t)$. By means of a mechanical pile oscillator fitted to the central experimental channel of the reactor, the function $x(t)$ as sample position with respect to time is measured using a multichannel analyser which is operated in multiscaling mode. This function is weighted with the result of the adjoint flux function $\rho(x)$ in order to get $\rho(t)$. The experimental result will be compared with an independent calculation.

2.1.6. Demonstration of neutron activation and decay of various radioisotopes

The production of radionuclides by neutron capture is one important purpose of many research reactors. Of course, in the low neutron flux density of a zero power training reactor, no real production of nuclides is possible on an industrial scale, but the effect of the generation of radioactive nuclei itself can be demonstrated absolutely. Activation, saturation and subsequent decay can be investigated with respect to reactor power (i.e. neutron flux density), time, capture cross-section, half-life time, sample size, etc. Students get the experience that materials are activated in the neutron field, but they likewise learn that activity decreases subsequent to the irradiation according to the law of radioactive decay. Experimental results are compared with calculations on a relative and absolute scale.

2.1.7. Identification of radionuclides by means of high resolution gamma spectroscopy

High resolution gamma spectrometry is one of the most used techniques for non-ambiguous identification of radionuclides. Physical properties of different kinds of detectors (e.g. simple traditional NaI scintillator/photomultiplier

combinations with their high efficiency but comparably poor energy resolution and actual HPGe semiconductor detectors) and required features of amplifiers and multichannel analysers are investigated. The HPGe spectrometer is calibrated in its energy scale, as well as with respect to absolute energy dependent efficiency using a set of well known calibration sources. The procedure of neutron activation analysis (NAA) is demonstrated by the activation of 'unknown' samples in an irradiation channel of the reactor and subsequent identification of the isotopic composition, qualitatively and quantitatively. Traces of radionuclides can be measured and identified in samples taken from the natural environment (e.g. K-40) or industrial goods (e.g. ancient tiles or watches). Another issue of NAA is the use of activation foils for the determination of absolute neutron flux densities in different energy ranges at nuclear facilities according to the conversion of the formula for activation with respect to the neutron flux as an unknown value.

2.1.8. Radiation protection and shielding measurements

In the radiation directed experiments, students obtain knowledge of dosimetric units and of dosimetric limits defined in the German Radiation Protection Ordinance. Characteristics of ionizing radiation are highlighted and from these characteristics, essential principles of radiation protection are derived. The student gets experience in the correct use of suitable types of measuring devices for different types of radiation and measuring conditions, as well as in the evaluation of measured values. Basic rules of radiation protection according to the ALARA principles are demonstrated experimentally, such as:

— Keep distance;
— Use shielding;
— Shorten length of stay inside radiation fields.

The basic physical dependences of dose rates or of total doses upon these parameters are explored. Efficiencies of several shielding materials, such as Pb, Fe or different qualities of concrete, are investigated and shielding coefficients are derived and compared with data in literature. Using these data, required thicknesses of shieldings can be calculated and experimentally verified in order to obtain defined dose rates meeting given conditions at the training reactor.

2.1.9. Radiation measurement techniques

Fundamental characteristics of different kinds of radiation detectors are explored with special attention to gas filled ionization detectors, such as

ionization chambers, and proportional and Geiger-Müller counters. The dependence of the counting rate from the high voltage is measured and discussed with reference to the adjustment of suitable parameters for proper application. Students become familiar with the statistical character of any measured value being essential for assessing accuracy (Gaussian distribution of measured data, arithmetic mean value, statistical uncertainties and standard deviation). Attenuation curves are measured for different materials, and linear as well as mass attenuation coefficients are determined and discussed.

2.2. Application in research projects

Due to its physical characteristics, a zero power reactor, such as the AKR, offers only limited possibilities for research. However, it can be used for all those projects that high neutron fluxes are not required for, but where variable operational conditions and low costs are requested. Consequently, the AKR was and is involved in research projects, such as:

- Investigations on sophisticated neutron detectors;
- Development of measuring techniques for safeguards purposes (the result is a so-called criticality tester) and a corresponding training of IAEA and EURATOM inspectors for the application of the devices developed;
- Radiation spectrometry in mixed neutron–photon fields, i.e. evaluation and improvement of existing measuring techniques and preparation of a compendium as a guideline for application;
- Experiment to calculation comparison of neutron and gamma energy spectra in reactor pressure vessel benchmark arrangements for reactor material dosimetry purposes.

2.3. Information centre for the public

Two advantages of small, low power training reactors on the campus of a university is their central location and that no strict admittance restrictions exist as they do in many other nuclear installations. Hence, these facilities are suitable for use as information centres for groups or for private individuals. Besides the transfer of basic knowledge, it is of high importance to enhance the quality of the public discussion about nuclear energy. Where else is nearly any person 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. TECHNICAL DESIGN AND SAFETY FEATURES OF THE AKR

The training and research reactor AKR is a thermal, homogeneous, solid material moderated zero power research reactor with maximum continuous power of 2 W. Its features are:

- Chain reaction is maintained mainly by thermal neutrons;
- Nuclear fuel and moderator material are distributed homogeneously in the fuel plates;
- Moderator material is solid polyethylene (without liquids such as water);
- Extremely low nuclear power of only 2 W ('zero power') allows the effects of temperature, fuel burnup, formation of nuclear waste, activation of structural materials; Xe poisoning and others to be neglected.

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 became critical for the very first time on 28 July 1978. It was developed analogously to the well known types AGN-201 [3] and SUR100 [1], and has proved to be efficient and reliable in practice since. Advantages of the AKR design are:

- Application of low enriched uranium (LEU), i.e. ^{235}U content in the fuel is $\leq 20\%$;
- Low absolute amount of nuclear fuel (total mass of ^{235}U in the reactor <1 kg);
- Desired characteristics of the reactor with respect to its inherent safety features (negative temperature coefficient of reactivity; integrity of the core maintained in 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 ($\Phi_{max} = 5 \times 10^7$ n·cm^{-2}·s^{-1});
- Free space above the core for possible installation of additional experimental equipment.

The structure of the training reactor is shown in Figs 2 and 3. 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% enriched in ^{235}U, 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 g ^{235}U). Within certain restrictions, the AKR is a minimum critical mass reactor.

For safety reasons, the core consists of two separable sections. The fuel elements of each section are enclosed in a hermetically sealed aluminium 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 supported barrier prevents an uncontrolled leakage of radioactive fission products even in the

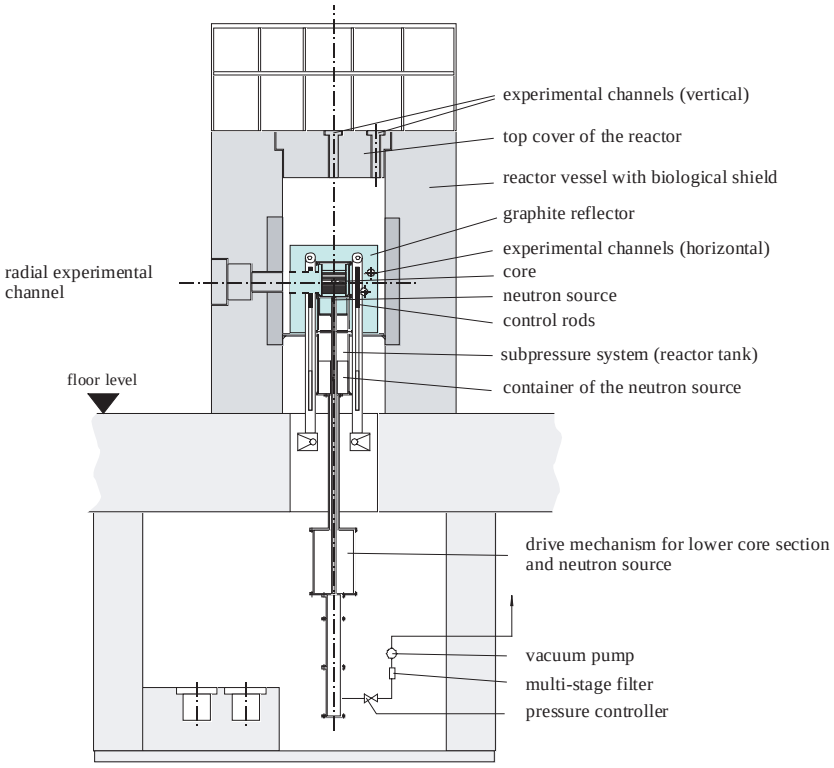

FIG. 2. Vertical section of the AKR training and research reactor.

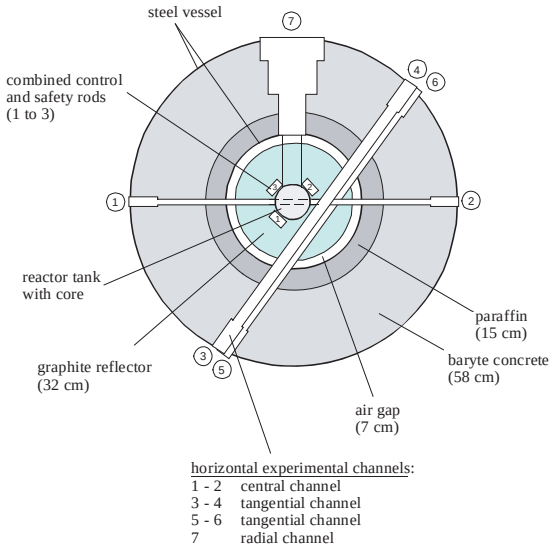

FIG. 3. Horizontal section of the AKR training and research reactor at core level.

unlikely case that all the other internal retention barriers fail. The subpressure in the tank is maintained continuously by a prevacuum pump, its control is automatically performed by means of a pressure controller even when the AKR is shut down. A multistage filter is positioned between the tank and the vacuum pump.

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. Through a tube within this mechanism, the startup neutron source (Am–Be, neutron yield 2.2×10^6 n·cm^{-2}·s^{-1}) is moved from the source container to the bottom side of the core.

The AKR is controlled by three cadmium absorber plates. 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, both falling down into their shutdown positions by gravity.

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 power level is limited only by the effectiveness of the biological shield. It consists of paraffin and baryte heavy concrete with a total thickness of 750 mm. Starting from the condition that the equivalent dose just outside the shield should never exceed 10 mSv/a, even under the worst circumstances, a 2 W continuous operation is possible. At 2 W power level, the maximum flux density of thermal neutrons in the central experimental channel amounts to about 5×10^7 n·cm^{-2}·s^{-1}.

As already mentioned, the safe and foolproof operation of the reactor is guaranteed by a combination of inherent safety, engineered safety and administrative procedures. The design and operation conditions of the AKR are based upon the considerations that prompt criticality must not occur at all and undue increase of the power must not endanger the operators, the environment or the reactor itself. Thus, the nuclear 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 of them is sufficiently high to shut down the reactor and keep it subcritical. Thus, even in the case of failure of any two of the single rods, the AKR will be shut down reliably;
— Independently 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. This ensures 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 ± 0.05) ¢/K was found. Estimates have shown that power excursions would be self-limiting before damages occur at the reactor itself or at 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 were dimensioned such that the rates of reactivity changes are below 0.0001 s^{-1} (expressed in Δk/k).

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

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10.3. CONTRIBUTION OF A SMALL UNIVERSITY REACTOR TO NUCLEAR RESEARCH IN EDUCATION AND TRAINING

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1. INTRODUCTION

The Triga Mark II reactor in Vienna, operated by the Vienna University of Technology, is the research reactor facility closest to the IAEA. Its main tasks are nuclear education and training in the fields of neutron and solid state physics, nuclear technology, reactor safety, radiochemistry, radiation protection and dosimetry, and low temperature physics and fusion research. Academic research is carried out by students in the fields mentioned. The students are coordinated and supervised by about 70 staff members with the aim of a Master's Degree or PhD in one of the areas mentioned. In addition, the Atomic Institute of the Austrian Universities cooperates closely with the IAEA, located nearby, in research projects, coordinated research projects (CRPs) and in supplying expert services. Regular training courses are performed for the IAEA for safeguard trainees. Moreover, fellowship places are offered for scientists from developing countries and staff members carry out expert missions to research centres in Africa, Asia and South America. Special nuclear material (SNM) belonging to the IAEA is stored for calibration purposes at the Atomic Institute.

A summary follows of how and to what extent low power research reactors can efficiently be used to serve university education and training, cooperation with international and national networks, as well as for the IAEA in various fields, such as nuclear safeguards and participation in international coordinated projects.

2. UNIVERSITY EDUCATION AND TRAINING

The Triga research reactor in Vienna has a maximum continuous power output of 250 kW(th) and is therefore considered a low power research reactor. Since the moderator, zirconium hydride, has the special property of moderating less efficiently at increasing temperatures, the Triga research reactor can also be operated in a pulse mode (with a rapid power rise to 250 MW for roughly

40 ms). That power rise is accompanied by an increase in the maximum neutron flux density from 1×10^{13} n·cm^{-2}·s^{-1} (at 250 kW) to 1×10^{16} n·cm^{-2}·s^{-1} (accompanied by a rapid power rise to 250 MW for roughly 40 ms).

In accordance with its purpose as a research reactor, the Triga Mark II is equipped with a number of irradiation devices (Fig. 1), which are used for neutron and solid state research, as well as for neutron activation analysis (NAA) and in-house radionuclide production. These facilities consist of the following installations:

— Five reflector irradiation tubes;
— One central irradiation tube;
— One slow pneumatic transfer system (transfer time 4 s);
— One vertical fast pneumatic transfer system (transfer time 0.3 s);

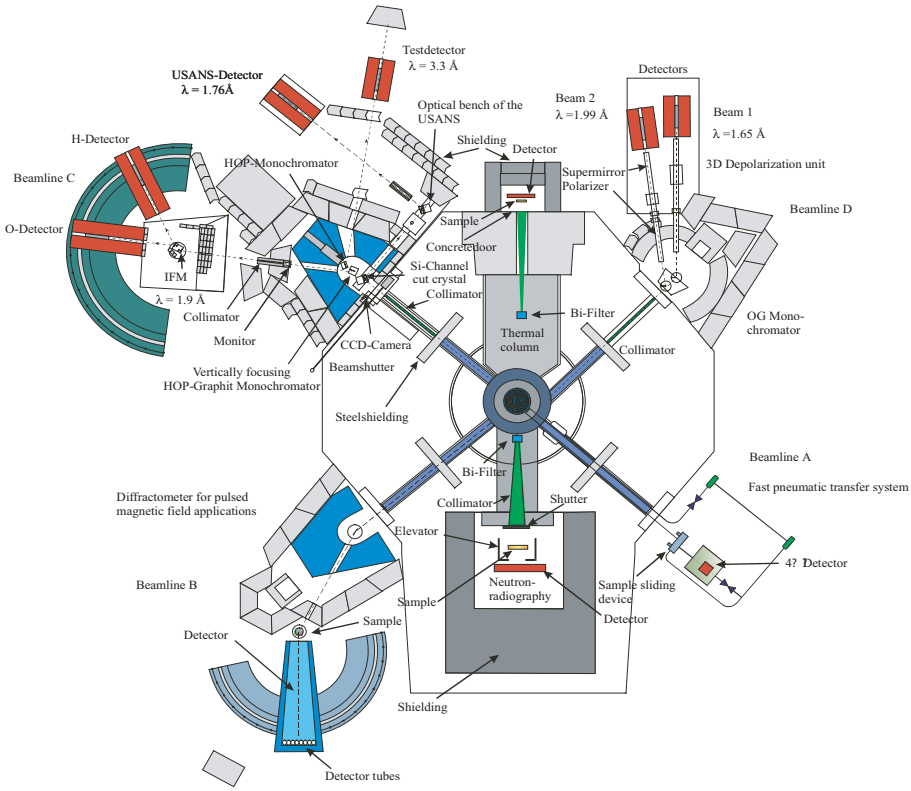

FIG. 1. Experimental set-up at the Triga Mark II reactor, Atomic Institute of the Austrian Universities.

— One horizontal fast pneumatic transfer system (transfer time 20 ms);
— Four neutron beam holes;
— One thermal column;
— Two neutron radiography facilities.

During the past 15 years, about 580 students graduated from the Atomic Institute in the fields of neutron and solid state physics, nuclear technology, reactor safety, radiochemistry, radiation protection and dosimetry, low temperature physics, and fusion research. In addition, there is intense, ongoing international cooperation with the Institut Laue Langevin (ILL) in France, the Paul Scherrer Institut (PSI) in Switzerland, the Forschungszentrum Jülich (FZJ) in Germany and the Rutherford Analytical Laboratory (RAL) in the United Kingdom.

3. COOPERATION WITH INTERNATIONAL AND NATIONAL NETWORKS

In addition to the devices mentioned, a number of special courses on reactor physics and kinetics, on reactor instrumentation, and on radiochemistry are held at the Atomic Institute for national and international organizations. These practical courses containing different exercises, composed of the regular course programme, are also carried out for commercial external customers, such as reactor staff from various NPPs, regulatory bodies and guest scientists from other universities.

Further, the Atomic Institute participates in the European Nuclear Engineering Network (ENEN). ENEN was initiated in Belgium in mid-1999 with the aim to coordinate and harmonize nuclear education in Europe and to improve the mobility of students and academic personnel. It started in October 2000 with representatives from 17 European countries: Austria, Belgium, Czech Republic, Finland, France, Germany, Greece, Hungary, Italy, Netherlands, Romania, Slovenia, Slovakia, Spain, Sweden, Switzerland and the United Kingdom.

Within this network, a model of an international training course, the Eugene Wigner Training Course for Reactor Physics Experiments, was initiated and performed in 2003 and repeated in 2004, together with the Slovak University of Technology, the Czech Technical University and the Budapest University of Technology and Economics. This course was also financed by the IAEA and by the Slovak Company CNES. The basic features of this course are listed as follows:

— Three weeks in duration;
— In English;
— Main emphasis is to perform reactor physics experiments at three different research and training reactors in three different cities (Vienna, Prague, Budapest);
— Experimental work is preceded by theoretical lectures aiming to prepare students for the experiments (Bratislava);
— Students' work will be evaluated and, if successful, the students will get a certificate;
— ENEN recommended European Credit Transfer System (ECTS)[1] of the course is between 6 and 8;
— Final value of the course will be determined by the students' home university accreditation body.

To supply at least one real example of an international training course, details of the content of the course are given in the Appendix. There were 21 students from 11 different countries participating in that course.

In addition to that special training, many other theoretical courses are offered at the Atomic Institute to complement the practical training by subject, such as nuclear engineering, alternative nuclear energy systems, reactor physics, instrumentation and control of reactors, nuclear fuel cycle, neutron physics, radiation protection and dosimetry.

Some of the courses mentioned are in English and are also presented in that language within the regular university curriculum in physics.

4. COOPERATION WITH THE IAEA

The main areas of cooperation with the IAEA are:

— Tests of newly developed detectors;
— Safeguards;
— Training of staff.

[1] ECTS is a system established within the Bologna Agreement and classifies courses at European universities with a uniform scale. In that scale a student has to reach 30 credits per semester and 300 credits in total to obtain a Master's Degree.

4.1. Tests of safeguards detectors

Safeguards inspectors increasingly use miniature CdZnTe detectors for practical applications in the fields of verification of nuclear material which have many advantages compared with previously used high purity germanium (HPGe) detectors, as they are all solid state, do not need any cooling and their resolution is between Na(I) and HPGe. The preparation and planning of applications requires good knowledge of the characteristics and spectral performance of these detectors. The great success of previous gamma spectra catalogues for Na(I) and HPGe detectors has shown this need. In one project, a spectral data catalogue for room temperature semiconductor detectors has been initiated. The catalogue is not only for users and application planners but also for researchers modelling gamma spectra and developing spectrum processing software. In the current project, a total of 26 nuclides have been measured with three different CdZnTe detector types. The resulting gamma spectra have been catalogued and are available through the IAEA. Furthermore, different types of these detectors have been tested in terms of their performance in practical applications.

4.2. Other safeguards related projects

Being the closest research reactor to IAEA headquarters, various departments of the IAEA utilize extensively the Triga facility, mainly in the fields of nuclear safeguards. For example, various SNM samples are stored at the Atomic Institute to be used frequently for calibration purposes for safeguard instruments to be later applied in NPPs. For this purpose, two WWER 440 fuel assemblies are stored in the fresh fuel storage facility of the Triga in Vienna. Video cameras and electronic seals used for surveillance of sensitive areas in NPPs are tested for their radiation resistance to neutron and gamma exposure. Video cameras, radiation sensitive electronic parts and electronic seals are exposed to fast and thermal neutron flux or pure gamma radiation, as well as to mixed radiation fields at various irradiation facilities. The equipment is monitored on-line and any failure is analysed. Finally, the exposure dose is determined at the point of time that the instrument fails. Subsequent to these experiments, selective improvements of the relevant electronic parts can be performed.

4.3. Training

Every two years, a four week training course is carried out within the Safeguards Traineeship Programme of the IAEA to train future safeguards

inspectors from developing countries to fulfil the application criteria for safeguards inspectors. In the past 15 years, approximately 80 trainees have passed through these courses at the Atomic Institute. Finally, it has to be mentioned that the Atomic Institute readily accepts IAEA fellows from developing countries to be trained in various nuclear fields for periods from one month to one year. During the last 15 years, more than 100 fellows from 30 different countries have received their training at the Atomic Institute and continue the cooperation with the Atomic Institute following their return to their home institutes.

5. CONCLUSIONS

Although Austria does not operate an NPP and, in fact, objects to nuclear power as part of its national policy, it is necessary to keep a certain level of nuclear competence in the country. This is also necessary for other countries following the same policy, such as Denmark, Italy or Portugal. Students graduating from the Vienna University of Technology/Atomic Institute have received broad and intensive training. Many of them are hired by governmental institutions, nuclear medical hospitals, the IAEA, Preparatory Commission for the Comprehensive Nuclear-Test-Ban Treaty Organization,[2] research centres or universities within the European Community, such as Ispra, Italy, or Petten, the Netherlands, or the European Space Agency. At present, there is an excellent balance between the number of graduates and jobs available for qualified candidates.

[2] The Preparatory Commission for the Comprehensive Nuclear-Test-Ban Treaty Organization (CTBTO Preparatory Commission) is an international organization established by the States Signatories to the Treaty on 19 November 1996. It carries out the necessary preparations for the effective implementation of the Treaty, and prepares for the first session of the Conference of the States Parties to the Treaty.

Appendix

PROGRAMME OF THE EUGENE WIGNER COURSE

This appendix puts together the theoretical lectures held at the Slovak University of Technology (STUB), Bratislava, Slovakia, and the experiments performed at the three training and research reactors in Budapest, Hungary; Vienna, Austria; and Prague, Czech Republic.

A.1. The lectures

A.1.1. Survey of research reactors and associated systems

The survey consisted of a three hour lecture comprising:

— Survey of utilization of research reactors (overview, reactor utilization in physics, chemistry, medicine, biology, training and education, and industrial applications);
— Selected systems and components:
  • Fuel elements and assemblies for research reactors;
  • Safety and control systems (including control rods and neutron detectors);
  • Cooling and ventilation systems;
  • Experimental facilities.

A.1.2. Data evaluation techniques (related to the practical exercises)

The data evaluation consisted of a three hour lecture comprising:

— Principles of parameter estimation;
— Maximum likelihood method; method of least squares;
— Some illustrative examples;
— Solution of the least squares equations;
— Statistical properties of the estimated parameters (expectation value, standard deviation, etc.);
— Confidence intervals;
— Quality of test fittings.

A.1.3. Radiation protection and dosimetry

This part of the course consisted of a three hour lecture comprising:

— Physical and biological dose: definitions, units, interpretations;
— Harmful effects of ionizing radiations on the human body: deterministic and stochastic effects;
— Structure of health physics regulations (laws, decrees, guidelines, standards) for limiting the emission and immission of radioactivity;
— Sources of radioactivity in the environment: natural and artificial radionuclides, management of radioactive wastes;
— Measurement techniques associated with the determination of external and internal dose;
— Migration of radioactivity in environmental media, role of environmental monitoring;
— Activity of health physics services at workplaces.

A.1.4. Radiation detectors

The part of the course on radiation detectors consisted of a two hour lecture comprising:

— Principles of detection;
— Gas filled detectors: ionization chambers, proportional counters, Geiger-Müller (GM) counters;
— Scintillation detectors: scintillators, photomultipliers;
— Semiconductor detectors;
— Neutron detectors.

A.1.5. Devices for nuclear radiation measurements

The part of the course on radiation detectors consisted of a two hour lecture comprising:

— Single channel analyser;
— Multichannel analyser;
— Multiscaler mode;
— Differential and integral discriminators;
— Coincidence devices.

A.1.6. Nuclear measuring methods

The part of the course on nuclear measuring methods consisted of a two hour lecture comprising:

— Alpha spectrometry;
— Gamma spectroscopy (energy calibration, efficiency, evaluation of spectra);
— Neutron activation analysis.

A.1.7. Nuclear safety

The part of the course on nuclear safety consisted of a two hour lecture comprising:

— General information about nuclear safety;
— Defence in depth (accident prevention and mitigation);
— Safety culture;
— Legislative rules;
— Physical protection;
— Safeguards at research reactors;
— Application of nuclear safety at research reactors;
— Emergency preparedness.

A.2. Timetable of the Bratislava part

Table 1 provides the timetable for that part of the course which took place in Bratislava.

TABLE 1. TIMETABLE OF THE BRATISLAVA PART OF THE EUGENE WIGNER TRAINING COURSE

Day	Time	Activity
Tuesday	14:00 19:00	Arrival, occupying accommodation Opening ceremony, common dinner
Wednesday	08:00–16:30	Theoretical lectures (12:00–13:00 lunch break)
Thursday	08:00–16:30	Theoretical lectures (12:00–13:00 lunch break)
Friday	08:00–16:30	Technical tour to the Bohunice NPP and the Radwaste Storage Facility
Saturday		Free programme (excursion, barbecue, etc.)

A.3. Reactor experiments at Budapest

The reactor experiments were performed at the 100 kW training reactor of the Institute of Nuclear Techniques (INT) of the Budapest University of Technology and Economics (BUTE) at Budapest, Hungary.

A.3.1. NAA

NAA was applied to analyse environmental samples by comparison with standard reference materials.

A.3.2. Measurement of delayed neutron parameters and determination of uranium concentration

Half-lives and relative intensities of some delayed neutron groups are determined. The ^{235}U concentration of a depleted uranium sample is estimated by comparing the delayed neutron intensities in the sample of interest and in a uranium standard.

A.3.3. Exercises on reactor operation and control

The students learn and practise the operation and control of the reactor at different power levels.

A.3.4. Determination of the diffusion length of thermal neutrons in graphite

The measurement is performed by activation method in the thermal column of the reactor.

A.3.5. Measurement of the reactivity importance of neutron absorbers

The reactivity importance of a perturbation is studied as a function of its position in the active core.

A.4. Timetable of the Budapest part

The timetable of the experiments performed at INT-BUTE during two weeks for two groups of students (Gr. 1, Gr. 2) is given in Table 2.

TABLE 2. TIMETABLE OF THE BUDAPEST PART OF THE EUGENE WIGNER TRAINING COURSE

	Monday	Tuesday	Wednesday	Thursday	Friday
08:30 to 12:30	Gr. 1: Delayed neutrons… (A.3.2)	Gr. 1: (10:00 h –13:30 h) n-Activ. Analysis (A.3.1)	Gr. 1: Reactor operation (A.3.3)	Gr. 1: Diffusion length… (A.3.4)	Gr. 1: Reactiv. importance (A.3.5)
	Gr. 2: Reactor operation (A.3.3)	Gr. 2: Reactivity importance (A.3.5)	Gr. 2: Diffusion length… (A.3.4)	Gr. 2: Delayed neutrons… (A.3.2)	Gr. 2: (10:00–13:30) n-Activation analysis (A.3.1)
14:00 to 17:00	Gr. 2 & Gr. 1: Preparation and evaluation of the reports	Gr. 2 & Gr. 1: Preparation and evaluation of the reports	Gr. 2 & Gr. 1: Preparation and evaluation of the reports	Gr. 2 & Gr. 1: Preparation and evaluation of the reports	Gr. 2 & Gr. 1: Preparation and evaluation of the reports

A.5. Reactor experiments at Vienna

The subsequent programme of reactor experiments is the part of the course performed at the Triga Research Reactor Facility of the Atomic Institute of the Austrian Universities in Vienna.

A.5.1. Thermal neutron flux measurement

Thin Au foils are irradiated in the Triga core at 10 W both Cd covered and uncovered in different radial and axial positions. The radial and axial neutron flux distribution is determined from these Cd difference measurements.

A.5.2. Influence of void coefficient on reactor power

At 10 W reactor power a small container with different air volumes is pulled axially through the reactor core while the reactor is in automatic operation mode. The influence of volume and position in the core on the reactivity is determined.

A.5.3. Critical experiment

Ten Triga fuel elements are removed from the reactor core and consecutively reloaded. The neutron count rate is measured after each step. At each

step, measurements are performed with all control rods up and then down. Criticality is reached with all control rods up after reloading of five of the removed fuel elements.

A.5.4. Determination of the neutron absorption cross-section according to the danger coefficient method

After reactivity calibration of the reactor with a known absorber, several unknown materials are inserted at the same core position. From the reactivity response of the reactor, conclusions are drawn on the absorption cross-section of the unknown materials.

A.5.5. Reactivity values of fuel elements in different core positions

While the reactor is on automatic control at 10 W, one Triga fuel element is removed from a position at each of the five fuel rings. The movement of the regulating rod compensates the loss of reactivity. From the rod position difference and using the rod calibration curve, the reactivity value of each of the fuel elements in the different core positions can be determined.

A.5.6. Reactor power calibration and temperature coefficient of reactivity

The reactor is operated at 10 W, rod positions, water and fuel temperature is noted, then the reactor power is raised to 100 kW, again the values are noted. From the difference in rod position and fuel temperature, the fuel temperature coefficient can be determined. Then the reactor is operated for 90 min only with convection cooling and the increase of water temperature is monitored. Comparing the temperature increase with the value from a previous calibration, the thermal reactor power can be determined.

A.5.7. Demonstration of a prompt critical power excursion

The demonstration of a prompt critical power excursion requires only a short time. Due to the strong negative temperature coefficient of reactivity, the Triga research reactors can tolerate prompt critical excursions up to 1000 times the normal power mode without any damage to the core. This is demonstrated using a pneumatic rod which is removed promptly from the critical core. Typical power levels of 250–300 MW are reached for a time period of about 40 ms.

A.6. Timetable of the Vienna part

Each group of students performs the experiments described in Section A.5 over two days (Monday/Tuesday, and Thursday/Friday, respectively, as shown in Table 3).

A.7. Reactor experiments at Prague

The reactor experiments are performed at the research reactor facility VR-1 of the Czech Technical University (CTU) in Prague, Czech Republic.

A.7.1. Properties of neutron detectors for nuclear reactor control

The programme on properties of neutron detectors comprises neutron gas detectors for the experiments at the VR-1 reactor, dead time and differential characteristic of the neutron detector SNM-10, measuring of the differential characteristic and setting the optimum of the discrimination level, as well as measuring of the detector dead time by using the two source and the maximum rate method.

A.7.2. Measurements of reactivity by various methods

An introduction into the methods of measurement of reactivity, its definition and its units, is given comprising reactivity measurements in the subcritical state by source jerk method (using a quickly movable external neutron source), reactivity measurements in the supercritical reactor by positive period method, as well as reactivity measurement applying the rod drop method (control rod worth measurement).

TABLE 3. TIMETABLE OF THE VIENNA PART OF THE EUGENE WIGNER TRAINING COURSE

Time	Monday/Tuesday	Thursday/Friday
08:00 to 17:00	Flux measurement (A.5.1), n-abs. cross-section determ. (A.5.4), Reactivity values (A.5.5)	Critical experiment (A.5.3) Void coefficient (A.5.2) Power calibration (A.5.6) Power excursion (A.5.7)

A.7.3. Calibration of control rods

An introduction into the calibration of control rods comprises the differential and integral characteristics of the control rod, the inverse rate method and the mutual calibration method.

A.7.4. Study of nuclear reactor dynamics

The part of the course on reactor dynamics includes a mathematical model of the research reactor dynamics, a study of the reactor response to a negative/positive/periodical reactivity change, a study of the influence of the 'bubbly boiling' to the VR-1 research reactor operation, a simulation of the selected operating states of a power reactor of the WWER type, i.e. approaching the critical state with modelling of the dilution of boric acid concentration and of the change of the operating group of the control rods.

A.7.5. Startup and operation of the VR-1 research reactor

The startup of the VR-1 research reactor, its operation in manual as well as in automatic mode, the variation of the power level (increasing, decreasing), and the shutdown of the reactor for various reasons can be demonstrated, and various inspections and checks of the digital control and safety system can be performed.

A.7.6. Digital control systems of research reactors

An introduction to the digital control and safety system of the VR-1 research reactor is based on the microprocessor with the relay emergency chain, short description of the main parts of the digital control and safety system comprising measuring, safety and communication channels, peripheral channel and control desk.

A.8. Timetable of the Prague part

Each group of students performs the experiments at the VR-1 reactor at Prague during two days (Monday/Tuesday and Thursday/Friday, respectively) as compiled in the overall timetable of Table 4.

TABLE 4. TIMETABLE OF THE PRAGUE PART

	Monday/Tuesday	Thursday/Friday
09:00	Neutron detectors (A.7.1)	Reactivity measurement (A.7.2)
to	Digital control systems (A.7.6)	Control rod calibration (A.7.3)
16:00	Reactor operation (A.7.5)	Reactor dynamics (A.7.4)

10.4. INTEGRATING THE McCLELLAN NUCLEAR RADIATION CENTER INTO A CAMPUS BASED EDUCATION PROGRAMME

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1. INTRODUCTION

The McClellan Nuclear Radiation Center (MNRC) is centred on a 2 MW Triga research reactor, built by the US Air Force (USAF) for the purpose of radiography of aircraft parts. In October 2000, the University of California at Davis (UC Davis), United States of America, took over the operations of the facility with the purpose of integrating the research reactor facility into the University's research and education programmes. There are two problems to overcome in achieving this goal: firstly, the facility is located off-campus; and secondly, there is a gradual decline nationwide in the recruitment of nuclear engineering students and faculty. Creative solutions have therefore to be sought in order to fully utilize the research reactor facility.

The MNRC was purpose built for neutron radiography and associated imaging techniques. Four tangential neutron beam tubes provide four highly collimated thermal neutron beams for radiography in four bays. The unique configurations of these beam tubes are discussed in Section 6.6, however, it is important to emphasize that, in contrast to other research reactor facilities, the MNRC has been optimized for neutron radiography. The MNRC is, therefore, uniquely suited to developing a research and education programme using advanced imaging techniques, yet is hampered by a lack of associated nuclear engineering and applied nuclear science programmes on the UC Davis campus. A solution to this problem was found in collaboration with other research reactor facilities in the western USA in the formation of the Western Nuclear Science Alliance (WNSA).

2. WESTERN NUCLEAR SCIENCE ALLIANCE

2.1. Western Nuclear Science Alliance educational mission

The WNSA is the result of a successful solicitation of funds to the US Department of Energy through the Innovations in Nuclear Infrastructure and Education (INIE) programme. A series of 'manpower' reports over the past few years has emphasized the problem which will face the country in terms of the availability of sufficient numbers of nuclear engineers, health physicists and radiochemists to replace the large number of scientists in the nuclear industry and the national laboratories who are within ten years of retirement. Despite this demand for nuclear scientists, there has been a decrease in the number of universities offering nuclear engineering degrees and in the number of students enrolled in PhD, Master of Science and Bachelor of Science programmes. For example, in 1975 there were 47 universities offering degrees in nuclear engineering, while in 2001 there were only 24 similar programmes. The INIE programme is the DOE response to the current crisis facing the nuclear industry in the USA, that of a lack of recruitment in nuclear engineering courses and a rapidly ageing workforce.

Research reactors across the USA have been traditional training centres for skilled nuclear scientists employed by government, industry and academic institutions. Through their affiliations with universities, research reactors also have been important in developing a wide variety of analytical techniques (neutron activation analysis, non-destructive testing and radioisotope production) which are used to cater to the expanding needs of environmental, earth, biological, biomedical and materials scientists working on environmental remediation, nuclear waste management and stockpile stewardship. Students in these fields require training in nuclear physics and chemistry, and research reactor facilities offer the resources to meet these future needs of education and research in nuclear science and engineering.

The WNSA is an alliance of several institutions with research reactors and/or appropriate educational programmes. The WNSA builds on the unique strengths of these institutions and provides an effective means of addressing the nation's future requirements for nuclear scientists. The WNSA consists of the largest research reactors in the western USA and the only nuclear engineering departments on its west coast. By joining these reactors, faculty and university resources, the WNSA will provide unique opportunities to train undergraduate, graduate and postdoctoral students in a way that universities themselves could not provide. The WNSA will combine world class facilities, for example, in neutron imaging, with diverse, interdisciplinary research and

educational programmes in nuclear science and applied nuclear science, providing a mechanism for:

— Sharing and enhancing the capabilities of WNSA's reactor facilities;
— Developing new curricula to better meet future educational requirements;
— Promoting the west coast reactors to a wider scientific community.

The WNSA consists of the research reactors at the MNRC, Oregon State University (OSU), and the nuclear engineering programme at UC Berkeley, the radiochemistry programme at Washington State University (WSU) and the programme at Idaho State University.

The aim of the WNSA is to promote the scientific and technical strengths of the WNSA research reactors to the broader scientific community through the synergistic relations between the alliance members (schematically illustrated in Fig. 1). Each member of the alliance has their own diverse interests and research agenda; the strength of the alliance comes through the exchange of ideas and scientific information between the group members. The vectors for communication between group members will be the WNSA graduate students and WNSA fellows.

These strategic interactions:

— Enable high quality science to be performed;
— Best utilize existing resources and expertise without duplication;
— Promote innovation in education and research through collaborative projects;
— Enable the development of new technologies and techniques;
— Provide the academic, government and industrial communities with an increasing supply of scientists educated in the principles and applications of nuclear science and engineering;
— Promote further growth of a research community with interest in nuclear and applied nuclear sciences.

The success of the alliance depends on a large community of graduate and undergraduate students who will be given a unique opportunity to engage in diverse, interdisciplinary research programmes.

2.2. Research programmes at the MNRC

One of the most unique facilities that the MNRC can offer the research community is a computed tomography (CT) capability. On takeover from the

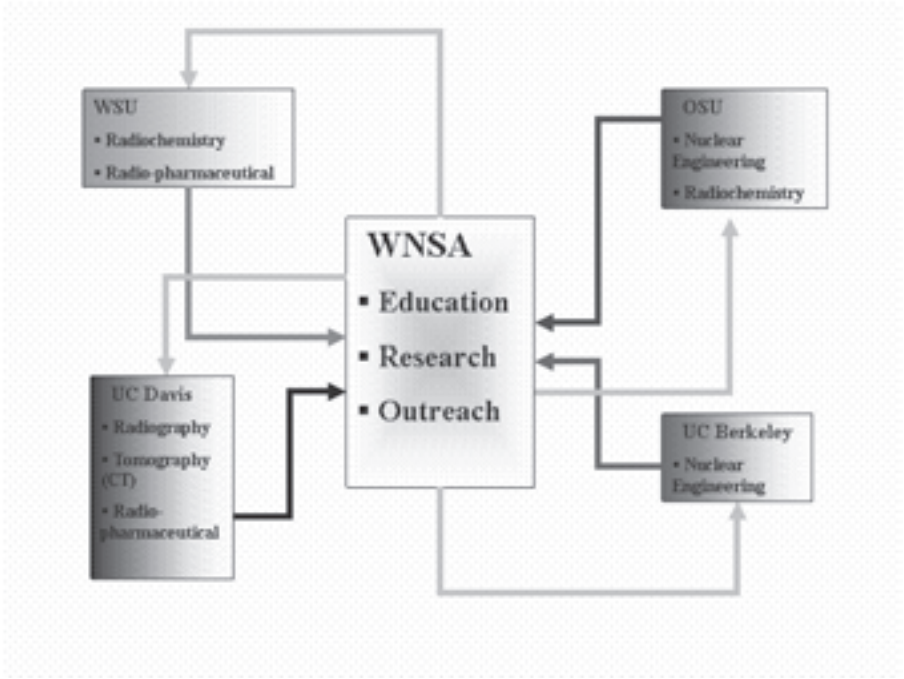


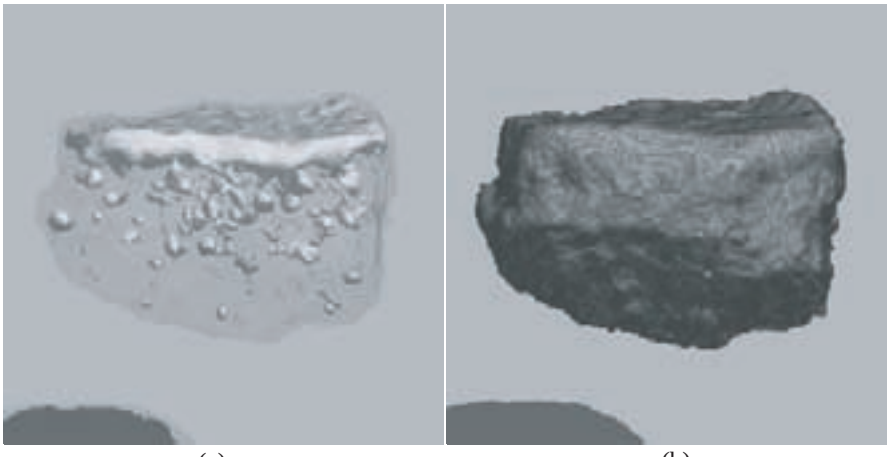
FIG. 1. Schematic diagram of the organizational structure of the WNSA. Each of the four nodes has its own research and educational strength. When combined, the interactions and transfer between nodes will provide a new generation of nuclear scientists and applied nuclear scientists.

USAF, a small research programme was developed at the MNRC by the UC Davis Geology Department to utilize this unique capability to examine complex geological materials. In these initial studies, an effort was made to identify the areas of geological interest in which neutron CT could make a unique contribution to the understanding of three-dimensional textures. Four target areas have been selected for further study.

In contrast to the more commonly used X ray CT technique, neutron CT is sensitive to light elements, especially hydrogen. This means that in geological samples containing hydrous mineral phases or organics, the three-dimensional structure can be appraised. In the initial feasibility studies, samples containing hydrogenated materials have been studied together with complex textured igneous rocks which have been compared with X ray CT. Through technique development and optimization, the neutron CT can be applied with an instrumental resolution comparable with that of X ray. The current instrumental resolution is limited to the available CCD camera and optical imaging system

and we propose enhancement of the neutron CT facilities at the MNRC to enable resolution of smaller features in the rock samples. The development of a high resolution radiography and tomography capability with associated improvements in image collection and processing software will be the first step towards the construction of a neutron microscope. This facility upgrade would fit into existing graduate student programmes at the UC Davis Geology Department; in addition, it would facilitate development of non-destructive testing components of UC Davis engineering courses.

Four subject areas have been used to test the application of the neutron CT technique to geological samples. Firstly, the thickness of partly hydrated glass fragments in under sea volcanic rocks (hyaloclastites) and the altered volcanic ash matrix has been examined for drill core samples obtained from Hawaii (Kilauea Volcano). Secondly, the three-dimensional relations of remnant bacterial structures in carbonate samples have been determined. Thirdly, the porosity and density changes in sulphide 'chimneys' from deep sea hydrothermal vents have been investigated. Finally, the complex textural features developed in layered igneous intrusions have been imaged. These four areas of research form the basis of the research programmes at the MNRC facility involving graduate students and UC Davis faculty (Fig. 2).



(a) (b)

FIG. 2. Examples of neutron computed tomography performed at the MNRC. The CT reconstruction (a) of a bacterial colony in sandstones from desert environments sample (~20 mm in maximum dimension) has a colony of bacteria which is visible in the iso-surface (surface of equal attenuation), view (b). The high attenuation of neutrons results from the presence of hydrogen in the organic bacterial colony.

2.3. Programmes at other WNSA nodes

Each of the other nodes of the WNSA has its own research strength. OSU and Washington State University maintain strong health physics and radiochemistry programmes. In addition, OSU offers a programme for neutron activation analysis, based on over 30 years of successful analysis. The latter capability offers a facility not currently offered at the MNRC and provides the opportunity for geochemical and soil science research at OSU for undergraduates and graduate students from other WNSA members, as well as allowing the transfer of knowledge and experience to the fledgling NAA laboratory at the MNRC.

The research programmes that WNSA provides are thus greater than the sum of their parts. For instance, WNSA expertise in reactor thermal hydraulics combined with expertise in neutron imaging will result in a stronger research programme than a single university could offer. The opportunity to link the strong radiochemistry programmes of WNSA universities and the national laboratories will result in benefits to faculty, students and the national laboratories. The research programmes the WNSA will undertake are innovative (e.g. dynamic imaging of thermal hydraulic events and the neutron microscope) and will result in societal benefits (e.g. the remediation of contaminated soils, plant root nutrition uptake). The opportunities for the universities, national laboratories and the private sector to work together are being implemented in the WNSA research programmes (for example, the neutron microscope and the radiochemistry of soils).

The WNSA university participants in conjunction with the national laboratories will provide calculations and documentation of radioisotope production capabilities within the research reactor community. Furthermore, the WNSA in cooperation with the DOE Office of Isotope Production will provide the only ^{125}I supplies for medical and research uses. It is expected that these radioisotope programmes will expand as the WNSA capabilities emerge.

The WNSA has the resources and a comprehensive plan to implement innovative and high quality university research, training and education in nuclear science and technology. The WNSA plan integrates resources from the universities, national laboratories and the private sector. One important aspect of the educational programme is that it actively involves participation from non-nuclear fields. The goal of this participation is to increase the knowledge base for engineers and scientists working in nuclear and non-nuclear fields. The strongest attribute of the WNSA is the combining of resources in areas of educational and facilities strength that perfectly fit the capabilities of the faculty and facility personnel.

2.4. WNSA educational mission

The WNSA participants and their individual institutions are committed to the goals of the INIE proposal. To meet these educational needs and enhance the research opportunities in neutron science, unique resources and expertise at research reactors and their host institutions are combined. The two major research reactors in the west of the USA provide an excellent example. The host institution, OSU, has a strong academic programme in nuclear engineering and health physics, as well as the highly regarded Oregon State Triga Reactor (OSTR) and radiochemistry programme. The UC Davis reactor at the MNRC is the nation's newest facility with the most advanced neutron beam imaging system in the country. UC Davis is renowned for its strong inter-disciplinary programmes in the environmental, earth, biological and material sciences. These complimentary institutional strengths, coupled with the proximity of the OSU and UC Davis facilities on the west coast of the USA, offer the opportunity to construct a bridge of cooperation and shared resources that will advance neutron science education and research in the 21st century.

This bridge will be in the form of the WNSA and will involve the development of new education and research programmes in neutron science. The WNSA will provide the mechanism for (a) sharing and enhancing the capabilities of its reactor facilities; (b) developing a curriculum that better meets the educational needs in fields using nuclear technologies; and (c) expanding research opportunities for the scientific community at research reactors in the western USA. To further the WNSA educational mission, other university participants include the WSU research reactor and radiochemistry programme, UC Berkeley Department of Nuclear Engineering, and Idaho State University.

3. EDUCATION PROGRAMMES AT THE MNRC

To offset the logistical difficulties resulting from the off-campus location of the MNRC, novel teaching approaches will be used. In designing courses, we will use existing expertise to expand and develop more traditional nuclear engineering courses.

Preliminary educational programmes have been designed to introduce graduate students to applied nuclear techniques. This has resulted in a small nucleus of research personnel who use the MNRC as an integral part of their studies. The taught part of this portion of the educational programme has been limited to a series of seminars and short courses, for example, on neutron activation analysis, run in conjunction with WNSA partners (OSU). The next

stage is to provide an intense course on radiography. This uses MNRC expertise and also has the added advantage of providing a tangible qualification, separate from a university degree.

The central mission of the MNRC is neutron radiography and imaging. Non-destructive testing, which included radiography, is an integral part of mechanical and chemical engineering curricula. To offer an industry recognized qualification in radiography will be an important contribution that the MNRC can provide to UC Davis graduates, faced with an increasingly competitive employment market. We will offer, through the university extension programme, a training programme that will be preparatory for Level I radiography certification. Through web based courses, timed quizzes and video link, the taught portion of this course can be held without students necessarily having to be present at the research reactor continually, practical portions and supplementary taught portions can be held at the MNRC, utilizing the extended working hours of the facility (two shift operation).

This self-contained, qualification programme will be a required component for researchers intending to use the MNRC facilities and provides both independence for research as well as a tangible qualification. This programme will also form the template for a nuclear engineering programme.

The WNSA research reactors offer potential for training and education for students in nuclear reactor operation, neutron applications and computer simulation. Very few locations in the USA are equipped to offer qualifications in reactor operations and this provides a unique opportunity for the MNRC, as a fully fledged nuclear engineering programme develops. Using the radiography course as a guideline for development for reactor qualifications, training for reactor operations will be offered. This again offers a unique, tangible qualification additional to any degree component, as well as offering the additional advantage of providing operations staff for the MNRC.

Operations qualifications are a component of the more long term goal of provision of a nuclear engineering component at UC Davis, to be taught in collaboration with UC Berkeley.

The taught course will consist of three components, and will be taught over two academic quarters. The three components comprise an introduction to nuclear engineering, an introduction to radiological physics, and radiation detection and measurement. The taught course will have practical sessions run at the MNRC and will include the reactor operations training as an integral component.

Students will be able to use experimental facilities for the measurement of neutron interactions with matter, neutron flux and neutron detection. Computer simulation of reactor operations can be compared directly with measurements at the research reactors and will enable students to gain

experience in experimental design, planning and the prediction of nuclear events. In addition, neutron detection, instrumentation and imaging components will expand the training and education segment of the INIE programme to promote nuclear technology in environmental, earth, biological, biomedical and materials sciences. The programme will build upon the existing UC Berkeley programme and it will be integrated into existing programmes within the UC Davis College of Engineering.

4. CONCLUSIONS

A combined educational and research programme has been developed at the MNRC. A combination of the lack of an existing nuclear engineering programme on the Davis campus and the location of the MNRC facility (a 30 min drive) has led to the expedient of developing a regional alliance with other western US nuclear institutions, which have been faced with the related problems of a lack of students, staff and integrated educational programmes.

An alliance between these institutions, the WNSA, has been forged to combine the research and educational strength of each node facility. The purpose of this alliance is to provide a robust education and research programme for the next generation of nuclear engineers and applied nuclear scientists. These educational programmes require the dynamic interaction with the transfer of ideas, personnel, students and expertise between each node.

At the MNRC, the educational programme will offer educational programmes based on existing certification for radiography which will be extended to reactor operations. New technologies will be used to augment the taught components. These form the template for a more extensive nuclear engineering programme which is to be taught in combination with UC Berkeley.

10.5. TEACHING AND TRAINING AT THE RA-6 REACTOR

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1. INTRODUCTION

The RA-6 research reactor is owned and operated by the National Atomic Energy Commission (CNEA) of Argentina. It is located at the Bariloche Atomic Center, Bariloche (Argentina).

The reactor was entirely designed and built in Argentina and it has operated since October 1982. The main design goal was for a training reactor to support nuclear engineering education. Moreover, CNEA wanted to own a plant suitable for nuclear experiments to be performed for its Nuclear Engineering Department. Figure 1 shows the building of the RA-6 research reactor.

FIG. 1. RA-6 reactor building.

2. REACTOR DESCRIPTION AND IN-BUILT TRAINING FEATURES

The RA-6 is a pool type research reactor, cooled and moderated by light water, having a nominal power of 500 kW. Figure 2 shows the reactor block.

The reactor core configuration is variable inside an 8 × 10 grid and consists of about 28–32 fuel assemblies (FAs) with 90 wt% enriched uranium and a number of reflector elements. There are three different assemblies:

— Standard FA (horizontal cross-section 81 mm × 77 mm, height 750 mm) has 19 fuel plates loaded with about 8 g of ^{235}U each;
— Control assembly that is similar to the standard FA but 4 of the 19 fuel plates (numbers 2, 3 and 17, 18) were removed and replaced by two channels in which a control drive can move a fork type absorber up and down;
— Reflector elements from graphite canned with an aluminium box having the dimensions of an FA.

The absorbers of the control rods are made from cadmium cladded with stainless steel. Normally, the core configurations house five control assemblies, four in the central area and one in the periphery. Their control drive mechanisms are installed on the top of the pool at the mechanisms' bridge.

FIG. 2. RA-6 reactor block.

The open pool itself is a stainless steel tank of 2.4 m in diameter and 10 m in depth.

Two kinds of irradiation devices can be used to irradiate different types of specimens or samples. For long period irradiation, hermetic aluminium cans are normally used and positioned in special boxes inside the core. A pneumatic irradiation facility is installed inside the core, which can serve for the short time irradiation of samples. The plastic rabbit capsule can be easily inserted and removed during reactor operation.

Fifteen years ago, an external graphite thermal column and an internal one were positioned in one lateral core face, but in 1997 they were replaced by the internal filter and the external port of a BNCT facility (see Section 5.6). Additionally, there have always been five beam tubes, two of which reach up to the core border and the other three end at the external pool liner such that special devices are needed to reach the core border.

As it was, one of the design goals to maximize the available teaching and training capabilities of this small plant, the following special characteristic features were established.

The instrumentation and the logic system for both, the reactor protection system (RPS) and the logic control system (LCS) are overdesigned and overequipped. These systems are more complex than would be necessary for this small reactor, in order to be a versatile tool to familiarize students and trainees with reactors and to get practice with the systems. Those systems include redundancy and diversity for all nuclear and conventional parameters. The startup channels comprise three fission chambers applying 'two out of three' logic for high neutron flux and for core power gradient. The same applies for the three compensated ionization chambers which serve as power channels. The startup and the power system have overlapped for more than seven decades. This implementation of the control logic exists as well for the water flow rate, the inlet and outlet core temperatures, the underpressure of the reactor confinement, the radiation level at the top of the reactor pool, etc. The air dissipated from the reactor building to the environment passes through charcoal and absolute filters. Another arrangement of such filters cleans the air inside the reactor hall when it is isolated under accident conditions.

A training control room provides all the information about the parameters of the plant. It is permitted to move one control rod up and down during experiments supervised from that training control room. This latter action is permitted after an authorization given from the main control room 'by hardware', i.e. by a key which enables turning on a hard-wired interlock logic. Only smooth reactivity changes are permitted, otherwise the main control room regains control of the reactor.

3. TEACHING AND TRAINING

The research reactor has been used intensively in the nuclear engineering education and postgraduate studies of the Balseiro Institute in terms of courses at all grades as well as Master's Degree and PhD theses over the last 22 years.

During the phase subsequent to the basic studies at the university, the RA-6 is used for the following courses: reactor physics, radiation protection, activation analysis, radiation physics and dosimetry, nuclear instrumentation and reactor control, nuclear measurements, and plant maintenance. Within these courses, experiments are performed, such as:

— Critical experiments;
— Determination of the critical control rod position;
— Neutron flux profile measurements;
— Neutron spectrum unfolding;
— Determination of thermal power;
— Evaluation of core reactivity due to Xe poisoning;
— Determination of temperature and void reactivity coefficients;
— Control rod calibration;
— Calibration of nuclear and conventional detectors.

During the ninth semester of the education of a nuclear engineer, the students perform one supervised project to improve one of the existing research devices or systems or develop a new one. The general idea of each project is to implement all the steps necessary when introducing a device or a system into the reactor, such as design, construction, safety analysis, qualification, assembling and mounting, and cold and hot test performance, as well as quality assurance. Examples of the systems or devices that had been modified following this approach are:

— ^{16}N power control system;
— Neutron radiography system;
— Analysing system for prompt gamma;
— Neutron transmutation doping system, which was fully developed by students as a teaching tool;
— Failed fuel assembly detection system based on the detection of delayed neutrons in the primary cooling loop.

The reactor provides an important support for courses at the postgraduate level of Master's Degree in medical physics, for specialists in

nuclear energy technological applications and for the IAEA–CNEA radiation protection course.

The RA-6 and its courses have also been involved in training teams from the NUR multipurpose reactor (Algeria) and the MPRR (Egypt), which were commissioned during the last decade, and personnel involved in the development and operation of those plants. Moreover, operators of the Argentine nuclear power plants Atucha I and Embalse received specific training in reactor physics at the RA-6 and a large quantity of professionals from all over the world were trained at the plant, supported by IAEA fellowships.

The relationship between teaching by the reactor group and research at the RA-6 can be highlighted by the fact that several engineers/scientists used the RA-6 for their PhD theses in recent years (Refs [1–3]) and several are carrying on.

4. FACILITY DEVELOPMENT

To further support the role of the RA-6 as a plant comprehensively providing facilities for training and education and for experimental work, the irradiation facilities initially included in the reactor design were improved, focusing on specific interests. The appropriate personnel involvement together with student's cooperation allowed different grades of development and improvement to be achieved. Facilities involved in that process were:

— Neutron Activation Analysis Laboratory (LAAN);
— Boron neutron capture therapy (BNCT) facility (Refs [4, 5]);
— Neutron radiography facility, which was designed as an academic tool mainly for teaching and training purposes, but also for industrial services;
— Prompt gamma neutron activation analysis (PGNAA), designed for academic purposes and being upgraded in the framework of the current IAEA project 'New Applications on PGNAA'.

The RA-6 will remain a plant capable of providing excellent capabilities and facilities for education and training.

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