IAEA

Atoms for Peace: The First Half Century
1957–2007

International Conference on Research Reactors

Safe Management and Effective Utilization

Sydney
5–9 November 2007

Organized by the
International Atomic Energy Agency (IAEA)

Hosted by the
Government of Australia

through the
Australian Radiation Protection and Nuclear Safety Agency (ARPANSA)

Australian Government
Australian Radiation Protection and Nuclear Safety Agency

IAEA web site
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Where to with the Code of Conduct on the Safety of Research Reactors?

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The Code of Conduct on the Safety of Research Reactors was adopted by the Board of Governors of the IAEA and endorsed by the General Conference in 2004. The development of the Code took place over several years and followed letters to the Director General on research reactor safety from the International Nuclear Safety Advisory Group.

The Code is a non-binding international legal instrument designed to 'serve as guidance to States for, inter alia, the development and harmonization of policies, laws and regulations on the safety of research reactors.' It contains guidance on best practice directed to the State, to the regulatory body and to the operating organization.

As it is non-binding in nature, the Code does not itself include a mechanism for implementation based upon the process of ratification and the participation in formal review meetings that implements the Convention on Nuclear Safety and the Joint Convention. Nonetheless, processes for information exchange and a form of peer review are being considered by interested Member States. In 2006, the General Conference supported a proposal that periodic meetings be organised to discuss application of the Code in Member States. It looked forward to discussion of implementation of the Code including at this International Conference.

The non-binding status of the Code and the consequently more informal nature of mechanisms for implementation may be an advantage in allowing for a graded approach to the different types of research reactors, their status and the safety issues they face.

Feedback from the Regional Meetings on the Application of the Code of Conduct and Updating of the IAEA Programme on Research Reactors Safety

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The information related to the safety status of research reactors worldwide were collected by the IAEA through a number of channels. These include the regional meetings on application of the Code of Conduct on the Safety of Research Reactors, safety review missions, evaluation of the INSARR (Integrated Safety Assessment of Research Reactors) mission reports, and reports collected via the Incident Reporting System of Research Reactors (IRSRR). The analysis of this information helped identifying issues that present safety concern and need an increased attention.

Three regional meetings on application of the Code of Conduct were held by the IAEA in 2006 and 2007. The results of the self-assessments made by the participating Member States (MS) showed that there is a common need to:

- Improve the capabilities of regulatory bodies in the review and assessment of safety submittals;
- Increase the attention to commissioning of modifications and experiments. In particular, safety aspects of core conversion need to be taken into account in a more effective manner;
- Perform site re-evaluation for existing research reactors as a part of maintaining their safety assessment up to date;
- Develop comprehensive emergency plans and establish preparedness and response capabilities at the national level;
- Improve the safety culture in operating organizations and address human factors issues in all phases of research reactors lifetime;
- Improve the capability to prepare the safety documentation for decommissioning, and establish criteria for research reactors in extended shutdown and for release from regulatory control for decommissioned research reactors.

The analyses of the results of the IAEA safety review missions were useful in identifying common safety issues and trends, although the observations made and the issues raised by these missions are specific to the problems of each individual reactor. The results of the recent missions indicated out-of-date safety documentation (safety analysis report, operational limits and conditions, emergency plans, etc.), which need to be updated to conform the actual status of the facility. These missions showed also the need to:

- Improve the role and responsibilities of the safety committees in many operating organizations;
- Establish integrated management systems/quality assurance programmes;
- Put in place a clear strategy for the management of radioactive waste generated from research reactors.

The analysis of the incidents reported to the IRSRR showed that ageing of components is one of the most important root causes of the incidents, representing more than 50 %. Two-thirds of the operating research reactors are over 30 years old. Although many of these old reactors have been refurbished to comply with today's safety requirements, ageing of systems and components, including obsolescence of the instrumentation and control systems, is considered to be an important safety issue.

In addition to the safety issues mentioned above, future challenges include development of self-assessment capabilities for safety review in all MS and coordination with other international organizations having activities regarding research reactors safety. The IAEA programme on research reactors safety was developed taking into account these safety issues as well as the future challenges. It entails the following four projects:

1. Enhancing the safety of research reactors: This project is focusing mainly on promoting the application of the Code of Conduct and providing assistance for its effective implementation and completing the corpus of comprehensive safety standards that support the application of the Code. The project gives priority to the improvement of the regulatory supervision of research reactors and self-assessment capabilities of MS, conduct of safety review missions and assistance in implementation of the recommendations given by these missions. The other activities under this project include assisting in developing systematic ageing management programmes and decommissioning plans, increasing awareness of safety of experiments and modifications including core conversion projects, and establishing integrated management systems.

2. Monitoring and safety enhancement of research reactors under agreement: There are 35 research reactors in 27 MS under Project and Supply Agreement with the IAEA. In accordance with this project, safety review missions are conducted at these reactors to ensure their compliance with the requirements of the IAEA safety standards. This project will ensure the continuous operation of the follow-up system for monitoring the safety of these research reactors, which is based on collection and analysis of data on safety performance indicators and the dissemination of the operating experience.

3. Fostering international exchange of information on research reactors safety aspects: The objective of this project is to facilitate the improvement of operating practices and the exchange of information on events with lessons learned. Under this project, IAEA will continue operating the IRSRR, conducting reviews of incidents and assessment of Safety Review Mission reports to identify significant issues, and organizing meetings for the exchange of operating experience feedback. The other important activities include the development of a Research Reactor Information Network (RRIN) and the conduct of Coordinated Research Projects (CRPs) on subjects related to research reactors safety.

4. Assisting on safety aspects related to protection against sabotage for research reactors: This project aims at enhancing the awareness and understanding the synergy between safety and security and improving protection of research reactors against sabotage, including development of the regulatory authority's capabilities on methodologies for assessment and improvement of the synergy between safety and security.

In the final paper, the safety status of research reactors including common safety issues and trends, feedback from the regional meetings on the application of the Code of Conduct, and details of the IAEA programmes on research reactors safety will be presented along with discussions.

Effectiveness of Safety Regulation at Russian Research Reactors in Compliance with International Practice

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Russian research reactors base was created in 50-80th years of XX century to develop fundamental and applied sciences in the sphere of physics, power engineering, and science of materials, biology and medicine.

After the USSR collapse the changing of Russian economy was attended with decreasing of state financial support for safety ensuring and development of nuclear research facilities (NRF)[1] and reduction of their amount. Nevertheless the Russian fleet of NRF remains the largest in the world and posed more than 20% of worldwide quantity.

Nowadays the Russian atomic industry plans to start commissioning of new units of nuclear power plants on the basis of evolutionary designs and innovating research of power reactors, implementation of "small" and regional nuclear power facilities, problem-solving of closed fuel cycle, spend fuel management, and utilization of radioactive wastes. The high flux beam research reactor is being in commissioning to develop fundamental neutron investigation. Russia supports the conversion of the research reactors situated in development countries to reduced enrichment fuel. Thereupon the lay down development of nuclear science and technology should demand efficient utilization and modernization of Russian research reactors multitude.

Russian state regulatory body for nuclear and radiation safety has to be ready to internal and external challenges and its influence shall become stronger on the basis of priorities of safety in problem-solving of the use of atomic energy.

There were a few stages of improvement and harmonization of the Russian nuclear regulatory system on the basis of IAEA safety standards and transferred Western European safety principles and practice.

Rostechnadzor carries out major functions of the regulatory body in environment, different sphere of industry and nuclear activity.

The nuclear and radiation safety regulation in Russia is based on: development of the legislative framework including international agreements, federal laws in sphere of the use of atomic energy, national safety standards and regulations of executive bodies; realization of the licensing system of activities in sphere of the use of atomic energy and system of permissions for personnel; fulfillment of conformance evaluation of equipment and systems;

1 NRF - will be interpreted structures and complexes with civil research nuclear reactors, critical and subcritical nuclear assembles, which have been designed for utilization of neutrons and ionizing radiation for research purposes.

improvement of inspection programs and promotion of effectiveness of enforcement measures.

The legislative and regulatory infrastructure in the sphere of nuclear and radiation safety of NRF in accordance with fundamental safety principles and regulatory approach of IAEA especially under the Code of Conduct on the Safety of Research Reactors is considered.

The information on current status of Russian NRF and their state of nuclear and radiation safety is provided. The results of NRF licensing are reviewed as in-process improvement of safety during lifetime of the facility commensurate with grade of facility hazard and nuclear activity.

The qualitative indicators of nuclear regulatory efficiency are based on model of quality management being developed by international organizations.

The quantitative performance indicators of safety regulation at NRF are appraised in framework of methodology based on final indicators of achievement the plan targets of the regulatory body.

With regard to analysis result of current state of safety at Russian NRF the issues of further improvement and harmonization of regulatory system and enhancement of safety at NRF are posed in view of regulatory approach of IAEA and advanced practices of Western European countries.

Some Operational Aspects of BRR's Management System with respect to the Code of Conduct

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The Code of Conduct (CoC) provides a basis for the self-assessment of all aspects of management practice. In line with this idea the evaluation of the BRR's (Budapest Research Reactor) past 15-year management practice with respect to the CoC has resulted in good feedback, which has supported the applied practices and also highlighted deficiencies or weaknesses. The period of self-assessment was an obvious choice as it encompasses the time since the full-scale reconstruction of the reactor (which was completed in 1992); that, more over, represents a sufficient amount of time for drawing useful conclusions, summarising the experiences, and showing trends. In this paper, from the perspective of operation organization, some aspects of management practice are highlighted. These include: operation and utilization issues; conformity and safety reviews; and safety culture focusing on human factors. The paper also highlights how this management practice relates to and supports nuclear safety and demonstrates this safety to the public. It is hoped that the highlighted aspects of management system described here may serve as a general methodology for the RR community.

Operation and utilization practice. The BRR has adopted a range of management practices since the time of first criticality in 1959. The development of this practice was influenced in a number of ways. Among others self-improvement methodology, regulatory requirements and the sense of practicality also played a significant role. One of the practical matters concerned the operation and utilization practice that arose during the two-year period following the political changes in Hungary in 1990. This was a period of uncertainty during which it was essential to demonstrate the necessity of the BRR. For this reason a consortium, the Budapest Neutron Centre (BNC), was founded by four academic institutes to coordinate the reactor utilization, but first of all help to win public support for reactor start-up. Following the reactor start up, as the BNC put into operation the facilities around the reactor and started to manage the utilization strategy, it became obvious that the BNC could effectively represent the user interests, thereby leaving the reactor management to focus on safe reactor operation. Now this management system is highlighted as being one of the best operational practices. [1].

Conformity and safety reviews. During the last 15 years, two conformity reviews (CRs) and one periodic safety review (PSR) have been conducted at the BRR. The CRs were passed following the development of Nuclear Safety Regulations (NSRs) in 1997 and then in 2005. The CRs essentially consisted of a number of comparison assessments, where the conformity of the design requirements and the implementation of operation requirements prescribed in the NSRs were reviewed and evaluated on the existing reactor structures and the applied operation practices including the status survey of mandatory documentation. Regarding the PSR, the Hungarian legal system obliges operators to prepare a safety review every decade. As the BRR's operation licence was issued in 1993 the operating organization was obliged to conduct a PSR in 2002-2003, which ensured a complex overall review of the BRR, treating the reactor as a complex system whilst taking into account its service life. The general philosophy of the PSR was to assess the condition of the reactor structures with respect to the 10-year operation record, and evaluate all operational experiences including event records. [2].

Safety culture focusing on human factors. Safety has been given the highest priority at the reactor since the time of the first criticality. This safety-committed approach formed a deep-rooted nuclear culture at the BRR that has been inherited down through the generations of employees. As it is essential, the importance of safety is understood both by the top reactor management and by all staff involved in the operation organization. During the PSR in the frame of examining human factors, thorough and painstaking reviews were performed involving all safety issues, including all organization and administrative factors relating to safety. The review statements certified that the 10-year service life of the reactor was safe, there were no violations of the OLCs, and that the operation and maintenance practices met with both local and Licensing Authority's regulations. However due to some safety increasing measures following the PSR, at present the accident conditions from the perspective of human factors have already been analysed in the final SAR. A new version of the QA program was also issued which classifies the human function as a safety critical barrier and stipulates the expected safety policy. That is to say each staff member is committed to implementing a policy of safety awareness, which should be based on perception and prevention.

Demonstrate the safety. It has been obvious since the PSR that there is an increasing demand by society for a nuclear facility not only to be safe, but also to be seen to be safe. Thus the BRR's operating organization continuously strives to demonstrate the safety. There are many traditional (regular reporting, open door policy, etc.) and modern practices (internet technologies) that ensure near continuous access to the reactor indicators and events. On the basis of general experience in this field it can be stated that to meet the demand (demonstrate the safety) the common approach of any public information system is to increase the transparency and traceability of the activities. The public information must not only contain the information necessary to justify the conformity and demonstrate the safety, but must do so in an easily accessible manner. Based on feedback from the public, the most important considerations are that the information be authentic, coherent and comparable with previous years. It must also clearly report unscheduled events and provide comparison to well-known pointers. To meet this public demand one of the best comparisons to make is to demonstrate conformity to operation regulations and records. Another strong comparison to make is between the safety indicators of the reactor and the appropriate aspects of the IAEA's standards and recommendations, and (above all) the IAEA's 'Code of Conduct on the Safety of Research Reactors'.

In summarizing the general management practice with respect to the CoC two important statements can be made. Firstly: the CoC does not contradict the everyday practices (including the legislative and safety standards and regulation system). Secondly: the CoC clarifies the duties and responsibilities of all, be they organizations, regulatory bodies, or individuals (regardless of their position relative to the management hierarchy). Hence, the CoC acts as a compass that harmonizes and directs the safety approach taken at all levels, from the top to the implementation. It may be said therefore that it shapes the unity of content and form of a RR's safety.

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[2] S. Tőzsér, J. Gadó, K. Késmárky, I. Vidovszky: Periodic Safety Review of the Budapest Research Reactor. International Conference on Topical Issues in Nuclear Installation Safety: Continuous Improvement of Nuclear Safety in a Changing World (IAEA-CN-120/4), pp. 186-190. Beijing, China, 18–22 October 2004.

Session 3: Sustainable Utilization and Strategies

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Utilization Programmes For Low And Medium Flux Research Reactors

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We consider various major utilization programmes that may be pursued at any research reactor and discuss relevance to low flux (10^{12} neutrons/cm^2/s) and medium flux reactors up to $1x10^{14}$ neutrons/cm^2/s. Low-flux reactors inherently have low radiation field and much simplicity, and therefore, despite their limitations of flux, enable greater flexibility in designing new applications and testing of new ideas. These could generate valuable experience that could form the basis for advanced applications at higher flux reactors.

1. Neutron beam research

Thermal neutrons have been used to investigate various nano-scale structural, dynamical and magnetic properties of materials of scientific interest and technological importance. A variety of specific techniques are used such as diffraction, small-angle scattering, reflectometry, neutron depolarization, and inelastic and quasi-elastic scattering. The main advantages of using neutrons over much brighter photon beams are that the former provide (i) a sharp contrast between isotopes or between neighbouring atoms in the periodic table, (ii) high energy resolution of sub-meV, (iii) investigation up to large wave-vector transfer needed for liquids and amorphous structures, (iv) intense magnetic probe, (v) characterization of bulk samples and (vi) particularly large cross-section of scattering from hydrogen atom. All the techniques may be fruitfully exploited at medium-flux facilities. However, experiments at low-flux facilities might be limited to using diffraction techniques on samples of relatively simple crystal and magnetic structures. The experimental results are usually of fundamental scientific interest, and are expected to lead to newer ideas and possible applications.

2. Neutron beam applications

These involve characterization of various materials for their properties at microscopic or mesoscopic length scale. Typical examples are (i) residual stress analysis; (ii) surface and interface characterization including corrosion in magnetic and nonmagnetic multilayers; (iii) hydrogen sensing including environment around the hydrogen atoms; (iv) size and shape of large molecular systems such as hydrocarbons, micells, polymers and proteins and (v) voids in bulk materials such as steel, ceramic, coal and cement. These applications, usually possible at medium-flux facilities, are expected to lead to improved materials and processes.

3. Neutron radiography

This technique is complementary to gamma-ray and x-ray radiography. Specifically large engineering systems can be characterized by neutron radiography due to bulk penetrability of thermal and epithermal neutrons and their sensitivity to hydrogen. Of particular interest are investigations of internal defects in thick samples. Special techniques, such as hydrogen-sensitive epithermal neutron radiography, could detect rather low concentration of hydrogen atoms. Real-time images may be observed by dynamic neutron radiography. These applications are well suited at both low and medium-flux reactors.

4. Radioisotope production

A large variety of radioisotopes with different half-lives for applications to industry, health care, education and research, and other societal benefits can be produced by neutron absorption in reactors. These applications are well suited at medium-flux facilities.

5. Medical applications such as boron neutron capture therapy (BNCT)

While such applications are indeed highly desirable, they require large infrastructure involving nuclear and medical professionals.

6. Industrial applications such as neutron transmutation doping.

Neutron irradiation can provide excellent uniformity of dopants by transmutation in large size ingots of silicon etc.

7. Neutron activation analysis (NAA) and prompt gamma neutron activation analysis (PGNAA)

These form major programmes at low and medium-flux reactors for trace-level detection of many elements in a variety of samples of geological, nuclear and other materials.

8. Research and development support to nuclear energy programme

Research reactors may be used for testing of fuel and other power reactor components, neutron irradiation and radiation damage studies.

9. Human resource development for nuclear energy programme

10. Education and training at university level in basic sciences and technology

Most of the above programmes have relevance to both low and medium flux reactors though the scales may differ. At low-flux facilities the focus would be more on NAA and radiography while crystal structure investigations and some isotope production are feasible.

Experiences at several medium-flux facilities show that preliminary and even complete research investigations are often possible. These could form the basis for advanced research at high flux facilities.

In India a national facility for neutron beam research is operated at the research reactor Dhruva (10^{14} neutrons/cm^2/s). It includes single-crystal and powder diffractometers, a polarization analysis spectrometer, inelastic and quasi-elastic scattering spectrometers in the reactor hall, and small-angle scattering instruments and a polarized neutron reflectometer in the neutron-guide laboratory. A diffractometer for residual-stress measurements is being built. In addition a neutron radiography facility and a detector development laboratory are located at APSARA reactor (10^{12} neutrons/cm^2/s). All the instruments including the detectors and electronics have been developed within India. The National facility is utilized in collaboration with various universities and other institutions.

Various applications at the reactor facilities in India include neutron beam research and applications; neutron radiography; neutron activation analysis, support to nuclear energy programme, radioisotope production; as well as education and training, manpower development and teaching.

A Corrosion Monitoring Programme for Research Reactor Spent Fuel Basins.

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The main reason for conducting a corrosion surveillance or corrosion monitoring programme (CMP) at a RR facility or at a spent fuel wet-storage facility is to evaluate the effect of the prevailing water parameters at either facility on the corrosion of the spent fuel cladding and/or of other structural materials. A programme of this nature gives an insight into the extent of corrosion of the metallic materials. It is well known that good quality water is essential to prevent corrosion in a spent fuel basin. However, certain water parameters like conductivity, chloride ion content and some other ions, in quantities well below levels of concern, have a synergistic effect on the pitting corrosion behavior of aluminium alloys. Hence, maintenance of water parameters within specified limits is not reason enough for complacency about corrosion of fuel cladding. A well-planned and executed CMP would give the RR or spent fuel basin manager an insight into the state of fuel cladding and/or metallic structural materials in terms of corrosion. A CMP involves the exposure of test coupons to the RR or basin water for a pre-determined period followed by its evaluation to detect for corrosion. The CMP also involves the determination of water parameters at periodic intervals. Thus, the RR or basin manager is informed of any transients in water parameters, (which often go unperceived in the absence of a CMP) and its effects, if any, on the corrosion of the coupons, and consequently on that of the fuel cladding and other structural materials.

A CMP involves 3 stages, the planning stage, the execution stage and the action stage.

The **planning stage** includes: (1) listing metallic materials that are exposed to the RR or spent fuel basin water; (2) specifying their composition, microstructure, the heat treated condition and surface condition; (3) selecting the materials for the programme; (4) specifying the duration of the programme, if any; (5) specifying the frequency of corrosion monitoring; (6) specifying the frequency for monitoring water parameters; (7) specifying the location within the RR or spent fuel basin to place the coupons; (8) determining availability of sufficient materials in the desired states, a manufacturer or supplier adequately equipped to manufacture the test coupons and racks, all accessories required to identify, introduce, follow-up and withdraw the rack of test coupons; (9) determining laboratory facilities for water analysis, sediment analysis, test coupon evaluation; (10) specifying dimensions of the coupon, the insulating separator and the rack; (11) specifying configuration of the coupons with respect to nature of corrosion – crevice and bimetallic; (12) specifying if settled solids need to be evaluated in terms of quantity and composition, and if so, aspects mentioned in sub-items 7-9.

The **execution stage** involves all actions related to preparing the test coupons and racks, conducting the corrosion programme and evaluation of the coupons. A test protocol was elaborated by the authors during the IAEA sponsored Coordinated Research Project (CRP) on ‘Corrosion of Research reactor Aluminium Clad Spent Fuel in Water’ containing all details relate to the execution stage.

The **action stage** involves: (1) correlation of the results of coupon corrosion evaluation and the water parameters; (2) correlation of the results of coupon corrosion with that of the spent fuel cladding and/or other structural components; (3) taking of appropriate actions, if necessary, to alter water parameters, reduce extent of settled solids and verify fuel cladding surface integrity.

Guidelines to be followed during the three stages of a corrosion surveillance programme for RR spent fuel basins will be presented and discussed.

THE IEA-R1 RESEARCH REACTOR: 50 YEARS OF OPERATING EXPERIENCE AND UTILIZATION FOR RESEARCH, TEACHING AND RADIOISOTOPES PRODUCTION

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ABSTRACT

This paper describes almost 50 years of operating experience and utilization of the IEA-R1 research reactor for research, teaching and radioisotopes production. The current and future program of upgrading the reactor is also described. IEA-R1 research reactor at the Instituto de Pesquisas Energéticas e Nucleares (IPEN), Sao Paulo, Brazil is the largest power research reactor in Brazil, with a maximum power rating of 5 MWth. It is being used for basic and applied research in the nuclear and neutron related sciences, for the production of radioisotopes for medical and industrial applications, and for providing services of neutron activation analysis, real time neutron radiography, and neutron transmutation doping of silicon. IEA-R1 is a swimming pool reactor, with light water as the coolant and moderator, and graphite and beryllium as reflectors. The reactor was commissioned on September 16, 1957 and achieved its first criticality. It is currently operating at 3.5 MWth with a 64-hour cycle per week. Originally charged with 93% enriched U-Al fuel elements it currently uses 20% enriched uranium (U_3O_8-Al and U_3Si_2-Al) fuel that is produced and fabricated at IPEN.

In the early sixties, IPEN produced ^{131}I, ^{32}P, ^{198}Au, ^{24}Na, ^{35}S, ^{51}Cr and labeled compounds for medical use. In the year 1980, production of ^{99m}Tc generator kits from the fission ^{99}Mo imported from Canada was started. This production is continuously increasing, with the current rate of about 16,000 Ci of $^{99m}T_C$ per year. The ^{99m}Tc generator kits, with activities varying from 250 mCi to 2,000 mCi, are distributed to more than 260 hospitals and clinics in Brazil. Several radiopharmaceutical products based on ^{131}I , ^{32}P , ^{51}Cr and ^{153}Sm are also produced. These radioisotopes amount to more than US$15 million in sales per year for IPEN.

During the past few years, a concerted effort has been made in order to upgrade the reactor power to 5 MWth. One of the reasons for this decision was to produce ^{99}Mo at IPEN, thus minimizing the importation cost and reliance on only one or two world suppliers. The reactor cycle will be gradually increased to 120 hours per week continuous operation. The reactor modernization plans include the following during 2003-2009:

- Pool water treatment and purification system.
- Replacement of four reactor control elements.
- Replacement of one of the primary heat exchanger.

- Installation of a new core grid plate.
- Replacement of the reactor control panel.

The other infrastructure modernizations include the following:

- Modernization of the reactor fuel element fabrication facility in order to increase the production capacity to 15-18 U_3Si_2 type fuel elements per year.
- Optimization of radiochemical facilities to process ^{99}Mo using the gel process.
- Development of an effective project for spent fuel management and storage.

It is anticipated that these programs will assure the safe and sustainable operation of the IEA-R1 reactor for several more years, to produce important primary radioisotopes ^{99}Mo, ^{125}I, ^{131}I, ^{153}Sm and ^{192}Ir. The production of these isotopes will result in less dependence on the world supply, and reduce importation costs. Routine radiation monitoring is carried out at 25 different locations in the reactor building. This regular monitoring practice has helped in maintaining the equivalent dose of reactor workers below the established limits. Following the guidelines of IAEA an ALARA program is being implanted at the reactor and associated laboratories, which handle radioactive materials.

In order to achieve the goals of modernization and reactor aging management, IPEN undertook the following tasks under a technical cooperation program funded by the International Atomic Energy Agency (IAEA) and Brazilian government agencies.

- Replacement of some electrical and refrigeration systems; radiometric analysis system for water and air samples; reactor control instrumentation; radiation monitoring equipment.
- Neutron flux mapping facility using self-powered neutron detectors (SPNDs).
- Improved computational facility for neutronic calculations.
- A highly radioactive sample handling facility.
- Training of personnel engaged in electrical and mechanical maintenance, water chemistry, and irradiation services.
- Installation of a continuous vibration monitoring system for rotating machinery.

Currently, all aspects of dealing with fuel element fabrication, fuel transportation, isotope processing, and spent fuel storage are handled by IPEN at the site. Spent fuel assemblies are visually inspected routinely using underwater cameras. Seeping analysis is performed if there is an indication of fission product release in the pool water. IPEN has an on-going project with IAEA on spent fuel management. The reactor modernization program is slated for completion by 2009.

IAEA-CN-156/U-43

THE UTILIZATION OF 10Mw RESEARCH REACTOR IN TASHKENT

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ABSTRACT

We present the short review of basic and applied research as well as data on the mass production of reactor isotopes with use of 10Mw water-water research reactor of the Institute of Nuclear Physics of Uzbekistan Academy of Sciences in Tashkent. Despite the relatively long time of operation (since 1959) the several projects on the modernization of machine have been done. The reactor operates more than 5000 hours a year and serves also for elemental analysis (neutron activation), radiochemistry, radiation hardness and fission products studies as well as for changing the properties of optical and semiconductor materials. Until 1997 reactor was operating with use of highly enriched fuel (90% of enrichment) and starting from the middle of 1997 it has been converted to use 36% enrichment fuel. In the second half of 2007 the preparatory works on the full conversion to the 19.7% enrichment fuel should be completed. We also present the results of our experience in sending highly enriched spent fuel back to country-origin (Russia) for first time in last 16 years.

The external one-arm spectrometer of secondary fission products made it possible a study of properties (the charge, angular and momentum distributions) of fission products which revealed quite interesting features inconsistent with some standard models of fusion. The special method has been developed to carry out elemental analysis which once applied, for example, to pure metals allows to determine impurities with concentration up to 10^{-10} % (for almost 60 elements simultaneously). The reactor is also using for changing the properties of some materials like semiconductors, ceramics and natural crystals bringing their quality to market demands.

The main activity of our reactor is the mass production of isotopes for medical and scientific needs. In particular, the isotopes like Tc-99m (from irradiation of Mo wires), P-32, P-33, I-125, I-131, S-35, Au-186, Sr-89, Fe-55 and some others are producing at the level of hundreds and thousands Curies and, in addition, ready-to-use products like Tc99m generators for medical clinics and labeled compounds for research are delivering to customers in many countries. The high chemical purity and specific activity are making those products competitive at international market.

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The 30 kW Research Reactor Facility in Ghana: Past, Present and Future Programmes

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The Ghana Research Reactor-1 (GHARR-1) is a small, simple, reliable and safe reactor design and constructed by China Institute of Atomic Energy (CIAE). GHARR-1 adopts the pool-tank structure and employs highly enriched uranium as fuel, light water as moderator and coolant, metal beryllium as reflectors. The reactor is cooled by natural convention. The rated maximum thermal power of GHARR-1 is 30 kW; the corresponding neutron flux is $1.0x10^{12}$ $cm^{-2}s^{-1}$. The refueling mode of the reactor is to totally change the old core with a new one, the lifetime being more than ten years.

Since the commencement of operation of the low-flux miniature neutron source reactor (MNSR) in 1995, a significant number of research and development in the field of neutron activation analysis have taken place. During its 12 years of operation, after the first criticality, the reactor has been used as a neutron source for research, teaching and training to support several graduate and post graduate careers for students from universities in Ghana and the West African sub-region. Owing to the stable flux of the reactor and rapid proliferation in utilization, several analytical techniques have been developed. As a national neutron source reactor facility, Ghana's MNSR also known as GHARR-1 is now successfully utilized in various areas of neutron activation analysis (NAA), teaching, research and training.

The GHARR-1 application in neutron activation analysis included: (i) Food analysis; (ii) Heavy metals determination in environmental samples; (iii) Determination of major, minor and trace elements in geological samples; (iv) And mineral prospecting among others. The educational programmes in place at the center are teaching and learning in nuclear engineering, nuclear physics, nuclear and radiochemistry and other related fields.

The paper will focus on the past and current status of GHARR-1 with respect to utilization and management and future programmes to enhance its uses in the fields of teaching, research and training. The present and future collaboration programmes with national institutions and technical cooperation among some AFRA member states shall be given.

IAEA-CN-156/U-18

The Halden Reactor Project: Experience gained in international research

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The Norwegian government has with the successful completion of the June 2006 "Symposium and Technical Workshop on Minimisation of Highly Enriched Uranium in the Civilian Nuclear Sector" [1] confirmed its engagement for minimizing the risk related to the continued use of fissile material in general and HEU in particular. One suggested concept in reducing risk associated with fissile material required for research is the concentration of research efforts in shared facilities. The increased global demand for nuclear energy also points towards the need for first class research facilities. Pooling resources makes sense as a way to maximise research efforts. But how should a shared facility work and function? The OECD-Halden Reactor Project [2] is a good example of operation of a shared research facility which has been in operation for nearly 50 years. An overview of how this cooperation has worked and how the research facility has grown and developed will be presented in this paper.

First a look at the historical perspective. The initial idea in the 50's for the building of the Halden Reactor was that Norway wanted experience in building nuclear reactors for use in ship propulsion. It soon became clear that Norway could not afford this project alone, even the close cooperation between Norway and the Netherlands was not enough to complete the project. Finally in 1958 when the reactor was opened the project had become international with the Institute for Atom Energy (Now called the Institute for Energy Technology) being the host.

FIG. 1. Cutout view of Halden Reactor.

FIG. 2.Control room simulation and observation gallery, MTO-Laboratories

Then as now the research was organised in 3 year programmes. Countries participating provide financial support, are part of the steering committee which decides on the content of the research. Visiting research scientists are also sent to Halden to gain experience and to participate in the operation of the experiments. Which and how many countries are members may change. There is no obligation to re-join after a 3 year period has finished. This flexibility is probably one of the main reasons that this organisation form has remained unaltered in now almost 50 years. It has also meant that the research program is constantly changing so as to be as up to date as possible.

Another factor contributing to the projects success is the design of the reactor. The Halden reactor is a boiling heavy water reactor typically operated at 235 °C. The neutron flux level is typical of commercial power reactors. What makes the Halden Reactor unique is a combination of instrumentation, availability and versatility.

With two operational periods each year a typical availability of 50% is achieved. This combined with many experimental positions more than compensates for the moderate flux level. In-pile instrumentation has been under development since the start of the reactor in 1958. To date almost all physical parameters can be measured; fuel temperature, stack elongation; cladding outer temperature, diameter; rod internal pressure, to name but a few.

The versatility of the reactor is also important; several different conditions can be simulated simultaneously: BWR, PWR, PHWR. All with their own unique water chemistry, temperature and pressure. Power ramps can be performed for individual fuel channels without affecting the whole reactor; and recently a LOCA test was performed in realistic conditions.

In the beginning of the 80's it was recognised that reactor research should not be solely concentrated on research reactors. Other aspects of commercial reactor operation were of equal importance, such as computer technology, human factors and psychology. This is where the dynamic nature of the projects organisation proved its usefulness, with an increased emphasis being placed on these new areas in the 1982-84 three year programme. To date these new areas in safety, Man-Technology-Organisation (MTO) represent 40% of the projects activities and include areas such as: operator procedures, computerised surveillance, visualisation techniques, use of virtual reality, advanced communications, software reliability, and human reliability in power station environments.

In addition to providing training, experience and countless amounts of experimental data for the projects member countries, the Halden Project has also been extremely beneficial for the hosting country Norway. Despite the fact that the Norwegian nuclear industry did not materialise there have been several benefits for conventional Norwegian industry (in particularly petroleum) and has lead to several commercialisation successes. The most recent being in electronic energy trading and cable fault detection. In addition the Institute provides commercial services based on the reactor and associated technology with turnover comparable to the research project.

In short we believe that the Halden Project is a good example of how to operate an international research project. The key ingredients being flexible project membership, active participation of project members, and a constantly evolving research programme. The research data obtained at Halden is vital for the safe operation of many nuclear power plants as well as helping the industry gain economic savings. The hosting of the Halden Project has also been beneficial for Norway and the Institute.

[1] "Minimization of Highly Enriched Uranium (HEU) in the Civilian Nuclear Sector", Oslo 2006, http://www.nrpa.no/symposium/

[2] "The Halden Reactor Project", http://www.ife.no/hrp/index_html?set_language=en&cl=en

The utilization of research reactors and related facilities for the supporting of innovative power reactor and related fuel cycle

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Abstract

Over 30% of electricity in Japan was produced by the existing power reactor "LWR". Nuclear power has an advantage to reduce the emission of greenhouse gas "CO_2". In near future, the LWR are utilized to produce the electricity and it is planned to extend the lifetime to go up the economic efficiency and availability. To meet this technical request, the experimental resolution for the ageing degradation of materials under neutron environment and upgrading the fuel performance are so important issues.

From the viewpoint of energy resource and the reduction of environmental burdens, the innovative nuclear reactor systems are investigated in the worldwide, such as generation IV systems and INPRO (International Project on Innovative Nuclear Reactors and Fuel Cycles). In Japan, the demonstration fast breeder reactor is expected to start the operation around 2025. As a key for the realization of the fast breeder reactor system, it is important to improve the fuel performance and to develop the new materials. For this purpose, it is necessary to investigate the fuel behavior to make clear the fuel stability limit and safety margins.

The role of research reactor is to figure out the degradation of materials and fuel performance, include the development of new reactor materials. The important thing for the research of fuel performance using the research reactor should be collocated with the fuel fabrication facility for the irradiation test, post-irradiation examination facility, and so on.

In Oarai Research and Development Center, there are some research reactors and post-irradiation examination facilities to investigate the fuels and materials for innovative power reactor, such as Fast Reactor, High Temperature Reactor. That is, the research reactors as the thermal neutron irradiation field are Japan Material Test Reactor (JMTR) and High Temperature Test Reactor (HTTR). The Japan Experimental Fast Reactor "JOYO" is utilized as the fast neutron irradiation field.

The JMTR has been used as the irradiation facility for the research the fuels and materials of the existing power reactor and the fundamental research of materials. The JOYO has been used for the research and development of the fuels and materials of the fast breeder reactor and the fundamental research of new materials development for the fusion reactor. The many type of irradiation devices to meet the objectives of irradiation experiments are provided include the instrumented vehicle for JMTR and JOYO.

In addition, there are four post-irradiation examination (PIE) facilities and a fuel specimen fabrication facility in Oarai Research and Development Center. Three PIE facilities are to treat the Plutonium fuels. And also it is possible to fabricate the fuel pellets remotely. Furthermore, the reassembling of irradiation vehicles in the hot cell to reload into the reactor is possible. From the view point of the sophisticated irradiation technology, how to contribute in the field of irradiation effects of reactor materials for fission and fusion reactor is under discussion.

Thus, it is possible not only to fabricate the fuel specimens, examine the fuels and materials after irradiation, but also to treat and storage the radioactive waste. Besides, the research center of universities for the irradiation behavior of materials is located in the same site. Hence, Oarai Research and Development Center is able to support the research of irradiation behavior for reactor materials and upgrading the reactor fuels, effectively and efficiently.

RACE-T Experimental Activities An overview of the subcritical measurements preliminary to the accelerator coupling experiment

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As it is well known, the concept of accelerator-driven systems (ADS) can provide a solution for the nuclear waste management issue by burning minor actinides. The subcritical nature of ADS makes them safer regarding prompt criticality accidents. However, the question of the reactivity control of ADS remains a main concern since their power is inversely proportional to their reactivity level.

The objective of the European Integrated Project EUROTRANS of the EURATOM 6th Framework Program is to bring answers to the high level nuclear waste transmutation in ADS. The EUROTRANS experimental activities have been joined into the ECATS domain, namely Experiment on the Coupling of an Accelerator, a spallation Target and a Sub-critical blanket.

In the period between February 2004 – February 2006, in view of the past TRADE project and the subsequent RACE experiment planned in the frame of the EUROTRANS Integrated Program of the 6th European Framework Program, different experimental activities were carried out in the 1 MW TRIGA reactor operated by ENEA in his Casaccia Research Center near Rome.

In that time the reactor operation was mainly in sub-critical conditions, with the exception of the critical reference core assessment at the beginning of the campaign where the maximum power was 50 W.

The critical reference configuration was characterized by spectrum index measurements and flux distribution as well.

Measurements of spectrum indices (ratio of various fission rates) in different core positions were performed using ^{235}U, ^{238}U and ^{237}Np miniature (diameter 1.5 mm) fission chambers for radial and axial measurements on locations situated on the main diagonal in the reference configuration, and a complementary measurement performed to check a dissymmetry effect on the radial profile. These measurements were realized with the use of a special fuel pin that will be described.

The determination of the neutron flux inside the TRIGA core at a critical configuration with a reactor power of 10 watts was performed with the well-known activation technique. A pneumatic Fast Rabbit device was settled down inside the reactor hall in order to send gold and indium samples into the reactor. Those samples were irradiated with and without cadmium covers for the assessment of axial and radial flux distributions for both thermal and epithermal neutrons. The cadmium ratios were also calculated. Results deduced from gold and indium sample measurements were compared. From the aluminum material of the irradiated shuttles, an estimate of the fast neutron flux was derived.

The RACE-T experiments allowed improving the knowledge of the experimental techniques for absolute reactivity calibration at either startup or shutdown phases of accelerator-driven systems. Various experimental techniques for assessing a subcritical level were inter-compared through three different subcritical configurations SC0, SC2 and SC3, about -0.5, -3 and -6 dollars, respectively. The area-ratio method based of the use of a pulsed neutron source appears as the most performing. When the reactivity estimate is expressed in dollar unit, the uncertainties obtained with the area-ratio method were less than 1% for any subcritical configuration. The sensitivity to measurement location was about slightly more than 1% and always less than 4%. It is noteworthy that the source jerk technique using a transient caused by the pulsed neutron source shutdown provides results in good agreement with those obtained from the area-ratio technique.

In the ADS projects, one of the most important aspects needed for approval of the demonstrator is the experimental verification of the simulation. The determination of the neutron spectrum is particularly interesting (i.e. neutron flux as a function of the neutron energy) for different configurations of the sub-critical device. As well known, the neutron flux in ADS consists of neutrons produced via spallation reactions in the target and fissions from the multiplying blanket. Unfortunately the neutron spectra cannot be measured using only one type of detector. To cover the complete energy range of the produced neutrons a new neutron detector concept based on Micromegas technology, widely used in particles or nuclear physics experiments. One of the main qualities needed for that detector is its high resistance to the radiation. A test with sealed prototype placed inside an empty rod of the reactor has been performed in-core in the RACE-T experimental campaign: the main outcomes will be illustrated.

In order to corroborate the outcomes from the previous MUSE experiment, and to improve our understanding about the experimental methodologies needed to monitor the subcritical level in ADS, a common calculation benchmark is activated, in the Frame of the IAEA-CRP *Analytical and Experimental Benchmark Analyses of Accelerator Driven Systems (ADS)*, aiming to evaluate the spatial/energy effects to be applied to some selected reactivity determination measurements obtained in the frame of the RACE-T experimental campaign. The rationale and the status of the benchmark activities will be illustrated.

Acknowledgements

The authors appreciate the efforts and support of all the scientists and institutions involved in EUROTRANS and the presented work, as well as the financial support of the European Commission through the contract FI6W-CT-2004-516520.

Session 4: Safety Management and Operational Safety

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Improvements in the Management of Safety in Research Reactor Operation through Appropriate Application of Selected Power Reactor Good Practices

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Research reactor managers are increasingly implementing improvements in their management of safety through the application of good practices originally developed as power reactor programs. This paper considers ways to select practices to emulate, effectively incorporate them into a research reactor program and evaluate their contribution to safety.

Relative to research reactors, power reactor programs look relatively homogeneous when considering source terms, stored energy, core power density, operating cycles, plant systems and staff sizes. They have potential hazard consequences that require effective safety management programs. Finally, power reactors generate a stream of revenue to fund these programs. The power reactor community has combined their resources with the homogeneity of their challenge to create impressive safety management tools, many of which can be effectively implemented in the research reactor community. However, not all programs can be effectively implemented in all research reactors.

A number of power reactor programs are analyzed in the paper with consideration of their effective implementation and potential contribution to research reactor.

Safety Management at NRG-Petten

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NRG (Nuclear Research and consultancy Group) develops knowledge, products and processes for safe applications of nuclear technology on behalf of energy, environment and health. NRG has acquired more than 45 years of experience in the nuclear field. Amongst others this includes experience in the operation of a wide range of nuclear facilities, namely:

- Operation of the 45 MW High Flux Reactor for materials testing, (medical) isotope production and BNCT;
- Operation of a 30 kW Argonaut Low Flux Reactor for training and biological applications;
- Operation of Hot Cells for materials testing and isotope separation;
- Overall responsibility for a Molybdenum Production Facility
- Operation of a Decontamination and Waste Treatment facility for both internal and external services.

Management of safety

As a part of NRG's mission statement safety has the highest priority. In respect of nuclear operations, safety requirements have an over-riding priority above of the production demands. To as-sure this mission statement is satisfied outstanding safety management is of paramount importance. Overall safety management is achieved through a combination of a safety-management system and the company safety culture.

At NRG, safety management is an integral part of operational practices, with the safety procedures and instructions fully incorporated into NRG's integrated management system. The basis for our safety culture are the shared values and shared basic assumptions which are formalized in our code of conduct. The visible characteristics of our safety management range from our Mission and Policy, through the skills and competences of our staff to the most visible characteristics the written procedures and instructions of our management system.

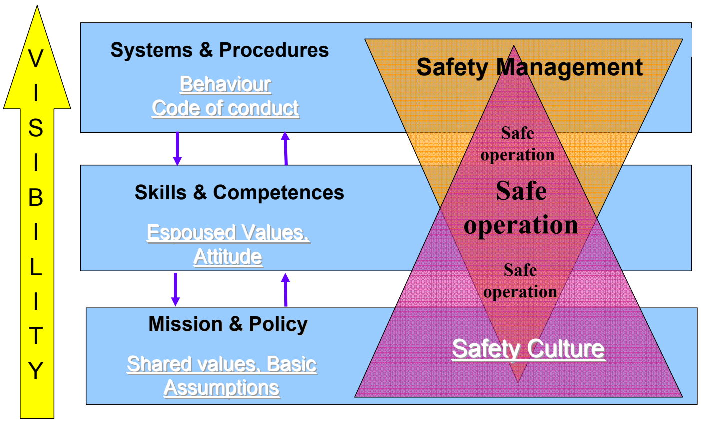

FIG. 1. Visibility of safety culture versus Safety management (based on Schein and Frischknecht)

The characteristics of a safety culture are less visible but are based on the shared values and basic assumptions of an organisation. The attitude of the staff is mostly based on the espoused values within an organisation. The visible aspects of the safety culture are the behaviour of the staff and the formal code of conduct of an organisation.

NRG's integrated management system

NRG developed a management system which satisfies NEN-ISO-9001:2000 as well as NEN-ISO-17025:2000. Where necessary our Management System has been further developed to fulfill additional requirements of the Dutch Nuclear Safety Requirements NVR-1.3, which are based on IAEA safety requirements DS 338 and the corresponding safety guides. These adaptations mostly concern issues of: management review, control of modifications of installations, operations and maintenance of the nuclear installations, control of experimental and testing programmes and registration, storage and transport of fissile and radioactive material. The ISO-14001 standard has been taken as the basis for our environmental management procedures, which are also integrated in our Management System. The structure of our Management System is presented in figure 2. The corporate procedures consist of flowcharts defining the tasks and responsibilities of all relevant processes and apply to all product groups. Where necessary independent control- or review activities and quality registrations are prescribed. Every product group has a set of specific procedures and instructions. These documents provide a more detailed description of the specific product group related processes such as detailed operational, maintenance, calibration and testing instructions. NRG's management system is based on the skills and competences of our staff.

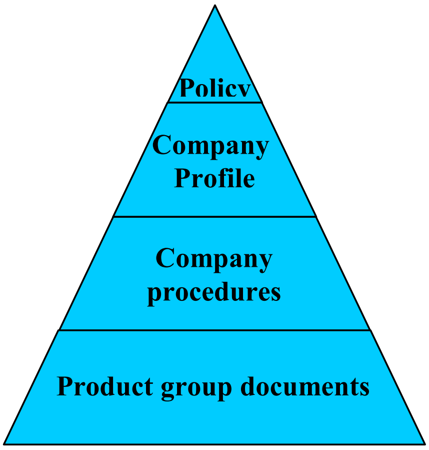

FIG 2. Structure of NRG's management system

NRG's organisation improvement programme

Since organisation culture aspects are essential layers of the defence in depth NRG launched in 2003 an extended organisation improvement programme. This programme has the objective to transforming NRG from a technocratic institute into a human oriented learning organisation. This programme started with a dedicated conscious management development training of the senior management, followed by conscious management development training for the second management layer and young potentials. As a part of the improvement programme NRG developed also a new vision and long-term business strategy, emphasising safety as key business element, to be shared and borne by all company staff. Individual responsibility, open communi-cation with mutual respect and trust, a learning attitude of management and staff, positive engagement and a blame free culture are the key elements which are also incorporated into our code of conduct. An open learning attitude is pro-actively being stimulated by top management encouraging an environment in which all safety issues and the behaviour of all staff are being discussed openly. This approach forms the basis for our safety culture in which in addition to our customers the public, environment and the competent authorities are seen as our stake-holders too. Through continuous improvement NRG strives to be an excellent organisation with an excellent safety culture.

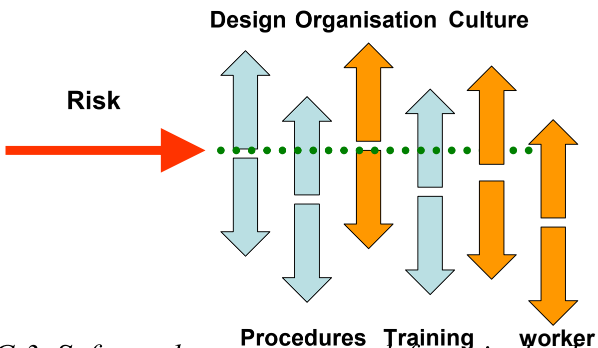

FIG 3. Safety culture aspects at defence in depth

Safety Management and Effective Utilization of Indian Research Reactors APSARA, CIRUS and DHRUVA

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Research Reactors Apsara, Cirus and Dhruva are located at Bhabha Atomic Research Centre, Mumbai, India. Apsara, a swimming pool type reactor with maximum design power of $1MW_{th}$ is in operation since 1956. Cirus ($40MW_{th}$) and Dhruva ($100MW_{th}$), both tank type reactors were commissioned in 1960 and 1985 respectively. These research reactors have played very important role in developing India's nuclear energy programme and in establishing the safety basis for the related activities. Due to diversity in their design, operating power levels and utilization aspects, a slightly flexible and graded approach is necessary for managing safety in these research reactors. While the major safety aspect for research reactors are similar to that of power reactors, special attention is necessary on account of several irradiations and experiments that are carried out in these facilities.

Several good operating practices like special work permits for maintenance activities, issuance of written valve slips for effecting valve status changes, special procedures for non-routine activities, pile irradiation requests for irradiation of samples, close attention to chemistry of fluid systems, prompt reporting of faults and incidents etc. have been practiced over the years in these reactors. Following good operating practices meticulously results in development of good operating culture which in turn provides right environment for developing safety culture. It is well known that training is essential for development of good operating culture. Strong emphasis is therefore laid on formal training and qualification of personnel in research reactors right from their inception. These practices led to development of a strong safety culture wherein all plant personnel are conscious about safety importance of their actions and are proactive about maintaining and enhancing safety. In fact, research reactors laid the foundation of a strong safety environment in reactor operations and this trickled down to the nuclear power plants also.

Track record of safety of research reactors in India has been excellent. Root cause analysis of every incident is carried out and appropriate corrective measures are implemented. Detailed surveillance and In Service Inspection programmes have been evolved based on specific reactor design and operating experience and are adhered to. Ageing of old reactors has been managed by systematic assessments and refurbishing actions. The refurbishing outage has been also utilized for making several safety upgrades to meet present day requirements, as in the case of refurbishment of the Cirus reactor. Safety improvements have been made on a continuing basis based on operating experience and new knowledge. At times, these improvements have gone beyond the requirements of design and safety analysis giving credence to the slogan **AHARA "Safety – As High As Reasonably Achievable".**

For ensuring continued safety during the operating life of the research reactors a well planned safety management system is in place. There exists a well defined hierarchical structure and line of communication among operating organization, regulatory agency, health and safety organization, maintenance and services organization and various safety committees / groups, quality groups, and experimenters, to facilitate smooth functioning of the operational activities of the research reactors at BARC. The paper would outline the safety management system practiced in these research reactors.

All the three research reactors have been well utilised for basic and applied research, neutron radiography, nuclear detectors testing, radioisotope production, material testing and human resource training and development. The National Facility for Neutron Beam Research (NFNBR) has been created at BARC during early nineties to cater to the needs of the Indian scientific community. Scientists from, universities and national laboratories also use these facilities in research reactors through collaborative research projects. Many of these collaborations are being supported by University Grant Commission - DAE Consortium for Scientific Research (UGC-DAE CSR), Board of Research in Nuclear Sciences (BRNS), and other agencies. Besides conventional uses, these research reactors have been utilised for conducting various engineering experiments also.

A refurbishment and upgradation plan has been drawn up for Apsara which has completed 50 years of service. The refurbishment would incorporate reactor design modifications to obtain enhanced neutron flux and a larger irradiation volume. Reactor core would be converted to a LEU Silicide fuel core. The paper would describe some important utilisation of research reactors Apsara, Cirus and Dhruva, modifications incorporated and up gradation proposed to enhance utilisation. Our unique experience in developing ISI programme for Dhruva and Cirus will also be covered in brief .

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Ageing Management of Pakistan Research Reactor-1 (PARR-1)

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Introduction to PARR-1

Pakistan Research Reactor-1 (PARR-1), a swimming pool, MTR type research reactor went critical on 21 December 1965 and attained full power of 5 MW on 22 June 1966 with 93 % Highly Enriched Uranium (HEU) fuel. The reactor was shut down in 1990 for core conversion to commercially available < 20 % Low Enriched Uranium (LEU) fuel. During the process of core conversion the reactor power was also upgraded to 10 MW. PARR-1 went critical with LEU fuel on October 31, 1991 and attained the upgraded power level of 9 MW on 07 May 1992. The reactor power was raised to 10 MW on 27 February 1998 after enhancing the primary flow rate.

Ageing Issues

The physical ageing of PARR-1, due to normal running of the plant and equipment, wear and tear, corrosion, vibration, stressing, thermal and mechanical fatigue and the general deterioration of the plant etc., has been manifested and dealt with almost from the beginning. The non physical ageing issues have been demonstrated in satisfying changing regulatory requirements, updating safety and administrative documentation, coping with technical obsolescence of facilities, and maintaining essential staff skills keeping in view the loss of safety knowledge that occurs with the loss of staff due to either retirement or organizational changes. Ageing issues are still there and need continuous attention.

Life Limiting Factors

Embedded primary piping from pool bottom to valve pit is seen as the life limiting factor for PARR-1. The pipe is embedded about 1.5 m deep in concrete and if this pipe becomes leaky due to corrosion, erosion or cracks, the replacement will not be possible and the reactor will have to be shut down permanently. Components like core support structure, grid plate and plenum are theoretically replaceable but, in reality, a replacement may be impracticable due to high doses and economic considerations.

Ageing So Far

The physical ageing issues tacked so far include pool civil repairs, heat exchangers descaling, renovation of instrumentation and control, core conversion and power upgradation during which a number of systems were renovated, new systems were installed and visual inspections of key components were made, and cooling tower replacement. The non physical ageing issues have been fulfilment of regulatory requirements resulting in updating the safety and administrative documentation e.g. introduction of a preventive maintenance programme, revision of operation manual, revision of safety analysis report, introduction of maintenance procedures, establishment of a formal quality assurance programme, and failure mode and effect analysis of the power supply system.

Further Ageing Issues

Ageing issues are still there and need continuous attention. An extensive ageing management programme is seen essential to cope with physical degradation and technological obsolescence of certain components and systems, and ensure the safe operation of key components by demonstrating their integrity. An ageing management programme shall primarily address replacement of lead thermal shield; replacement of fire alarm system; replacement of emergency generator # 2; instrumentation and control renovation; fabrication of new graphite reflectors; building civil repairs; replacement of transfer port control system; replacement of rabbit station control systems; inspection of welds; thickness inspection of heat exchanger shells, heat exchanger tubes, pool stainless steel lining, and primary and secondary system piping; visual inspections; and most important of all manpower ageing and loss of knowledge.

Problems

Certain problems are foreseen before taking up a full-fledged ageing management programme which include lack of experience, manpower shortage, unavailability of equipment, high doses on reactor components of major interest, financial constraints, and reactor under utilization.

The Safety Reassessment of Research Reactors in France

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In France, all the research reactors are subject by law to periodic safety reassessment which is necessary to maintain them in operation.

These safety reassessments follow strict guidelines in terms of methodology, requirements and rules, similar to the ones used for Nuclear Power Plants. Compulsory, they are done every ten years. The objectives of such reassessments are:

- to check that the facility is in conformity to its safety system of reference in use,
- to perform an inventory of the nuclear safety of the facility related to present rules.

This process, which is in fact based on two separate steps (conformity exam and safety reassessment), takes into account feedback experiment, obtained during the operation of the reactor and on other similar nuclear facilities. It allows us to identify the gap between the present situation and the requirements due to the current rules and to check the possibility of implementation.

At the end of these safety reassessments, refurbishment work is defined (seismic reinforcement, fire protection, command/control upgrade …) according to the life expectancy forecast for the facility. In this way, the safety level of the research reactor increases.

These Safety Reassessments can give also the opportunity:

- to increase the experimental capabilities of the reactors, the impact of such modification being included in the new Safety Analysis,
- to solve problems linked to the obsolescence of materials even if these materials are not linked to the safety level of the reactor.

A Safety Reassessment Report is examined by the French Nuclear Safety Authority and its technical supports (IRSN: Radioprotection and Nuclear Safety Institute). The French Nuclear Safety Authority:

- expresses its needs which represent a significant amount of studies and work,
- gives its formal approval to the modifications of the reactor,
- and, finally, gives its authorisation to continue the operation of the reactor.

The definitive Safety Report is then produced in order to take into account all the modifications of the facility.

The most sensitive points which appear on these safety reassessments in the French Research Reactors concern:

- the dimensioning of the confinement barriers,
- the treatment of internal risks (fire, flood…),
- the treatment of external risks (earthquake, airplane crash…),
- the incidental and accidental situation analysis and their radiological impact according to the new rules.

The French Atomic Energy Commission (CEA) operates ten research reactors covering the whole scope of use (neutrons beams reactor for ORPHEE, Zero Power Reactor for neutron physic studies for EOLE, MINERVE and MASURCA, safety test dedicated reactors such as CABRI and PHEBUS…).All these reactors are subject to the safety reassessment described above.

As an example, the safety reassessment of MASURCA in progress at the moment leads to:

- a complete refurbishment of the main systems of the facility (ventilation, electricity, command/control…),
- the realisation of important building work for seismic protection reinforcement of the nuclear material storage,
- the realisation of important work for reinforcement of fire protection.

Considering the CEA strategy on its research reactors which consists of performing these safety reassessments for the large majority of its research reactors, CEA owns a scope of modern and efficient tools with compliance to the latest safety rules.

For the Material Testing Reactors, CEA strategy consists of an improvement of the OSIRIS reactor safety level for a few more years of operation. This is not a complete safety reassessment as described above, due to the fact that its replacement has been decided. Nevertheless, the continuity of service between OSIRIS and the Jules Horowitz Reactor (JHR), which will start operating by 2014, is foreseen.

All these research reactors are open to international collaboration in order to realise various experimental programs and welcome scientists worldwide to participate in these programs.

After a brief review of these reactors, the paper will focus on the safety reassessment of MASURCA and will give an overview of the refurbished reactors ready to operate for the next decades.

Extended synopsis **IAEA-CN-156**

Compatibility of safety and security

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Nuclear research reactors play an important role in the development of nuclear science and technology, contributing to important progress on nuclear power, radioisotope production, neutron beam research and analysis, nuclear medicine and personnel training, materials development, component testing, computer code validation... Many countries have developed their own nuclear program thanks, in part, to an effective development and efficient utilization of their research nuclear reactors.

However, concerns over the safety and security of the research reactors have increased due to the national and international contexts.

Regarding safety issues, more than half of all operating research reactors worldwide are nowadays over 30 years old. Many research reactors face concerns with management, obsolescence of equipment, lack of experimental programmes, lack of funding for operation and maintenance and loss of expertise through ageing and retirement of the staff. Also, as safety requirements have progressively been enhanced in the field of power reactors, additional safety requirements are often also applied to research reactors, adding to their operation costs.

Concerning security, it can no longer be assumed that research reactors cannot become a target for terrorists, either for the purpose of their destruction, or for the acquisition of sensitive materials. Hence, the security of nuclear research reactors has become an issue which should focus more attention worldwide. In particular, high priority should be given to improving their physical protection, and to converting the reactors to use Low Enriched Uranium fuel.

In this context, synergy between safety and security is of the utmost importance to meet the challenges associated with these issues.

Safety culture includes characteristics and attitudes in organizations and individuals which establish that safety issues relating to nuclear research reactor receive the attention warranted by their significance. Security culture includes characteristics and attitudes in organizations and of individuals which establish that the issues relating to protection against the loss, theft and other unlawful taking of nuclear material on one hand and deliberate malicious acts against nuclear research reactor or during transport of nuclear materials on the other hand, receive the attention warranted by their significance.

It should be understood that, despite differences between the safety culture and the security culture, the establishment of close links between the two is beneficial to both. The conference will present the key elements on which to establish such a continuity:

- **General framework**, which can be considered in the light of 4 concerns:
    - **Objective:** Security and safety cultures are based on the same principles: the general objective being the protection of man and environment, by

establishing and maintaining effective defenses, for the research reactor, against radiological risks in the event of an accident of functioning or by malicious acts.

  - **Governmental support:** The two cultures can only develop and be maintained if they are promoted at State level. Attention should be focused on the serious erosion in the level of governmental support, management commitment and available resources to the infrastructure necessary for effective research reactor operations.
  - **Regulatory framework:** For both safety and security, appropriate and coordinated regulatory frameworks are necessary. Given the large variety of types and modes of utilization of research reactors, preference should be given as far as possible to objective driven regulations rather than to detailed technical prescriptions
  - **International cooperation**: taking into account the large number of countries involved in the operation of research reactors, and the differences in the overall strength of their nuclear industry, international cooperation is essential to contribute to the achievement in every country of a sufficiently high level of both safety and security. International cooperation should in particular be active in the following areas:
    - Harmonization, and promotion of best practice in the regulatory field,
    - Development of scientific cooperation in order to maximize the rational utilization of existing and future research reactors,
    - Cooperation in order to promote worldwide a high degree of implementation of both safety and security cultures,
    - Provision of technical assistance, including training, to countries facing specific concerns with respect to safety or security in their installations
    - Security issues may at times be difficult to address in full at the international level. Bilateral or other more focused cooperation arrangements may be seen at times as more appropriate channels, but, in this case, care should be taken to ensure that such approaches remain compatible with the results of international consensus.

- **Methodology:** the concept of in depth defense, based of a good knowledge of the research reactor, can be applied to the 2 areas (prevention, supervision / detection, intervention, accident management...). In addition, in order to get a license, the operator must submit reports demonstrating that his installation is both safe and secure. In France, the Institute for Radiological Protection and Nuclear Safety (IRSN) - the technical support of the national authorities - is entrusted to carry out the assessment of this demonstration.

- **Reliability:** In security, as in safety, much depends on the individuals, on their reliability. Moreover, the operator remains globally responsible for his reactor.

- **Risks notion:** That encompasses the notion of "acceptable risk" (according to the consequences) as well as the notion of "design" (design-basis accident and design-basis threat (DBT)).

The enhancement of these commonalities should help avoiding the pitting against each other of safety and security. The two should coexist, interact and complement each other, although they have different attributes and objectives. Through concrete examples, the conference will give an insight on how such interactions may be managed, particularly in areas where safety and security appear on first sight to lead to opposite types of prevention measures, and will propose some key principles for the guidance of a process of optimization respectful of both safety and security objectives.

Operational Safety Experience at 14 MW Research Reactor from Institute for Nuclear Research Pitesti

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The safe operation of TRIGA reactor installed in Institute for Nuclear Research in Pitesti, started in 1979 in conditions of existing infrastructure, Regulatory Authority was established in 1962, a 2 MW VVR-S Research Reactor was in operation since 1956 (from 23 years), the University of Bucharest and the Polytechnic University teaching nuclear physics and nuclear engineering.

The Regulatory Authority having the independence for regulation and control, issued the basic norms and regulations for safety of nuclear reactors, radiation protection, transportation of radioactive materials, safeguard, updated in 1976. Cooperation with an experienced supplier of numerous TRIGA Research Reactors ensured the transfer of specific knowledge of core design, safety principles and operation of first 14 MW TRIGA SSR core built by US General Atomics.

The quality assurance for commissioning and operation of the reactor was the formal requirement of Regulatory Authority. In the above presented circumstances, the Institute for Nuclear Research built, commissioned, and operates the TRIGA Research Reactor under IAEA Project and Supply Agreement. The Safety Analysis Report structure and content, was adopted following the IAEA recommendation having in that time 17 chapters. Later, sections concerning Quality Assurance, Emergency Preparedness, Decommissioning, were developed and added.

The continuous development of safety requirements, new standards, guides and norms were reflected in national legislation and lead to a rather complete Legal Regulatory Framework, in this regard, based on Safety Analysis Report and the proposal of Operational Limits and Conditions sustained by Operating Organization, a full formal licensing process was developed and reinforced by the Regulatory Authority in Romania. Continuous development of safety standards and guides by IAEA, development of a new and complete set of internal regulations since 2001 by Regulatory Authority and experience of 20 years of TRIGA reactor operation lead to periodic revision of SAR. Most of the sections, chapters of SAR were updated considering the modification of the reactor, core conversion, taking advantage of the new computational methods and solutions. Reevaluation of safety studies with new methods including Probabilistic Safety Analysis, considering real condition of operation, updating the specifications, the procedures, the SAR and the OLC's allow to enhance the safety of TRIGA Research Reactor. The accomplishment of safety objectives of operation and utilization of research reactor operated by an organization which is in the same time Technical Support Organization for Nuclear Activities in Romania is the challenging task correlated with the sustainability of the institute.

Operational safety of TRIGA Research Reactor is sustained by Organization for Operation of Research Reactor Department and overall management at the level of institute. Organization of Quality Assurance evolved in institute since 1978 in agreement with continuous evolution of standards and applications for products and services of institute, lead now to an integrated Quality Management System following ISO 9001/2000 accredited by Lloyds in 2003.

Concept of defense in depth is applied to all activities related to safety organizational, design related, operation, modification and utilization, emergency preparedness. Operational Limits and Conditions of TRIGA Research Reactor are approved by Regulatory Authority with the occasion of Safety Analysis Report updating and any time when the license is renewed.

The IAEA INSARR Missions performed at TRIGA Research Reactor in Pitesti, performed in 1983, 1992 and 2002, and follow-up in 2003, ensured the evaluation of safety. The suggestions and recommendations formulated, were progressively implemented.

Fulfillment of OLC's requirements in operation ensure the level of prevention, that means that normal operation values of operational parameters and administrative requirements are established with conservative margins, the warning of deviation or/and violation of OLC's will bring the reactor in safe condition.

Program of monitoring, verification and surveillance, testing of equipments and systems, ageing evaluation is presented as well as the results of program and feedback in the decision of research reactor systems maintenance and modernization.

A special program for radiological protection, environmental monitoring of releases and effluents is derived from SAR requirements and is applied by the Radiological Protection Department.

A comprehensive Emergency Intervention Plan conceived in correlation with the results of safety analysis of each internal and external initiating events and some events beyond design basis provide means of action for mitigation of accidents in correlation with offsite emergency plans.

Safety Culture of Operating Organization is appreciated in regards of actions oriented to foster responsible behavior and the manner in which conditions and resources for safety are allotted. In correlation with IAEA Research Reactor Incident Reporting System proposal, a set of performance indicators was selected. The Annual Operation Report contains the synthesis of performance indicators and trends derived from multi annual analysis based on self-evaluation of safety issues and concerns, diverse challenges need diverse approach.

The main challenges identified in TRIGA Research Reactor operated in Institute for Nuclear Research in Pitesti, Romania, are in fact similar with challenges of many other research reactors in the world, those are:

- Ageing of work forces and knowledge management;
- Maintaining an enhanced technical and scientific competences;
- Ensuring adequate financial and human resources;
- Enhancing excellence in management;
- Ensuring confidence of stakeholders and public;
- Ageing of equipment and systems.

To ensure safety availability of TRIGA Research Reactor in INR Pitesti, the financial resources were secured and a large refurbishment programme and modernization was undertaking by management of institute. This programme concern the modernization of reactor control and safety systems, primary cooling system instrumentation, radiation protection and releases monitoring with new spectrometric computerized abilities, ventilation filtering system and cooling towers. The expected life extension of the reactor will be about 15 years.

The international cooperation is a useful resource for nuclear safety. It is worth to mention the IAEA support sustaining Research Contracts, Technical Cooperation Projects related to specific safety issues, trainings, workshops and meetings, as well as bilateral cooperation with organizations from Europe and USA.

Ljubljana TRIGA Mark II, 40 Years of Successful Operation

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The research reactor TRIGA Mark II is part of the Jožef Stefan Institute, located near Ljubljana. It was built by General Atomics. The research reactor was commissioned in 1966 and in 1991 it was reconstructed and equipped for pulse mode operation.

The reactor TRIGA Mark II is a typical 250 kW light water reactor cooled by natural convection. It is designed for training in reactor operation and technology, research with neutrons and isotope production. It has been used for experiments in the following fields: solid state physics, neutron radiography, reactor physics including burn up measurements and calculations, boron neutron capture therapy, environmental studies and researches of advanced materials.

The reactor has accumulated 40 years of continuous operation without any failure of major equipment or any event violating safety standards. There has been no release of radioactivity into the environment exceeding limiting values prescribed by the regulatory requirements. Major refurbishment included installation of a pulse rod, reconstruction of control mechanisms and control units, replacement of the primary coolant pumps with new ones, modification of a spent fuel storage pool and installation of new pneumatic mail.

The United States nuclear non-proliferation policy provided Slovenia with the opportunity to return the spent fuel from the research reactor TRIGA Mark II back to the USA. After reconstruction, the reactor was loaded with fresh low enrichment fuel elements and all spent fuel elements were shipped back to the USA in July 1999. At present there are 94 fuel elements with 20 % enriched uranium on site.

All questions related to nuclear safety are treated in detail in a Safety Analysis Report. Its operation is regulated by several national and international nuclear laws, regulations and standards. The enforcement is provided by national and international bodies: Slovenian Nuclear Safety Administration (SNSA), Health Inspectorate of the Republic of Slovenia and International Atomic Energy Agency.

A set of new regulations for research reactors are currently under preparation (regulation on safety analyses report with specific attachment for research reactors, regulation on operation and reporting requirements of a licensee with some specific issues for the research reactor, regulation on design basis including design basis for the research reactor too, etc.).The requirement for a research reactor periodic safety review will be included in new regulation. The graded approach is taken into account. Application of the Code of Conduct on the Safety of Research Reactors is accomplished through the new safety regulations pertaining to all stages in the life of the research reactor.

The SNSA also undertake regulatory inspections and assessments of the TRIGA Mark II research reactor. Based on a request from the SNSA an IAEA INSARR (Integrated Safety Assessment for Research Reactor) mission was performed at the TRIGA research reactor in

March 1992. The mission identified a number of good practices and confirmed that the nuclear activities in the research reactor centre were being conducted in accordance with the purposes of the Agency's safety standards and measures. Several SNSA inspections of the TRIGA reactor in the past decade covered the program of implementation of the INSARR recommendations.

The reactor has been playing an important role in developing nuclear technology and safety culture in Slovenia. Technical conditions of the reactor allow for at least 10 years more operation. It is planned that the reactor will operate at least until 2016.

Ageing Management Program of Kartini Reactor for Safe Operation

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Kartini Reactor in Yogyakarta is in operation for 28 years. In order to assure and maintain the safety of the reactor, ageing management program has been conducted. Inspection to the Reactor tank was conducted firstly in 2001 during major shutdown by using non destructive inspection methods, such as visual, ultrasonic, replicas, dye penetrant and hardness survey. Through visual inspection indicated that the tank has only minor defects. The origin of the two swelling features seen on the bottom tank indicated as S1 and S2 could not be verified, however they are not considered to be a serious issue for future safe operation. The wall thickness was predominantly in original condition about 6 mm. A few thickness reading were found to be low (less than 5 mm), however, the origin of these low readings is uncertain and could be some defect from the original manufacturing process. Replication of the surface defects indicated that defects observed are minimal and are not of concern from a safety perspective. The small crack was detected through replicas and dye penetrant approximately 6mm long was located in the upper belt weld will not cause fracture and is an original manufacturing defect confined to the weld cap. This is a hot-tear from the welding process. Hardness values are consistent with published data and minor neutron damaged has occurred. In general, inspection result revealed that the Kartini Reactor tank is still in good and safe condition for continued operation [1]. Under IAEA TC Project INS/9/022, in year 2004 BATAN received new inspection equipment such as video scope for visual inspection (under water camera) and ultrasonic wall thickness. By using this equipment, re-inspection to the reactor tank to obtain latest features of the reactor tank condition and to observe the swelling closer and more detail has been conducted in year December 2004 and July 2005. Thickness measurement performed on the reactor tank wall at various of locations to determine the tank wall thickness. Reading of the wall thickness of between 5.3 mm 5.5 mm were obtained. Thickness measurement was then taken on swelling areas of S1 and S2 was found also to be between 5.3mm and 5.5 mm. It was mean that thickness of the reactor tank wall and bottom tank were still in original [2]. To provide the swelling profile such as height and areas of swelling, a replicate technique by using dental putty has been applied for. This technique was one of non destructive testing method, which can be applied for dry and wet conditions and also to measure accurately the dimension of defect and its profile on the surface of a component. Principle of this technique is by making mold on the examined component using dental putty. Result of the first mold called negative replicate. Furthermore, by using the same way the second mold on the negative replicate is made. Result of the second mold called positive replicate became a representative feature of swelling. Detail measurement of swelling profile to obtain height and areas of swelling in several point is conducted on the positive replicate utilizing height gauge tool [3].
Meanwhile Kartini Reactor license was expired in November 2005. To process a renewal license, one of BAPETEN requirements (Regulatory Body of Indonesia) was safety status of Kartini Reactor must be demonstrated well. Therefore re-inspection to the reactor tank using visual, ultrasonic and replicas were conducted again in December 2005 and September 2006. Through visual inspection indicated that the defects in the reactor tank were minor. Ultrasonic to measure the thickness of reactor tank wall at various of locations includes on swelling areas indicated that the thickness was still between 5.3 mm 5.5 mm. It was mean that thickness of

the reactor tank wall and bottom tank were still in original [4]. Swelling areas has been measured using replicate technique. Data of swelling areas from inspection results in December 2004, July and December 2005 and September 2006 were tabulated in Table 1 [3,5]. Figure 1 was the measurement of swelling area. Figure 2 was the graph of S1 and S2 of swelling areas. Data obtained in the Table 1 show that swellings areas had grown slowly in size and became relatively stabilize (does not grow). Although swelling has been occurred on the bottom tank of Kartini Reactor as an ageing product of component and relatively constant in size after in operation for 28 years, Kartini Reactor tank inYogyakarta is still in good and safe condition for continued operation.

A routine and regularly inspection schedule is established for Kartini Reactor. Swelling area will be monitored every 3 months utilizing visual equipment and every 6 months utilizing replicate and ultrasonic. A comprehensive inspection will be conducted every 5 years.

As an outcome of regularly inspection has been performed and safety status has been demonstrated well over this period, last November 2006 Kartini Reactor received a new 4 years license from the BAPETEN.

TABLE 1: AREA OF SWELLING [3,5]

Area of Swelling S1 (in mm^2)				Area of Swelling S2 (in mm^2)			
Dec. 2004	July 2005	Dec. 2005	Sept. 2006	2004 Dec.	July 2005	Dec. 2005	Sept. 2006
1365.07	1380.17	1389.40	1404.59	1082.78	1749.00	1839.08	1839.67

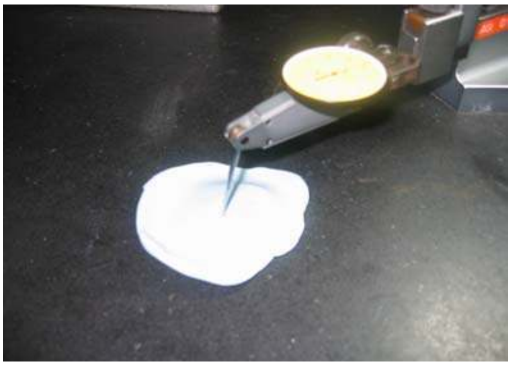

FIG. 1.
Measurement of Swelling Area

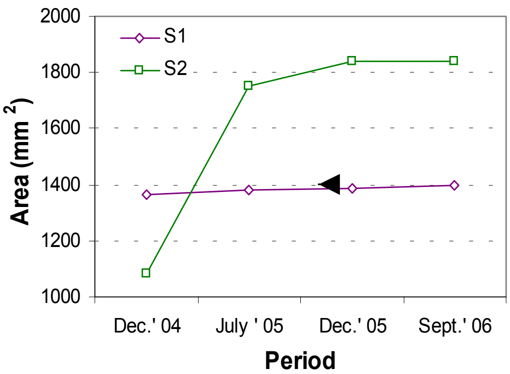

FIG. 2.
The Graph of Swelling Area of S1 and S2

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Periodic Safety Review Management for French Research Reactors - Technical Support Organization Approach

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Periodic Safety Review (PSR) is a key issue in the life of a Basic Nuclear Installation (BNI). Indeed, the safety review is an efficient means to improve the safety level of an installation and to take a decision for the continuation of the reactor operation towards the next decade. Based on the French practice and experience, the objective of this paper is to give an overview of the PSR management for Research Reactors (RR) from a Technical Support Organization (TSO) point of view.

After a presentation of the new French regulatory context since the law on transparency and security in the nuclear field has been published in the *Official Gazette* on 13 June 2006, the expectations/requirements of the Nuclear Safety Authority (ASN) and the Institute for Radiological Protection and Nuclear Safety (IRSN) are given.

All the organization and contents of PSR, from the submission by the operator of the main options report until the implementation of hardware modifications and the update of reference documents, are described; however, the paper mainly focus on the two basic parts of PSR, which are the conformity check, and the safety reassessment.

For the PSR, the safety review shall consider especially:

- the state of the installation,
- the operational experience feedback,
- the improvement of technical knowledge,
- the applicable rules for similar installations.

Then, based on the latest PSR conduct on French research reactors and analyzed by IRSN, the paper emphasizes the main conclusions. In this regards, items such as:

- the seismic resistance,
- the reliability of the safety systems,
- the structure ageing,
- the improvements in the Defence-in-Depth application,

are widely concerned.

Anyway, if now in France the great safety principles applied for the safety evaluation of any research reactors are very similar to those used for the Nuclear Power Plants (NPP), adaptations and graded approaches are used, due to specific features of research reactors.

The full paper presents and discusses all those points; it gives also many examples.

Experience of IPEN-CNEN/SP in the Execution of the First Phase of the Safety Culture Enhancement Programme at IEA-R1 Brazilian Research Reactor

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This work describes the experience gained in the execution of the first phase of the Safety Culture Enhancement Programme being implanted in the IEA-R1 Research Reactor. This phase covered a period from September 2002 to December 2006. The main results of the study are presented.

IEA-R1 is a 5 MW pool type reactor, cooled and moderated by light water, and it uses graphite and beryllium as reflectors. The reactor building is located within the premises of IPEN-CNEN/SP, one of the Brazilian institutes for nuclear and energy research. The operation, maintenance and irradiation services of IEA-R1 reactor are currently being administered by the Research Reactor Center.

The general manager of the Research Reactor Centre instituted a Safety Culture Enhancement Working Group, with the objective to formulate and implement the first phase of the Safety Culture Enhancement Programme. The group included senior professionals representing diverse areas such as radiation protection, quality management, probabilistic safety assessment, managers of reactor operation and services and the general manager of the Research Reactor Centre. The group meetings were held every week to discuss different issues involved in the programme.

The first task of the working group was to acquire knowledge about the state of the art in safety culture. This task was accomplished through seminars given by invited speakers and panel sessions where the participants made presentations based on literature studies in this area. The relevant literature studies consisted of guidelines and reports published by the International Atomic Energy Agency (IAEA) [1], the U.S. Nuclear Regulatory Commission (NRC), the Health and Safety Executive (HSE) [2], DuPont, and Brazilian nuclear organizations.

After the completion of literature survey and the detailed analysis of different approaches which possibly could be used in our context, the working group developed a proper methodology to be used in the first phase of the Safety Culture Enhancement Programme at IEA-R1 Research Reactor consisting of the following steps:

a) Safety culture assessment at IEA-R1 Reactor;

b) Identification of weak points related to safety;

c) Elaboration and implementation of an action plan aiming at the enhancement of safety culture in the organization.

The safety culture assessment was carried out in two stages adopting different approaches. 1) Evaluation of the reactor employee's perception regarding safety by conducting a survey in order to evaluate the main aspects of safety culture according to the employee's attitude, opinion and perception [3]. The survey method used was a quantitative written questionnaire composed of 42 questions. The questionnaire was answered by 34 people involving only part of the staff of the Research Reactor Centre, more specifically those who work at the Reactor Operation and Maintenance Division, the Irradiation Service Division as well as the technicians of the Radiation Protection Division. The data were compiled, statistically analyzed, documented in a technical report, and the main conclusions were presented in the form of a seminar to the employees. 2) Self Assessment of Safety Culture at IEA-R1 where the aim was to evaluate safety culture at the reactor objectively. For this purpose, the working group used the self assessment questionnaire proposed in the document INSAG-4 published by the IAEA [4]. Each question was accurately analyzed, discussed and compared to evidences before being answered in a consensual way by the group. The results of the self assessment task as well as an outline of improvement actions were documented in a technical report which was placed at the disposal of the staff of the IEA-R1 Reactor.

A workshop is planned to be held in the first semester of 2007, with the participation of employees of the IEA-R1 Reactor. In this workshop the main concepts of safety culture and the results of the two previously mentioned safety assessment approaches will be presented. Discussions on these issues among the participants should lead to the identification of the weaknesses and strengths of the safety culture of the organization. The conclusions and recommendations would be taken into consideration for the elaboration of a formal action plan for the safety culture enhancement.

The second phase of the Safety Culture Enhancement Programme at the IEA-R1 Research Reactor will be started after the implementation of the action plan when further evaluations will be performed. The refinement of the methodology applied in the first phase is also expected.

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Development of Safety Performance Indicators for HANARO

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Synopses

The nuclear facilities need an extensive basis for ensuring their safety. An operating organization should conduct its operation and utilization important to the safety in accordance with approved procedures and regulations [1]. The general aims of a management system for nuclear facilities are to improve the safety performance through a planning, control and supervision of safety related activities and to foster a strong safety culture [2]. The effectiveness of a management system can be monitored and measured to confirm the ability of its processes to achieve the intended safety performance by an assessment of the operational performance. The Operational Safety Performance Indicators, also known as SPI, help an organization define and measure a progress with regard to safety activity goals. The elements of a SPI are quantifiable measurements that reflect the critical success factors of an organizational safety. Since 1995, efforts have been directed towards the elaboration of a framework for the establishment of an operational safety performance indicator program in nuclear power plants (NPP). IAEA-TECDOC-1141, "Operational safety performance indicators for NPP" attempted to provide a frame work for an identification of performance indicators which have a relationship to the desired safety attributes, and therefore, to a safe plant operation. Three key attributes of a smooth operation, an operation with a low risk, and an operation with a positive safety attitude, were recommended, which are associated with a safe operation. Because these attributes cannot be directly measured, an indicator structure is expanded further until a level of easily quantifiable or directly measurable indicators is identified [3]. The intention of this approach is to use quantitative information provided by the specific indicators and to analyze performance trends relative to established goals.

The safety activities in HANARO have been continuously conducted to enhance its safe operation. HANARO has made an effort to select operational safety performance indicators which are specific for a research reactor operation and utilization. The main frame is nearly the same structure recommended by IAEA-TECDOC-1141 except for the attitude of reactor utilization for a research and an application. The measuring elements of the HANARO Operational Safety Performance consist of 4 safety attitudes, 10 overall indicators, 22 strategic indicators and 42 specific indicators. The operational safety attitudes of HANARO include following categories;

A. Plant operates smoothly
B. Plant operates with low risk
C. Plant operates with a positive safety attitude
D. Plant operates with a safe utilization

42 specific indicators were selected to cover the most adequate parameters to monitor the safety attributes in HANARO. Through reviewing these indicators, we can obtain the following information;

- Plant safety status
- Safety parameter trends
- Safety information, for example, reactor operation status and radiation safety

These indicators will be very useful to review and evaluate the safety performance of the reactor operation and utilization. We will continuously pursue the trends of the operational safety attitudes for an effective safety management of HANARO.

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EMERGENCY INTERVENTION PLAN FOR 14 MW TRIGA - PITESTI RESEARCH REACTOR

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The decision to initiate a protective action is a complex process. The benefits of taking the action is weighed against the risk and constraints involved in taking the action. In addition the decision will be made under difficult emergency conditions, probably with little detailed information available. Therefore, considerable planing is necessary to reduce to manageable levels the types of decisions leading to effective responses to protect the public in the event of a nuclear incident.

The sequence of events for developing emergency plans and responding to nuclear incidents [1] will vary according to individual circumstances, because the international recommendations and site-specific emergency plans cannot provide detailed guidance for all accident scenarios and variations in local conditions. Flexibility must be mentained in emergency response to reflect the actual circumstances encountered (e.g. source term characteristics, the large number of possible weather conditions and environmental situations such as time of the day, season of the year, land use and soil types, population distribution and economic structures, uncertainties in the availability of technical and administrative support and the behaviour of the population). This further complicates the decision-making process, especially under accident conditions where there are time pressures and psychological stress.

Therefore one of the most important problems in the case of a nuclear emergency is quantifying all these very different types of off-site consequences. Last years, and in particular since the Cernobyl accident, there has been a considerable increase in the resources allocated to development of computerised systems which allow for predicting the radiological impact of accidents and to provide information in a manageable and effective form to evaluate alternative countermeasure strategies in the various stage of an accident.

The emergency plan is structured on 4 sections: Section I: General data on INR site, Section II: On-site intervention procedures in event of nuclear accident having off-site consequences, Section III: Specific intervention procedures for INR nuclear facilities having strictly limited consequences on facilities or site and Section IV: Procedures for beta and gamma radioactivity determination

During the normal operation of the nuclear facilities, the sources are under control.

The effects on the population health are in the stochastic range, and the radiation protection has the role to maintain the probability of consequences at a low range.

In case of a nuclear accident the sources are no more under control. The reduction of doses – and the implicit reduction of health risks – is done by taking measures in what concerns the employees, the population and certain environmental factors.

The dose that the INR personnel can bear in case of a hypotetical foreseen accident (destruction of a cassette or of the active zone) varies among a few mSv and Sv. The effects on their health belong as much to the probabilistic domain as to the deterministic one. Few persons in the reactor’s building may get lethal doses.

The intervention's management must be as well organized as to prevent serious deterministic effects, this is why the on-site emergency plan has a modular structure.
The procedures included in the third section describe the way of operating in case of certain possible accidents at the nuclear installations of ICN and NFP (nuclear fuel plant). The foreseen events may have cosequences on the installations and maybe on the personnel. It is unlikely that the effects of these events have off-site consequences. For every foreseen event a procedure has been settled. These procedures have a strong technical character, specifically describing the maneuvers that must be accomplished on the installations.
The procedures from the fourth section are the monitoring procedures for radiation levels during emergency and fallow- up. These procedures establish the way of sampling and the preparation of samples for beta measurements and for the gamma spectrometry analyses of the radioactive content in air, soil, vegetation, foodstuff etc. The procedures from the fourth section have the role of providing integrated methods of sampling, preparation, measurement, interpretation and recording of the results for each laboratory and for all the people involved.
The schematic structure of activities covered by emergency intervention plan are presented in Figure 1.

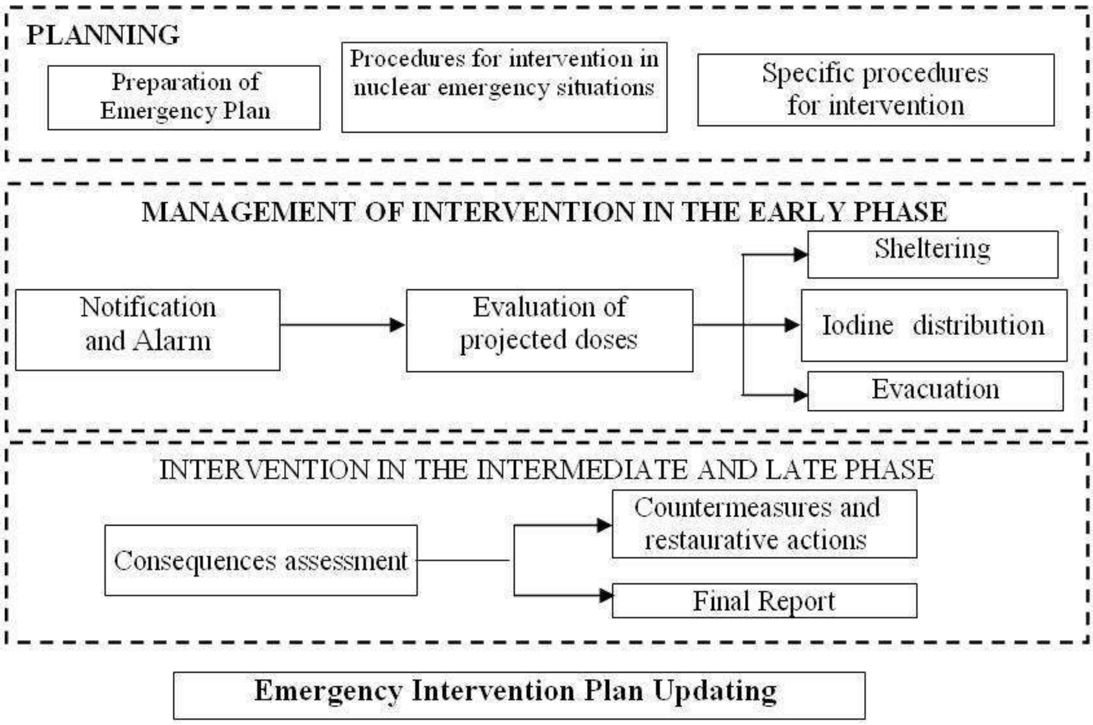

FIG. 1. The schematic structure of activities covered by emergency intervention plan.

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Session 5: Fast Flux Research Reactors

Synopses no. IAEA-CN-156/	**Synopses Title**	**Main Author**
U-3	Analysis of sodium cooled fast reactors operation and consequences for future reactor design and operation	Guidez, J.
U-31	Two Decades of Operating Experience with the Fast Breeder Test Reactor	Gopala Iyengar, S.
U-5	Budget uncertainty and minimum detectable concentrations using relative, absolute and k0 – IAEA standardization for the INAA laboratory of the ETRR-2	Khalil, M.Y.
U-20	Usage of the BOR-60 reactor for investigation of advanced fuel cycles and materials	Bychkov, A.V.

Analysis of Sodium-Cooled Fast Reactor Operations World-wide and Consequences for Future Reactor Design and Operation.

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In 2007, the sodium-cooled fast reactors which are currently operating or have operated throughout the world will have accumulated collective experience of 379 years of operation for 18 different reactors.

This paper summarizes the various incidents and problems which have impacted these reactors' operations, ranks them by function and lists the solutions which were brought. This paper solely looks at the operating problems which occurred in the "sodium" part of the fast reactors. Problems encountered in the classic water/steam part are not described herein

A review is made for all these reactors on

-The water/sodium reactions occurred in the steam generators.
-The technical difficulties on the primary components (pumps and exchangers) and the repair operations.
-The incidents in handling operations
-The spurious leaks or transfers of sodium and their consequences.
-The intakes of air or impurities
-The experience from fuel and clad failures.
-The neutronic operations and control
-The material behaviour problems.(as 321 or SPX1 drum)
-The difficulties due to sodium aerosols.
And all the specific difficulties due to sodium related technological problems.

This paper also describes how the experience gained has been taken into account in the safe operation of these plants and also in the design of future reactors.

In conclusion it appears that a significant experience has been accumulated that allow today good availability for the remaining operating plants and also good design possibilities for the future.

Two Decades of Operating Experience with the Fast Breeder Test Reactor

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Fast breeder reactors constitute the second stage of India's three-stage nuclear energy programme, for effective utilization of the country's limited reserves of natural uranium and exploitation of its large reserves of thorium. The Indira Gandhi Centre for Atomic Research was established at Kalpakkam, 80 km south of Chennai (India) in 1971, with the mission to develop the technology of sodium cooled fast reactors. The heart of this Centre is the sodium cooled test reactor, named Fast Breeder Test Reactor (FBTR), built to serve as a test bed for irradiation of fuels and materials and provide experience in large scale sodium handling and fast reactor operation. FBTR was built on the lines of the French Rapsodie-Fortissimo reactor.

FBTR is a 40 MWt, sodium cooled, loop type fast reactor. Heat generated in the reactor is removed by two primary sodium loops, and transferred to the corresponding secondary sodium loops. Each secondary sodium loop is provided with two once-through steam generator modules. Steam from the four modules is fed to a common steam water circuit comprising a turbine-generator and a 100% dump condenser. The reactor went operational in Oct 1985 with a core of high Pu carbide fuel. Being a unique fuel of its kind without any irradiation data, the core size was made small, with 23 fuel subassemblies of Mark-I composition (70% PuC+30%UC) , rated for 10.5 MWt. The fuel was rated for a linear heat rating of 250 W/cm and the target burn-up was set at 25 GWd/t. The target burn-up has been progressively increased based on Post-irradiation examination at 25, 50 & 100 GWd/t. The fuel has seen a burn-up of 155 GWd/t without any pin failure. Over the years, the core has been slowly enlarged by adding fresh fuel subassemblies to compensate for burn-up loss of reactivity. The present core has 49 fuel subassemblies- viz. 27 subassemblies of Mark-I composition (70% PuC+30%UC), 13 subassemblies of Mark-II composition (55% PuC+45%UC) and eight MOX fuel subassemblies (44% PuO_2 + 56% UO_2).

The experience with sodium systems has been extremely good, except for an incident of leak of 75 kg of sodium into an inerted cabin. The sodium purity has been maintained consistently well for the past two decades by cold-trapping of oxides. Each of the four sodium pumps has been in trouble-free service for more than 125,000 h. The steam generators, which have high pressure water & steam on the tube-side and sodium on the shell side, have operated for more than 20,000 h without any incident of water leak into sodium.

With its flux one order higher than in the Pressurised Heavy Water Reactors, the Fast Breeder Test Reactor was used for studying the irradiation creep behaviour of Zr-Nb being used in the Indian Pressurised Heavy Water Reactors. The present mission of the FBTR is to irradiate the MOX fuel (29 % PuO_2) chosen for the 500 MWe Prototype Fast Breeder Reactor being built at Kalpakkam at a linear heat rating of 450 W/cm. To get the required linear heat rating in the high

Pu core of FBTR, U^{233} has been added to the test fuel. This test fuel has seen a burn-up of 60 GWd/t so far, without clad failure. It is planned to test it upto 100 GWd/t.

Several physics experiments have been conducted in FBTR so far. In addition to routine measurements of control rod worths and reactivity coefficients of inlet temperature and power after every core configuration change, void coefficients of the core at different locations have been measured; response of delayed neutron detection system to detect clad failure has been studied; flux mapping in sodium above the core has been done. A series of safety related engineering tests was conducted in 1994-95, basically to validate the codes used in incident analysis. The most important tests were natural convection tests in the secondary and primary loops at a reactor power of 170 kWt. The evolution of the temperatures of biological shield concrete and structural concrete of the Reactor Assembly in the absence of shield cooling was studied.

Several problems have been faced in the course of reactor operation. During commissioning, reactor vessel deflection was found high, beyond a sodium temperature of 270˚C, due to circumferential variation of the reactor vessel temperatures in the cover gas region. It was suspected to arise due to convection loops formed by the cover gas in the annular gap between top shields and reactor vessel. This was solved by injecting helium in the gap, while admitting argon over the free sodium level below. In 1987, there was a major fuel handling incident, in which a sturdy tube which guides the gripper of the Charging Flask got bent by about 320 mm. The bent tube was cut in-situ using a specially designed remote cutting machine and removed in two pieces. All care was taken to ensure that the cutting chips do not fall into the reactor. In 1996, the core cover plate which houses the core thermocouples got stuck in the fuel handling position, and could not be restored. This has led to inaccurate core temperature measurements due to plenum hydraulics. There was an incident of water leak from the biological shield cooling coils. The leaks were reduced by chemical sealing twice. Measures have been taken to monitor the leaks and to mitigate further aggravation. We had three incidents of anomalous positive reactivity- in 1994, 1995 and 2000. Except that all were positive, they were different in their characteristics and magnitude. Though the exact reasons could not be found out, they are all suspected to arise out of the large temperature gradients inherent in a small core with high power density. The incident of leak of 75 kg of primary sodium was due to a defective valve. The system was restored within two months, with practically no exposure or contamination to personnel.

The radiological impact of FBTR is absolutely negligible. The cumulative dose to personnel in two decades is only 60 person-mSv. The cumulative stack release is less than 18.5 TBq of Ar^{41} (500 Curies), testifying to the fact that fast reactors are eco-friendly.

The encouraging experience with FBTR operation has been a major factor in the launch of the 500 MW(e) Prototype Fast Breeder Test Reactor at Kalpakkam. This reactor has been designed and developed by IGCAR. The reprocessing of the high burn-up carbide fuel has also been successfully demonstrated by IGCAR. With these, India is set to launch upon a major programme of fast reactors in the next two decades.

Budget Uncertainty and Minimum Detectable Concentrations Using Relative, Absolute and K0 –IAEA Standardization for the INAA Laboratory of the ETRR-2

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The objective of this work is to determine the uncertainty budget and sensitivity of the INAA laboratory measurements according to some standard methods [1, 2, and 3]. The relative, absolute and k0 methods are used competitively for that purpose [4, 5]. Concentrations of 9 elements, Ca, Co, Cr, Cs, Fe, K, Mn, Na, and Rb, were measured against a certified test sample. Relative, absolute, and K0-IAEA standardization were employed.

Two reference materials were used. One is certified rock-type test sample (P1), and the other is Certified Reference Material (CRM). The sample and reference were provided from a proficiency test study under the AFRAIV-7 project. The flux monitors were gold, cadmium cover (foil, tube), gold diluted with aluminum of (gold 0.1% and aluminum 99.9%), nickel, and zirconium, all with purity (99.9%).

Two different reference rock samples with certified analysis were used. One was considered a test (unknown) sample and the other as a comparator. Some elements under consideration produce long half-life radionuclides after irradiation and the others produce short ones. Hence, samples were divided into two groups. One group, for counting of short lived isotopes, was irradiate in thermal column position by means of the rabbit system. The other group, for counting of long lived isotopes, was irradiated in the irradiation grid.

Measurements of samples were carried out at 7 cm from the top of HPGe detector with relative efficiency 100%. Detector resolution FWHM is 2.1keV for the Co-60 line at 1332.4 keV. The detector is coupled to a computer controlled gamma ray spectrometric system through a chain of associated linear electronics which contain a multi-channel analyzer, and adapted nuclear data software (Gamma Vision) for neutron activation analysis employed for online spectral evaluation.

Concentrations using relative method

Test and reference samples were irradiated sequentially and later measured under the same counting conditions. The net area correction was applied and the concentration of the elements under consideration was calculated.

The sources of standard uncertainty are grouped according to the individual steps of analysis into three categories uncertainty in sample preparation, uncertainty in sample irradiation, and uncertainty in sample counting. The uncertainties arising from each individual source were evaluated. The final stage is to multiply the combined standard uncertainty by chosen coverage factor in order to obtain the expanded uncertainty.

Concentration using absolute standardization

The combined standard uncertainties were calculated using the absolute standardization equations. Finally the expanded uncertainties were calculated.

Concentration using the K0-IAEA method

The concentration in the K0-IAEA method is determined in three steps. One is to calibrate the detector using a calibration source. Second is to characterize the irradiation positions using gold and zirconium samples for the rabbit system, and gold, zirconium, and nickel samples for the grid positions. Finally, the spectrum and the GammaVision report associated to every sample and comparator were fed to the k0-IAEA software. The program interpreted every series of samples plus comparators and produced the elements concentration.

Calculating the minimum detectable concentration

Using the values of uncertainties obtained in the relative and absolute standardization methods the minimum detectable concentration evaluated for the three standardization methods.

Conclusion

The uncertainty budget in the relative, the K0-IAEA, and the absolute standardization methods ranged from 2-11%, 3-15%, and 6-27%, respectively. Despite that the relative method is the most accurate among the three methods tested, the K0 method is more handy and easier to employ when large amount of data must be processed. The minimum detectable concentration was the lowest for Cs ranging between 0.36 and 0.59 ppb and the highest being for the K in the range of 0.32 to 8.64 ppb.

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IAEA-CN-156/U-20

Usage of the BOR-60 reactor for investigation of advanced fuel cycles and materials

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FSUE "SSC RIAR"

Abstract

Currently fast research reactor BOR-60 is actually the only one of its kind in the world that has been demonstrating safe and continuous operation for more than 35 years. One of the major reactor objectives is irradiation of advanced fuel and structural materials in different operating conditions. Materials can be irradiated in any cell of the reactor core and lateral screen except for seven cells intended for control rods. A number of fuel assemblies loaded in the reactor can vary from 85 to 124 according to fuel burnup, reactor core configuration and fuel properties. No less than 20 experimental fuel assemblies can be loaded in different cells of the reactor core because the design allows for changing the reactor core size in a wide range. The BOR-60 reactor is a unique experimental facility whose neutron spectrum can vary from the "rigid" spectrum in the reactor core to the "intermediate" in the lateral screen. It also has got a high neutron flux density of 3.5×10^{15} $cm^{-2}s^{-1}$ that can vary three times as much in some cells of the reactor. This allows the reactor to be loaded with different fuel compositions and actually achieve different fuel burnup values.

A long-term study of the reactor performance made it possible to analyze its behavior under different conditions in greater detail, develop a set of verification codes and different techniques for on-line support of the reactor operation and performance of a great number of experiments.

A system of special-purpose experimental devices comprising capsules and dismountable fuel assemblies is used to irradiate numerous grades of materials and articles under different conditions and at different parameters. A simple design and convenience of installation in any cell of the reactor core and lateral screen actually should be considered among the advantages of these devices. In this case it is possible to monitor neutron flux and temperature in addition to a precision computation of irradiation test conditions. There is a special thermometric channel in the reactor. It allows for installing the experimental devices in the reactor core directly and outputting data on the irradiation test conditions of materials through 30-50 communication lines. A number of autonomous instrumented loops-capsules, special instrumented fuel

assemblies etc. have been developed for this channel. The main objective assigned in development of these devices is to provide the desired temperature conditions on the specimens. For this purpose, heat insulating gaps, blast cooling or additional heating at the expense of energy release during irradiation or fuel fission are used. As a result, different advanced types of fuel and structural materials are tested in the reactor at high thermal loads (100 kW/m), temperatures (1000 °C), burnups (33 % h.a.) and fluences (1.8×10^{23} cm^{-2} with E>0.1 MeV). As may be required, temperature is stabilized through changing thermal resistance in the heat supply circuit or intensifying heat removal with liquid boiling metal. These devices make it possible to provide a predetermined azimuthal temperature non-uniformity and temperature non-uniformity along the height. These experimental devices can be used for irradiation of fuel elements down to 15 mm in diameter, which are enclosed in different arrays (triangular, square etc.) and environments (sodium, lithium, lead, different gases etc.).

The available pilot reprocessing facility of irradiated fuel and production facilities of fuel and fuel elements for fast reactors make it possible to develop a closed fuel cycle. The Institute performs examinations and irradiation tests of refabricated fuel with involvement of minor actinides and long-lived fission products in the closed fuel cycle. The efficient operation of the BOR-60 reactor as a burner of minor actinides as well as power-grade and weapon-grade plutonium has been demonstrated. Taken together these factors allow for solving important tasks on reduction in value of fuel factor, decreasing the amount of radioactive waste and improving the ecological situation in the nuclear power engineering.

The BOR-60 reactor made it possible to obtain a wide diversity of experimental data on irradiation of oxide, metal, ceramic-metal, carbide and nitride fuel compositions for reactors of different applications, in particular for fast reactors with sodium coolant. Regularities of gas release, deformation and structure formation were studied. The obtained results made it possible to justify the usage of fuel elements for fast reactors BN-350 and BN-600 as well as other types of reactors. Uranium pellet fuel was used in the reactor since 1969 and then in 1981 it was converted to vibropac mixed oxide fuel with the use of power-grade and even weapon-grade plutonium in the last years. So it is possible to perform a large-batch testing of different types of fuel. A special loop-capsule was used for vibropac fuel testing under emergency conditions until fuel element failure. These tests proved the possibility of fuel testing not only under steady state and transient but emergency conditions also.

The results obtained during examination of different fuel compositions form the basis for fuel cycle development of advanced fast reactors with the enhanced safety. Among them are the BREST-OD-300 reactor with the nitride fuel and other advanced reactors. The use of carbide and nitride fuel that possess high fissile atom density and high thermal conductivity as opposed to the oxide one allows for increasing thermal loads and safety indexes significantly.

A loop device was used for performance of the first test runs of the BREST fuel elements in lead at the irradiation test parameters similar to the design ones. A burnup of 0.5 % h.a. was achieved. Currently the same fuel elements are being irradiated as a part of fuel assemblies cooled with sodium to enhance representativeness of the results. The achieved fuel burnup is 2.2 % h.a. and damage dose - 23 d.p.a. in steel.

Nowadays the reactor continuous its successful operation and thus provides for performance of a large scope of scientific research. The Institute holds a license for its operation till 2010. Work is being done to extend the service life of the reactor till 2015. In doing so there are preconditions for continuation and expansion of activities in the field of fuel cycle justification of advanced reactors.

Session 6: Research Reactor and Network Corporation

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DEVELOPING RESEARCH REACTOR COALITIONS AND CENTRES OF EXCELLENCE

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Research reactors continue to play a key role in the development of peaceful uses of atomic energy. They are used for a variety of purposes such as education and training, production of medical and industrial isotopes, non-destructive testing, analytical studies, modification of materials, for research in physics, biology and materials science, and in support of nuclear power programmes. The IAEA Research Reactor Data Base lists about 250 operational research reactors worldwide, many of which have been operating for more than 40 years.

Through both statistical and anecdotal evidence, it is clear that many of these reactors are underutilized, face critical issues related to sustainability, and must make important decisions concerning future operation. These challenges are occurring in the context of increased concerns over global non-proliferation and nuclear material security, due to which research reactor operators are coming under increased pressure to substantially improve physical security and convert to the use of low enriched uranium (LEU) fuel. Thus, there is a complex environment for research reactors, and one in which underutilized and therefore likely poorly funded facilities invoke particular concern.

Many research reactors are challenged to generate sufficient income to offset operational costs, often in a context of declining political and/or public support. Many research reactor operators have limited access to potential customers for their services and are not familiar with the business planning concepts needed to secure additional commercial revenues or governmental or international programme funding. This not only results in reduced income for the facilities involved, but sometimes also in research reactor services priced below full cost, preventing recovery of back-end costs and creating unsustainable market norms.

Parochial attitudes and competitive behaviour restrict information sharing, dissemination of best practices, and mutual support that could otherwise result in a coordinated approach to market development, building upon strengths of various facilities. Moreover, belief that the markets for research reactor products and services are a "zero-sum" game, with market gains by one research reactor coming at the expense of another facility, result in a general lack of openness within the research reactor community.

Yet there is evidence to suggest that the market for research reactor services is supply limited, rather than demand limited. A number of factors limit the ability of research reactors to expand their user base and to generate new sources of revenue:

- Many potential customers do not know how, or where, to contact the research reactor community, and have only limited knowledge or awareness of the range of research reactor services, equipment and locations available.
- The standards of quality control and quality assurance between research reactors are not uniform, impede business development, and may result in a lack of confidence in service reliability. As a consequence, customers need to conduct due diligence for each facility to be used, reducing the enthusiasm and financial rationale for developing additional sources of supply.
- Transport of radionuclides is becoming increasingly difficult, with examples of shipments held in customs, prevented from leaving the country of origin or from entering the customer destination, and requires specific expertise and experience to manage this issue.

In order to address the complex of issues related to sustainability, security, and non-proliferation aspects of research reactors, and to promote international and regional cooperation, the IAEA is initiating the Research Reactor Coalitions and Centres of Excellence initiative. This activity is supported by a two-year grant from the Nuclear Threat Initiative, Inc. (NTI), and by a 2007-2008 IAEA Technical Cooperation Project, "Enhancement of the Sustainability of Research Reactors and their Safe Operation Through Regional Cooperation, Networking, and Coalitions" (RER/4/029). These two activities will work in an integrated manner, along with other relevant national and regional IAEA Technical Cooperation projects and complementary IAEA regular and extra-budgetary funded programme activities in research reactor utilization, safety, security, and the fuel cycle. These activities were endorsed by the IAEA Board of Governors in its March 2007 meeting which encouraged regional cooperation and networking among research reactors.

The aim of this initiative will be to establish a pilot project involving the formation of at least one voluntary, subscription-based, self-financed coalition of research reactor operators (possibly including other participants, sponsors, etc.), which may serve as a model for the establishment of additional coalitions.

IAEA's Subprogramme on Research Reactors: Technology and Non-Proliferation

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For nuclear research and technology development to continue to advance, research reactors (RRs) must be safely and reliably operated, adequately utilized, refurbished when necessary, provided with adequate proliferation-resistant fuel cycle services and safely decommissioned at the end of life.

The IAEA has established its competence in the area of RRs with a long history of assistance to Member States in improving their utilization, by taking the lead in the development of norms and codes of good practice for all aspects of the nuclear fuel cycle and in the planning and implementation of decommissioning. The IAEA Subprogramme on RRs is formulated to cover a broad range of RR issues and to promote the continued development of scientific research and technological development using RRs. Member States look to the IAEA for coordination of the worldwide effort in this area and for help in solving specific problems.

The IAEA coordinates and implements an array of activities that together provide broad support for RRs. As with other aspects of nuclear technology, RR activities within the IAEA are spread through diverse groups in different Departments. To ensure a common approach a Cross-Cutting Coordination Group on Research Reactors (CCCGRR) has been established, with representatives from all departments actively supporting RR activities.

Utilization and application activities are generally lead from within the Department of Nuclear Applications (NA). With respect to RRs, NA is primarily carrying out IAEA activities to assist and advise Member States in assessing their needs for research and development in the nuclear sciences, as well in supporting their activities in specific fields.

Safety and Security aspects of RRs are handled by the Department of Nuclear Safety and Security (NS).

The technological, fuel cycle and operational aspects of RR management are supported by the Department of Nuclear Energy (NE). NE is primarily working to support RR organizations in their pursuit of often diverse strategic objectives within the context of modern RR operational constraints. Today RR operating organizations must overcome challenges such as the ongoing management of ageing facilities, pressures for increase vigilance with respect to non proliferation, and shrinking resources (financial as well as human) while fulfilling an expanding role in support of nuclear technology development.

In addition, the Department of Nuclear Safeguards is responsible for the control of the fissile material for RR and the Department of Technical Cooperation (TC) supports RR activities for the principal benefit of RRs in developing countries. TC is subsequently supported by NA, NS, and NE who assist in the development and implementation of relevant TC projects within their specific fields of expertise.

The Subprogramme on RRs is under IAEA's Programme D on Nuclear Science. Implementation of the IAEA Subprogramme on RRs (IAEA code D.2) is shared between NE and NA while separate subprogrammes, managed by NS, deal with RR safety and security. In this paper, only the activities managed by NE and NA under the subprogramme on RRs are presented, including a complete description of the ongoing projects and planned activities for the years 2008-2009. Special emphasis is put on new international collaborative undertakings, like the IAEA's Technical Working Group on RRs.

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Research reactor utilization in the Mediterranean region
The experience of Montenegro and possibilities for future cooperation

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Being a small developing country, Montenegro is hardly advanced when research in nuclear sciences is concerned. The latter is limited to two institutions: Centre for Eco-Toxicological Research of Montenegro, Department of Radiation Protection and Monitoring (CETI) and University of Montenegro, Faculty of Sciences, Department of Physics (FS), both in Podgorica, the capital. Strange enough, even though there has never been a research reactor in the country, there is quite some tradition and experience with RR utilization, particularly with reactor neutron activation analysis (RNAA). Namely, already from the years 70's onwards, physicists from FS were educated&trained in prominent nuclear research centers in former Yugoslavia (Vinca NRI near Belgrade; Jozef Stefan NRI, Ljubljana, Rudjer Boskovic NRI, Zagreb) and abroad (University of Ghent NRI, Belgium; Dubna NRI near Moscow; University of Charlottesville RR, USA; NIST, Gaithersburg, USA; Rez NRI near Prague; Saclay NRI near Paris, Forschunszentrum Garching near Munchen, Germany, etc.).

Quite extensive international scientific cooperation existed on various RNAA topics under national and international reserch projects: environmental pollution, geological, pedological and metallurgical studies, reactor neutron flux characterization, RNAA nuclear data standardization, k0-RNAA method development, semiconductor detector gamma-spectrometry software optimization, true coincidence effects, etc. - the output being counted by hundreds of publications, reports, studies, Ph.D. theses, etc. Some of them are listed below [1-11]. In addition, ANGLE software for semiconductor detector gamma-efficiency calculations, developed by nuclear spectrometry group at the University in Podgorica, Montenegro, is nowdays in use in numerous gamma spectrometry (including NAA) laboratories all around.

Unfortunately, these activities practically collapsed in the 90's, following long lasting political and economical crisis in the country. Recently, former research cooperation links are being re-established and new ones created, to mention just the two in the Balkan region involving RR utilization: with "Jozef Stefan" Institute, Ljubljana, Slovenia, Environmental Sciences Department, k0-RNAA group (topic: gamma-spectrometry software upgrading) and "Demokritos" Research Institute, RR Laboratory, Athens, Greece (topic: bulky samples RNAA characterization). There is also a formal cooperation agreement with Nuclear Research Institute of the Bulgarian Academy of Sciences, Sofia, Bulgaria (topic: environmental pollution measurements and monitoring, however no RR utilization included up to now).

Experience of Montenegro in research reactor utilization through international cooperation, in particular in the Balkan/Mediterranean region, was presented during IAEA Technical Meeting on Strategic Planning for Sustainability-Mediterranean Region - Research Reactor Utilization, Vienna, March 19-22, 2007. On this ocassion Montenegro was pointed out as a good example of how a “no-RR country” could incorporate well into RR community. It was subsequently proposed that Montenegro coordinates the activities on initiating/formulating a regional (mediterranean) IAEA project on neutron activation analysis selected topics. The latter will include i.a. preparation of reference materials, irradiation under controlled conditions, measurements, software assessment and data interpretation. Applications in areas like Environment, Geology, Health, Agriculture & Cultural Heritage are particularly emphasised.

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Role of the Oarai Branch as a facility open for university researchers in utilization of research reactors

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Research reactors incarnate an invaluable theater for materials irradiation, from fundamental points of views, namely, its relatively homogeneous flux distributions, long and stable operations, controllability of irradiation atmospheres, and an electronic-excitation-and-nuclear-displacement ratio relevant to actual application conditions. In the meantime, researches utilizing fission reactors are expensive and time&manpower consuming, which could be affordable only for a long-term and large scale project. The Oarai Branch of Institute for Materials Research in Tohoku University (Hereafter denoted as the Oarai Branch) has been acting as a coordinator for university researchers' utilizing fission reactors for their fundamental studies, under the close collaboration with Japan Atomic Energy Agency (JAEA). Related research fields extend into a variety of topics such as cosmological and geological age estimation, detection of a trace amount of actinide elements in highly pure semi-conductors, materials issues related with safety and extension of life of water cooled power reactors, and developments of materials for advanced reactors including a next generation fission reactors and nuclear fusion reactors. The paper will report a role of the Oarai Branch in utilizing research and test reactors and will describe present status of university-related research activities utilizing reactors. A variety of researchers in universities have a variety of interests in utilizing research reactors such as Japan Materials Testing Reactor (JMTR; water cooled mixed spectra reactor) and JOYO (sodium cooled high flux fast reactor). University researchers of about 2000-3000 man days are visiting the Oarai Branch every year to carry out their own research projects.

For utilization of research reactors for fundamental studies, some irradiation techniques and intimate linkages between reactors and post irradiation facilities(PIFs) are needed to develop. The JAEA is planning to establish a comprehensive framework for advanced reactor-irradiation-studies in the Oarai area, whose details will be described in a separate paper. In this framework, the Oarai Branch is planning to play an unique role not only for convenience of university researchers but also for some large-scale national projects which will be managed mainly by the JAEA and industries. Typical Examples are prompt transportation of irradiated specimens to PIFs, and prompt implementation of post irradiation examinations (PIEs). Shortening of an iteration period from specimen preparation to implementation of PIEs is essential for the university researchers, where education of post-graduate students in a defined period is one among major targets. To realize this, legitimate and prompt transfer of irradiated specimens from the JAEA to the Oarai Branch is established with needed software and hardware. Examples will be a development of shuttle irradiation rig for the JOYO irradiation. Other roles which the Oarai Branch tries to take an initiative are development advanced analyzing instruments for studies of nano-scale evolution of radiation induced microstructural changes and development of advanced irradiation techniques. Finally, it is essential for fundamental studies of university researchers to control irradiation conditions such as a temperature and an atmosphere, independently of a reactor operating mode and to monitor irradiation conditions and some property changes of materials under irradiation in-situ. By 1980s, no-instrumented rigs were dominant for simple irradiation of materials with possible neutron fluences. In late 1980s, it was seriously recognized through accumulated studies that well controlled reactor irradiation is inevitable to make reactor irradiation studies compatible with advanced material science. Since then, specially designed irradiation rigs were developed. One among most important topics is a temperature control being independent of a reactor operation mode. Usually, temperatures of irradiated specimens were seriously influenced by a nuclear heating (a gamma-ray dose rate), which is dependent on a reactor power. So, an irradiation temperature will vary when a reactor power varies, especially at transient periods of reactor operations such as a startup and a shutdown. It was clearly revealed that the variation of irradiation temperature will result in very complicated evolution of radiation induced microstructures, which could not be rationally understood from fundamental points of view. In-situ measurements of some properties of irradiated materials under reactor irradiation will be another topic. Electrical and optical measurements were realized in JMTR, where radiation induced electrical conductivity (RIC) of ceramic insulators were systematically measured and radiation induced luminescence from ceramics was measured through radiation resistant optical fibers during JMTR irradiation.

Now, the JMTR is under refurbishing and a new framework is under survey for its utilization more effective and more convenient for customers. The Oarai Branch is planning to reorganize its role in utilizing research reactors for university researchers under close collaboration with JAEA and university-research-network for utilizing research reactors.

Session 7: Regulatory Aspects and Experience with Current Research Reactor Issues Including Safety Aspects of Core Conversion

Synopses no. IAEA-CN-156/	**Synopses Title**	**Main Author**
S-29	Canadian Experience in Implementing Modern Regulations to Existing Research Reactors	Alwani, A.
S-31	The Role of Regulatory Authority in Safe Operation of Research Reactors	Mikulski, A.T.
S-33	Causal Factors Guide For The Evaluation Of Accidents In Research Reactors	Perrin, C.D.
U-30	Contribution of Research Reactors to the Programmes for Research and Technological Development on the Safety	Couturier, J.
S-59	Review of the ANSTO Application for a Facility Licence to Operate the OPAL Research Reactor in Australia: Case study review of operational readiness	Ward, J.S.
S-38	Research Reactors in Germany: An Overview	Schneider, M.
S-42	The French approach for the regulation of research reactors	Conte, D.
S-30	MARIA research reactor conversion to LEU fuel	Krzysztoszek, G.

Canadian Experience in Implementing Modern Regulations to Existing Research Reactors

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Research reactors in Canada are regulated by the Canadian Nuclear Safety Commission (CNSC). The CNSC is an independent agency of the Government of Canada and operates in a transparent manner. The mission of the CNSC is to regulate the use of nuclear energy and materials to protect health, safety, security and the environment and to respect Canada's international commitments on the peaceful use of nuclear energy.

Currently, there are seven research reactors in operation in Canada: The National Research Universal (NRU) reactor, 135 MW; McMaster Nuclear Reactor (MNR), 5 MW, and five Safe Low-Power Kritical Experiment (SLOWPOKE) reactors, 20 kW. This is in addition to two Multipurpose Applied Physics Lattice Experiment (MAPLE) reactors, 10 MW which are in the commissioning stage.

The operating research reactors in Canada have a long history of operation and established safety track records. The NRU was build in 1957, the MNR in 1959 and the SLOWPOKEs in the seventies. The Atomic Energy Control Board, the predecessor of the CNSC, was the national regulatory body in Canada since 1946 and regulated the research reactors among other nuclear facilities and materials for decades.

In May 2000, the Nuclear Safety and Control Act replaced the Atomic Energy Control Act of 1946 with a modern statute that reflects public expectations for the regulation of nuclear energy. The new Act provides for more explicit and effective regulation of nuclear activities. It ensures high standards in the areas of health, safety, security and protection of the environment, sustaining our environment and for ensuring a modern regulatory regime to meet the needs of the 21st century.

While the licensing regime for nuclear facilities including research reactors continues with the new Act, several changes occurred to the requirements and the process. Both the licensee and the regulator made particular efforts to face the challenges of defining the implications of the new requirements, managing the expectations, and bringing the licensed activities up to compliance with the new standards.

One main aspect of the modern regulatory practice of the CNSC is to focus on programs to be in place to address safety areas. Examples of these areas are training, criticality safety, and fire protection. These aspects have always been subject to a continuous regulatory oversight. However, the CNSC now expects the research reactor licensees to have formalized documented programs in place to address these aspects. Several regulatory guidance and standard documents have been published since 2000 and which provide recommended or approved methods to formalize the safety area programs.

Another change with the modern regime is the requirement that research reactor licensees, among other large facilities' licensees, establish financial guarantees to ensure that costs of decommissioning their nuclear facilities, here research reactors, are borne by the licensees and not by the taxpayers.

Quality Management in another area of new requirements formally established with the new Act and the regulations pursuant to the Act. The requirements to have a quality management system in place are now standard to all research reactor licensees. Associated with it are incident investigation process and capabilities as well as human factors engineering reviews.

The implementation of the modern regime presented a number of challenges to the research reactor facilities. These facilities have usually small organizations and fewer resources to devote to address the new demands. Also, with the mature operations and the established good performance and safety records, the new requirements may be perceived as an added burden without additional safety benefits.

The CNSC approach is characterized by the following:

- Risk Informed Approach: Many CNSC requirements were originally designed for nuclear power reactors. They have been tailored to research reactor to take into account the smaller risks imposed by the research reactor facilities. The graded risk approach has been used to define both the applicability of specific requirements and the acceptance criteria.

- Transitional Periods: The CNSC recognizes the need for transitional periods for rolling in and implementing the various new requirements. Transitional periods for full compliance with each requirement are imposed after careful assessment of the priority and feasibility of the required action. This is to ensure that the upgrade to the current standard is done within a reasonable time frame and also effectively.

- Compliance Promotion: As expected in any change, there is a need to communicate fully with the licensees and provide the rationale for each new requirement. The notion that issuance of a new requirement by the CNSC does not mean that what was safe yesterday is not safe today, is explained fully to the licensees. All is aimed toward a common goal of ensuring that the Canadian worker, public and environment are protected from undue risk from research reactors by adapting the best safety standards.

The Role of Regulatory Authority in Safe Operation of Research Reactors

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To be presented at
International Conference on Research Reactors: Safe Management and Effective Utilization
5-9 November 2007, Sydney, Australia

EXTENDED SYNOPSIS

The safe and secure operation of research reactors depends on several factors. One of them is the activity of local (country) regulatory authority. It is not enough to follow the well known international guidance, e.g. the IAEA Code of Conduct for research reactors, Safety Guidance, etc. but training and experience of inspectors contributes significantly in this field. The activity of regulatory activity should be concentrated in many directions, as follows:
(1) issuing licences for reactor operation,
(2) giving permission for technological or procedural changes in operating instructions,
(3) requiring and/or authorizing technical modifications,
(4) verification of reports submitted by operating staff,
(5) performing own analysis also by invited specialists or experts.

The paper summarises the activity of Polish regulatory authority in this field during the last five years and is referred to the MARIA research reactor. The reactor is used mainly for isotope production and experiments in nuclear physics. It is in operation since more than 30 years and undergoes systematic improvements. It is operated on a one-week long cycles with two longer breaks per year for regular overhauls made by reactor staff. Regulatory authority requires detailed plan of the work to be performed and verifies results before starting of new period of reactor operation.

The reactor undergoes several modification and technical improvements. They are prepared by reactor staff and approved by regulatory authority. In case of more complicated improvements or requiring detailed analysis, especially numerical calculations, they are analysed by independent experts, mainly from technical universities. The most important examples of such improvements are:
(a) transformation from HEU to MEU fuel (from 80 to 36% enrichment, which were performed in years 1999-2003 and required installation of a special fuel assembly equipped with thermocouples in order to performed verification of thermo-hydraulic calculations),
(b) preparation for transformation to LEU fuel (less than 20% enrichment, to be started in year 2007),
(c) agreement for decrease number of heat exchangers in fuel channel cooling system (from 4 to 3 exchangers in a case when one is leaking),

(d) restoring of cooling pipes connecting two fuel channels (dismantled many years ago in order to place a special test loop),
(e) modernization of dosimetric system,
(f) installation of new system for measurement of technological parameters,
(g) preparation of new locations in a reactor core for isotope irradiation in higher neutron flux,
(h) improving of cooling conditions for a natural convection during a decay heat removal from fuel elelemnts, etc.

The other role of regulatory authority is suggesting technological changes and improvements in safety equipment, to be done in future as: modernization of vibration monitoring system, development of more friendly procedures for visualisation of information stored by signalling system, etc. The authority has some funds for this activity, which are coming from a state budget and granted once a year.

According to Polish Atomic Law, reactor operator is obliged to present quarterly reports of reactor operation and send an immediate notification in case of any abnormal situation, including shortening of operation cycle. These reports are carefully analysed and in many cases used as suggestions for improvements in operating instruction or installing new equipment. The reactor operator is obliged to submit the actual version of a Safety Analysis Report before applying for a new operting licence, which is granted for period specified in an appl;ication and this period was equal to five years in the past.

All these problems will be described in detail, having in mind an experience of almost 30 years of the MARIA reactor operation in Poland and may be of interest for regulatory authorities in other countries.

IAEA-CN-156/S-33

Causal Factors Guide For The Evaluation Of Accidents In Research Reactors

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ABSTRACT

In the field of radiological and nuclear safety, the Nuclear Regulatory Authority (ARN) of Argentina controls three research reactors and three critical assemblies, by means of evaluations, audits and inspections, in order to assure the fulfillment of the requirements established in the Licenses, in the regulatory standards and in the mandatory documentation in general.

From the Nuclear Regulatory Authority point of view, within the general process of research reactors safety management, the management of operating experience plays an outstanding roll.

In this aspect the ARN has established specific requisites in the Operation Licences in relation to the communication, evaluation, investigation of causes, and adoption of corrective measures, for the happened events.

From the experience collected in the analysis of the reports sent by the operators it has been verified some weakness in relation to the methodology of analysis of events and in the determination of the causal factors.

In such a sense, with the purpose to establish a help for the analysts and to homogenize the treatment of the events, two reference guides were designed: a guide for the evaluation of events and another with a grid of causal factors

This paper describes the main aspects of the operating management system for research reactors and critical assemblies in Argentina, and the guides developed for the event analysis and determination of causal factors.

Contribution of research reactors to the programmes for research and technological development on the safety

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The most significant current use of nuclear reactors is for the generation of electrical power, for instance with pressurized water reactors (PWR) which constitute the most commercial type of reactor. Nevertheless, since the beginning of nuclear activities, important applications of nuclear reactors are research applications including: the providing of neutrons source for the basic research (beamline experiments), the development of nuclear technology, and of new types of reactors, for example to bring solutions of the radioactive waste management. Moreover, nuclear reactors are used for training, nuclear propulsion, production of radioisotopes for medicine and industry.

Therefore, the research reactors constitute major equipments to the scientific and technological research. Research is of benefit for both reactors currently in operation and for future reactors as the phenomena occurring during a severe accident are the same. Nevertheless, if severe accidents were not taken into account in the initial design of nuclear power plants currently in operation, improvements have been made in France in terms of prevention and limitation of consequences.

The objective of the present article is to illustrate the contribution of the research reactors to the programmes for research and technological development led to increase the safety of the nuclear power plants, more particularly in the field of the accidents of insertion of reactivity (RIA), loss of primary coolant (LOCA) and partial or total melting of the core; indeed, the physical phenomena occurring during a severe accident are extremely complex. Research aims to better understand these phenomena and to reduce the associated uncertainties in order to assess the extent to which the state of acquired knowledge can be used to make reliable predictions. The examples will concern accidents that may occur in pressurized water reactors and in sodium fast reactors (SFR).

Some given examples will show the fundamental importance of experiments in research reactors to establish and support the safety demonstrations; indeed, such in-pile tests:

- allow to characterize badly known physical phenomena,
- can put in evidence some not foreseen phenomena,
- contribute *in fine* to the development and in the global validation of the codes used in the safety demonstrations, such as ASTEC (Accident Source Term Evaluation Code) used to simulate an entire accident from the initiating event to the possible radionuclide release outside the containment.

Regulatory Review Methods: Review of the ANSTO Application for a Facility Licence to Operate the OPAL Research Reactor in Australia: Case study review of operational readiness

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This paper will examine the review methods used by assessors of the Australian Radiation Protection and Nuclear Safety Agency in advising the CEO of ARPANSA on the application by the Australian Nuclear Science and Technology Organisation (ANSTO) for a facility licence authorising it to operate the OPAL reactor. In particular it will focus on an aspect of that review that considered the operational readiness demonstrated by ANSTO in relation to the OPAL reactor as an aspect of the review of safety of the OPAL reactor.

The review of operational readiness examined the extent to which the managerial, procedural and administrative controls that were proposed for the reactor were appropriate and the extent to which they demonstrated underlying support for the operation of the reactor.

The review utilised guidance from International Atomic Energy Agency documents on the operations of research reactors and also an internal ARPANSA regulatory guide that has been based on a review of current international best practice in nuclear safety in this area. The review in particular focused on the adequacy and maturity of the information on matters such as effective control, safety management, radiation protection, radioactive waste management, ultimate disposal, security and emergency planning in relation to the OPAL reactor.

The paper will focus in particular on the manner in which each of these plans and arrangements were reviewed and the overall importance that was placed on this review in the regulatory decision making process undertaken by ARPANSA. Key themes that will also be explored in the paper are:

- The development and implementation of ARPANSA regulatory guidance, its basis in key international guidance and its use in undertaking regulatory reviews;
- The importance of key safety management plans and the review of those plans in establishing regulatory confidence in the operator.
- The importance of open and questioning safety management policy and procedures, with the development of a set of safety performance indicators (SPI) that can provide the regulatory body with assurance of nuclear safety.
- The challenge of assurance for the regulator when systems are new and untested, and staffing profiles are under development

Research Reactors in Germany: An Overview

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In Germany, substantial experience was gained in the field of research reactors during the last five decades. In this paper, an overview about the legislative and regulatory framework in Germany is given particularly with respect to research reactors, as well as a survey of the plant and licensing status of the facilities in Germany.

A comprehensive legislative and regulatory framework is established to govern the safety of nuclear installations in Germany. The various nuclear safety regulations are structured hierarchically. At the top level, the Atomic Energy Act comprises the general national regulations for the safety of nuclear installations in Germany and constitutes the basis for the associated ordinances, e.g. the Radiation Protection Ordinance or the Nuclear Licensing Procedure Ordinance. The Atomic Energy Act and the ordinances are applicable to all kind of nuclear installations, and therefore are applied to nuclear power reactors as well as to research reactors in a common approach. After the amendment of the Atomic Energy Act in 2002, one of its purposes is to phase out the use of nuclear energy in a controlled manner. In fact, this is in force for nuclear power reactors for commercial generation of electricity, but not for research reactors.

Below the legal level, the safety provisions and regulations of the Atomic Energy Act and its ordinances are put into concrete terms by general administrative provisions, by regulatory guidelines, by safety standards of the Nuclear Safety Standards Commission (KTA), by recommendations from the Reactor Safety Commission (RSK) and the Commission on Radiological Protection (SSK), and by conventional technical standards (e.g. DIN, ISO, IEC). The nuclear safety regulations concerning nuclear installations are in compliance with the international accepted safety standards, e.g. the “Safety Fundamentals” of the IAEA.

The regulatory guidelines and the KTA Safety Standards are mainly developed for nuclear power plants. In practice, they are applied in analogy or with some interpretation for research reactors, in accordance with the potential hazards of the specific research reactor by means of a graded approach. There are only very few regulations implemented specifically for research reactors, e.g. two regulatory guidelines relating to the technical qualification of research reactor personnel or one KTA Safety Standard relating the monitoring of the discharge of radioactive substances from research reactors. Moreover, some recommendations from the RSK regarding specific licensing procedures of individual research reactor facilities have been made.

In 2004, the IAEA Board of Governors approved the “Code on Conduct on the Safety of Research Reactors”. To comply with its recommendations, the German Federal Government included research reactors in the report for the third review meeting in April 2005 [1] for the “Convention on Nuclear Safety”.

According to the Atomic Energy Act, a licence is required for the construction, operation or commissioning of research reactors. The licensing procedure and the continuous regulatory supervision of the facilities lies within the responsibility of the individual Federal States ("Länder"). To preserve the legal uniformity for the entire territory of the Federal Republic of Germany, the Federal Government supervises the licensing and supervisory activities of the "Länder"-authorities regarding lawfulness and expediency.

In Germany, a total of 46 research reactors were built and operated. In the meanwhile, most of them are in decommissioning or have already been dismantled completely. Concerning the design, there is, or has been, a very broad range of different types of research reactors. The variety of facilities includes large pool or tank reactors with a thermal power of several tens of megawatt as well as small educational reactors with a thermal power in the order of only hundred milliwatts.

At present, 13 research reactors are still in operation in Germany. The newest facility, the high flux neutron source FRM-II, became critical for the first time in March 2004 and started routine operation in April 2005. As an example, an overview about the licensing procedure and the specific plant characteristics of this facility will be given in the presentation.

The actual decommissioning and dismantling projects comprise 9 research reactors. Furthermore, 24 facilities have been dismantled completely and the sites are released from regulatory control. Examples for the regulatory procedure of decommissioning projects will be given also in the presentation.

[1] Federal Ministry for the Environment, Nature Conservation and Nuclear Safety, Convention on Nuclear Safety, Report by the Government of the Federal Republic of Germany for the Third Review Meeting in April 2005, Bonn (2004)

The French approach for the regulation of research reactors

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There are several types of research reactor currently in operation in France. Their usage includes neutronic studies, technological irradiations, neutron beam utilisation, safety research and teaching purposes. Most of these were built during the 60s and because they are all different each type presents particular hazards. The French Nuclear Safety Authority (ASN) works to ensure that the regulatory framework maintains a high level of safety for research and experimental activities.

After a description of French research reactors and their hazards and operation this paper will summarise the French regulatory framework and its evolutions over the last few years. For instance, the law of June 13th 2006 on Transparency and Security in Nuclear Field is now the most important piece of French legislation in the field of nuclear safety. It builds upon the high requirements of the existing regulatory framework and, sets a new base for the control of nuclear activities and facilities. Also it creates an independent nuclear safety authority (ASN) and includes the principle of periodic (10 yearly) safety reviews for all nuclear facilities.

The safety analysis and methods of control of research reactors safety have tended to become more and more similar in France to those of power reactors. For instance, even if a graduated approach has always been used, the safety analysis approach applied to operating conditions on research reactors is the same as that used for power reactors. Moreover, design codes used are often the same. The French regulatory framework is applicable to all nuclear facilities, including research reactors. To comply with the law and this regulatory framework, each facility must deliver a safety analysis report (SAR) which determines its particular operating limits. This analysis is assessed by ASN and its technical support organization, IRSN (Radioprotection and Nuclear Safety Institute).

To ensure that the licensee assume all its responsibilities and, to allow the necessary flexibility in the ever changing operations of research reactors, ASN has utilised the principle of internal authorizations. Even if the operation is not explicitly described in the SAR, the licensee has the opportunity, under certain conditions, to self authorize the operation provided that it is of minor safety significance and is bounded by another operation in the SAR. This principle has been applied to the use of experimental devices in specific conditions. Since its introduction in 2002 the internal authorization process has generated a significant amount of plant operations information which has been subject to review and feedback by both the licensee and ASN. ASN is about to extend its application to core management the long stop periods, especially for refurbishment in research reactors.

This paper then deals with the feedback provided over the last few years from events reported at French nuclear research reactors. There has been a significant increase in reported events between 1999 and 2006. Last year (2006) 29 events were reported on research reactors in France. Most of these were rated at level 0 and are minor events but this relatively high number is probably due to the ageing of many facilities. Nevertheless, it’s also the result of the introduction of new criteria for declaration for instance when an automatic shutdown occurs. However this increase in the number of events provides the opportunity for analysis to acquire more knowledge about research reactors operation and also an opportunity to tackle the subject of the ageing of those facilities.

Research reactors are essential support tools for the nuclear industry and for the design of the next generations of power reactors. Thus it is important to keep these facilities operational. To conclude, this paper give ASN's perspectives for research reactors for the coming years and also highlight the regulatory challenges associated with keeping those facilities operational with a high level of safety. The regulation of the projects RJH (Jules Horowitz Reactor), ITER (International Thermonuclear Experimental Reactor) or the next prototype of 4^{th} generation reactor will be one of those challenges.

Maria research reactor conversion to LEU fuel

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The multipurpose high flux research reactor MARIA is a water and beryllium moderated reactor of a pool type with graphite reflector and pressurised fuel channels. The main areas of reactor application is production of radioisotopes and physics research with usage of a neutron beam from horizontal channels.

Conversion programme of MARIA research reactor is closely associated with the "Global Threat Reduction Initiative" (GTRI). The MARIA reactor is a Russian-designed research reactor that operates with Russian highly enriched uranium (HEU) fuel. The reactor initially was operated with 80% enriched HEU and converted to 36% HEU in 1999. The programme objective is to convert the current core to LEU fuel. The proposed silicide fuel (U_3Si_2) with a density of 4,8 g/cm^3 has been qualified to achieving very high burnups. The assembly design proposed for use in the MARIA reactor will be unique, however, it needs to be qualified for the MARIA operating conditions by irradiating two lead test assemblies in the MARIA reactor.

The Institute of Atomic Energy (IAE) have prepared a neutron-physics and thermo-hydraulic analysis [1], [2], [3]. Also the IAE with cooperation of the BR-2 reactor specialist elaborated the final fuel specification for fuel procurement contract. This document contains the specification and detailed design of fuel element, mechanical, chemical and thermal constraints, materials specification, design of the laminated fuel as well as drawings and tubes specification. Under the IAEA bidding process the technical evaluation team's conclusion was that the CERCA and INVAP are qualified to manufacture U_3Si_2 fuel elements with flat or slightly curved plates at densities of 4.8 g/cm^3.

The first stage of LEU fuel qualification will include testing of mockup assemblies. IAE plans to perform hydraulic testing to measure the pressure drop through the assembly and thermal extention of tubes.

The scope of work required for the preparation of the Safety Analysis Report (SAR) for the insertion and irradiation of lead test assemblies (LTA) in the irradiation of lead test assemblies (LTA) in the conversion to LEU was developed.
The following transients analysis and specification of limits are required:

1. Decrease of core cooling capability
2. Deterioration of the cooling feasibility by the secondary circuit
3. Insertion of Positive Reactivity and Power fluctuation
4. Failures of the core structural components or experimental equipment
5. Partial Meltdown of core
6. Safety Limits for forced flow of coolant in primary circuits
7. Safety Limits at natural Convection of water in the fuel channel

8. Safety Thresholds at forced flow of coolant in the primary circuits.

The main stage for LEU fuel qualification will be irradiation of the LTAs in the MARIA core. Irradiation of the LTAs to the burn-up for qualification (~ 60%) is expected and it will take about 18 months. During the time of irradiation there will continuous monitoring of LTAa by means of the Radiation Monitoring System (RMS) to detect a fission product release based on delayed neutron measurements. On achieving the required burn-up the LTA's will be discharged from the reactor core and subjected to the examination in the storage pool. Within the PIE the gaps between individual fuel tubes excluding the possible swelling of the fuel elements will be measured.

The affirmative results of the aforementioned examination will allow for accomplishing the core conversion of the MARIA reactor by successive replacement of the fuel channels with high enrichment by the fuel channels with low enrichment.

[1] Andrzejewski, K. et al., „Neutron-physics characteristics of the U_3Si_2 fule with 19.7 % enrichment", Report IEA Nr: B-12/2005

[2] Andrzejewski, K. et al., „Comparision of neutron-physics characteristics for two types (MR-6 80%/350 and 36%/540) of reactor MARIA fuel and considered U_3Si_2 fuel elements with 19,75 % enrichment", Report IEA Nr: B-23/2005

[3] Bykowski, W. et al., „Thermal characteristics of heat exchangers system of MARIA reactor fuel channels primary cooling circuit", Report A IAE 124/A

Session 8: Specific Utilization Applications

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Experience with different methods for on- and off-line detection of small releases of fission products from fuel elements at the HOR

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During the HEU to LEU-fuel conversion of the Hoger Onderwijs Reactor (HOR) we encountered four cases in which the fission product concentration in the pool water increased significantly. Although this increase never caused any exceeding of intervention levels as they are mentioned in our permit and safety analyses, the ALARA principle urged us to analyse the situation in more detail. All cases have been related to a failure of the cladding of four elements respectively, all from the first batch of LEU fuel. Once an element has been identified to have a defect, it was removed from operation. The first two LEU cases are described in [1]. In this paper we present the strategy we used to handle such cases.

Already before having these cases, different detection techniques were used for on- and off-line determination of (increased) activity within the reactor pool and containment, such as:

- Automated dose-rate measurement in containment;
- Direct integral measurement of deposits in ion-bed exchanger;
- Cooling water delayed neutron activity detection;
- Off-line analyses of ion-bed exchanger;
- Off-line analyses of pool water at the end of a production week;
- Direct measurement of air-borne activity concentration just above the reactor pool;
- Measuring primary cooling water delayed neutron activity;
- Contamination check at the exit of the containment.

In our effort to obtain fast and reliable indicators for the reactor core performance, we developed the following instruments in addition to the listed techniques:

- A wet sipping device (NIP) to determine the fission product ‘leak rate’ of a separated, isolated element at suspense. Furthermore the idea is to set a reference level for every element and to detect minor anomalies in the ‘leak rate’ at an early stage by regularly re-measure it during the lifetime.
- A model to relate the pool water activity with the mean core fission product ‘leak rate’. This approach turns out to be a sensitive and reliable indicator even at very short reactor operation times as a few hours. Understanding the different factors from this analyses and combining it with the merits of the above mentioned direct air system led to the development of a new device.
- The ‘RID-cascade’, which measures on-line the air-borne activity concentration as it builds up from a direct coolant water-air interface. Important aspects of the device are: high efficiency for fission or decay products, easy maintainability and easy to operate.

The paper will give an overview of the different methods for on-line and off-line detection of activity measurement devices used at the HOR. The newly installed techniques with their pro's and con's are discussed in more detail and a precise description of the 'RID-cascade' is given with the gained experience of this new instrument.

[1] Verkooijen, A.H.M. and J.W. de Vries, "Experience with MTR fuel at the HOR", Kerntechnik 69 (2004) 3.

Neutronic Analysis for the Fission Mo-99 Production by Irradiation of a LEU Target at RECH-1 Reactor

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ABSTRACT

The RECH-1 is a 5 MW open pool type research reactor, cooled and moderated by light water, reflected by beryllium, using MTR type fuel with low enriched uranium. The reactor is operated by the Chilean Nuclear Energy Commission and is located at La Reina Nuclear Center. The LEU fuel elements were built by the Chilean Fuel Fabrication Plant with a uranium density of 3.4 g/cm^3. The technical specifications of these fuel elements were developed by the Chilean Manufacturer based on the original HEU assembly and approved by the reactor operator. The present core configuration, the first one with LEU fuel elements only, was configured in May of 2006 and has 32 fuel elements containing U_3Si_2-Al.

The Chilean Nuclear Energy Commission has initiated studies to produce fission ^{99}Mo by the irradiation of a low enrichment uranium target at RECH-1 reactor, thanks to the IAEA Coordinated Research Project (CRP) "Developing Techniques for Small Scale Indigenous Molybdenum 99 Production using LEU Fission or Neutron Activation", supported by ANL, USA. This target is made of a foil of 13 gram LEU metallic uranium inserted between two concentric aluminium cylinders.

Results for the neutronic and activity calculations are presented, taking into account different irradiation and decay times. A criticality safety analysis for the storage of the originating radioactive wastes of the process was performed using MCNP code. The neutronic calculations were carried out for the present core configuration of RECH-1 reactor using WIMS and CITATION codes, supposing that the target would be introduced in the reactor grid position of the maximum thermal neutron flux. Reactivity change due to target loading has been examined to confirm the reactor safety. The mean neutron thermal flux and the power generated for the target were calculated.

With the target power, the fission product activities have been calculated using ORIGEN-S code from the SCALE-4.4a system, taking into account different irradiation and decay times. At the end of the irradiation, the LEU foil target will be allowed to decay in the reactor pool. Prior to reactor start up, the irradiated target will be removed from the reactor pool and transported to the hot cell for disassembly.

Root Cause Analysis of Swelling Problem in *Kartini* Reactor

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Kartini reactor is an open pool type reactor of TRIGA Mark II family, with aluminium pool liner of 6 mm thickness. The reactor has been in operation since 1979. In 2001 the pool was emptied of fuel and control system to enable a complete inspection of the pool liner after more than 20 years of service. Several NDT methods had been used in the inspection i.e.: a comprehensive visual examination, hardness survey, dye penetrant examination, ultrasonic thickness survey, and replication of features of interest. In general the inspection revealed that the pool liner was in good condition and the results of the thickness and hardness survey were consistent with the service history. Three areas of interest were observed; a small area with apparent thinning, a small crack that was analysed as an original manufacturing defect, and there were two areas of swelling (bulges) observed under the thermal column and these were assigned as S1 and S2 for identification purposes.

New inspection equipment was acquired by BATAN through the IAEA TC Project No. INS 9022, and the features re-examined in 2003, 2004, 2005 and 2006. The bulges had increased in size over this period. Visual inspections by the video equipment and replication indicate that the swelling parts observed in 2001 had grown in size. In 2004 the dimension of S1 and S2 were observed to have 7.72 mm and 7 mm of height, and 1365 mm^2 and 1083 mm^2 of area, respectively. While in 2005, the heights of S1 and S2 were seen to increase to 7.78 mm and 7.56 mm, and the areas to be 1389 mm^2 and 1839 mm^2 respectively. Recent examination (Sep. 2006) showed that the size of bulges relatively constant, and the peak of the bulges appears to contain tears (cracks).

Therefore, it is wise to undertake a review of the above features, starting with a root cause analysis related to the above swelling problem. Root cause analysis (RCA) is a method to identify the root cause of an event or adverse trends associated with corrective actions to a set of events. The RCA result shows that probable root cause of swelling are as follows:

- It is probable that the seal on the cover plate in the service pool has deteriorated and allowed water to enter both the thermal column and the space between the aluminium reactor pool liner and the concrete. The water will also saturate the concrete and has the potential to corrode the steel reinforcement close to the surface of the concrete. It is believed that water leakage from the service pool has entered the area behind the aluminium pool liner and has saturated the concrete.
- It is believed that carbon steel reinforcement close to the inner surface of the reactor block has corroded. The expanding corrosion product (rust) has the forced layer of concrete covering the steel reinforcement and subsequently pushing the aluminium pool liner inwards, causing the swelling.
- There are two issues: the mechanism for the creation of the swellings or bulges dominated by Iron corrosion, and the potential for corrosion of aluminium dominated by the pH of water in contact with the aluminium. As the evidence for this condition

were: the apparent corrosion of steel reinforcement behind aluminium liner and formation of Ca(CO3) on outside of concrete reactor block indicating water saturation of the concrete block. (see Fig. 1.)

Fig. 1. CaCO3 forming on the outer surface of the reactor block near the service pool

The condition is advancing and will most probably result in a loss of pool liner integrity, this may or may not result in a loss of pool water due to the concrete backing behind the pool liner. The situation is not urgent at this time because a loss of cooling accident (LOCA) is not credible from the defects observed. A slow leakage of pool water may result. The issue is predominantly one of maintenance not safety. It is very probable that water is leaking from the sealing plate in the bulk shielding facility (service pool) and has saturated the area behind the liner and probably the thermal column. This provides an environment highly susceptible to corrosion for these components and it can be reasonably expected that accelerated pool liner corrosion will occur.

Based on the above analysis, the remedial actions that can be considered are as follows:

- The pool liner would be patched with a welded panel that would allow the features we have observed to keep growing in a harmless way. It is expected that they will reach a finite size as the iron is consumed by corrosion.
- The area behind the pool liner would be dried out, this should limit further growth of the bulges by improving the corrosion conditions behind the pool liner.
- The bulk storage facility would be lined with stainless steel (the potential to use the thermal column could be preserved for future use). This would remove what is thought to be the root cause of the bulges and keep the reactor block dry.

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IAEA-CN-156/U-23

Technique of testing the VVER-1000 high burnup fuel rods in the MIR reactor at the design basis RIA parameters

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(FSUE"SSC RIAR")

International Conference on Research Reactors:Safe Management and Effective Utilization, Australia,Sydney, 5-9 November 2007

To study the VVER-1000 fuel behavior under design basis RIA accident conditions, a technique was developed and implemented in the MIR reactor for conducting reactor dynamic experiments with simulation of parameters responsible for thermomechanical state of fuel rods.

A power pulse of the experimental assembly is formed in the reactor channel operating at constant power when the absorber shield (that surrounds fuel rods in the initial state) is removed from the assembly. Hafnium is used as the absorber allowing long-term operation at temperature of 400 ... 500 ^{0}C. According to the nuclear safety conditions pulse formation must be accompanied by slight and negative reactivity addition. A reactivity compensator that moves synchronously with the absorber shield was introduced to reduce the effect on the core. The absorber shielding and reactivity compensator are designed as a combined shielding device. In order to move the shielding device at a high speed (up to 200 mm per sec), a fast acting hydraulic drive was developed. It uses the potential pressure energy of the loop primary circuit. The driving force for the shielding device movement appears when the space above the hydraulic cylinder piston contacts with the atmosphere. To limit and regulate the pressure drop across the hydraulic cylinder piston, an original device is applied, in which the throttling effect is achieved when two preliminary separated medium flows meet each other.

A three-element fragment of the VVER-1000 fuel assembly including two high burnup fuel rods is used as an experimental fuel assembly. Total length of experimental fuel rods is 230 mm and the fuel column length is 200 mm. The specified fuel rod parameters –fuel column temperature and fuel enthalpy – are determined from the pulse amplitude and time of radiation exposure at the maximum power. A precise experiment control system was developed to set this time.

The method for defining parameters, which determine the thermomechanical fuel state, is based on processing the experimental curve of temperature change in the fuel center and readings of other measuring sensors installed in the experimental fuel assembly and primary circuit of the loop facility under stationary and transient conditions. The rate of sensor signals measurement under dynamic conditions is equal to 100 Hz.

The diagram below demonstrates the calculation results for distribution of the fuel rod linear power and fuel center temperature as a function of time.

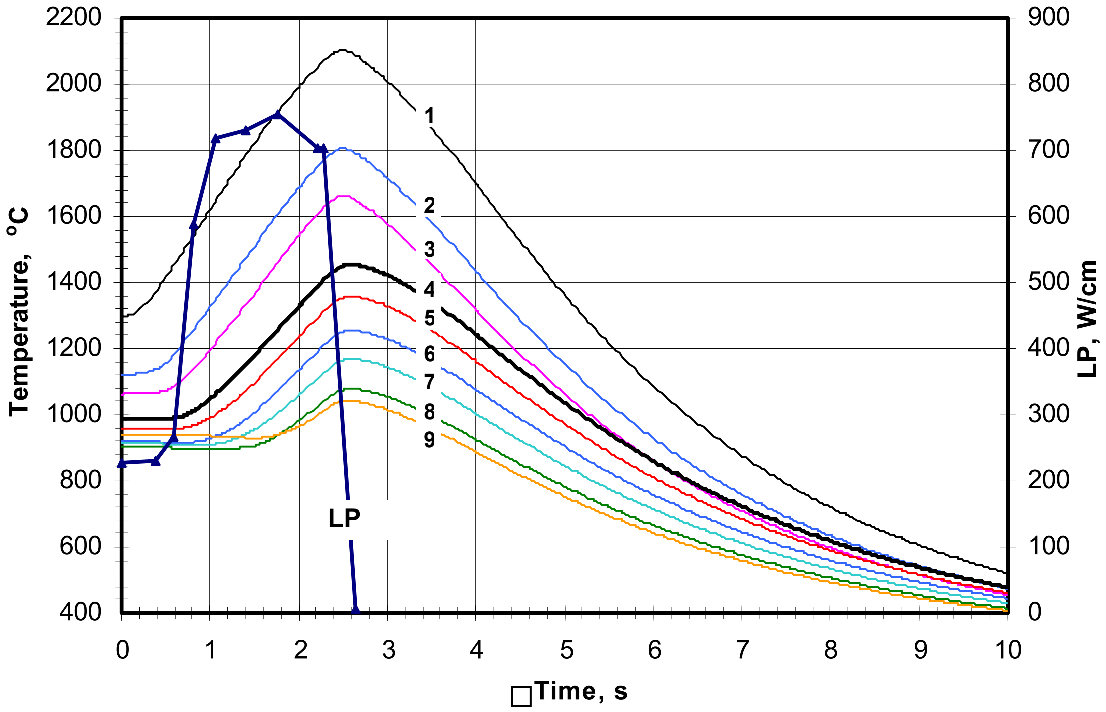

Diagram – Temperature of the fuel column center at different height (the fuel column is marked starting from its bottom and then in each 2.5 cm) and change of the linear power (LP) in section 4

Separation of the experimental fuel rods into short parts for calculations enables to obtain fragments of fuel rods with different pulse parameters within one experiment (amplitude ranges from 2.63 to 4.25 and half-width – from 0,8 s to 2.2 s). As a result it considerably increases the value of a single experiment. Parts 50 mm long are considered most appropriate.

Realization of the IBR-2 Research Reactor Modernization Program

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The pulsed fast reactor IBR-2 is a pulsed reactor of periodic action (pulsed reactor) and its original difference from other reactors consists in mechanical reactivity modulation with a movable reflector (MR). The movable reflector is a complex mechanical system with a total mass up to 60 t providing for reliable operation of the two parts, which determine reactivity modulation: the main movable reflector (MMR) and the additional movable reflector (AMR). The MMR and AMR rotors rotate in the same direction with different velocities. When both reflectors coincide near the reactor zone, a power pulse is generated. The factors determining the duration of a fast neutron pulse are fast neutron lifetime, configuration and rotation velocity of the rotors [1].
The IBR-2 reactor was put into operation in February 1984. At present an average power of the reactor is 1,5 MW and pulse repetition rate is 5 Hz. Due to its pulse power, equal to 1500 MW, IBR-2 possesses the highest in the world pulsed thermal neutron flux for beam investigations, which is 10^{16} n/cm²·s. Pulse duration is 245 μs for fast neutrons and 350 μs for thermal neutrons (behind the 4 cm thick water moderator) [2].
The IBR-2 reactor is used principally for beam studies in solid-state physics (solids and liquids), biology, and material science. The experience of the IBR-2 reactor operation proved it to be a rather effective neutron source, which for many applications is as good as the best sources, based on proton accelerators. Moreover, the development of neutron experiment technique and application of modern developments at the IBR-2 reactor have shown that neutron flux magnitude is of fundamental importance for high efficiency of a pulsed source.
At the same time pulse duration can be different in different experiments. This methodical conclusion can be essential for further development of neutron sources throughout the world. Operating experience of IBR-2 is especially important at present when the interest is aroused in pulsed neutron sources with large pulse duration [3].
The IBR-2 reactor possesses a record neutron flux and yet is a very economic and rather cheap machine. Its construction cost is ~20 M$ including the cost of buildings, and operating expenses are 1 M$/year. Due to low average power, activation of the equipment and burning up of the zone go on slowly. Under the established operating regime of 2500 hours per year for physical experiments, the service life is about 20 years for the zone and 7 years for the movable reflector. Accounting for the time under this operating regime, one can see that the service life of the main units of the reactor should end in 2002.
With this account it was considered appropriate to replace units of the reactor where necessary after 2002. The funding difficulties, however, made us to revise the established operating regime to slow down the wearing out of IBR-2 remaining service life, so that the reactor will have its rated resource exhausted by 2007. In this connection the reactor modernization program has elaborated for the period up to 2010. The present concept of the IBR-2 reactor modernization involves carrying out work including development, manufacturing and installation of the reactor equipment. At the same time, accounting for the experience of reactor operation and physical research, the given concept contains a number of

novel technical solutions that substantially improve operation and physical reactor characteristics, which permits one to assert that actually in the process of modernization a new IBR-2M reactor is being created [4].

In the report the following basic themes are stated: general description, main characteristics and current state of the reactor. Also the realization of the IBR-2 reactor modernization program and its main directions, terms, results, financing, postnatal modernization IBR-2 parameters are given.

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Session 9: Decommissioning and Waste Management

Synopses no. IAEA-CN-156/	Synopses Title	Main Author
DE-3	Overview of Research Reactor Decommissioning	Rowling, J.
DE-2	The transition from a research reactor in operation to a facility undergoing decommissioning	Nielsen, K.H.
WM-2	Carbon steel construction removal from spent fuel storage pool	Pešic, M.

Overview of Research Reactor Decommissioning

SYNOPSIS

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The purpose of this paper is to present a wide view of Research Reactor decommissioning. The focus will include historical terms, showing progress, trends, and state-of the-art strategies for remaining and emerging issues. Some of the issues relate directly to the decommissioning program planned by the Australian Nuclear Science and Technology Organisation (ANSTO). ANSTO has operated HIFAR, the 10MW research reactor since 1958. HIFAR was shutdown at the beginning of this year after 49 years of successful operation. The reactor provided neutron beams for research purposes, radioisotopes for medical and industrial use and irradiated silicon for the semiconductor industry. In addition, although ANSTO has successfully mothballed MOATA, a small Argonaut type reactor, there are still substantial planning requirements to progress MOATA's final decommissioning program.

ANSTO faces a number of challenges heading into the decommissioning of HIFAR. These include: the establishment of a modern decommissioning strategy in the absence of a long-term waste management facility or waste acceptance criteria for the material generated by the decommissioning. The impact of closure of the facility on staff morale and retention of key staff, plus the regulatory requirements as well as the Organisation's needs are now the major issues for the Decommissioning Team.

These challenges are compounded by competition for skilled resources now required by the full power operation of the newly commissioned research reactor (OPAL) at the same site and to meet the needs of the researchers, isotope production and the silicon customers.

The technical problems are now not insurmountable for decommissioning; however the areas of policy, planning, timing, costs, waste disposal, safety criteria and regulatory aspects need further development.

The industry now must demonstrate its maturity and keep up with safety, environmental and regulatory requirements, with pressure of change in political perceptions and expectations. There are 832 research reactors, of which there are 287 operating, 524 either shutdown or decommissioned plus 205 that are greater than 40 years old. There are 90 reactors planned or progressing to unrestricted use or safe enclosure (11%). There are 233 with a current status of decommissioning completed as "in unrestricted use" or "safe enclosure" (28%). This is based on data from the IAEA at May 2006 from Technical Report Series No. 446.

The Community have an expectation of the nuclear industry to clean up their backyard. This is much easier for the nuclear power industry where the income from electrical tariffs make provision for decommissioning, such as in France. The nuclear research reactor industry has supported the power industry by its research but has been left out of the funding equation in most cases.

In Australia there is no nuclear power industry as coal-fired power stations dominate electrical power production.

Fortunately the federal government will fund the decommissioning of MOATA and HIFAR. Two years ago we commenced the planning for the decommissioning of HIFAR, this included the application for funding.
HIFAR was performing better over its last five years than it did over its previous 40 odd years. However the plant and equipment was old. The processes were manual and time consuming - hence the reason to construct OPAL. It was determined that ANSTO had four choices for HIFAR after shutdown: 1) do nothing, 2) immediate dismantling 3) deferred dismantling, and 4) entombment. 1 & 4 were perfectly reasonable alternatives. The site now had OPAL for perhaps another 40 years. The preferred option was 2) immediate dismantling, taking advantage of the expertise of the existing highly skilled staff but this required a national waste repository. The only good option then left was 3) deferred dismantling. If immediate action was taken to dismantle the reactor block there would be significant cost implications in the storage and double handling of activated waste. The fuel and heavy water have been removed. This work was performed under the existing operating licence.
With this strategy in place we have prepared an application to the regulator for a "Control or Possess" licence for a period up to ten years. It is planned that during this period we can remove all non-active components and shrink the footprint of the HIFAR facility as a whole to just the reactor building shell and the auxiliary plant room.
Economic sustainability is not an issue as ANSTO is located in the outer metropolitan area of Sydney. ANSTO has operated HIFAR for years and there is a well developed fuel and waste management policy.
The next major issue for ANSTO is the active waste release/clearance criteria taking into account the newly proposed waste repository. There is no agreed location for the repository and no waste acceptance levels set yet.
Planning for decommissioning did start some 2 years ago using the above deferred dismantling strategy and similarly we have followed the IAEA guidelines for a structured approach consisting of generic tasks namely:

a) Preparatory work and planning for shutdown;
b) Final shutdown;
c) Removal of radioactive sources (including liquids);
d) Radiological characterisation (delayed until prior to application for decommissioning licence);
e) Decontamination and inactive dismantling;
f) Refurbishing of essential systems (HVAC and Electrical Supply)
g) Demolition of structures and buildings (Under P or C licence);
h) Surveillance and maintenance (Through whole project);
i) Final dismantling of reactor block and building
j) Waste Management
k) Site clearance and release (Planned for 2017-18)

During this total process substantial time will be given to the collection and recording of all data. The operating records have been collected and analysed to capture past events that may have significance in the decommissioning work.
"Lessons learnt" for me have been such items as staffing the decommissioning team, maintenance of a good safety culture during final stages of operation, working towards regulatory approval for decommissioning and strategies for knowledge retention.

The transition from a research reactor in operation to a facility undergoing decommissioning

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Danish Decommissioning (DD) was established as an institution under The Ministry of Science, Technology and Innovation based upon a resolution passed by Parliament in March 2003. The task is to decommission all the nuclear facilities at Risø National Laboratory (Risø) to the state of "Green Field" where all buildings, equipment and materials that cannot be decontaminated below established clearance levels are removed, and to maintain the nuclear facilities until these have been fully decommissioned.

The six nuclear facilities to be decommissioned are: Three research reactors - DR 1, DR 2 and DR 3, Hot Cells, Fuel Fabrication and a Waste Management Plant. Figure 1 below shows the location of the nuclear facilities at Risø:

FIG. 1. Location of the nuclear facilities at Risø.

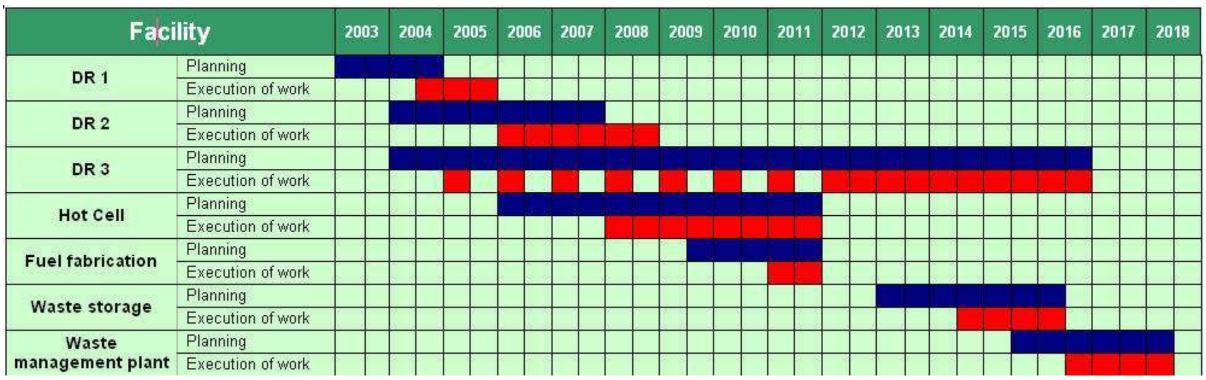

FIG. 2. Schedule.

The time frame for decommissioning the nuclear facilities at Risø is 11-20 years (DD expects to have completed the task by 2018). The time schedule is shown above in Figure 2. DR 1 has been decommissioned in 2004-2005 and the hall and surrounding areas were released for unrestricted use in January 2006. The decommissioning of DR 2 is ongoing and clearance measurements will be done in 2008.

Reactor DR 3 – which this paper will be focusing on – was a 10 MW heavy water cooled and moderated research reactor of a design similar to the British "PLUTO" type. Originally, DR 3 was built as a Materials Testing Reactor, but ended as a Multipurpose Research Reactor. With a cold neutron source, six three-axis spectrometers and a small-angle neutron scatter instrument DR 3 was appointed a Large European Beam Facility and these neutron beam instruments were used intensively by researchers from Risø and from the other EEC-countries. The main production activities were Neutron Transmutation Doping of Silicon (NTD), isotope production and activation analysis. By the end of its operating period, DR 3 supplied 1/3 of the world market of NTD.

DR 3 was in operation from 1960 until 2000 in a 4-week-cycle with 23 days of continuous operation and 5 days of shut down. After the final shut down the fuel elements have been removed and shipped to the US. The heavy water has been stored in drums and later exported to Canada. Characterization of the reactor block has been done in 2005-2006 and characterization of DR 3 storage is carried out in 2006-2007.

Most of the auxiliary systems outside the reactor building have been removed by now (e.g. the secondary and tertiary cooling systems) or modified (e.g. the power supply). Inside the containment the removal of auxiliary systems are ongoing as part time tasks along with decommissioning of other nuclear facilities at the site and are to be finished by the end of 2011. Planning the decommissioning of the reactor block has started and the work is to be carried out in 2012-2016.

Originally, the DR 3 reactor was supposed to be running until 2006, but in 2000 it was decided to close all the nuclear facilities at Risø. Therefore no planning of the decommissioning had been done in advance and the organization was not prepared for the new purpose. During operation a large part of the work was well known – operating and maintaining the reactor – according to the defined running and shut down periods. After the final shut down of DR 3 the type of work gradually changed from routine tasks into more project-based. The organization has undergone major changes in order to adapt to the new conditions.

This paper describes the transition from operating a research reactor to be decommissioning a nuclear facility and deals with the change of the organization from being a department of Risø to be an independent institution. The “treading water” period form right after the decision to finally shut down DR 3 was made until it was decided how to get on with the decommissioning will be discussed. The relations to the authorities, who were as little prepared for the new situation as we were, will be described along with the issue of preparing the staff for the change – retraining and replacing a large part. The current status of DD – how did we manage the decommissioning of the two small reactors – and how are we preparing the decommissioning of the largest reactor, DR 3? How do we maintain nuclear knowledge in house when key personnel retire and how do we keep the staff until 2018? These are some of the issues, which will be covered in this paper.

Carbon Steel Construction Removal from Spent Fuel Storage Pool

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Carbon steel structure in basin no. 4 of the spent nuclear fuel pool the RA research reactor at Vinca Institute of nuclear sciences, Belgrade, Serbia was constructed in 1959 for spent fuel assembly management. It consists of a robust metal construction with so called 'working table' and two sets of two connected coaxial tubes. The construction is shown in Figure 1.

Fig. 1. Carbon steel construction with tubes

The construction was the main source of corrosion process products in the pool filled with stagnant tap water. Inappropriate water chemical parameters [1] and carbon steel corrosion particles have also initiated corrosion of aluminium cladding of spent fuel elements stored in aluminium storage barrels. Fission products (^{137}Cs nuclide) leakage to the pool water was detected. Also this construction was the main source of sludge creation that was deposited to basins bottom walls, the construction itself and spent fuel storage containers.

For that reason, a complex technological project was initiated, in 2004, with the IAEA TCP assistance and Russian R&D company OEC NIKIMT to design and manufacture a proper equipment and develop safe technological process to remove (under water) the carbon steel construction from the pool.

After long and careful preparation processes, facility modification and upgrading many elements of radiation protection systems, including training RA reactor operational staff in procedures prepared and at mock up and after thorough review of the Safety analysis report of the whole operation, the Serbian regulatory authority issued approval for the operation in November 2007.

The removal of the structure from the spent nuclear fuel pool was carried out from mid-November 2006 to March 2007. The main structure was safely cut in few big pieces (due to unforeseen high gamma dose rate of the deposits at the construction further cutting to smaller pieces was abandoned). Tubes were safely cut in two pieces for the same reason. All removed elements were packed in new designed metal storage containers and temporary and stored at the Vinca RAW storage facility area, until commission of the new designed Waste Processing Facility (foreseen at the end of 2008).

Figure 2 shows operators' actions on removal of the biggest part of metal construction, 'working table' and two inner tubes, all heavy corroded and their storage at metal containers.

Fig. 2 Removal of the biggest part of metal construction, 'working table' and two inner tubes

Acknowledgement

Authors and the all members of the Vinča VIND SNF/RadProt/RAW team acknowledge to the IAEA TCP Department (Contract SCG4003-890905), IAEA experts from NFCWT /NFCM and DNIS/RRS Sections, Ministry of Science and Environmental Protection of the Republic Serbia, and the experts of the OEC NIKIMT, Obninsk, from the Russian Federation for their support and engagements during the whole operation.

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Session 10: Core Safety and Utilization Parameters

Synopses no. IAEA-CN-156/	Synopses Title	Main Author
S-36	Status Report on Preparation of IAEA Guidelines for Qualification of Research Reactor Fuels	Snelgrove, J.L.
S-13	Simulation of Flow Behavior in the HANARO Reactor Pool by Using the MARS Code	Park, C.
S-27	Experimental Measurements for Plate Temperatures of MTR Fuel Elements at Sudden Loss-of Flow Accident and Comparision with Computed Results	Sevdik, B.
S-60	Contract Performance Demonstration Tests in the OPAL	Hergenreder, D.F.

Status Report on Preparation of IAEA Guidelines for Qualification of Research Reactor Fuels

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Because development of high-density research reactor nuclear fuels (RRNF) is being pursued in many countries and because no comprehensive document addressing the rationale of qualification of RRNF has been published, the IAEA is carrying out a project[1] to produce a set of guidelines for RRNF qualification. These guidelines are intended to provide points of reference for the type, quality, and completeness of the information to be generated in order to ensure the acceptable performance of high density LEU fuels to be used and / or retrofitted in existing and new research reactors. It is anticipated that this guidelines document will be in the final editing stages by the time of the Sydney Conference. This paper provides an overview of the content and conclusions of this document.

The Guidelines Document addresses basic definitions; approaches and processes relevant to the qualification of research reactor nuclear fuel and defines essential information required for the licensing and use of fuels in research reactors. The most basic definition is that fuel qualification is a process carried out both by a fuel developer and a fuel manufacturer. The fuel developer must demonstrate the acceptable irradiation performance under conditions, including fission rate and ^{235}U burnup, that exceed any conditions that might exist during normal operation of a reactor using the fuel and in geometric configurations used in the reactors that are candidates to use the new fuel. The fuel manufacturer must demonstrate that it can reliably and consistently manufacture fuel assemblies acceptable to the customer.

The information that should be provided by the fuel developer for fuel qualification falls naturally into five groups: (1) basic fuel properties, (2) as-fabricated fuel meat properties, (3) fuel meat irradiation properties, (4) fuel element (plate, tube, or rod) properties, and (5) fuel assembly properties. This information provides the basis for fuel element and fuel assembly design and the basis for licensing the use of such fuel assemblies in research reactor cores. It should be noted that the fuel developer should endeavor to obtain as broad a range of data as possible at each step of development, within constraints of schedule and funding, in order to maximize the applications for which the fuel can ultimately be licensed. The ultimate proof that a fuel is qualified from an irradiation-performance perspective is the approval by a national regulatory authority for use of the fuel in a research reactor.

Manufacturing process qualification occurs after the fuel has been developed and is limited to activities performed to demonstrate that the process and product meet specified requirements. However, in most cases, the manufacturer has participated in the fuel development process. At the very least, it is generally expected that the manufacturer will have produced fuel

[1] Current Guidelines Document Committee Members: P. Adelfang (IAEA); H. Taboata (Argentina, CNEA); P. Lemoine (France, CEA); D. S. Sears (Canada, AECL); C. Jarousse (France, CERCA); A. Enin (Russia, NCCP); N. Arkhangelsky (Russia, Rosatom); C.-K. Kim (South Korea, KAERI); J. L. Snelgrove, T. Totev (USA-ANL); M. Nilles (USA-BWXT); D. Wachs, D. E. Keiser (USA-INL)

elements and fuel assemblies for final irradiation-behavior qualification tests. Nevertheless, ultimately, manufacturer qualification is determined by the customer, guided by requirements of the particular country and its national regulatory authority. In contrast to the usual one-time irradiation-behavior qualification for a particular reactor or reactor type, manufacturer qualification may be required more than once if the manufacturer produces fuel for a particular reactor only sporadically.

In summary, qualification of RRNF is a complex process involving the fuel developer, the fuel manufacturer, the fuel customer (usually the reactor operating organization), and the national regulatory authority. This new IAEA Guidelines Document is expected to clarify the process for all of these parties and, thus, lead to a more-efficient qualification process in the future.

Simulation of Flow Behavior in the HANARO Reactor Pool by Using the MARS Code

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Synopses

Generally speaking, no robust computer codes such as RELAP5, RETRAN, and MARS are available for the analysis of a research reactor, and it is highly recommended that these computer codes developed for a transient analysis of power reactors be applied carefully after assessing their applicability to a specific research reactor. The MARS code [1] is a realistic system transient analysis code that can be used for the simulation of a wide variety of PWR system transients. This code is a unified version of a 1-D reactor system analysis code, RELAP5/MOD3 and a 3-D reactor vessel analysis code, COBRA-TF coupled with a 3-D reactor kinetics code, MASTER and a containment code, CONTEMPT4. Some assessment calculations [2] have shown that the MARS code can also be applied to the thermal-hydraulic analysis of research reactors with careful evaluations.

The HANARO is an open-tank-in-pool type research reactor of 30 MWth [3], where a 3-D analysis of its flow behaviour is often necessary. One of major areas concerned for the 3-D flow behaviours in HANARO is reactor pool where the bypass flow may rises up to the pool surface.

In this paper the simulation results of the flow behaviour in the HANARO reactor pool by using the MARS code are described.

The measurement and prediction by the CFD code [4] for the flow behaviour in the HANARO reactor pool shows that a counter clockwise circulating flow exists in the entire reactor pool if one looks into the pool from the top, as shown in Fig. 1. It may be caused by an unsymmetrical pouring of the bypass flow into the bottom of the reactor pool. A large swirl flow rises up slowly toward the pool surface, but it soon turns downward at near the bottom of the hot water layer and then it flows downward to the core through the chimney.

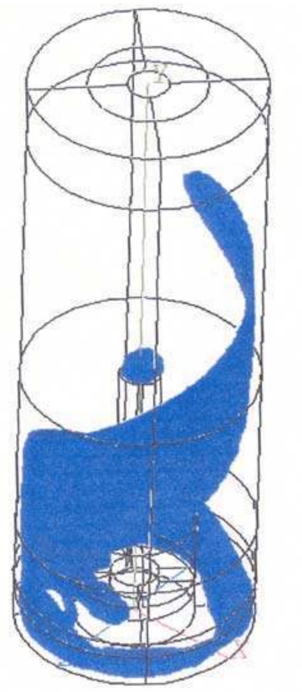

FIG. 1 Flow Behavior in the HANARO Reactor Pool

The dimensional flow behaviours in the HANARO reactor pool mentioned above have been simulated by using the 3-D model of the MARS code in order to inquire about the applicability of the MARS code to such a case. The nodalization of the HANARO reactor pool for the MARS code simulation is shown in Fig. 2. The reactor pool was modelled by using the MARS code multi-dimensional model. The numbers of Z, R, and Θ cells are 19, 8 and 8, respectively. So, the total number of volumes for the reactor pool are 1216. The reactor assembly part (green colour) is modelled as a 1-D part. The reactor pool surface is connected to a time dependant volume as boundary condition of the atmosphere. Other components such as the piping and pumps are modelled with proper components in the code.

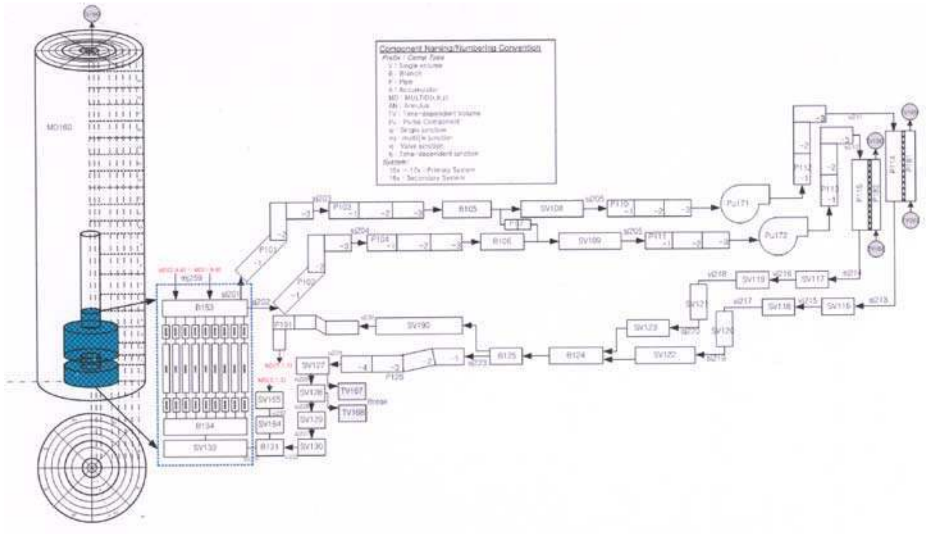

FIG. 2 Nodalization of the Reactor Pool and System of the HANARO

The results showed reasonable predictions for the 3D flow behaviour in the HANARO pool. This capability may be useful to predict the effect of 3D flow phenomena on a core during a flow reversal transient in research reactors.

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Experimental Measurements for Plate Temperatures of MTR Fuel Elements at Sudden Loss-of Flow Accident and Comparison with Computed Results

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The aim of this study is to generate experimental data to be used for sensitivity analysis and assessment calculations on the thermal-hydraulic codes written for Loss of Flow Accident (LOFA) analysis of research reactor with MTR- type (plate type) fuel elements.

In an open pool research reactor with downward coolant flow, an accident such as shaft breaks between pump and flywheel can lead to sudden loss of flow. Flow reversal occurs in many MTR-type research reactors during transient after shut-down (flow scram). It can be mismatched with the decay heat removal by natural convection when the coolant flow rate becomes so low. A critical heat flux may occur even at a low heat flux. It is essential to demonstrate that such a sudden loss of flow does not lead to excessive temperature increase at the fuel plate and consecutively not cause fuel melting.

In the TR-2 Research Reactor, Sudden Loss of Flow Accident which happens by shaft break between pump and flywheel is simulated by closing the core outlet valve. There is a butterfly type valve at the core outlet and total closing time is about 30 seconds but effective flow decrease occurs in 10 seconds. Beginning of the flow decrease by closing of the valve was accepted as time zero and the decrease of flow rate versus time was plotted. An instrumented fuel element, which has five thermocouples along the vertical length of the fuel plate, was used for the experiments. This fuel element was placed in a certain position of the TR-2 core and the plate temperatures were measured and recorded by a digital recorder. At the same time primary flow rate and core pressure drop (ΔP) at the grid plate were measured and recorded. Power distributions of the fuel elements in the reactor core were determined experimentally by using copper wire activation technique. According to the existing core loading of the TR-2 Reactor, core positions, U-235 weights, relative neutron flux distributions and power distributions of the fuel elements were given as the inputs of the calculations to simulate the loss of flow accident at the computer codes.

Four experiments were performed for sudden loss of flow accident. Three of these experiments were repeated with different initial conditions such as different core inlet temperatures and different reactor operation times at 5 MW nominal power level and 750 m^3/h primary flow rate. The reactor was shutdown by flow scam at these three experiments.

<u>Experiment I:</u> The TR-2 Reactor had been operated at 5 MW for 5 minutes. The primary flow rate was $750 m^3/h$ and core inlet (pool) temperature was 23^oC. Primary coolant flow rate began to decrease from $750 m^3/h$ (time zero) by closing of the core outlet valve. The primary flow scram happened after 7 seconds and natural convection flappers opened after 25 seconds. Plate temperatures and the decrease in primary flow rate were recorded for 1 second time intervals.

Experiment II: The TR-2 Reactor had been operated at 5 MW power for 5 minutes. The primary flow rate was $750m^3/h$ and core inlet (pool) temperature was 17^oC. Primary coolant flow rate began to decrease from $750m^3/h$ (time zero) by closing of the core outlet valve. The primary flow scram happened after 9 seconds and natural convection flappers opened after 35 seconds. Plate temperatures and the decrease in primary flow rate were recorded for 1 second time intervals.

Experiment III: The TR-2 Reactor had been operated at 5 MW power for 2 hours. The primary flow rate was $750m^3/h$ and core inlet (pool) temperature was 29.5^oC. Primary coolant flow rate began to decrease from $750m^3/h$ (time zero) by closing of the core outlet valve. The primary flow scram happened after 9 seconds and natural convection flappers opened after 28 seconds. Plate temperatures and the decrease in primary flow rate were recorded for 1 second time intervals.

The sudden loss of flow accidents performed in these three experiments were simulated and analyzed by using PARET computer code. The measured plate temperatures were less than the calculated plate temperatures. The differences may come from the decay heat calculation or modeling of the natural convection for narrow, rectangular flow channels at PARET code. To investigate the source of these differences a fourth experiment was performed at 50kW power level, which allows the reactor to operate with natural convection without primary flow scram.

Experiment IV: The TR-2 Reactor had been operated at 50kW power for 5 minutes. The primary flow rate was $750m^3/h$ and core inlet (pool) temperature was 19^oC. Primary coolant flow rate began to decrease from $750m^3/h$ (time zero) by closing of the core outlet valve. The primary flow scram not occurred (50kW neutronic power) and natural convection flappers opened after 35 seconds. Plate temperatures and the decrease in primary flow rate were recorded for 1 second time intervals.

The results of the experimental measurements were given in graphics (plate temperatures and flow rates versus time). Computational results obtained from PARET code were compared with the experimental values.

In conclusion, it was determined that the residual decay heat calculation and the natural convection modeling for narrow rectangular channels of PARET code are not quite realistic. PARET uses the decay heat generation rate within the reactor core based on the standard fission-product decay heat curve for uranium-fuelled thermal reactors published by the American Nuclear Society as a proposed standard (ANS-5.1/N18.6). ANS decay heat curve is too pessimistic. For the calculation of the residual decay heat, special attention must be given to the fact that “*Approximately one-half of the decay heat is due to gamma radiation with energies in the range of about 0.2-2.0 MeV. The e-folding length for 1 MeV gamma radiation in aluminum is 6 cm, i.e., the source gamma intensity is attenuated to 1/e of its original value after passing through 6 cm of aluminum. This corresponds to a penetration of 40-50 fuel plates and hence shows that a significant portion of a given level of fuel element decay heat will be deposited at the outside of the fuel element*”.

Realistic calculations can be done for LOFA and LOCA analysis by using the half of the values of decay heat generation curve of ANS.

Contract Performance Demonstration Tests in the OPAL

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Extended Synopsis

This paper will describe the measurements and calculations that were done in the OPAL Reactor to demonstrate compliance against contractual Design Features and Performance Acceptance Criteria.

The OPAL Research reactor is a multi-purpose open-pool type reactor. The nominal fission power of the reactor is 20 MW. The core is located inside a chimney, surrounded by heavy water contained in the Reflector Vessel. The whole assembly is at the bottom of the Reactor Pool, which is full of de-mineralized light water acting as coolant and moderator and biological shielding.

The reactor has two operative neutron sources: a cold neutron source and a thermal neutron source, each of them feeding two tangential beams containing several supermirror neutron guide lines. The reactor also provides support for a third (hot) neutron source feeding one neutron beam tube.

The reflector vessel has also 17 vertical irradiation tubes, with 5 targets each, for bulk radioisotope production (Ir, Mo and I) and 19 pneumatic rigs with 57 multipurpose target positions: radioisotope production, neutron activation analysis (NAA), etc. Additionally, it also has 6 neutron transmutation Si doping (NTD) facilities.

The contract specifies several neutronic aspects to be fulfilled by the core, the irradiation and the beam facilities design, which have to be verified during the commissioning tests. The following list gives a summary of such warranted values:

Neutron Flux Level.

- Several irradiation facilities require a minimum value for the neutron flux; but some of them require more strict flux control demanding a minimum and a maximum neutron flux value.
- Minimum flux levels are also specified for the cold and thermal beam facilities at the outlet of the different NG lines.

Neutron Spectra.

There are different requirements on the flux spectra associated with:

- Cold Neutron Beam Facility: Max. allowable peak in cold spectra)
- Thermal Neutron Beam Facility: Max. allowable peak in thermal spectra)
- Radioisotope Production Facility: neutrons with energy lower than 0.6 eV
- Irradiation Facilities: Fast neutron flux (E>1.0 MeV)

Flux Homogeneity. There are requirements on the flux homogeneity on the different irradiation facilities, i.e. the axial homogeneity on NTD facilities, inside an irradiation can, and within several targets of the same flux level.

Flux Perturbation. There are strict requirements on the flux perturbation at the irradiation positions due to the movement of the irradiation samples.
There is also a requirement on the irradiation facilities flux perturbation during normal operation due to control rod movement..

Summarizing, all the facilities were designed taken into account a set of performance criteria related with the flux level, spectrum and perturbations.

Guaranteed flux values will be taken as being for equilibrium core conditions. The relationship between values measured during commissioning (First Core) and the guaranteed values is largely based on calculations.

The First core with 3 types of FA was specially designed to minimize the difference on the flux levels and distribution during the transition between the first core and the equilibrium core.

Measurements performed when equilibrium conditions have been more closely approximated will be performed where possible and considered necessary.

A time-of-flight instrument with a flight path greater than 1m was constructed to measure the Energy Spectrum. To account for possible variation of the energy spectra horizontally across the guide, measurements were performed at multiple spatial positions.

In the neutron guides, the thermal neutron flux value was measured using pure gold foils located at multiple spatial positions.
In the irradiation facilities, the thermal neutron flux was measured using diluted gold wires. Axial and radial flux profiles were measured in the Bulks Irradiation Facilities and Neutron Transmutation Doping (NTD) Facilities.
The Cadmium Ratio Method was used to obtain the absolute thermal flux values.

The fast flux was measured in the NTD Facilities and Fast Flux Facilities using Nickel wires.

The measured values are compared with the acceptance values and also with the calculated values. The calculated values are obtained modelling with full detail the measurement conditions using the INVAP traditional calculation lines: CITVAP and MCNP calculation lines.

CITVAP code is used to follow the core burnup taking into account the operation conditions, (temperature and Control Plate positions). The CITVAP calculation line generates the Fuel Assembly numerical densities that will be required by the MCNP calculation line to model the measurement conditions.

The MCNP is used to obtain the calculated flux values in the irradiation facilities. The calculation model includes a full description of the irradiation conditions and core burnup.

Session 11: Programmes for the Minimization of the Use of HEU

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F-11	Conversion of Research and Test Reactors to Low Enriched Uranium Fuel: Technical Overview and Programme Status	Roglans-Ribas, J.
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STATUS OF THE UNITED STATES DEPARTMENT OF ENERGY'S NUCLEAR FUEL RETURN PROGRAMS

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SYNOPSIS

The U.S. Department of Energy/National Nuclear Security Administration's Global Threat Reduction Initiative (GTRI) is a cooperative program that is intended to provide international support for countries' national programs to identify, secure, recover and/or facilitate the disposition of vulnerable, high-risk nuclear and radioactive materials that pose a threat to the international community. GTRI has three complementary fuel return programs that strive to recover and/or facilitate the disposition of such materials. These three programs focus on the return of U.S.-origin fresh and spent fuel to the United States (The Foreign Research Reactor Spent Nuclear Fuel [FRR SNF] Acceptance Program), return of Russian-origin fresh and spent fuel to Russia (The Russian Research Reactor Fuel Return [RRRFR] Program), and disposition of other vulnerable, high-risk nuclear materials in the United States and elsewhere (The Gap Material Program). This paper will discuss the current status of each of these removal programs, future plans for each program and efforts to streamline operations and communications between the three programs.

Conversion of Research and Test Reactors to Low Enriched Uranium Fuel: Technical Overview and Program Status[1]

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Nuclear research and test reactors worldwide have been in operation for over 60 years. Many of these facilities operate with high enriched uranium (HEU – U^{235} enrichment ≥ 20%) fuel. In response to increased worries over the potential use of HEU from research reactors in the manufacturing of nuclear weapons, the U.S Department of Energy (DOE) initiated a program – the Reduced Enrichment for Research and Test Reactors (RERTR) - in 1978 to develop the technology necessary to reduce the use of HEU fuel in research reactors by converting them to low enriched uranium (LEU) fuel. The reactor conversion program was initially focused on U.S.-supplied reactors, but in the early 1990s it expanded and began to collaborate with Russian institutes with the objective of converting Russian-supplied reactors to the use of LEU fuel.

Increased security concerns in recent years have led to the establishment of the Global Threat Reduction Initiative (GTRI) by the U.S. DOE's National Nuclear Security Administration. The overall GTRI objectives include securing radiological and fissile materials. A follow up conference for the International GTRI partnership held at the IAEA in September 2004 [1] established the framework for international collaborations in meeting the goals of the program. As an integral part of the GTRI, the Conversion Program has accelerated the schedules and plans for conversion of additional research reactors operating with HEU. A total of 129 reactors are included in the scope of the Program.

The major technical activities of the Conversion Program include: (1) the development of advanced LEU fuels; (2) conversion analysis and conversion support; and (3) technology development for the production of Molybdenum-99 (Mo^{99}) with LEU targets.

The key factor in enabling the conversion of a research reactor lies in the availability of a fuel with much greater uranium content, to compensate for the reduction in the content of U^{235} in the LEU material. Several high density LEU fuels have been developed. The RERTR program developed the uranium disilicide dispersion fuel. General Atomics developed LEU fuel for TRIGA reactors. Oxide tube-type LEU fuels for the conversion of Russian-supplied reactors have also been qualified for use in several reactors. An accelerated fuel development and qualification program focusing on UMo alloy fuel is currently underway with the objective of qualifying very high density LEU fuels to enable the conversion of high performance research reactors that are not convertible with the existing qualified fuels.

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The conversion analysis and support activity provides the required analytical and design evaluations to support the program. Since the inception of the RERTR program, analysis methods and codes have been developed specifically for the analysis of research reactors. The methods and codes are currently evolving to incorporate the latest tools and data and have been validated with experimental data. Conversion analysis in general includes three major tasks:

- *Feasibility studies* to determine suitable LEU fuel assembly designs for each reactor. The design of the fuel assembly may be an iterative process with the fuel development activities.
- *Operational and safety analysis*, necessary to demonstrate that the transition from HEU to LEU fuel can be done safely and without interrupting normal operations
- *Resolution of regulatory issues* to obtain regulatory approval for the conversion to LEU fuel**.** It must be demonstrated that all safety requirements are met.

The programmatic objective of the technology development for Mo^{99} program element is to eliminate the use of HEU targets in production of Mo^{99}. The program intends to accomplish this objective through the development of LEU targets and chemical processing methods that do not significantly impact the isotope production yields, costs and waste treatment and disposal with respect to current production with HEU targets.

Since the inception of the Conversion Program, 51 of the 129 reactors have been converted to LEU fuel or have shutdown prior to conversion. The current goal is to convert the remaining 78 reactors in the list of candidates by the year 2018. Of the 78 remaining research reactors within the scope of the Conversion Program, 50 can be converted with existing LEU fuels, the high density UMo fuel under development will allow the conversion of 19 additional reactors, and the remaining 9 reactors require further analysis.

A key element of the Conversion Program is the coordination with multiple organizations, ranging from facility operators to regulatory bodies, and significantly, the International Atomic Energy Agency (IAEA). The IAEA supports the objectives of the Conversion Program and is currently leading coordinated research projects for conversion analysis of Miniature Neutron Source Reactors and technology development for Mo^{99} production, in addition to country-specific conversion support projects under the Technical Cooperation Department.

The paper will provide a more detailed overview of the status of the program, the technical challenges and accomplishments, and the role of international collaborations in the accomplishment of the Conversion Program objectives.

1. IAEA, *Global Threat Reduction Initiative, International Partners' Conference, Summary of the Proceedings and Findings of the Conference*, Vienna, Austria, September 18-19, 2004, available at IAEA Web site, www.iaea.org.

High density fuel development for Research Reactors

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An international effort to develop, qualify, and license high and very high density fuels has been underway for several years within the framework of multiple national RERTR programs. The current state of development is the result of significant contributions from many laboratories, specifically CNEA in Argentina, AECL in Canada, CEA in France, TUM in Germany, KAERI in Korea, VNIIM, RDIPE, IPPE, NCCP and RIARR in Russia, ANL and INL in USA.

These programs are mainly engaged on UMo dispersion fuels with densities from 6 to 9 gU/cm^3 (high density fuel) and UMo monolithic fuel with density as high as 16 gU/cm^3 (very high density fuel)

The limitations discovered under severe (high power) operating conditions during tests on first generation high density fuel, postponed initial plans and schedules for qualification tests and the subsequent licensing program [1], [2], [3].

These failures, firstly observed in French FUTURE experiment (see Fig. 1 & 2), and confirmed in the US miniplates and Russian tubes, are clearly attributed to the excessive formation of the interaction product and the inability of this compound to retain fission gas in the form of stable bubbles.

Fig.1: FUTURE metallography in the pillowed region (plate transverse section)

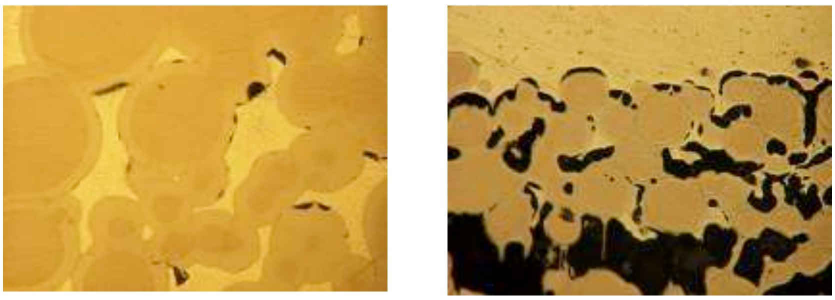

Fig.2: FUTURE metallographies with $(UMo)Al_x$ interaction product around the UMo particles, lenticular shape porosities at the interfaces with Al matrix, and final meat decohesion

Thus, it was clear that the only way to improve UMo dispersion fuel behavior for medium to high power operating conditions was to drastically reduce the formation of the interaction compound.

To solve the problem, full scientific and experimental programs were launched to understand the mechanism of UMo/Al interaction, develop and test in-pile solutions to avoid an inappropriate behavior under irradiation:

1. Thermodynamic studies to determine, through the U-Mo-Al ternary system, the phase¨s equilibriums at different temperatures (CEA, CNEA, ANL)

2. Metallurgical studies on diffusion couples to get a better understanding of the mechanisms of the UMo/Al interaction phenomena and underline the parameters which could prevent this reaction (KAERI, INL, CNEA, CEA, VNIINM)
3. Numerical simulation studies on effect of some additives in particles or matrix (CNEA, ANL)
4. High energy ion irradiation, typically 100 MeV Xenon, in order to simulate the damage created by fission products (TUM, CEA, INL).
5. In reactor tests on promising solutions (INL, ANL, CEA, KAERI, RIARR)

The in-pile irradiation programs are specifically designed to evaluate and optimize the irradiation behaviour of UMo dispersion fuels (high density fuel) as well as UMo monolithic (very high density fuel). This experimental work will culminate with irradiation tests on full size plates that will demonstrate the integrated package that includes both manufacturing and irradiation aspects, as has been previously demonstrated by the French UMo Group

A number of modifications to the fuel and/or the matrix aluminium, based on available literature and experience, have been evaluated specifically in US RERTR Experiments [4] and will be used to help select fuel compositions for future full size plate testing.

The most promising remedies for dispersion fuel appears to be an addition of silicon to the matrix or the oxidation of the particles without any additive in Al matrix [5]. For monolithic fuel, tests are in progress to improve cohesion between foil and clad during and after irradiation. Promising solutions include the application of a silicon rich layer to the interface between the cladding and fuel, application of a zirconium diffusion barrier to the interface, or by using Zy cladding co-rolled with the UMo foil [6].

The final paper will give the details of this entire program and will provide an updated schedule for qualification and licensing of the different solutions.

[1] J.L. Snelgrove, P. Lemoine, N. Arkhangelsky, P. Adelfang, ¨ Qualification and Licensing of U-Mo fuel¨, Proceedings of the 7th Int. Topical Meeting Research Reactor Fuel Management, Aix en Provence, France, March 9-12, 2003.

[2] P. Lemoine, J.L. Snelgrove, N. Arkhangelsky, L. Alvarez, ¨UMo dispersion fuelk: Results and status of Qualification Program¨, Proceedings of the 8th Int. Topical Meeting Research Reactor Fuel Management, March 21-24 2004, München, Germany.

[3] J.L. Snelgrove, P. Lemoine, L. Alvarez, ¨High density UMo fuels ¨ Latest results and reoriented Qualification Programs¨, Proceedings of the 9th Int. Topical Meeting Research Reactor Fuel Management, April 10-13 2005, Budapest, Hungary.

[4] M. Meyer & al ¨Progress in the RERTR Fuel Development Program¨, Proceedings of the 10th Int. Topical Meeting Research Reactor Fuel Management, 30th March -3rd May 2006, Sofia, Bulgaria.

[5] S. Dubois, J. Noirot, J.M. Gatt, M. Ripert, P. Boulcourt, P. Lemoine, ¨Comprehensive overview on French Experiments in Irradiation Tests and PIE on High density UMo/Al dispersion fuel¨, Proceedings of the 11th Int. Topical Meeting Research Reactor Fuel Management, March 11-15, 2007, Lyon, France

[6] D. Wachs, P. Lemoine, C. Jarousse, ¨Status on the LEU fuel development¨, Proceedings of the 11th Int. Topical Meeting Research Reactor Fuel Management, March 11-15, 2007, Lyon, France

Measuring Progress in Reactor Conversion and HEU Minimization Towards 2020 – the Case of HEU-fuelled Research Facilities

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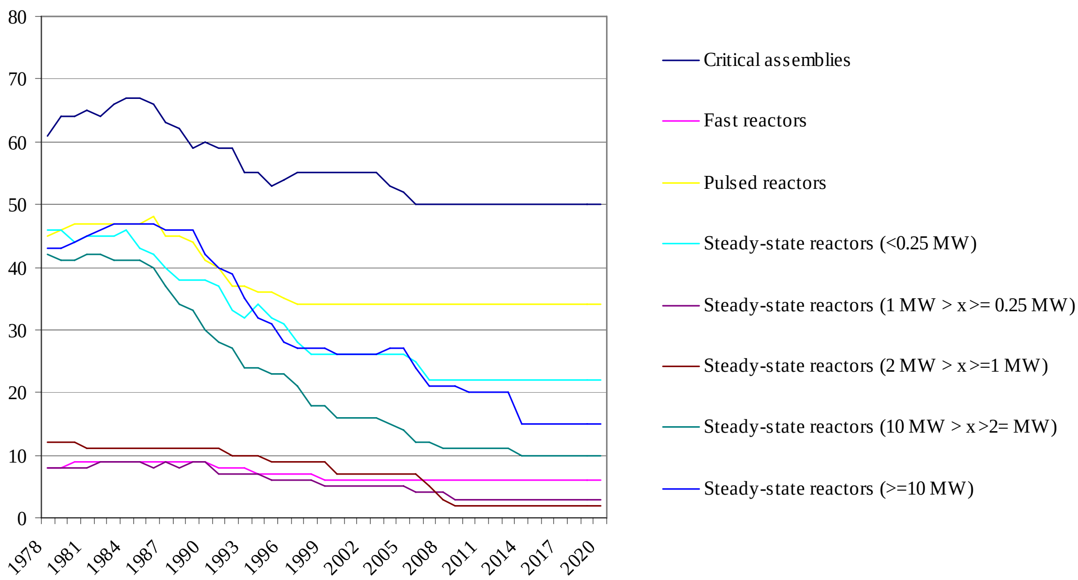
The primary impediment that prevents nuclear proliferation is the lack of access to fissile materials. Thus, a recognized objective internationally has been to minimize the use of HEU and reduce the number of locations with HEU present. Yet, nearing the 30 year anniversary of this objective, the number of HEU-fuelled research facilities in operation remains high, HEU is still being used in large quantities, and significant quantities of HEU is still to be found in a large number of unsecured locations worldwide. This paper identifies the most important indicators for measuring progress for the historical and future national and international efforts for research reactor conversion and decommissioning of vulnerable facilities.

Figure 1. Categories of HEU-fuelled Research Facilities in Operation 1978 - 2020[1 2]

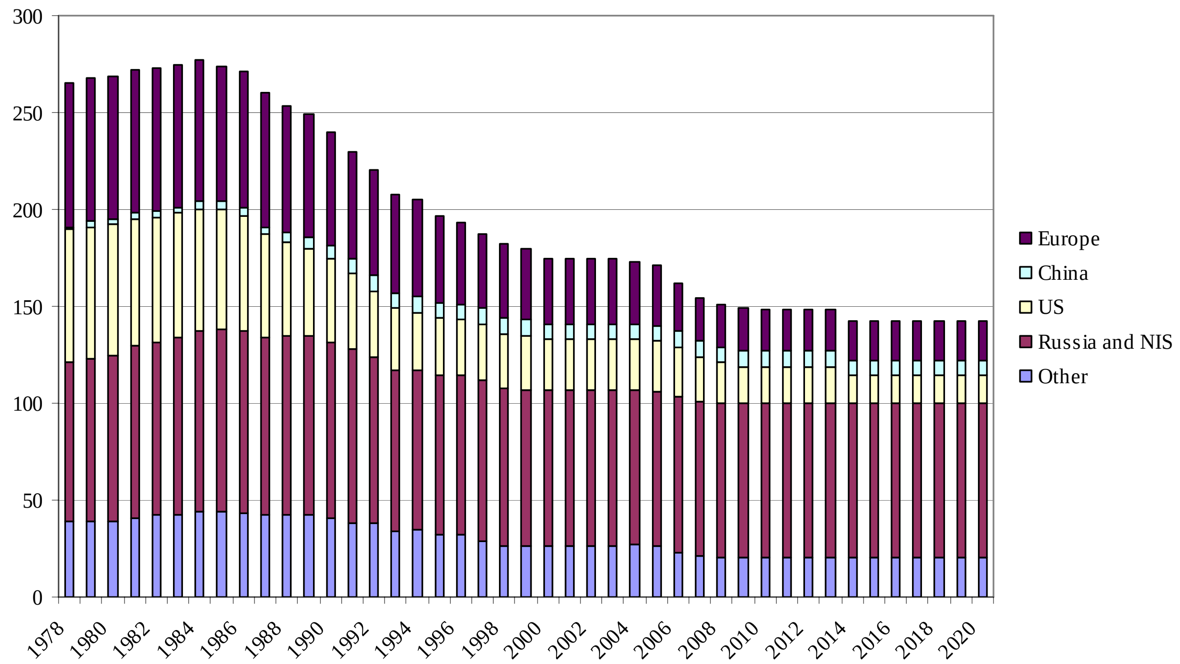


The primary impediment that prevents nuclear proliferation is the lack of access to fissile materials. Thus, a recognized objective internationally has been to minimize the use of HEU and reduce the number of locations with HEU present. Yet, nearing the 30 year anniversary of this objective, the number of HEU-fuelled research facilities in operation remains high, HEU is still being used in large quantities, and significant quantities of HEU is still to be found in a large number of unsecured locations worldwide. This paper identifies the most important indicators for measuring progress for the

historical and future national and international efforts for research reactor conversion and decommissioning of vulnerable facilities.

The present concern for HEU could be traced back to the International Fuel Cycle Evaluation (INFCE) initiative.[3] The INFCE report recognized that there were over 140 HEU-fuelled research reactors with significant power-output (between 10 kWt and 250 MWt) in operation in more than 35 countries, with nominal power in excess of 1700 MW annually, consuming each year more than 1200 kg U-235.After re-establishing this baseline measurement, this paper assesses that similar figures for 2006 are 1150 MW, 72 facilities, primarily as part of the steady-state reactor categories given in Figure 1, with an HEU consumption in excess of 700 kg. In addition, the potential for conversion of other types of facilities not included in the INFCE study, among others pulsed facilities and various types of critical assemblies, are discussed. When including all HEU-fuelled research facilities converted, commissioned or decommissioned after 1978; more than 310 facilities are identified and considered when identifying reductions in HEU consumption (e.g. converted vs. shutdown facilities). This assessment includes facilities within and outside the scope of current international programs, in particular the Global Threat Reduction Initiative (GTRI) and the program for Reduced Enrichment for Research and Test Reactors (RERTR).[4] In total, when also including HEU-fuelled isotope production facilities, more than 150 HEU-fuelled facilities still remain in operation today. Additional measures for assessing the overall risk and need for conversion, in addition to the different categories of material as suggested in InfCirc 224, Rev. 4, then considering also average core inventory and HEU consumption together with the fuel burn-up, are suggested as part of this paper.

Figure 2. HEU-fuelled Research Facilities in Operation in Different Global Regions 1978 - 2020



The main conclusion is that a more comprehensive approach to HEU minimization and conversion is needed to achieve a real HEU clean-out with respect to all types of research facilities. While the idea of a HEU global cleanout indeed has gained increased political attention, there is no clear and unified international policy and agreed measures on progress and completion. Real progress in all relevant areas may not be expected until a higher degree of consensus amongst all states on the final objectives is established. Again, the IAEA – maintaining an international rather than a national focus on the HEU clean-out and minimization activities – seems the obvious forum for information exchange as well as technical support, coordination, and cooperation. On the political level, renewed conversion interest may be accomplished by a second International Fuel Cycle Evaluation initiative. Particular support should be given to the current initiatives sponsored by the GTRI; for example the RERTR program is instrumental in assessing gaps and issues which need increased attention. This paper suggests, based on the findings above, that a another GTRI global conference should take place to facilitate the needed interaction and attention to prioritize the next steps and address urgent needs – for example in relation to spent fuel from HEU-fuelled facilities.

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[1] The statistics presented in this extended synopsis is primarily based different electronic and printed versions of the IAEA Research Reactor Database (www.iaea.org/RRDB). Specific entries on individual facilities may have been included on the basis of documentation worked out of individuals and/ or national competent authorities; however, this is thoroughly discussed in the final paper in relation to each figure.

[2] For research reactors, the power levels for categorizing research reactors suggested in International Atomic Energy Agency, *The Application of Research Reactors*, IAEA-TECDOC-1234, August 2001 has been applied in this paper.

[3] INFCE, *Advanced Fuel Cycle and Reactor Concepts*, Report of INFCE Working Group 8, IAEA, 1980, p. 137: *"The trade in and widespread use of highly enriched uranium and the production of fissile materials constitute proliferation risks with which INFCE is concerned. Proliferation resistance can be increased by: 1. Enrichment reduction preferably to 20% or less which is internationally recognized to be a fully adequate isotopic barrier to weapons usability of ^{235}U; 2. Reduction of stockpiles of highly enriched uranium; 3. Reduction of the annual production of fissile materials in research reactors, although attainment of weapons-usable material would require spent fuel reprocessing. For example, for some research reactors fuelled with natural uranium the proliferation resistance might be improved by utilizing slightly enriched uranium, which reduces the annual plutonium production."*

[4] Global Threat Reduction Initiative, GTRI Strategic Plan 2007, January 2007 (as accessed April 17, 2007 http://www.nnsa.doe.gov/na-20/docs/GTRI_Strategic_Plan_2007.pdf) and Dr. Armando Travelli, *Fuel Issues – Replacement of HEU*, Presentation at the IAEA Scientific Forum 2004, September 2004 (as accessed http://www-pub.iaea.org/MTCD/Meetings/PDFplus/2004/gcsfSess3-Travelli.pdf).

Session 12: New Research Reactor Projects

Synopses no. IAEA-CN-156/	Synopses Title	Main Author
S-21	Commissioning the new OPAL Research Reactor	Irwin, A.
S-23	Reactor PIK status of construction	Konoplev, K.A.
S-37	Licensing of the OPAL Reactor during construction and commissioning	Summerfield, M.
S-24	Concept for a new research reactor in Ukraine	Lobach, Y.

OPAL: Commissioning a New Research Reactor

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In 1997, the Australian Federal Government decided to commence the process of replacing ANSTO's 10 MW HIFAR reactor, which had operated since 1958, with a new 20 MW multi-purpose reactor for radioisotope production, irradiation services and neutron beam research.

The contract which included design, construction and commissioning was awarded to INVAP in 2000. INVAP designed an open pool type reactor, LEU fuelled, light water cooled with a heavy water reflector surrounding the core. The design includes a cold neutron source.

ANSTO established the organisation for the project in accordance with the IAEA Safety Guide for Commissioning of Research Reactors [1]. A joint ANSTO-INVAP *Commissioning Management Group* ensured resources were available and authorized the start of each commissioning stage. The ANSTO *Replacement Research Reactor Project Group* was responsible for all contractual matters, overseeing the construction, participating in pre-commissioning and acting as the focus for the licensing work. ANSTO also established a *Commissioning Operations Group* to take part in commissioning. This group formed the nucleus of the final Reactor Operating organisation.

INVAP design activities were based in Bariloche, Argentina but throughout construction and commissioning, INVAP maintained a strong presence in Australia, arranging for appropriate staff to be brought to site to support each stage. Although INVAP was responsible for commissioning, it was agreed during pre-commissioning that ANSTO Commissioning Operations Group would be involved in all commissioning activities. In this way the group gained valuable experience, and by the time of fuel loading was actually carrying out all operations, under the supervision of INVAP. In addition to training by INVAP and plant experience, an extensive classroom training program, including training on a simulator, was completed before fuel loading. Establishment of a good operating safety culture was of fundamental importance.

The Operating organisation is based on the IAEA Safety Guide [2]. ANSTO decided to maintain HIFAR operating 24/7 during the commissioning and early operation of the new OPAL reactor. There was not enough existing staff to operate two reactors, but some experienced operations staff were able to be released from HIFAR to join the OPAL organisation. One of the success of the project was to combine these experienced staff with ten new young engineering/science graduates, persons with experience of other reactors, staff seconded from other divisions on site and contract staff, to form an effective operating team.

Stage A commissioning was completed in 74 days in early 2006 without any major problems. All systems were operated with dummy fuel assemblies in the core to obtain the correct flow characteristics.

In Stage B1 (5 days) , 14 of the 16 LEU fuel assemblies were loaded and the reactor taken critical for the first time on 12 August 2006. The main issue during Stage B1 commissioning was spurious trips from the nucleonics instrumentation due to noise. This problem was solved by close attention to earthing, connections and cable screening.
In Stage B2 (25 days), the core loading was completed and the key nucleonics and reactivity parameters of the core measured at powers up to 400 kW. Minor problems with the nucleonics instrumentation were solved by instrument tuning.
Stage C commenced in October 2006 and the reactor power was increased in stages until the full power 20 MW was achieved on 3 November 2006. A problem with the core outlet temperature measurement was resolved by a simple modification to the primary cooling flow around the temperature detector. The cold neutron source could not be tested in cold mode during this stage, but a problem with the turbine has now been resolved and cold testing completed. The secondary cooling tower performance, although adequate for full power operations, did not meet the design specification for extreme weather conditions in Australia and further system tuning has been necessary.
During December, a degradation of the heavy water in the reflector was observed. Tests identified minor leaks in a non-structural seal weld between the reflector vessel and the neutron beam tubes. As of April 2007, the reactor was still operating at full power whilst INVAP proposals for repair were being discussed.

In addition to producing thousands of project documents, INVAP produced the design operations and maintenance manuals. Each manual has been reviewed to incorporate commissioning experience. ANSTO established the OPAL Business Management System (BMS), obtained ISO 9001/ISO 4001 accreditation and the INVAP manuals are now being incorporated into this BMS.

The Operating Limits and Conditions (OLC) are the basis for safe operation of the facility. In drafting the OLCs, ANSTO used IAEA guidance and also an IAEA Review Team examined the OLCs in detail.

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Draft Safety Guide Commissioning of Research Reactors, NS-G-4.1 (NS 259), Vienna

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Draft Safety Guide The Operating Organisation and the Recruitment, Training and Qualification of Personnel for Research Reactors, DS 325, Vienna

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Draft Safety Guide Operational Limits and Conditions for Research Reactors, NS-G-4.3 (NS 261), Vienna

REACTOR PIK STATUS OF CONSTRUCTION

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Report gives description of the current status of reactor PIK that is under construction at Petersburg Nuclear Physics Institute. Powerful research reactor has thermal neutron flux $5 \cdot 10^{15}$n/cm^{2}sec.

Plans of complete construction work and putting reactor PIK in operation are considered. Second part of report gives results of critical experiments supported reactor first criticality program.

Last years the rector construction has been financed in poor level less then 200 million rubles annually. We are waiting remarkable increase and prepare three construction phase for finalize design.

The equipment for the first phase is mounted on about 90%. Practically all subsystems of the primary coolant circuit are completed. Water test on this systems and primary coolant circuit itself begins. Electrical and instrumentation systems also are in test. Fire control system and physical protection systems for the first phase partly are in operation.

On the rector full-scale critical facility startup core and steps for increasing power are examined.

The PIK reactor features high experimental potential not only due to the high intensity neutron beam, which is approximately by one order higher than that at the existing medium-power reactors, but due to availability of the sources of hot, cold and ultracold neutrons as well. Therefore, as compared to the research reactors created in 50-60s, the PIK reactor will give unique opportunities for extending the neutron beam research activities conducted currently in Russia, as well as for launching new researches that are technically impossible at the moment.

Choice of Area - New research reactor projects.
Type of presentation – Oral presentation.

LICENSING OF THE OPAL REACTOR DURING CONSTRUCTION AND COMMISSIONING

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This paper presents a description of the licensing activities associated with the construction and commissioning of the Australian Nuclear Science and Technology Organisation's (ANSTO) OPAL reactor. It addresses the Construction Licence, the interface between ANSTO, INVAP (the contractor with responsibility for design and construction of the facility) and the Australian Radiation Protection and Nuclear Safety Agency (ARPANSA, the Australian nuclear regulator) during the construction of OPAL, specific licensing issues that have arisen during the construction and commissioning process, and the Operating Licence Application. Particular emphasis will be given to the way in which the licensing process is integrated into the overall project program and the lessons learnt that may be of benefit to other licensees and regulators.

1. The OPAL Construction Licence

The Application for the Facility Licence, Construction Authorisation was submitted to ARPANSA in May 2001. This licence was required in accordance with the ARPANS Act [1] and Regulations [2] in order to commence construction of OPAL. Following an extensive review process, the CEO of ARPANSA granted the Facility Licence, Construction Authorisation in April 2002. It incorporated 18 Licence Conditions, some of which had significant impacts on the construction and commissioning of OPAL, as described below.

1.1 Licence Condition 4.6: Construction of Items Important to Safety

ARPANS Regulation 54 states that "*The holder of a licence, or a person covered by a licence, must not construct an item that is important for safety, and that is identified in a safety analysis report, as part of the construction of a controlled facility, unless the CEO has given the holder, or the person, approval to construct the item*".

It defined "items important for safety" as all the Safety Category 1 and Safety Category 2 structures, systems and components.

This licence condition had a significant impact on the construction of OPAL, due to the number of items that required such approval.

During the course of the construction of OPAL, over 120 individual submissions were made to ARPANSA under Licence Condition 4.6. The preparation of such a large number of submissions by ANSTO and their subsequent review and evaluation by ARPANSA imposed a significant workload on both organisations.

1.2 Licence Condition 4.7: Commissioning of Items Important to Safety

Licence Condition 4.7 is an extension of Licence Condition 4.6, and states that the approval of the CEO of ARPANSA was required to commission individual items important for safety as part of the cold commissioning of OPAL. However, in the light of the experience gained in the implementation of Licence Condition 4.6, the CEO of ARPANSA subsequently revised

Licence Condition 4.7 such that it was split into two parts, one covering the overall arrangements for cold commissioning and the other identifying specific systems and components for which detailed information about the commissioning tests was required to be provided to ARPANSA.
As a result, instead of the 120 plus submissions under Licence Condition 4.6, there was a single submission of the overall arrangements for the cold commissioning, together with the formal submission of specific pre-commissioning and commissioning test procedures as identified by the CEO of ARPANSA. A total of 24 tests were specified by ARPANSA, including those of the first and second shutdown systems, the containment permeability, and the establishment of natural circulation cooling.

2. Licensing issues arising during construction

As can be fully expected with a project of this type, a number of licensing issues arose during the course of construction and cold commissioning of OPAL. The most significant with respect to their impact on the project were:

- Discovery of a geological fault during site excavations
- Concrete cracking in the reactor building basement
- Reactor pool heavy water penetration cut-outs
- Repairs to reactor pool

3. The OPAL Operating Licence

The Application for the Facility Licence, Operating Authorisation was submitted to ARPANSA in September 2004. It was sub-divided into 5 parts as follows:

Part A: General information on the purpose and location of the Reactor Facility
Part B: The plans and arrangements for managing safety of the Reactor Facility
Part C: The Safety Analysis Report (SAR) for the Reactor Facility, together with associated safety and licensing documentation
Part D: The Operational Limits and Conditions (OLCs) for the Reactor Facility
Part E: The plans and arrangements for hot commissioning the Reactor Facility

The Application was supported by a large volume of documentation, including detailed engineering and analysis reports supporting the SAR, the plant design, operation and maintenance manuals prepared by INVAP and the OPAL Business Management System (BMS). Also considered as part of the supporting documentation to the Application were all the submissions made under Licence Condition 4.6.
The CEO of ARPANSA granted the Facility Licence, Operating Authorisation in July 2006. This operating licence authorised ANSTO to load fuel, perform hot commissioning and operate OPAL. In addition to the Licence Conditions inherent in the ARPANS Act and Regulations, it also identified six Licence Conditions covering periodic safety reviews, periodic security reviews, safety culture, quarterly reporting, discharge authorisations and an index of licensing documentation.

[1] Australian Radiation Safety and Nuclear Safety (ARPANS) Act 1998,
[2] Australian Radiation Safety and Nuclear Safety (ARPANS) Regulations, 1999

Concept for a new research reactor in Ukraine

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Currently in Ukraine there is only one high-power research reactor – the WWR-M – in operation. The final shutdown of this reactor is expected in 2015 and, therefore, the construction of a new research reactor is under current consideration. The first stage was to develop a Concept for a new multi-purpose research nuclear reactor.

The Concept substantiates the need for a new research reactor in Ukraine and determines the optimum alternative. The Concept defines the basic requirements for the reactor's design, construction and operation at an appropriate safety level. The Concept involves considerations about the reactor type and relevant technical parameters, the reactor's main uses, possible locations, and the necessary scientific and technical infrastructure.

The following prerequisites underly the Concept:
- the new reactor is intended as a replacement for the existing WWR-M reactor;
- the new reactor will be designed and constructed with the aim of satisfying current and future needs of the State as a power source of neutrons;
- it will have multi-purpose use for both fundamental and applied investigations;
- the new research reactor will be an integral part of the Ukrainian nuclear power industry;
- the design will follow modern international trends for increased reactor utilization whilst maintaining all nuclear and radiation safety requirements;
- the benefits of construction to the domestic industry and economy;
- the new research reactor should be the core installation of a new national nuclear centre.

The main aim of a national nuclear centre based around the new research reactor will be an effective infrastructure to investigate and develop technology in the following areas:
- nuclear power (primarily, testing of structural materials for the power reactors under neutron irradiation; testing of new materials, heat-carriers, fuels, devices and operational regimes);
- fundamental science (neutron physics and nuclear spectroscopy; high-temperature semiconductors; biology, chemistry, medicine, ecology etc);
- applied nuclear technologies (radionuclide production for medicine and industry; silicon doping; neutron-activation analysis; sterilization technologies; modification and strengthening of polymers, metals and alloys etc);
- personnel training and retraining.

The reactor is expected to be a pool-type with thermal power in the range of 20-30 MW and having an averaged neutron flux about $2.0\text{-}4.0\times10^{14}\ n\cdot cm^{-2}\cdot s^{-1}$. Low-enriched uranium fuel will be used for the reactor operation. Reactor and technological systems will be located in the containment. The following laboratories and complexes are planned around the reactor:
- laboratory for the radionuclide production;
- laboratory for the neutron doping of silicon and another materials;
- laboratory for the neutron-activation analysis;

– material testing complex;
– medical complex.

The selection of the reactor site will be made according to established procedures at the design stage. The suburbs of the town of Slavutich in the Kiev region are considered now as the most preferable location; however, some other places should be considered too. Preliminary estimates are that the area of reactor site will be about 40-50 hectares. Taking into account the duration of the reactor’s construction, operation and decommissioning, the term of site use will be about 100 years. Implementation of the Concept tasks is foreseen during 2007-2015.

Poster Presentations: Safety

Synopses no. IAEA-CN-156/	Synopses Title	Main Author
S-9	Safety analysis of a 1-mw pool-type research reactor	Hamidouche, T.
S-12	Safety Core Parameters Prediction in Research Reactors using Artificial Neural Networks: A Comparative Study of various Learning Algorithms	Mazrou, H.
S-18	Determining Degradation of Sensors used for Fuel Assemblies Cooling Temperature measurement by the Analysis of Thermal Fluctuations Noise	Saadi, S.
S-20	Physical calculations for the design of irradiation device and nuclear heating calculation in silicon ingot at Es-salam research reactor	Widad, T.
S-43	Upgrading I&C for the Es Slam Research Reactor	Djaroum, B.
U-48	Training and Qualification of Reactor Operating Personnel	Kulak, R.
S-26	Experimental Heat Transfer Analysis of the IPR-R1 TRIGA Reactor	Mesquita, A.
U-1	Appraisal for the Operation Safety of SPRR-300 in recent years	Chen, W.
S-14	Calculations and Measurements at the Training Reactor VR-1	Rataj, J.
S-54	Improvements of the Training Reactor VR-1 – Only Way how to be Attractive for Students and Scientists	Sklenka, L.
S-44	Study of temperature effects on ULYSSE reactor for training and qualification of operating personnel	Foulon, F.
S-17	Monte Carlo Simulation of GRR-1 Core and Neutron Irradiation Positions Using MCNP	Stamatelatos, I.E.
S-28	Radiological characterization of the GRR-1 pool	Tzika, F.
S-71	The Next 20 Years Operation of the 36 Years Old Hungarian Training Reactor	Aszódi, A.
S-15	Validation of the Monte Carlo Method of MVP Code on the First Criticality of Indonesian Multipurpose Reactor	Sembiring, T.M.
S-16	Neutronic Analysis of RSG-GAS Silicide core with Uranium density of 4.8 gr/cc	Setiyanto
S-48	Fault Position Estimation Using Color Segmentation Approach to the Thermal Infrared Image of a Cabling Network in the Nuclear Reactor	Nugroho, D.H.
S-56	Water Chemistry Surveillance for Multi Purpose Reactor 30MW GA Siwabessy, Indonesia	Sunaryo, G.R.
U-51	Assessment of qualification of RSG-GAS Reactor operator in INDONESIA	Pandi, L.Y.
S-49	Developing an Ultrasonic NDE System for a Research Reactor Tank	Perets, Y.

S-70	Neural networks application to CRDM and thermohydraulic data validation	Sepielli M.
U-32	Investigation of JRR-3 control rod worth changed with burn up of follower fuel elements	Hosoya, T
S-72	Project to Replace the Control and Protection System at the WWR-K Research Reactor	Chakrov, P.
S-11	Modeling of Thermal Hydraulics Behaviour in Reactor Core	Mohamed, K.N.
S-67	PUSPATI TRIGA Reactor: 25 Years of Safe Operation and Strategies for Ensuring Safety and Security	Z. Masood
S-68	Core Calculation of 1MW PUSPATI TRIGA Reactor (RTP) using Continuous Energy Method of Monte Carlo MVP Code System	Abdul Karim, J.
S-10	Operational Experience and Programmes for Optimal Utilization of the Nigeria Research Reactor-1	Jonah, S.A.
S-32	Regulatory control of the Nigerian Research Reactor (NIRR-1)	Ogharandukun, M. O.
S-47	Comparative dose calculation for TRIGA HEU and LEU fuel in nuclear accident situations	Margeanu, S.
S-8	Safety Analysis of MNSR Reactor during Reactivity Insertion Accident Using the Validated Code PARET	Hainoun, A.
S-19	Fuel management methodology upgrade of Thai Research Reactor (TRR-1/M1) using SRAC computer code	Tippayakul, C.

IAEA-CN-156/

Safety Analysis of a 1-Mw Pool-Type Research Reactor

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The degree of consequences for any event postulated in safety analysis of a research reactor is determined by the response characteristics of the reactor system, which is a combination of non nuclear characteristics (like control systems) and inherent or intrinsic characteristics which are strongly nuclear.

In order to perform a safety analysis of a reactor, certain intrinsic key parameters namely Prompt neutron lifetime Λ, Delayed neutron fraction β, Reactivity feedback coefficients, and Reactor control systems, are important to understand the dynamic behaviour of the reactor core during any transient situation [1]. Indeed, most of these key parameters are highly dependant upon the core configuration such as the lattice dimensions, the disposition of the fuel and control assemblies, the control systems, the reflector elements in the grid matrix ...etc; all these related configuration's data are used to determine the well known inherent core intrinsic parameters.

In this order, the first part of this work concerns the determination of some inherent characteristics of two different configurations of an MTR pool type research reactor: one configuration contains 16-fuel elements (C-16) and the other, 25-fuel elements (C-25) [2] and the second part of the work concern the analysis of some postulated initiating event using the key parameters calculated in the previous step and the importance of such inherent parameters are emphasized by considering the response of the two different core configurations during similar protected postulated initiating events. Furthermore unprotected cases as well as transients without feedback are also considered.

The determination of the inherent characteristics of any given nuclear system passes throughout the use of some validated computer tools and techniques. For this purpose, a computer code package namely COMPACK-LHW [3, 4] has been developed for MTR research reactor applications. The individual modules of the package and the overall computational strategy along with the techniques of modelling used have been already assessed and qualified in a previous international benchmark problem [5-9]. This is being performed into two major steps:

The first step, require a rigorous analysis of the reactor component by an identification of the different types of cells, which may represent physically any region of the core configuration taking into account the fuelled and non-fuelled regions of the core loading, along with the grid plate and the reflector assembly. Thus, in present case, cell calculations are performed using the WIMS-D/4 [10] code for five (05) different identified and homogenized unit cell types of fuel and non-fuel elements present in the core. The calculations of macroscopic cross sections for the fuel and non-fuel cells were carried out using 69 energy groups with collapsing procedures version. A 5-energy groups cut-off was adopted for the following materials identified in the core: Fresh fuel element; H_2O – water reflector; Graphite reflector; Al+H_2O for the external region; Stainless steel + H_2O for the external region in the control blade without absorber.

The broad group cross sections were also generated for different fuel temperatures ranging from 20°C to 150°C and for water temperature ranging from 20°C to 100°C and for change in water density. These cross sections were used to calculate the isothermal feedback reactivity coefficients. In addition, the excess reactivity values for the fresh core considered were also computed at different times of full power operating reactor using the diffusion theory model.

In the second step, global core calculations with the locally two dimensional multigroup diffusion code MUDICO-2D [11] are performed. Effective multiplication factor, reactivity feedback coefficients (Fuel temperature, moderator temperature and density/void coefficients), power peaking factors, power defect are then obtained. Perturbation calculations module of the code is used to evaluate kinetic parameters, like effective delayed neutron fraction and prompt neutron life time.

The accident analysis is related to the simulation of some postulated initiating events in two different core configurations of the same reactor. The purpose is to assess the dynamic response of the core and to investigate whether the cladding failure (melting) could be induced. To reach this goal, the in-house coupled thermal-hydraulic-point kinetic RETRAC-PC [12, 13] computer code was applied. The results obtained showed that the clad melting threshold is not reached over a wide range of transient situations even when the scram is disabled (unprotected transient). The importance of the inherent safety parameters of a reactor core under different configurations has been emphasized. The results of the protected RIA transients are given on Table 2 whereas some results of the unprotected transients (without scram) are given on figures 1 and 2.

Table 2. Transient Parameter for protected RIA

Configuration / Parameter	C 16		C25	
	RETRAC-PC	PARET	RETRAC-PC	PARET
Trip time (sec)	0.547	0.547	0.546	0.547
Peak Power (MW)	51.676 (0.612)	51.756 (0.611)	57.325 (0.613)	60.087 (0.616)
Energy at Peak power (MJ)	1.41	1.37	1.59	1.67
Peak Fuel Temperature (°C)	157.60 (0.640)	150.60 (0.635)	146.86 (0.644)	142.73 (0.643)
Peak Clad Temperature (°C)	135.22 (0.654)	125.83 (0.651)	131.62 (0.661)	120.66 (0.659)
Coolant Temperature (°C)	63.93 (0.928)	63.47 (0.985)	63.97 (0.999)	70.60 (1.170)

()time of occurrence in seconds

As can be seen on figure 1, in case of no scram, the power continues its excursion until it is stopped by the inherent feedback's mechanisms of the reactor. For this case, the power reaches a maximum of 150 MW and quenches due to the strong feedback effect even the scram system is disabled during the course of the accident. No damage of the cladding occurs according to the results shown in Fig. 2. The maximum temperature reached (less than 186°C) is far below the melting point of aluminium that is 600 °C or even from the safety margin of 450°C fixed due to swelling of Aluminum.

Furthermore, the strong influence of the feedback mechanism on the stability of this research reactor are emphasized in the following by investigating the reactor response under protected RIA (with scram) but in absence of any feedback mechanisms. The results emphasized what has been stated in previous sections concerning the strong feedback effects on the excursion runaway before any effective scram. This exercise confirms also the importance of these inherent core parameters for the safety of the considered reactor (self shut-down).

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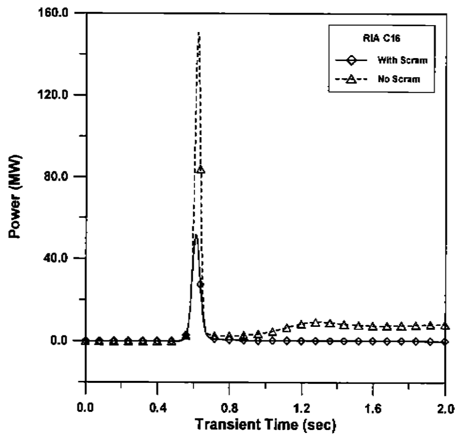

Fig. 1 : Power behaviour during RIA transient with and without scram (C16 configuration)

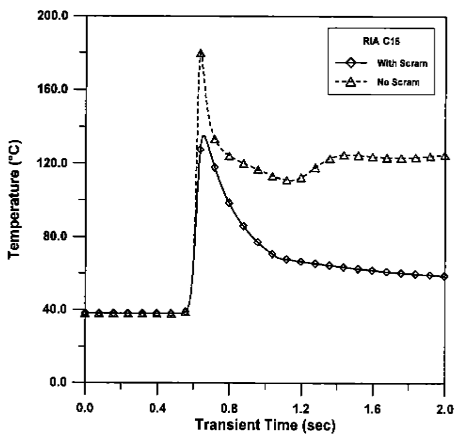

Fig. 2 : Cladding temperature behaviour during RIA transient with and without scram (C16 configuration)

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ANNEX B

IAEA-CN-156/

Safety Core Parameters Prediction in Research Reactors using Artificial Neural Networks: A Comparative Study of various Learning Algorithms

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In recent years, Artificial Neural Networks (ANNs) were applied successfully [1-2], as an advanced and promising tool for simulating several reactor physics parameters in nuclear engineering applications. The main objective in using such Artificial Intelligent (AI) methods, in the field of nuclear engineering, is to develop simple and first estimate models capable of simulating adequately, with reasonable error, important reactor physics parameters in relatively short time comparatively to time consuming and cumbersome reactor physics computer codes.

The feasibility of this application has been demonstrated through a previous work [3], done for a typical benchmark 10 Mw IAEA LEU (Low Enriched Uranium) core research reactor [4], using an adaptive learning rate procedure in a typical back-propagation algorithm in the training process. However, even tough the predictive results achieved are within ± 0.7% for K_{eff} and within ± 8.5% for P_{max} , the convergence time spent during the training phase were of about 36 and 24 hours, respectively for both cited parameters, on a small computational system (300 Mhz Pentium II PC).

Hence, this paper suggests one of the suitable ways explored to speed up the training process and to improve neural networks performances by carrying out a comprehensive sensitivity studies on an iterative and multistage calculation process using Neural Network MATLAB Toolbox [5].

In reality, it is very difficult to know beforehand what would be the suitable algorithm that can be used during the training phase for any given neural network problem and among the consulted literature there is no information about the precise concept to determine whether an algorithm is better for a specific application. In fact, it depends on several factors, i.e. the complexity of the problem to be treated, the optimal network architecture, the number of data in the training phase, the number of weights to be used in the network, selected error function and goals etc....

The methodology suggested was to find first optimal neural network architecture by applying a suitable transfer function in a typical feed-forward multilayered perceptrons (MLPs) network considered as the most commonly adopted for the predictive tasks. Once the optimal network architecture was set, the following computational step was dedicated to test several algorithms in the training phase in conjunction with the use of appropriate performance functions and adequate error goal and range of data to apply.

For this purpose, two types of algorithms have been assessed: the first one based on heuristic techniques i.e. Resilient Propagation (*RPROP*) and variable learning rate Back-Propagation (*GDA*), the second one is based on standard numerical optimization techniques i.e. Conjugate Gradient (*CGF, CGP, CGB and SCG*), quasi-Newton (*BFGS and OSS*) and Levenberg-Marquardt (*LM*). These fast training algorithms have been used in combination with four types of performance functions: Mean Square Error *(MSE)*, Sum Square Error *(SSE)*, Mean Absolute Error *(MAE)* and Mean Square Error with Regularization *(MSEREG)*.

For practical considerations, only part of the obtained results was given below. Figure 1, gives the mean square error versus execution time for several representative algorithms in predicting K_{eff} safety parameter during the training process.

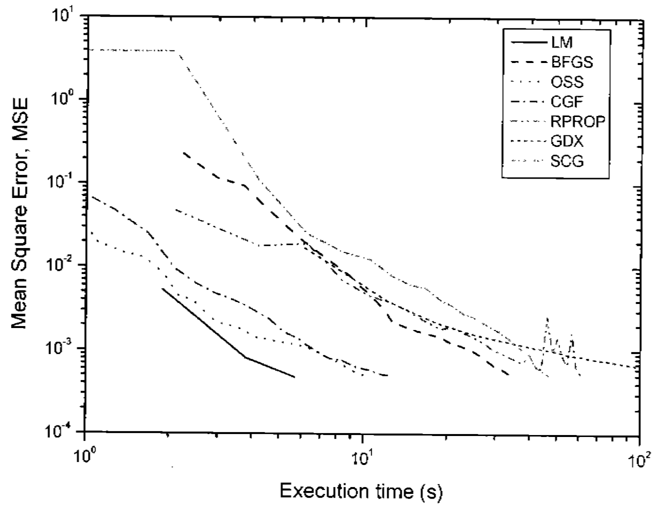

FIG. 1. Convergence speed of representative algorithms for K_{eff} predictions during the training process.

The performance of the various algorithms can be affected by the accuracy required for predictive purposes in the studied test problem. This is well demonstrated in the above shown figure. One can see that three algorithms *(LM, OSS and CGF)* converge well and fast in comparison to the others. The error in the Levenberg Marquardt *(LM)* algorithm decreases much more rapidly with time than the other algorithms shown.

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Determining Degradation of Sensors used for Fuel Assemblies Cooling Temperature measurement by the Analysis of Thermal Fluctuations Noise

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1.Abstract:
This work describes opportunities for continuous passive surveillance with noise analysis as the methodology, it is aimed to detect sensor degradation through response time factor calculation, to determine whether sensors have the response times within the plant safety analysis definitions, to ensure that the safety limits are not exceeded. The noise analysis techniques provide the capability for on-line monitoring of the response time of the installed sensors.

2. Introduction:

Reactor control is usually accomplished by the utilisation of sensors. The aim of these sensors is to reflect the exact behaviour of the physical process under surveillance; and to allow the user to ensure a well functioning of the plant under control.
The characteristics and properties of the sensors being used are critical factors whenever safety and high efficiency are required, especially when dealing with sophisticated plants such as nuclear plants. In fact, the achievement of these two requirements necessitates a continuous passive surveillance of sensors, however, this is traditionally accomplished by periodic tests perturbing the sensors and this direct sensor testing requires an access to all the sensors. The direct sensor testing, as it is shown in figure 1, is usually performed by applying a known signal to the sensor under test to determine the sensor response time as a performance testing factor. Drawbacks of this method are:
1 - It requires an access to each sensor; therefore it must be performed when the reactor is at a cold shutdown.
2 - It does not provide response time at normal operating conditions, so it does not yield the in-service response time of the sensor.
3 - Since sensors are continuously subject to degradations in their performances, it is required that the frequency of periodic tests must be increased, which is not practically and economically efficient. Therefore, the need for a passive on-line method for sensor performance testing is more than a requirement.
The noise analysis technique, which will be presented in detail in our work, provides the means for implementing a passive sensor surveillance, that yields the in-service response time, on the one hand, and that overcomes the drawbacks of the direct technique, on the other hand.

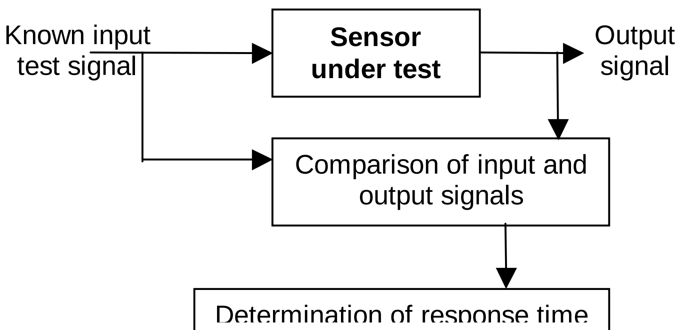

Figure 1: Traditional (direct sensor testing)

3. Noise analysis technique:

Sensor response time testing is performed in the reactor to determine whether sensors have the response times assumed in the plant safety analysis. Since response time degradations are known to occur, these tests must be performed periodically to ensure that safety limits are not exceeded. Furthermore, since the response time of some sensors depends on the process condition and installation, the tests must be performed at normal operating conditions to yield the in-service response time of the sensor.

The application of the noise analysis technique to response time estimation is shown in the block diagram of figure(2).

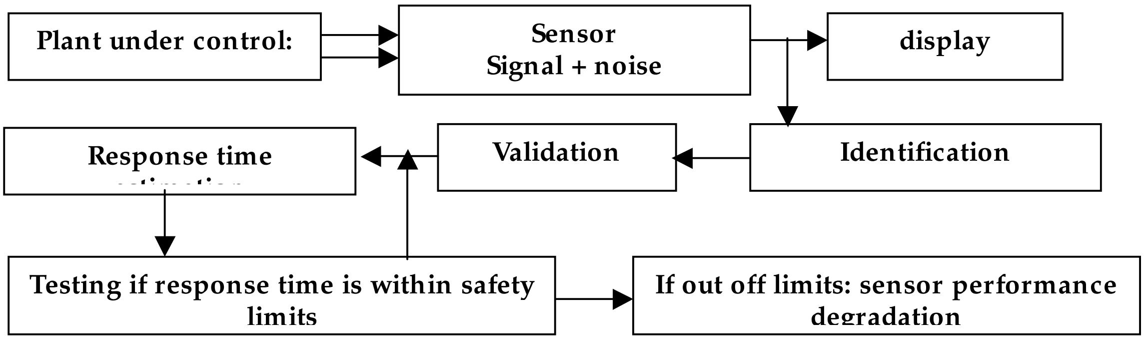

Figure 2: On-line sensor surveillance using noise analysis techniques.

4. Results:

A program was established and initial tests were performed to evaluate long term performance of resistance temperature detectors RTD of the type used in our research reactor to measure the difference in temperature in the fuel assemblies. The effort addressed the effect of aging on RTD calibration accuracy and response time. Sensors have been simulated by small random heat generators. For pressure transmitters, we used a hydraulic ramp generator in the laboratory. The in situation response time testing of FOXBORO force balance pressure sensors involves tricking the sensor, through the removal and reinitiating of power to the sensor, into the simulation of a pressure perturbation. The response transient of the sensor is measured, and the ramp response calculated. This transient is identical to one generated by a step change in pressure. This test can be completed in ten minutes from the control room with the reactor operating at full power.

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PHYSICAL CALCULATIONS FOR THE DESIGN OF IRRADIATION DEVICE AND NUCLEAR HEATING CALCULATION IN SILICON INGOTS AT ES SALAM RESEARCH REACTOR

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Abstract

To insure homogeneity of the neutron flux through Silicon irradiated at a Es-salam research reactor an irradiation device have to be conceived. The main criteria of its design are to fulfil all the necessary safety restrictions and recommendations while inserted into the core of the reactor. The excess of reactivity has to be also evaluated as well as nuclear heating in irradiated material.

This work consist on the evaluation of excess reactivity misleads by the insertion of the device in the reactor core. To do so, the reactor core has been divided into several homogeneous regions and a cartesian three dimensional XYZ model of the reactor core was performed using the three dimension diffusion code CITATION [1, 4]. The fewgroup library was generated by WIMS/D4[2,4].

The obtained excess reactivity of the irradiation device is negative, and with regards to neutronic consideration some positive reactivity is needed to make up for its insertion. This can be achieved through appropriate choice of the control rods insertions.

As for nuclear heating in the Silicon both neutron and photons contribution have been considered. The 1D transport code ANISN [3, 4] was used to perform the heating calculations. We have choice Pl= 3 order of scattering, S8 quadratic order and a distributed source in 207 energy group, 171 neutron group and 36 photon group to perform the calculations.

As it was expected photon heating is predominant on neutron part. This will allow us to neglect the neutron contribution to the heating and performing the design of the necessary cooling system.

In this paper, the adopted methodology is specified and the obtained results will allow us to decide on the cooling system and the rate of coolant to be fulfilled, also the necessary actions to maintain the reactor critical can be drawn.

Keywords: Silicon, Irradiation device, Excess reactivity, Nuclear heating.

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Upgrading I&C for the Es Salam Research Reactor

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The Es Salam research reactor was built at the nuclear research centre of Birine, 36 kilometers from Ain Oussera, Algeria. Es Salam, whose construction started in 1987, reached criticality for the first time on February 1992. The load was gradually increased until the reactor reaches its full power during performance tests in July 1992.

The Es Salam is a multi-purpose research reactor intended for the production of the radio elements, material tests, education and training. It serves also as a source for neutron beams used by chemists, biologists, metallurgists and physicists for fundamental research and applications. The Es Salam reactor is a tank type reactor of a nominal power of 15 MW(th) and of a neutron flux of 2.10^{14} (n/cm2-s). It is moderated and cooled by heavy water and has graphite reflector. Es Salam offers experimental possibilities in the fields of neutron spectrometry, production and materials studies.

The main task of the Es Salam Reactor Operation Division is to ensure a safe and reliable operation of the reactor. With this objective in hand, the Operation Division is responsible for updating the safety measures and conditions in the installation for the reactor equipments and systems. Due to the increased demand for experiences and to the ageing effects, modification and modernization of some safety items become necessary in comparison with state of the art installations. Furthermore, the technological advances and the development and introduction of new instruments, components and systems increased the need for modification. In this paper, we present the new I&C system which will be used for the Es Salam research reactor.

The Es Salam I&C System

At present, the Es Salam is still operated with the original I&C system which employs a technology from the 80's. This original, aged and analog, I&C system used discrete-component, solid-state devices and is no longer state-of-the-art and is in need of renovation and upgrade. A schematic diagram of the existing I&C system is shown in *FIG. 1 below.*

I &C upgrading

The existing analog I&C system becomes increasingly more difficult to maintain and to upgrade because of obsolete, ageing equipment and spare part procurement problems. It has been thought necessary to modernize and replace it with a digital control system which improves reliability, offers greater flexibility, increased data availability and enhanced features such as automatic self-testing and diagnostics.

The I&C modernization project for the Es Salam research reactor will be developed and implemented using distributed computer control with data communication networking possibilities. A modernization program is thus in preparation and will be launched soon.

The Replacement I&C system (*see FIG. 2 below*) includes the following equipment and systems: Digital reactor protective system, Digital power regulation system, Digital monitoring system, Fault diagnosis system, and Operator Station.

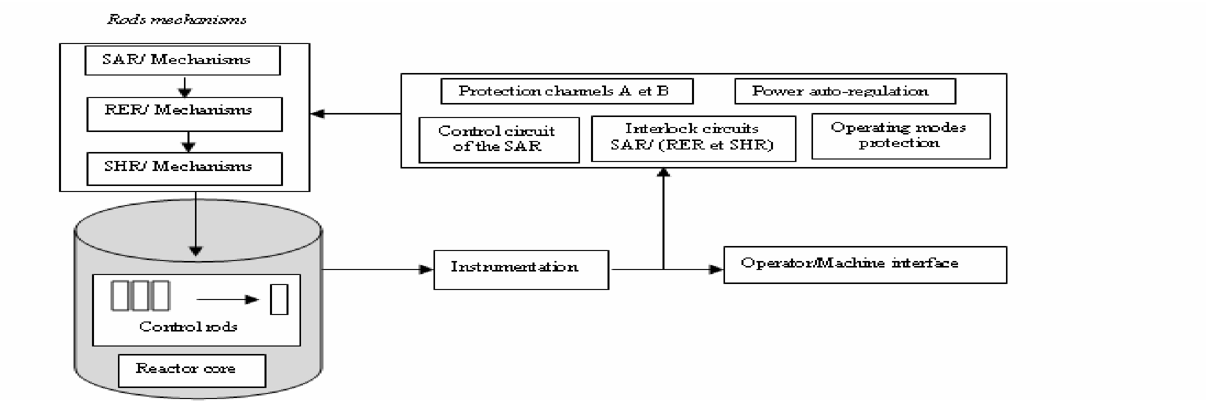

FIG. 1. Schematic Diagram of the Existing I&C System.

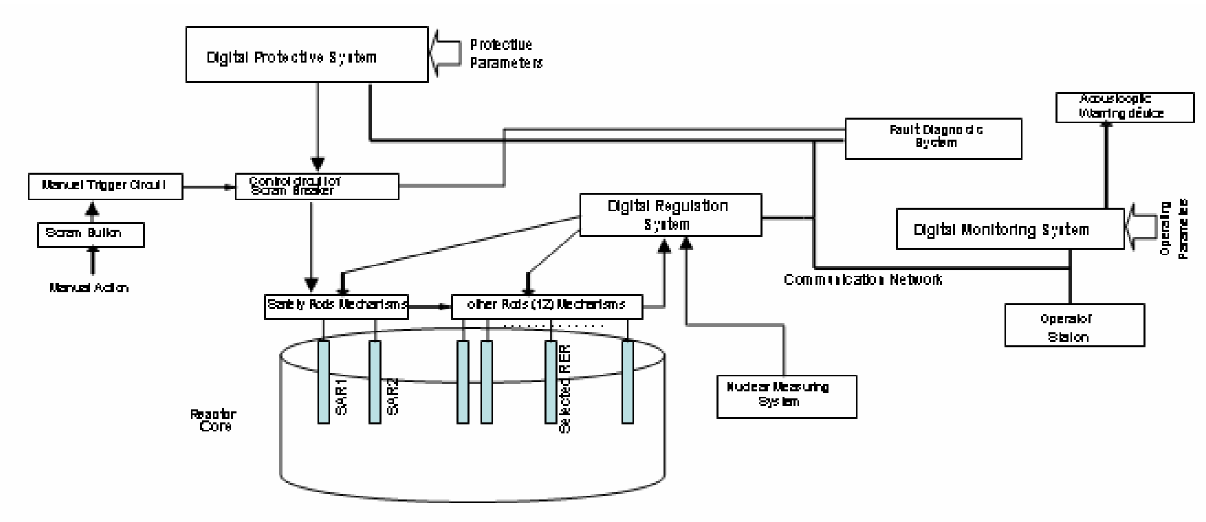

FIG. 2. General Layout of the New Digital I&C System.

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Training and Qualification of Reactor Operating Personnel

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Synopsis

INTRODUCTION
The training and qualification of OPAL Reactor Operations personnel is based on IAEA guidelines which recommend a Systematic approach to training. This involves performance and task based training which when used correctly establishes and maintains the competency and qualifications of staff. OPAL operator training is broadly grouped into classroom theory training, simulator training and practical plant training. The training is carefully structured to ensure that the selected candidate fulfils not only his personal ambitions but also the organisations needs. Our discussions will outline this approach

ROLES
OPAL Operations shifts consist of a Shift Manager, Reactor Operators and a Utilisation Operator.

The Shift Manager position is occupied by a professional engineer or scientist with extensive experience in the reactor operator role. The Reactor Operator and Utilisation Operator position is generally occupied by a technically competent person have a trade/post trade qualifications.

The responsibilities of each of these roles aims to achieve the required breadth of activities to enable safe and efficient operation of the reactor and utilisation of the irradiation facilities.

RECRUITMENT
The recruitment of Shift Managers and Reactor Operators for the commissioning and initial Operation of OPAL commenced in 2004, three years prior to the commencement of commissioning. The first Reactor Operator training course took place over 6 months in 2006, and full 24 hour shift manning commenced in July 2007, prior to fuel loading.

TRAINING PROGRAM
The training program is comprised of an appropriate combination of classroom instruction, simulator training, practical hands on sessions and supervised self-study. In addition to this, fundamental safety principles are also incorporated into the training to promote and improve safety culture and good work practices.

There are six basic components to the training which are carried out in a staged process:

1. Reactor Fundamentals
2. Classroom Design and Operations Theory
3. Simulator Training
4. Practical Training
5. Ongoing Training consisting of Safety Culture awareness, good work practices and facility update training
6. Shift Manager rotation

Shift Managers undergo additional detailed training in the areas of emergency response, safety analysis and application of Operating Limits and Conditions.

Training is assessed by written examinations, panel interviews, simulator assessment and completion of competency based checklists.

Training needs are identified in order to prepare training plans which are incorporated into the Quality Management System. The prime objective of training is to ensure the safe and efficient operation of the reactor.

Experimental Heat Transfer Analysis of the IPR-R1 TRIGA Reactor

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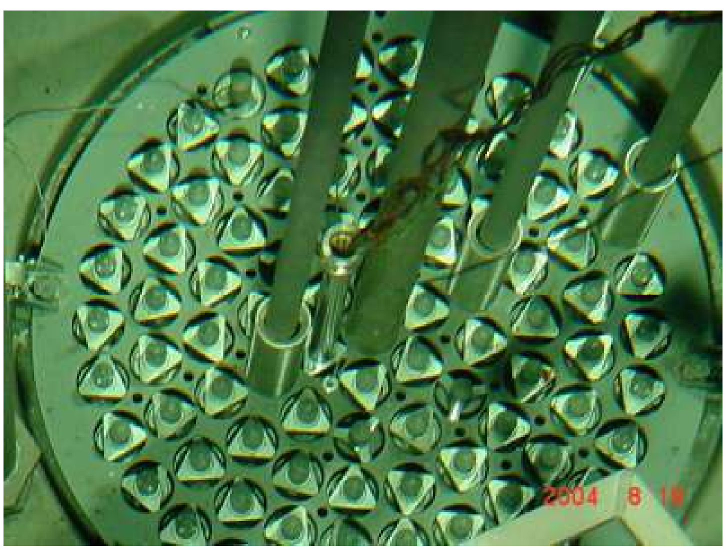
The heat generated by nuclear fission is transferred from fuel elements to the cooling system through the fuel-to-cladding gap and the cladding to coolant interfaces. The fuel thermal conductivity and the heat transfer coefficient from the cladding to the coolant were evaluated experimentally. A correlation for the gap conductance between the fuel and the cladding was also presented. As the reactor core power increases, the heat transfer regime from the fuel cladding to the coolant changes from single-phase natural convection to subcooled nucleate boiling. Results indicated that subcooled boiling occurs at the cladding surface in the central channels of the reactor core at power levels above approximately 60 kW.

The IPR-R1 TRIGA Nuclear Research Reactor (Fig. 1), installed at Nuclear Technology Development Center (CDTN), is a pool type reactor cooled by natural circulation, and having as fuel an alloy of zirconium hydride and uranium enriched at 20% in ^{235}U. The core contains 59 aluminum-clad fuel elements and 5 stainless steel-clad fuel elements. One of these steel-clad fuel elements is instrumented in the center with three thermocouples (Fig. 1).



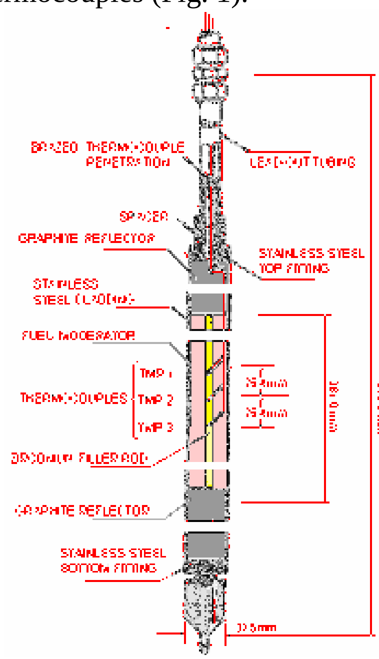


FIG 1. Core upper view with the instrumented fuel element in ring B and instrumented fuel element scheme

The objective of the thermal and hydrodynamic projects of the reactors is to remove the heat safely, without producing excessive temperature in the fuel elements. The regions of the reactor core where boiling occurs at various power levels can be determined from the heat transfer coefficient data.

The thermal conductivity (*k*) of the metallic alloys is mainly a function of temperature. In nuclear fuels, this relationship is more complicated because *k* also becomes a function of irradiation as a result of change in the chemical and physical composition (porosity changes due to temperature and fission products). Many factors affect the fuel thermal conductivity. The major factors are temperature, porosity, oxygen to metal atom ratio, PuO_2 content, pellet cracking, and burnup. The second largest resistance to heat conduction in the fuel rod is due to the gap. Several correlations exist [1] to evaluate its value in power reactors fuels, which use mainly uranium oxide. The only reference found to TRIGA reactors fuel is General Atomic [2] that recommends the use of three hypotheses for the heat transfer coefficient in the gap. The heat transfer coefficient (*h*) is a property not only of the system but also depends on the fluid properties. The determination of *h* is a complex process that depends on the thermal conductivity, density, viscosity, velocity, dimensions and specific heat. All these parameters are temperature-dependent and change when heat is being transferred from the heated wall to the fluid. An operational computer program and a data acquisition and signal processing system were developed as part of this research project [3] to allow on line monitoring of the operational parameters.

Subcooled pool boiling occurs above approximately 60 kW on the cladding surface in the central channels of the IPR-R1 TRIGA core. However, the high heat transfer coefficient due to subcooled boiling causes the cladding temperature be quite uniform along most of the active fuel rod region and do not increase very much with the reactor power. The IPR-R1 TRIGA Reactor normally operates in the range from 100 kW until a maximum of 250 kW. On these power levels the heat transfer regime between the clad surface and the coolant is subcooled nucleate boiling in the hottest fuel element. Boiling heat transfer is usually the most efficient heat transfer pattern in nuclear reactors core [4]. Another important aspect of the reactor operation safety is that it is far from the occurrence of critical heat flux [5].

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Appraisal for the Operation Safety of SPRR-300 in Recent Years

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SPRR-300 reached first criticality in 1979. During its 27 years of safe operations, the reactor has never been renovated comprehensively. So the condition of the devices and the reactor must be analyzed to appraise the safety of the operation.

A management networks for the safety and the quality of the reactor operation were built within the operation department. A general outline for quality of SPRR-300 during operation stage went into effect in 2004. A set of files have been written or rewritten to ensure that the outline is operated smoothly. The rules and regulations of the reactor are insisted on. The emergency scheme and its execution programs had been established around 2002. Emergency practices were held in 2003, 2004, and 2006. Through this series of practices, the abilities for emergency treatment of the staff have been boosted continually.

The minimalist technical transformation of the reactor had only been done once around the year of 2002. The failure ratio of the reactor goes down year by year. Before 2002, 22 out of the 31 scrams occurred without even one warning signal showed at the same time. Since the minimalist technical transformation finished, that kind of scram has not occurred any more. The reactor resumed normal operation in 2003. In this year, there were 27 scrams, in which nine were caused by staff faults. 18 scrams in 2003 were cause by devices breakdowns, while most of them were originated from malfunctions of the old sets of power protection instruments. There were 14 scrams in 2004 and 2005. Five of them were caused by staff faults. More than half of the nine scrams caused by devices breakdowns were related to the electricity net lapse.

It is reasonable that there are small quantities of manual failure. Every time before and after operation, the group of technical management would organize some kind of meeting to discuss the problem that may appear or have appeared. With the continued fostering of safety culture, the operation quality of the staff is enhanced constantly. To those old reactors like SPRR-300, devices are the key of the reactor safety although some suitable modifications have already been done. In this period, usually some new kinds of device failure would turn up. So it is a must that the quality of the staff should be enhanced steadily to solve every accident correctly and resolutely.

There were 12 cases in 2003 that the power supplies for ionization chambers gave alarms of lower voltage because the voltage of local power grid was too low. The reactor was removed of automatic operation afterwards because of malfunction of the power regulate system. To solve this kind of hidden peril, the institute managing the reactor began to work together with local power supply bureau to stabilize the voltage during every period of operation. Also in 2003, the second regulate rod system worked abnormal two times. Its lifting speed was much lower than that of normal. In the first case, the reason was that the contact of its speed-adjust electric resistance was poor. In another case, the reason was short-circuit between the turns of the stator coil of the servo motor. Once in 2004 the drop-fuse of one transformer outside the reactor building struck arc. The reactor was forced to be shut down temporarily so the transformer could be disconnected to be repaired later.

The bad installation of the second motor for the ventilation of the coolant tower was one of the inherent defects in SPRR-300. The retarder lapsed eventually. Thorough repair of the retarder had been accomplished in 2005.

As a result, the most of threats to the normal operation of the reactor come from instability of power grids outside the reactor building. Because the inherent safety of the reactor and the developing of the ability for accident treatment within the reactor staff, instability of power grids can only influences the usages of the reactor to a certain extent. It will not produce an effect on the safety of the reactor.

Through maintaining and repairing in accordance with certain cases apart from overhauling at regular intervals for the devices, all systems of the reactor are kept in good condition, which meets the demand to guarantee the safety of the reactor. So long as the regulations are abided by, the training of the staff is strengthened over and over, and the safety culture is fostered continually, the safety and the quality of the operation can be ensured.

Calculations and Measurements at the Training Reactor VR-1

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The Paper presents basic information about verifying calculations of the experimental measurements at theVR-1 reactor and their evaluation. The training reactor VR-1 has been in operation since 1990. The VR-1 reactor is a pool-type light-water reactor based on low enriched uranium fuel IRT-4M with maximum thermal power $1kW_{th}$. The reactor is successfully used for training the students of Czech universities and preparing the experts for the Czech nuclear programme. The reactor VR-1 can be used for basic research, when respecting small power of the reactor.

The calculation preparations and verifications of the experiments are integral part of the VR-1 reactor operation. The neutronics calculations and analyses are very often necessary condition to gain the authorization to experiments realization at the reactor VR-1.

Deterministic codes WIMS and DIFER were used formerly for the neutronics calculations of the VR-1 reactor at the Department of Nuclear Reactor. Since 1998 neutronic calculations are performed by the statistical code MCNP. The Department of Nuclear Reactors currently works with the MCNP version 5. For the last nine years the MCNP model of the training reactor VR-1 has been permanently adjusted and improved. Now the MCNP model of the VR-1 reactor contains all important parts. The main reactor components in the model are:

- fuel assemblies IRT-4M (4-tube, 6-tube and 8-tube assemblies),
- 6-tube assemblies with control rods,
- fuel dummies,
- vertical and horizontal channels,
- external neutron source AmBe type,
- core grid,
- beryllium and graphite reflectors.

MCNP model of typical VR-1 reactor core configuration is shown on Fig. 1.

The main part of MCNP calculations is focused on different parameters and characteristics of the new core configurations at the VR-1 reactor, e.g. control rods worth calculation, determination of reactivity excess and calculations of neutron flux distribution. In Table 1 calculated and measured values of control rod worth are given. A lot of MCNP simulations are focused on verification of neutron activation analysis experiments or determination of delayed neutrons parameters.

The Department of Nuclear Reactors made detailed criticality calculations for core configurations with LEU fuel using MCNP code. These calculations were the necessary condition to gain the authorization to operating the training reactor VR-1 with the LEU fuel IRT-4M. The department of Nuclear Rectors has already performed calculating verifications of 15 core configurations at the VR-1 reactor. The calculated results prove good agreement with experimental results.

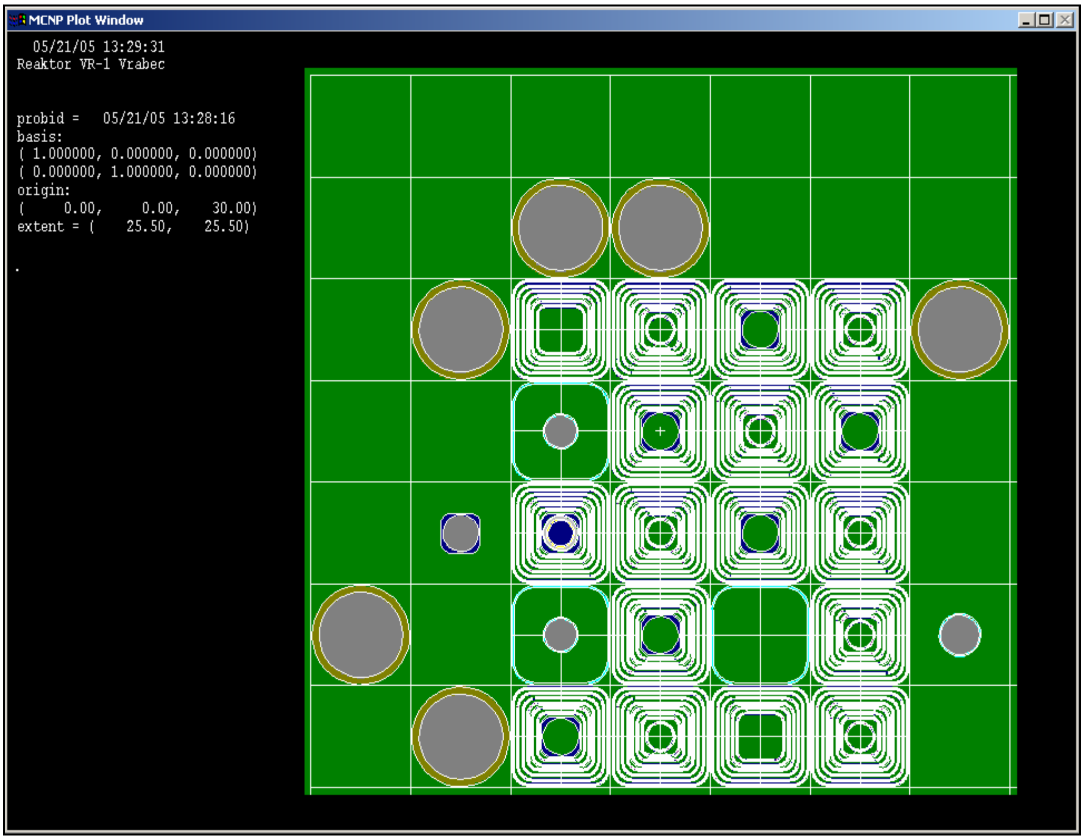

Fig. 1 MCNP graphical output - model of the VR-1 reactor core

control rod	MCNP		Experiment	
	worth [%]	deviation	worth [%]	deviation
B1	1.50	0.0848	1.37	0.0967
B2	2.11	0.0861	1.89	0.0606
B3	1.64	0.0878	1.46	0.0789
E1	1.22	0.0811	1.00	0.0482
E2	0.96	0.0901	0.93	0.0642
R1	0.46	0.0888	0.42	0.0601
R2	0.84	0.0849	0.90	0.0838

Table 1 Calculated and measured control rods worth

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Improvements of the Training Reactor VR-1 – Only Way how to be Attractive for Students and Scientists

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The VR-1 Reactor is operated for training of university students and nuclear power plant personnel, R&D, and information services for non-military nuclear energy use. During last four years a large number of improvements has been achieved. The paper describes four of them which are the most important for the operation from the safety-point of view.

1. Conversion of the reactor and operation with LEU fuel

During the autumn of 2005 VR-1 reactor has been successfully converted from the Russian IRT-3M fuel containing HEU (enrichment 36 % ^{235}U) to the Russian IRT-4M fuel containing LEU (enrichment: 19.7 % ^{235}U). This task was completed within the scope of the RERTR programme, that was initiated by the US Department of Energy, consistent with the global non-proliferation policy goal of minimizing the use of highly-enriched uranium in civil programmes worldwide. The fuel swap involved a direct cooperation of DOE and the RERTR programme, a NZCHK (Russian fuel supplier), IAEA, and a SOSNY (Russian company, which was repatriating the HEU fuel IRT-3M) [3].

During the first critical experiment with LEU fuel and the testing regime all criticality and neutronics characteristics of the LEU core were measured; hence the necessary conditions to license the core were fulfilled. Then the other reactor's operational characteristics were measured, such as thermal neutron flux at different positions in the core or reflector. Also the reactor's dynamics was studied, namely responses to positive, periodic and negative reactivity changes [4]. All the results confirm theoretical calculations when was expected that the reactor would behave equally with the new LEU fuel practically in a same way as with the HEU fuel [1]. The measurements confirmed the difference, that was expected in the calculations, of a lower neutron flux density with the LEU fuel compared with the HEU fuel. Training and teaching was already started during the testing regime for the undergraduate and graduate students. All standards experiments were implemented and the effects of the new LEU fuel were closely watched [1].

2. Upgrade of the control and safety system

The reactor was put into operation in the 1990. The original reactor I&C seemed to be obsolete and their replacement (upgrade) was being carried out. The upgrade is being done gradually during holidays in order not to disturb the reactor utilization during teaching and training. The new I&C substantially improves the reactor safety, the comfort of the reactor operation, and facilitates the maintenance. The I&C upgrades started in 2001 [2] with the human-machine interface, continued in 2002 with the control rod motors, drives and safety circuits upgrade. The control system upgrade followed in 2003 [2].

The next upgrade stage was the independent power protection system upgrade in 2004/05. The last stage is going to be the upgrade of the operational power measuring system in autumn 2007. The upgrades bring the reactor I&C to the top conditions and enable a prolongation of its functionality and maintainability for at least next 10 years. The new protection system consists of the operational power measuring (OPM) and the independent power protection (IPP) system [5]. The new OPM system is a full power range system that receives signals from fission chambers. The signals are evaluated according to the reactor power either in the pulse or current mode. The current mode utilizes DC current and Campbell techniques. The new IPP system working in the two highest power range decades receives signals from boron chambers and evaluates them in the pulse mode. The computers of both systems calculate the reactor power and power rate, compare them with the safety limits, and if they are exceeded, the safety action is initiated. There are 4 independent channels (3 in active mode) in each system and evaluated in the 2 out of 3 logic. The OPM and IPP systems are diverse different types and location of chambers, completely different hardware, software algorithms, development tools (PharLap based tools and Keil µVision 2 system) and teems for software manufacturing. Quality assurance, configuration management, verification and validation accompanied the software development [5].

3. Upgrade of the radiation monitoring system

The original radiation monitoring system STADOS was developed in the mid-80s. The new monitoring system RMS was implemented at the reactor in 2004. New RMS system is fully computerised state-of-art system connected with reactor information system. RMS system consists of 2 neutron and 10 gamma detectors, aerosol detector, external gamma detector located at the roof, control unit, archive system, user friendly LCD touch screen for control with all necessary information and connection with reactor console.

4. Upgrade of the physical protection system

After the reactor conversion the nuclear materials category was decreased to the no. 3, but the physical protection system corresponding to the category no. 2 has not been changed. The new comfort and more reliable physical protection system has been installed at the reactor in 2006. The system respects all changes at the reactor hall during I&C upgrade and allows to protect the reactor hall at higher level using new technologies in the alarm communication and display between the physical protection system, reactor staff and respond forces.

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Study of temperature effects on ULYSSE reactor for training and qualification of operating personnel

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Part of the French Atomic Energy Commission, the National Institute for Nuclear Science and Technology (INSTN) provides both academic and professional courses in all disciplines related to nuclear energy applications, including the physics and the operation of nuclear reactors.

Theoretical courses and training courses on simulators are completed by experimental work on a training reactor which ensure a practical and comprehensive understanding of the reactor operation. For the training and qualification of reactor personnel and regulators, this global approach that includes experimental work contribute to the improvement of the safety of the reactor operation.

We report here experiments conducted in this framework on the ULYSSE training reactor which is a 100 kW Argonaute type reactor. The main experiment developed here concerns the study of the temperature effects and the determination of temperature coefficients.

ULYSSE reactor is a water moderated and cooled, graphite reflected system, using enriched MTR fuel elements. The core has an internal diameter of 90 cm. 24 fuel elements form an annular crown around a graphite reflector. Thus, both water and graphite play a important role in temperature effects. The global temperature coefficient α_{global} corresponds mainly to the sum of the water α_{water} and the graphite $\alpha_{graphite}$ temperature coefficients.

The experiment is carried out by maintaining the reactor critical at low power and adjusting the position of the regulation rod while the temperature of water is intentionally changed through the use of the different elements that constitute the primary and secondary water circuits.

During the experiment, the water temperature in the core and the position of the regulation rod are measured as a function of time. Figure 1 shows the evolution of the rod position and of the water temperature during the experiment.

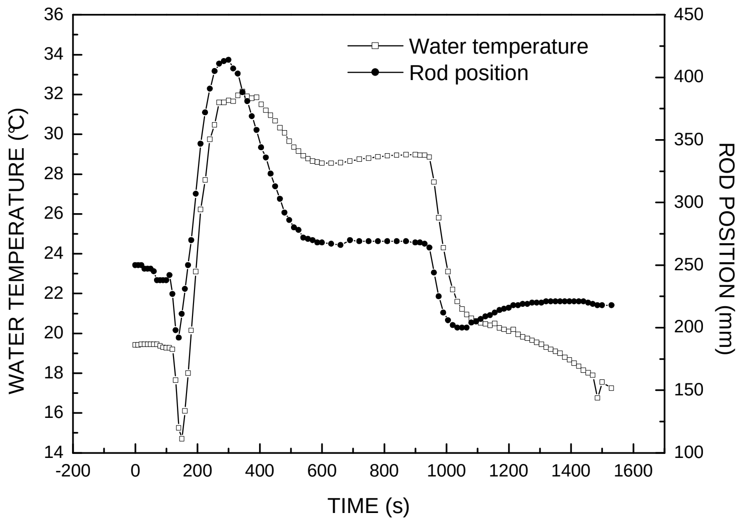
Using the calibration curve of the regulation rod, the evolution of core reactivity as a function of time is determined. The core reactivity varies as the inverse of the rod position. The evolution of graphite temperature is extrapolated from the water temperature measurements, showing evidence of the thermal inertia of graphite. The analysis of the experimental results, give an insight on the temperature effects.

At first, one can observe that the core reactivity varies as the inverse of water temperature, i.e. $\alpha < 0$. An increase of the reactor power and thus an increase of the water temperature produces a decrease of the core reactivity, which in turn produces a decrease of the water temperature. This shows that, by conception, the reactor is sub moderated and thus autoregulates when the power is increased, participating to the safe regulation of the reactor.

Using the recorded data, one can then successively calculate the global, the water and the graphite temperature coefficients, i.e about - 2 , - 10 and + 8 pcm/°C, respectively, showing that α can take either positive or negative values depending on the physical effect inducing the temperature effect. The negative value of α in water is related to dilatation effect that reduces neutron moderation, while the positive value in graphite is related to the decrease of the capture cross section of graphite σ with the neutron kinetic energy ($\sigma \approx 1/T^{1/2}$).

The experiment also shows that the apparent temperature coefficient that is "lived" by the operator who regulates the reactor is not constant through the time. Indeed, when the water temperature rapidly change, while the graphite temperature is constant due to its inertia, the apparent α is close to α_{water}, i.e - 10 pcm/℃. This contribute to a rapid change in the core reactivity with time that needs a rapid variation of the control rod position. But, when heat transfer contribute to an equilibrium in water and graphite temperature, then the apparent α is close to $\alpha_{global,}$ i.e - 2 pcm/℃. Thus, what is important, is not only the value of the temperature coefficients, but also the physical effects to which they are related and the kinetics of this effects.

From the understanding of these temperature effects, empathise will then be given on there impact on the safety of reactor operation, particularly in the occurrence of an accidental situation.



Monte Carlo Simulation of GRR-1 Core and Neutron Irradiation Positions Using MCNP

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Prediction of neutron flux at the irradiation devices of a research reactor facility is essential for the design and evaluation of experiments involving material irradiations. The Monte Carlo technique offers significant advantages, since the complex geometrical configuration of the reactor core and irradiation devices can be modeled in detail.

A model of the Greek Research Reactor (GRR-1) core configuration was developed using the Monte Carlo code MCNP with continuous energy neutron cross-section data evaluations from ENDF/B-VI library [1]. The model includes detailed geometrical representation of the fuel and control assemblies, beryllium reflectors, graphite pile and irradiation devices [2]. In this work, the MCNP model of GRR-1 core was applied in order to predict the neutron field characteristics at

(i) a special in-core irradiation element at core lattice position D-4
(ii) irradiation devices used for neutron activation analysis and radioisotope production
(iii) the graphite thermal neutron column assembly

The MCNP estimated neutron flux was compared with measurements using gold foils with and without cadmium covers to obtain the thermal and epithermal neutron flux in the irradiation positions. A good agreement between calculated and experimental results was observed.

Since experimental determination of the flux characteristics at the reactor irradiation devices is time and labor consuming, the developed model is a useful tool in predicting neutron flux in samples for purposes of designing and evaluating experiments involving material irradiations. Moreover, it was proven particularly useful in the time period of gradual conversion of GRR-1 core from HEU to LEU fuel since it enabled estimation of the neutron field characteristics at the irradiation devices under the actual mixed core configuration conditions.

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Radiological characterization of the GRR-1 pool

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GRR-1 is a 5MW open pool type research reactor with MTR-type fuel elements cooled and moderated by light water with beryllium reflectors at the two opposing sides of the core. A graphite thermal neutron column is adjusted to one side of the core. Six radial horizontal beam tubes are available, of which three contain in-pile collimators for neutron scattering instruments. The reactor is currently out of operation for inspection and refurbishment purposes. The core has been dismantled and the fuel elements are stored in the used fuel storage tank.

The GRR-1 inspection and refurbishment plan involves inspection and eventually replacement of the reactor's primary cooling circuit. The health physics procedures to be implemented during inspection of the main water outlet are divided in three stages: a) pool dose rate survey from pool top, b) pool drainage by decreasing water level in steps and c) inspection of the water main outlet.

Purpose of the present work is the evaluation of the gamma radiation fields inside and around the pool during the above procedures. The Monte Carlo neutron and photon transport computer code MCNP5 was employed in order to asses the activation of materials in the pool and predict gamma radiation levels during pool drainage. The code enabled simulation of the reactor core, pool and activated components within the pool i.e. experimental tubes with in pile collimators, beryllium blocks, reactivity control rods, lead shields and irradiation rigs. The actual photon source in the reactor pool depends upon alloy, impurity content, neutron spectrum, reactor operation history, and decay time. The MCNP predicted gamma dose rate results were calibrated against experimental measurements performed using a submersible GM detector.

The results of this study were used to derive the gamma dose rate profile inside and around the reactor pool. The dose rate profile data will be used in order to set the radiation protection requirements, such as implementation of certain procedures or manufacture and placement of special shielding, during the inspection and refurbishment operations.

The Next 20 Years Operation of the 36 Years Old Hungarian Training Reactor

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The four units of the Paks NPP generate 38% of the Hungarian electricity production, which properly indicates the stressed importance of nuclear energy in Hungary. One decade prior to the construction of the Paks NPP, in the early 60’s, the country had begun preparations for domesticating nuclear technology and the preliminary steps of construction of the training reactor had been taken. The construction finished in 1971 at the Technical University of Budapest, first criticality was reached in May, 1971.
The training reactor is a pool-type reactor with 100 kW nominal power, operating with Russian-made EK-10 fuel assemblies of uranium enriched to 10%, and based on Hungarian reactor design. The training reactor has been successfully serving Hungarian nuclear expert education, and technical education of physicists, teachers and engineers in the last 36 years.

Hungary prepares for extending the design lifetime of the four VVER-440/213 type units; in that case they will finish operation between 2032 and 2037. Discussion on possible new nuclear units in Hungary was recently commenced.

Lifetime extension and the construction of new units require continuous expert supply. For this purpose the technical possibility of further operation of the training reactor provides a good base, but numerous renewals are to be performed and the generation-change of the operating and educational personnel of the training reactor is needed.

The colleagues participated in the start-up of the reactor and assured the safe operation of the reactor during the last 36 years, are now retired or preparing to retire. Their replacement requires a special program, which has been planned and was implemented in the last 3 years. In the most important operating positions the number of employees has been doubled in 2 years, so that the young colleagues can work together with the experienced ones day-by-day. This ensures the most effective transfer of special operational and maintenance knowledge. Up to know the young experts have successfully obtained their licenses for reactor operation, and they are able to operate the reactor alone.

For a more effective technical knowledge transfer a modern, computer-based 3D CAD model has been developed based on the 40-45 years old drawings. This has been proved to be effective from more viewpoints:

- differences between the old drawings and the implementation have been revealed;
- the young employees could learn reactor construction in a more effective manner through the problems arisen during the development of the 3D CAD model;
- there are up-to-date drawings, which are may continuously be updated according future modifications of the reactor.

A long-term technical development plan was also outlined to ensure the extended operation of the reactor.

In the last 3 years the following main actions were made at the training reactor:

- Renovation of one part of the ventilation system;
- Renewal of the pneumatic rabbit system;
- Restoration of the classroom in the reactor building;
- Application of state-of-the-art 3D reactorphysical and thermohydraulical simulations in the safety analysis;
- Building of a new document archive for old and new safety reports, procedures, manuals, research reports and books;
- Development and use of a modern intranet site serving as general information source for the reactor operators and the collective of the institute.
- Actions for the conservation of collective knowledge.

The capacity utilization of the training reactor is quite high due to the different education programs. The reactor participates in the education of engineering-physicists, energetic, mechanical, electrical and chemical engineers. There are about 2 to 3 thousands of secondary school students visiting the reactor annually, which helps the teachers in the basic nuclear education.

We are participating also in the ENEN (European Nuclear Education Network Association), aiming at the integration of the European nuclear education. The first and most successful course of the ENEN is the *Eugene Wigner Course for Reactor Physics Experiments*, the main emphasis of which is to perform reactor physics experiments to enhance the knowledge of the students in nuclear engineering and reactor safety. The course is based on 3 research- and training reactors in three different countries (Vienna – Austria, Prague – Czech Republic, Budapest – Hungary). The 21 days course was started in 2003. In the last 4 years 58 participants from 12 countries accomplished the course. The participants are mainly MSc students, but PhD students and young experts can be found among them as well.

In 2006 and 2007 the Periodic Safety Review (PSR) was carried out for the training reactor. Based on this PSR the Hungarian nuclear safety authority is to issue the operating license for further 10 years. During the PSR the safety of the reactor has been re-evaluated, a new 3D neutron kinetics code has been developed, and modern Computational Fluid Dynamics (CFD) methods have been applied for the safety analysis of the special EK-10 fuel (aluminium-coated, filled with magnesium and uranium-oxide mixture) for LOCA and RIA transients.

The paper will describe among others the above mentioned actions in human resource management and knowledge management, and also the new safety analysis methods which were applied during the recent Periodic Safety Review of the Hungarian Training Reactor.

IAEA-CN-156/S-15

Validation of the Monte Carlo Method of MVP Code on the First Criticality of Indonesian Multipurpose Reactor[1)]

Tagor Malem Sembiring[2)], Surian Pinem[2)] and Setiyanto[2,3)]

Extended abstract

The validation research works in BATAN are focused using Monte Carlo method with recent nuclear data on the experimental results. In this paper, the validation results of Monte Carlo method of MVP code on the first criticality experimental of Indonesia Multipurpose Reactor (RSG GAS reactor) are presented. The MVP code is a vectorized and continuous energy Monte Carlo code developed by Japan Atomic Energy Agency (JAEA). The objective this paper is to show the accuracy of the code using recent nuclear data of JEF-3.0, JENDL-3.3 and ENDF/B-VI.8. The final goal of this research is to use the code as a in-core fuel management code since the code has a module of burn-up calculation (MVP-BURN).

RSG GAS reactor is a beryllium (Be)-reflected, light-water-moderated and -cooled, 30 MWth (max.) multipurpose reactor. Presently, the reactor uses MTR-type LEU (19.75 w/o) silicide fuel (U_3Si_2-Al) elements (FEs). On the 10 x 10 core grid positions there are 40 standard FEs (each consisting of 21 fuel plates), eight control elements (CEs, each consisting of 15 fuel plates) initially loaded with 250 and 178.6 g ^{235}U respectively, Beryllium reflector elements and other irradiation facilities. This fuel loading corresponds to a uranium meat density of 2.96 g/cm^3.

The equilibrium core is achieved through some transition cores with smaller core and lower power. In the first transition core, all FEs and CEs were fresh oxide fuel with same uranium meat density of 2.96 g/cm^3. Some experiments were carried out in the core, including criticality experiments, as a part of commissioning activity. Since the first core using the fresh fuels, so the core can be used a bench mark core to validate the accuracy of a selected code. In the criticality experiments, there were some types of experiment have been performed as follows:

- first criticality by adding FEs and CEs
- criticality condition by adding the Beryllium reflector elements
- criticality condition by adjustment various positions of six control rods
- excess reactivity and total control rod worth

In this paper, those criticality experimental results are compared to the calculated results by using the MVP code. The calculated results showed that the calculated results

using the selected nuclear data are very close to the experimental results. For example, the calculated core excess reactivity using JENDL-3.3 is in the range of 8.17 %Δk/k – 8.35 %Δk/k. The calculated result is very close to the experimental result of 8.41 %Δk/k.

It can be concluded that the MVP code with JENDL-3.3 and ENDF/B-VI.8 nuclear data can be applied for the MTR type reactor with bulky Beryllium reflector.

1) Submitted to International Conference on Research Reactors: Safe Management and Effective Utilization, Sydney, Australia, 5-9 November 2007
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Neutronic Analysis of RSG GAS Silicide Core with Uranium Density of 4,8 g/cm^3 [1)]

Setiyanto[2,3)], Surian Pinem[2)], T.M. Sembiring[2)] and Lily Suparlina[2)]

Extended Abstract

Fuel conversion program of the RSG GAS multipurpose reactor is to convert the fuel from oxide, U_3O_8-Al to silicide, U_3Si_2-Al. The aim of the program is to gain longer operation cycle by having, which is technically possible for silicide fuel, a higher density. Since 1999, a core conversion program of the RSG GAS reactor from oxide to silicide fuel with the same fuel density of 2.96 g U/cm^3 has been started and at the end of year 2002 all silicide core was achieved. However the conversion study of using higher uranium density fuel in the RSG GAS reactor is still on-going. The study is focused on silicide fuel with density of 3.55 gU/cm^3, 4.8 gU/cm^3 and 5.2 gU/cm^3.

Previous research showed that 3.55 g/cm^3 uranium density can extend the operation cycle length from 615 MWD to 900 MWD significantly without any material and core configuration change. This paper presents an advanced research on the 4.8 g /cm^3 uranium density utilization on RSG-GAS core. This research focused on the neutronic aspect, including reactivity balance, power peaking factor and kinetic parameters. The calculations were performed using computer codes WIMS/D4, Batan-2/3 DIFF and Batn-EQUIL-2D.

During the design, the following constraints are used:

1. No major modification to the reactor balance of plant, shielding, core main structural components, and civil buildings may be made.
2. The number as well as the performance of irradiation positions and facilities must be maintained.
3. The existing one-stuck-rod reactivity margin and thermal-hydraulic safety requirements must be fulfilled.
4. Maximum discharge burn-up is limited to 70 % for licensing purposes.

The calculations showed that the reactor cycle length can be extended until 1400 MWD with the maximum radial power peaking factor 1.31 less that the limit value of 1.4. However, the two safety rods should be added to higher the shutdown margin (one stuck rod criteria). The reactivity balance and the in-core fuel management of the core are also presented in this paper.

For kinetic parameter, the calculated delayed neutron fraction, delayed neutron decay

constant and average neutron life time are 7.03256×10^{-3}, $7.8520 \times 10^{-2}\ s^{-1}$ and 55.49 μs respectively. Those parameters are not change, significantly, compared with the existing core.

It is concluded that the proposed equilibrium silicide of 4.8 gU/cm^3 core configuration could be applied in RSG-GAS operation, safely.

1) Submitted to International Conference on Research Reactors: Safe Management and Effective Utilization, Sydney, Australia, 5-9 November 2007

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Fault Position Estimation Using Color Segmentation Approach to the Thermal Infrared Image of a Cabling Network in the Nuclear Reactor

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Currently, predictive maintenance has become an indispensable part of many plants maintenance planning strategies which contribute to the ageing management of a nuclear reactor. It uses projected data or trends from condition monitoring techniques to determine the trouble-free service life of equipment, thereby eliminating unscheduled shutdowns. These techniques monitor deterioration, processing conditions, and specific events that precede the development of equipment failures.

One of the predictive maintenance techniques utilizes thermal infrared imagers. This technique detects hot spots in electrical equipment in the plants by scanning the object such as substations, distribution lighting panels, or electrical motors. Problems are **easily and quickly eliminated** before they cause system failure. **Thermal infrared imagers** are detector and lens combinations that give a visual representation of infrared energy emitted by all objects.

Fault position in a system is estimated in this research using color segmentation approach to a thermal infrared image of a cabling network which can be implemented for ageing management in a nuclear reactor. Image segmentation is a process to convert the real image into some certain area black and white[1]. Implementation of color segmentation for fault diagnosis in the nuclear reactor cabling network is conducted in this research utilizing image from thermal infrared imagers. In this case, the operator should determine the color in infrared image which indicate the faulty. Image input is first converted to RGB (*Red, Green, Blue*) image model and then converted to CMYK (*Cyan, Magenta, Yellow, Key for Black*) image model. Assume that the yellow color in the image represented the faulty, the CMYK image model is then diagnosed using color segmentation model to estimate the fault position in a cabling network. The system diagram block is represented in the Figure 1.

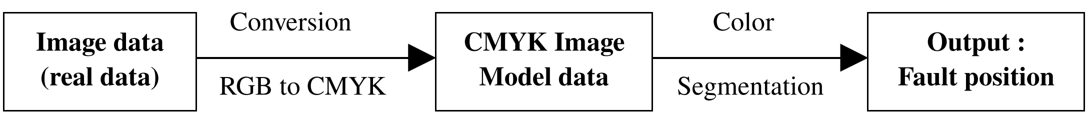


Figure 1. Diagram Block of the Fault Diagnosis System

If k_o is output value, and the input value is represented by chromatization value k_i, then the color segmentation represented in the equation (1) would become binary image which has 2 value as such 1 (white) if the chromatization value greater than threshold (τ), and 0 (black) if the chromatization value is same or less than the threshold (τ),

$$k_o \begin{cases} = 1 & \text{if} \quad k_i > \tau \\ = 0 & \text{if} \quad k_i \leq \tau \end{cases} \qquad (1)$$

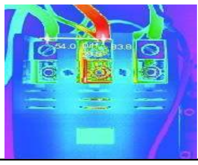
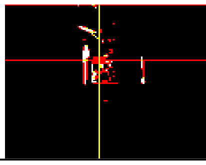
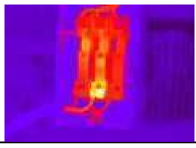
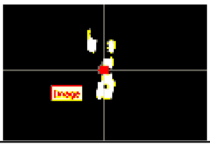
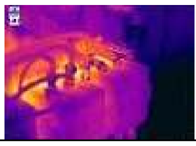
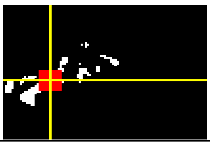
Based on the segmented binary image then the fault position can be estimated using equation (2) as follows [2] :

(13)

$$\bar{x} = \frac{\sum_{x=i}^{m} \sum_{y=1}^{n} xO(x,y)}{A} \quad \text{and} \quad \bar{y} = \frac{\sum_{x=i}^{m} \sum_{y=1}^{n} yO(x,y)}{A} \qquad (2)$$

Input data image, output image, fault location center in cartesian coordinate and fault indication judgement are provided in the Tabel 1.

Table 1. Fault Diagnosis

Image number	Input Image	Output Image	Fault Location (x,y)	Fault Indication
1			(83,75)	occured
2			(114,83)	occured
3			(24,42)	occured

It can be concluded from experimental results provided in the Table 1 that the color segmentation approach from input image capable to estimate the location of faulty center in the whole picture of three sheets image acquired, if assumed that the fault is represented by yellow color in the input image.

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WATER CHEMISTRY SURVEILLANCE FOR MULTI PURPOSE REACTOR 30MW GA SIWABESSY, INDONESIA

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There are three research reactors in Indonesia, all operated by BATAN. The major and the largest one is the 30 MW Multi Purpose Research Reactor (MPR) G.A. Siwabessy located in Serpong, about 30 km from Jakarta. It is an open-pool type reactor cooled and moderated by light water and fueled by 19.75% enriched U_3O_8Al/U_3Si_2Al yielding averaged neutron flux of $1x10^{14}$ n/cm^2 s. The reactor construction began in 1983 and completed in July 1987 when the reactor reached its first critically, however, its full power of 30 MW was reached in March 1992. The reactor cooling system comprises primary and secondary systems. The primary coolant is subjected to gamma and neutron irradiation by the core, producing radiolysis species, such as oxidator O_2 and H_2O_2 which induce oxidation reactions of dissolving substances within them. This causes corrosion reactions in the reactor tank components and other undesirable effects like forming compound deposits inside the cooling system, in particular, inside the heat exchanger cores, thereby adversely affecting its performance and life-span. As such, it is imperative to monitor, control and maintain the primary water chemistry to the required specifications at all times in order to ensure longevity and performance of the reactor system. The secondary cooling water, on the other hand, is not subjected to irradiation, and its water chemistry is aimed at suppressing corrosion and algae growth.

A special surveillance program has been prepared and put into practice to study the material behaviour when subjected to the current water chemistry regime and to acquire experimental data upon which BATAN can effectively improve the primary water chemistry.

The main objectives of the surveillance program are to:

(i) establish standard procedures for corrosion monitoring and surveillance; and
(ii) provide technical guidelines for continued wet storage of spent fuels.

Experiments have started with clear understanding of the current water chemistry profile, followed by the manufacture of a series of appropriate coupons to study the effect of the primary water on homogeny, galvanic and crevice corrosion. The test coupons were made of the same grade of materials as used for aluminium fuel cladding and reactor tank, and stainless steel fuel storage racks. As shown in Figure 1, the coupons are made of a series of discs (ϕ95 mm with ϕ15 mm centre hole; 1 mm thick for stainless steel and 5.5 mm thick for aluminium) put together as an assembly using a stainless steel centre pin and insulated with a ceramic bushing and 5 mm thick spacers between the discs. The coupons are also assembled in two orientations to allow vertical and horizontal installations in the pool (Fig. 2). The first batch of six coupon assemblies were strategically positioned in the service pool in January 2007 and are due to be withdrawn in stages for inspection after 1, 2, and 3 years of exposure.

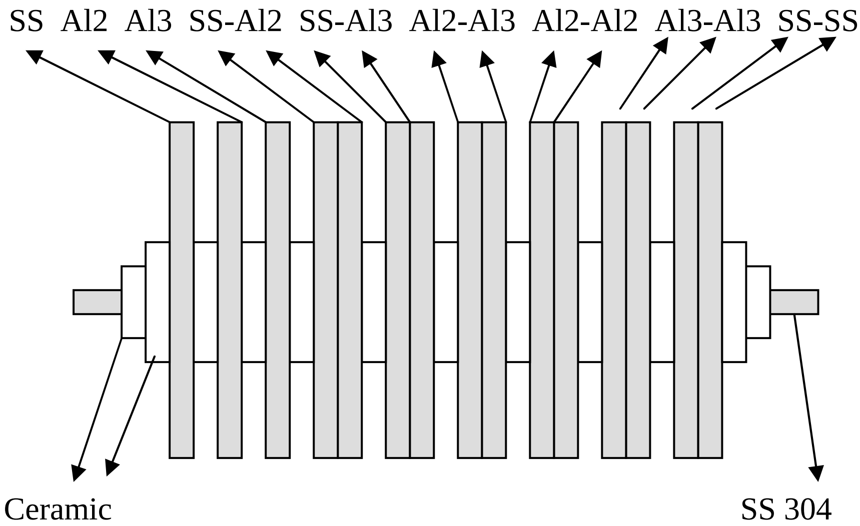

Figure 1. Coupon Assembly.(SS=SS 304, Al2=Aluminum for fuel cladding, Al3=Aluminum for tank)

(a) Vertical

(b) Horizontal

Figure 2. Coupon Orientations

[1] IAEA, TRS 418, 'Corrosion of Research Reactor Aluminum Clad Spent Fuel in Water'.

[2] IAEA TECDOC 927, 'Influence of Water Chemistry on Fuel Cladding Behaviour'.

[3] Diyah Erlina Lestari, Geni Rina Sunaryo, Yusi Eko Yulianto, Sentot Alibasyah H. and Setyo Budi Utomo, 'G.A. Siwabessy Research Reactor Water Chemistry', Module B2, Water Chemistry Course I, Serpong, Indonesia, 2004.

IAEA-CN-156/U-51

ASSESSMENT OF QUALIFICATION OF RSG-GAS REACTOR OPERATOR IN INDONESIA

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Abstract

The reactor operator and supervisor who operate the reactors have important tasks to ensure the safe operation of the facilities. For that reason, personnel who are in charge to operate the reactors, both operators and supervisors, shall possess working license issued by Bapeten according to the Act No. 10/1997.

To implement the Act No. 10/1997 especially relating to the personnel who operates a nuclear reactor in Indonesia, Bapeten has enacted the guidance which is required by operator/supervisor to fulfill the requirements obtaining license to nuclear installation and ionization radiation utilization installation, such as Government Regulation No. 64/2000 and Decree of Bapeten Chairman No. 17/1999.

In this paper, the objective of the assessment of the operators and supervisors of RSG-GAS Reactor is performed to identify that operating organizations (in that case, for Center for Multi Purpose Reactor, the operating organization has managed the RSG-GAS Reactor) has implemented the Bapeten provisions and requirements. This paper describes the qualification for operators and supervisors of RSG-GAS Reactor and the assessment of the operators and supervisors of RSG-GAS Reactor.

1. INTRODUCTION

Indonesia has three research reactors i.e. RSG-GAS Reactor at Serpong, TRIGA 2000 Reactor at Bandung and Kartini Reactor at Yogyakarta. For operating nuclear reactor, it is needed some reactor personnel. Nuclear installation operating organizations shall ensure that the designated reactor personnel fulfilling the requirements to operate nuclear reactor according to the Act No. 10 Year 1997 Article 19 clause (1). It stated that any employee who operates a nuclear reactor and other nuclear installation and in the installations that use ionizing radiation sources shall possess a license.[1].

2. REGULATION

The training, qualification and working license are based on the following regulations:

- The Act No. 10 year 1997 on Nuclear Energy
- Government Regulation No. 64 Year 2000 on Licensing of Nuclear Energy Utilization

- Government Regulation No. 43 Year 2006 on Licensing of Nuclear Reactor
- Decree of Bapeten Chairman No.06 /Ka-BAPETEN/V-99 on The Construction and Operation of Nuclear Reactor
- Decree of Bapeten Chairman No. 17/Ka-Bapeten/IX-99 on The requirement to Obtain a Working License for Personnel in Nuclear Installation And Radiation Installations
- Decree of Bapeten Chairman No. 04P/Ka-Bapeten/I-03 on The Guidance on the Training of Nuclear Reactor Operators and supervisors.

According to Chapter 3 of Government Regulation No. 64 Year 2000, the operating organization shall have qualified personnel for nuclear energy utilization [2], and the personnel operating or control the reactor operation or personnel related to safety shall obtain working license issued Bapeten before the personnel performs the duty. [2, 3, 4].

In Decree of Bapeten Chairman No.06 /Ka-BAPETEN/V-99 Article 10 is governed that [4]:

1. The expert and the personnel who act as a nuclear reactor operator shall have a working license from Bapeten.
2. The working license is issued by Bapeten after the personnel has passed the exam.
3. The working license is issued for some period and to be renewed.

According to Decree of Bapeten Chairman No.06 /Ka-BAPETEN/V-99 Article 24 clause 2.c., the operating organization shall establish the training programme for reactor personnel, and in Decree of Bapeten Chairman No. 17/Ka-Bapeten/IX-99 in Chapter 5 is governed that any reactor should be operated by some capable and skill personnel, at least consisting of [5]:

a. one operator
b. one supervisor
c. one radiation protection officer
d. one maintenance personnel

Article 6 stipulates that the personnel shall have the training and pass the examination to proof her/his qualifications. The Decree of Bapeten Chairman concludes that reactor personnel shall perform the training in the accreditation organization/institution. He/she has carried out an examination by Bapeten for a working license.

3. TRAINING AND QUALIFICATION

A. Preliminary Training Programme (Qualification Programme)

Preliminary training programme shall provide the understanding on the basic principles of nuclear technology, nuclear safety, radiation protection, reactor design and reactor utilizations, etc. for application in field.

That training for operators and supervisors shall contain an adequate theoretical and practical knowledge on function, location and operation mode of reactor, and also shall emphasize to the importance of limiting and condition of operation, and the consequences of violation the limit, and the safety consequences of fault procedures.

B. Programme of Qualification Examination

After passed the training examination, the candidate of operator and supervisor shall be examined and evaluated in the final training before examination by BAPETEN for a working license suitable for his professionalism.

The method of evaluation by Bapeten consists of:

1. Written examination
2. Oral examination (can be done in the class or as a part of practical examination in facility), and or
3. Practical examination on normal and emergency operation condition.

C. Requalification Programme

Requalification programme is based on the systematic approximation to ensure that the knowledge, skill and attitude of operating personnel can be preserved. If necessary his skill and knowledge can be increased. Any personnel having task in the field of reactor safety shall perform this programme. The programme is carried out every two years. The requalification programme includes the selected topics from preliminary training, supporting important tasks for reactor safety operation, infrequent and difficult task. The programme shall also include periodical training and case studies in emergency conditions and events by operating organization.

D. Programme of requalification Examination

Operators and supervisors which will renew the working licensing shall take part in programme of requalification examination. The programme of requalification examination is conducted by Bapeten.
The method of evaluation on the requalification examination by Bapeten is the same with the method of evaluation of the qualification examination.

E. Promotion

The promotion of their occupations is used to be provided in all facilities. When the operator is assured to be capable of a reactor supervisor, his/her experiences, leadership and communication capability shall be taken into account. He/she as a new professional worker shall be trained to increase his/her knowledge and skill.

4. THE REQUIREMENTS OF QUALIFICATION EXAMINATION

The requirements of operator and supervisor examination are following :

a. The operator shall be graduated at least from scientific or engineering senior high school. The supervisor shall be graduated at least from Bachelor of science or engineering. He/she should have two year experiences in nuclear field.

b. In general, the operator and supervisor shall have good health and physical condition. Therefore it is assumed that there is no unanticipated failure or fault resulting in the hazard for the health and safety publics, for example: epilepsy, psyche, diabetes mellitus, hypertension, heart, unconsciousness, ear or eye disability and other mental and physical condition.

5. APPLICATION OF WORKING LICENSE

The licensing applicant shall fill the application form and enclose:

i. Documents of applicant abilities on the methods of safe reactor operating. For supervisor, the applicant shall understand his/her job and responsibilities. These documents include a certificate or recommendation from the examination team. The certificate or recommendation consists of : the detail course, course hours, start-up and shut down experiences.

ii. Physician recommendation from his/her medical check up.

6. WORKING LICENSE

Bapeten will give a working license for a participant passing the exam. The working license is one of significant documents presented capabilities as a reactor operator or supervisor. The working licenses has been valid for two years.

The expired license will be come into effect, if the licensee renews his/her license. The renewal application must be enclosed as following:

i. Documents on applicant experiences including total hour.

ii. A letter of recommendation from operating organization stated that the applicant has implemented his/her working very well.

iii. Medical check up results.

iv. A certificate of requalification training.

The duration of working license can be renewed if:

a. The operator or supervisor has good health and physical condition so that there is no unanticipated failure or fault resulting in the hazard on the health and safety publics.

b. The operator/supervisor has performed his/her task actively or and he/she is able to continue his tasks.

c. He/she has passed the examination arranged by Bapeten. For participant not passing this exam, he/she may take the examination at another occasion.

Reactor operator or supervisor license can be revoked if :

a. The operator or supervisor has not a good health and physical condition. The requirements in section 4.b. has not yet been fulfilled.

b. His/her fault can cause any accident increasing the radiation hazards and contamination for personnel, member of publics and environment.

7. ASSESSMENT OF QUALIFICATION FOR RSG-GAS REACTOR OPERATOR AND SUPERVISOR

A. Method of Assessment

The assessment of RSG-GAS operators' and supervisors' qualification has performed by comparing the RSG-GAS operators' and supervisors' licensing documents with the DCB No. 17/ 1999 and DCB No. 04P/2003.

B. Result and analysis/assessment

RSG-GAS reactor is research reactor in Indonesia, and has potential radiation hazard, therefore to operate it, it is needed some qualified operators and supervisors. The requirements of the operator and supervisor license has been established by Decree of Bapeten Chairman No. 17/Ka-Bapeten/IX-99. For identifying whether the RSG-GAS operator and supervisor have fulfilled the qualification in accordance with the provisions issued by Bapeten, it is needed to assess the qualifications of RSG-GAS reactor operators and supervisors.

The background education requirements to be a reactor operator or supervisor is according to the examination requirements of sec. 5.a. In this case, RSG-GAS reactor operators and supervisors graduated at least from scientific or engineering senior high schools and Bachelors of science or engineering, and having experience in nuclear field minimum of 10 years.

The requalification for operator and supervisor will renew the working license. The training requalification programme is performed periodically by Center for Multi Purpose Reactor (CMPR) for minimum 2 years.

According to sec. 5.b. the requirement of physician and health condition, CMPR is to perform medical check up for the operator and supervisor candidates. The operator and supervisor may renew the working license.

In table 1 can be shown the results of assessment of the qualification for operator and supervisor to fulfill the provisions/requirements issued by Bapeten.

TABLE 1: ASSESSMENT RESULT OF THE QUALIFICATION RSG-GAS REACTOR OPERATOR AND SUPERVISOR

No.	Provision/Requirement	Regulation	Assessment Result
1.	a. The operator shall be graduated at least from scientific or engineering senior high school and experienced in nuclear filed minimum 2 years b. The supervisor shall be graduated at least from Bachelor of science or engineering. He/she should have two year experiences in nuclear field	DBC No. 17/1999 Annex II. 2.1	a. Education for operators : generally scientific or engineering senior high school : 22 persons with experience between 2- 14 years b. Supervisor education: scientific or engineering senior high school : 6 persons and Bachelor of science or engineering 4 persons with experience more than 10 years.
2.	having good health and physical condition. Therefore it is assumed that there is no unanticipated failure or fault resulting in the hazard for the health and safety publics, for example: epilepsy, psyche, diabetes mellitus, hypertension, heart, unconsciousness, ear or eye disability and other mental and physical condition.	DBC No. 17/1999 Annex II.2.1.	The physician and health condition of RSG-GAS reactor operators and supervisors are good condition, is proved by the medical check up reports.
3.	Attending the preliminary training programme for the operator and supervisor candidates and the requalification programme for the operator and supervisor which will renew the working licensing programmed by operating organization.	DBC No. 17/1999 article 6 DCB No. 04P/03	Operators and supervisors have followed the qualification (preliminary programme) and requalification programmed by CMPR, is proved by training certificates.
4.	Performing the qualification and requalification examination arranged by Bapeten.	DBC No. 17/1999 Article 6, 10, Annex II.2.1. and 5.1 DCB No. 04P/03	Passing the examination, proving by the working license issued by Bapeten.

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8. CONCLUSION

Reactor personnel operating nuclear reactor shall fulfill the requirements in regulation, because they will determine whether the reactor operation is safe or not. One of the requirements is working license. This license will be submitted after the personnel has attended the training by the accreditation training organization/institution and has passed the examination by Bapeten.

From Table 1 can be concluded that RSG-GAS reactor operating organization –CMPR has complied the Bapeten provisions for arranging the qualification and requalification training programme for the operators and supervisors. The RSG-GAS reactor operators and supervisors have followed the qualification and requalification trainings and examinations.

9. REFERENCES

[1] Act No 10 Year 1997 on Nuclear Energy, Jakarta (1997).
[2] Government Regulation No. 64 Year 2000 on Nuclear Energy Utilization Licensing, Jakarta(2000).
[3] Government Regulation No. 43 Year 2006 on The Licensing of Nuclear Reactor, Jakarta (2006).
[4] Decree of Bapeten Chairman No.06 /Ka-BAPETEN/V-99 on The Construction and Operation of Nuclear Reactor, Jakarta (1999).
[5] Decree of Bapeten Chairman No. 17/Ka-Bapeten/IX-99 on The requirement to Obtain a Working License for Personnel in Nuclear Installation And Radiation Installations, Jakarta (1999).
[6] Decree of Bapeten Chairman No. 04P/Ka-Bapeten/I-03 on Guidance on training of nuclear reactor operator and supervisor, Jakarta (1999).
[7] Licensing Document of RSG-GAS Reactor Operators and Supervisors.

Developing an Ultrasonic NDE System for a Research Reactor Tank

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Ultrasonic testing is one of the established tools for routine in-service inspection of reactor tanks [1]. As part of the preventive maintenance of the IRR2 reactor, an ultrasonic scanning system was developed for the inspection of the reactor tank wall. Here, we present the main features of the special equipment developed for this task. In addition, we describe the procedure used for validating the inspection method.
The inspection will be done from inside the tank, using the immersion technique [2], when the tank is empty of fuel and control rods but filled with coolant/ moderator water.
The construction of the mechanical apparatus that was designed for the task is sketched in figure 1a. The ultrasonic scan-head is carried by a custom-designed manipulator that is inserted in the center of the tank. The manipulator has two motion axes that are the elevation Z and the rotation Φ.

The scanning will be done in three stages:

1. A preliminary, low resolution (1 mm X 4 mm), scan of the whole tank wall using two 5 MHz straight transducers and six pairs of 2.25 MHz angular transducers operated in both Pulse-Echo and the Pitch-Catch modes [2]. The aim of this stage is to detect areas in the tank that may contain flaws.
2. A high resolution (1 mm X 1 mm) scanning of suspect areas, detected in stage 1, using the same scan-head as in stage1.
3. A local scan for the evaluation of suspect- flaws that may be detected in stages 1 and 2. This stage is performed using a 10 MHz straight transducer and 3 pairs of 2.25 – 10 MHz angular transducers. The scan-head used has an additional degree of freedom ω.

In order to enable the thorough evaluation of the results, all the acquired A-scan signals are saved on a 2 TByte RAID system. A dedicated software package was developed for the processing of this huge amount of data.
The validation of the design was carried out in two stages. First, the ultrasonic technique was calibrated on specimens with known flaws, using a laboratory-scale ultrasonic scanner. In the second stage, the whole inspection system was tested in a full-sized mock-up. In this mock-up installation, critical procedures such as the insertion and the folding of the manipulator arm were practiced. Also the mock-up tank wall contained flaws that were unknown to the inspectors. In figure 1b, a sample C-scan of an area containing such validation-flaws is shown.

Conclusions
A special apparatus was developed for the ultrasonic scanning of a research reactor tank wall, the operation of which was practiced using a full-scale mock-up. The inspection technique was validated using a variety of flaws that were unknown to the operators.

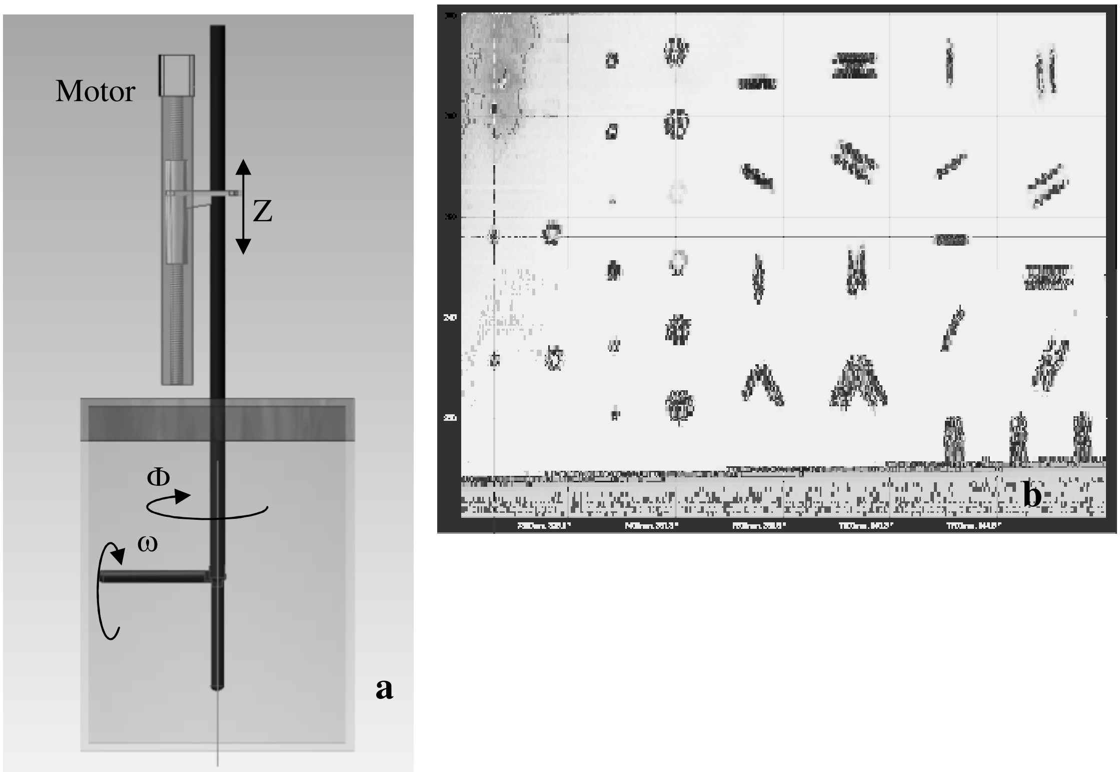

FIG. 1. a) The special manipulator designed for scanning the reactor tank. The manipulator is inserted vertically through a central hole and folded to scanning position. b) A C-scan of a defect-containing area in the mock-up.

[1] ASME Boiler and Pressure Vessel Code, Section XI.
[2] ASM Handbook, Vol. 17 Nondestructive Evaluation and Quality Control Section: Methods of Nondestructive Evaluation, Ultrasonic Inspection

Neural Networks application to CRDM and thermo-hydraulic data validation

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The activity is aimed at validating data obtained by reactor measurements through soft-computing models based on neural networks (NN).
Application is made on the CRDM rod position and the fuel temperature which are calculated by NN-based algorithms and compared with the real data coming from sensors. The bias is used to train the NNs for improving the performance in the next experiments.

General Description of the Reactor

The reactor is a typical light-water reactor with an annular graphite reflector cooled by natural convection, with a power of 1MW.

Reactor Core

The core is placed at the bottom of the 6.25-m-high open tank with 2-m diameter. The core has a cylindrical configuration.
In total there are 91 locations in the core, which can be filled either by fuel elements or other components like control rods, a neutron source, irradiation channels, etc. The core lattice has an annular but not periodic structure.

Reactor instrumentation

Core temperature is measured by 20 thermocouple situated above and under the core (see Fig.1).
Fuel temperature is measured in two fuel elements instrumented with thermocouples.

Neural Network application

In this first application we try to validate the position of the control rod. To do this we use a neural network that elaborates the measured data on the thermo-fluid-dynamic system of the reactor, in parallel with the control system.
The used net is feed-forward 3-layers network trained using second-order algorithms (Levenberg-Marquart).
It is constituted from two nets, Fig.1 that work in parallel: the first used for the validation of the data (trained initially through the use of reactor data set); the second one used for the training that update its parameters every time occur an error given from the difference between the exact value of the first net and the measured value from the instruments (validated from the operator).
The new parameters of training will come therefore transferred to the first net.

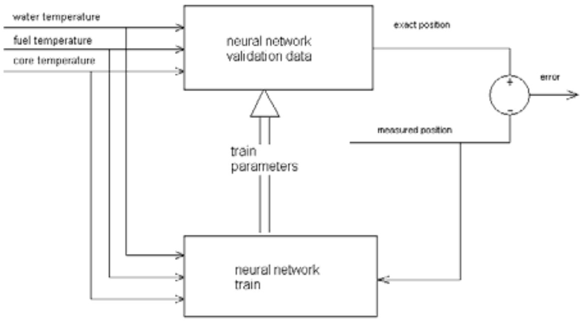

Fig.1 Validation network

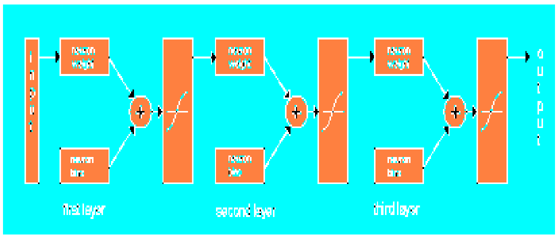

Fig.2 Neural network

The net's geometry is the following: input layer 35 neurons; hidden layer 22 neurons; output layer 2 neurons Fig.2.

Input data: core temperature (20 thermocouple); fuel temperature; water temperature.

Output data: rod position (2 channels).

The outcome of training is displayed below

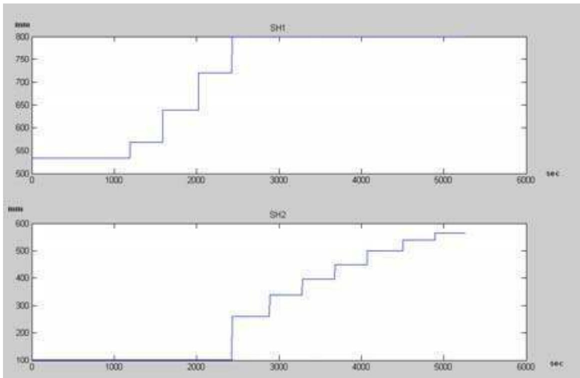

Fig.3 Rod position channels

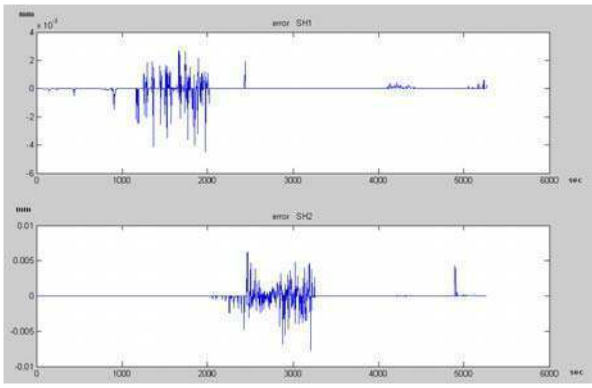

Fig.4 Rod position error

In this second application we wont validate the value of the fuel temperature. To do this we use a neural network trained with the measured power and fuel temperature.

The used net is feed-forward 4-layers network trained using first-order algorithms (steepest descent).

The outcome of training is displayed below

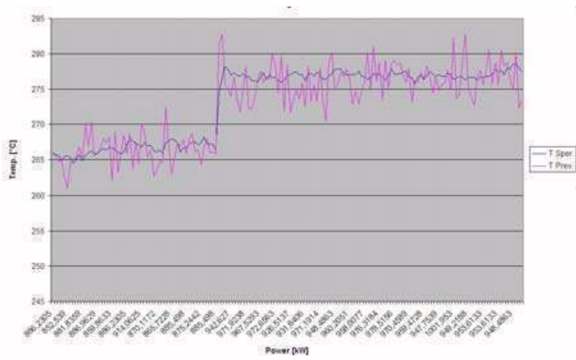

Fig.5 Experimental data (blue),
Simulated data (pink).

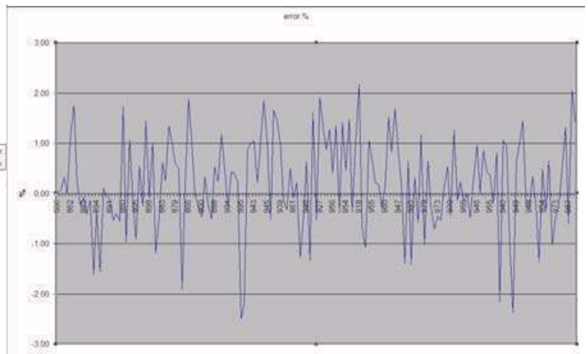

Fig.6 Percentage error

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Investigation of JRR-3 control rod worth changed with burn up of follower fuel elements

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JRR-3 (Japan Research Reactor No.3) is a swimming pool type research reactor with thermal power of 20MW. The core is composed of 26 standard fuel elements, 6 control rods (Sa-1, Sa-2, S-1, S-2, R-1, R-2), beryllium reflectors, and vertical irradiation holes as shown in Fig.1. Control rod is composed of follower type fuel and hafnium neutron absorber. Length of an operation cycle is 26 days, and seven operation cycles are conducted for a year.

The control rod worth (CR worth) has been measured in an annual maintenance period by inverse kinetics method (IK method). The CR worth is used as important and basic data for management of reactor operation, such as prediction of control rod position at start-up and estimation of an irradiation sample reactivity and excess reactivity changed with burn-up of fuel elements. However, because the control rod worth would be affected by the change of burn up of fuel, the reliability of the measured CR worth would be degraded when the burn-up would be different. In particular, the difference in the CR worth would be enlarged before and after the fuel exchange. For these reasons, it must be necessary to grasp the CR worth precisely to operate the reactor safely.

Table 1 shows the CR worth measured in 7 years. It is shown that the maximum difference in the total worth is about 3.1%Δk/k and would be caused by the difference in burn-up. In order to evaluate the difference, the CR worth in different burn-ups are calculated by simulating the measurement with IK method, and compared with the measured one. This report shows the results of evaluation and also proposes a convenient method to estimate the CR worth with higher reliability. Whole core calculation is carried out by the Monte Carlo Code MCNP5.

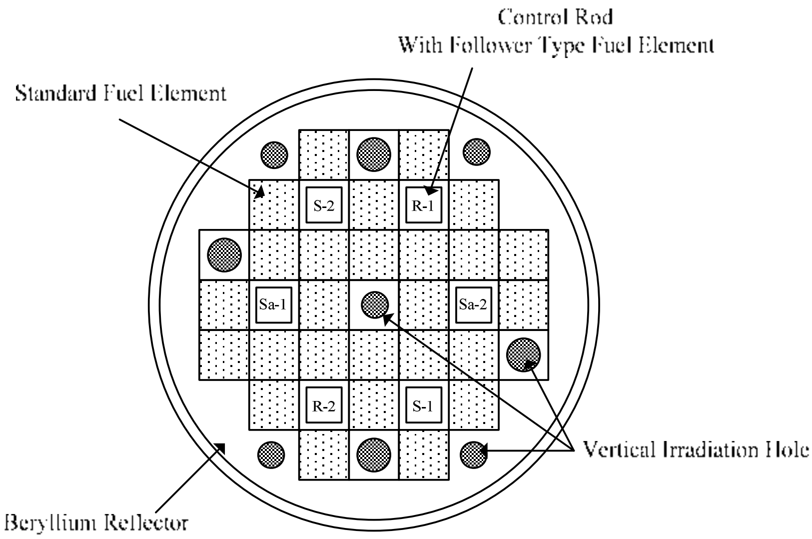

Fig.1 Horizontal Cross Section of JRR-3 Core

Table 1 Control Rod Worth of JRR-3 Measured with IK method

Fiscal Year \ Control Rod	R-1	R-2	S-1	S-2	Sa-1	Sa-2	Total
2000	3.949	3.942	4.140	4.237	4.818	5.017	26.103
2001	3.344	3.263	3.393	3.553	4.665	4.768	22.986
2002	3.562	3.552	3.752	3.764	4.219	4.319	23.168
2003	3.490	3.500	3.858	3.806	4.327	4.196	23.177
2004	3.755	3.947	3.752	3.663	4.029	4.127	23.273
2005	3.728	3.671	3.768	3.941	4.371	4.490	23.969
2006	3.890	3.792	3.820	3.950	4.242	4.319	24.013

(Unit: %Δk/k)

[1] M. Takeuchi, et al., “The development of the Measurement Technique of the Control Rod Worth Source”: JAERI-Tech 2000-054 (2000)

[2] X-5 Monte Carlo Team, “MCNP — A General Monte Carlo N-Particle Transport Code, Version 5” LA-CP-03-0245 (2003)

Project to Replace the Control and Protection System at the WWR-K Research Reactor

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The WWR-K is a tank-type, light water moderated and cooled multipurpose research reactor with a nominal power rating of 6 MW and a peak thermal neutron flux of over 10^{14} n/cm^{2}s. The reactor is operated by the Institute of Nuclear Physics of the National Nuclear Center of the Republic of Kazakhstan (INP), and is located at Alatau, near Almaty, the largest city in Kazakhstan. The reactor was commissioned in 1967 with a power rating of 10 MW, then shut down between 1988 and 1998 while seismic safety upgrades were completed. Currently, INP is undertaking the planning and technical work to allow the reactor to be converted from HEU fuel with a U-235 content of 36% to LEU fuel with a U-235 content of 19.7%. The reactor has an important role in Kazakhstan, and is producing isotopes for medical and industrial uses, providing material testing services and neutron activation analysis.

Replacement of reactor control and protection system (CPS) is a part of the wider program of the reactor modifications related to its conversion from HEU to LEU fuel, supported by the US Department of Energy, the Nuclear Threat Initiative and the Kazakhstan government. The program includes development of a new LEU fuel assembly design and re-configuration of the reactor core, with change in number and positions of the control rods, which requires modification of the CPS. Furthermore, replacement of the existing instrumentation, some of which is 40 years old, will improve reactor safety, bring the CPS up to current international standards, and provide an upgraded control interface.

The existing CPS at the WWR-K relies to a large extent on old designs of detectors and electronic equipment that are no longer in production and the system as a whole does not meet the necessary standards for prolonged operation of the reactor. Consequently, the Kazakhstan Atomic Energy Committee (KAEC), the nuclear regulatory body of Kazakhstan has mandated that the project for conversion of the WWR-K to use LEU fuel must include replacement of the CPS. By using the support services available from the IAEA for procurement of the replacement CPS, INP will ensure that the new system meets current safety standards.

Organizationally, the project will take full advantage of the assistance available from the IAEA Technical Cooperation program for the procurement of equipment and services for the replacement of the CPS at the WWR-K reactor. As a first step, INP is developing a draft technical requirements document for the new system. IAEA technical and procurement staff will work closely with the team from INP, as will the reactor designer and the funding parties in reviewing the technical requirements document for completeness and compliance with applicable safety regulations and standards. The completed technical requirements document will be submitted to the KAEC, for approval. After approval by KAEC, it will be formally submitted to the IAEA along with a request for procurement, to initiate a standard tender procedure. The contract between IAEA and the selected supplier of the replacement CPS is expected to be signed in late 2008.

Replacement of the CPS includes the following activities:
- Replacement of the reactor power monitoring, automatic control and emergency shutdown systems based on in-core neutron flux measurements;
- Installation of a new module for collection and processing of data from instruments measuring non-nuclear parameters. The seismic detection instrumentation will be replaced, although the existing instrumentation will be retained for measurement of other non-nuclear parameters;
- The existing control rod drive motors will be retained but the control rod position monitoring instruments will be replaced and a new control system will be installed. The control rod position indicators on the reactor control desk will also be replaced;
- The control desk in the main control room and all reactor operations and emergency system displays will be replaced;
- The control desk in the stand-by control room will be replaced (including all safety system equipment, neutron flux and rod position indicators and operational controls);
- All audible and visual warning systems will be replaced;
- A new data management system will be installed, to provide upgraded collection, storage and processing of information on operation of the reactor and ancillary equipment, plus improved control room data readouts;
- Provision of an auto-diagnostic system to provide pre-startup testing of the relevant instrumentation and monitoring equipment and testing of the serviceability of operating equipment.

Both to ensure maximum reliability and to simplify maintenance, the replacement CPS will, to the extent possible, be based on standard industrial controllers. Performance of the power level and rate of power increase control systems and the emergency protection system will be tested under reactor operating conditions during commissioning of the new CPS.

Procurement and installation of the replacement CPS will include the following steps:
- development of the technical requirements document for the new system compatible with other core modifications,
- development of the system design for the new system,
- equipment manufacturing and factory acceptance tests,
- installation of the new CPS at the reactor,
- commissioning of the new CPS and staff training.

These activities will meet the regulations, technical and safety norms and standards of Kazakhstan, IAEA, IEC and other relevant bodies that set safety standards for the operation of research reactors and replacement of research reactor control and protection systems.

Modeling of Thermal Hydraulics Behaviour in Reactor Core Of Reaktor TRIGA PUSPATI (RTP)

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Abstract

Reactor TRIGA PUSPATI (RTP) in Malaysian Nuclear Agency (Nuclear Malaysia) is the one and only research reactor in Malaysia and had been used exclusively for research and development (R&D), training for reactor operators and education purposes. The RTP is a 1 MWt pool type reactor with natural convection cooling system and pulsing capability up to 1200 MWt. It went critical on 28 June 1982 and the core configuration has been changed twelve times to date. The core is a mixed type using 20% enriched U-ZrH fuel element containing 8.5, 12 and 20wt% uranium.

This paper will discuss the modeling of thermal-hydraulics behaviour in reactor core of RTP using computer code namely PARET. The results of the calculation that were carried out at RTP are modelled and temperature profiles of the thermal hydraulics data at different locations and power levels are developed.

As a comparison to the thermal hydraulics calculation using PARET, an experiment were carried out at several different locations and power levels in the reactor core for temperature profile in the core to compare the result obtained from PARET. Finally, an overall analysis of the result of PARET calculation and experimental measurement were exhibited in this paper.

Keywords : TRIGA, PARET, thermal-hydraulics behaviour

PUSPATI TRIGA Reactor: 25 Years of Safe Operation and Strategies for Ensuring Safety and Security

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The PUSPATI TRIGA Reactor (RTP) is a pool-type 1 MW research reactor which has been in operation since 28 June 1982. During its 25 years of safe operation, no incidents as listed in the unusual event reporting categories have been reported. A pragmatic approach to safety has ensured that the reactor is operated within its operating limits and controls, while regular maintenance has been carried out according to manufacturer's recommendations. In addition to this, several reactor systems or components have been repaired, refurbished or replaced over the years, in order to maintain the reactor integrity. However, ageing systems and components especially the instrumentation and control systems are of pressing concern. This has lead to the initiation of a project to upgrade the present analog control console. An in-service inspection has also been initiated with the assistance with the non-destructive testing group in Malaysian Nuclear Agency.

An important element in ensuring reactor safety is the personnel involved in the reactor operation and maintenance. Previously, insufficient personnel were allocated to the reactor facility however this has changed since early 2006 where the number of personnel has more than doubled. Retraining and upgrading of the current reactor operators are underway and new reactor operators are also being trained.

The last safety analysis report (SAR) for RTP was written in 1983, a year after the reactor started its operation. Hence, in 2006 and 2007, there was a concerted effort to update the SAR and it has been submitted to the regulatory body in mid 2007 as part of the requirement for the issuance of the operation license of RTP.

In 2004 RTP was awarded the ISO9001:2000 for its irradiation services for neutron activation analysis. However, in order to be in line with the practice in the nuclear industry, a quality assurance programme (QAP) was developed in 2006 and implemented in mid 2007.

An assessment of the safety culture at RTP was carried out by an international peer review team in early 2006. In response to the recommendations of this peer review, several changes and improvements to the facility and procedures were implemented. Subsequently, the peer review revisited the facility and commended on the improvement and prompt implementation of the majority of the recommendations.

The recent emphasis on the security of research reactors world wide has lead to the reassessment of the physical protection system and emergency preparedness at the reactor site. Improvements are being implemented and systems are being enhanced.

In conclusion, the strategies taken to enhance the safety and security of the RTP in line with international practice will assure the safe operation of the last 25 years is continued in the future.

Core Calculation of 1MW PUSPATI TRIGA Reactor (RTP) using Continuous Energy Method of Monte Carlo MVP Code System

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The RTP is a light-water moderated and pool-type TRIGA MARK II reactor with power capacity of 1MWt. It was built in 1979 and attained the first criticality on 28 June 1982. The RTP was designed mainly for neutron activation analysis, small angle neutron scattering, neutron radiography, radioisotope production, education and training purposes. It uses standard TRIGA fuel developed by General Atomic in which the zirconium hydride moderator is homogenously combined with enriched uranium. It has a cylindrical core with which possibility of locating 127 of fuel elements. Both of the coolant and moderator uses light water system and the reflector is made of high purity graphite. Because of its relatively small power, it uses natural convection for its cooling system. To ensure the integrity of the core, fuel shuffling have been carried out several times. Until now, there were 12 configurations of the core, the most recent change being in July 2006. This paper will describe the RTP core calculation using the Monte Carlo MVP code system.

MVP is a general multi-purpose Monte Carlo code for neutron and photon transport calculation in order to have an accurate and fast Monte Carlo simulation of neutron and photon transport problems. The MVP Monte Carlo code calculation is based on the continuous energy method. This code is capable of adopting an accurate physics model, geometry description and variance reduction technique. When compared to the conventional scalar method, this code could achieve higher computation speed by up to a factor of 10 on the vector super-computer.

The RTP core has been modelled using cylinder geometry along the z-coordinate geometry with the MVP code system while its material cross section data is calculated beforehand. The JENDL3.3 data library was used in the whole calculation. The objectives of the calculation are to calculate the multiplication factor values (k_{eff}), fission density and flux distribution from the tally data. The calculation also gives control rod worth value and comparison with experimental data was made to evaluate the safety of the reactor.

Operational Experience and Programmes for Optimal Utilization of the Nigeria Research Reactor-1

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Abstract
The Nigeria Research Reactor-1 (NIRR-1) is the nation's first nuclear reactor and it is sited at the Centre for Energy Research and Training, Ahmadu Bello University, Zaria, Nigeria. It is a Miniature Neutron Source Reactor (MNSR) that attained criticality on February 03, 2004 and was licensed to operate at a maximum power of 31 kW three days a week in June 01, 2004. This presentation enumerates the measures put in place to ensure safe operation and adequate maintenance regime as well as the strategic plans for optimal utilization of the reactor. Some of these measures, which bothers on safe operation and sustainable maintenance culture that have been implemented include: strict adherence to the periodic preventive maintenance routines; standard procedures for pre-startup, startup and shut down procedures; provision of a quick access to reactor top to facilitate rapid response in case of emergency, especially in the case of rod-stuck incident. Similarly, on the basis of experience gained since the commissioning vis-à-vis the neutron flux spectrum characteristics of the MNSRs, experimental protocols are presented for the analysis of elements producing short-lived, medium-lived and long-lived activation products in geologic materials with negligible nuclear interferences especially for the analysis of Al in the presence of Si. Furthermore, research and development activities in core physics analysis and thermal hydraulics with regards to conversion from the current HEU core to a LEU core under the aegis of the IAEA Coordinated Research Project entitled "Conversion of MNSR to LEU" are outlined.

REGULATORY CONTROL OF THE NIGERIAN RESEARCH REACTOR (NIRR-1)

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Abstract
The Nuclear Safety and Radiation Protection Act 1995 (Act) established the Nigerian Nuclear Regulatory Authority (NNRA) and charged it with regulatory oversight of ***nuclear research reactors and also*** with responsibility to ensure Nigerian compliance with International safety, security and safeguards legal framework. Under Technical Cooperation with the Agency, Nigeria installed and commissioned **NIRR-1**, which was licensed by the NNRA. The NNRA has since exercised regular oversight on the facility, ensuring its safe operation.

1. Introduction - Legal and Regulatory Infrastructure

The Nuclear Safety and Radiation Protection Act 1995 (Act), established the Nigerian Nuclear Regulatory Authority (NNRA), with responsibility for nuclear safety and radiological protection regulation in Nigeria. Amongst others, NNRA shall *perform all necessary functions to enable Nigeria meet its national and international safeguards and safety obligations in the application of nuclear energy and ionizing radiation*; *advise the Federal Government on nuclear security, safety and radiation protection matters*; *regulate the safe promotion of nuclear research and development, and the application of nuclear energy for peaceful purposes*; and *liaise with and foster cooperation with international and other organizations or bodies concerned having similar objectives*. The NNRA has powers to license operators ***nuclear research reactors and critical assemblies, nuclear power reactors, mining and milling of radioactive ores and other facilities of the nuclear fuel cycle.***

Nigeria signed the **Nuclear Non Proliferation Treaty** (**NPT**) and voted for its indefinite extension in 1995. Nigeria also signed the **Comprehensive Safeguards Agreement** in 1988 and the **Protocol Additional to the Application of the Safeguards Agreement** (**Additional Protocol**) in 2001. Nigeria ratified the **Convention on Early Notification of a Nuclear Accident** and the **Convention on Assistance in the Case of Nuclear Accident or Radiological Emergency**. Furthermore, the President has recently signed the instruments ratifying the *protocol Additional to the Agreement between the Federal Republic of Nigeria and the Agency for the Application of Safeguards; Agreement on the Privileges and Immunities of the Agency; Convention on Nuclear Safety; Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management; Convention on the Physical Protection of Nuclear Material; Amendment to the Convention on the Physical Protection of Nuclear Material; Vienna convention on Civil Liability for Nuclear Damage.* These shall be deposited at the IAEA.

An Agency International Nuclear Security Service (INSServ) mission was conducted in 2004, which concluded amongst others that ***Nigeria should consider a review of the 1995 Nuclear Safety and Radiation Protection Act to ensure that it adequately covers nuclear security***. Consequently an Inter-Ministerial Committee has reviewed the Act and strengthened it with provisions that *Established and empowered the National Nuclear*

Security Committee; Broadened NNRA's scope to include the regulation of nuclear installations; Imposed strict liability for nuclear damage on a nuclear installation licensee; Specified duties regarding nuclear accidents and incidents, and Provided for emergency planning for nuclear accidents or incidents. The revised Act is awaiting passage by the Federal Parliament.

2. NIRR-1

Nigeria's compliance with international conventions, the promulgation of the Act and the establishment of the NNRA in 2001 facilitated the installation and commissioning of the **Nigeria Research Reactor-1** (or **NIRR-1**), covered by INFCIRC/526 - Project and Supply Agreement signed at the Agency in 1996, between the Agency, the Governments of Nigeria and China. **NIRR-1** is located at the Centre for Energy Research and Training (CERT), Ahmadu Bello University, Zaria. It is designed for teaching, research and for use in Neutron Activation Analysis (NAA) and limited radioisotope production. It is a tank-in-pool type reactor with 90% HEU (U-Al alloy) as fuel, light water as moderator and coolant, and metallic beryllium as reflector. It has a built-in clean cold core excess reactivity of 3.77 mk. It can operate for a maximum of 4.5 hours at full power (i.e. equivalent to a thermal neutron flux of $1x10^{12}$ $n.cm^{-2}.s^{-1}$ in the inner irradiation channels.

2.1 Authorization of NIRR-1

In August 2001, the NNRA inaugurated a Technical Advisory Committee (TAC) for the **Establishment of Licensing Procedure for Nuclear Research Reactors** and for the **Licensing of Reactor Operators in Nigeria**. TAC witnessed the fuel importation, installation, criticality experiments, commissioning tests, other initial operations and submitted its report in July 2002. It also developed and submitted the **Guide to Licensing Nuclear Research Reactors in Nigeria** and the **Guide to Licensing Nuclear Research Reactor Operators in Nigeria**. CERT applied for authorization to operate NIRR-1 early in 2004 on completion and submission of a Commissioning Test Report. NNRA's first reactor safety inspection in May 2004 assessed the Programmes for **Physical Security**, **Emergency**, **Access Control**, **Maintenance**, **Quality Control** and **Radiation Protection** of NIRR-1. NNRA also observed the Reactor Operators run NIRR-1 for the first time without Chinese assistance. The **Reactor Operation Licence** and **Reactor Operators Licence** for **Categories A**, **B** and **C Operators** were later issued in June 2004.

2.1 Inspection, Enforcement and Safety of NIRR-1

NNRA commenced its regular schedule of inspection in September 2004. These included compliance and quarterly safety inspections to assess CERT's compliance with the terms and conditions of the Reactor Operations Licence. By May 2005, some inspection findings revealed certain major non-compliance that could potentially impede strict adherence to acceptable limits. The necessity for closer monitoring compelled the NNRA to institute a regular monthly safety inspection schedule. Further violations considered to be potentially detrimental to skill acquisition, confidence building and sustainability, and which may negatively impact on safety were uncovered in June 2005. Thus the NNRA in August 2005 directed CERT to stop the operation of the reactor until proper measures were emplaced to ensure overall reactor safety. Further NNRA interventions ensured improvement of QA, QC and ultimately helped the renewal of CERT's licences. The

Agency has also conducted regular annual Safeguards Inspections since 2003 and in May 2005, sent an ***International Nuclear Security Advisory Service (INNServ) Mission*** to Nigeria. Additionally, the Agency also sent an ***Incident Reporting for Research Reactor Mission*** in February 2005. These have helped CERT to focus on enhancement of the safety aspects for reactor operation and on improvement of the safety culture.

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Comparative dose calculation for TRIGA HEU and LEU fuel in nuclear accident situations

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INR-Pitesti TRIGA research reactor is basically a pool type reactor with a special design in order to fulfill the requirements for material testing, power reactor fuel and nuclear safety studies. The dual-core concept involves the operation of a TRIGA-SSR high-flux, steady-state research and material testing reactor at one end of a large pool, and the independent operation of an annular-core pulsing reactor (TRIGA-ACPR) at the other end of the pool. The steady-state reactor is used for long term testing of power reactor fuel components (pellets, pins, subassemblies and fuel assemblies), and the annular-core pulsing reactor is used for transient testing of power reactor fuel specimens.

The safety evaluation involved a several design basis accidents [1]: single-pin cladding failure in water, 25-pin fuel bundle failure in water and in air, accidental reactivity insertions, loss of flow from main coolant pump accidents, and interaction between the two cores within the tank.

One fission product release category were analyzed - a design basis release. The failure of a single fuel pin classing (due to material deficiencies) with the consequent release of fission products is an event that has a small but significant probability and, over the life of the core, it could be expected that such a failure could occur during normal operation. For the design basis release has been assumed to be no retention of volatile fission products in the fuel-moderator material. This failure was analyzed using the following assumptions:

a) The fuel pins that fail have operated at an average power density of twice as great as the average power density in the core;
b) The core has operated continuously for a total of 7700 MWd;
c) For the anticipated release, experimentally determined fraction of volatile products released from the fuel material [2] will be about 6.3E-04;
d) For the design basis release, there is no retention of volatile fission products in the fuel-moderator material;
e) 100% of the noble gases in the fuel-clad gap are released from the fuel bundle and, subsequently, are transferred directly to the reactor hall. 25% of the halogens are released from the fuel bundle (with the remainder assumed to plate-out on the relatively cool clad). As regarding the halogens that escape for the anticipated (single-pin failure) release, 10% are assumed to form organic compounds which escape in the pool water. Only 1% of the balance is undissolved in the pool water and appears in the reactor hall air. The net halogen release to the reactor room and potentially outside is 2.725%. All other fission products remain in the pool or are otherwise unable to escape from the reactor room. For the design basis release (in air) fully 25% of the halogens are released to the reactor hall. All other fission products are assumed unable to escape from the reactor room because of plate-out on cool surfaces [2];
f) The release height is assumed to be 50 m above the ground level;
g) For this case, the fuel bundle is in air. The assumption for this case, that there is no retention of fission products in the fuel, implies that the fuel involved is at temperatures

approaching the melting point of U-ZrH. That is inconceivable because the fuel bundle is in water.

For training purposes, and to exercise our ability to conduct Level-3 PSA studies, a severe accident scenario involving 14-MW INR-TRIGA research reactor has been developed. In this scenario is assumed that a large part of the reactor hall roof or a heavy object escaped from the crane hook is dropped over the 14-MW TRIGA-SSR core, resulting in mechanical damage of the core. It is assumed, also, that no core melting is occurring, but only fuel-cladding rupture being involved for several 25-pins fuel bundles.

The accident was analyzed using the following assumptions:

a) The core has operated continuously for a total of 7700 MWd;
b) The affected fraction of the core is 45%
c) For the release, experimentally determined fraction of volatile products released from the fuel material [2] will be about 6.3E-04;
d) There is no retention of volatile fission products in the fuel-moderator material;
e) 100% of the noble gases in the fuel-clad gap are released from the fuel bundle and, subsequently, are transferred directly to the reactor hall. 25% of the halogens are released from the fuel bundle (with the remainder assumed to plate-out on the relatively cool clad). As regarding the released halogens, 10% are assumed to form organic compounds which escape in the pool water and 1% appears in the reactor hall air. There is no retention of fission products in the considered fuel (as shown above), and due to the high temperature of fuel a 2.0E-02 release fraction was assumed for other fission products than noble gases and halogens;
f) During the accident evolution, an emergency ventilation system occurs and charcoal traps are not available, so no fission products will be retained by traps;
g) The release height is assumed to be 60 m above the ground level;
h) The radiological consequences assessment has been performed with PC-COSYMA computer code [3], considering a site specific meteorological file (being generally known that the dispersion model -MUSEMET- used in PC-COSYMA version of the computer code have a bug for single stability class type calculation, for the D stability class). By using a meteorological file with hourly changes in wind direction for each hour at each phase option considered, the arise of bug is avoided;
i) The calculation was performed by considering both possibilities regarding countermeasures: no countermeasures for the general public and the environment are taken, and countermeasures are implemented considering site specific intervention parameters [4, 5];
j) As regarding the calculations, both deterministic and probabilistic approaches have been used.

The evaluation of the radiological consequences considers both early and late consequences.

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Safety Analysis of MNSR Reactor during Reactivity Insertion Accident Using the Validated Code PARET

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In the frame work of the IAEA's CRP project (J7.10.10) on "Safety significance of postulated initiating events for various types of research reactors and assessment of analytical tools" the Syrian team contributed in the assessment of computational codes related to the safety analysis of research reactors [1]. During the project implementation the codes PARET and MERSAT have been tested, modified and verified regarding specific phenomena related to safety analysis of research reactors [2]. In the framework of this contribution the code PARET has been applied to model the core of the Syrian MNSR reactor. The code analysis includes the simulation of steady state operation and a group of selected reactivity insertion accident (RIA) including the design basis accidents dealing with the insertion of total available excess reactivity.

For this purpose a PARET input model for the core of MNSR reactor has been developed enabling the simulation of neutron kinetic and thermal hydraulic of reactor core including reactivity feedback effects. The neutron kinetic model depends on the point kinetic with 15 groups of delayed neutrons including photo neutrons of beryllium reflector. In this regard the effect of photo neutron on the dynamic behaviour has been analysed through two additional calculations. In the first the yield of photo neutrons was neglected completely and in the second its share was added to the sixth group of delayed neutrons. In the thermal hydraulic model the fuel elements with their cooling channels were distributed to 4 different groups with various radial power factors. The pressure loss factors for friction, flow direction change, expansion and contraction were estimated using suitable approaches.

Figure 1 presents the evolution of relative reactor power after a step reactivity change of 1 mk starting from 20% of reactor nominal power. Figure 2 shows the developments of average core temperature following a complete withdraw of reactor control rode from the cold core condition corresponding to a reactivity insertion of about 6 mk .

The results of these RIA simulation show good agreement with the experimental data. Thus, it can be concluded that the code PARET possess good ability to model the expected thermal hydraulic and neutron dynamic phenomena. Particularly the results indicate the correct simulation of inherent safety features of MNSR reactor resulting from the strong reactivity feedback effects of coolant temperature under natural circulation conditions.

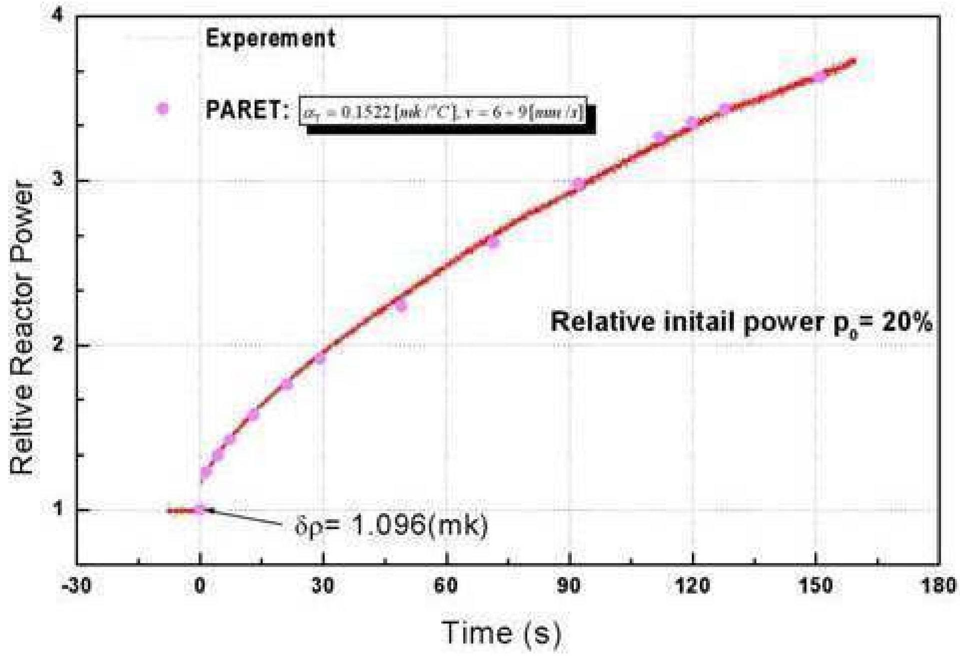

Fig.1. Relative power distribution after a step reactivity change starting from the initial power level of 20% of nominal reactor power.

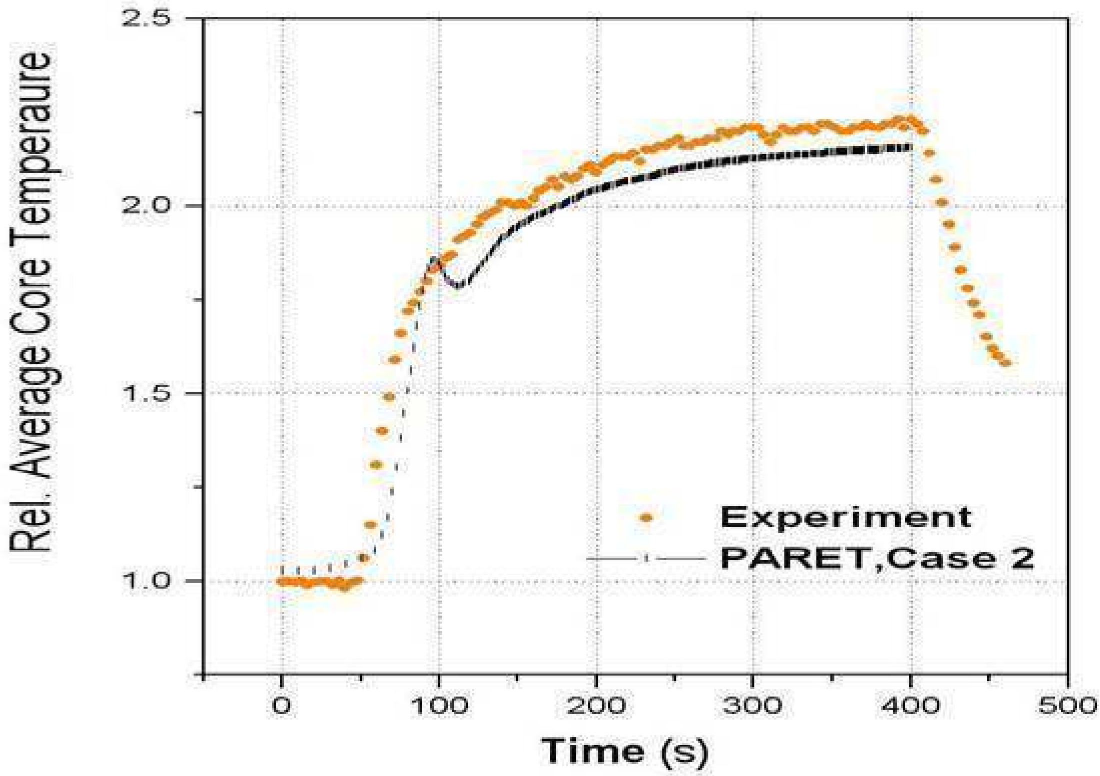

Fig.2.Evolution of relative average core temperature following a complete withdraw of reactor control rode.

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Fuel management methodology upgrade of Thai Research Reactor (TRR-1/M1) using SRAC computer code

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Thailand Institute of Nuclear Technology (TINT) is currently responsible for the nuclear research reactor called “TRR-1/M1” which is located in Bangkok. The existing fuel management tool for TRR-1/M1 is a computer program called TRIGAP [1] which was developed in Slovenia during the 80’s. Although TRIGAP is capable of calculating reactor parameters such as core excess reactivity or neutron fluxes, this tool has several drawbacks. Since TRIGAP only models the spatial distribution of neutrons in cylindrical geometry, the TRR-1/M1 core, which is formed in hexagonal lattices, needs to be homogenized into cylindrical rings. As a result, TRIGAP is unable to provide pin-wise data such as normalized power distribution of the reactor. To overcome this, an upgrade to the existing methodology is proposed. The upgraded methodology is actually similar to the existing methodology that it is executed in 2 steps. However, both steps are performed by more advanced computer programs collectively packaged into one system called “SRAC” [2] which has been developed in Japan since 1978. The first step, which is the group cross section generation, is performed by the PIJ module of SRAC system utilizing 2D collision probability method. Typically, the group cross section generation is performed using infinite arrays of 2D lattice models corresponding to unique lattice regions of the reactor. In essence, each 2D lattice model represents an axial node which has the same material throughout axial direction. For TRR-1/M1, there are three types of rods: fuel element, fuel follower control rod and air follower control rod. Each type of the fuel rods is divided axially into different types of 2D lattice models. Furthermore, the water lattice model is created to represent the lattice filled completely with water. To generate the group cross sections, these 2D lattices are classified into two model types, i.e., fuel type lattice and non-fuel type lattice. The fuel type lattice is performed as it is in the group cross section generation while the non-fuel type lattice is performed in a special color-set model to simulate the environment effects from surrounding fuel rods. Moreover, the group cross sections of the fuel type lattice are generated at various burnup points from fresh to very high burnup and they are also generated at different power levels which correspond to different equilibrium temperatures. Following the group cross section generation step, the reactor calculation step is performed. The upgraded methodology uses the COREBN module of the SRAC system to perform the reactor core calculation. The COREBN module has a capacity of performing burnup calculation and modeling hexagonal lattice of TRR-1/M1 core.

To validate the upgraded methodology, the group cross sections of different lattices needed for the reactor core calculation were generated by the PIJ module of the SRAC system and the reactor core calculations of TRR-1/M1 core number 1 and 2 were performed afterwards. The K_{eff} of the “all-rods-out” models of the reactor cores were derived and the excess reactivity was calculated by (K_{eff} -1)/(Beta*K_{eff}) where Beta is fraction of delayed neutrons (0.007). Table I presents the core excess reactivity results of TRR-1/M1 core number 1 and 2 with the comparison against operation data from the operation log book.

Table I: Core excess reactivity results of TRR-1/M1 core number 1 and 2

Model	Keff by SRAC code	Calculated core excess reactivity by SRAC code	Operation core excess reactivity
Core #1	1.05809	7.84$	7.43$
Core #2	1.05252	7.13$	6.87$

As it can be seen from Table I, the excess reactivity calculated by SRAC system agrees well with the operation data when considering that the operation value has inherently some amount of measurement uncertainty. There seems to be a bias of around 0.40$ between the calculated results and the operation data. Also, the change of core excess reactivity as a function of power level is obtained from series of reactor core calculations at various equilibrium temperatures. Fig. 1 presents the changes of core excess reactivity as a function of power level of TRR-1/M1 core number 1.

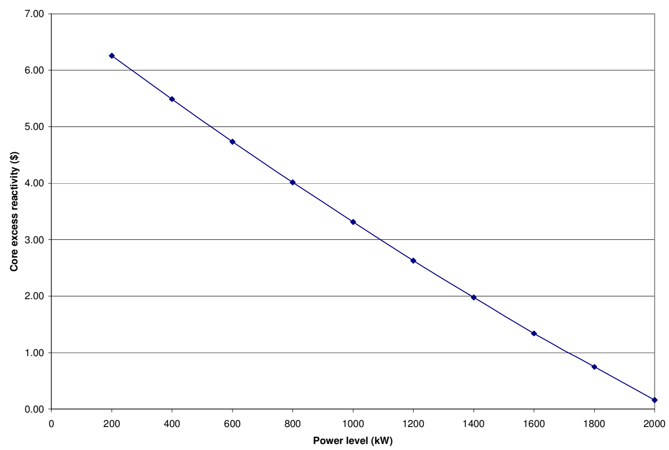

Fig. 1: Core excess reactivity of TRR-1/M1 core #1 as a function of power level (kW)

From Fig. 1, the core excess reactivity decreases quite linearly as a function of power level as expected. This result confirms the negative temperature feedback of the TRR-1/M1. It can be concluded that the upgraded methodology is functioning well and it can be used routinely as a fuel management strategy for TRR-1/M1.

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Poster Presentations: Utilization

Synopses no. IAEA-CN-156/	Synopses Title	Main Author
U-54	Operating Experience Utilization Programmes of the BAEC 3 MW TRIGA Mark-II Research Reactor in Bangladesh	Haque, M.
U-55	Studies on environmental pollution in Bangladesh using reactor based neutron activation analysis technique	Abdul Latif, S.
U-8	Improvement on Sensitivity for the Track - Etch Neutron Radiography	Pugliesi, R.
U-10	New Perspectives for the TRIGA IPR-R1 Research Reactor	Soares Leal, A.
U-15	First experiments in the IPEN-CNEN/SP PSD Neutron Powder Diffractometer	Parente, C.B.R.
U-45	Future utilization of the Research Reactor IRT IN SOFIA after its reconstruction	Ilieva, K.D.
U-38	Utilization of the LVR-15 Research Reactor at Rež	Marek, M.
U-49	Utilization of the VR-1 Training Reactor	Matejka, K.
U-28	The 250 kW FiR 1 TRIGA Research Reactor - International Role in Boron Neutron Capture Therapy (BNCT) and Regional Role in Isotope Production, Education and Training	Auterinen, I.
U-6	Chemical characterization of early fine-ware pottery by neutron activation analysis: analytical and statistical approaches to production and trade	Kilikoglou, V.
U-56	Feasibility Study of I^{125} Brachytherapy in Indonesia	Soentono, T.M.
U-17	Characterization of Airborne Particulate Matter at Urban and Rural Area in Bandung and Lembang Indonesia using Instrumental Neutron Activation Analysis	Santoso, M.
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U-33	Optimizing Conditions Suited for Stress Determinations in Q-Space Focusing Configurations	Ionita, I.
U-34	SANS facility at the Pitesti 14MW TRIGA Reactor	Ionita, I.
U-19	RIAR Capabilities in Support of the Innovative Nuclear Technologies	Bychkov, A.V.
U-21	SM Reactor After Core Modernization	Klinov, A.
U-22	Set of Investigations of HFR Fuel Rods in Justification of their Serviceability and Safe Operation	Tsykanov, A.V.
U-12	Utilization of irradiation holes in HANARO	Lee, C.S.
U-27	Design and Installation of Fuel Test Loop in HANARO	Ahn, S.H.
U-42	Design Characteristics of Cold Neutron Source in HANARO	Wu, S.I.

U-47	Pure Commercial Gold Foils As Neutron Flux Monitor Neutron Self-Shielding Assessment	Helal, W.
U-50	Implementation of TRR-1/M1 for Thailand's Nuclear Engineering Program	Nilsuwankosit, S.

Operating Experience and Utilization Programmes of the BAEC 3 MW TRIGA Mark-II Research Reactor of Bangladesh

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Synopsis

The 3 MW TRIGA Mark-II research reactor of Bangladesh Atomic Energy Commission (BAEC) has been operating since September 14, 1986. The reactor is used for radioisotope production (^{131}I, ^{99m}Tc, ^{46}Sc), various R&D activities, manpower training and education. The reactor has been operated successfully since it's commissioning with the exception of a few reportable incidents. Of these, the decay tank leakage incident of 1997 is considered to be the most significant one. As a result of this incident, reactor operation at full power remained suspended for about 4 years. However, the reactor operation was continued during this period at a power level of 250 kW to cater the needs of various R & D groups, which required lower neutron flux for their experiments. This was made possible by establishing a temporary by pass connection across the decay tank using local technology. The reactor was made operational again at full power after successful replacement of the damaged decay tank in August 2001. At that time, several modifications of the reactor cooling system along with its associated structures were also implemented and then necessary testing and commissioning of the newly installed component/equipment were carried out. The other incident was the contamination of the Dry Central Thimble (DCT) that took place in March 2002 when a pyrex vial containing 50g of TeO2 powder got melted inside the DCT. The vial was melted due to high heat generation on its surface while the reactor was operated for 8 hours at 3 MW for trial production of Iodine-131 (131I). A Wet Central Thimble (WCT) was used to replace the damaged DCT in June 2002 such that the reactor operation could be resumed. The WCT was again replaced by a new DCT in June 2003 such that radioisotope production could be continued.

The facility has so far been used to train up a total of 27 personnel including several foreign nationals to the level of Senior Reactor Operator (SRO) and Reactor Operator (RO). The reactor is operated 4 days a week at a power level of 3 MW for production of Iodine-131. During the other one weekday, the reactor is operated at lower power levels (250 – 500 kW) to cater the needs of NAA and NR groups. At present the total burn-up of the core stands at about 484 Megawatt Days. BAEC has a plan to increase the production of Iodine-131 to install more dry tubes in the core so as to meet the total demand of RI in the country. There is also plan to develop unused experimental facilities such as, thermal column and radial beam ports for strengthening the R& D activities around the reactor. A total of 1093 irradiation requests (IRs) have been catered so far for different reactor uses. The total amount of RI produced stands at about 4000 GBq. The total amount of burn-up-fuel is about 11607 MWh. Efforts are on to undertake an ADP project so as to convert the analog console and I&C system of the reactor into digital one. The paper summarizes the reactor operation, maintenance experiences and utilization programmes focusing on troubleshooting, rectification, modification, RI production, various R&D activities and training program being conducted at the facility.

Keywords: Reactor, Dry Central Thimble (DCT), Wet Central Thimble (WCT), demineralize water, irradiation request (IR), pyrex vial, TeO2 powder, radioisotope, burn up.

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Studies on environmental pollution in Bangladesh using reactor based neutron activation analysis technique

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Environmental and health related problems have become a major global concern in the recent years. Bangladesh is now facing a serious problem about arsenic (As) and chromium (Cr), which contaminate our environment. Arsenic exposure is a potential health risk to local populations in most of the parts of Bangladesh.

Hazaribagh is a densely populated area of Dhaka city in Bangladesh, where about 149 leather-processing industries are in operating and discharge a lot of solid and liquid waste directly to the low-lying areas, river and natural canals without proper treatment. These tanneries follow the practice of chrome tanning. In this practice the leather takes only 50-60% of the applied chromium and the remaining is discharged as waste. The pollution load emanating from tanneries is directly affecting surface water, ground water, soil and air. A number of people have been affected directly and indirectly with chromium of wastes from tannery industries.

Environmental research using instrumental neutron activation analysis (INAA) for the determination of trace and ultra-trace element pollutants has a great potential in relation to human health. The total element concentration has been traditionally used to assess environmental impact and health risk of the element. We have a 3MW TRIGA Mark-II research reactor at Atomic Energy Research Establishment, Savar, Dhaka, Bangladesh. Our interest is to study environmental pollution due to As and Cr through distribution in environment over Bangladesh. Particularly, this work was undertaken for determining As content in water, soil and herbal plants, and Cr-content in soil of tannery industrial areas as a part of our systematic studies.

For Cr-determination, thirty soil samples were collected at several depths of different locations (L-1, L-2 ......) from both of Hazaribagh tannery area and the Tatuljhura proposed tannery area. The samples were dried in an oven at a temperature of about 70° C until they attained constant weight followed by ground. Two sets of samples for Hazaribagh and Tatuljhura were prepared for separate irradiations. The samples and standards (IAEA Soil-7 & SL-1 and NIST Coal Fly Ash (CFA) 1633b) were irradiated with thermal neutrons of ~ 2 $\times 10^{12}$ n cm^{-2} sec^{-1} for 3 hours at 250 kW in Lazy Susan of the 3 MW TRIGA Mark-II research reactor. The activities of irradiated samples were measured by HPGe gamma-ray spectroscopy system coupled with computer based S100 MCA acquisition software.

Arsenic levels in water, soil and herbal samples collected from 10 individual locations of Sonargaon in Narayanganj district were determined by INAA technique. These samples were irradiated at the same irradiation conditions as chromium.

Quality control (QC) test is performed to investigate the reliability of the analysis by measuring chromium concentration in certified reference materials CFA, SL-1 and Soil-7 relative to primary standard. The experimental results are varied with the certified values within 6%. The deviation was achieved within the uncertainties quoted with the certified values. Therefore, INAA is an efficient method to determine chromium in soil samples.

The Cr-element was identified via the $^{50}Cr(n,\gamma)^{51}Cr$ reaction. The gamma ray emitted from ^{51}Cr at 320.1 keV was not interfered from other short-lived radioactive nuclides since the samples were measured after long cooling time. In Hagaribagh, the Cr-content in the range of 880 to 33550 ppm was found in surface soils and it was 71-90 ppm for Tatuljhura. For Hagaribagh, the Cr-concentration decreases with the increasing of depth upto 180 cm, and then scattered results were found as shown in Fig.1. The Cr-concentrations of soils in Hagaribagh are rather high and in the most cases these are above permissible level. The Cr-concentration of Tatuljhora was in the range of 50.87 to 93.63 ppm, which is below the permissible level reported in worldwide (Sandia Corporation, 2000).

Arsenic was detected in each of ten soil samples in the concentration range of 0.63-11.35 ppm, where only one was under permissible level (2 ppm). Five water samples out of ten contain arsenic above permissible level (0.05 ppm). Surprisingly, arsenic was also found

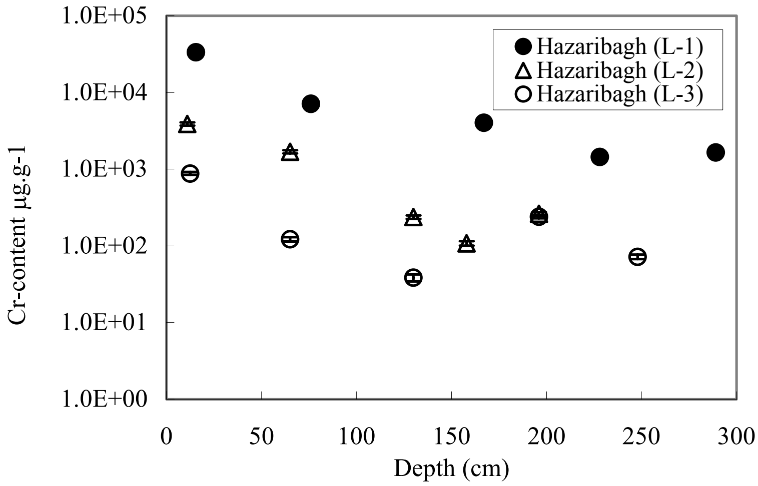

Fig. 1 *Chromium content in soil samples of the tannery area in Hazaribagh*

considerably high in some herbal samples. The arsenic was not found in some herbal samples corresponding to the water samples where arsenic was not detected. Therefore, the presence of arsenic in ayurvedic herbal medicine may be done through the contamination of herbal plants with arsenic contaminated water and soil.

The obtained results will play important role to create public awareness on contamination with As and Cr. This study reports baseline data for the proposed tannery industry in Tatuljhura that will help to assess the level of contamination when the tannery industries will discharge their waste in the environment.

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Improvement on Sensitivity for the Track - Etch Neutron Radiography

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The use of solid state nuclear track detectors(SSNTD) to record neutron radiography(NR) images is a well known technique. The radiography is obtained by irradiating a sample in an uniform neutron beam and a converter screen transforms the transmitted neutron intensity into ionizing radiation which is able to cause damages into the detector. Usually boron based converter screens are used and in this case alpha particles and lithium ions cause the damages. By means of a chemical etching the latent damages are enlarged and are called tracks and, they form a two-dimensional image which is visible by the naked eye. One of the main restrictions to the employment of track detectors for neutron radiography is the low-intrinsic optical contrast achieved in the recorded image that leads to a poor sensitivity to discern thickness changes of the materials. The sensitivity is determined by measuring the light transmitted through the radiographed image and for such purpose, conventional analog optical microphotometers are typically employed.

In the present work a digital system, consisting of a photo enlarger, video camera, capture frame grabber and a computer has been employed to measure the light transmitted and a significant improvement in the sensitivity was achieved. The light intensity is evaluated in a 8 bit gray level scale. The track detector CR-39, 500μm thick was used to record the images and the chemical etching was performed in a KOH(30%) aqueous solution at a constant temperature of 70^{o}C [1]. The sensitivity has been determined for two materials, Plexiglas and iron. The samples are step wedges with thicknesses varying from 2 mm to 12 mm which have been radiographed in a NR facility installed at the radial beam-hole 08 of the 5MW IEA-R1 Nuclear Research Reactor of the IPEN-CNEN/SP [2]. The sample-detector set has been irradiated at a neutron exposure of $E = 4x10^9 n/cm^2$ and the detector etched for 25 minutes. For these conditions the recorded image exhibits the highest intrinsic optical contrast in the detector[3]. The sensitivity expressed as the minimum detectable thickness, is numerically evaluated by:

$$\Delta x = -\Delta(GL)/C \qquad (1)$$

where "Δ(GL)" is the minimal discernible gray level intensity in the image and "C" is a constant depending of the sample and of the digital system.

The *FIG 1* shows the behavior of the gray level intensity(including the detector background) as functions of the sample thicknesses. For the present digital system Δ(GL) = 2.5 and the obtained results for the sensitivity are shown in Table I. The uncertainties in the results have been determined by the standard propagation method applied to (1).

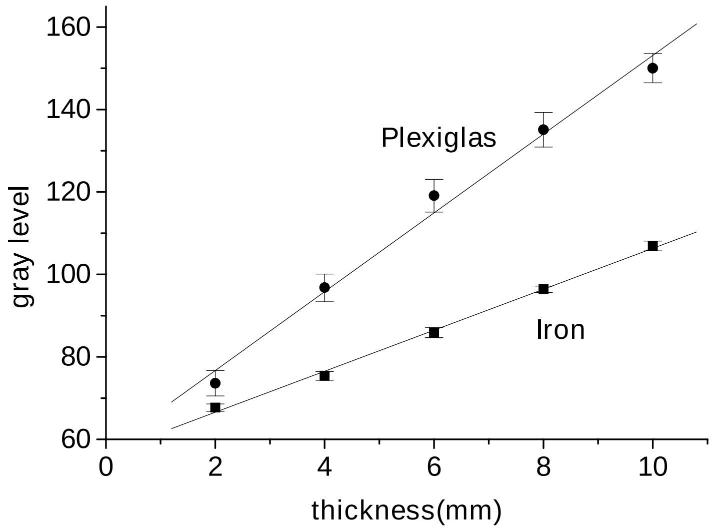

FIG. 1: Behavior of the gray level intensities as functions of the sample thickness.

	Δx(mm)	
	Plexiglas	Iron
Digital system	0.26±0.01	0.50±0.02
Microphotometer	0.47±0.01	0.75±0.03

TABLE 1. COMPARISON OF THE SENSITIVITY VALUES.

In order to compare the potential of the present digital system the Table 1 shows also the values of the sensitivity as evaluated, according to the same procedures, by using a microphotometer [3] to analyze the light transmission. As can be seen the sensitivity provided by the digital system overcomes the one provided by the microphotometer, demonstrating that the former can be an important tool to improve the neutron radiography technique.
The rapid data acquisition is also an important characteristic of this system. Each gray level intensity and its corresponding uncertainty are evaluated by averaging the intensities of about 1700 individual pixels in an area corresponding to about 0.4 cm^2 of the image. This procedure takes some few seconds. For the microphotometer the reading procedure takes about 30 minutes in an area approximately 200 times smaller.

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New Perspectives for the TRIGA IPR-R1 Research Reactor

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The TRIGA IPR-R1 CDTN´s research reactor is of the type MARK I, which core is below the floor level, as it is shown in Figure 1. It is operating since 1960 and the main activities have been the neutron activation analysis and the training of nuclear power plant operators [1]. Since 2001, new projects of utilization of the reactor were initiated as the production of some special labeled molecules to be used in medical purposes the improvement of color of Brazilian gemstones by neutron irradiation [2,3].

More recently, CDTN started a project to evaluate the possibility of obtain a characterized neutron beam from a vertical tube from the TRIGA's core. The availability of a neutron beam with appropriate characteristics as intensity, spectrum and collimating, will enlarge the possibilities de application in these fields and also will open interesting perspectives of new applications such as the study of DNA structures by neutron irradiation and neutrongraphy [4].

This work presents the recent results of these new applications, about gemstones, special activated molecules and also the preliminary simulated results of the neutron extractor of the using the MCNP code.

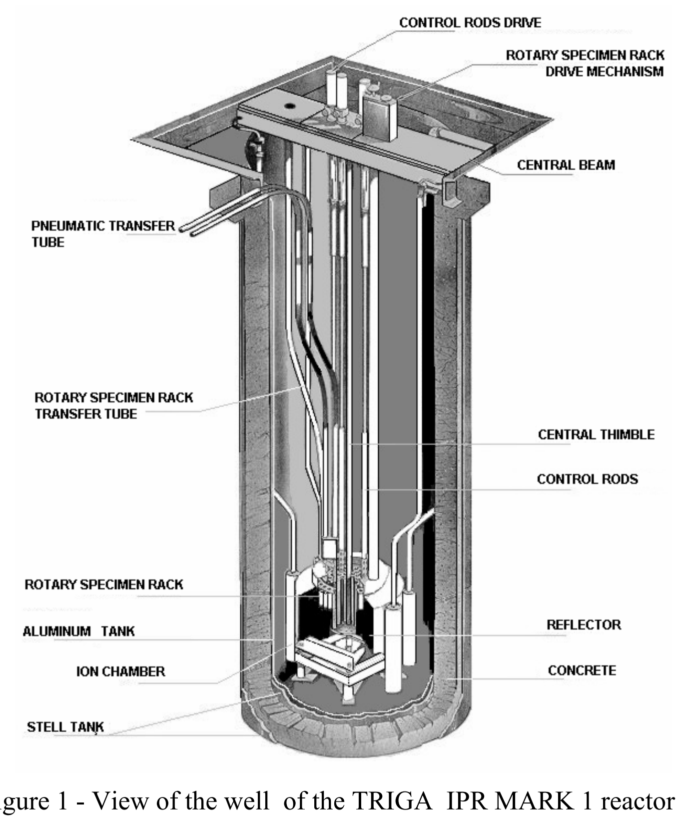

Figure 1 - View of the well of the TRIGA IPR MARK 1 reactor [5].

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First experiments in the IPEN-CNEN/SP PSD Neutron Powder Diffractometer*

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The neutron powder diffractometer, recently installed at the IEA-R1 3.5 MW research reactor, is equipped with a position sensitive detector (PSD) [1]. The PSD spans 20° of a neutron powder diffraction pattern. An extended pattern can be obtained by measuring intensities in 20° segments in a 2θ angular interval ranging from 5 to 125°. At a take-off angle of 84°, a focusing Si monochromator [2], installed in the instrument, can be easily positioned in order that the following wavelengths become available: 1.111, 1.399, 1.667 and 2.191 Å. A rotating-oscillating collimator (ROC) was also installed in the new instrument. The ROC eliminates parasitic scattering from furnace or cryorefrigerator heat shields in the vicinity of the sample. It also makes the PSD less sensitive to the ambient background. The new diffractometer can measure a neutron powder diffraction pattern in a matter of ten to twenty hours with a good statistics. (See reference [3] for more details about this instrument). Figure 1 is a photograph of the IPEN-CNEN/SP PSD neutron powder diffractometer.

FIG. 1. The IPEN-CNEN/SP PSD Neutron Powder Diffractometer. (Photo by Marcello Vitorino).

After calibration of the instrument, a series of experiments were done in order to determine its operational characteristics. In all experiments, wavelength used was $\lambda = 1.399$ Å. Samples used for this purpose were powdered Ni, Si and Al_2O_3. Figure 2 is a comparison between two neutron powder patterns for Ni measured with the old and the new diffractometers. The old

* Instrument financed by Fundação de Amparo à Pesquisa do Estado de São Paulo (FAPESP) under Project no. 95/05173-0.

diffractometer (commissioned in 1966) was a single detector (BF_3)/single wavelength (λ = 1.103 Å) instrument. It was installed at the same place where is now installed the new one.

Observing Figure 2, it is worthwhile to note the improvement in resolution of peaks when using the new instrument particularly for large values of 2θ. Another point to be highlighted in the comparison is that, although having four times the number of intensity points (for a same angular 2θ interval), the new pattern took *ca*. one third of the time spent in the measurement of the old one. Several other patterns of many different compounds as, for example, Fe_2O_3 (magnetite), CeO_2, $SrTi_{0.65}Fe_{0.35}O_{3-\delta}$ and BaY_2F_8:Nd were also obtained with the new diffractometer.

Utilization of the IPEN-CNEN/SP PSD neutron powder diffractometer is open to brazilian and other latin-american scientific and technological communities.

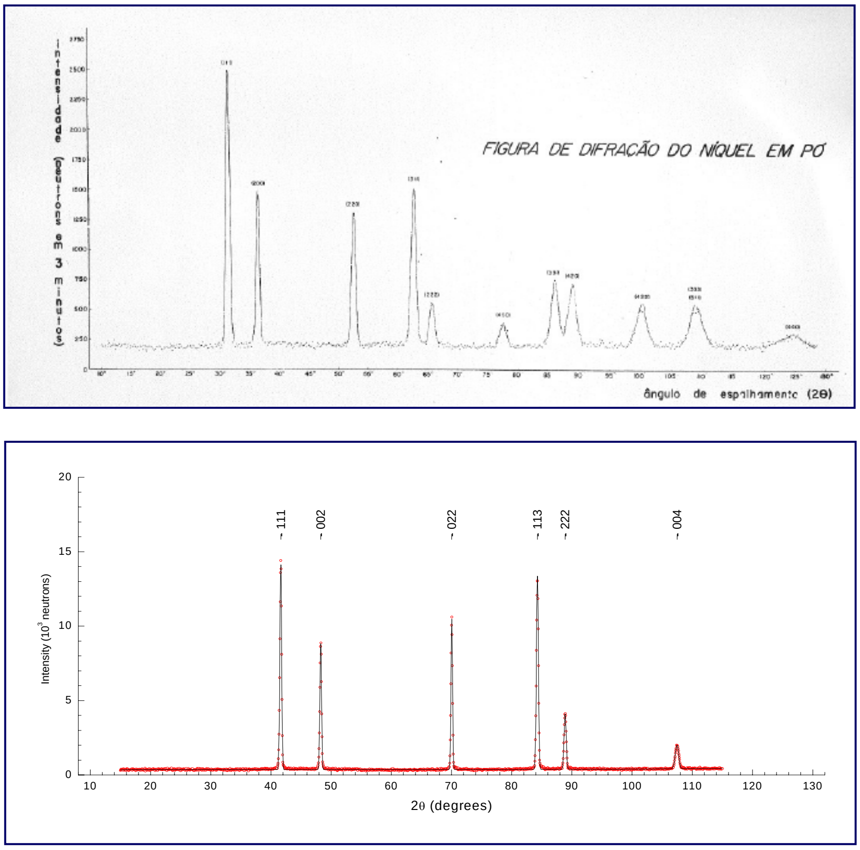

FIG.2. Comparison between neutron powder patterns for Ni measured with the old (above) and the new (below) instruments.

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FUTURE UTILIZATION OF THE RESEARCH REACTOR IRT IN SOFIA AFTER ITS RECONSTRUCTION

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The research reactor IRT-2000 (IRT) in Sofia to the Institute for Nuclear Research and Nuclear Energy (INRNE) was built and put into operation in 1962. It was temporarily shut down in 1989 for improvement. The reconstruction of the IRT is being carried out under the decision of the Council of Ministers of Republic of Bulgaria from 2001. The strategy for sustainable utilization considers the IRT as a national base and aims to satisfy the society needs for:

⇨ education of students and training of graduated physicists and engineers in the field of nuclear science and nuclear energy,
⇨ implementation of applied methods and research,
⇨ development and preservation of nuclear science, skills, and knowledge.

The IRT Technical Design is being in process of elaboration. The IRT will be reconstructed into a reactor:

- of thermal power 200 kW;
- with low enriched fuel, with uranium-235 enrichment below 20% in accordance with the current requirements of the security of transportation and storage of nuclear and other radioactive materials which are vulnerable to theft by terrorists;
- with ten vertical and seven horizontal experimental channels which will supply maximal fast neutron flux about 3. 10^{12} n/cm^{2}s, and maximal thermal flux about 8.10^{12} n/cm^{2}s;
- with channel which will supply epithermal neutron flux about 0,9.10^9 n/cm^{2}s suitable for medical Boron Neutron Capture Therapy (BNCT) application.

The INRNE together with the Technical University in Sofia have proposed to the Ministry of Education a new programme for education of students in nuclear energy. The Nuclear Energy course will be obligatory for obtaining the Master of Science Degree of the Technical University in Sofia. The educational classes refer: types of research reactors, main characteristics and design of the reconstructed IRT, safety assuring and licensing, reactor physics and thermo-hydraulic characteristics determination, accident analyses, fresh and spent fuel management, radioactive waste management and governmental categorization norms and rules. Acquaintance with calculational codes as the MCNP code for neutron transport and criticality calculations, WIMS-ANL code – for preparing of neutron cross sections for diffusion calculation, REBUS code for the fuel burn depth calculation, SCALE code system for spent fuel transport and storage devices safety assessment, PLTEMP/ANL code for calculation of thermo-hydraulic steady-state, and RELAP5 code – for transient operation, etc. is planned too. Preliminary acquaintance with the neutron activation analysis and BNCT is included in the educational programme.

The classes will be held in the INRNE and the reconstructed IRT will be used for carrying out specific training exercises on the reactor: reactor start, manual and automatically control, control rod calibration, delayed neutron group measurements, sub-critical multiplication/shutdown margin measurements, excess reactivity and shutdown margin measurements; reactor-physics measurements of static and kinetic reactor parameters, reactor dosimetry, measurements of the spent fuel characteristics in the hot laboratory, radiological characterization survey - alfa, beta and gamma measurement techniques, contamination measurement, etc.

The reconstruction of the IRT includes an arragement for a BNCT facility. Preliminary neutron transport calculations for BNCT channel regarding the geometry and material composition design have been carried out (Fig.1). Feasibility studies within the national network of the Medical University in Sofia, the National Centre of Radiobiology and Radiation Protection, the Institute of Experimental Pathology and Parasitology and Institute of Electronics of the Bulgarian Academy of Sciences, and the Faculty of Physics of Sofia University are carried out. Contacts with institutes, experienced in BNCT as EC JRC, Petten, the Netherlands, VTT, Finland and NRI-Rez, the Czech Republic, were established. Human, social and economical results due to the BNCT for patients from Balkan region are expected.

Besides the financial support of the Bulgarian government the IRT has the IAEA support through the project BUL/4/014 “Refurbishment of the Research Reactor” and the support of the US Department of Energy in the frame of the RERTR program.

The reconstructed IRT is a basis for keeping up specialists with researcher’s approach and skills who are able to give adequate responses to the challenges of complex modern technologies and the associated environmental problems. The reactor will be used for production of isotopes needed for medical therapy and diagnostics; it will be the neutron source in element activation analysis having a number of applications in industrial production, medicine, chemistry, criminology, etc.

Nuclear energy has a strategic place within the structure of the country’s energy system. A new nuclear power plant Belene with two reactors of 1000 MeV will be built. The extremely high requirements regarding nuclear safety call for the availability of scientific and technical potential, and for an adequate culture of safe use of nuclear energy. The acquired scientific experience and qualification in reactor operation is a basis for participation of the country in the international cooperation within the European structures. In that aspect, the operation and use of the IRT brings economic benefits for the country.

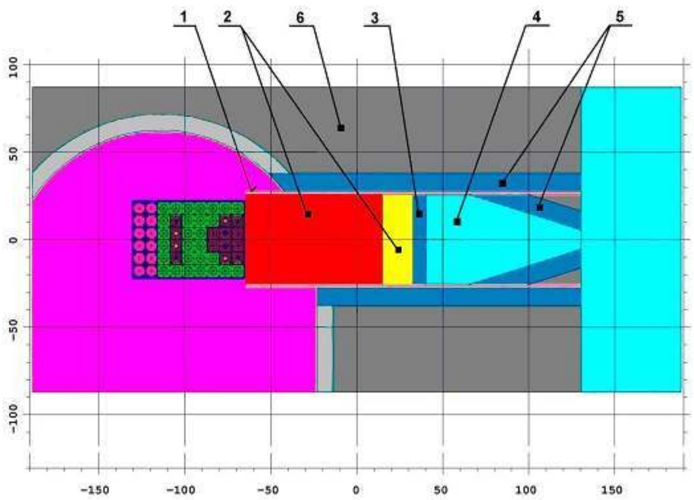

Figure 1. The BNCT beam tube model: 1. Vessel of Channel; 2. Filter 3. Lead Shielding; 4. Collimator; 5. Lead Shielding of Channel; 6. Concrete.

Utilization of the LVR-15 Research Reactor at Řež

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The LVR-15 research reactor commenced operation in 1957 as a multipurpose source of neutrons for basic research at horizontal channels and user-oriented research at mostly vertical loop channels and rigs as well.

Since 1957 the reactor has undergone two reconstructions. During the last one in 1989 all the reactor components and systems were replaced, including the reactor vessel. The LVR-15 is a tank type reactor (Fig. 1) and currently uses IRT-2M fuel of 36 wt.% ^{235}U enrichment manufactured by the NZCHK Company in Novosibirsk, Russia. The fuel features limit the output reactor power to 10 MW. The thermal and fast neutron flux reach up to 1.5 x 10^{18} n/m^{2}s and 2.5 x 10^{18} n/m^{2}s, respectively. Due to the nature of the reactor use, the reactor working cycle is 21 days and the number of the cycles is 8-10 per year.

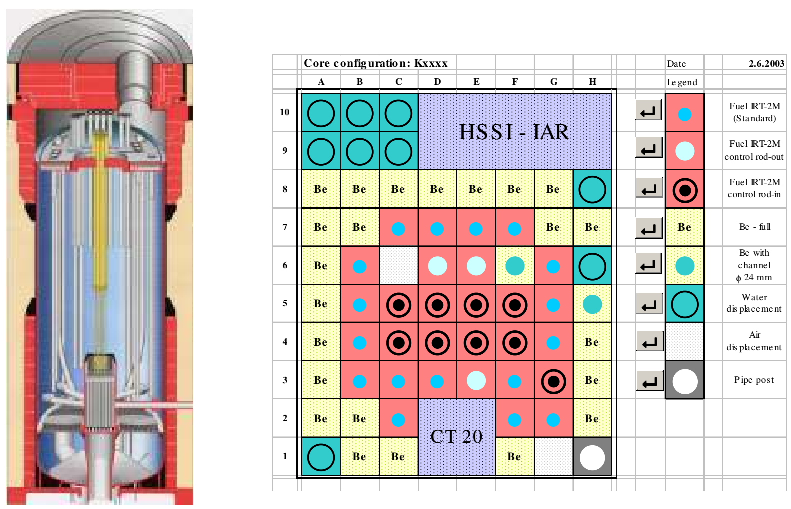

Fig. 1 LVR-15 research reactor; example of LVR-15 core (axial cross section)

Reactor Use For Material Research

The advantage of the reactor arrangement results from flexible diameter of irradiation channels, good access to the upper parts of the channels, and the fact that the core can be refueled without outage of the irradiation facility. The main fields of the reactor utilization are neutron beam research including BNCT, fuel and material irradiation tests, and radioisotopes and silicon production. LVR-15 special reactor features in the field of material research can be summarized:

- Core and irradiation channel size flexibility
- Irradiation rigs for irradiation of small (ring, tensile) to large (1CT, 2CT) specimens
- Five big loops with specialized mechanically loaded or heated irradiation channels
- Water chemistry and dosimetry control ensuring the conditions in testing facilities to be as close as to the conditions in power plants.

Other, more limited uses are in the areas of medicinal and industrial radioisotope manufacturing, production of radiation doped silicon and development of boron neutron capture therapy.

The reactor is equipped with hot cells for post-irradiation sample manipulation, disassembling and assembling of core channels.

The LVR-15 reactor specializes – due to its output and achievable neutron fluxes – in the study of combined effects of radiation and ambient media on materials. The reactor is equipped with experimental facilities such as loops and rigs, which permit exposure under conditions corresponding to those in power reactors. The generally utilized procedure is that the material is pre-irradiated in rigs and then is further exposed in loops enabling also the simulation of the thermal flux or physical stresses.

Irradiation rigs permit the exposure starting from small samples (ring, tensile) up to very large samples (1CT, 2CT). Total five loops simulating either PWR or BWR conditions in various irradiation channels and other specialized facilities are in operation in the reactor.

Fuel cycle

Reactor has joined the Russian Research Reactor Fuel Return (RRRFR) initiative to be converted from HEU to LEU. The present fresh fuel stock consists of 86 fuel elements that are sufficient for the reactor operation till the end of 2010 year. The spent fuel FA's are stored either in the reactor pools (113 pieces of IRT-2M, 36 wt.%, 80wt.%)or in the NRI interim storage (240 pieces of IRT-2M, 80 wt.%) or in dry barrels (206 pieces of EK-10, 10 wt.%). At present the transportation of spent fuel to Russia is being prepared. Spent fuel will be shipped to Russia in 9/2007 in the VPVR casks (Fig.2) that were designed for all the Russian fuel types used in the LVR-15 similar research reactors of the Russian origin round the world. The cask is licensed in the Czech Republic and Russia as well.

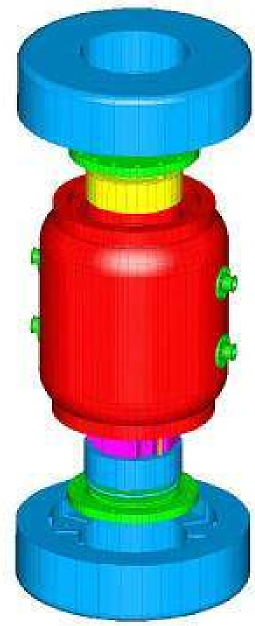

Fig.2 SKODA VPVR/M cask

Conclusions

The LVR-15 reactor is an important facility, which supports and contributes to research of nuclear materials and water chemistry. Experience that has been achieved operating the reactor during the last 50 years can be now transferred to the new irradiation facility designs including those which performs the research for Generation IV reactors, e.g. such as reactors cooled by high-temperature helium or water with supercritical parameters.

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Utilization of the VR-1 Training Reactor

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The Paper presents basic information about utilization of the training reactor at Faculty of Nuclear Sciences and Physical Engineering, Czech Technical University in Prague, signed VR-1 and named VRABEC (which means "Sparrow"). The reactor has been used very efficiently especially for education of university students and specialists in favour of the Czech nuclear programme for more than 15 years. It is the only reactor of this type in the Czech Republic. Therefore, students from several Czech technical universities and also from universities in Central Europe participate on its use. The operator and main user of the VR-1 Reactor is the Czech Technical University in Prague.

The VR-1 Reactor is well equipped for education and training not only by the experimental facility itself but also by carefully developed training methods. These are divided into several basic areas. Typical examples of them are as follows:

- Nuclear reactor control (start-up, operation, power changes, shut down),
- Reaching the critical state and critical parameters measurement,
- Dynamic experiments (periodical reactivity changes, delayed neutrons examination, examination of "bubbly boiling" influence on reactivity),
- Reactivity measurement and control rods calibration,
- Measurement of spatial distribution of neutron flux density,
- Neutron detection,
- Simulation of selected operational states of WWER type of power reactors,
- Neutron activation analysis.

The education experiments can be combined into training courses attended by students according to their study specialization. The methods enable to choose an appropriate level of student participation in task completion and result evaluation. The training programme of university students at VR-1 Reactor covers overall information on nuclear safety, radiation protection, emergency preparedness, and physical protection principles.

Every year, approximately 250 university students undergo training at VR-1 Reactor. Their stay at reactor site means an enormous benefit for their study process. The Czech Republic has a well-developed system of university nuclear education. The technical universities are cooperating in CENEN association (See Figure) In addition to the education of university students, the utilization of the VR-1 Reactor covers selected parts of training of specialists under the Czech nuclear programme and it also covers a supplemental research programme. The research programme is limited by a relatively small reactor power, however, a large variety of experiments and tests is still possible to be performed (from neutron detector calibration to verification of selected components developed for the transmutation technologies). The visits to the VR-1 Reactor are very popular. Part of the visit is a performance of reactor operation.

Overall experience with VR-1 Reactor utilization is very good. There is a big call for reactor training. Detailed instructions and report forms for record and evaluation of measured values are available for every exercise. The forms usually exist in digital shape. The operation organization and reactor utilization is ruled by safety culture principles.

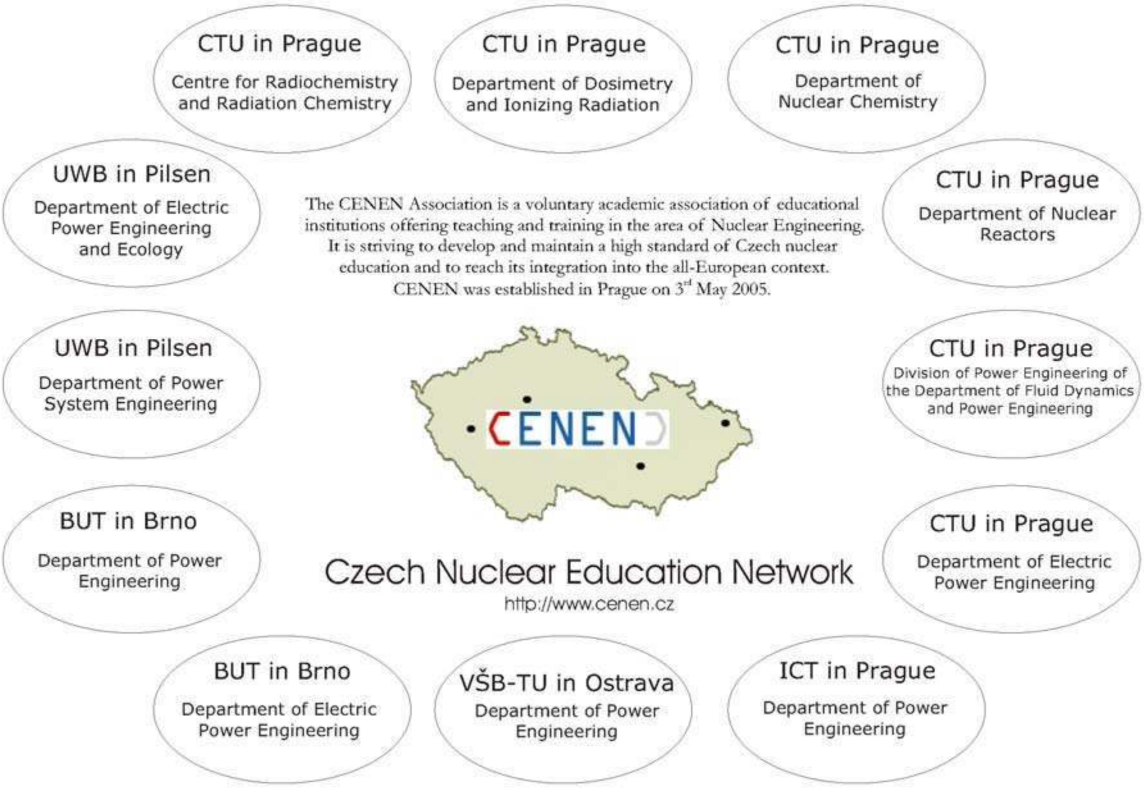

Figure: A Scheme Diagram of Cooperation in CENEN (Czech Nuclear Education Network) Framework.

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The 250 kW FiR 1 TRIGA Research Reactor - nal Role in Boron Neutron Capture Therapy (BNCT) and Regional Role in Isotope Production, Education and Training

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The Finnish Triga reactor, FiR 1, has been in operation since March 1962. The reactor has been delivered by General Atomics, USA. The reactor belonged first to the Department of Technical Physics at the Helsinki University of Technology. The activities of the reactor were defined as training in nuclear technology, research and production of radioactive isotopes. In 1972 the reactor was transferred under the administration of the Technical Research Centre of Finland (VTT).

From its early days the reactor created versatile research to support both the national nuclear program as well as generally the industry and health care sector. The volume of neutron activation analysis was impressive in the 70's and 80's when the reactor was operated close to daily only for activation analysis. Now other analysis methods have nearly totally replaced neutron activation analysis. In isotope production a small research reactor is competitive only in producing short lived isotopes for local markets.

Boron neutron capture therapy (BNCT) treatments dominate the current utilization of the reactor: three days per week are for BNCT purposes and the rest for other purposes such as isotope production, neutron activation analysis and education.

In the 1990's a BNCT treatment facility was build at the FiR 1 reactor. A special new neutron moderator material Fluental™ ($Al+AlF_3+Li$) developed at VTT ensures the superior quality of the neutron beam. Also the treatment environment is of world top quality after a major renovation of the whole reactor building in 1997. The FiR 1 reactor has proven to be a reliable neutron source for the BNCT treatments; no patient irradiations have been cancelled because of a malfunction of the reactor. Over one hundred patient irradiations have been performed at FiR 1 since May 1999, when the license for patient treatment was granted to the responsible BNCT treatment organization, Boneca Corporation. Currently three clinical trial protocols for tumours in the brain as well as in the head and neck region are recruiting patients.

FiR 1 reactor has made already now a major contribution in the research and development of BNCT. Significant breakthroughs have been achieved in the application of BNCT for cancer treatment. FiR 1 has an important international role in the development of boron neutron capture therapy for cancer as it is one of the few facilities in the world providing this kind of treatments. The successes in the BNCT development have now created a demand for these treatments, although they are given on an experimental basis.

Due to the BNCT project FiR 1 has become an important research and education unit for medical physics. Since the early 1990's several graduate and postgraduate students from the medical physics program of the University of Helsinki have been working at the FiR 1 BNCT facility. In research projects funded by the Finnish Academy, the Finnish Center for Technology Development (Tekes) and the EU the dosimetry, radiation transport modelling, treatment planning, prompt-gamma imaging and other medical physics aspects of the BNCT have been studied and developed. Over ten academic theses and dissertations have been produced in these projects, along with over hundred scientific publications. The research has been performed in close international collaboration with European, American as well as Japanese researchers. Also at Helsinki University of technology masters theses works have been made in connection to the BNCT research at FiR 1. The students aiming at the hospital physicist exam credit up to one year of required hands-on experience when working at the FiR 1 BNCT facility.

In addition to BNCT FiR 1 continues to support research in radiopharmaceutical applications. In the 1980´s FiR 1 had a major role in the Finnish radiopharmaceutical research and development. The spin-off company established at the reactor at that time, MAP Medical Technologies, has been successful but relies now on other sources for its radioisotopes, the accelerator at the Jyväskylä University in Finland and traditional international isotope producers. Now FiR 1 is utilised by researchers at the University of Helsinki for studying the use of samarium as a tracer in pharmaceutical formulation development.

Education and training play also a role at FiR 1 in the form of university courses and training of nuclear industry personnel. Helsinki University of Technology has a right to use the reactor for its purposes. Yearly there are at least two courses for technical physics and energy technology students in reactor and neutron physics that utilize the reactor and the BNCT facility. Reactor physics demonstrations are also organized for the students of the Lappeenranta Technical University.

FiR 1 is utilized also in the continuing education and training of the personnel at nuclear power companies, both in Finland and in Sweden, and other organisations connected to nuclear power. These are typically one day intensive courses with hands on exercises, or demonstrations and excursions in connection to longer lecture courses.

The main part of the working time has been reserved for the work at the BNCT irradiation facility. Still we have also for 20% - 40% of the time the possibility to irradiate samples either to produce some short-lived isotopes or to do neutron activation analysis. Main isotopes for tracer studies produced in the reactor are ^{24}Na, ^{82}Br and ^{140}La. The incomes from the production of isotopes are 15 % to 20 % of the turnover.

The reactor is operated by four reactor operators and five shift supervisors. All of them are part time operators or supervisors in addition to their work as research scientists or research engineers. This amount of operators and supervisors ensures that the reactor is easy to keep in operation during normal working hours and also during exceptional hours.

Chemical characterisation of early fine-ware pottery by neutron activation analysis: analytical and statistical approaches to production and trade

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Neutron activation analysis (NAA) is a well-established method for the chemical analysis of ancient pottery, particularly in the pursuit of provenance studies. The significance of NAA in such research is its eminent suitability in the measurement of those trace elements which are characteristic in geochemical terms of clay sources used in ceramic production. Since the chemical variation between different clay sources generally can be considered to be larger than natural variation within a particular clay source, pottery production sites, or even workshops that use a specific clay paste, can be identified by a so-called 'chemical fingerprint'.

At the N.C.S.R. "Demokritos", NAA has been used for more than 30 years for the routine analysis of ceramics. Approximately 130 mg, exactly weighed, of the powdered and dried sample are placed in a polyethylene vial, which is then heat-sealed. The vials are irradiated for 45 min in the "Demokritos" swimming pool reactor, at a thermal neutron flux of about $6{\cdot}10^{13} n{\cdot}cm^{-2}s^{-1}$, in batches of 10. Usually two batches are irradiated together, each of them including eight samples with unknown composition and two standard reference materials (SRM). Seven days after irradiation the vials are placed in the sample changer and counted for one hour with a Ge γ-detector covering the energy range of 80-1600 keV. The concentrations of As, Ca, K, La, Lu, Na, Sb, Sm, U and Yb are determined by this count. Three weeks after irradiation, samples and standards are counted again for two hours each and the concentrations of Ba, Ce, Co, Cr, Cs, Eu, Fe, Hf, Nd, Ni, Rb, Sc, Ta, Tb, Th, , Zn and Zr are determined. The principle standard employed is SOIL-7, issued by the International Atomic Energy Agency. Calibration tests have been carried out against SL-1 (IAEA), SRM 679 (brick clay-NIST), SRM 2711 (Montana soil- NIST) and in-house standards used in other laboratories.

The applications and potentials of NAA in archaeological ceramic studies are demonstrated by two case studies recently completed at N.C.S.R. "Demokritos": those of Late Neolithic 'black-on-red' pottery and Early Bronze Age 'sauceboats', both from the prehistoric Aegean. We highlight these case studies as they involve the detection of provenance and trade of fine ware pottery, which is difficult to address with mineralogical techniques. Fine pottery is relatively uncommon in these Neolithic and Early Bronze Age assemblages, and finding their origin is a matter of priority.

'Black-on-red' painted pottery is one of the most characteristic ceramic types, which appears during the final phases of the Late Neolithic in Northern Greece. The largest quantities of this kind of pottery have been found in Eastern Macedonia and Thrace. However, significant amounts have also been excavated in Central and West Macedonia, but also, in lesser amounts, in the upper Strymon Valley, in South-west Bulgaria, and in Thessaly. The importance of this ware has been appreciated by the researchers of the Neolithic Period in the Aegean, using it as a diagnostic element for the relative dating, as well as for the cultural attribution of different ceramic assemblages. The purpose of our study was to search for production patterns for this ceramic group. More specifically, to examine questions concerning the degree of standardisation in production, the location of production centres, indications of the scale of production, along with the differentiation of ceramic recipes within the group, which could reflect variation (geographic or other) in pottery traditions. Altogether 198 samples were selected for NAA in an attempt to answer the questions produced by the stylistic study of this material. Statistical evaluation of the chemical compositions of the 'black-on-red' pottery samples analysed from Eastern Macedonia resulted in the identification of four distinct groups, each of which represents a distinct production area. The pottery production within each of the four geographical areas had common technological and, in a broader sense, stylistic characteristics. It was found that the products of individual production centres were exchanged beyond their region of production and that the products of one particular centre were found as widely distributed as the island of Thasos and the plain of Thessaly. This demonstrates not only that these early fine wares were produced in separate centres which shared technological characteristics, but also that they actively participated in extensive exchange systems.

The second case study concerns specialised shapes from the Early Bronze Age of the Aegean, known as 'sauceboats'. These distinctive vessel forms appear during the middle of the third millennium BC, as a crucial component of drinking sets which accompanied the introduction in the Aegean of true, flat-based 'table-wares'. Decorated in three broad ways: dark-on-light painted, yellow-blue mottled and a glossy black slip known as 'urfirnis', the different sauceboat types have been attributed to two main areas of *production*: the Helladic mainland (in the case of the urfirnis and yellow/blue mottled) and the Cycladic islands (in the case of the dark-on-light painted). A large number of sauceboat samples were taken from various sites on the mainland, throughout the Cyclades, on the Asia Minor coast and from the island of Crete. NAA was able to distinguish a number of chemical groups which represented the widely distributed products of specific centres of production. The picture was far more complex than anticipated, with the production of urfirnis sauceboats on both the Cycladic island of Melos and on Crete. While the products of a West Cretan centre have a limited distribution, those manufactured on Melos were found on the Asia Minor coast, as well as Northern Crete and throughout the islands. Yellow-blue mottled sauceboats from the area of Attika on the mainland and the island of Kea were also found to share their centre of production. In this case, NAA demonstrates its suitability to provide detailed information about the origin of important fine wares, whose mineralogical study had proved problematic.

We use these examples to demonstrate the effectiveness of NAA in the study of fineware pottery groups, making an important contribution to ceramic studies through their characterisation according to trace element concentrations, which prove far more distinct than petrographic or mineralogical. In this effort, compared to other chemical analytical techniques applied to ceramics, NAA still provides advantages in terms of sample preparation, precision and accuracy.

FEASIBILITY STUDY OF ^{125}I BRACHYTHERAPY IN INDONESIA[1]

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ABSTRACT

Cancer is a term for diseases in which abnormal cells divide without control [(1)]. Many cancer treatments such as chemotherapy or immunotherapy, as well as radiotherapy, are intended to kill tumor cells. Brachytherapy can be performed by using tiny titanium cylinders which contain a small amount of radioactive material such as radioisotopes I^{125}. These seeds are used as implants for prostate cancer or breast cancer. This Study is to develop and produce innovative radioisotopic products as brachytherapy seeds.
Research reactor in Serpong Indonesia has been producing radioisotopes I^{125} from the mid year 2005 through 4 time irradiation. From these radioisotopes the research would continue to produce I^{125} brachystherapy. This study is focused on the technology producing of I^{125} brachystherapy production, for Indonesian people who has a cancer.

BACKGROUND

1. Research reactor in Serpong has been producing radioisotopes I^{125}
2. Brachytherapy I^{125} has low risk of radiation exposure for the pasien, health care workers and those around them [(1)]

CANCER CASES IN INDONESIA

The statistic from Health Department of Indonesian Government showed that 13,2 million peoples or 6 % of Indonesian population 2005 (around 241.973.879) has cancer desease. 65 % is too late to go to the hospital for diagnosis and treatment. A number of them frightened of operations. The cancer desease ranks fifth in the list of mortality after Hearth attack, Stroke, Lung and Diarrhea. The relative frequency of cancer desease in Indonesia is 23,66 % in age group of less than 40 years old and 22,7% in that of more than 45 years olds. [(2)]

TECHNOLOGY OF BRACHYTHERAPY I^{125} PRODUCTION

Research Reactor in serpong has been producing radioisotopes I^{125} from the mid year 2005 through 4 time irradiation of ^{124} Xe gas. From these radioisotopes the research would continue to produce I^{125} brachystherapy.

TABEL 1: The Activity of ^{125}I Product in 2005

No	dates	Activities
1	13 June 2005	9.5 Ci
2	5 July 2005	9.8 Ci
3	25 July 2005	17.2 Ci
4	12 December 2005	9.4 Ci

Brachytherapy is internal radiation therapy using an implant of radioactive material such as ^{125}I, ^{103}Pd and ^{90}Y placed directly into or near the tumor. The size of this brachytherapy is 4.5 mm long and 0.8 mm in diameter.

The benefit of placing sources near or inside the tumor is that they are able to deliver a cell-killing radiation dose to the tumor while sparing the healthy surrounding tissue. In terms of efficacy, cost and maintaining the patient's quality of life.

Technology of I^{125} brachytherapy production consists of two methods. First, Ion implantation to manufacture the radioisotopes I^{125} provides the seed, this method is more difficult than the second because this seeds are tiny and the dispensing of radioisotopes I^{125} to the seeds is buried has to be done the hotcell.

In the second methods pure isotopes of non-radioactive Xenon-124 gas beneath the surface of the seed core. This method provides safety to the operator and easy for dispensing but leaking analysis of gas before irradiation must be done.

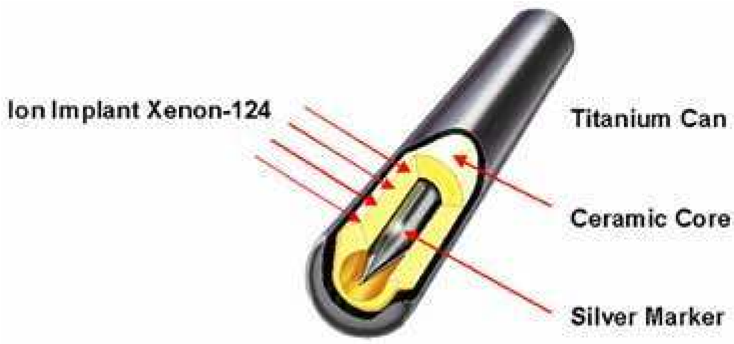

Figure 1: brachystherapy seeds [3]

CONCLUSION

The research for producing I^{125} brachytherapy needs to bee continued to decrease people fear for operation. The I^{125} brachytherapy enables patient to leave hospital on the same day as that of admission.

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Characterization of Airborne Particulate Matter at Urban and Rural Area in Bandung and Lembang Indonesia using Instrumental Neutron Activation Analysis

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Samples of fine and coarse fractions of airborne particulate matter were collected twice a week for 24 hours in Center of Nuclear Technology for Materials and Radiometry Bandung (urban) and Meteorological and Geophysical Agency station Lembang (rural) from January 2004 to December 2004. The samples were collected using a Gent stacked filter sampler in two fractions of < 2.5 µm fine and 2.5 – 10 µm coarse sizes. The samples and synthetic standards were analyzed for elemental concentrations by instrumental neutron activation analysis (INAA) at Bandung TRIGA 2000 reactor. Black carbon concentrations were determined using Smoke Stain Reflectometer. The synthetic standards were prepared by pipetting 100 uL of the solution containing one or more elements into a polyethylene vial. Irradiations were carried out under two experimental conditions. Short irradiation of 5 min at the pneumatic transfer tube with thermal neutron flux of approximately 10^{13} n cm^{-2} s^{-1}were used to determine the short half-life elements, cooled for approximately 60 s, and counted for 300s (live time) in the Radiometry Analysis Technique Laboratory. For determining the medium-long half-life elements, the samples were then irradiated again in the fixed irradiation system with a neutron flux of 10^{13} n cm^{-2} s^{-1} for 48 – 60 hours, cooled for 1 - 2 days and counted for 3000s or more depending on their activities by a gamma spectrometer with coaxial detector coupled to Integrated Signal Processor and System MCA Card both from APTEC. The identification of radioisotopes was carried out by characterizing its half-life and its gamma-ray energies.

The NIST's standard reference material SRM 1648 airborne particulate matter were analyzed in the same experimental conditions used in the sample analysis as method validation to evaluate the precision and accuracy of the results. The result of SRM 1648 analyses mostly have good agreement with the value quoted in the NIST's certificate (see Table I). The radioisotopes measured in this study were related to 22 elements as follows : C, Na, Al, V, Mn, Br, Cl, I, Cr, Fe, Zn, Sc, Sb, Co, La, Mg, Sm, K, Ca, Ti, As, and Cs. (*see FIG.1*). Most of the element concentrations in Bandung (urban) are higher than in Lembang (rural). These data of elements can be used to characterization of pollutant sources by correlating the relationship among these elements, such as Al, Ca and Ti for soil, black carbon and K for wood burning source. The correlation between those crustal elements Al against Ti and Ca of coarse fractions both at Bandung and Lembang show how well these elements correlated to each other and demonstrates that they are basically related to the one source, soil. The black carbon versus Br of fine faction at Bandung plot also show good correlation, that they are related to motor vehicles source. This result associated with the number of vehicles registered in Bandung approximately increased 30% compared to the last two years.

Table I. Elemental concentration obtained for NIST SRM 1648

Element	Unit	This work $X^a \pm SD^b$	Certificate value (NIST value)[c]
Al	%	3.43 ± 0.12	3.42 ± 0.11
As	mg/kg	124 ± 8.5	115 ± 10
Br	mg/kg	445 ± 14	500
Cl	%	0.45 ± 0.03	0.45
Co	mg/kg	18 ± 2.4	18
Cr	mg/kg	378 ± 84	403 ± 12
Cs	mg/kg	3.67 ± 0.13	3
Fe	%	3.98 ± 0.38	3.91 ± 0.10
I	mg/kg	19 ± 2	20
La	mg/kg	37 ± 6	42
Mg	%	1.3 ± 0.3	0.8
Mn	mg/kg	788 ± 27	786 ± 17
Na	%	0.403 ± 0.031	0.425 ± 0.002
Sb	mg/kg	45.8 ± 2.2	45
Sc	mg/kg	6.74 ± 0.28	7
Sm	mg/kg	4.5 ± 0.85	4.4
Ti	%	0.41 ± 0.05	0.40
V	mg/kg	126 ± 7	127 ± 7
Zn	%	0.418 ± 0.024	0.476 ± 0.014

[a] mean value
[b] one standard deviation
[c] NIST does not provide uncertainties for uncertified elements

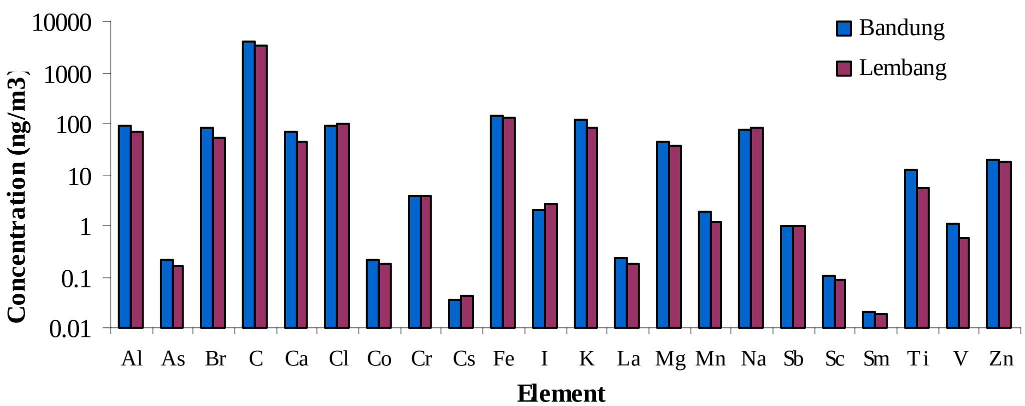

FIG.1. Annual average of elemental concentration of APM in Bandung and Lembang

INAA as one of nuclear techniques utilizing research reactor provides a very good, simultaneous, multi-elemental analysis methods for airborne particulate studies. In particular, INAA is known to be a reliable analysis technique with low detection limits and effective for a large number of samples. The ability to provide elemental concentration information on all of the most significant elements enables statistical techniques to be applied in the data and determine the source contribution of the pollutant. Analysis the airborne particulate samples collected in Bandung and Lembang were reported to demonstrate the advantage of INAA method, especially to get a better understanding about the condition of atmosphere in Indonesia due to air pollution.

Characterization of a Neutron Collimator for Neutron Radiography Applications.

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TRIGA Mark II reactor of ENEA's Casaccia research Center (in Italy named RC-1) reached first criticality in 1960 and, after an upgrade completed in 1967, its power was increased till 1 MW. At this power level it is still running, mainly for short mean life time radioisotopes production (for medical purposes) and neutron radiography. In the year 2003 started a program for a new neutron radiography/tomography facility. This facility will utilize the "Tangential Channel" of the Triga RC-1 reactor. After some preliminary studies and the design, a suitable collimator was installed in the reactor channel.
This paper will show the technical description of the facility, some theoretical studies on the collimator design and the experimental characterization of the collimator. Several configurations of the internal filters were arranged and analyzed experimentally.
For the collimator characterization, the Kobayashi method [1] was used in order to evaluate the L/D ratio, the geometrical resolution and the divergence angle. A map of the spatial distribution of the neutron flux, γ-ratio and Cadmium-ratio was also analyzed and optimized.
A neutron collimator is a device that focalize the neutron beam towards an imaging plane "modifying" the neutron emission from the neutron source into a parallel or slightly divergent beam. Usually, a neutron beam, could be classified in three different categories:

- Radial : when the collimator starts radially from the neutron source;
- Tangential : when the collimator lies tangentially from the neutron source;
- Beam Guide : when the collimator, starting from the neutron source, has a complex shape that "drives" the neutron beam.

The facility installed in the Triga RC-1 Reactor is a Slightly Divergent Tangential Collimator (see Fig.1).

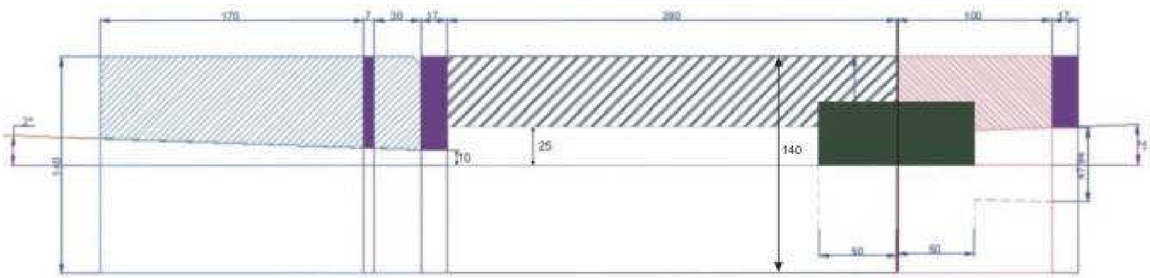

Fig. 1. Triga RC-1 Tangential Channel Collimator

The main feature of this particular type of device is that the γ emission from the beam is less than in the case of a similar collimator installed radially from the core of the reactor.
The main design parameters of such a type of collimator are: filters (γ/n), neutron entry hole diameter, absorbing walls and filling gas. The main features of the device (depending from the optimization of the design parameters) are: Neutron Flux, Cadmium Ratio, γ/n Ratio, Effective Beam Diameter and L/D Ratio, Background Components.

The Neutron Beam/Collimator characterization was performed by a campaign of γ/n measures: Space/Energy distribution measurements by Activation Foils, Cadmium Ratio Measurements by Activation Foils and Cadmium Capsules, γ measurements by TLD devices and L/D Ratio measurements by Special Equipment and Imaging Plates (Kobayashi Method). The spatial distribution of the neutron flux was performed using some gold foils and the optimization of the channel performances was done by analyzing three different conditions: with or without a Bismute filter or with a Graphite filter. After the optimization of the Neutron Flux, the collimator was characterized using the Kobayashi Method in order to define the L/D Ratio and the Effective Beam Diameter. In Fig 2 is shown the special device (2 aluminum plates with a Cadmium plate inside) used to perform the measure.

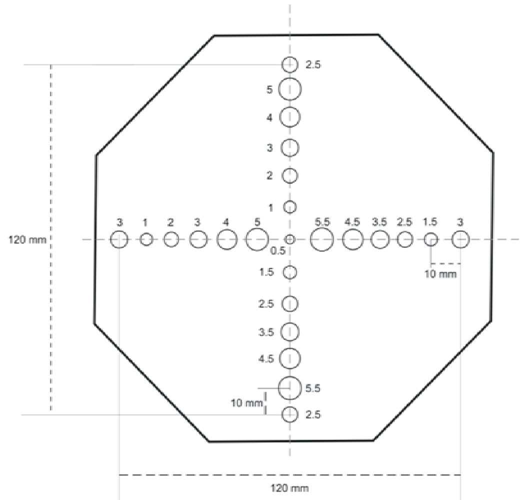

Fig. 2. Kobayashi Method Sample Device

The image of the Sample Device, projected on an Imaging Plate placed at a proper distance from it, was analyzed and the calculations from which the L/D Parameter is derived, are reported and discussed.
The final design optimization, the future improvements of the collimator and the design of a new concept of a Neutron/γ shutter are reported and discussed too.

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Optimizing Conditions Suited for Stress Determinations in Q-Space Focusing Configurations

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During the last decade a new concept of high-resolution focusing configuration has been developed, using Q-space focusing effects, which proved to be an alternative to the existing conventional configuration.

If such a focusing configuration is to be used for stress determinations, special problems arise. As, for a given scattering angle, thin peaks are obtained only for a certain position of the sample, severe limitations appears in choosing the scattering angle or in amount of information possible to be obtained.

A convenient solution is to limit the dimensions of the sample zone for which strain determinations are realised. That means not only the use of corresponding diaphragms but also getting real-space focusing at sample position.

In conclusion if strain determinations are to be realised using Q-space focusing configuration both real space focusing at sample position and the phase-space focusing getting thin diffraction peaks must be obtained. The corresponding focusing conditions are deduced in this paper for three neutron diffractometer configurations: crystal diffractometer, 2 crystals diffractometer and time-of-flight diffractoemeter using steady-state neutron source. For this extended abstract only 1 crystal diffractoemeter configuration is considered.

1. Crystal neutron diffractometer

1.1 **1 crystal monochromator unit**

The considered configuration is given in fig.1. The Bragg constraints are given by:

$$\gamma_0 + \gamma_1 = \frac{2l_m}{R_m} sign\left(\theta_m + \chi_m\right) \tag{1}$$

The configuration geometry gives:

$$\begin{aligned} L_0\gamma_0 &= l_m \sin\left(\theta_m + \chi_m\right) - l_0 \\ L_1\gamma_1 &= l_s \cos\left(\theta_s + \chi_s\right) + l_m \sin\left(\theta_m - \chi_m\right) \\ L_2\gamma_2 &= l_d - l_s \cos\left(\theta_s - \chi_s\right) \end{aligned} \tag{2}$$

where θ_m and χ_m are the monochromator Bragg angle and the cutting angle respectively, l_i (i= 0, m, s, d) are the widths in the horizontal plane of the source, monochromator sample and detector respectively , R_m is the radius of curvature of the monochromator and Li (i=0, 1, 2) are the distances between source and monochromator, between monochromator and sample and between sample and detector respectively; γ_i are angular variables in the horizontal plane, representing deviations from the most probable values and sign(α) = abs(α)/α.

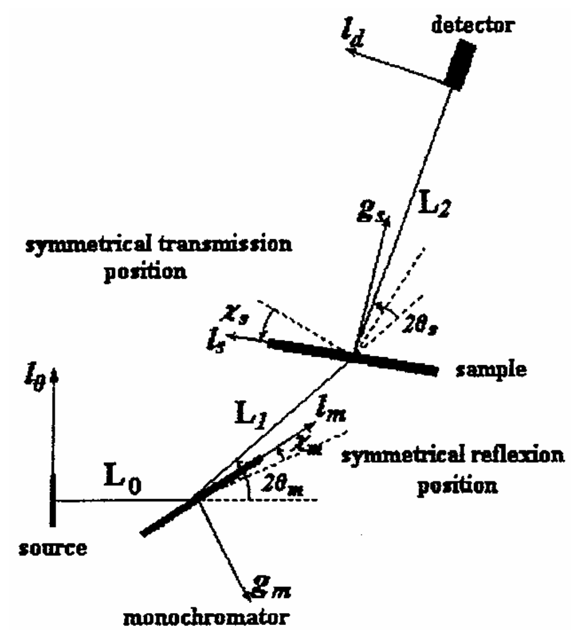

Fig.1 Experimental setup for an one crystal diffractometer

If l_{seff} is the effective sample width (the part of the sample irradiated by the neutrons) in the direction normal to the beam incident on the sample, using (1) and (2) one obtains:

$$l_{seff} = l_m \left[\frac{2L_1}{R_m} sign(\theta_m + \chi_m) - \sin(\theta_m - \chi_m) - \frac{L_1}{L_0} \sin(\theta_m + \chi_m) \right] + l_0 \frac{L_1}{L_0} \quad \textbf{(3)}$$

The minimum value of l_{seff}, i.e. the minimum width of the beam at the sample position, is obtained cancelling the l_m coefficient. It is obtained:

$$R_m = \frac{2L_1 sign(\theta_m + \chi_m)}{\sin(\theta_m - \chi_m) + \frac{L_1}{L_0}\sin(\theta_m + \chi_m)} \quad \textbf{(4)}$$

The relation (4) is the real space focusing condition giving the monochromator radius of curvature for which the contribution to the beam width at the sample position given by the monochromator length is compensated. If we want to have a thin diffraction line for the 2θ value of the scattering angle, the "Q space" focusing condition should be fulfilled, [1], [2], [3] also:

$$R_m = \frac{a}{2a-1} \frac{2L_1 sign(\theta_m + \chi_m)}{\sin(\theta_m - \chi_m)} \quad with\ a = -\frac{\tan\theta_s}{\tan\theta_m} \quad (5)$$

The value given by the (4) condition depends only on θ_m and χ_m while that given by (5) depends on θ_m , χ_m and θ_s. In order to have the best conditions for the stress measurements, both (4) and (5) have to be fulfilled. These two conditions are satisfied, for a given value of θ_m, only for a certain value of the scattering angle given by**:**

$$\tan\theta_s = \frac{\tan(\theta_m)}{1 - \frac{L_1}{L_0} \frac{\sin(\theta_m + \chi_m)}{\sin(\theta_m - \chi_m)}} \quad (6)$$

This work has been partially supported by IAEA Vienna under the contract ROM/13579.

SANS FACILITY AT THE PITESTI 14MW TRIGA REACTOR

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At the present time, an important not yet fully exploited potentiality is represented by the SANS instruments existent at lower power reactors and reactors in developing countries even if they are, generally, endowed with a simpler equipment and are characterized by the lack of infrastructure to maintain and repair high technology accessories. The application of SANS at lower power reactors and in developing countries nevertheless is possible in well selected topics where only a restricted Q range is required, when scattering power is expected to be sufficiently high or when the sample size can be increased at the expense of resolution.

The need for the installation of a new SANS facility at the Triga Reactor of the Institute of Nuclear Researches in Pitesti, Romania become actual especially after the shutting down of the VVRS Reactor from Bucharest.

Experimental SANS configurations suited for a steady state reactor

Any SANS configuration is formed by a monochromator unit, an analyser unit and a detecting system. In the most used instruments the analyser unit is a simple one given by the detecting system itself. The monochromator unit can by formed by one or two single crystal. For the two crystals system the spatial extension around the beam of the experimental setting is decreased significantly while the luminosity is lowered because of the second refection. A quite common configuration has a crystal monocromator and a crystal analyser the sample being positioned between monochromaror and analyser. The analyser crystal is rotated around an axis normal to the beam direction and a “rocking” curve I(θ) is recorded, representing the angular distribution os the scattered neutrons. For the case of a mechanical monochromator the monochromatic beam has the direction of the beam channel.

The selection of the optimum configuration; The selection criteria definition

For SANS the recorded curve is I(Q), $Q = k_i - k_f$. As the dimensions of the components are finite, a given number of counts correspond not to a certain value Q, but to Q+ΔQ. The value of Δθ is given by the overall collimation while Δλ is given only by the monochromator. We shall choose as selection criterion the value of the monochromatic beam intensity at sample corresponding to a given value of Δλ/λ. Another criterion is the instrument extension around the beam channel; the involved sum of money to realize the instrument is important as well.

For the case of crystal monocromator unit, that with one crystal is the most desirable though the instrument is somewhat extended from the beam channel axis. The SANS instrument with double crystals monochromator is significantly less extended while that with crystal monocromator and analyser allows for the very low Q values determinations. For both of them, because of the two reflections, the luminosity is rather poor. These instruments could be preferred for the case of large flux reactors as neutron source.

The SANS configuration with mechanical monochromator has significantly increased luminosity, in comparison with those having crystal monocromator units, while the spatial extension of the instrument is quite reasonable. Tacking account that TRIGA reactor existing at INR Pitesti is a medium flux reactor and that the available dimension for the SANS instrument is severely limited by the dimensions of the room where the instrument has to be installed, this experimental configuration has been chosen as the most suited for the situation existing in our institute. For usual collimation values of about 30 minutes, and for an inclination angle of the monochromator axis of about 2-3 degrees, $\Delta\lambda/\lambda$ is about 20-30%, i.e. quite reasonable value. sample width may be fixed between 10 mm and 20 mm. The minimum value of the scattering vector is $Q_{\min} = 0.005\ \overset{o}{A}^{-1}$ while the maximal value is $Q_{\max} = 0.5\ \overset{0}{A}^{-1}$. The relative error is $\Delta Q / Q_{\min} = 0.5$. In the case of our SANS instrument a monochromatic neutron beam with $1.5\overset{0}{A} \le \lambda \le 5\overset{0}{A}$ is produced by a mechanical velocity selector with helical slots. The distance between sample and detectors plane is (5.2 m).

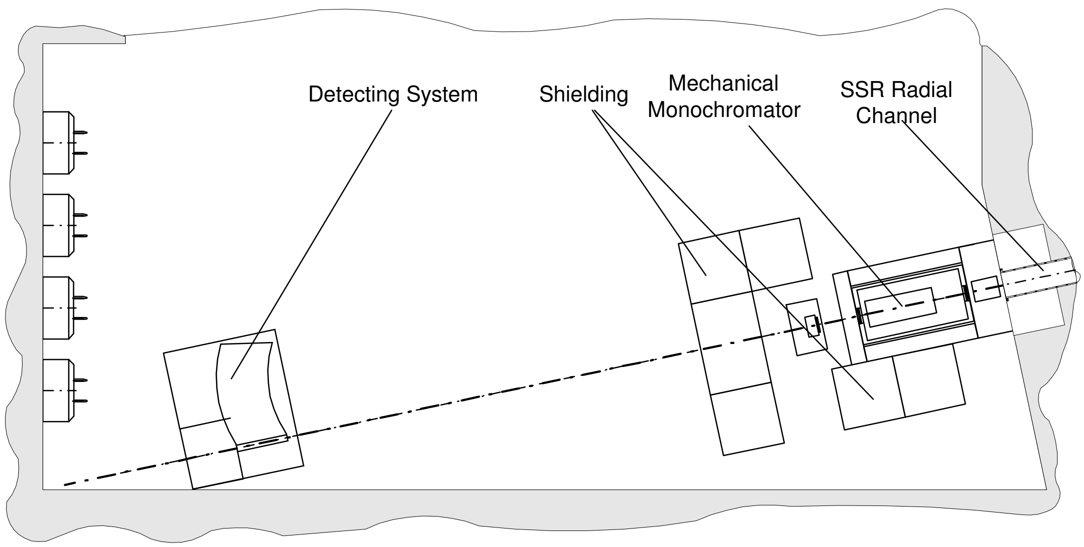

Fig.1 The instrument layout

The SANS instrument (fig. 1) has the following components: The mechanical monochromator, shielding, the Bi filter, the sample table and holder, the detecting system formed by two rows of 40 He^3 detectors each, 2 flux monitors, the beam stop. Paraffine blocks and recipients with water form the shielding. The flux monitors are positioned one before the sample (in front of the monocromator window) the other in front of the beam stop. The sample can be rotated using a step by step motor. A cadmium slit system actioned by a step by step motor allows for the determination of the monochromatic beam center at the beam stop position.

RIAR Capabilities in Support of the Innovative Nuclear Technologies

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The RIAR installations capabilities for provide for solving problems of nuclear power engineering development, safety ensuring, environmental acceptability, efficiency and effectiveness are described in the report.

RIAR reactor complex including high-flux research reactor SM-3, research fast reactor BOR-60, research material testing reactor MIR and water pool-cooled research reactor RBT (see table 1).

Table 1

Research reactors operated in RIAR

Name	Type	Power, MW	Reactor start-up	Refurbishment
SM-3	Vessel-type	100 (50)	1961	1974, 1992
MIR-M1	Channel-type	100	1966	1975
BOR-60	fast (liquid-metal)	60	1969	
RBT-6	Pool-type	6	1975	
RBT-10/1	Pool-type	10	1983	
RBT-10/2	Pool-type	10	1984	

All RIAR research nuclear reactors provided with experimental devices and in-pile examination techniques to solve pressing tasks of nuclear power engineering and reactor material science with regard to the reactors of different application and examinations in support of new technologies and engineering solutions for modern and advanced reactors of a new generation.

RIAR have got the unique material science complex also.

The Complex based on hot cells and it is the largest material science complex in Russia: in its 3 buildings there are about 50 shielded chambers and over 100 heavy and medium-weight shielded boxes. Fuel element and FA non-destructive examinations, some fuel and absorbing element fragments as well as other specimens with emergency parameters are studied in one building. Material science investigations on samples, fuel and absorbing element fragments cut out from the standard or pilot products with taking into consideration non-destructive examination results are carried out in another building. In the third building there are cells and shielding box chains equipped to manufacture the experimental and pilot fuel elements with different fuel compositions.

In the report the examples of RIAR international collaboration and proposals on development of international collaboration on the basis of the research nuclear reactors are described.

IAEA-CN-156/U-21

SM Reactor After Core Modernization

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International Conference on Research Reactors: "Safety Management and Efficient Usage"
Sidney, Australia, November 5-9, 2007

The modernization purpose was to make possible the irradiation testing of structure materials for power water fission and fusion reactors at a damage rate of ≥ 20 dpa, helium accumulation rate - > 1000 appm per year and fast neutron dose - 100 dpa and more. At the same time, conditions must be created for simultaneous irradiation of representative sample arrays in water environment with necessary chemical parameters at a temperature regulated in the range 100-300^{0}C and pressure up to 17 MPa by using irradiation rigs well-instrumented with devices for on-line parameters monitoring, and temperature and neutron spectrum control. Actually, it was necessary, in addition to the available irradiation positions, to provide places for loop and capsule channels of large diameters, which replaced some part of fuel without significant changes of the reactor design and operational procedures.

In the process of work on the modernization project a new fuel assembly design was developed that had an experimental channel ∅24.5mm in the fuel rods array, and the possibility was substantiated to place up to four such assemblies in the reactor core. Instead of two positions for fuel assemblies, two positions for loop or capsule channels ∅ 69mm were provided. Reactivity losses were compensated by the replacement of stainless steel wrappers with those of zirconium alloys and by increase of U^{235}content in the fuel rods by 20%. This required mastering the fabrication process of the modified fuel rod and experimental verification of its serviceability at a thermal flux density up to 15 MW/m^2 on its surface, as well as of the serviceability of three types of fuel assemblies with such fuel rods, including the assemblies with the experimental channels.

In the course of post irradiation examinations the data on fuel rod swelling as a function of burn-ups were obtained and deformation and failure temperatures at different burnups were determined. The fuel rods showed to be characterized by the sufficient serviceability under normal operation conditions and to have the necessary temperature margin before initiation of deformation and failure under overheating.

In January 2005, during the routine fuel reloading without reactor shutdown the standard fuel assemblies were replaced by the fuel assemblies with the increased U^{235} content. The reactivity characteristics of the reactor core were found to be slightly changed that is explained by the great effect of fuel self-screening. The fuel cycle parameters were noticeably ameliorated.

Fuel utilization performance before and after modernization

Parameter	Value of 2004	Value of 2005
Energy production, MW-d	21900	21600
Number of spent fuel assemblies	101	77
Fuel assembly consumption per 1000 MW-d	4.61	3.56
^{235}U mass in the spent fuel assemblies,kg	92.6	82.9

Characteristics of a new configuration of the core ensure the requirements for operation and experiment conditions.

Comparative core characteristics before and after modernization (experiment)

Parameter	Value before modernization	Value after modernization
Reactivity margin, β_{ef}	10.6±1.0	10.7±1.0
Control rod efficiency, β_{ef}	12.7±0.7	13.2±0.8
Temperature coefficient of reactivity under nominal conditions, β_{ef}/C^0	$-(1.8\pm1.0)\cdot10^{-2}$	$-(2.0\pm0.15)\cdot10^{-2}$
Rate of reactivity loss as a function of fuel burn-up, β_{ef}/MW-d	$(6.1\pm0.5)\cdot10^{-3}$	$(5.5\pm0.4)\cdot10^{-3}$
Power coefficient of reactivity for "hot", "poisoned" state, β_{ef}/MW-d	$-(4.0\pm1.3)\cdot10^{-3}$	$-(4.7\pm0.3)\cdot10^{-3}$

Simplicity and possibility to rapidly introduce the development results are accompanied by certain losses of neutron flux density in several reflector channels due to their shielding by the new ones loaded instead of fuel assemblies at the boundary between the core and reflector.

The investigations and developments performed on the second stage of the modernization include, along with other activities, the development of a new fuel rod with small parasitic neutron absorption and of a fuel assembly with rod-type burnable absorbers, as well as the core rearrangement that allows us not only to compensate the disadvantages of the first stage but to obtain additional advantages too.

Set of Investigations of HFR Fuel Rods in Justification of Their Serviceability and Safe Operation

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RIAR perform works on the upgrading of the SM high-flux research reactor core. The main objective of these works is to improve effectiveness of the reactor utilization by increasing a scope of experiments with a high density of neutrons intended for isotopes accumulation, irradiation of structural and other materials, as well as for the performance of different kinds of experiments. The objective may be achieved by the arrangement of irradiation channels within the SM reactor core. The resulting reactivity losses can be compensated in two ways: by increasing of ^{235}U loading into the fuel assemblies or using new fuel rods with a less cross-section of neutron absorption. At the initial upgrading stage it was decided to use standard fuel rod designs with the uranium loading increase by 20 percent in order to compensate the reactivity losses.

Standard fuel rods of the SM reactor are made on the basis of dispersion fuel composition UO_2 – Cu, while the fuel rod claddings are made of steel EI-847. The fuel rod section is cross-shaped. The SM standard fuel rods are successfully operated for 40 years. They have shown high serviceability and operation safety at a high density of the heat flux from the surface of 15 MW/m^2

However, current commissioning of the new or modified nuclear fuel provides for a performance of the required set of tests and investigations in justification of its serviceability. In this connection, RIAR implement a program for a study of serviceability and safe operation of the modified SM fuel rods with the increased uranium loading.

A set of works presented in this paper incorporates the results of calculations and investigations of the experimental fuel rods tested under specially performed reactor experiments with a simulation of a wide range of neutron-physical and thermal-physical irradiation parameters on the fuel rods. Simulated parameters of the reactor tests of the fuel rods changed within the following ranges: fission density ($4.42*10^{14}$ 1/cm^3 – $1.08*10^{15}$ 1/cm^3), thermal flux density (6.62 MW/m^2 – 15.7 MW/m^2), temperature (295ºC – 582ºC), and concentration of the fission products up to 1,1g FP/cm^3. Reactor test conditions different in temperatures and thermal loadings allowed to obtain detailed data on the behavior of fuel rods in general and of their components in particular.

The paper describes a connection between neutron-physical and thermal-physical parameters of the reactor tests and a change in the properties of the fuel rod components. The following regularities of the change in macro- and microstructures of the fuel column were revealed: further sintering and its influence on the porosity in the fuel particles and matrix, migration of voids and fuel particles in the fuel columns. Peculiarities of the radial and local swelling of the fuel rods, as well as the swelling at different fuel rod elevations, were revealed. Dependencies of the radial swelling on the accumulation of fission products, fission density, heat flux density and testing temperature were plotted. The dependencies allow for a qualitative evaluation of the fuel rod swelling at different stages of operation and different values of the thermal-physical parameters, as well as for a selection of safe parameters subject to the ultimate swelling of the fuel rods and technical conditions.

Utilization of Irradiation Holes in HANARO

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Since 1996, HANARO has been widely used for radioisotope production, material and fuel irradiation tests, beam application research and neutron activation analysis. Seven irradiation holes are provided in the core and twenty holes including two NTD(Neutron Transmutation Doping) holes, large hole and NAA holes are located in the reflector tank. The fuel and material irradiation tests that require long irradiation time are performed by using the seven holes where the forced circulation of core flow exists. But the targets to produce RI(Radio Isotope) with the short half life are mainly irradiated in the holes at reflector tank where the natural convection of the pool water is available.

In the initial stage of normal operation, the utilization of irradiation holes was not active because the development of capsules for irradiating material, RI targets and test fuel was delayed. The capsule for RI production at holes in reflector tank was developed easily because the coolant flows by natural convection. But in case of the capsules using holes in the core, a locking device should be provided to prevent an inadvertent removal during power operation and fixing device to reduce the vibration of the guide tube for instrument such as thermo couple and SPND by coolant flow. And it should be confirmed for any kind of irradiation tests to safely remove the heat induced by fission events or gamma heating during tests. The development of these capsules made the utilization of the irradiation holes vigorously. Especially the development of the instrument capsule has accelerated utilization of material and fuel irradiation. Figure 1 shows the status of utilization of the irradiation holes in the core by year.

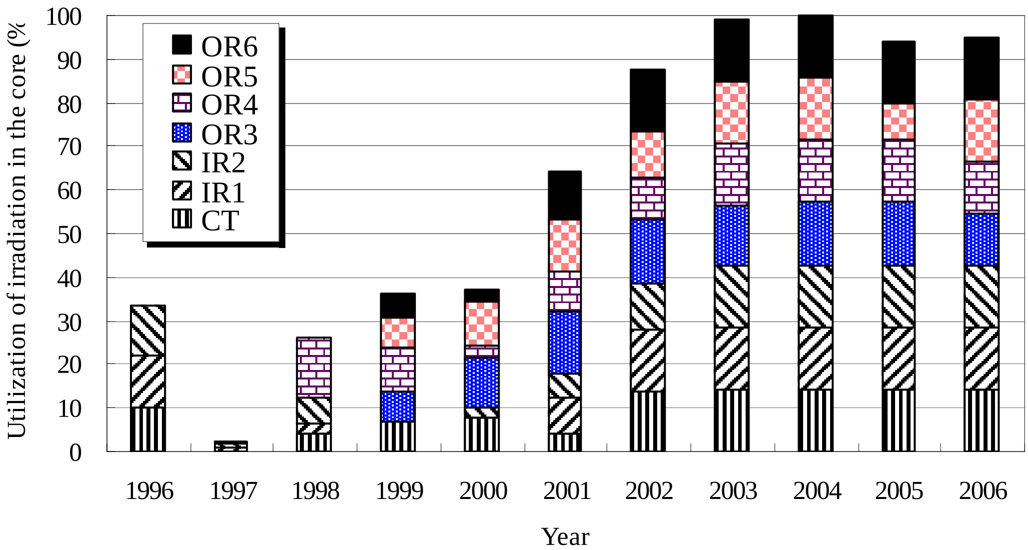

FIG. 1. The utilization status of irradiation holes in core by year.

In 1996, the first irradiation test at HANARO started with the HANARO test fuel containing 3 rods as the HANARO fuel qualification program at high power that was required to resolve a conditional licensing prerequisite. During developing the capsule for material and RI, the test fuels manufactured for localization of the HANARO fuel had been irradiated in the irradiation holes in the core until 1998. The development of RI capsule for producing Ir-192 and the instrument capsule for material test requiring the high fast and thermal flux condition made the irradiation holes in the center area of the core used continuously.

Another reason for increasing the utilization of HANARO is the fuel irradiation tests to develop the new fuels. In 1996, the first LEU U3Si fuel produced by atomization process was fabricated into a mini-assembly. After irradiation test of this one, the full-length bundle test was performed. From the results of these irradiation tests, HANARO fuel manufacturing facility was completed on May 2004 and local manufacturing of the HANARO fuel started. A qualification program for a rod type fuel of atomized U-Mo was initiated in 2000. The first and second irradiation tests were carried out in 2001 and 2003. The third test started in 2006 and will be finished in 2007. The U-Mo fuel irradiation tests will contribute to new fuel development for research reactor. KAERI has been developing a small and medium reactor, SMART for electricity generation and sea water desalination. U-Zr fuel was adopted for the SMART reactor. The first irradiation test of U-Zr fuel with three 8.1w/o enriched fuels and the second test with three 8.9 ～ 10.0 w/o fuels were completed. The third test had been performed with 19.75w/o fuels from 2004 to 2006. As a power reactor fuel, the large grained UO2 pellet has been developed aiming high burn-up in KAERI. Two test assemblies are loaded in the upper and lower positions of a capsule. The test assembly of the upper position was unloaded for PIE (Post irradiation examination) and the lower position assembly will be irradiated by the burn-up of 70MWD/kgU. KAERI has been studying DUPIC fuel through international co-operative research together with CANADA, U.S.A. and IAEA. Up to the present, irradiation tests of DUPIC (Direct Use of spent PWR fuels in CANDU reactors) fuel have been conducted 6 times. Through the DUPIC irradiation tests, thermal behavior of DUPIC pellet was analyzed and the technology for remote assembling and handling has been developed.

Following the experiences of the fuel irradiation test, FTL (Fuel Test Loop) has been installed in the core to extend the utilization of the fuel irradiation. This facility will be used for irradiation of PWR and CANDU fuel pellets under the environments of power reactor with high pressure and high temperature and will operate from 2008 after commissioning. The several irradiation holes at the reflector tank are not used yet. It is required to strengthen the technology for supporting users and to enlarge the utilization area.

Design and Installation of Fuel Test Loop in HANARO

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FTL (Fuel Test Loop) is a test facility which could conduct fuel irradiation test at HANARO reactor [1]. The fuels can be tested in the IR1 irradiation hole of HANARO under the commercial power plant operating conditions. The conceptual design of the FTL started at the end of 2001 and the detailed design was finished by March 2004. The installation of FTL was finished successfully on March 2007. The commissioning of the FTL will be conducted by January 2008. The FTL would be used for the irradiation test of high burn-up PWR fuels from January 2008. The design characteristics and the installation of the FTL facility are introduced in this paper.

The FTL is composed of an IPS (In-Pile test Section) and an OPS (Out Pile system). Fig. 1 shows the schematic diagram of the FTL. The IPS is to be loaded into the IR-1 position in the HANARO core. This implies that the environment around the IPS is subjected to a high neutron flux (Thermal neutron flux : 1.2×10^{14} n/cm^2/sec, Fast neutron flux : 1.6×10^{14} n/cm^2/sec). The IPS can accommodate up to 3 pins of fuel and has instruments such as thermocouple, LVDT and SPND to measure the fuel performances during the test. The IPS is composed of IPS head, outer pressure vessel, inner pressure vessel, flow divider and test fuel carrier. Inlet nozzle and outlet nozzle for the main cooling water located in the IPS head and insulated from the HANARO pool. Neon gas is filled into the gap between the outer pressure vessel and inner pressure vessel to insulate the IPS from the HANARO pool. A flow divider divides the outlet cooling water from the inlet cooling water. The test fuel carrier is composed of a fuel carrier support stem (with 6 slots for the hot cooling water injection), fuel carrier leg (3 legs are arranged through the 120° angles) and a fuel carrier head.

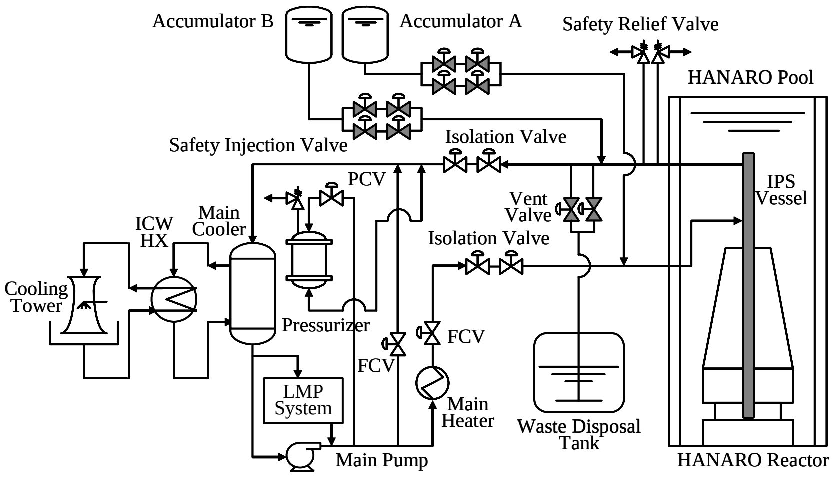

FIG. 1. Schematic diagram of FTL.

The OPS contains pressurizer, cooler, pump, heater and purification system which are necessary to maintain the proper fluid conditions. In addition, the OPS contains a engineered safety system that could safely shutdown both HANARO and the FTL if an accident occurs. The FTL simulates the irradiation conditions of the commercial power plants such as pressure, temperature and neutron flux levels to conduct for the irradiation and thermo hydraulic tests. The FTL coolant is supplied to the IPS at the required temperature, pressure and flow conditions that are consistent with the test fuel. The nuclear heat added within the IPS is removed by the main circulating water cooler. The main circulating pump provides the motive power to circulate the FTL coolant within the loop. After pump discharge, an in-line heater provides the capability to increase temperature for startup and for positive temperature control. A pressurizer is provided to establish and maintain the coolant pressure to the test fuel type. A purification and de-gasification system is provided to maintain the coolant inventory and the chemistry conditions. The emergency cooling system is provided to maintain the experimental fuel cooling in the event of the anticipated operational occurrence or the design basis accidents. The OPS components are located in FTL room 1 (safety related components), room 2 (non-safety related components) and the FTL control room which are dedicated rooms for the FTL.

The control system for FTL operation is divided into the safety control system and the non-safety control system [2]. The safety control system is used for controlling of the safety related FTL process systems and shutdown the HANARO reactor against the abnormal operating conditions. The non-safety control system consisted of a computer control system controls the non-safety related process systems. The application fields of the FTL are as follows.

- Nuclear fuel irradiation behavior test at the operating condition of the commercial power plant.
- Fuel burn-up and mechanical integrity verification.
- Irradiation data generation for the analysis model
- Technical improvement of design and fabrication for the advanced fuel development.
- Fuel rod irradiation test for performance verification.

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Design Characteristics of Cold Neutron Source in HANARO

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The HANARO has been operated for 12 years since its initial criticality in February of 1995. The reactor power has been gradually increased to 30 MWth through out its service period. In order to enhance the utilization capacity of HANARO, a CNS (Cold Neutron Source) development project has been underway since 2003. As of now, the detailed design of the CNS has been completed. The overall design concept of the CNS is to ensure that the reactor safety systems and the on-site personnel and equipment are not adversely affected by the hydrogen-oxygen reaction from the CNS. The safety design criteria of the CNS are a defence-in-depth approach that provides several means to avoid any accidental contact between the hydrogen in the system and the air. Therefore, the principles of a conservatism, simplicity, redundancy, fail-safe design, and passive safety features are included to design it with an enhanced safely and efficiency. According to the safety classification based on ANSI N51.1 [1], the HANARO reactor assembly is classified as safety class 3. A vacuum chamber, which will be installed into the reflector tank of the reactor, is the highest safety class of the CNS components. As the vacuum should maintain its integrity in the case of a hydrogen-oxygen reaction, it is defined as an ultimate pressure boundary in accordance with the ASME Sec. III NB code requirements; So, the safety class of the vacuum chamber is specified as safety class 3. Every component except the vacuum chamber is classified as a non-nuclear safety class.

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The Vacuum System (VS) is to act as thermal insulation for the cryogenic part of the IPA and act as a safety barrier against an irruption of liquids and / or gases from the outside. The thermal insulation is of relevance to the performance of the IPA cooling system process.
The VS consists of a pumping station, valves and gauges, and connecting pipes. The vacuum level for the cryogenic insulation shall be at least lower than 10^{-5} torr, which will be achieved by means of two vacuum pumping sets. One of the pumping sets is in operation while the other is on standby. All of the VS are installed in a vacuum box filled with nitrogen blanketing gas. The discharging gas from the vacuum pump is collected in a gas collection tank and then the collected gas is released into the reactor hall in the case of the hydrogen contact not being higher than 3.5% of the total volume. Fig. 1 shows a final schematic drawing came from the detailed design of CNS in HANARO.
This paper describes the design characteristics of a CNS in HANARO. The detailed design based on the safety criteria and user's requirements has been completed and it will provide the basis by which the manufacturing, installing, and commissioning of the CNS will proceed. All of the CNS equipment and systems except for the IPA will be installed in August 2008 and it will start its commissioning of the end of 2008. The IPA will be installed in the first half of 2009. It is expected that the CNS will be available from the beginning of 2010.

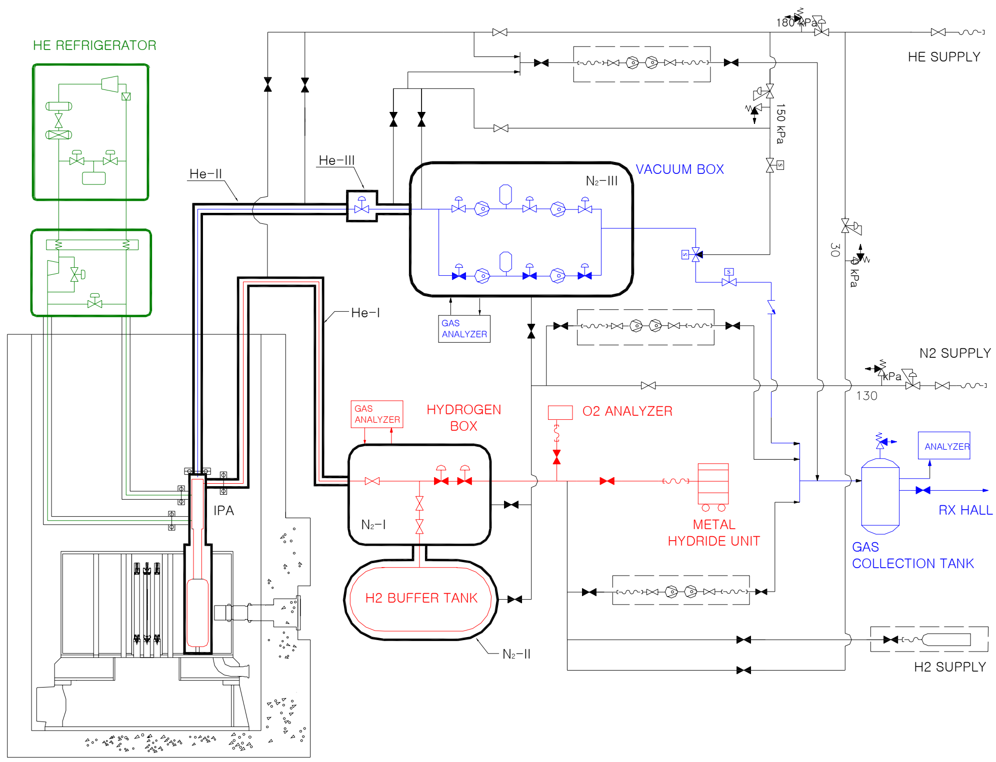

FIG. 1. Schematic Drawing of CNS in HANARO

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[2] S.I. Wu, Y.K. Kim, and Y.J. Kim, 'Safety Design Criteria for Facilities of Cold Neutron Source in HANARO', HANARO Workshop 2004, May, 2004

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PURE COMMERCIAL GOLD FOILS AS NEUTRON FLUX MONITOR: NEUTRON SELF-SHIELDING ASSESSMENT

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1. EXPERIMENTAL

Certified activation monitors such as IRMM-530R Al-0.1% Au are used to determine neutron flux [1]. Considering the high cost and other complications, pure (99.9 %) commercial gold foils was introduced as an alternative material. The minimum available thickness of the commercial gold foil was 0.1 mm. According to that the principal problems which we must solve are: results repeatability investigation and determination of the neutron self-shielding factors for both thermal and epithermal neutrons for the prepared foils. Two groups of pure commercial gold foils of 2 mm diameter were prepared. The first one was of 0.1 mm thickness and was used to investigate the results repeatability. The second group was five couples of samples of thicknesses ranges from (0.1 to 0.5) mm and was used to determine the neutron self-shielding factors for both thermal and epithermal neutrons. All samples were irradiated in the same conditions in the inner irradiation sites of MNSR. The specific saturated activities resulting from thermal (α_{th}) and epithermal (α_{epi}) activations were determined experimentally [2]. Table1 shows the results repeatability.

Table 1: The results repeatability

	bare			Cd-covered		
d, mm	**n***	**α_1, Bq/nuclide**	**stdev%**	**n***	**α_2, Bq/nuclide**	**stdev%**
0.1	15	7.62E-11	5.64	6	1.55E-11	5.23
0.2	3	6.73E-11	7.67	3	1.10E-11	8.93
0.3	3	6.12E-11	11.72	2	9.04E-12	0.25
0.4	3	5.34E-11	10.44	3	8.57E-12	8.86
0.5	3	5.37E-11	7.19	3	7.62E-12	8.28

** - test frequency*

The second group foils were divided into two subgroups. The first subgroup foils were irradiated bare and the second subgroup foils were irradiated in the cadmium boxes. Induced specific activities were measured. The neutron self-shielding factor G for a foil was determined (Fig. 1).

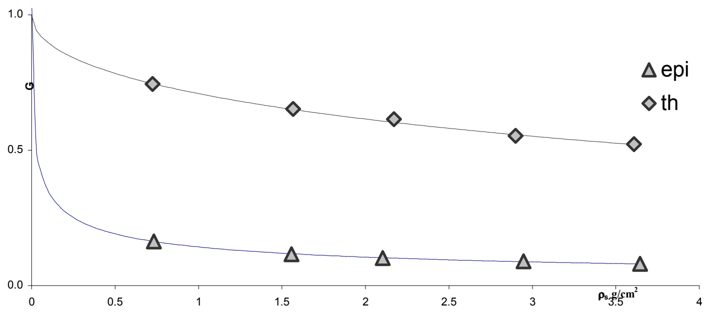

Fig.1: The curves of the neutron self-shielding factors versus the foil thickness for thermal and epithermal range.

The resulted values of the thermal and epithermal neutron fluxes were compared with the corresponding values resulted by using certified activation monitors (IRMM-530R Al-0.1%).

Table 2: Comparison between the results of certified activation detector and in-house one.

detector	φ_{th}	er. %	φ_{e}	er. %
standard	9.21E+11	2.5	5.86E+10	1.0
in-house	9.27E+11	3.1	6.31E+10	3.6

2. CONCLUSION

An assessment of pure commercial (99.9 %) gold foils as neutron flux monitor was performed. A thin foils of pure commercial gold were prepared as an in-house reference material for neutron flux measurement. The assessed foils are available commercially and its cost is much less than the certified ones. Determination of the neutron self-shielding factors in these foils for both thermal and epithermal neutrons have been done experimentally. These foils show good results repeatability and good agreement with certified activation monitors. According to the well-known physical constants of the nuclide and its low cost comparing with certified foils, it can be used as an in-house reference monitor.

The authors would like to thank Prof. I. Othman, the Director General of the Atomic Energy Commission of Syria, for his continuous support
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Design Characteristics of Cold Neutron Source in HANARO

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The HANARO has been operated for 12 years since its initial criticality in February of 1995. The reactor power has been gradually increased to 30 MWth through out its service period. In order to enhance the utilization capacity of HANARO, a CNS (Cold Neutron Source) development project has been underway since 2003. As of now, the detailed design of the CNS has been completed. The overall design concept of the CNS is to ensure that the reactor safety systems and the on-site personnel and equipment are not adversely affected by the hydrogen-oxygen reaction from the CNS. The safety design criteria of the CNS are a defence-in-depth approach that provides several means to avoid any accidental contact between the hydrogen in the system and the air. Therefore, the principles of a conservatism, simplicity, redundancy, fail-safe design, and passive safety features are included to design it with an enhanced safely and efficiency. According to the safety classification based on ANSI N51.1 [1], the HANARO reactor assembly is classified as safety class 3. A vacuum chamber, which will be installed into the reflector tank of the reactor, is the highest safety class of the CNS components. As the vacuum should maintain its integrity in the case of a hydrogen-oxygen reaction, it is defined as an ultimate pressure boundary in accordance with the ASME Sec. III NB code requirements; So, the safety class of the vacuum chamber is specified as safety class 3. Every component except the vacuum chamber is classified as a non-nuclear safety class.

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Poster Presentations:
Fuel and Waste Management

Synopses no. IAEA-CN-156/	Synopses Title	Main Author
F-4	Converting HIFAR to Low Enriched Uranium Fuel	Storr, G.J.
F-6	The regulatory role of the Australian Radiation Protection and Nuclear Safety Agency in relation to spent fuel arising from research reactors in Australia	Sarkar, S.
F-7	Design and construction of a Decay Pool at IAN-R1 Research Reactor	Sarta Fuentes, J.A.
DE-1	Safety Assessment for Decommissioning of Research Reactors	Kaulard, J.
WM-4	Further Development in Characterization of Radioactive Waste Drums by Non Destructive Gamma Spectrometry at GRR-1	Savidou, A.
F-1	Al-Clad Spent Nuclear Fuel Corrosion Studies at Magurele site, Romania	Dragolici, A.C.
F-2	Vinca site preparation for spent fuel shipment	Pešic, M.
WM-5	Experimental study of systematic errors of gamma technique for assay of radioactive waste drums	Tran, Dung

Converting HIFAR to Low Enriched Uranium fuel

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The Australian Nuclear Science and Technology Organisation (ANSTO) began operating the High Flux Australian Reactor (HIFAR) in 1958, a DIDO-class research reactor operated at a thermal power of 10 MW. On 30th January 2007, after more than 49 years of successful and safe operation HIFAR was finally shutdown. Since that time all the fuel has been successfully removed from the reactor containment building. HIFAR was primarily used for neutron scattering science, service irradiations and isotope production. Over the nearly 50-year operating life of HIFAR a variety of fuel designs have been used. After the 1970s fuel enrichment was reduced in stages from over 90 percent to 19.75% in 2006.

The reactor core consisted of 25 fuel elements with uranium-aluminium alloy fuel sections, arranged in concentric tubes. HIFAR was moderated and cooled by heavy water, and the coolant contained within an aluminium tank, which in turn was surrounded by a graphite reflector and concrete biological shielding. Reactor control and shutdown were achieved with six europium tipped cadmium control blades, which moved as a bank between the rows of fuel elements. Two cadmium shutdown rods provided additional shutdown capacity.

In May 2006 the HIFAR reactor was fully converted to Low Enriched Uranium fuel. The conversion commenced in October 2004. The LEU fuel was procured from RISO National Laboratory in Denmark, was originally made for use in the DR3 reactor, and was modified to be compatible with HIFAR. This type of fuel was used safely in DR3 before its closure. A safety analysis report for the approval and use of the LEU fuel which was prepared well in advance of loading the fuel into HIFAR, provided detailed analyses of issues important to reactor and general fuel safety, including, criticality safety outside the reactor, reactor physics, eversafe times, thermal hydraulics and accident analyses. Many of the issues studied for LEU fuel reanalysed operational and accident conditions that had been previously analysed for HEU fuel. In most cases the conclusions provided in each analysis demonstrated there was little difference in behaviour between HEU fuel and LEU fuel in HIFAR under operational and accident conditions. However, there was one significant difference between HEU and LEU fuel as it was shown that in general eversafe times for LEU fuel are greater than for HEU fuel. Consequently, procedures were modified for some operations to ensure compliance with safe heat limits.

The paper will present the process undertaken for the conversion of HIFAR, including the development of the safety case, requirements for regulatory approvals, and results from the conversion program.

The regulatory role of the Australian Radiation Protection and Nuclear Safety Agency in relation to spent fuel arising from research reactors in Australia

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This paper will describe the elements and performance of ARPANSA's regulatory management of spent fuel arising in Australia, with particular emphasis on the experience of ensuring compliance with the Code of Practice Code of Practice for Safe Transport of Radioactive Materials in relation to in land surface transport of spent fuel within Australia.

The Australian Radiation Protection and Nuclear Safety Agency is the regulatory authority for Commonwealth entities, such as the Australian Nuclear Science and Technology Organisation (ANSTO), who operate nuclear installations in Australia.. Nuclear installations that operate under ARPANSA facility licence include research reactors and plants for the storage and management of research reactor fuel. ANSTO is the only operator of nuclear installations in Australia.

The Australian Radiation Protection and Nuclear Safety Agency is also the competent Authority for inland surface transport. ARPANSA has adopted the IAEA Safety Regulations for Safe Transport of Radioactive Materials domestically in the form of the ARPANSA Code of Practice for Safe Transport of Radioactive Materials (RPS 2).

As the competent authority ARPANSA approves the shipment and design of a new cask, validate original certificate applying the requirements of the RPS 2.

ARPANSA's regulatory oversight of compliance with the requirements of its own legislation and the requirements of the Code emphasises assurance of safety in the operation of nuclear installations and the shipment of spent fuel is achieved principally by prior assessment of the operator/consignors safety case, and by compliance monitoring through regular reporting (quarterly and annually), as well as planned and reactive inspections. During the operating life of these facilities for several decades there have been no incidents which have had off-site or significant on-site, consequences.

This paper will examine that experience and in particular focus on the regulatory experience of oversight of recent shipment of spent fuel arising from the HIFAR reactor, including the methods for reviewing requests for approval and the issues that have emerged.

Design and construction of a Decay Pool at IAN-R1 Research Reactor

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IAN-R1 Research Reactor is a swimming pool type with a concrete shield and two beamports. IAN-R1 is licensed for steady state operation at 30 Kw and it is located at Instituto Colombiano de Geologia y Mineria INGEOMINAS in Bogotá D.C., Colombia. IAN-R1 is a research reactor pool type which was initially fueled with MTR–HEU enriched to 93% U-235 [1], operated since 1965 to 10 kW (t) and was upgraded to 30 kW(t) in 1980. General Atomics (GA) achieved in 1997 the conversion of HEU fuel to LEU fuel TRIGA ($UzrH_{1.6}$) type enriched to 19.7%,, and upgraded the reactor power to 100 kW(t) [2].

With cooperation of the International Atomic Energy Agency (IAEA) and the Department of Energy (DOE) of the United States, several calculations and tasks related to the waste disposal of spent MTR-HEU fuel enriched nominally to 93% were carried out for the conversion of the IAN-R1 Research Reactor from MTR-HEU fuel to TRIGA-LEU fuel. In order to remove the spent MTR-HEU fuel of the core and store it safely a project was established at the Instituto de Ciencias Nucleares y Energìas Alternativas (INEA). This project included training, acquisition of hardware and software, design and construction of a decay pool, transfer of the spent HEU fuel elements into the decay pool and his final transport in 1996 to Savannah River Site in United States. In this paper are presented neutronic and shielding calculations for the decay pool concerning the storage of spent MTR-HEU fuel elements.

The standard MTR-HEU fuel element includes 10 fuel plates having each one a 593,7 mm (length) x 62,3 mm (width) x 0,508 mm (thickness) active area. This fuel is metallic uranium alloyed with the 90% purified aluminum in the U^{235} isotope. The unit effective density is 3,32667 g/cm^3 and is contained by a 0,508 mm-thick Al-1100 clad with. 150 g of ^{235}U and 16.6715 g of ^{238}U. The core contains 16 fuel assemblies, 13 standard MTR-HEU fuel elements with 10 fuel plates/element, and 3 of the 16 fuel assemblies are control elements with 6 fuel plates/control element.

The facility designed to store spent fuel at IAN-R1 consists of a storage rack located into a cylindrical tank of 4.0 m in height with 2.0 meter in diameter and 6 mm thickness Type 304 of stainless steel. The rack has a total capacity to store 30 fuel assemblies and it is made with stainless steel. The biological shielding is provided by the water inside the tank and by the concrete (2.3 g/cm^3) which surrounds the tank on the radial direction. A water purification system provides corrosion control and optical clarity. Control of the water purity is performed by analysis of the water conductivity with values as low as 5 µS/cm. The decay pool is physically independent of the reactor pool and is located inside the reactor building.

In order to establish the sub-criticality was evaluated a configuration of 30 storage positions occupied with standard MTR-HEU fuels. In this analysis neutron cross sections are generated for five neutron energy groups. All neutron cross sections for thermal and fast energies are generated using the WIMS code [3]. From diffusion code CITATION [4] a value of the effective multiplication factor $K_{eff} = 0.42161$ was obtained for all the storage rack loaded with standard MTR-HEU fuels.

A model with MERCURE code [5] has been developed for 16 storage positions occupied with MTR-HEU spent fuel elements after 120 days the reactor was shutdown. To estimate the average irradiation burning level in the standard MTR-HEU fuel, the reactor operating record was considered. The kW-hour datas since the year 1965 to the last operation of the IAN-R1 with MTR-HEU fuel in 1994 were taken into account. The photon spectrums for fission and activation products for evaluating dose rate distribution were taken from reference [6]. The radiation levels for the horizontal direction due to the direct radiation from MTR-HEU fuel elements into the decay pool were evaluated through the biological shield at the mid-plane of the MTR-HEU fuel.

Due to the low nominal power and short operation cycles, the IAN-R1 Research Reactor at INGEOMIAS actually does not generate spent LEU fuel and it is not considered to be problem. Nevertheless IAN-R1 has a facility to store all the TRIGA-LEU fuel of the reactor core and there is enough operational storage capacity of spent LEU fuel for many years of reactor operation, even if there is an increase in reactor utilization.

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Safety Assessment for Decommissioning of Research Reactors

International Project on Evaluation and Demonstration of Safety during Decommissioning of Nuclear Facilities (DeSa)

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Decommissioning is the final stage in the life cycle of each nuclear facility, including research reactors. In the past, there was a tendency to address related aspects at a very late period of operation of the facility or even after final shut down stage. While this tendency has been changing especially in case of nuclear power plants and research reactors, the international community recommended to IAEA in the Berlin Conference in 2002 to stipulate an early addressing of decommissioning [1]. This emphasis took also into consideration the fact that worldwide there are over 500 research reactors and critical assembly units that will eventually require decommissioning. In addition, the international community recommended to IAEA to provide general assistance in development and review of safety assessments related to decommissioning. Accordingly, in its International Action Plan on Decommissioning of Nuclear Facilities, approved by the IAEA Board of Governors in 2004, IAEA reflected these recommendations [2] and initiated the international project on *Evaluation and Demonstration of Safety during Decommissioning of Nuclear Facilities (DeSa Project)* [3].

For the last three years about fifty experts from over thirty Member States have been working in the DeSa project on; (i) the establishment of a harmonized general safety assessment methodology for decommissioning; on (ii) the development of recommendations for a regulatory review process of such safety assessments; (iii) development of recommendations on the application of a graded approach to performance and review of safety assessment, which ensures that the extent of the safety assessment is commensurate which the risks posed by the facility and the proposed decommissioning activities, and finally (iv) the application of the assessment methodology, the regulatory review procedure and graded approach recommendations to three test cases with different complexities and hazard potentials – a nuclear power plant, a research reactor and a nuclear laboratory.

This paper provides an overview on current status of the DeSa project activities, and their application to the development of the Research Reactor Test Case, including the presentation of preliminary lessons learned from applying the graded approach on a research reactor. The DeSa project results, including the outcomes of the review of the Research Reactor Test Case by applying the review procedures are envisaged to be summarized at the 4th Joint DeSa meeting in October 2007, where the scope and objectives of a follow-up project will be also discussed.

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Further Development in Characterization of Radioactive Waste Drums by Non Destructive Gamma Spectrometry at GRR-1

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In the present work a non destructive technique based on gamma spectrometry and application of the Monte Carlo method for detector efficiency calibration, was used to assay radioactive content waste drums.

Exploranium™ GR-130 miniSPEC portable gamma ray spectrometer was used to externally monitor 26 waste drums containing radioactivity of ion exchange resin waste from the water demineralization system of GRR-1 open pool-type research reactor facility at NCSR "Demokritos". A three 20 min consequent measurements (one every 120° of drum rotation) scheme was employed. The GR-130 gamma ray spectrometer was equipped with a 38 mm diameter x 57 mm long NaI(Tl) scintillation detector of 7% resolution for ^{137}Cs at 662 keV.

Monte Carlo simulation of the drum and detector configuration was carried out using MCNP-4C2 code and cross-section data from the ENDF-VI-b library. The code was used to perform numerical simulations taking into consideration the energy of the gamma ray emitter, the matrix material, the detector efficiency, the geometric configuration employed, the size of the drum, and the wall material and thickness of the drum. The detector was modelled as a cylinder of sodium iodide surrounded by an aluminum layer. The model geometry included a homogeneous distributed cylindrical volume source within a cylindrical iron drum. Runs were performed for cylindrical sources of diameter 55.6 cm and heights ranging between 60 cm and 40 cm, thus representing different drum loads. Three resin densities of 0.75, 0.84 and 0.92 g cm^{-3} were modelled. The NaI crystal active centre was positioned at 57 cm from the geometrical centre of the drum at the drum mid-height level with its axis vertical to the drum's main axis of symmetry.

The results of MNCP simulations of a point source positioned at the geometrical centre of an air filled drum were calibrated against experimental measurements performed under the same conditions. Measurements were performed using ^{60}Co, ^{137}Cs and ^{152}Eu standard point sources and appropriate adjustment factors for the MCNP calculations were derived for given photon energies. Efficiency curves in the energy range 60 to 1500 keV were predicted. A relative error of less than 5% was achieved in all simulated cases.

Satisfactory agreement was observed by comparing the results of the non destructive method against analytical results of samples obtained from each drum. Fig. 1 and 2 show the activity concentrations in the waste drums as estimated by the non destructive drum assay, using efficiencies derived by the Monte Carlo method, and by the sample analysis technique, for ^{108m}Ag and ^{60}Co radioisotopes, respectively. In these figures the z-scores representing the difference between the two measuring techniques in combined standard deviation units are also shown.

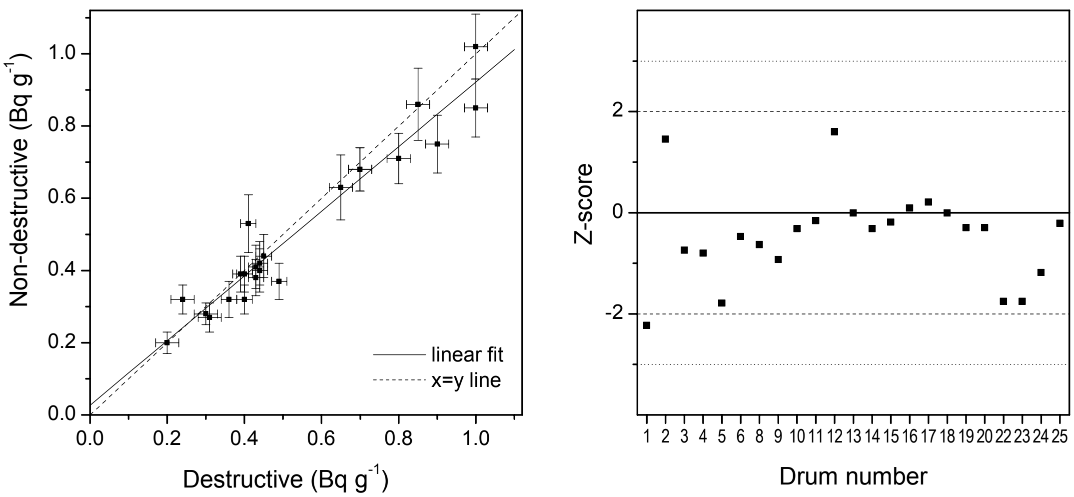

FIG. 1. ^{108m}Ag activity concentration in the waste drums as estimated by the non-destructive drum assay and determined by the conventional gamma spectrometry technique.

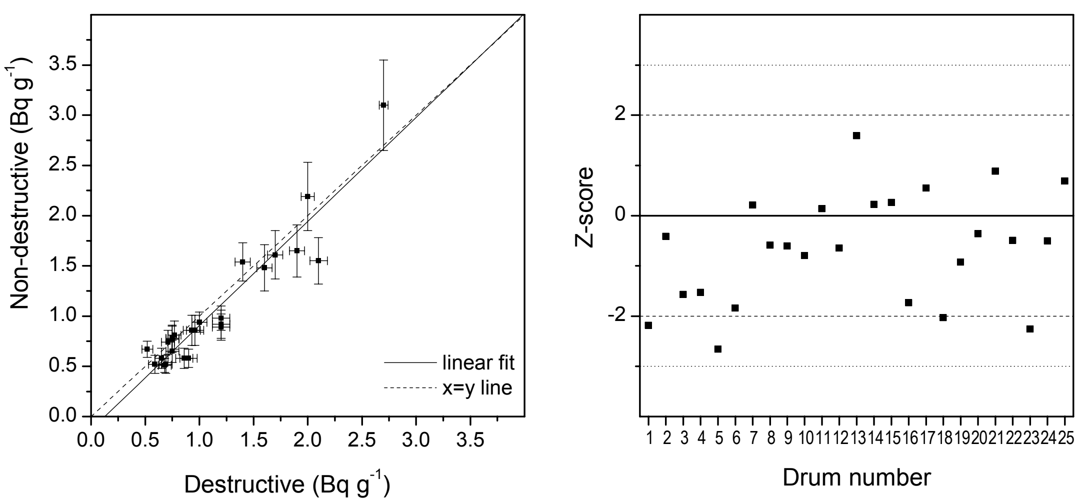

FIG. 2. ^{60}Co activity concentration in the waste drums as estimated by the non-destructive drum assay and determined by the conventional gamma spectrometry technique.

The derivation of detector efficiency was performed assuming a homogeneous distribution of the source activity within the matrix material. Moreover, the MCNP model was used to examine the effect of inhomogeneities, variation of matrix material density and drum filling height on the accuracy of the technique and estimate the measurement bias.

The simulations of the present study can easily be extended to model other materials and container types and sizes as well. Therefore, the technique can also be applied in other radiation protection applications such as biological radioactive waste from hospitals or other physical forms materials that can be met in radioactive waste.

Al-Clad Spent Nuclear Fuel Corrosion Studies at Magurele site, Romania

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This paper is an overview of the scientific investigations of the IAEA Coordinated Research Project (CRP) on “Corrosion of Research Reactor Aluminium-Clad Spent Fuel in Water (Phase II)”, as carried out at IFIN-HH institute, Magurele, Romania. The results of corrosion surveillance activities on racks containing coupons (see Table I) immersed either in vertical or horizontal position in individual fuel storage basins are presented in this paper. IAEA instructions for coupon preparations and immersion [1] were strictly followed.

TABLE I: ORDER AND POSITION OF COUPONS IN RACKS.

No.	IAEA order	BASIN 2		BASIN 3	
	TOP	TOP (Site order)	ID #	TOP (Site order)	ID #
CR1-	Ceramic ring	Ceramic ring		Ceramic ring	
Coupon 1	Site specific alloy	Site specific alloy	2-1	Site specific alloy	3-1
CR2-	Ceramic ring	Ceramic ring		Ceramic ring	
Coupon 2	Site specific alloy	Site specific alloy	2-2	Site specific alloy	3-2
Coupon 3	SS304	SS304	320	SS304	323
CR3-	Ceramic ring	Ceramic ring		Ceramic ring	
Coupon 4	Szav-1	Szav-1	385	Szav-1	369
CR4-	Ceramic ring	Ceramic ring		Ceramic ring	
Coupon 5	Szav-1 (P-O & S)	Szav-1 (P-O & S)	316	Szav-1 (P-O & S)	304
CR5-	Ceramic ring	Ceramic ring		Ceramic ring	
Coupon 6	Szav-1	Szav-1	339	Szav-1	370
Coupon 7	Szav-1	Szav-1	324	Szav-1	355
CR6-	Ceramic ring	Ceramic ring		Ceramic ring	
Coupon 8	Szav-1	Szav-1	348	Szav-1	330
Coupon 9	SS304	SS304	346	SS304	339
CR7-	Ceramic ring	Ceramic ring		Ceramic ring	
Coupon 10	AA6061	AA6061	310	AA6061	304
CR8-	Ceramic ring	Ceramic ring		Ceramic ring	
Coupon 11	AA6061 (P-O & S)	AA6063 (P-O & S)	200	AA6063 (P-O & S)	209
CR9-	Ceramic ring	Ceramic ring		Ceramic ring	
Coupon 12	AA6061	AA6061	340	AA6061	319
Coupon 13	AA6061	AA6063	246	AA6063	236
CR10-	Ceramic ring	Ceramic ring		Ceramic ring	
Coupon 14	AA6061	AA6063	233	AA6063 (P-O & S)	216
Coupon 15	SS304	SS304	324	SS304	357
CR11-	Ceramic ring	Ceramic ring		Ceramic ring	
	BOTTOM	BOTTOM		BOTTOM	
P-O & S = pre-oxidized and scratched					

Specific procedures for coupon preparations for examination and further analysis [2] were implemented for obtaining consistent results. The experimental investigations have been done on aluminium alloys ($AlMg_3$, AA 6061, AA6063 and Szav-1) and stainless steel 304 coupons. Single coupons, bimetallic and crevice coupled have been investigated. Visual examinations, metallographic and ultrasonic investigations [3] were used. Also, an image analysis software was applied to measure the surface areas affected by corrosion. The experimental data on the most affected coupons by galvanic corrosion have been statistic analyzed. The single $AlMg_3$ coupons have presented a non-significant pitting corrosion, for SZAV-1, AA6063 the susceptibility to corrosion phenomenon is reduced, excepting AA6061 (*see Fig. 1*);

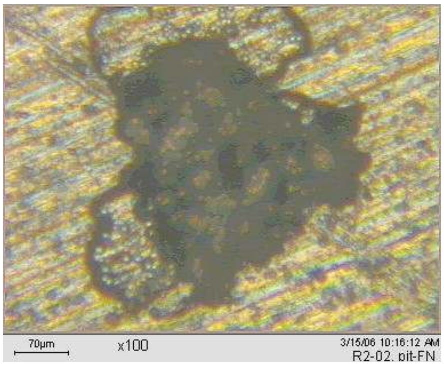

FIG. 1. Coupon R2-02, pit aspect.

The oxidized and scratched coupons are not affected by corrosion;
The coupled coupons made from identical material are affected by crevice corrosion, the maximum pits depth being less than 0.1 mm;
The bimetallic coupled discs present a galvanic corrosion, with material dissolution of extended areas (1.1%-1.7% of the surface); the effect of weight reducing is more pronounced on this type of coupons. Also the pitting factor presents the highest value = 3.2;
The most affected coupons are those immersed in Basin 3, in the coupled conditions.
On horizontal rack, the appearance of pitting corrosion is clearly shown only at the first coupon owing to the deposit foreign particles from the basin lids. These particles are cast iron and paint and also dust from the air carried by the ventilation system.

[1] Test Protocol, IAEA CRP "Corrosion of Research Reactor Aluminium-Clad Spent Fuel in Water (Phase II)".
[2] ASTM – G1, Standard Recommended Practice for Preparing, Cleaning and Evaluating Corrosion Test Specimen.
[3] ASTM – G46, Standard Recommended Practice for Examinations and Evaluation of Pitting Corrosion.

Vinca Site Preparation for Spent Fuel Shipment

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Heavy water research reactor RA at theVinca Institute of nuclear sciences, Belgrade, Serbia was constructed according design developed in the Institute for Theoretical and Experimental Physics in ex-USSR in late 40es and mid-fifties of last century. The reactor at Vinca site was in operation from 1959 to 1984. It has operated using Russian origin TVR-S type fuel elements made of uranium metal (LEU) from 1959 to 1976. From 1976 for the reactor operation was used the same type fuel elements, but designed as uranium dioxide (HEU) dispersed in aluminium matrix. The reactor was temporary shut down in 1984 for refurbishment, but the operation was never started again.

Serbian government has decided in 2002 to shut down the RA reactor permanently and to proceed with development of the decommissioning planning. At the same time the Government made decision to ship fresh and spent nuclear fuel (SNF) arose from the reactor operation back to the country of origin – Russian Federation within the frame of the Russian Research Reactor Fuel Return Program. IAEA has coordinating this Serbian SNF shipment program since 2003. A three-party (IAEA, Vinca and Russian Consortium Mayak/Sosny/Tenex) contract was signed at the IAEA on September 2006 for Vinca's spent fuel repackaging and shipment.

This paper covers the all activities on Vinca site initiated already in 2003 for SNF preparation and the RA facility and Vinca site modification within the frame of the contract. These activities include:

- Collection of the facility relevant documentation and converting to electronic form
- SNF identification and verification
- SNF condition assessment and investigation
- SNF nuclide inventory calculation
- Dose rate measurements from SNF storage containers and fuel elements for verification of calculation results
- SNF burn up measurements for verification of calculation results
- Monitoring of activity of ^{137}Cs nuclide in pool water and in SNF storage containers
- Removal of sludge from water from SNF pool to increase water transparency
- Upgrading facility ventilation systems
- Upgrading monitoring systems for detection of emission of radionuclide from ventilation stack of the facility
- Examination (and possible increasing) of loading capacity of floor in reactor room and storage room
- Verification of operation of tools and equipment used for SNF management
- Verification of operation and upgrading of internal transport devices
- Preparation of RAW management strategy to meet requirements of SNF repackaging and transportation technology
- Preparation of basins of SNF storage pool, reactor core and reactor block for SNF repackaging

- Preparation of SNF room and basins for repackaging equipment and temporary storage of repackaged SNF
- Removal carbon steel structure from SNF water storage basin
- Preparation of the reactor room, SNF room and basins for loading repackaged SNF to transport casks
- Review of Institutes roads and examination for possible modification
- Upgrading radiation protection equipment and health physics instruments
- Increasing safety culture of operators through intensive courses and training on mock-ups

Acknowledgement

Authors and the all members of the Vinča VIND SNF/RadProt/RAW team acknowledge to the IAEA experts for TCP/Europe Department, IAEA experts from NFCWT /NFCM and DNIS/RRS Sections, Ministry of Science and Environmental Protection of the Republic Serbia, and the experts of the Sosny/Mayak/Tenex Consortium from the Russian Federation for their support, efforts and engagements in this project.

EXPERIMENTAL STUDY OF SYSTEMATIC ERRORS OF GAMMA TECHNIQUE FOR ASSAY OF RADIOACTIVE WASTE DRUMS

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1) Basic principle of measurement

The operation of nuclear reactors results in the production of a considerable amount of radioactive low density waste, mainly consisting of organic materials which are usually stored in large sealed drums (208 l). The drums must be checked to satisfy regulations of radioactive waste management.

The Segmented Gamma Scanner (SGS) is an traditional tool for the isotopic composition measurement and for determination of the activity level in gamma contaminated waste drums [1, 2]. The systematic error of this technique is still large because of: non-uniform distribution of radioactive source within the drums frequently causes the largest error [3,4]; non-uniform distribution of non-radioactive materials (matrix) [4,5]; particles size of the nuclear material, the lump effect, specially for uranium and plutonium assay [2,6,7,8]; the drum-to-detector distance [4].

The other measuring technique has been studied by Cesana [9] for assay of the drums containing low density waste, mainly consisting of organic materials such as contaminated paper, rags, protective clothing, shoes, etc. The measuring arrangement consists two identical detectors set at equal distances from two bases of drum. This technique was developed because of the reasons: first, the measure is usually limited to rather hard gamma rays emitted by Cs-137, Cs-134, Co-60..., the mass-absorption coefficients are nearly independent of the atomic number of matrix, and the linear attenuation coefficients are very low (typically 0.01-0.03 cm^{-2}) because of the low waste density (0.2-0.4 g/cm^3). Therefore, the gamma attenuation in whole drum can be considered by means of on average linear attenuation coefficient, independent of the position in a drum; second, the number of drums to be examined is supposed to be very large, so that a detailed scanning by SGS is practically impossible; third, the measuring arrangement is very simple, so it can be used for any situation. However, in this measurement the drum must lie down. Thus, it spends time and is not convenient for use in practice. Moreover, the error is still very large (223 %) if the activity is distributed in the vertical large region. In order to overcome these disadvantages a modification to the measurement arrangement with the geometric coefficient for cylindrical sample was proposed [10]. The measuring principle is given in Figure 1. A detector is set perpendicularly to the drum axis at its middle point at equal distances from the curve face of drum.

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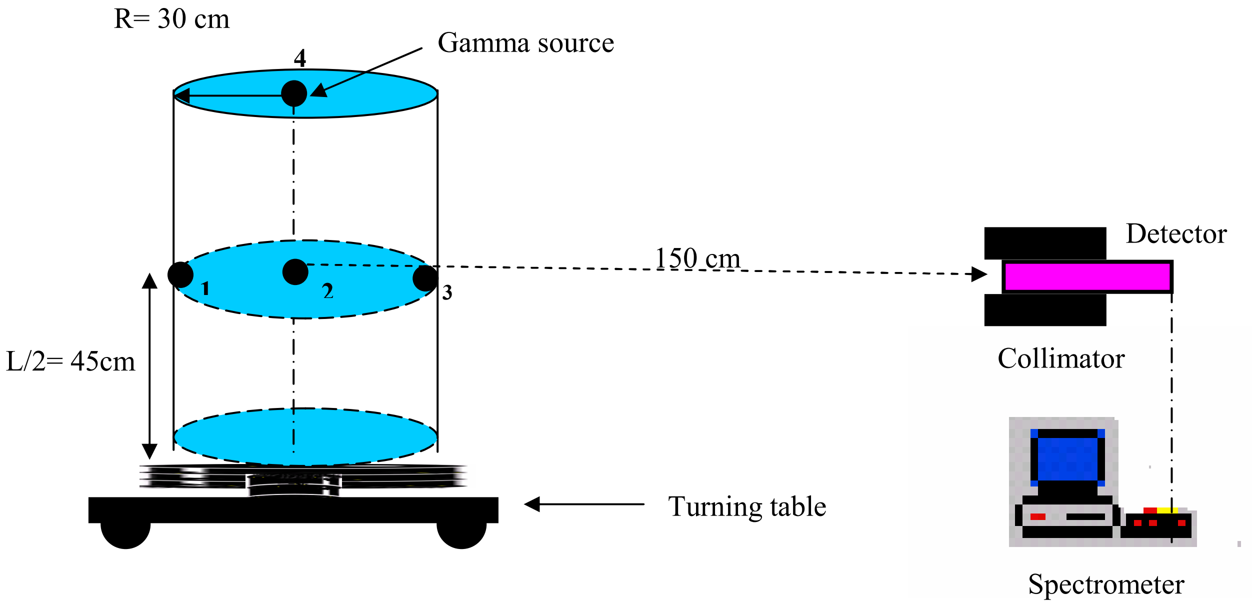

Figure 1: Waste drum counting geometry. The drum is measured two times. In the second measurement the drum is rotated 180^0.

The activity I in the drum can be determined as

$$I = \frac{(C_1 C_2)^{1/2}}{G} \quad (1)$$

with

$$G = \frac{\alpha}{S^2} \exp -(\mu_1 \times 0.832 \times R) \quad (2)$$

where C_1, C_2- the count rates of the detector in two measurements; S=K-0.823.R; K- distance from detector to the drum. This distance is larger the diameter of the drum several times; $\mu_1 = 2/S + \mu$; α- coefficient, that is a function of the gamma-ray energy, intrinsic efficiency of detector and the effective distance K. μ - the linear attenuation coefficient in the waste mixture. R- radius of drum; The values of α/K^2 versus gamma ray energy can be determined by using appropriate standard source.

2) Experimental study and discussion

The proposed coefficient G in formula (2) is based on the some approximations [9,10]. The distance between the centre of the drum and the detectors is about four times the half of the height of the drum. Thus, the geometric mean of the efficiencies of the sources at different positions of vertical axis can be considered as the same. When the height of the drum is 86 cm, K= 120 cm the maximum variation is less than 6 %. Therefore, the vertical count rate variation can be ignored. The systematic errors are caused mainly by the horizontal inhomogeneity of the source in the drum because the important assumption of this technique is that the activity is distributed in a small region of the drum volume.

In order to develop the technique to measure the activity of the radioactive waste drums released during operation of Dalat Research Reactor, Dalat City, Vietnam, the experimental study are carried out to confirm the calculation results of the systematic errors and evaluate the performance of this technique in practice. The results are also basic for establishment of measuring procedure for assay the radwaste drums. In this paper, the results of homogenous matrix and point source in different positions in the drum are presented.

In the experiments, a standard drum was used. Two point sources Co-60 of 0.36 MBq and two point sources Cs-137 of 0.64 MBq were located at four specified radial positions in the different measurements as in Fig. 1. Many pieces of matrix with various shapes and densities are made from clothing materials. The linear attenuation coefficients are in range 0.011- 0.031 cm^{-1} corresponding to gamma rays emitted by Cs-137 and Co-60 sources. A High-purity germanium detector- Gem 50P4, and a standard digital spectrometer, model DSPEC.JR-312-Ortec, with GAMMAVISION 32 version 6.1 and COLEGRAM version 2.01 were used for detection and analysis of gamma spectra. The results are given in Table I. Here the experimental and calculated values of errors are presented as ratio of the experimental and calculated to true value of activity.

The experimental results confirm prediction of the theoretical results of systematic errors of this technique. When the activity concentrates as a point source the systematic errors are minimum and independent on the attenuation coefficient of matrix as given in Table I.a. In these cases the errors depend only on the position of the sources. The distance between the centre of the drum and the detectors is equal to many times the half of the height of the drum. Thus, the geometric mean of the efficiencies of the sources at different positions of vertical axis can be considered as the same. When the height of drum is 86 cm, and detector- to –drum is 120 cm the maximum variation is less than 6 %. This is proved in case of the source in position 5. Therefore, the vertical count rate variation can be ignored. The errors are caused mainly by the horizontal inhomogeneity of the source in the drum as case of two sources with the same activity in position 1 and 3 (see Table I.b).

3. Conclusion

The experimental results confirm performance of this gamma technique for assay of radioactive waste in practice. The maximum systematic errors are not larger than about 70% when the attenuation coefficients of matrix are in the range of 0.01-0.03 cm^{-1}. The maximum error (~ 41%) of this method is larger than that (~ 20%) of SGS techniques. However its measurement apparatus is very simple, perform of measurement is faster, and it can be used for most of the situations encountered in practice.

TABLE I

a) Comparison of the experimental and calculated values for estimation of the systematic error when a point source in different positions.

Case of study		*One source in position 1*		*One source in position 2*	
Gamma Energy (kev)	Linear attenuation coefficient (cm^{-1})	Experimental value of error	Calculated value of error	Experimental value of error	Calculated value of error
661	0.016	0.97 ± 0.01	0.99	0.91 ± 0.02	0.95
661	0.024	0.92 ± 0.01	0.95	1.04 ± 0.02	0.91
661	0.031	0.92 ± 0.02	0.91	1.01 ± 0.02	0.87
1173	0.013	0.99 ± 0.01	1.00	0.90 ± 0.02	0.96
1173	0.014	0.99 ± 0.01	1.00	0.88 ± 0.02	0.96
1173	0.023	0.96 ± 0.01	0.95	0.91 ± 0.02	0.91
1332	0.011	0.99 ±0.01	1.01	0.90 ± 0.02	0.97
1332	0.013	1.00 ± 0.01	1.00	0.91 ± 0.02	0.96
1332	0.019	0.97 ± 0.01	0.97	0.91 ± 0.02	0.93
1332	0.021	0.96 ± 0.02	0.96	0.93 ± 0.02	0.92

TABLE I (continued)

b) Comparison of the experimental and calculated values for cases of large systematic errors.

Case of study		*One source in position 5*		*Two sources in position 1 and 3*	
Gamma Energy (kev)	Linear attenuation coefficient (cm^{-1})	Experimental value of error	Calculated value of error	Experimental Value of error	Calculated value of error
661	0.016	0.81 ± 0.02	0.85	1.15 ± 0.02	1.21
661	0.024	0.87 ± 0.02	0.81	1.27 ± 0.03	1.30
661	0.031	0.76 ± 0.02	0.77	1.38 ± 0.03	1.41
1173	0.013	0.82 ± 0.02	0.87	1.16 ± 0.02	1.18
1173	0.014	0.81 ± 0.02	0.86	1.15 ± 0.03	1.19
1173	0.023	0.78 ± 0.02	0.81	1.31 ± 0.03	1.29
1332	0.011	0.80 ± 0.02	0.88	1.14 ± 0.02	1.17
1332	0.013	0.81±0.02	0.87	1.25 ± 0.03	1.18
1332	0.019	0.90 ± 0.02	0.83	1.17 ± 0.03	1.24
1332	0.021	0.88 ± 0.02	0.82	1.30 ± 0.03	1.26

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