



**18th Conference of the
International Group on Research Reactors
&
IAEA Workshop on Safety Reassessment of
Research Reactors in Light of the Lessons
Learned from the Fukushima Daiichi Accident
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Book of Abstracts

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IAEA Activities to Support Sustainable operation of and Access to Research Reactors

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Abstract. Research reactors (RRs) have strongly contributed for more than six decades and continue contributing to the development of nuclear science and technology programmes in IAEA Member States (MS), including nuclear power programmes. The sustainability of their life-cycle is an issue of major relevance and MS are increasingly seeking IAEA's assistance in addressing the main challenges related to RR sustainable operation, including effective utilization, as well as in building new and accessing existing RRs for developing their national nuclear programmes and strategies, including human capital development.

The presentation will provide an overview of IAEA's activities and recent developed tools to provide assistance to MS in addressing the issues related to life-cycle sustainable operation of research reactors and access to research reactors. This includes assistance to: (1) improve operational performances and to address long-term operation issues, (2) strengthen capability to deal with fuel cycle issues, (3) enhance capability for provision of products and services and stakeholders' involvement, (4) develop the national nuclear infrastructure in MS embarking in new research reactor projects, (5) build and/or preserve national capacity, including human capital development, for nuclear science and technology programmes, through access to research reactors.

**The CEA Scientific and Technical Offer as a Designated ICERR
(International Center based on Research Reactor) by the IAEA:
First Feedback with the Prime Affiliates**

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Abstract. The IAEA Director General has approved on September 2014 a new initiative, namely the IAEA designated International Centre based on Research Reactors (ICERR), which will help Member States to gain access to international research reactor infrastructures. In fact, for the agency, one of the main goals of this ICERR scheme is to help Member States, mainly without research reactors, to gain timely access to research reactor infrastructure to carry out nuclear research and development and build capacity among their scientists.

CEA has decided to be candidate to its designation as an ICERR and consequently has established a candidacy report following criteria given by the IAEA in the Terms of Reference (logistical, technical and sustainability criteria). The CEA offer is covering a broad scope of activities on the 3 following topics:

- Education & Training
- Hands-On Training
- R&D Projects.

The perimeter (facilities and associated scientific and technical skills) proposed by CEA on this ICERR is centered on JHR project; its future international Material Testing Reactor under construction in Cadarache. Ancillary facilities in operation proposed in this offer include:

- ORPHÉE research reactor in Saclay, neutron beams reactor used for science, academic and industrial research, training and education to the use of neutrons scattering;
- ISIS EOLE and MINERVE zero/low power reactors located in Saclay and in Cadarache, dedicated to Core Physic and Education & Training in nuclear engineering; and
- LECA-STAR and LECI hot laboratories for fuel and Material Post Irradiated Examination, located in Cadarache and in Saclay.

The designation was the result of a rigorous process, including the review of the application and support documentation, an audit mission performed at the CEA sites, as well as a comprehensive evaluation and recommendation by an international selection committee made up of representatives from the global research reactor community and IAEA staff.

CEA Cadarache and Saclay centers are the first designated ICERR by the agency; this has become official during the last General Conference on the 14th September 2015. The Director General of the agency indicated the agency motivations at a ceremony during which he awarded the designation to CEA: *“Such centers will enable researchers from IAEA Member States, especially developing states, to gain access to research reactor capabilities and develop human resources efficiently, effectively, and, probably, at a lower cost. The ICERR scheme will also contribute to enhanced utilization of existing research reactor facilities and, by fostering cooperation, to the development and deployment of innovative nuclear technologies”.*

Following this designation, CEA has established a generic template as an agreement to be signed between CEA and any institutes, organization from Member State wishing to become Affiliate to CEA through this ICERR Scheme (it is question here of a bilateral agreement, the IAEA being only a facilitator). This template indicates rights and duties of both parties willing to collaborate through this ICERR scheme.

The 3 first Affiliates to CEA signed this agreement in September 2016 (JSI from Slovenia, CNSTN from Tunisia and CNESTEN from Morocco) followed by 3 others Affiliates during the first semester of 2017 (BATAN from Indonesia, COMENA from Algeria and JAEC from Jordan). Some first scientific and technical topics are now going-on giving some concrete examples of collaboration.

This paper presents in detail the CEA offer as an ICERR, the template agreement and describes, as examples, some first scientific and technical actions recently launched with the Affiliates.

RIAR As IAEA ICERR: Pilot Technical Cooperation Projects and Future Prospects

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Abstract. JSC "SSC RIAR" is the largest Russia's and world's research centre able to carry out irradiation and post-irradiation examinations over the whole range of existing nuclear power technologies and innovative reactor concepts.

JSC "SSC RIAR" operates the world's largest fleet of research reactors and experimental nuclear facilities (five research reactors: MIR.M1, SM-3, BOR-60, RBT-6 and RBT-10/2, pilot demonstration boiling power reactor VK-50, two critical assemblies).

The RIAR's Reactor Materials Testing Complex (more than 50 hot cells and heavy-shielded boxes) is equipped with modern experimental and analytical devices to conduct a wide range of non-destructive and destructive assays of any fuel and structural materials (including examinations of irradiated items with activity achieving $1.9 \cdot 10^{16}$ Bq).

The RIAR's research & scientific activity focuses on the experimental provision of the Russia's nuclear engineering development program and performance of research for a wide range of foreign customers.

The designation of RIAR as the IAEA International Centre based on Research Reactors (ICERR) in 2016 marked the worldwide recognition of RIAR's unique competencies and broadest experimental capabilities and confirmed the readiness of RIAR's infrastructure and specialists for further expansion of both international and bilateral technical cooperation with foreign partners.

The IAEA ICERR covers unique and the most-in-demand research reactors (loop-type reactor MIR.M1, high-flux reactor SM-3 and the world's only fast test reactor under operation BOR-60), reactor materials testing complex and research facilities under construction (including the poly-functional radiochemical complex).

In addition, at the disposal of foreign customers and international partners, JSC "SSC RIAR" has engineering and research infrastructure as well as a developed cycle of fuel supply and SNF and RW management (including handling of minor-actinides and production of advanced nuclear fuels).

The paper presents detailed outline of the experimental capabilities and competencies of JSC “SSC RIAR” as well as maps out prospects and sets forth specific proposals on the development of the international technical cooperation and implementation of joint research projects on the basis of JSC “SSC RIAR” as IAEA ICERR.

The First 50 Years of Operation of the ATR at the Idaho National Laboratory

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Abstract. The Advanced Test Reactor (ATR) is a key nuclear facility at the Idaho National Laboratory (INL). This year, ATR celebrated 50 years since initial criticality was achieved on July 2, 1967. This milestone is significant because of the many nuclear materials research programs supported over the past five decades but also as a waypoint because ATR is expected to continue operation out to at least 2050. The strategy for long-term operation includes a comprehensive age and reliability management program and engagement with government funding agencies to support the program. When considering the operation of a reactor facility beyond 80 years it is important to look beyond those systems required to simply keep the reactor operating and include inspections and condition evaluation of support infrastructure. This infrastructure evaluation must include water sources, waste disposal pathways, electrical substations and distribution, support facilities, and roads. ATR is evaluating reactor and non-reactor systems and executing a long-term operation and age management plan that may serve as a model for other facilities considering long term operation.

A Qualitative Study for Establishing the Conditions for the Successful Implementation of Public Private Partnerships in Research Reactor Projects in Newcomer Countries

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Abstract. The UAE is currently developing a peaceful nuclear energy program as part of its low carbon energy strategy to meet future energy demands. Research of nuclear energy technologies is required to support nuclear energy generation projects and maximize their performance. Research of this type will require building and operating a research reactor (RR), a costly undertaking in most circumstances. Collaboration between government and private parties through public private partnerships (PPP) can maximize the benefits expected from the adoption of a RR project. The aim of this research is to establish the conditions for developing a RR project for newcomer countries, with the UAE taken as a case study, through the utilisation of public private partnerships (PPP). The results of this study were arrived at through the use of semi-structured interviews conducted with 10 experts in the field of research reactors, using grounded theory method. Ineffective project initiation work was found to be the main causal condition influencing the success of PPPs in research reactors, governmental and political interventions were the intervening conditions, the local/regional justification and viability were the contextual conditions.

Keywords: Public Private Partnerships (PPP); Research Reactors (RR); Grounded Theory (GT).

IAEA Activities on the Safety of Research Reactors: 2017 Update

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Abstract. The IAEA conducts a broad range of activities to enhance the safety of research reactors worldwide. The objective is to support Member States in achieving and maintaining a high level of safety throughout all the stages in the lifetime of a research reactor. The focus is on improving regulatory effectiveness and enhancing the management of the safety of the facilities through the effective application of the Code of Conduct on the Safety of Research Reactors.

One of the key IAEA activities is the development of safety standards and supporting technical publications. These standards reflect international consensus on what constitutes a high level of safety, cover all areas important to the safety of research reactors, and form the basis of the IAEA safety review services for research reactors, including Integrated Safety Assessment for Research Reactors (INSARR) missions, safety reviews and expert missions. The IAEA safety standard on the safety of research reactors was revised and published in September 2016 as IAEA Safety Standards Series No. SSR-3, Safety of Research Reactors. While most of the technical requirements are largely unchanged, SSR-3 covers subcritical assemblies, addresses the interface between safety and security and incorporates the relevant feedback from the accident at the Fukushima-Daiichi nuclear power plant.

The IAEA also supports capacity building, disseminates operating experience, and promotes information networks and the exchange of operating experience. This is done through training workshops and technical meetings, group fellowship training courses and coordinated research projects. The IAEA operates the Incident Reporting System for Research Reactors (IRSRR) and organizes regular meetings for the exchange of operating experience. The IAEA supports Member States embarking on new research reactor projects to develop the necessary safety infrastructure through advisory services, expert missions and publications on implementing new research reactor projects.

This paper describes the safety issues and challenges for research reactors and the IAEA activities to enhance the safety of research reactors. Recent progress and achievements in improving research reactor safety worldwide and the strategy for implementing further safety improvements are also presented.

New Safety Requirements Addressing Feedback from the Fukushima Daiichi Accident

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Abstract. It is well known that regulatory body activities are constantly aimed at improving the safety of the supervised nuclear facilities. These generally include amending safety requirements, implementation of licensing and periodic safety reviews (PSRs), and inspections, and this was common practice of many states prior to the Fukushima Daiichi NPP accident (hereinafter – F-D accident). The F-D accident experience has revealed a wide range of new areas for safety improvement, most of which are applicable also to research reactors (RR), especially those, subjected to extreme external events. It was proposed that the RR safety regulations should be supplemented with new safety requirements which addressed among others: increasing of the original beyond design basis accident groupings; strengthening emergency preparedness and response (EPR) infrastructure; ensuring effective communication procedures. Rostekhnadzor is making improvements ranging in scope from significant to minor changes in its day-to-day processes. A number of amendments to the regulatory framework have already been made or are in progress, including requirements for PSRs. There is a need to improve arrangements on emergency exercises, which so far have involved preferably local emergency services with minimal involvement of external response organizations. The supplementary safety assessments based on lessons learned from the F-D accident showed the need to strengthen role and capabilities of the regulatory body in EPR, and this has been done as a part of the overall improvement of EPR system at the national level. It has also been recognized that application of the graded approach to safety and safety assessments is reasonable and appropriate, but there is a need in practical guidance on grading performance. The main aspect of using the graded approach in safety requirements is the classification process for reactors, systems, structures and components (SSCs), facility modifications, and procedures. The paper will focus on new safety requirements for research reactors in the regulatory framework of the Russian Federation, which have been developed in consistence with the provisions of the Code of Conduct on the Safety of Research Reactors and the IAEA Safety Report Series No. 80 (SRS-80).

Some Thoughts on Operator Intervention Arising from Safety Reassessments of Research Reactors in Light of the Fukushima Daiichi NPP Accident

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Abstract. Many research reactors have undertaken a safety reassessment of their facility in the light of the Fukushima Daiichi NPP accident, often using the guidance contained in IAEA Safety Report Series No.80. Such safety reassessment have often resulted in revisions to the training and qualification programme for the reactor operating personnel to cover the operator's response to beyond design basis events as well as to ensure that operators are able to fully recognize the potential for an event to be beyond the design basis and to respond effectively. However, an extension to this is what happens if the reactor operating personnel do not recognise or respond to a beyond design basis event, particularly slow evolving events?

This paper considers this issue and identifies some suggestions regarding claiming intervention by reactor operating personnel in response to beyond design basis events (or design extension conditions as they are now referred to) through examples for a generic pool-type research reactor.

Regulatory requirements to carry out *Complementary Safety Assessments* and to implement *Hardened Safety Core* provisions in Light of the Lessons Learned from the Fukushima Daiichi Accident

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Abstract. Following the Fukushima Daiichi Accident, ASN has established regulatory requirements for all the Operators to carry out a *Complementary Safety Assessment* for each nuclear facility in line with the European Stress Tests and the French Prime Minister request. It was expected that this process would re-evaluate the available safety margins beyond design basis and, identify potential *Cliff-Edge Effects* and *Feared Situations* following an extreme event. Extreme events, such as natural hazards significantly higher than design basis, lead to the loss of heat sink and/or electrical supplies and, affect several nuclear facilities on a same site. Then, regulatory requirements have been introduced to deal with potential unacceptable consequences resulting from these extreme events. The strategy includes two options. The first option is to decrease the level of hazard in the nuclear installations by reducing the amount of nuclear or hazardous substances. The second option is to provide the demonstration to reach a safe state for the nuclear installation following an extreme event. This demonstration should be supported by a set of existing or new provisions withstanding to extreme events with a high level of confidence, the *Hardened Safety Core*. In January 2015, ASN has established in its resolutions the basis of the implementation of the *Hardened Safety Core* for the Research Reactors, the Fuel cycle Facilities and the associated nuclear sites. They set specific requirements for the design of the *Hardened Safety Core* provisions and deadlines for their implementations. To reach a high level of confidence for levels of natural hazards with a very low annual frequency of exceedance is a real challenge. It is not always supported by available data and guidance. In order to comply with deadlines set in the regulatory programme, the stakeholders have assumed uncertainties and defined framework as workshop, to progress on these topics. The experience feedback from this programme would be useful in the context of the development of the Design Extension Conditions and the related safety requirements.

Present Status of Kyoto University Research Reactor, KUR

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Abstract. By reflecting the lessons learned from the accident of TEPCO's Fukushima-Daiichi Nuclear Power Plant which occurred on 11th March, 2011, the Nuclear Regulation Authority (NRA), Japan has formulated the new law regulating the nuclear facilities including the research reactors. Then, all the research reactors in Japan had to temporary shutdown, and they must have the safety review by NRA under the new law.

For the Kyoto University Research Reactor, KUR, we also shut it down in May, 2014 and then had the safety review by NRA. In September, 2016, we got the new license for KUR. After that, we have made refurbishment of the facility to comply with the safety requirement under the new regulation. Then, in the end of August, 2017, we finished the refurbishment works and the inspections by NRA, and we have started KUR operation.

In this report, we will describe the NRA's new regulation for research reactors and our response to it for the case of KUR.

Experimental Devices in Jules Horowitz Reactor and First Orientations for the Experimental Programs

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Abstract. This paper describes the experimental capacity and the status of the development of the test devices in the Jules Horowitz Reactor (JHR), under construction at the CEA¹/Cadarache in France.

JHR is a MTR whose design allows performing 20 experiments simultaneously with a large range of neutron fluxes and neutron spectra. The first set of test devices is dedicated to LWR: MADISON, ADELIN and LORELEI for fuel studies respectively under nominal, incidental (power ramp), and accidental (LOCA) conditions; MICA, OCCITANE, for material studies (behaviour under representative thermal conditions, neutron fluxes and possibly under stresses), respectively for SS and/or Zirconium alloys and for pressure vessel steel; and CLOE for IASCC (Irradiation Assisted Stress Corrosion Cracking). Other test devices are under conceptual design.

The paper describes the performances of these test devices, and their status of development. The guideline for the construction of the experimental programmes is also expounded, for fuels (with a priority for LWR, for basis properties and for the behaviour under incidental – accidental situations) and for materials (for claddings, reactor pressure vessel steel, internal and absorbers).

Keywords: Jules Horowitz Reactor, experiment, test device.

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Research Reactor Design Drivers

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Abstract. This paper shows the experience of the authors on participating in the design of several research reactors for more than 30 years.

The design drivers for research reactors have been changing over the years as it is demonstrated by the leading projects envisaged by or executed in several countries. Setting aside the widespread deployment of successful standard designs such as the TRIGA reactors, the unique design requested by some countries for their national facilities, defined to be aligned with particular local interests, demonstrates the evolution of the main design drivers along the years.

In general terms, in the earliest research reactor's projects, the focus was set in developing national infrastructure, including wartime objectives. With time, the need for research activities involving thermal neutrons, thus requiring graphite thermal columns and beam tubes, led to low and mid power configurations implemented in universities or colleges pool-type facilities.

Simultaneously, high power facilities were also built for providing services in the development of new fuels and materials required for expanding nuclear power programs. Nowadays, much of these facilities underwent large-scale refurbishment processes for attending niches never envisaged by the original designers, thus also demonstrating the evolution of their design drivers.

In recent years, the tendency of having multipurpose facilities dominated many projects, aiming at attracting support from various communities (scientific, medical, industry), which would be also included in funding the project. Commercial applications such as the production of radiopharmaceutical drugs, with their associated revenues, compelled to develop particular designs oriented at supporting a business endeavour. This paper analyses how factors such as the availability of funds, the demanding licensing processes or the attractiveness of certain applications are dominant design drivers in current projects.

Finally, the impact that present day operating facilities have in the design of future installations is discussed under the light of strategic alliances and emerging markets.

Contributions of Previous Projects to the Design of New Research Reactors

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Abstract. For the successful achievement of a new research reactor design, it is necessary to meet on one hand the project specifications and on the other hand Safety Authorities requirements. Customer needs are usually centered on a common set of applications, however their balance is tailored to each specific project, hence resulting in a dedicated specification.

Safety Authorities have implemented in their own way the requirements of the international framework in the applicable national regulation. Currently, there is an increase in their expectations as regards the implementation of defense in depth, the robustness against internal and external hazards and qualification requirements of SSCs.

The common traits of customer’s and safety authorities’ expectations are to decrease the risk:

- as regards utilization and operation performances and the smooth project achievement
- as regards potential consequences for the operators, the public and the environment.

Both are expecting up to date and fitted to purpose practices while using a proven design, qualified SSCs as well as state-of-art qualified methodologies.

Both customer’s expectations and safety requirements are progressing and constantly pushing the designer to challenge his solutions while keeping the best level of proven design.

Meeting all these expectations at once is a big challenge for designers.

To address this issue, the paper presents and illustrates on the basis of TechnicAtome (formerly AREVA TA) practices how the past and ongoing project experience is implemented in the design of new research reactors.

The topics of concern are:

- the reactor overall architecture as regards utilization and operation performances
- the experimental devices ensuring experimental and production applications
- the SSCs ensuring utilization, operation or safety functions as regards their proven design characteristics and qualification requirements
- the methodologies, approaches and tools as regards their qualification.

The way to address these issues is discussed in the paper showing the cross-cutting contributions of past and current projects and are illustrated with some examples in the light of AREVA and TechnicAtome experience.

Commissioning of the Jordan Research and Training Reactor (JRTR)

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Abstract. The Jordan Research and Training Reactor (JRTR) is an open tank-in-pool type reactor with a downward core flow during normal operation. JRTR is built on the campus of the Jordan University of Science and Technology (JUST) to be a hub for excellence in nuclear sciences and technology in the region. JRTR is a multipurpose, 5 MWth upgradable to 10 MWth reactor. Currently, the JRTR is in the operational phase. Prior to the start of JRTR operation, a set of commissioning tests have been performed. The normal purpose of the commissioning process is to verify that systems and components of research reactors and fuel cycle facilities, after they have been constructed, are made operational and meet the required safety and performance criteria.

In the commissioning process of the JRTR, the IAEA safety guides NS-G-4.1 has been followed. As recommended in the IAEA safety guide, JRTR commissioning process was divided into three main stages with hold points at the end of each stage. These stages are; tests prior to fuel loading, fuel loading tests and initial criticality tests which include low power tests; and the last stage constitutes power ascension tests and power tests up to rated full power. These stages have also been divided into sub stages. The performed tests have proved that all design and performance parameters have been achieved. For instance, the thermal power of 5 MW, maximum thermal neutron flux of 1.5×10^{24} (n/cm²·s) and negative reactivity feedback have been achieved. This paper describes each commissioning stage of the JRTR and the final results and conclusions.

Completion of Jordan Research and Training Reactor Construction Project

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Abstract. In June 2017, the Consortium of Korea Atomic Energy Research Institute (KAERI) and Daewoo E&C handed over the Jordan Research and Training Reactor (JRTR) to the Jordan Atomic Energy Commission, the owner and operator of JRTR. This official hand-over took place in the end of almost seven-year long endeavour since the project had launched in August 2010. This paper presents the chronology of the JRTR construction project along with focal points to which the Consortium paid special attention. For the chronology, we divide the whole period of project into three stages: a design stage, a stage of construction in a narrow sense including manufacturing and installation of equipment, and a commissioning stage. The design stage was three-year long from launching of the project to the issuance of Construction Permit (CP) in August 2013. The Consortium focused on the design that meets international safety standards while meeting performance requirements. In practice as agreed with the owner, we adopted primarily the Korean regulation, codes and standards. Those conform to international standards including IAEA's guides and requirements. The construction stage took about two years and half from CP until the start of Site Acceptance Tests in December 2015. One of the focal points was the minimization of non-conformity to the design. Lastly, the commissioning stage took a year and half: four months for so-called cold commissioning until the fuel loading in April 2016 and another fourteen months for nuclear commissioning and for fulfilment of contractual obligations until the official hand-over. During the commissioning stage, the Consortium paid special attention to not only confirming the completeness of the construction but also enhancing the operation capability of Jordanian staff. KAERI is proud of the successful completion of JRTR construction. Furthermore, it is continuing collaboration with Jordan in the areas of operation and utilization of JRTR such as RI production technique.

Preliminary Results of In-Core Irradiation Tests of Fluoride Salt and Materials at the MIT Research Reactor

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Abstract. Fluoride salts are a promising coolant option for advanced nuclear reactors because of their high volumetric heat capacity, thermal conductivity, melting and boiling temperatures at atmospheric pressure, and chemical stability. One particular reactor concept, the fluoride-salt-cooled high temperature reactor (FHR), combines successfully demonstrated technologies from other innovative reactor designs in order to expedite its required time to commercialization. The baseline FHR concept combines a fluoride salt coolant called FLiBe (2:1 mixture of LiF and BeF₂), with a graphite-matrix, coated-particle fuel. Three in-core irradiation experiments have been performed at 700_C which marks the first demonstration of FLiBe irradiation capability at the 6 MW Massachusetts Institute of Technology Research Reactor (MITR). The irradiation tests are part of an ongoing joint research program being conducted at universities including MIT, the University of California-Berkeley, and the University of Wisconsin-Madison. The objective of the overall research program is to develop a path forward to a commercially viable, fluoride-salt-cooled, high-temperature reactor (FHR). The objectives of these FHR irradiation experiments are:

- (1) to assess the corrosion and compatibility of proposed FHR materials 316 stainless steel, Hastelloy[®] N, SiC and SiCf/SiC composites, nuclear graphite, Cf/C composite, and surrogate TRISO fuel particles in molten FLiBe,
- (2) to measure the fast neutron activation products ¹⁶N (t_{1/2} = 7.1 s,) and ¹⁹O (t_{1/2} = 26.9 s,) that are significant radiation dose contribution in the gas phase, and
- (3) to examine the partitioning of tritium, produced from neutron interactions with flibe, among the various media in the experiment.

New irradiation facilities were specifically designed to ensure the success of these high-temperature in-core molten salt corrosion tests. Following the in-core tests, a serial of post-irradiation examinations (PIE) were carried out. Both Fe- and Ni-based alloys experienced 1000-hour continuous corrosion tests in MITR shows acceptable corrosion rates, 0.3 - 2.1 mg/cm² weight loss mainly due to Cr depletion through grain boundaries and grains. Additionally, the presence of graphite in molten FLiBe salt accelerated corrosion rate. Microstructure observations under scanning electron microscope (SEM) and transmission electron microscope (TEM) evidence the high-temperature corrosion attack in terms of deep intergranular corrosion. A large number of irradiation-induced Mo- and Cr-rich precipitates, and structure defects were also observed in both alloys. The PIE results show that SiC and graphitic materials are stable in high-temperature molten FLiBe with normal redox potential.

Some cracks were observed on the outer coating layer of tested TRISO fuel particles which is related to the combined effects of neutron irradiation and molten salt freezing cycle.

On-line Condition Monitoring Tool for Nuclear Research Reactors Coolant System Components

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Abstract. Machine condition monitoring is a world wide spread technology to improve predictive maintenance, availability, reliability and productivity. Additionally, in nuclear facilities, machine condition monitoring is an activity that could be used to improve nuclear safety. However, in experimental reactors, commercial technology for machine condition monitoring can be expensive and difficult to implement. This disadvantage leads to the development of an economic focused system capable of a complete on-line diagnosis for the principal rotating components of the experimental reactor, the nearby principal pipe and the decay tank. This requires development of sensors, electronics, software and specific knowledge, in order to obtain an early failure prediction system.

In RA-6 nuclear research school reactor, an off-line condition monitoring technology was developed in the early 80's during the startup and has been used since then. In 2008 and on-line and automatic condition monitoring system was developed and installed and continue reporting to this day. This system is being used as a platform for development of early failure detection techniques. The objective of this project was the development of an automatic condition monitoring system applied to the RA-6 primary coolant pump. The system is capable of performing the identification of the cause of the anomaly detected. Some typical problems in rotating machinery like unbalance, misalignment, shock and loss parts are identified by the system.

In this work a general description of the Condition Monitoring System is presented. Some result in anomaly detection issues and dynamical computations feedbacks are included. The system use anomaly detection algorithms, unsupervised machine learning, to determine unusual behaviour conditions and it is capable of monitoring these novelties and automatically generates a rule for detection using only relevant ranked features. These results show the potential of the on-line tool to react to early failure conditions, particularly in the primary coolant circuit including the nuclear core.

Ageing Management and Structures, Systems and Components Improvements at IRR1

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Abstract. The Israel Research Reactor #1 (IRR1) is a 5 MW swimming-pool type Research Reactor (RR), located at Soreq Nuclear Research Center. IRR1 is one of 144 RR around the world that were commissioned more than 40 years ago. During the last several years, IRR1 underwent several major improvements as part of an ageing management, to ensure continued adequacy of the safety level and reliable operation of the reactor. The improvement plan was executed as part of the long-life cycle plan. This comprehensive plan included: (a) improvements in safety and safety related systems, (b) a full review and reassessment, in view of Fukushima Daiichi accident, of the original Postulated Initiated Events (PIE's) that may lead to DBA\BDBA (and associated update to the Safety Analysis Report), and (c) extensive upgrades to mechanical and electric components including: replacement of the analog recorders and the control console with a modern system, installation of high quality accelerometers, modification of the emergency water supply system, enhancing remote monitoring to front command room, upgrading of the electrical power supply systems, replacement and improvement of the main valves, improvements in the Rabbit system, removal of unused utilities from the reactor, and more. Three independent processes contribute to the successes of the plan: firstly, professional and well-trained operators, safety culture in terms of honest reporting, events documentation, and debriefing; secondly, operation and maintenance strategy using routine maintenance and in-house reactor management software; and lastly, independent inspections (regulatory body and INSARR peer review). The objective of this paper is to share the collected experience and knowledge with other RR groups so that they may consider such structures, systems and components (SSCs) improvements in the context of their ageing management processes.

Improvements in the Safe Availability and Reliability of the OPAL Reactor

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Abstract. The organisational business objective for the OPAL Multipurpose Reactor of achieving safe but high availability operations, with a reliability greater than 98%, lead to the commencement of a journey to align our people, processes and decision making. Improvements in operating practices, risk management, long term planning, shutdown management and system engineering have supported the transition of the OPAL Reactor to a highly available, reliable and predictable Multipurpose Reactor. Future continuous improvement will be driven within the asset management framework.

The Impact of Changes in Utilization on Human Performance

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Abstract. The utilization of the High Flux Reactor in Petten has changed over the last decade. The primary focus of utilization was to conduct research in support of the nuclear energy industry. This has for a large part been replaced by the production of medical and industrial isotopes. This now takes up around 75% of the capacity of the reactor.

This change in utilization also changes the work of the reactor staff. Instead of working on one-off experiments that try to push scientific and technical limits, production of isotopes requires reliability and repetition, while keeping flexibility for the changes in market demand in product type and volume. The likelihood of the different types of errors that humans can make changes accordingly. This demands new requirements for the utilization process ranging from design to the management system and competences of personnel.

This presentation will focus on the process used to encompass human factor engineering during the HEU to LEU conversion of the production of Molybdenum-99 and the lessons learned during this process.

Progress in Conversion Study of the MIT Research Reactor from Highly Enriched Uranium to Low Enriched Uranium Fuel

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Abstract. The Massachusetts Institute of Technology Reactor (MITR) is a 6 MW research reactor operating with highly enriched uranium (HEU) finned plate-type fuel. It is designed as a multi-purpose research reactor for advanced nuclear materials and instrumentation tests using in-core irradiation facilities, neutron science research using neutron beam ports and research and education. It delivers a neutron flux comparable to light water reactors in the compact core, suitable for materials and fuel irradiation tests. The conversion study objective is to design a low enriched uranium (LEU) fuel element that could safely replace the current 15-plate HEU fuel element with longitudinal finned clad and maintain performance while requiring minimal changes to the reactor structures and systems. The LEU fuel matrix is a high-density U-10Mo monolithic fuel currently under qualification tests. Recent design analyses of alternatives to 0.25 mm clad finned LEU fuel plates have shown a 19-plate unfinned LEU fuel element with increased cladding thickness and thinner fuel meat thickness on the outer plates to be a feasible design. With the LEU fuel conversion, reactor power will increase to 7 MW in order to maintain thermal neutron flux. This paper provides an overview of MITR conversion study including the proposed LEU fuel element design, core neutronic and thermal hydraulic analyses, accident analyses, transitional cycle study, and impact of LEU conversion on in-core experimental facilities.

Observations on Experimental Fluid Structure Interactions of Plate-Type Fuel

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Abstract. The Materials Management and Minimization Program is pursuing the goal of converting all United States nuclear civilian research and test reactors from highly enriched uranium to low enriched uranium (LEU) fuel. At present 49 reactors have successfully been converted, leaving five reactors yet to have their fuel changed. These five reactors have much higher neutron fluxes than their counterparts and as such have been termed ‘high performance research reactors’ (HPRRs). Presently qualified fuel options in LEU composition would significantly reduce these reactors’ performance characteristics. As such, the Fuels Development Program led by the Idaho National Laboratory (INL) is presently working to qualify a new ultra-high-density LEU fuel composition consisting of a monolithic uranium molybdenum alloy. A necessary activity to support the qualification of this fuel is to characterize the prototypic fuel’s mechanical response under hydraulically loaded conditions. Oregon State University is working in collaboration with the INL to perform a comprehensive set of flow tests in a large-scale flow loop with this prototypic fuel as well as alternate materials. The element to be tested has been termed the Generic Test Plate Assembly (GTPA) as it is not representative of physical attributes of any of the HPRRs, but rather is designed to prescribe well-known boundary conditions on a series of flat test plates. This comprehensive testing campaign includes a statistically significant number of tests for each independent variable considered. In this case, three independent variables are investigated, the response of plates that (1) comprise aluminum 6061-T0, (2) are aluminum clad and have a surrogate material (stainless steel) dispersed within the fuel-meat region, and (3) are aluminum clad and have a monolithic DU10Mo fuel meat region. The objectives of this parametric testing effort are to (1) demonstrate the relative difference in mechanical response of each material type, (2) develop an experimental data-set that is of the quality, rigor, and resolution to support the validation of modelling tools, and (3) develop a better understanding of the mechanical instabilities of plate-type fuel under extreme hydraulic loading.

In order to satisfy these objectives the testing program is utilizing state-of-the-art instrumentation to characterize material properties, hydraulic boundary conditions, and mechanical response before, during, and after flow testing. While the testing campaign is early in its execution, a number of substantiated observations have been made that already shed light on objectives 2 and 3 detailed above. This paper presents the methods, outcomes, and observations made from these early NQA-1 compliant flow tests, demonstrating a successful observation of plastic deformation of this plate-type fuel and a basis for its occurrence when compared against previous computational studies’ results.

Feasibility Studies for Simultaneous Irradiation of NBSR & MITR Fuel Elements in the BR2 Reactor

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Abstract. The BR2 reactor is involved in preliminary neutronics feasibility studies for irradiation of four Design Demonstration Element (DDE) tests foreseen in the US High Performance Research Reactor (USHPRR) conversion program: MURR, MITR, NBSR and HFIR. The purpose of the irradiation will be to qualify the new LEU fuel material for each DDE Lead Test Assembly in the BR2 reactor at conditions that are similar for the reactor of origin. The present paper focuses on the preliminary feasibility scenarios for simultaneous irradiation of two DDE's: the MITR and the NBSR Lead Test Assemblies in the BR2 reactor. The MITR and the NBSR fuel elements will be loaded in the 200 mm diameter channels (H5 and H3). In order to 'mimic' the irradiation conditions for each DDE in the BR2 reactor as in the original reactors, a number of optimization scenarios are considered. The flux and burn-up calculations are performed using the fully automatic option of the MCNP6/CINDER90 code.

Keywords: LEU, conversion, MITR, NBSR, MCNP6

Modeling and Simulation of Dispersion Particle Fuels in Monte Carlo Neutron Transport Calculation

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Abstract. The dispersion particle fuel has advantages of high burnup, strong ability of containing fission products and good thermal conductivity. It is widely used as an advanced fuel element in next-generation nuclear reactors, such as the Molten Salt Reactor and High Temperature Gas Cooled Reactor. However, the dispersion fuel element in which the fuel particles randomly distributed in the matrix material presents new challenges for the conventional reactor physics methods. Based on the Monte Carlo method, the sub-fine lattice method for modeling of dispersion particle fuels in neutron transport simulation was studied, and the impacts on the modeling efficiency and calculation accuracy of the lattice grid sizes were given. The numerical results showed that the sub-fine lattice modeling with optimal lattice size can meet the requirements of reactor physics simulations for the dispersion particle fuel.

Keywords: Monte Carlo, particle fuel, sub-fine lattice model, reactor physics

Study of an Integrated Passive Safety System for a Research Reactor

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Abstract. An integrated passive safety system for a research reactor was suggested to improve the safety of the research reactor. This integrated system has three roles in a facility as a decay tank, siphon breaker, and long-term cooling tank. To design and optimize the decay tank and the siphon breaker of the integrated passive safety system, CATIA program was used to design 3D model, and flow fields were simulated by ANSYS Fluent. From the simulation result we could satisfy the design requirement of the decay tank, that is 60s minimum flow residence time. Also, the performance of new type siphon breaker was tested. An 18-inch diameter size siphon breaker was designed at the roof of decay tank's 3rd section, and we could observe siphon breaking phenomena that can prevent a severe accident of the research reactor. By locating siphon breaker at 3rd section of decay tank, we could also use the coolant of the front three sections for a long-term cooling of the research reactor.

Secure Enterprise Integrations for Multipurpose Research Reactors

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Abstract. The OPAL research reactor operated by the Australian Nuclear Science and Technology Organisation (ANSTO) is a multi-purpose scientific and manufacturing facility. Information produced by the reactor's operational technology (OT) systems is relied upon by engineering, scientific, and manufacturing information systems.

We present an overview of the requirements for, and the techniques applied to achieve secure enterprise integration between the OPAL Research Reactor's Operational Technology (OT) and associated these information systems, while conforming to international and national guidance including:

1. Minimising what "information system" functionality is present on reactor OT systems;
2. Providing a controlled, uni-directional security gateway for data flows from the OT systems to the ERP; and
3. Limiting data entry to reactor OT systems to operator controlled barcoded paper forms using strictly defined data formats.

First Periodic Safety Review of the FRM II After 10 Years of Routine Operation

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Abstract. The FRM II is Germany's most modern and most powerful research reactor and is operated by the Technical University of Munich (TUM) on its research campus in the North of Munich. Due to its 20 MW thermal power and the very compact core it offers a very high neutron flux for more than 25 beam tube instruments and a set of irradiation facilities. The construction of the FRM II started in 1996 and the reactor achieved its first criticality in March 2004. In the nuclear commissioning phase the FRM II reached its nominal power of 20 MW in August 2004 and the startup cycle was finished end of October 2004. Routine operation was started with the beginning of the second reactor cycle end of April 2005.

According to the German nuclear energy act and a dedicated requirement from the operating license also the FRM II as a research reactor has to perform a regular periodic safety review (PSR) every 10 years. This PSR includes a full description of the facility, a deterministic safety status analysis, a probabilistic safety analysis and a security analysis. The PSR documents were sent in time to the regulatory body and its technical support organization (TSO) in May 2015. The paper presents the graded approach for the PSR for a research reactor in respect to the mandatory German regulations for NPPs, some results of the PSR and the status of the follow up process underway to finalize the PSR.

Status of a Periodic Safety Review of HANARO

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Abstract. The first Periodic Safety Review (PSR) for HANARO² is now being conducted to meet the mandatory requirements in accordance with the National Nuclear Safety Act, which was amended to impose PSR on the research and training reactors in Korea. The set of documents of the HANARO periodic safety review including all safety significant findings is supposed to be submitted to the regulatory body by the end of 2018. HANARO has been operating over 20 years for the radioisotope production, material irradiation, neutron transmutation doping, neutron activation analysis, and neutron beam utilization. Periodic inspections and In-service inspections have been fulfilled to ensure the safe operation of the facility and also a special safety review after the Fukushima Daiichi Accident was done to check if the facility could be in safe conditions by natural or external events such as an earthquake, flooding, loss of offsite electric power, station blackout, etc. But the whole comprehensive safety review for the reactor facility had not been performed before. Hence the ongoing periodic safety review will provide the first opportunity to obtain an overall view of the actual plant safety and the quality of the safety documentation, and to determine reasonable and practical modifications to ensure and improve the safety. In this paper, the implementation status of HANARO periodic safety review will be presented.

²Hi-flux Advanced Neutron Application Reactor

MARIA Reactor Safety Improvements and Research Capacity Update

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Abstract. Reactor MARIA is located in the National Centre for Nuclear Research, Świerk-Otwock (Poland). First critical experiment of MARIA has been done in 1974 and during past forty years the reactor has been significantly modernized. Actually every year brings some upgrading's of MARIA operational infrastructure. Recent years improvements are, among others, core conversion from Medium Enriched Uranium fuel elements (MEU: 36% of U²³⁵ enrichment) into Low Enriched Uranium fuel elements (LEU: 19.75% of U²³⁵ enrichment), primary cooling system enhancement – new main cooling pumps and new emergency cooling system with stand-by pumps, emergency power supply voltage converters replacement and implementation of the new irradiation facilities. Currently works are focused on every fuel element power measurement system replacement. Another works are planned in the short term inter alia diesel generators, beryllium blocks replacement and cooling towers renovations. In parallel reactor personnel is regularly trained to raise their qualifications and new employees are being engaged to secure safe and floating operation.

Thanks to systematic reactor safety improvements, MARIA reactor is perceived as a still modern European research reactor with the relatively long perspective of the operation. This allows to plan long strategy research objectives and ensure radiopharmaceuticals production using MARIA reactor for next decades. Indeed new research programs are performed and planned in the frame of international collaboration. Among all the most distinguish projects are:

- ✓ New types of fuel elements for testing,
- ✓ New 14 MeV neutrons irradiation channel with flux $> 1 \cdot 10^9 \text{ cm}^{-2} \text{ s}^{-1}$,
- ✓ New irradiation probes for material modification in high energy neutron fluxes,
- ✓ New irradiation positions inside fuel elements for high fluxes of fast neutrons,
- ✓ New methods of irradiation for holmium and other radioisotope production,
- ✓ New facility for neutron-boron therapy experiments,
- ✓ New instruments dedicated to neutron studies delivered by Helmholtz-Zentrum,
- ✓ New high gamma flux irradiation positions,
- ✓ Research on beryllium blocks poisoning,
- ✓ New molybdenum production methods,
- ✓ Nuclear instrumentation and measurement methods development,
- ✓ Development of nuclear calculation tools.

MARIA research reactor in spite of constantly broadening research activity, still has wide capacity for research update and to increase international cooperation. Fluxes densities inside MARIA reactor may reach up to $3 \cdot 10^{14} \text{ cm}^{-2} \text{ s}^{-1}$ (thermal neutrons) and maximal thermal power can be 30MW. Core height is around 1.1m (1m of fuel part) and it is placed in the open water pool at depth of 10.5 m. Fuel elements are separated from the pool by

pressurized channels with individual coolant temperature and flow measurement. This unusual construction allow to measure power distribution between all fuel elements by measuring coolant temperature before and after Fuel Assemblies (FA's). This power control system measure the individual fuel element power and burn-up during operation. In the same time all irradiation channels are available for irradiations and research. This feature gives wide possibilities of performing various experiments.

Safety Analysis For Prototype MNSR HEU Core Unloading And Storage

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Abstract. Prototype Miniature Neutron Source Reactor (MNSR), a low-power research reactor, was designed and fabricated by China Institute of Atomic Energy. It adopts HEU as fuel, beryllium as reflector, light water as moderator. The heat of reactor core is removed through natural circulation for cooling. There is one fuel cage with dimension of $\phi 241 \times 278$ mm in the reactor core, U-Al4 alloy a total of 376 rod fuel element. The activity of core is calculated by ORIGEN2 program according to the operation history of the prototype MNSR, and the temporary storage cask of the spent fuel (HEU) is designed according to calculating result of the core activity by MCNP program; The K_{eff} values are calculated at the different positions of spent fuel cage during the unloading of spent fuel cage with MCNP program and the accident critical safety analysis is carried out with the reactivity change during the unloading. Results show that the source term activity of reactor core fuel element is 5.22×10^{12} Bq after 12 months of shutdown, of which γ radioactive activity is 3.74×10^{12} Bq. K_{eff} value is less than 1 during the unloading of spent fuel (HEU) cage from the reactor core, which meets the requirements of radioactive safety. The inner dimension of the temporary cask is $\phi 280 \times 309$ mm, outer dimension is $\phi 540 \times 691$ mm, and the thickness of Pb is 90mm; the max value of γ rate on the cask surface is 0.54 mSv/h, which is less than 2 mSv/h.

Completion of Seismic Rehabilitation Project at HANARO after the Fukushima Daiichi Accident

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Abstract. Right after the Fukushima accident in 2011, KAERI immediately carried out necessary safety reconfirmation program in accordance with the government's safety reassurance policy. The Nuclear Safety and Security Commission (NSSC) of the Korean government formed a special nuclear safety inspection team with the participation of civilian experts. The team initiated an overall safety check-up program for all major nuclear facilities in the country including KAERI's HANARO. As a result of the safety mission, the NSSC requested KAERI to reassess the seismic qualification of the HANARO with particular emphasis on the reactor building and the stack.[1] The NSSC also recommended KAERI to develop necessary seismic reinforcement measures in order to reassure the safety of HANARO. The NSSC therefore, strongly recommended KAERI to reinforce an outer wall of the reactor building to protect the Reactor Concrete Island (RCI) more effectively from possible seismic issues. This paper is to report the completion of HANARO reactor building reinforcement project that began in 2015 as part of the HANARO safety reinforcing endeavours.

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Enhancement the Safety of the Jordan Research and Training Reactor (JRTR)

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Abstract. The JRTR is a multipurpose reactor designed and constructed to be used for education and training, research and radioisotope production. All safety aspects of the JRTR fall under the category of SC-3 according to the ANSI/ANS 51.1 classification system of nuclear reactors. For example, the Reactor Structure Assembly (RSA), Primary Cooling System (PCS), CRDM/SSDM, Reactor Protection System (RPS), Confinement Isolation Dampers, Siphon Breaking Valves and UPS are classified as SC-3 components.

However, in the wake of Fukushima-Daiichi accident, and learning the lessons of the accident and following the recommendations, the safety measures of the JRTR have been extensively investigated to enhance the safety of the reactor. Therefore, design changes of systems and equipment due to the reinforced international safety norm after Fukushima disaster, addition, expansion and modification of facilities to accommodate the design changes have been implemented. As a result investigation, several aspects of the JRTR safety have been improved. As examples of these, the quality class has been upgraded for several components such as Process Instrumentation and Control System (PICS), Radiation Monitoring System (RMS), Information processing System (IPS) and Operator WorkStation (OWS). Additionally, expansion and modification of facilities to accommodate systems and equipment have been applied. The seismic monitoring system has been improved by upgrading quality class and by adding a function generating the automatic seismic trip signal when a seismic motion exceeds Operating Basis Earthquake (OBE). Pool Liner Integrity has been enhanced by improving the welder qualification process and by enhancing the weld quality. Furthermore, the emergency conditions have attracted special attention. The emergency water storage capacity has been increased, and two mobile diesel generators have been placed in a building of seismic category I.

This paper describes the safety aspects of the JRTR and the improvements after the Fukushima-Daiichi accident.

Findings and Results of Safety Reassessments and Safety Improvements on the ORPHEE Research Reactor

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Abstract. The Orphée reactor is a 14 MWth research reactor located at the CEA center in Saclay, France. The main function of the reactor is to supply neutron beams for fundamental research. The reactor first went critical on December 19th, 1980.

Periodic safety review (PSR) and post-Fukushima stress tests have been conducted on the reactor over the past few years. Since the commissioning of the reactor, the Orphée reactor has gained experience in conducting PSRs and implementing safety improvements. The second PSR was released in 2009. The on-going third PSR will be submitted to the ASN in 2019.

The safety reassessments are realized to verify that the reactor is operated with a good level of safety, meaning that the reactor is in conformity and that the safety cases still meet the requirements even after the evolution of regulation and safety analysis standards. For the Orphée reactor, the findings from the conformity analysis and the safety reassessments have resulted in the implementation of both operational improvements and safety related equipment.

After the Fukushima accident, in respect with the regulatory requirements, the safety margins to reach and maintain safe state have been evaluated for extreme external hazards and relevant cumulative losses. For the Orphée reactor, a hard core of robust equipment has been defined for the monitoring of a safe state in extreme situation. The robust design of the Orphée reactor enables to withstand long time loss of heat sink or loss of electrical power thanks to passive residual heat removal design and important inertia of the water capacities. Regarding external hazards, the site of Saclay has a low seismicity risk. The safety improvements mainly consisted in the implementation of reactor emergency shutdown on seismic signal, implemented to strengthen the control rods drop in case of extreme seism. An ultimate emergency control panel with the report of minimal information for the monitoring of safe state has been installed in a local with sufficient seismic margins.

This paper will describe the relevant findings from the periodic safety reassessment and post-Fukushima stress tests, and safety improvements that were implemented on Orphée reactor to fulfil the requirements of safety levels from the regulatory body.

Safety Reassessment of German Research Reactors in the Light of the Accident at the Fukushima Daiichi Nuclear Power Plant – Current Status of the Improvements focused on Emergency Preparedness

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Abstract. In the light of the accident at the Fukushima Daiichi nuclear power plant, in summer 2011 the first comprehensive safety assessment was requested for all German research reactors with a continuous thermal power of more than 50 kW. The general approach for the safety assessment of research reactors was based on the stress test for nuclear power plants. The main goal was to verify the robustness of research reactors against severe conditions. The statement of the Reactor Safety Commission (RSK) identifying the robustness of the facilities and improvement measures was published on 3rd May 2012. In 2015, RSK performed a reassessment of research reactors verifying the implementation status of measures identified in the first assessment.

Assessment of Lessons Learned from the Fukushima Dai-ichi Nuclear Accident to Research and Test Reactors in the United States

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Abstract. The United States Nuclear Regulatory Commission (NRC) staff evaluated the thirty-one NRC-licensed research and test reactors (RTRs) to assess the applicability of lessons learned from the Fukushima Dai-ichi nuclear accident to these facilities. The NRC staff assessed two categories of RTRs based on the licensed thermal power level of each facility. Category 1 consisted of the twenty-six research reactors licensed to operate at power levels lower than 2 megawatts-thermal (MW_t). The remaining five NRC-licensed RTRs licensed to operate at 2 MW_t or higher comprised Category 2. The NRC staff's assessment concluded that all of the Category 1 and the two lowest-powered Category 2 research reactors were highly resilient to the loss of electrical power, active decay heat removal systems, and heat sink. These twenty-eight research reactors generate minimal decay heat, which can be adequately removed via air cooling to prevent fuel cladding failure. In contrast, the three largest Category 2 RTRs rely on the availability of water for adequate decay heat removal. As such, the NRC staff performed an additional assessment of these three facilities to determine the resilience of their primary coolant systems to a beyond-design-basis seismic event. For the 20 MW_t test reactor, the NRC staff also assessed the resilience of emergency power, active decay heat removal systems, and coolant make-up systems for flooding and beyond- design-basis seismic event. The results of the NRC staff's assessment revealed that the existing design-bases for NRC-licensed RTRs adequately protect against fuel cladding failures and the release of radioactive material during a beyond-design-basis external event.

The Construction Stage in the RA-10 Reactor Project

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Abstract. The RA-10 is a new multipurpose research reactor that is under construction in Argentina. It is a 30 MW thermal power reactor and it is designed to achieve high performance neutrons production to fulfil the stakeholder's requirements in compliance with stringent safety regulations. The principal objectives of the facility are to consolidate and increase the radioisotope production in order to cover future demands, to provide fuel and material testing irradiation facilities in order to support national technology development in this field, to offer new applications in the field of science and technology based on modern neutron techniques.

The project is supported by the National Administration and is conducted by the National Atomic Energy Commission (CNEA).

The construction stage was begun last year. Previously, the construction license and the environmental aptitude certificate were obtained and a social perception study was performed. The first concrete was built on last May 6. The reactor is planned to be operative in 2020.

The main ongoing activities developed during this stage involve the civil work construction, the industrial components manufacturing and mounting, the nuclear supplies and component provision and the development of the operation team. The plant documentation elaboration and preoperational tests preparation are planned for the next year. Regarding licensing, beside the upgrade of the Preliminary Safety Report and the commissioning preparation, the main objective is to assure and demonstrate that the licensee, and all safety and regulatory requirements, are correctly implemented.

According to the chosen model for the project organization, responsibilities were assigned to in-house groups and external companies within the frame of a few main contracts and internal agreements. The project organization had to be adapted for this stage reinforcing capabilities related to coordination, integration and controlling. CNEA has updated the project management system with specific procedures, modifying the corresponding quality system for handling this stage. The contractor's quality systems had to be reviewed and adapted. Specific schedules had to be integrated. The commitment with safety culture was reinforced by assuming that, beside the contractors' participation, the responsibility for implementing the license relies on the operating organization.

This paper describes the experience of managing the construction stage, particularly for the different involved activities, with emphasis on their interactions and the application of control tools.

JHR Project. Irradiation Devices. In-Service Inspection of Nuclear Pressure Equipment's. Investigation of Non Destructive Examinations for Inspection Purposes.

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Abstract. The Jules Horowitz Reactor currently being built at the Cadarache center in the south of FRANCE will be a modern Material Testing Reactor (MTR) designed to perform irradiation experiments while complying with today's safety, quality and regulatory requirements. The JHR Reactor will be used to irradiate fuels and materials samples under experimental conditions representative of current and future nuclear power plants. The facility will also be used to irradiate fuel targets (Mo99) for medical purposes.

The experimental irradiation loops to be installed in the reactor will generally comprise an in-pile section (device, underwater lines, pool penetrations) and an out-of-pile section (hydraulic cooling system, auxiliary systems, vessel, power distribution system, instrumentation & control command). These loops or experimental devices will operate under thermal hydraulic conditions that are representative of the reactor technology being studied (LWR, Gen IV). This implies operating the loop components under specific pressure and temperature conditions (155b, 320°C for PWR).

The use of nuclear pressure equipment's necessarily entails a number of periodic inspections. These inspections often required the prior disassembly of compartments forming this equipment so as to gain access to the different internal and external surfaces to be inspected.

Within the scope of irradiation devices designed for the Jules Horowitz Reactor (JHR) comprising both internal and external irradiated compartments separated by a thin gap filled with gas (5/10 mm), their periodic disassembly and reassembly in hot cells for soundness checks will be a complex operation.

To overcome such problems, the possibility of using non-destructive examination (NDE) techniques are investigated to obtain the data needed to appraise the soundness of such equipment and thus meet the inspection requirements laid out in the regulations.

Keywords: JHR, Irradiation devices, nuclear pressure equipment, In-service inspection, Non-destructive tests.

Thermal-hydraulic Conceptual Design of the New Multipurpose Research Reactor Succeeding to JRR-3

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Abstract. The core thermal-hydraulic conceptual design of the new multipurpose reactor succeeding to JRR-3 was performed for the forced convection cooling modes. The key criteria are first to avoid the nucleate boiling anywhere in the core and second to have enough safety margin to departure from nucleate boiling for normal operation conditions.

The results of the thermal-hydraulic conceptual design and analysis show that the optimum coolant velocity in the standard fuel element is about 6.0 m/s with the core inlet pressure of 1.49 kg/cm² abs at 30 MW of thermal power. At the coolant velocity of 6.0, the temperature margin against ONB temperature is 7.2 °C and the minimum DNBR is 2.3. The results obtained in this work establish the preliminary technical specifications for the core thermal-hydraulic design of the new multipurpose research reactor.

Monte-Carlo Coupled Depletion Codes Efficiency for Research Reactor Design

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Abstract. In the field of industrial studies, burnup calculations and associated neutronic parameters are most often determined using deterministic methods. Even though they are faster than Monte-Carlo methods and efficient concerning the determination of best estimate physical quantities, they however have drawbacks: these tools rely on certain conditions and reactor characteristics (such as energetic spectrum, lattice pattern, types of fuel, etc.) and are based on neutronic assumptions and numerical approximations. On the other hand, Monte-Carlo methods are not problem-dependent. They do not require approximations, all kinds of complex geometries can be modelled and the whole interaction processes can be treated with the best state-of-the-art knowledge. Due to flexibility and accuracy, Monte-Carlo burnup calculations are frequently used in R&D studies, where faster deterministic burnup calculations are more suitable for industrial studies (best cost/accuracy ratio). With the improvement of CPU power, Monte-Carlo codes ability for burnup calculation has to be tested in order to determine whether their performances are convenient for industrial studies. This paper proposes to perform a comparison between Monte-Carlo codes used by TechnicAtome (TRIPOLI-4[®] [1], MCNP6 [2] and Serpent [3]) and the deterministic scheme COCONEUT [4]. This comparison also provides the Validation and Verification process (V&V) undergone by COCONEUT [5]. This is carried out on standard fuel assembly (SFA), absorber fuel assembly (AFA) of a core with burnable poison and on the full 2D core. Such geometry is used by TechnicAtome to set new methodology studies. Parameters such as reactivity, isotope concentration and neutronic flux are studied on a burn-up calculation reaching around 100 GWd/tU per assembly. Excellent agreement is observed between all codes.

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INL Transient Reactor Restart Progress

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Abstract. Final preparations for restart of the TREAT are under way. The TREAT reactor was used from 1959 to 1994 to conduct more than 2,800 nuclear fuel transient tests, and was placed in standby in 1994. The plant was extensively upgraded shortly before it was placed in standby. The reactor systems and infrastructure were renewed as required, procedures and configuration management documents have been updated, and a qualified operating organization put in place.

The United States Department of Energy authorized operations of the Transient Reactor Test (TREAT) facility on August 31st, 2017. This authorization completed the reactor restart program, more than twelve months ahead of the baseline schedule and approximately \$20 million less than the baseline cost estimate.

A controlled approach to restart of the TREAT reactor is being implemented with actual startup anticipated by early 2018. Following reactor performance evaluation and physics testing the experimental programs will be initiated.

Keywords: TREAT, Transient Test Reactor, INL

MOLY Production in the Jules Horowitz Reactor: Capacity and Status of the Development

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Abstract. This paper describes the capacity and the status of the development of Moly production facility in the Jules Horowitz Reactor (JHR), under construction at the CEA (Commissariat à l'Énergie Atomique et aux Énergies Alternatives)/Cadarache in France. JHR will contribute to the security of supply of medical radioisotopes, especially for the ^{99}Mo - $^{99\text{m}}\text{Tc}$.

Four locations are devoted to the ^{99}Mo production in the JHR reflector. A dedicated cooling circuit is associated to the Moly devices. In the way to perform the design, several mock-ups have been manufactured since 2014. The process of industrialization began in 2015 for the execution studies and the construction of the in-pile part of the facility (movable systems, devices, safety cooling injection). The process of industrialization for the out of pile part (normal cooling circuit and I&C systems) and for the tooling will begin in 2018.

It is scheduled that JHR will start producing radioisotopes at the beginning of reactor operation (providing completion of the qualification of the irradiation process).

Keywords: Jules Horowitz Reactor, radioisotopes, nuclear medicine, ^{99}Mo , MOLY facility

Core Elements improvements for optimization of radioisotopes production in an MTR-type core

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Abstract. The primary purpose of research reactors (RR's) is to provide a neutron source for research in natural sciences, industrial processing and nuclear medicine [1]. The latter takes place in 25% [2] (80 facilities) of the research reactors around the world. The most common method for producing radioisotopes is by the neutron activation process. Due to the cosine shape of the flux along the axis of every fuel assemblies (FAs), a limited area of maximal flux makes the activation process of large or multiple samples less efficient. The objective of this study is to analyze two design parameters of MTR-type core components in order to optimize isotope production: (i) FA linear fuel distribution loading [4] [5] [6] and (ii) Irradiation Position body material. The study presents a 3-D MTR fuel calculation using Serpent [7]. Using the parameter improvements in this process, in terms of flattening and maximizing the thermal flux, the production rate doubles while keeping the safety parameters of the fuel thermal- hydraulics properties. This improvement can lead to better gains for society and will also reap financial rewards.

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Production of Medical Radioisotopes at the FRM II Research Reactor

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Abstract. Due to its design as a heavy water moderated reactor with a very compact core FRM II, Germany's most modern and most powerful research reactor, offers excellent conditions for basic research using beam tubes. On the other hand it is equipped with various irradiation facilities to be used mainly for industrial purposes. From the very beginning of reactor operation a dedicated department had been implemented in order to provide a neutron irradiation service to interested parties on a commercial basis.

As of today the most widely used application is Si doping. The semiautomatic doping facility accepts ingots with diameters between 125 mm and 200 mm and a maximum height of 500 mm. The irradiation channel is located deep in the heavy water tank and exhibits a ratio of thermal/fast neutron flux density of > 1000 . This value allows the doping of Si to a target resistivity as high as $1100 \Omega\text{cm}$ within the tight limits regarding accuracy and homogeneity specified by the customer. Typically the throughput of Si doped in FRM II sums up to about 15 t/year.

Another topic of growing importance is the use of FRM II aiming the production of radioisotopes mainly for the radiopharmaceutical industry. The maybe most challenging example is the production of Lu-177 n.c.a. based on the irradiation of Yb_2O_3 to a high fluence of thermal neutrons of typically $1.5\text{E}20 \text{ cm}^{-2}$. The Lu-177 activity delivered to the customer is in the range of 750 GBq. With respect to further processing it turned out to be a highly advantageous to have the laboratories of ITG, the company extracting the Lu-177 from the freshly irradiated Yb_2O_3 on site FRM II.

Further irradiation facilities are available at FRM II in order to allow the activation of samples for analytical purposes or to irradiate samples for geochronological investigations using the fission track technique. Finally a project on the future installation of a facility dedicated to the irradiation of U-targets for the production of Mo-99 is in progress.

It is noteworthy that all of the irradiation facilities at FRM II have been certified according to the ISO 9001:2008 standard.

Reactor Laboratory for Nuclear and Biomedical Research

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Abstract. Currently, access to methods using neutrons is very limited in the world. Lately, the nuclear reactor, OSIRIS in Saclay was closed (2015) and BER II in Helmholtz-Zentrum Berlin is under closure (2019) – this results in closing down the activities of associated neutron laboratories. Research using neutrons is important for the scientific community due to the large European programs: ESS (European Spallation Source) in Lund and the construction of an ENS (Early Neutron Source – EURATOM). The MARIA reactor is continuously modernized. Lately conversion of the fuel elements from medium to low enriched has been done. Moreover, primary cooling system has been enhanced, new emergency cooling system installed and emergency power supply voltage converters replaced. The future plans are: beryllium blocks replacement and cooling towers renovations. Current license of the MARIA reactor is valid to 2025 and another improvements enable to plan research until at least 2035. Furthermore, NCBJ works are being conducted on renovation of the measurement and research facilities for all eight horizontal channels and separation of the reactor hall from the experimental hall for the security and radiological safety reasons.

Scientific interest of reactor laboratory can be divided into two sections: horizontal (outside the core) and vertical (in-core) experiments. Epithermal neutron beam with the flux of $2 \cdot 10^9$ n/cm²/s is under certification by the authorities. The scientific program “Neutrony H2” is a large multi institutional project of international significance (institutions from over a dozen countries) in the field of biomedical and material research. One of the interests is boron neutron therapy (BNCT). Exposure of the biological material (or small and large animals) was dictated by the renaissance of this method in the world due to the development of demonstrators of therapeutic neutron generators at medical universities in Japan with the participation of Sumitomo and Mitsubishi (2015). Moreover, the refurbishment of the reactor experimental hall is due to the world-class research instruments worth 12 million \$ transfer from HZB to MARIA reactor (four modern built between 2000 and 2016 spectrometers and diffractometers). One of the spectrometers has already been transferred to the NCBJ, the other 3 will be delivered in 2018.

The second group of research is based on high fluxes densities inside the MARIA reactor core (up to $4 \cdot 10^{14}$ cm²/s of thermal neutrons). This ensure research on radiopharmaceuticals production: holmium (2016), yttrium (2015), molybdenum (2017), nuclear construction testing (fuel elements, beryllium blocks poisoning, material modification). Furthermore, a new irradiation position has been installed: channel for 14 MeV neutrons, high fluxes of fast neutrons inside fuel elements MR-2 and high gamma flux (large cubature).

Training of the First Operation Team

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Abstract. The paper shows the experience of the authors in dealing with the training of the first operation teams for research reactors.

Training of the First Operation Team (FOT) is one of the most important activities on the verge of commissioning and of starting the operation phase of a new research reactor. Additionally, as both the facility and the operating documentation are yet unproven, the actions for developing the skills of the future operators require an all-encompassing approach. Despite the common objective of preparing the human resources to command the facility in a safe and efficient manner, different approaches are available for ensuring a successful implementation. The following approaches, which have been implemented by different projects worldwide, are discussed in this paper: - Transference of the technical knowledge required for the specific facility to be commissioned. It is implemented when a seasoned staff is available from other facilities (as was the case, for instance, of the RA-8 in Argentina) thus only requiring: o Technical knowledge of the peculiarities of the facility to be operated o Highlight of the potential new risks o Familiarisation with the new procedures and instructions - Full training program. Applicable when newly engaged personnel is being appointed for operating the facility, as was decided for the ETTR-2 reactor in Egypt, or when no previous facility was available in the country, as was the case of the NUR reactor in Algeria: o Development of a Safety Culture framework o Leverage of basic skills among the staff members o Team building activities including the identification of leaders and the development of command lines o Transference of the scientific and tech knowledge required - Training the trainers. An encompassing approach where the training program and material are checked while very skilled personnel are trained, which will be, on their turn, delivering the training in the future. This approach was implemented for the OPAL reactor in Australia. Finally, the advantages of including training elements during the design of the facility are presented, including: - Availability of mock-ups for hands-on training - Documentation of the “why-not” design options - Training manuals, procedures and instructions

Current and Prospective Tests in Reactor MIR

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Abstract. The MIR reactor was purposely designed to perform loop tests of fuel rods and fuel assemblies of various types of reactors. Nowadays, there are seven loops used for testing (two loops are cooled with pressurized water, two loops are cooled with boiling water, two loops are cooled with overheated steam and one loop is gas-cooled). The current key activities are loop tests of VVER fuel rods to study their characteristics under the conditions simulating the normal operational ones, deviations from normal conditions and design-basis accidents as well as fission gas release from leaky fuel rods. Moreover, high-dense low-enriched fuel is tested under the RERTR program. The paper presents the current and prospective test programs for fuels and structural materials of various types of reactors, describes the experimental-methodical base of tests, and gives some test results.

Competence Development of Research Reactors Personnel in Indonesia

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Abstract. Development of human resources is an absolute requirement in order to support the efforts of nuclear power utilization and its supervision so that the utilization of nuclear power contributes in improving the welfare of society. Self evaluation on human resources has been done and the result showed indication of demotivation and decline in employee competence since there are no major programs in the last 25 years, ageing of employees because of moratorium program for new recruitment, limited competency budgets, as well as existence of potential of knowledge lost. The results of the evaluation was also applicable for research reactor personnel. Competence development for research reactor personnel is expected to provide outcomes: government regulations are met, national programs are still in place, critical knowledge loss can be prevented, knowledge retention program can be done, and research reactors can be operated in a safe, secure and sustainable. Planning for the development of competency of research reactor personnel has been conducted, including study of human resource condition, competence map, critical knowledge and potential knowledge loss, and identification area of expertise. The competence development program is conducted using grading approach to set priorities based on national program, cooperative commitment with partners, and potential knowledge lost.

Keywords: Research reactor personnel, competence, training, grading approach.

Safety Reassessment of Egyptian Second Research Reactor post Fukushima Accident

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Abstract. In the light of Fukushima accident, reassessments of Egyptian second research reactor number two have been done. It shows that all of the basic safety functions are fulfilled during the extreme events. Review and updating of the safety analysis after in-core irradiation for LEU targets to produce Mo-99. Ensure that operation and limits conditions (OLCs) are maintained as they approved by the current operating license. Review the emergency power supply and Uninterrupted Power Supply System (UPS) systems. Review of the emergency preparedness and the emergency equipment. There are recommendations for enhancing the safety during normal and emergency cases.

Considering Fukushima lessons for the Budapest Research Reactor

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Abstract. Hungary participated in the stress test process of the nuclear power plants according to the specification laid down by the European Commission. In the mirror of experiences of the Fukushima Dai-ichi NPP accident the stress tests were meant to re-examine the design basis and the margins against beyond design basis events for NPPs including analysis of cliff-edge effects. In particular, the loss of heat sink, loss of external power and emergency power were assessed and the potential instruments for coping with severe accidents and emergency preparedness were put in the objective. The stress tests took place as a single, extra effort for the nuclear power plants, about which the licensees and the regulatory body summarized the results in national reports. The results were then reviewed by European expert teams including on-site examinations, workshops and review meetings. The conclusions were compiled in a National Action Plan. Concerning research reactors there have not been no such a single European effort. The methodology, however, could be applied to these facilities (and other fuel cycle facilities), at least concerning the assessment phase of the stress tests. Regarding the Budapest Research Reactor, the occasion to carry out the assessment came with the imminent Periodic Safety Review (PSR) that was due in 2012. In addition to the Hungarian legal requirements, the HAEA's practice is to issue a regulatory guideline to aid how to carry out the actual PSR of the facilities. In this case a separate chapter was dedicated to the post-Fukushima safety review of the Budapest Research Reactor where a short specification was provided on the contents of the assessment. The licensee then followed the instructions and performed the analyses according to the guidance of the regulator, which were finally assessed and approved at the end of the PSR process by the HAEA. The overall conclusion was that it did not reveal the need for any immediate action.

The paper summarizes the stress test methodology followed in Hungary, the PSR requirements for research reactors including the post-Fukushima chapter added in 2012, the analyses results and the conclusions of the Licensee and the corresponding HAEA assessments.

Safety Enhancement of Dhruva Reactor through Periodic Safety Review

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Abstract. Dhruva is a natural uranium metal fuelled, heavy water moderated, cooled and reflected research reactor of 100 MWth capacity. Dhruva attained first criticality on August 8, 1985. As part of the regulatory requirement every operating facility including research reactors, shall carry out a Periodic Safety Review (PSR) once in every 10 years. A comprehensive Periodic Safety review of plant was carried out to continue operation beyond May 2014, taking into account the cumulative effects of plant ageing, modifications carried out over the years and the feedback of operating experience. The review was carried out based on safety factors suggested in safety manual BSCS/SM/2010/1 Edition; R-0, March-2010 issued by the apex safety body of BARC. Before a PSR is started, an agreement between the operating organisation and the regulatory body indicating the scope and objectives of the PSR, its schedule and expected outcome was made. Based on these reviews, certain mid-term safety up grades in various systems of Dhruva Reactor was carried out. This paper provides an overview of overall safety enhancement of the research reactor carried out following PSR.

Safety Reevaluation of Indonesian MTR-Type Research Reactor

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Abstract. Safety analyses of 30 MW MTR-type research reactor have been conducted for three important cases, i.e. loss of flow accident (LOFA), loss of off-site power and reactivity insertion accident (RIA) due to inadvertent control rod withdrawal. The transient calculations were done using PARET/ANL, and for reactivity insertion rate MCNP5 code was employed. For both LOFA and loss of off-site power, the reactor was assumed to have been operating at nominal power of 30 MW. On the other hand, during accident of inadvertent control withdrawal the reactor was assumed to have been operating at 1 MW. The analyses are intended to confirm that the most bounding design basis accidents, the reactor can be maintained safe. For all the aforementioned cases, The fuel and clad temperatures can be maintained well below safety criteria of 200 °C and 145 °C, respectively. In addition, the safety margin against flow instability (S) is kept well above the criterion of 1.48.

Keywords: MTR type, LOFA, loss of off-site power, RIA, rod withdrawal.

WWR-K Reactor Safety Reassessment

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Abstract. WWR-K reactor is a unique multipurpose research reactor in the Republic of Kazakhstan. The first physical start-up of the WWR-K had been carried out in October of 1967. Research reactor WWR-K is a water-water heterogenic reactor with thermal neutrons and project power of 10 MWt and U-235 enrichment of 36 %. It worked without any deviations and emergencies till 1988. After Chernobyl accident in April of 1988 it was adopted by regulatory body of the former Soviet Union the decision to shut down it permanently in order to carry out works on enhancing of seismic defense of the reactor. Since 1998 the reactor had been put into commission at the power 6 MWt. Since 2003 it was began investigations on conversion from high enriched uranium fuel to low enriched uranium fuel with preservation of operation and experimental features. As a result of these investigations it had been elaborated a new fuel assembly VVR-KN. A new reactor core had been elaborated. Life test of this three fuel assemblies was carried out and it was reached more 50% of U-235 their burn up. Conversion of the reactor had being carried out within the framework of the IAEA programme Russian Research Reactor Fuel Return and the International Program on Reduction of Fuel Enrichment of the Research and Test Reactors (RERTR). Taking into account Fukushima Daiichi Accident it had been adopted the decision alongside with the conversion to carry out modernization of the reactor systems under the abovementioned programmes. I&C, emergency cooling, emergency core sprinkling, primary cooling, secondary cooling, radiation monitoring, gas and aerosol emissions monitoring systems have been modernized. Physical and power startups of the reactor had been successfully carried out in the first half of 2016. In September of 2016 the reactor WWR-K had been put into commission with low enriched uranium fuel.

TAPIRO Fast Spectrum Research Reactor Characteristics for Neutron Radiation Damage Analyses.

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Abstract. Research reactors have been always considered as a powerful tool to investigate the effects of neutron radiation damage on a wide class of structural components of interest for both research and industrial applications. In particular, depending on the damage level demanded after irradiation, high power or low power research reactors are usually selected as radiation fields. Currently important high neutron flux facilities, like JHR (France), MYRRHA (Belgium), MBIR (Russia) are under construction or in advanced planning stage. Nevertheless, there is an increasing attention also towards low power research reactors for neutron radiation damage analyses, essentially because they can provide very qualified, in terms of both intensity and energy spectrum, neutron radiation fields, being such characteristics very useful for irradiation of electronic components and/or innovative neutron detectors. The ENEA low power fast spectrum TAPIRO research reactor, located in the Casaccia Research Centre near Rome, Italy, complies with the above quality requirements.

This paper describes how the neutron flux stability can be monitored at TAPIRO, together with some reference procedures for the calibration of detectors in the TAPIRO neutron field. Characteristics of some main ASTM standard damage parameters, like 1 MeV neutron equivalent flux and hardness parameter, are provided for different positions all along the main irradiation channels. Finally this paper describes what kind of spin-off is expected, in terms of neutron flux characterization, from the first phase of the international experimental campaign AOSTA, foreseen the next year at TAPIRO.

The Nuclear Technology and its Development in the National University of Rosario – Argentina

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Abstract. Since 1972 The Faculty of Exact Sciences, Engineering and Surveying of the UNR has had a nuclear reactor SUR-100 called the RA-4. For different reasons a staff of teachers, researchers and technicians could not be consolidated to fulfill the university missions: Teaching, Research and Extension.

In 2014 after several meetings with CNEA, the University and the Secretary of University Policies (SPU) of the Ministry of Education, a program was elaborated and signed in order to consolidate the teaching, research and extension activities of the RA – 4. Nowadays there is an Institute called IENRI (Institute of Nuclear Studies and Ionizing Radiations) dedicated to the development of the subject. It works with a staff of professional teachers and technicians that have licenses to operate the Reactor. Regarding Teaching, the Institute it is developing practical experiences for the degree in physics as well as teaching the elective subjects “Nuclear Power Plants I and II” for engineering degrees. There are also being programmed experiences, laboratory work and pre – professional practices for the students of the Advanced Technical Degree in Nuclear Reactors. Furthermore, it is working on the drafts for the creation of a degree course: Degree in Nuclear Technology in cooperation with related sciences (CCC Cycle of Curricular Complementation).

In the area of Investigation we count with a Doctor in Physics and a Doctorate in Mathematics who make presentations in congresses and symposiums. The opportunities are expanded by disseminating the abilities of the facility to the various groups, laboratories and institutes of the University and CONICET.

In terms of Extension and Linkage, we signed agreements with Na-Sa, the National University of Cordoba (CUTeN RA-0), the Dan Beninson Institute belonging to UNSAM and the ISFT Technical Training Institute No. 195 of Lima, all Institutions dedicated to the teaching, research or production of nuclear energy.

Radiation Safety Training at the Open Pool Australian Light-water (OPAL) Multi-purpose Reactor

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Abstract. The development and implementation of effective workplace radiation safety training at the OPAL multi-purpose reactor has been achieved through the application of the Systematic Approach to Training (SAT) process. SAT is a multi-step, iterative process for the development and continuous improvement of training. Key components of SAT are the training needs analysis, overarching training program, learning objectives, content and assessment development, and training effectiveness evaluation.

The fundamentals of radiation safety as it applies at ANSTO facilities are covered by the Basic Radiation Safety course for new starters and the scenario-based Radiation Safety Workshop that classified workers enrolled on the ANSTO dosimetry service are required to complete every five years. To complement this radiation safety training has been developed on the radiological hazards encountered and the radiation protection arrangements at OPAL. This training consists of an instructor led OPAL specific radiation safety training course for staff new to the facility, and an online OPAL specific radiation safety training refresher that staff working at OPAL are required to complete every three years.

This paper describes the training program structure, learning objectives, content development and delivery format of the various courses, workshops, inductions and online modules in greater detail. It also examines how by using the SAT process an effective radiation safety training program has been developed for classified workers at the OPAL multi-purpose reactor to ensure they have the desired knowledge, skills, experience and attitude to radiation safety.

**Safety Reassessment of Research Reactors in Light of the Lessons Learned
from the Fukushima-PAA point of view. Selected Issues.**

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Abstract. The MARIA multipurpose high flux research reactor is at present the only operating nuclear reactor in Poland. In case of the MARIA research reactor, the safety assessment recommended by the IAEA in the SRS-80 report after the Fukushima accident was re-evaluated. Most of the safety analysis already exists and was appropriate, however, there are additional safety analyses needed. Therefore, appropriate PAA and operator actions will continue in the near future. This report contains description of activities performed by PAA and Maria research reactor operator. The study also includes information on some changes in the Polish Atomic Law.

Internal Peer Review (IPR) of Pakistan Research Reactor-1 (PARR-1)

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Abstract. The lessons learned from Fukushima Daiichi accident helped the nuclear community to revisit the design of nuclear installations and identify vulnerabilities for preventing accidents involving large releases of radioactive materials. In this context, corporate office of Pakistan Atomic Energy Commission (PAEC) conducted Internal Peer Review (IPR) of Pakistan Research Reactor-1 (PARR-1). Objective of IPR was to conduct operational safety review and safety assessment of PARR-1 and to prepare them for the proposed IAEA (INSARR) Mission. Review of PARR-1 was based on IAEA safety standards, national regulations and international best practices. For IPR of PARR-1 mutually agreed 19 areas were selected for review. Various issues were identified during the review process and recommendations and suggestions for rectification of these issues were made by the reviewers. As a result, Corrective Action Plan (CAP) was prepared for implementing necessary measures to enhance safety/robustness of PARR-1, which has recently completed 50 years of successful operation.

SAFARI-1 Safety Reassessment and Modifications in Light of the Fukushima Daiichi Accident

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Abstract. Following the Fukushima nuclear accident, a directive from South Africa's National Nuclear Regulator was received which required a safety reassessment of the SAFARI-1 research reactor.

The safety reassessment consisted of:

- Evaluation of the response of the SAFARI-1 Research Reactor when facing a set of extreme external events (EEE)) or Design Extension conditions and
- Verification of the preventive and mitigation measures chosen following a defence-in-depth (DiD) logic: initiating events, consequential loss of safety functions, severe accident management.

The safety reassessment process was performed in various steps. Site-specific natural external events (Design Extension conditions) were firstly identified. The full lists of EEEs identified that may have an impact on SAFARI-1 include earthquakes, external flooding, tornadoes and tornado missiles, high winds, sandstorms, storms and lightning, hurricanes and tropical cyclones, bush fires, explosions, toxic spills, accidents on transport routes, effects from adjacent facilities, biological hazards, and power or voltage surges.

For the set of situations proposed, a deterministic approach was used in which the sequential loss of the existing lines of defence is assumed, regardless of their probability of occurrence. The ultimate objective was to confirm the degree of suitability of the existing measures for accident management and, finally, to identify potential applicable improvements regarding both equipment (fixed and portable) and organisation (procedures, human resources, emergency response organisation and use of off-site resources).

This step was followed by the development of event trees which depict the progressive evolution of the EEE into plant damage states which could potentially lead to public exposure. These evaluations were carried out in accordance with the philosophy of DiD as proposed in the ENSREG stress test specification.

A number of recommendations were identified by the safety reassessment feasibility investigations, including stabilisation of fresh fuel store, emergency water return, external plugin power, re-flooding nozzle, emergency control room, independent seismic trip, second shutdown system, reactor building reinforcement and updating emergency procedures. Most of the recommendations have been taken to the implementation phase.

This paper will present the *summary of feasibility phase* outcomes, *results of the safety reassessment*, as well as some of the progress on resulting modifications and the future operations to conclude the post-Fukushima safety enhancements activities.

Thermal Hydraulic Analysis of 49-2 Swimming Pool Reactor with a Passive Siphon Breaker

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Abstract. In order to improve the safety of the reactor, a passive siphon breaker has been added to the primary cooling system of 49-2 Swimming Pool Reactor (SPR). The thermal hydraulic of the reactor with a siphon breaker of diameter 1.6 cm was analyzed by RELAP5/MOD3.3 code. The results show that: under the conditions of steady-state operation, small break loss of coolant accident (SBLOCA) and large break loss of coolant accident (LBLOCA), the siphon breaker is able to break the siphon phenomena, and maintain the pool water level above the reactor core when the reactor and the pump shutdown.

Neutronic Analysis of Control Rod Effect on Safety Parameters in Tehran Research Reactor

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Abstract. The measurement and calculation of neutronic parameters in nuclear research reactors has an important influence on control and safety of nuclear reactor. The power peaking factors, reactivity coefficients and kinetic parameters are the most important neutronic parameter for determining the state of reactor. The position of the control shim safety rods in the core configuration effects on these parameters. The main purpose of this work is using the MTR_PC package to evaluate the effect of the partially insertion of the control rod on the neutronic parameters at operating core of the Tehran Research Reactor. The simulation results show that by increasing the insertion of control rods (bank) in the core, the absolute values of power peaking factor, reactivity coefficients and effective delayed neutron fraction increased and only prompt neutron life time decreased. . In addition, the results show that the changes of moderator temperature coefficients value verse the control rods positions is very significant. The average value of moderator temperature coefficients increase about 98% in the range of 0-70% insertion of control rods.

Keywords: neutronic parameter, power peaking factor, moderator temperature coefficient, fuel temperature coefficient, kinetic parameters

Identification and Implementation of a Hardened Core in a Research Reactor in Light of the Lessons Learned from the Fukushima Daiichi Accident. The JHR Case.

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Abstract. The JHR reactor is a new high performance MTR (Material Testing Reactor) working as an International User Facility in support to Nuclear Industry, Public Bodies and Research Institutes. This new facility of a maximum power of 100 MWth will allow to reproduce on a small scale real power plant conditions and in some cases, more severe conditions for Material screening (comparison of materials tested under representative conditions), Material characterization (behaviour of one material in a wide range of operating conditions, up to off-normal and severe conditions) and fuel element qualification (test of one / several fuel rods (clad+fuel)). Its construction began in 2009 after getting the authorization by French Regulatory Body (ASN) with the best safety design specifications of the moment.

Following the Fukushima Daiichi accident as a consequence of the earthquake and tsunami occurred on 2011 March 11th, the French government asked all French nuclear facilities to perform complementary safety assessments to meet objectives under extreme situations exceeding licensing basis (with focus on “cliff-edge” effect prevention). JHR, which was under construction, provided its report on 2011 September 15th. The complementary safety assessments basically confirmed the sound design bases of the JHR. Nevertheless ASN asked CEA to propose a so called “Hardened Core” of material and organizational dispositions that can ensure the following three objectives in extreme situations exceeding the current licensing basis:

- prevent a severe accident or limit its progression,
- limit large-scale releases in the event of an accident which is not possible to control,
- enable the licensee to perform its emergency management duties.

After analysis of the consequence of postulated failure of components under extremely severe conditions, some were proposed to be part of the hardened core of material and a selected needs for extra equipment were identified. As an answer to French nuclear regulator requirements, CEA proposed a set of “hardened core” measures. This set of “hardened core” measures was assessed by the technical support of the French Regulator (ASN) in 2011 and reviewed in April 2013 by its standing advisory committee. ASN approved CEA proposal and add the containment deflation circuit.

Due to the fact that JHR construction is not yet completed, its situation is rather particular and the Hardened Core is constituted by already designed SSC (Structure, system and

component) but also by new SSC. For all Hardened Core SSC the so called Supports to SSC have to be identified. These SSC are the ones requested by Hardened Core SSC to perform its requested function, and identification methods have been developed. All margins on existing SSC designs have to be evaluated regarding the construction codes used for initial design, particular methodologies and new criteria must be defined in case of lack of margin. All new SSC are designed using the classical codes of construction with more severe operating conditions resulting of post Fukushima situations.

The completion of Hardened Core implementation on JHR will be performed without start up schedule modification.

Safety Reassessments and Actions Taken in HANARO since Fukushima Daiichi Accident

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Abstract. After the Fukushima Daiichi Accident, a special safety review was conducted for all nuclear facilities such as nuclear power plants, research reactors, and nuclear fuel fabrication facilities in Korea, by a special review team organized with government officials, regulators and civilian experts. HANARO, a 30MW research reactor, was also inspected to review the plant design and configuration, operation procedures, the accident management procedures, and the emergency preparedness plan over the design basis accidents and extreme natural events. As the results of the review, several recommendations were given to HANARO for the improvement of the safety. The measures for the recommendations have been implemented and reported to the regulatory body. One of the recommendations made by the special review team was to evaluate the seismic margin of the reactor building. A seismic margin assessment was performed and it showed that some part of the outer wall of the reactor building did not satisfy the seismic design criteria set to 0.2g in horizontal direction. Thus, the regulatory body officially asked a seismic reinforcement of the reactor building and the reinforcement work has been completed recently. This paper describes the summary of the safety reassessment and the actions taken to improve the safety of HANARO since the Fukushima Daiichi Accident.

Introduction of Nation-wide Inspection and Reassessment to Chinese Research Reactors after Fukushima Accident

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Abstract. Briefly introduce the nation-wide comprehensive inspection to NPPs and research reactors performed by regulatory body after Fukushima accident, as well as the inspection results to research reactors. Afterwards, reassessment of safety conditions of research reactors was carried out by experts, focus on the following aspects: Anti-flooding capability evaluation, Evaluation on seismic response and anti-seismic capability, reliability of external and internal electrical power supply, evaluation on nuclear accident emergency management, etc. Based on the reassessment results, some improvement measures were proposed by regulatory body, which require operators of research reactors to implement in few years, for instance evaluating seismic margin of some old reactors, increasing waterproof blocking of nuclear island facilities and buildings as well as interim waterproof measures adopted in key buildings, improving power supply capability of storage battery and installation of mobile EDG, and so on. Finally, some additional safety requirements are described for several reactors with potential risk.

Keywords: NNSA, research reactor, Fukushima accident, national inspections

Experience with Safety Reviews of Slovenian Research Reactor by PSR and IAEA INSARR Missions and the Stress Tests for the Krško NPP

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Abstract. The TRIGA Mark II research reactor of the »Jožef Stefan« Institute in Slovenia started its operation in 1966. The operational experience of the reactor and performance indicators reflect safe and stable operation. The first safety review of the reactor was done in 1992 by IAEA INSARR mission. In 2002 the new Ionizing Radiation Protection and Nuclear Safety Act introduced a requirement for nuclear facilities to perform Periodic Safety Reviews (PSR) which is a prerequisite for extension of facility's operating license. The PSR requirements were defined in 2009 in a new regulation Rules on operational safety of radiation and nuclear facilities. The operator of the TRIGA research reactor performed the PSR from 2011 to 2014. The PSR summary report confirmed that the reactor can safely operate for a further 10 years. In 2012, the second IAEA INSARR mission of the TRIGA research reactor was carried out. The INSARR Follow up mission in 2015 confirmed that the operator is making significant progress with recommendations implementation and included some additional recommendations. In 2011, after the Fukushima accident, the EU countries performed the stress tests. The Krško NPP prepared a Safety upgrade program to resolve stress tests findings and to upgrade the plant design to Design extension conditions.

RCC-MRx 2015 Code: Context, Overview and On-going Developments

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Abstract. RCC-MRx Code is the result of the merger of the RCC-MX 2008 developed in the context of the research reactor Jules Horowitz Reactor project, in the RCC-MR 2007 which set up rules applicable to the design of components operating at high temperature and to the Vacuum Vessel of ITER.

This code has been issued in French and English versions by AFCEN (Association Française pour les Règles de Conception, de Construction et de Surveillance en Exploitation des Matériels des Chaudières Electro-Nucléaires) in 2012, and a new edition has been published at the end of 2015.

A significant work has been performed for this edition to improve the code in order to facilitate its use and understandability, and also to have a better fit with the feedback of the users.

In parallel, in compliance with the EC's objectives and its own policy of openness, AFCEN proposes to make its codes evolve, taking into account the needs and expectations of European stakeholders (operators, designers, constructors, suppliers...) threw a workshop called CWA phase 2.

This paper gives an overview of the realized work and also will identify the work to be done for an opening of a standard such as RCC-MRx code.

Commissioning Experience for Reactor and Primary Cooling System of Jordan Research and Training Reactor (JRTR)

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Abstract. The primary cooling system is designed to cool the heat generated from the core of a pool-type research reactor (JRTR, Jordan Research and Training Reactor). The system penetrates the pool and is connected to the reactor. The reactor and pool should be kept clean as a caution during commissioning because re-cleaning or disassembling and re-assembling will require additional time and cost. Thus, the reactor, fluid equipment, instruments and pipes should be fabricated based on the cleaning procedure in accordance with the requirements of the related code and standard. The reactor and primary cooling system should be installed using the installation procedure because the interface between the reactor assembly and the related system including the fluid system, platform, pool door, instruments, detector conduits, and pool covers inside the pool is considerably complicated. After the primary cooling system is installed in the reactor and pool, system flushing is performed to remove any dust, particles, or other foreign matter using closed and open flushing methods. After the flushing and required CATs (Construction Acceptance Test) are completed and the demineralized water is filled in the pool and system, the SPT (System Performance Test) including measuring the system flow rate and pressure loss and checking the function of the pumps, valves, and system alarms can be started. Because the control valve is not used in the safety system, the pressure loss of the system is adjusted by replacing the system orifice plates with the planned procedure to meet the system flow rate during the SPT. The PST (Pre-Service inspection Test), which is a prerequisite for developing an in-service test program can be performed after the results of SPTs satisfy the acceptance criteria of the tests.

Jules Horowitz Reactor: RCC-MRx[1] Applicability for the Design Phase of Experimental Devices

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Abstract. The Jules Horowitz Reactor (JHR), currently being built at the CEA2/Cadarache in the south of France, will be a Material Testing Reactor (MTR) designed to perform irradiation experiments while complying with today's safety, quality and regulatory requirements. This paper introduces the fundamentals of the RCC-MRx[1], technical references and rules for designing, manufacturing & controlling of mechanical components for the JHR.

The RCC-MRx defines the rules and the recommendations for all the mechanical equipment, which will be used in the JHR, even the experimental devices which will have a safety and reliability functions. These rules must be applied, in particular, during the process of mechanical design and calculations. They are depending of the type of equipment (containment, structures, pumps, valves, etc.) and they are proportional of the safety/reliability levels defined by the safety studies.

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Progress in Conceptual Research on Fusion Fission Hybrid Reactor for Energy

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Abstract. Fusion Fission Hybrid Reactor for Energy (FFHR-E) is fueled by natural uranium and cooled by light water. A three dimensional code MCORGS, which is coupled by MCNP and ORIGENS, has been adopted in neutronics analysis of the three dimensional blanket model. A simplified pyro-reprocessing scheme is suggested. It is expected that the spent fuel can be heated up to 2100K by its decay heat, then the fission product elements whose boiling points below this temperature will be evaporated. The displacement per atom (dpa) of the blanket materials in one year was evaluated. A reload period around ten years is suggested · the spent fuel can be reused multiple times after reprocessing. The average Tritium Breeding Ratio (TBR) is about 1.15 and the blanket energy multiplication is about 12 in the first 60 years. While in the 2nd to 9th 60 years, the average TBR and M are 1.35 and 18 separately.

Networking Advanced Experimental Capacities in Operating European Materials Testing Reactors for Qualification of Innovative Nuclear Fuel and Materials: The FIJHOP R&D Program Proposal

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Abstract. The Jules Horowitz Reactor (JHR) is a Material Testing Reactor under construction at CEA Cadarache in Southern France and foreseen to be in operation by the beginning of the next decade. Operated as an User's facility at international level, it will promote coupled operation between the operating European MTRs (BR2, HFR, LVR-15, HBWR, MARIA...) for performing multilateral programs associating one or several MTRs and hot cell laboratories for post-irradiation examinations (PIEs). These programs will beneficiate from a strong support of state-of-art models and codes, either for defining the experimental protocol or pre-calculating the sample behaviour, or finally assessing results.

This last way has been recognized as essential by the JHR Consortium, which endorsed the objectives of a proposal called FIJHOP (Foundation for future International Jules Horowitz experimental Programs), submitted to the last H2020 call in October 2016. Supported by some communities such as NUGENIA, it addressed following scientific issues:

- For nuclear fuels (FIJHOP-F): to discriminate and quantify phenomena having an impact on clad loading and deformation during a power transient: fuel thermal expansion, fuel swelling, fission gas release and volume change at incipient fuel melting. Such final status will be reached with a high burn-up and multi- instrumented experimental fuel rod,
- For nuclear materials (FIJHOP-M): to study irradiation effects on Internals and more specifically to check effects of the neutron spectrum on damage accumulation kinetics, which may impact their mechanical properties. Associated particular interest is to harmonize interpretation of such evolution versus dpa (displacement per atom).

FIJHOP targeted an implementation in 3 European MTRs in operation (BR2, HFR and LVR-15) and 9 Hot Cell Laboratories, and was accompanied by a code benchmarking. Although the proposal was not accepted for the last H2020 call, partners decided to search for other possible tools/frameworks to maintain or even enlarge the objectives of the former FIJHOP proposal. In particular, contacts are in progress with the Nuclear Science Committee (NSC) of the OECD/NEA.

This paper presents the FIJHOP proposal, the scientific objectives in relation with stakes for power reactors operation, its innovative feature, and details on both experiments (fuel and materials). It highlights the added value provided by qualified models for optimizing the experiment preparation, and in turn the interest of expected results for participant's database improvement, development and qualification of their simulation tools.

Analysis of a Hypothetical LOCA in an Open Pool Type Research Reactor

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Abstract. An analysis of a hypothetical loss of coolant accident (LOCA) in a pool-type research reactor is presented. The study was implemented for the Israel Research Reactor 1 (IRR-1), which is a 5MW reactor using highly enriched MTR-type fuel plates reflected by Graphite elements. The reactor core is cooled by downward forced flow of light water during normal operation and by upward natural convection flow through a safety flapper valve during shutdown. LOCA in pool-type research reactors may be initiated by various incidents such as ruptures and leakages from pipes and valves in the primary cooling system, ruptures of beam tubes or cracking of the pool wall caused by, e.g., strong earthquakes. Each one of these scenarios results in a rapid drop of the pool water level after reactor SCRAM. If water flow through the break persists, the core could eventually uncover completely and be exposed to the ambient air. The present study analyzes the possibility of passively cooling an exposed reactor core by thermal radiation and natural convection to air. The core uncover time is estimated by conservatively assuming that the LOCA was initiated by a guillotine break of a 10 inch outlet cooling pipe at the bottom of the pool, causing the core to uncover about 20 min after reactor SCRAM. Longer uncover times were used for parametric comparison. Since the Graphite reflector elements surrounded the core are typically solid that do not generate heat, they have the potential to act as a heat sink. The effect of the reflector on the core cooling was studied by comparing the total heat transfer from the core with and without considering the thermal contact between the core and the Graphite reflector elements. It is shown that for an uncover time of 20 min the core could reach its melting point if thermal contact with the Graphite is neglected. On the other hand, considering perfect thermal contact between the core and the Graphite reflector, the core temperature is predicted to remain indefinitely below the clad melting point (580 °C). The decay heat generation rate after reactor shutdown plays an important role in the analysis of LOCA. Several empirical correlations and theoretical models are available for predicting the decay heat after shutdown of a continuously operating power reactor. These correlations could not be simply applied for research reactors that work intermittently. A conservative decay heat generation curve was, therefore, estimated by comparing numerical results obtained by the BGCore computer code with available semi-empirical fitting functions and the ANS 5.1 standard curves. It has been shown that the BGCore computer code predict the decay heat generation rate with a small deviation from the corresponding semi-empirical functions results and the ANS 5.1 standard curves.

Validation of the Stable Period Method against Analytic Solution

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Abstract. Control rod (CR) reactivity worth plays an important role in safety and control of reactors. There are several ways to calculate the control rod reactivity worth, the most common and frequently used method is the stable period method. In this method, the reactivity of the system is related to the stable reactor period (time for power to change by the factor e), through the reactor kinetics equations. From this study, it was found that there are three main areas of uncertainty that contribute to the accuracy of the method: the parameters used in calculations, the operator procedure, and the effectiveness weighting factor, γ , of the effective delayed neutron fraction. The former represents a source of random uncertainty while the latter two represent systematic sources of error.

The objective of this study is to analyse two practical applications of the stable reactor period method of calibrating a low-worth (regulating) absorber rod: (1) the doubling time method - where the absorber rod is withdrawn a percentage of its length and the operator measures the time until the reactor power is doubled, and (2) the 30 second method – where the absorber rod is withdrawn a percentage of its length, the operator waits 30 sec, and notes the power increase over the next 30 seconds. Comparing calculations of these two methods to analytic solutions found that in terms of shim rod uncertainties, the random uncertainty is up to $\pm 1.7\%$ relative error in the 30 seconds method, comparing to $\pm 1.4\%$ in the doubling time method. The systematic error found to be relatively low in both methods with 1.4% relative error, in the 30 seconds method, comparing to 1.1% relative error in the doubling time method. Additionally, this study demonstrates the importance of the effectiveness weighting factor of the effective delayed neutron fraction and its significant impact on reactivity values (can represent up to 32% systematic error).

Hydraulic Design and Validated Calculation Tool of the Jules Horowitz Reactor (JHR) Reflector

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Abstract. Optimisations of Research Reactors hydraulic design have to be performed on major components such as the core or reflector in order to reach neutronic performances while limiting the reactor cost. Validated calculation tools are developed for these components in order to support design and produce operating and safety studies.

TechnicAtome is in charge of both the design and construction management of the 100 MW Jules Horowitz Reactor on behalf of CEA. This modular Material Testing Reactor will show capabilities of radioisotope production and material testing. The JHR reflector outside the primary loop is composed of non-similar beryllium blocks arranged all around the core in 9 independent sectors cooled by one downward flow open pool circuit. Its design is completed and this paper is dealing with the hydraulic design of the reflector, tests which have been performed on dedicated hydraulic loops and all the actions which lead to provide validated calculation tools as STAR-CCM+ (PLM Software) for 3D computations and CATHARE2 (CEA) for circuits modelling.

JHR specifications led to a complex design especially for reflector beryllium blocks with heterogeneous gamma heating according to the presence of gamma shield. Taking into account the need to load and unload experimental devices while reactor is under operation, thermal-hydraulic design had to manage these cooling constraints whereas a downward flow cooling circuit limits the maximum mass flow and its head losses.

To dispatch the flow between structures of the reflectors (beryllium, aluminum or zircaloy) and experimental devices, each sector of the reflector has a water box with diaphragms located at the input. A torus water box collects these flows and makes a balance of the head losses. The JHR reflector hydraulic design can be compared with an organ where each channel is set to use the only necessary mass flow.

A coupled approach based on experimental hydraulic tests and computer codes simulations is performed. The first head losses characterization of beryllium blocks and water boxes are based on hydraulic loops tests with a Reynolds similitude to cover the operating and accidental domain. Whereas there are approximately 20 different types of beryllium blocks and 9 sectors due to core design, a predictive methodology is developed (based on a 1D calculation approach for beryllium blocks and a 3D calculation approach for water boxes) to limit the number of hydraulic experiments. This coupled approach (hydraulic tests/calculations) makes possible the extrapolation to different beryllium

blocks and water boxes designs by calculations only. Calculation tools are operational to finalize the cooling sizing of the reflector and to simulate various configurations of the JHR reflector.

Onset of Flow Instability in a Rectangular Channel under Transversely Uniform and Non-uniform Heating

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Abstract. Flow instability in a narrow rectangular channel (2.35 mm x 54.0 mm x 300 mm) is studied under uniform and non-uniform heating conditions since the power released from the nuclear fuel is not uniform in the axial and transverse directions. Transverse non-uniform heating may cause local boiling where local heat flux is relatively higher than other locations. This may occur boiling locally, which disturb or generate a different velocity profile compared with that under uniform heating. The velocity profile change is significant when the flow condition reaches the Onset of Flow Instability (OFI). In the present study, an experimental facility has been designed to study the effects of non-uniform heating on the velocity profiles. Experiments are carried out using two different ways to reach the OFI; (1) decreases flow rate with constant power and (2) increases power with constant flow rate. When the flow reached the OFI, the pressure drop changes show different trends. This is because the flow travels faster where there is a significant boiling than that where there is not. This study shows different boiling behaviours in a narrow rectangular channel under transversely uniform and non-uniform heating.

Application of the Next Generation of the OSCAR Code System to the ETRR-2 Multi-cycle Depletion Benchmark

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Abstract. The OSCAR code system is primarily used to perform day-to-day reactor calculations in support of the SAFARI-1 research reactor at Necsa. Recent developments in the OSCAR code focused on integrating high-fidelity and standard nodal diffusion analysis codes in a consistent way. A code independent front end system is used to create a detailed heterogeneous model of a reactor. This particular model can then be used to create input for various underlying target codes such as MCNP, Serpent and the OSCAR nodal diffusion solver (termed MGRAC), ensuring consistency between the models for target codes under consideration. The ETRR-2 benchmark, which is part of the IAEA CRP on multi-cycle depletion analysis, was modelled in aid of code validation. This benchmark is a good candidate to use for code validation because the experiments to be modelled span the first four cycles of the reactor life. All models will therefore have exactly the same initial fuel number densities, removing unnecessary uncertainty from the calculations. Prior to any burn-up, a series of control rod calibrations were performed at the plant, for which the experimental data is available in the benchmark. In this work, a detailed heterogeneous model is built in OSCAR from which a reference Serpent model is generated. In addition, cross sections and model input are also generated for use in MGRAC. Several control rod calibration experiments are simulated with Serpent to test the accuracy of the models. Depletion analysis for the four cycles is performed with Serpent and MGRAC in order to validate their burn-up capabilities. In particular, calculations are performed for three spent fuel elements. Comparisons were done between measured burn-up, Serpent and MGRAC.

IAEA Activities in Support of Operation and Maintenance of Research Reactors

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Abstract. Approximately 50% of the operating research reactors (RR) in the world are more than 40 years old. Although the life of such facilities could reach 60 years and beyond, it is of paramount importance that adequate life management programmes (ageing and refurbishment/upgradation programmes) are established well in time.

The IAEA provides support to Member States in management of all relevant activities related to operation and maintenance of operating research reactors with focus on enhancing their availability and reliability throughout the whole life cycle. This includes support for the development and implementation of plans for operation and maintenance (O&M), ageing management, human resource development, refurbishment and modernization and establishment of Integrated Management System, as well as of decommissioning plans. Apart from topical meetings and training workshops, a peer review service called Operation and Maintenance Assessment for Research Reactors (OMARR) is provided to Member States upon request. The OMARR mission provides advice and assistance to Member States in enhancing the performance of their research reactors. The mission is aimed at improving operation and maintenance (O&M) practices throughout the facility's operational life cycle. The service can also assist operating organizations carrying out major refurbishment or modernization of their facilities in identifying the structures, systems and components (SSCs) to be replaced. The expected results include a more efficient long-term operation, better performance, improved safety and safety culture, and optimized utilization of human and financial resources. In addition, Agency provides support to MS in addressing safety aspects related to RR operation through the INSARR peer review mission. Additional support is provided through the recently launched Research Reactor Ageing Data Base (RRADB) which compiles inputs from Member States on experience in tackling issues related to ageing degradations of SSCs. Additionally, as an outcome of a co-ordinated research program recently concluded, a Research Reactor Material Properties Data Base (RRMPDB) is being developed to provide consolidated information on properties of irradiated core structural materials.

ANSTO OPAL Reactor CNS Replacement

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Abstract. ANSTO is planning to replace the Cold Neutron Source (CNS) at the end of its design life. The project has a number of challenges due to the activity of the components. Working to ALARA principles and to minimise the reactor shutdown duration, a number of specialised tools and processes are being developed.

The topics that will be discussed in this presentation include the components to be replaced, computer modelling, tooling, mock-up, safety equipment, planning and training. In addition, some lessons have already been learnt and these lessons can be incorporated into future CNS designs.

The New Irradiation Infrastructure at the BR2 Reactor

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Abstract. Together with the third refurbishment of the BR2 reactor, which took place from March 2015 until July 2016, the experimental capabilities of the reactor are modernised. In the first phase, structure material irradiation rigs are designed constructed in order to meet the modern requirements for material irradiation programmes in support of ageing management for existing reactors and qualification of materials for new installations. The basic characteristics of these installations are fundamentally different and represent an evolution with respect to the capabilities of the BR2 reactor before its refurbishment. For irradiation of materials in support of the ageing management programmes of existing reactors, the RECALL device offers the possibility to irradiate standard size samples for fracture toughness testing of pressure vessel material. The specification of the device requires strict temperature control of the specimens before, during and after the irradiation (+/- 5°C in a range between 250 and 320°C), irrespective of the reactor power, so all neutron damage is accumulated at constant temperature. The in pile section can accept 4 sets of 5 standard Charpy specimens, alternative sample designs can be loaded in the same volume. The irradiation conditions are selected to achieve between 0.05 and 0.15 dpa (in steel) in a single reactor cycle of 3 weeks. The evolution with respect to current or past devices is the flexibility in loading position of the rig (with respect to the past CALLISTO loop) and the volume available for the specimens (with respect to the LIBERTY device).

For qualification of materials at high fast neutron flux and high service temperature, the HTHF device (High Temperature, High Flux) targets the irradiation of materials for fusion and Generation 4 reactors for use at high temperature (300°C to 1000°C) and high fast flux (up to $6 \times 10^{14} \text{ n/cm}^2 \text{ s}$, $E > 0.1 \text{ MeV}$). This device has a dedicated in-pile section for each irradiation demand, but has a generic design and out of pile control equipment. The HTHF rig is designed to be loaded inside a standard 6 plate fuel element, maximising the fast flux and loading flexibility, with the potential to accumulate damage dose up to 10 dpa in steel (total irradiation time of 45 weeks), under nuclear heating conditions from 8 up to 14 W/g inside a dry medium. The first irradiation project is focussed on tungsten irradiation at 800°C to achieve 1dpa (in Tungsten metal). The control system is designed to offer a stable irradiation temperature within 20°C for samples loaded over a range of axial flux positions (100% down to 70% of maximum flux). The irradiation campaign is complemented with irradiation of specimens in non-instrumented capsules, with target irradiation temperatures between 400 and 1200°C.

Modification of the CNS Helium Injection Logic

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Abstract. The Cold Neutron Source (CNS) located at ANSTO's OPAL Reactor, Australia, utilises a helium cryogenic refrigeration loop to keep deuterium in liquid state for the production of cold neutrons. There exists two types of damage to the CNS in-pile structure which may potentially occur: (1) hot damage, where the CNS in-pile thimble overheats due to decay heat from the reactor core warming the in-pile with no available heat sink, and (2) cold damage, where great thermal stresses are induced on the in-pile assembly due to the large temperature difference of the room-temperature injected helium gas and the cryogenic temperature in-pile. The original protection logic supplied from the reactors' designer primarily focused on prevention of hot damage. However, a once-off undesired injection of helium into the vacuum containment brought potential cold damage into focus. Although this event did not appear to cause damage to the CNS in-pile structure, an administrative control was put in place to prevent reoccurrence. Upon further analysis, we modified the injection logic and developed an engineering solution, removing the need for an administrative control.

OPAL Cold Neutron Source Moderator Performance

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Abstract. The OPAL reactor at ANSTO has a cold neutron source (CNS) that operates for over 300 days a year with near 100% reliability, providing cold neutron beams to eight neutron scattering instruments. The high performance of the OPAL CNS is primarily due to its single phase liquid deuterium moderator, cooled by cryogenic helium and maintained by a vertical thermosiphon. In this paper, we present computational and experimental characterisation of the LD2 moderator including sensitivities of CNS heat load and flux on moderator temperature and reactor plant conditions such as core configuration, control rod movement and heavy water purity.

The Cold Neutron Source is a Key Experimental Facility of the CARR Reactor in China

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Abstract. The Cold Neutron Source (CNS) is a key experiment facility of the CARR Reactor in China. It was designed to be a 11.2-litre single phase liquid deuterium source located in the heavy water reflector and able to deliver a cold flux of 1.0×10^{10} n/cm²·s at reactor face. The CARR CNS was commissioned in early 2017 and got into operation under reactor full nuclear power. During the commissioning operation the heat load was measured and the thermo-hydraulic performance have been tested and verified. It has been demonstrated that all aspects of the thermo-hydraulic design feature have been fulfilled.

The IAEA Internet Reactor Laboratory Project: Status, Feedback From Recent Broadcasting and Future Expansion

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Abstract. In the last few years the IAEA, together with the help of its Member States, has developed a specific scheme of services to facilitate the access to research reactors for nuclear capacity building. This scheme is addressed to countries operating or planning research reactors, interested in utilizing them as a primary facility to develop nuclear science and technology capabilities, or in some cases, as a supporting step to embark on a national nuclear power programme. This scheme is now composed of four complementary instruments, each of them being targeted to specific objectives and audience. One of those instruments is the web-based training, called Internet Reactor Laboratory (IRL). By using a Research Reactor Facility (host institution), the IRL provides access to students from remote locations (guest institutions), generally from countries without research reactors, to attend online the experiments carried out in a research reactor and that are relevant in the process of developing human capacity related to nuclear.

The IAEA's IRL project was established in 2015 with two host reactor facilities. The first one is based on the ISIS reactor, at CEA Saclay center in France and another one is based on the RA-6 reactor located in Bariloche Atomic Centre, CNEA, in Argentina. Since September 2016, two host reactors (CNEA – RA-6 in Argentina and CEA – ISIS in France) are broadcasting reactor laboratories (respectively, in Spanish and in English) to guest institutions located in Latin America, Europe and Africa regions. Transmissions from each reactor are carried once a year in the frame of the IAEA project with major support from Peaceful Uses Initiative (PUI) funds from United States Department of State. The expansion of the project includes additional research reactor facilities, already pre-selected and to be designated as host reactors during 2017. One, for the Asian-Pacific region, is the AGN-201K reactor, located at the Kyung Hee University's (KHU) Reactor Research and Education Center in Republic of Korea. The other one, to serve the African region, is the TRIGA Mark-II, located at the Centre National de l'Énergie des Sciences et Techniques Nucléaires (CNESTEN) in Rabat, Morocco.

In this paper, we explain the different approaches regarding the development and implementation of the IRL capabilities based on different research reactor facilities. We also present the results and feedback collected after the very first broadcasting sessions from ISIS and RA-6. Finally, we outline the plans and schedule for the expansion of the IAEA IRL project to Africa and Asia Pacific Regions with IRL host-reactors the AGN-201K in Korea and the TRIGA Mark-II in Morocco. Development of IRL for Russian speaking countries is also being considered in cooperation with the Russian Federation.

Installation of a Second CLICIT Irradiation Facility at the Oregon State TRIGA[®] Reactor

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Abstract. The Oregon State TRIGA[®] Reactor (OSTR) utilizes a cadmium-lined in-core irradiation tube (CLICIT) near the center of the core in support of Ar-Ar geochronological research. Due to significant demand on the CLICIT facility, it was desired to install a second CLICIT facility on the periphery of the core in order to simultaneously irradiate two samples. MCNP was used to model a variety of core locations to determine a feasible location that would not negatively impact current operations. Once the location was chosen, the core was reconfigured to optimize reactor operations. Reactivity effects and control rod worths were predicted through k-code calculations then compared to experimental results.

Post Irradiation Testing Capabilities of Experimental Reactor Components at the LECI Facility for Service Life Assessment

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Abstract. Assessment of the effect of irradiation on the mechanical and structural properties of experimental reactor core components plays a crucial role in the determination of the achievable service time and replacement schedule. As most of these parts are composed of aluminum alloys (5xxx or 6xxx series), thorough feedback is needed to correlate the microstructural evolution combining among others phenomena Al to Si transmutation, irradiation swelling and oxidation with changes in mechanical and physical properties of the parts. The methods and experimental capacities developed over the years in hotcell at CEA Saclay on highly irradiated material will be presented, regarding to precise measurement of the Si content and its repartition (through LIBS, EPMA and ICP-AES), extensive mechanical testing (tensile test and fracture mechanics), and microstructural investigations including optical, SEM and TEM microscopy. CEA has also developed the associated mechanical and structural analysis capabilities to provide a comprehensive answer to the issues regarding the evaluation of the component service time limit. Some example of past expertise work conducted on MTR components will be shown to illustrate these capabilities.

Power ramping and cycling testing of VVER fuel rods in the MIR reactor

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Abstract. Loop-type test reactor MIR was purposely designed to perform long-term lifetime tests of fuel assemblies, fuel rods and structural materials of various reactors. Since 1990s, the MIR reactor has been used to develop, implement and improve fuel rod power change test techniques, as well as to simulate abnormal operating conditions and power cycling. For these tests specific techniques, experimental rigs and in-pile gages have been developed and successfully applied.

During this period of time, about 120 VVER fuel rods have been tested under the power ramping and step-up modes (see FIG. below); about 20 VVER fuel rods have been tested under power cycling conditions [1...5].

The paper presents information about techniques applied at the MIR reactor as well as some tests and post-irradiation examinations results.

35yrs Experience in Operations and Utilization of the Malaysian PUSPATI TRIGA Reactor

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Abstract. The Malaysian PUSPATI TRIGA Reactor (RTP) reached its first criticality on 28th June 1982. It has successfully been operated up to 1-MW and pulsed up to 1300 MW. Initially some medical radioisotopes were produced. Currently its major utilization are irradiations for NAA of various samples from in-house research divisions and external customers. The analog console was replaced in 2013 with a digital console, however with pulsing deactivated. Since 2014, RTP was the site for practical training of local nuclear engineering students. Last year, RTP was one of two research reactors involved in regional nuclear school. Earlier this year, the analog console was successfully refurbished as a training simulator. Currently a project is ongoing for a spent fuel pool expected for completion by end of 2017. This paper will present the above activities and conclude with challenges facing further operation and utilization at RTP.

Experimental Study of the VVER-1000 Fuel Rods Behavior under the Design-basis RIA and LOCA in the MIR reactor

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Abstract. Since 2001 RIAR has been conducting irradiation tests in the MIR reactor under the design basis loss-of-coolant accident (LOCA) and reactivity-initiated accident conditions (RIA), which are targeted at obtaining experimental data on the VVER-1000 fuel performance under these conditions. Each experiment confined itself to examination of fuel, fuel-cladding interaction and analysis of gaseous fission products release from irradiated fuel. Several experiments were carried out under both the RIA and LOCA conditions with the use of the VVER-1000 fuel rods operated at nuclear power plants and attained a burnup of 40 to 70 MW·d/kgU. The irradiation experiments were followed by post-irradiation examinations. In order to conduct irradiation testing of fuel in the loop facilities of the MIR reactor under the VVER-1000 primary circuit conditions, it was necessary to develop appropriate test methods, manufacture fuel test rigs and related engineering equipment.

Modifications on TR-2 Reactor Against an Expected Earthquake

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Abstract. TR-2 Research Reactor is one of the two research reactors in Turkey which was shut down by the regulatory body at 1995 due to seismic safety and has been in the process of being in a long-term shutdown since then. Within the scope safety enhancement, some modifications have been made to take some extra safety precautions. The reinforcement work of the TR-2 Research Reactor, is perhaps the example for some of the reactors in similar positions in the world. The project, which was prepared according to the value of 0.69g PGA, was put into practice after the Fukushima accident and the building was strengthened to the expected seismic events. In this process, some safety measures, from the measures taken after Fukushima, have been evaluated and implemented with the recommendations made in the framework of the different missions made by the IAEA. These include essentially, emergency water spray system above the core and portable emergency energy supply generators and their external connection boxes. Additional safety precautions have been taken, such as fixing the overhead crane against seismic derailing, analyzing all the critical components for nuclear safety and upgrading or retrofitting where necessary.

Safety Reassessment of Ukrainian Research Reactors in Light of the Lessons Learned from the Fukushima Daiichi Accident

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Abstract. Ukraine operates two research reactors:

- WWR-M (total capacity – 10 MW), which is located on site of Kyiv Nuclear Research Institute of the National Academy of Sciences of Ukraine;
- DR-100 (total capacity – 200 kW), which is located on site of Sevastopol Nuclear Energy and Industry University.

Both of them have been operated since 1960.

One more research installation is under construction: the Neutron Source Facility - based on subcritical assembly driven by linear accelerator of electrons.

After the Fukushima-1 accident, Ukraine joined the initiative of the European Commission and European Nuclear Safety Regulatory Group (ENSREG) regarding performing stress tests for NPP units and spent fuel storage facilities based on the ENSREG stress test and peer review specifications (Declaration on Stress Tests, 24 June 2011).

In this period, both of Ukrainian research reactors were under the process of lifetime extension. Taking into account that most of countries worldwide, which operate research reactors, had performed or were conducting the complimentary assessment of their safety, State Nuclear Regulatory Inspectorate of Ukraine required Ukrainian Operating Organizations (OO) to carry out the complimentary assessment and to combine it with Periodic Safety Review for the research reactors.

The OOs had to perform additionally assessment in detail:

- external extreme natural events (earthquakes, flooding, fires, tornadoes, extremely high/low temperatures, extreme precipitations, strong winds, combinations of events, etc);
- loss of electrical power and/or loss of ultimate heat sink;
- severe accident management.

SNRIU took into account the results of stress tests during the assessment process of OO's application on the lifetime extension of WWR-M. According to the results of "stress tests" a set of measures for improving of RR's safety have been developed and implemented.

OO of DR-100 did not finish stress tests and preparation of the final version of lifetime extension justification materials since Crimea had been temporarily occupied by the Russian Federation. The license on operation of IR-100 was suspended by SNRIU in 16 June 2014.

Additionally the lessons learned from Fukushima Daiichi accident have been taken into account during the safety justification of the Neutron Source Facility - based on subcritical assembly driven by linear accelerator of electrons.

The paper contains the main results of reassessment and the status of implementation of developed measures for improving the safety of RRs.

Filling the Neutron Gap at the Canadian Nuclear Laboratories after Shutdown of the National Research Universal (NRU) Reactor

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Abstract. The NRU reactor, commissioned in 1957, has served three primary purposes: to be a supplier of industrial and medical radioisotopes used for the diagnosis and treatment of life-threatening diseases; to be a major Canadian facility for neutron physics research, and to provide engineering research and development support for CANDU[®] power reactors. Present uses for the NRU include irradiations in support of current and advanced reactors and isotope production. The NRU also hosts the Canadian Neutron Beam Centre, where materials science research is carried out.

On March 31, 2018, the National Research Universal (NRU) Reactor will be permanently shut down. Canadian Nuclear Laboratories (CNL), as Canada's national nuclear laboratory, needs access to neutrons to realize its vision [1], and is considering multiple options to fill the neutron gap.

CNL has developed a list of requirements based on current and projected work, and is leveraging studies that have been done both inside and outside CNL. The requirements for nuclear fuels testing, for example, are quite different than the requirements for materials research or isotope production. There are multiple options for filling the neutron gap, and each is being considered with respect to factors such as timeline, capabilities, cost (including transportation costs), available space, and feasibility. Options include utilizing multiple test reactors as needed, setting up long-term agreements for reserved space with one or more test reactors, leveraging the IAEA ICERR (International Centre based on Research Reactors) framework, and participation in the development of a new test reactor such as the Jules-Horowitz Reactor. The optimal solution may be a combination of these options.

This paper discusses the options under consideration and progress towards identification of the best way to fill the neutron gap at CNL.

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The Impact of Cultural Diversity on the Technological Innovation Process in the Nuclear Energy Corporations

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Abstract. The research paper intended to show the impact of cultural diversity on the technological innovation process in the nuclear sector in the UAE. The study is based on both of the secondary and primary resources of information. The focal emphasis areas on the paper is to show the benefits of cultural diversity in respect of innovation process, the pouring forces for nuclear technology innovation, the nuclear innovation program, the challenges of cultural diversity in respect of innovation process and the current strategies of handling cultural diversity for executing innovation process.

For the primary data, it has been gathered from a sample of the staff working in the Nawah Energy Company, which has been chosen for conducting the primary research. In the present context 75 employees of the Nawah Energy Company have been chosen for collecting the quantitative data, while 3 managers of the same organization have been selected for gathering the qualitative data.

The paper recommends several strategies for improving the positive impact of the cultural diversity practices in the nuclear sector to improve the innovation process in this critical sector. These included the communication cost, access of market, creative ideas, as well as, resource allocations are the highly crucial aspects of the innovation process in the nuclear sector.

Development of Transient Testing Capability to Support the TREAT Facility

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Abstract. The re-establishment of the TREAT facility at the Idaho National Laboratory as a fuel safety research instrument is well underway following formal authorization from the US DOE to begin reactor operations in August 2017. Three technical pillars are being pursued to maximize the scientific impact of the instrument. This includes a full exploration of TREAT's range to generate transient shapes relevant to current experimental needs, the development of irradiation devices to deliver the desired environmental boundary conditions, and the development of examination and instrumentation capabilities to enable sample characterization.

Public Education and Outreach for Supporting Nuclear Program in Indonesia

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Abstract. Public acceptance is an important aspect in the utilization of nuclear energy. Therefore, efforts to increase public understanding and knowledge on nuclear is one of the important efforts to increase the level of public acceptance. Currently, the trend of rejection of nuclear energy program is seen maybe due to the lack of understanding of the people. To address these challenges, the Indonesian government through relevant institutions and ministries conducts public education and outreach programs to increase public understanding and knowledge on nuclear issues. Public education and outreach programs are designed as a combination of educational and training approaches. Public education and outreach to young generation through education and training are important due to the following outcomes: younger generations may have higher interest to higher levels of education in science and technology, engineering, and math, (STEM), including nuclear science; students may choose a nuclear related career; future generations may have a proper knowledge on nuclear, even their careers are not directly linked to nuclear industry; it can help people and make rational judgments and decisions in their lives.

Activities in the formal education path consists of the introduction of nuclear content in the formal education curriculum at the secondary educational level, and the provision of smart books as a reference for high school teachers. In addition, BATAN provides higher education scholarships in the field of nuclear technology, which significantly contributes the increased interest of students to study nuclear. Training are also held on introduction of nuclear science and technology for high school teachers, practical works for university students in radiochemistry and medical physics. Outreach through informal engagement in the formal education pathway is done through the fostering of young communities, Nuclear Goes to School, Science Day, Facility Visit to Nuclear Science Competition. Public opinion polls are conducted to quantitatively measure the effectiveness of public education and outreach programs on improving community understanding and acceptance. Polls show an increasing trend of public acceptance of nuclear energy programs.

Multidisciplinary Engagement at Research Reactors: The NCSU PULSTAR

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Abstract. The PULSTAR reactor at North Carolina State University (NCSU) is the primary facility of the Nuclear Reactor Program; a Board of Governor's Center in the University of North Carolina system. It went critical in 1972 and has been operating at 1-MWth for the past 45 years. To enhance the engagement of the PULSTAR in the institutional mission of NCSU, a strategic plan with well-defined education, research and service/outreach objectives has been under implementation for the past 15 years. Educationally, this included pioneering modern modalities such as the Internet Reactor Laboratory (IRL), which is currently in its 12th year of implementation with international extensions to Jordan, Vietnam and other potential regions. In scientific research, several unique and high-performance instruments have been developed and operated including a neutron powder diffractometer, an intense positron beam, an advanced neutron imaging system, an ultracold neutron source, and a fission gas release and measurement loop. These instruments are currently utilized to address the needs of a multidisciplinary community of fundamental and applied researchers. In addition, projects supporting the nuclear data community and emerging fields such as cybersecurity have been launched. Furthermore, the engagement footprint of the PULSTAR is highly enhanced through national and industrial partnerships to support developments in important sectors such as nuclear energy, nanotechnology and radioisotope production. This includes memberships in two important consortia: the US Department of Energy's Nuclear Science User Facilities (NSUF), and the US National Science Foundation's Research Triangle Nanotechnology Network (RTNN). The combination of capabilities and partnerships has resulted in significantly enhancing the utilization levels of the PULSTAR to approach 10,000 user hours annually. Consequently, over the past 15 years the PULSTAR has succeeded in meeting or exceeding institutional metrics for educational impact, multidisciplinary engagement, academic/scientific performance, and the ability to be self-supporting. This trend is expected to continue as the PULSTAR upgrades to a power of 2-MWth and completes the implementation of its next generation of instruments and projects.

Disposing High-level Transuranic Waste in Subcritical Reactors

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Abstract. We propose a new method of burning high-level transuranic (TRU) waste combined with thorium-uranium fuel cycle in the subcritical reactors driven by the external fusion neutron sources. The thorium-based TRU fuel burns all the long-lived actinides by hard neutron spectrum while outputting power. The concept of the corresponding one dimension model is built by means of the ONESN_BURN code with new data libraries. The thorium-uranium fuel cycle produces less TRU, less radiotoxicity and fewer long-lived actinides. The thorium-uranium fuel cycle provides breeding of ²³³U with a long operation time (>20 years), hence significantly reducing the reactivity swing while improving safety and burnup. A detailed analysis is given in the paper.

Keywords: High-level Transuranic Waste; Thorium-Uranium Fuel Cycle; Thorium Base TRU Fuel; Actinide Burning.

Possible Shifts in MARIA Reactor Reactivity and Power Changes Caused by the Seismic Event

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Abstract. The paper is investigating the possible impact of seismic events on the change of reactivity and Power of the MARIA research reactor in Poland, caused by potentially occurring vertical oscillations of control rods. Using the measurements of the actual vibrations of the reactor, a calculation model was developed and was used to determine the scale of the threat. Data used to calculate the problem were actual waveforms of earthquakes registered in Poland, upscaled to meet international recommendations of reactor. They were scaled to the peak ground acceleration, recommended for the calculation of nuclear reactors safe shutdown earthquake limits.

Recent Developments of the OSCAR Computational System, as Applied to Selected Examples from IAEA Research Reactor Benchmarks

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Abstract. The OSCAR-4 code suite is a nodal diffusion based calculational system, which has been used over many years for research reactor support. It is primarily used to support the operation of the SAFARI-1 research reactor at Necsa, South Africa, but is also applied at various other international research reactors (such as HOR, HFR and MNR). Recently, the next generation OSCAR system (termed OSCAR-5) has been under development, with specific focus on the challenges which highly heterogeneous research reactor core designs pose – in particular with regard to core design and core-follow type analyses. The main aim of the new development is the seamless integration between high fidelity and standard core analysis methods. A detailed heterogeneous model is constructed in a code-independent front-end system, which is then capable of deploying the model to all codes connected to it. In particular, automatic input generation is available for Monte Carlo codes like MCNP and Serpent, as well as the nodal diffusion solver in OSCAR-4. The deployment to a nodal diffusion code uses advanced homogenization and nodal equivalence methods to ensure a theoretically minimized discrepancy between the heterogeneous and homogeneous solutions. The nodal model is developed in a staged process, allowing tight monitoring and control of the model error as compared to the reference heterogeneous Monte Carlo model. In particular, all non-fuel homogenized multi-group cross-sections are generated from a set of full-core heterogeneous calculations, while fuel models are generated from typical, often infinite lattice, environments. The use of infinite lattice models results in the so-called environmental error on the nodal equivalence parameters, which in the new system may be remedied via various correction schemes in the nodal diffusion solver. The numerical impact of these approaches is illustrated on the SAFARI-1, OPAL and IPEN reactor benchmarks as they appear in the ongoing IAEA CRP T12029 on multi-cycle depletion and activation analysis.

Investigation on Core Downward Flow by a Passive Residual Heat Removal System of Research Reactor

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Abstract. Most of existing research reactors has been designed with Active Residual Heat Removal System (ARHRS) to remove decay heat of reactor core when the primary cooling pump stops. However, ARHRS takes much cost and is hard to design. As an alternate system, Lee et al. [1] proposed a Passive Residual Heat Removal System (PRHRS) of research reactor. PRHRS consists of three main components, a flywheel linked to the primary cooling pump, Gravity Core Cooling Tank (GCCT), and flap valves. For the PRHRS, a research is needed to verify the safety system can be applied to design of research reactor. In this paper, the verification was focused on performance of the GCCT in terms of hydraulics, specifically mass flow rate. An experimental facility was manufactured replicating a research reactor with GCCT in small scale and experiment was conducted. Furthermore, a theoretical formulation and computational fluid dynamics (CFD) model were developed to provide a better understanding of fluid flow characteristics of GCCT. The maximum mass flow rate of two models was about 50% higher than experimental results. It is because the pressure loss of experimental facility is bigger than that of two models. Considering this reason of difference, theoretical and CFD models were improved to follow the experimental results by adjusting diameter of DPP. As a result, the difference of maximum mass flow rate decreased below 17%. To apply these models to real scale research reactors, the models must be verified through further experiments in many different conditions including heat transfer aspects.

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Investigation of siphon breaker simulation program through small scale siphon breaker experiment

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Abstract. When a research reactor which has a characteristic of the core down flow is designed, some important components like pump are located at a lower height than the core is. It is because of siphon phenomenon. It happens through a pipe when the main pipe of the primary cooling system is ruptured. As coolant leaks from the reactor pool, the water level of the pool gets lower as much as the coolant leaks. Thus, a core is exposed to air and this can lead to dangerous situations. To prevent it, siphon breaker is developed. However, as it is difficult to predict the results, a siphon breaker simulation program(SBSP) was designed by Lee et al.[7]. In this study, by using the SBSP, a small scales siphon breaker was designed to verify the SBSP. Range of experiments included previous experimental range by Kang et al.[4][5], and expended to improve the SBSP. The results of experiments follow the SBSP's one except for the extrapolation range. As a result, the SBSP is a good estimate for designing general siphon breakers, but it requires the model improvement for satisfying the wider range.

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A Dummy Core for V&V and Education & Training Purposes at TechnicAtome: In and Ex-Core Calculations

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Abstract. Core calculations, both deterministic and stochastic, constitute every-day activities in TechnicAtome (formerly AREVA TA) Core Physics Department for the design and operation of nuclear research facilities. Developments of calculation schemes but also methods are a key issue.

In order to enable objective comparisons in methodologies, TechnicAtome has designed a dummy core. This non-existing core provides a mixture of some common features found in research reactors throughout the world. Its characteristics enable TechnicAtome to validate and qualify both calculation and computational techniques on a single-of-a-kind core with significant heterogeneities, thus extending the validation to almost any kind of small light-water reactor.

In addition to comparing methodologies, this dummy core is also used for non-regression tests performed between incremental versions of home-made calculation schemes. It is finally also used for Verification and Validation purposes, when it comes to comparing different codes on one given object.

This paper presents the fictitious core itself, with its description, from the standard FAs (Fuel Assemblies), to the hafnium-plate controlled assemblies and the full core with its in and ex-core features. It then presents the main results in terms of reactivity and power distribution, both for a basic 2D infinite periodic assembly and for the full core.

The paper also describes a methodology developed to calculate ex-core detector responses. This method consists in the TRIPOLI4© “Green Function” feature to determine the importance of each assembly concerning detector signal. This knowledge then allows to instantly predict the detector response in case of different control rod configurations. Comparisons with full simulations are performed.

Finally, the paper illustrates the sketch of TRIPOLI4© and GEANT4 unification at geometry level with further outlooks.

Radiation Resistance of the U(Al, Si)₃ Alloy: Ion-induced Disorder

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Abstract. During the exploitation of nuclear reactors various U-Al based ternary intermetallides are formed in the fuel-cladding interaction layer. Structure and physical properties of these intermetallides determine the radiation resistance of cladding and ultimately the reliability and lifetime of the nuclear reactor. In current research U(Al, Si)₃ composition was studied, as a constituent of interaction layer. Phase content of the alloy of an interest was ordered U(Al,Si)₃ intermetallide, structure of which was fully characterized and revealed earlier, and pure Al which constituted less than 20 vol.% of the alloy. This alloy was investigated prior and after the irradiation performed by Ar ions at 30 keV. Experiment was performed on the Transmission Electron Microscopy (TEM) samples with approximate thickness (in the electron transparent area) of 100 nm. It was found that there is a dose threshold for disordering of the crystalline matter in the irradiated region of material with appearance of almost solely disordered phase. Using the programs for Stopping and Range of Ions in Matter (SRIM) and Transport of Ions in Matter (TRIM), the parameters of penetration of Ar ions into irradiated samples were estimated. Based on these estimations, the features of the dose dependences for ion-induced material disordering were explained. Experimental results are in agreement with calculated ones. In addition, experimental results point on stress relief due to irradiation.

Management of Safety and Licensing Requirements during the RA-10 Reactor Construction Stage

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Abstract. Construction stage of the RA-10 Reactor begun on DATE. Previously, the construction license was obtained from the Argentinean regulatory body, which is a requirement to start any manufacturing or construction activity. For this stage, CNEA has established a management system to ensure that, as a licensee, all safety and regulatory requirements are correctly implemented. A configuration management system is the key management tool for this stage in order to ensure that all regulatory requirements established in the construction license are reflected in the project documentation and are therefore correctly implemented during manufacturing and construction activities. Through configuration management, it is ensured that all relevant requirements and the valid design documentation are provided to manufacturing and construction contractors. Additionally, design changes, which is a very important aspect during this stage, are addressed in the configuration management procedures. The configuration management system follows IAEA and other international organizations guidelines [1][2]. Another important aspect is the management of communications with the regulatory body that on one hand need to be clear and precise and on the other hand need to be conducted within the time frames established in the construction license. This paper describes the aspects of the construction management system, particularly those related to the management of safety and regulatory aspects, with emphasis on management tools and procedures used in the field.

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Theoretical Study of Steady State Neutron Flux Re-construction in ADS subcritical Reactor by Using Higher-order Modes

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Abstract. Neutron flux in Accelerator Driven Subcritical Reactor(ADS) is formed by superposition of fundamental and higher-order modes neutron flux, which provides the physical foundation of modes expansion based studies in ADS subcritical reactor characteristics. Based on the bi-orthogonal properties of forward and ad-joint modes, modes expansion theory for steady state neutron flux in ADS subcritical reactor is established in this paper, and numerical studies on three dimensional four groups ADS subcritical reactor diffusion problem are performed. Results indicate that λ and prompt α modes can effectively re-construct steady state neutron flux of ADS subcritical reactor, neutron flux re-construction accuracy is enhanced by increasing expansion number of modes. Compared with prompt α modes, λ modes are more appropriate for steady state neutron flux re-construction. Since the symmetries of external neutron source and core pattern in this paper, only modes that have symmetrical properties have contributions to steady state neutron flux.

Radioactive Radon Effect of Spent Fuel Storage Pool Kr-85 Monitor

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Abstract. In the Post Irradiation Examination Facility (PIEF) spent fuel storage pool site of KAERI, there are three Kr-85 monitors in order to detect Kr-85 in real time. When if certain spent fuel cut and float above the pool surface while operating the facility, the Kr-85 Monitors can detect Kr-85 spurt from the damaged spent fuel.

The Detector of Kr-85 Monitor is proportional counter tube in sandwich geometry, and can detect radioactive noble gas. Noble gas flow into the input port not adsorbed to the pre-filter but counts in the count chamber. Measured values calculated by applying the count efficiency of Kr-85. At the same time Rn-222 which is also a Noble gas counts in the counter chamber as well, and the measured values can be shown.

Radioactive radon nuclides which are always in the work field can cause confusion whether these values are caused by Kr-85 or Rn-222. As a result the workers in that field cannot estimate whether the work field is in safe condition or not. For that reason, it is essential that conduct refinement and separation of radioactive gases, like Rn-222.

In this study, to estimate whether the measured value from the spent fuel pool site of PIEF, KAERI Kr-85 monitor is actually caused by Kr-85. Furthermore, behaviour of radioactive radon nuclides had been monitored in the pool site. The observation was conducted comparing continuous radon monitor (RAD7) installed in the workplace near the pool side and a Kr-85 monitor (FHT 57 E-S) which is continuously monitoring the pool surface.

During suspension of working in the area, the data trend of Kr-85 monitor and RAD 7 was about equal. As a result, it can be figured out that the measured data of Kr-85 monitor was from Rn-222 which is always in the area. Merely, the trend was not exactly equal, and there was some difference as time goes by. The reason was figured out that the difference of sampling port of Kr-85 monitor and RAD7 made some differences.



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INVAP is proud of its pursuit of innovative, cost-effective and timely solutions for nuclear clients. These have included tailor made reactor cores for maximum performance, neutron beams optimized for maximum neutron flux yields and innovative irradiation systems technology. For all the projects in the nuclear field INVAP implements an encompassing approach that results in the implementation of a healthy and robust safety culture. The approach results in a careful implementation of the defence in depth approach,



Korea Atomic Energy
Research Institute

The Korea Atomic Energy Research Institute (KAERI), which was established in 1959, is the first science and technology research institute in Korea to be mandated to achieve energy self-reliance through nuclear technology. Since its establishment, KAERI has played a leading role in the advancement of science and technology as well as national economic growth by producing landmark achievements during the 50-year history of nuclear energy development. Such achievements include the self-reliance of the Nuclear Steam Supply System design of Korean Standard Nuclear Power Plants, the localization of both CANDU and PWR fuels, the indigenous design of the HANARO research reactor and the nation's first ever exportation of a whole nuclear facility to Jordan. Meanwhile, the HANARO, which was shut down for the last three and a half years, is now ready to return to power and KAERI will shortly re-open the web page for the beam time allocation.

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